Study on Analysis of Incidents and Accidents at Nuclear Installations

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Norio Watanabe

Abstract

In order to assure the safety in nuclear facilities, it has been worldwide recognized an important and effective means to obtain the lessons and insights through the analysis of incidents and accidents and to feed them back to the design, construction, operation and management of facilities. Using event information, the activity called operating experience feedback (OEF) has been carried out on a national or international basis. This activity can be achieved by a systematic and comprehensive review and analysis of events from various points of view such as generic aspects, specific aspects, and risk significance aspects. Aiming at providing insights and technical issues useful for regulating and improving the safety of nuclear facilities, the present thesis addresses a series of studies on analysis of nuclear and radiological events as follows: the generic studies (including comprehensive reviews) of various events, topical studies on safety significant events, accident sequence precursor (ASP) studies, and review of the five investigation reports on the Fukushima accident. As well, described are new computer software tools developed to assist in the comprehensive reviews. Finally, provided are discussions on insights from these studies and precursors to the Fukushima accident to clarify the generic safety issues to be resolved.

In the generic studies, a number of event reports were reviewed to examine overall trends and characteristics of safety relevant events and to identify the safety significant events which (might) have affected the facility, environment and/or public health. The studies provide the observations on safety issues to keep in mind among the nuclear community and indicate that the safety significant events can be roughly categorized into three groups of events: recurring events, events associated with external hazards and the events related to new phenomena or unexpected aggravating conditions. As a supporting tool for comprehensive reviews, a new computer software package CESAS was developed to extract the event sequence from the event description in English. Through its applications, it is shown that the CESAS-extracted event sequences generally agree with manually-extracted ones, demonstrating the effectiveness and feasibility of CESAS.

Three topical studies are described on loss of decay heat removal during reactor shutdown at pressurized water reactors, criticality accidents at nuclear fuel fabrication and reprocessing facilities, and setpoint drift in safety or safety/relief valves at light water reactors. These studies derive the generic safety issues and insights useful for examining the measures against individual topics and show that the lessons learned from past events have not sufficiently been employed and effective measures have not taken, implying that sharing the relevant information on operating experience should be enhanced on a national and international basis. In particular, the lessons learned and insights obtained should be disseminated into the nuclear community.

In the ASP studies, the analysis of steam generator tube ruptures identifies the technical issues to be resolved for mitigating such an event, by indicating the significance of timely depressurizing the reactor in terms of core damage probability. The trending analysis with use of newly proposed quantitative risk indicators, that is, annual core damage probability and occurrence frequency of precursors, reveals that the core damage risks have been lowered and the likelihood of risk significant events has been remarkably decreasing at U.S. nuclear power plants. Also, the proposed risk indicators are shown to be useful for determining the risk characteristics of events, monitoring the risk level at nuclear power plants, and examining the industry risk trends.

The Fukushima accident is delineated with use of event trees to clarify the differences among accident sequences at Units 1 to 3 and the actual responses to avoid the severe accidents are discussed. Also, the review of five investigation reports on the accident identifies the technical issues to be further examined and discussed so that effective OEF can be performed at other nuclear facilities.

Finally, the present thesis discusses the insights obtained through the analysis of operating experience in terms of commonalities in the Fukushima accident and identifies precursors to the Fukushima accident. Highlighted are the importance of physical separation, the necessity of paying special attention to protection against common cause failures, the need to protect the site against external hazards and the importance of considering new phenomena or unexpected aggravating conditions. Based on the observations, generic safety issues to be resolved are addressed. The present thesis also indicates that the Fukushima accident should be analyzed more comprehensively and in detail considering the insights gained from past events.

The studies described in the thesis reveal the importance of bringing to light the generic safety issues from the operating experience by executing the systematic and comprehensive analysis of various events, in-depth analysis of significant events from various points of view, and accident sequence precursor studies.

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Acronyms

General Terms

AC : alternate current
ADS : automatic depressurization system
AFW : auxiliary feedwater system
AM : accident management
AOV : air-operated valve
ASDV : atmospheric steam discharge valve
ASP : accident sequence precursor
ATWS : anticipated transient without scram
BWR : boiling water reactor
CCDP : conditional core damage probability
CCF : common cause failure
CCW : condenser cooling water (system)
CDP : core damage probability
CHP : charging pump
CSR : containment spray recirculation
CST : condensate storage tank
DC : direct current
DHR : decay heat removal
ECCS : emergency core cooling system
EDG : emergency diesel generator
EM : evaluation models
ENS : emergency notification system (U.S.A.)
EP : electrical power supply
EPRY : events per reactor year
EPSW : emergency process seawater
ESWS : essential service water system
FBR : fast breeder reactor
FRP : functional recovery procedure
F&B : primary feed and bleed
GCR : gas cooled reactor
HFU : uranium hexafluoride

HPCI : high pressure coolant injection (system)
HPI : high pressure injection (system)
HPR : high pressure recirculation (system)
IC : isolation condenser
JIT : just-in-time briefings
LER : licensee event report (U.S.A.)
LCO : limiting conditions for operation
LOCA : loss of coolant accident
LOFW : loss of feedwater
LOOP : loss of offsite power
LPI : low pressure injection
LWGR : light water cooled graphite moderated reactor
LWR : light water reactor
MC : metal clad
MCR : main control room
MDAFW : motor-driven auxiliary feedwater
MFW : main feedwater
MLSB : main steam line break
MOV : motor-operated valve
MOX : mixed oxide (fuel)
MSIV : main steam isolation valve
MSSV : main steam safety valve
NPP : nuclear power plant
PC : power center
PHWR : pressurized heavy water reactor
PORV : pressurizer/power operated relief valves
PRA/PSA : probabilistic risk/safety assessment
PSV : pressurizer safety valve
PWR : pressurized water reactor
PWSCC : pressurized water stress corrosion cracking
RCIC : reactor core isolation cooling
RCP : reactor coolant pump
RCS : reactor cooling system
RPV : reactor pressure vessel
RPVHS : reactor pressure vessel head spray
PSW : process seawater
RHR : residual heat removal (system)

RWST : refueling water storage tank
SAR : safety analysis report
SBO : station blackout
SDC : shutdown cooling
SER : significant event report
SFPC : spent fuel pool cooling
SG : steam generator
SGTR : steam generator tube rupture
SGTS : standby gas treatment system
SI : safety injection
SLC : standby liquid control
SOER : significant operating experience report
SRV : safety relief valve
SSPS : solid state protection system
TAF : top of active fuel
TBV : turbine bypass valves
TCE : trichloroethane
TDAFW : turbine-driven auxiliary feedwater
TG : turbine generator
UPS : uninterruptable power supply (system)
VCT : volume control tank
WWW : world wide web

Organizations

AEOD : Analysis and Evaluation of Operational Data (USNRC's Office, U.S.A.)
ANO-1 : Arkansas Nuclear One Unit 1 (PWR, U.S.A.)
ANO-2 : Arkansas Nuclear One Unit 2 (PWR, U.S.A.)
ANS : American Nuclear Society (U.S.A.)
ANSI : American Nuclear Standards Institute (U.S.A.)
ASME : American Society of Mechanical Engineers (U.S.A.)
B&W : Babcock and Wilcox (U.S.A.)
CE : Combustion Engineering (U.S.A.)
IAEA : International Atomic Energy Agency
JAEA : Japan Atomic Energy Agency (Japan)
JAERI : Japan Atomic Energy Research Institute (Japan)
JNES : Japan Nuclear Energy Safety Organization (Japan)
JCO : JCO Co. Ltd. (private company for nuclear fuel conversion) (Japan)

LANL : Los Alamos National Laboratory (U.S.A.)
NISA : Nuclear and Industrial Safety Agency (Japan)
NSAC : Nuclear Safety Analysis Center (U.S.A.)
NSC : Nuclear Safety Commission of Japan
OECD/NEA : Organization of Economic Cooperation and Development/Nuclear Energy Agency
RJIF : Rebuild Japan Initiative Foundation (Japan)
STP-1 : South Texas Project Unit 1 (PWR, U.S.A.)
TEPCO : Tokyo Electric Power Company, Inc. (Japan)
TMI-2 : Three Mile Island Nuclear Power Plant Unit 2 (PWR, U.S.A.)
USNRC : United States Nuclear Regulatory Commission
WANO : World Association of Nuclear Operators
WH : Westinghouse

Others

AIRS : Advanced IRS
CESAS : Computerized Event Sequence Abstracting System
CFR : Code of Federal Regulations (U.S.A.)
ICNC : International Conference on Nuclear Criticality Safety
INES : International Nuclear Event Scale (currently, International Nuclear and Radiological Event Scale)
IRS : Incident Reporting System (currently, International Reporting System for Operating Experience)
NEWS : Nuclear Event Web-based System
U.S.(A.) : the United States (of America)
U.K. : the United Kingdom

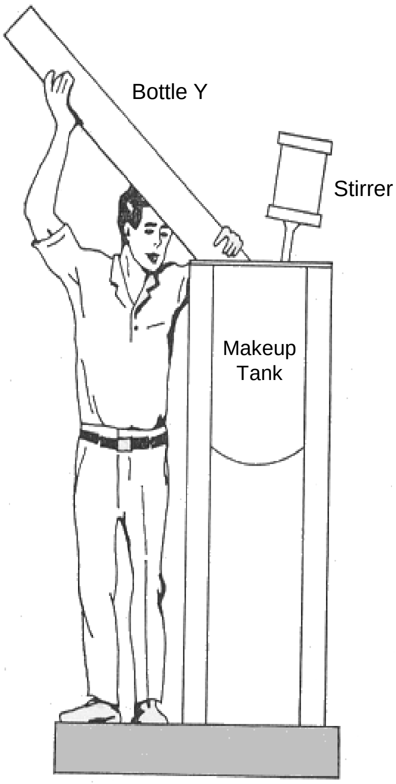
Chapter I

Introducing Remarks

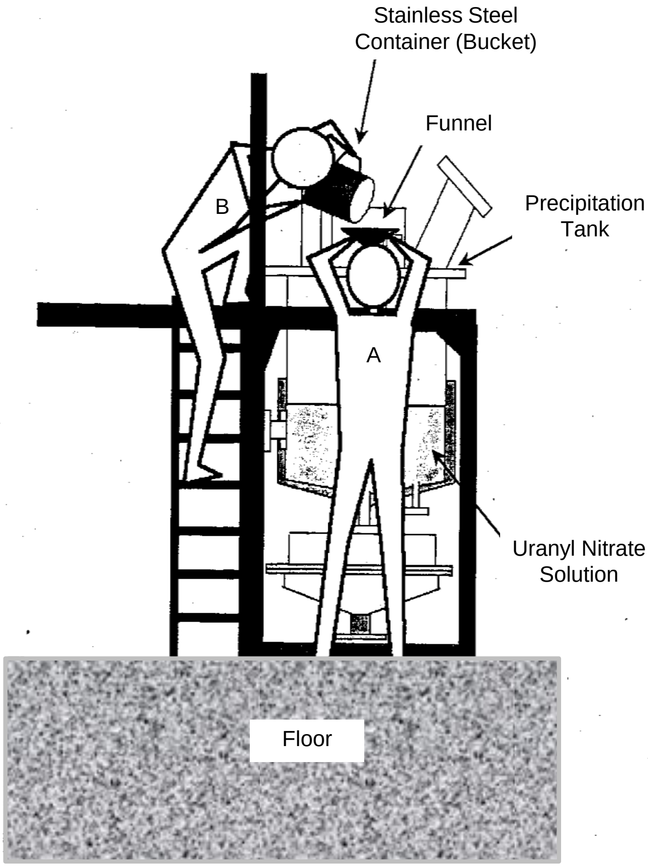
I.1 Background

It has been recognized an important and effective means to learn from experience and/or accidents, for identifying the matters that have not been noticed in the design stage and/or those to be considered for operation and management of facilities, not only in the field of nuclear technology but also in all types of technological fields. In order to assure the safety in nuclear facilities, in particular, one of the essential elements is to obtain the lessons and insights through the analysis of incidents and accidents that actually occurred and to feed them back into the design, construction, operation and management of facilities. Such activities have been carried out worldwide as “operating experience feedback,” and thus, the incident reporting has become an important aspect of the operation and regulation of nuclear facilities.

However, it may be the rare case that an incident or accident took place due to a completely new phenomenon and factor and in a large majority of events, similar occurrences have been experienced in the past. As shown in **Figure I.1**, for example, the criticality accident that occurred at the JCO uranium fuel processing plant in Tokai-mura in Japan on September 30, 1999(1) was extremely similar to the criticality accident at the Wood River Junction uranium recovery plant in the United States in 1964(2). Both accidents were caused by a massive amount of high-concentration uranium solution being injected into a container that was not designed for such use. The container reached a critical state, resulting in workers being exposed to lethal doses of radiation. Considering that these similar accidents had taken place, we may well wonder whether we have actually been learning the lessons from experience and accidents. It is highly likely that the criticality accident would have been avoided if the workers, who died as a result of the JCO criticality accident, had been given sufficient information about the Wood River Junction accident and had fully recognized the risk involved in what they were doing.



Criticality Accident at Wood River Junction
(July 24, 1964)
(Source: Ohnishi, et al., “Accidents in Nuclear Installations”, JAERI 4052, September 1970)



Criticality Accident at JCO
(Source: Nuclear Safety Commission of Japan, "The Uranium Processing Plant Criticality Accident Investigation Committee Report", December 24, 1999)

Figure 1.1 Circumstances at Onset of Criticality Accidents at Wood River Junction and JCO

Similar events have occurred at nuclear power plants. The accident at the Three Mile Island Unit 2 (TMI-2) that occurred on March 28, 1979[3-5] had some similarities to the event that occurred in 1977 at the same type of reactor at the Davis Besse Nuclear Power Plant[6]. However, the safety significance of this event had not been well recognized and it was said that the plant operating staff at TMI-2 had not known about this event. While, at that time, one nuclear engineer did realize the importance of the Davis Besse event and warned about the potential risk of this event, almost nobody did give attention to his warning, resulting in his prophecy coming true by the TMI-2 accident. These two events highlighted the importance of analyzing incidents and accidents and feeding the lessons learned back into the design, operation, and management of the plant.

At the Fukushima Dai-ichi Nuclear Power Plant, three units resulted in a severe core damage and the subsequent large release of radioactive materials into the environment, the major cause of which was the station blackout stemming from the tsunami-induced flooding in the buildings by the huge earthquake[7]. However, the tsunami hit a nuclear power plant in India, resulting from the earthquake in Sumatra Island in 2004 without any serious damage to the plant facilities. As well, the heavy rain with high winds resulted in the flooding of reactor building in a French nuclear power plant and the some safety-related systems were affected. However, the lessons learned from these flooding events had not been fed back into the design of the nuclear power plants in Japan.

I.2 Incident Reporting

Licensees or operators of nuclear facilities are required to notify and/or report the events, which occurred at their own plants, to the regulatory authorities immediately, analyze their causes, and take preventive measures. Such activities, that is, the event notification and/or reporting and the analysis of events, had mainly been carried out within individual national regulatory frameworks prior to the TMI-2 accident.

The TMI-2 accident prompted enhancement of the analysis and evaluation of operating experience at nuclear power plants. In the United States, partially in response to lessons from the TMI-2 accident, the United States Nuclear Regulatory Commission (USNRC) revised its immediate notification requirements via the emergency notification system (ENS) in 10 CFR 50.72 and modified and codified its written licensee event report system requirements in 10 CFR 50.73 in 1983[8,9]. While these reporting requirements range from immediate telephone notifications to written reports, covering a broad spectrum of events from emergencies to component level deficiencies, the USNRC wishes to emphasize that reporting requirements should not interfere with ensuring the safe operation of a nuclear power plant. Licensees' immediate attention must always be given to operational safety concerns.

In the light of the TMI-2 accident, as well, the Organization of Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) established the Incident Reporting System (IRS; so-called NEA-IRS) as a forum for the exchange of information on operational events at nuclear power plants and started its operation in 1981 with 13 member countries. Under this system, member countries are requested to submit an IRS report to the OECD/NEA Secretariat when an abnormal event or failure with safety significance occurs at their respective nuclear power plants, and then the Secretariat distributes the reports to member countries. As well, the International Atomic Energy Agency (IAEA) established a similar system (so-called IAEA-IRS) and started its operation in 1983 with 16 member countries. These two systems had been operated independently for a few years and the two agencies reached an agreement on the mutual exchange of IRS reports in 1988. Since then, OECD/NEA and IAEA member countries can receive all of the IRS reports submitted to both agencies. To promote the effectiveness of IRS, the IAEA and the OECD/NEA approved the implementation of joint guidelines(10) in 1997, resulting in the official integration of both IRSs into one system, the IAEA/NEA-IRS. In accordance with the integration and joint operation, the reporting formats were unified in 1998. After that, the Advanced IRS (AIRS), an IAEA's computer database of the IRS reports, was developed and the CD-ROM containing the database updated had been periodically distributed to member countries. In recent years, a new database system for IRS, Web-based IRS, was developed and has been operated by the IAEA in cooperation with the OECD/NEA, and member countries can enter the events that occurred at their nuclear power plants and retrieve the report in the database(11). In addition, the official name of IRS was changed from "the Incident Reporting System" to "the International Reporting System for Operating Experience." As of the end of December 2012, 32 countries are participating in the IRS, and more than 3,750 events had been reported since 1981. As the IRS was designed to facilitate the exchange of information on operational events among regulatory agencies in member countries, access to the IRS reports is, in principle, limited to the individuals registered in the database(12,13). (However, decisions on access are left to the IRS national coordinators of the individual member countries.)

As well, the IAEA, in cooperation with the OECD/NEA, operates the International Nuclear Event Scale (INES: currently the International Nuclear and Radiological Event Scale)(14). The INES is a means for promptly communicating to the public the safety significance of events that occurred at nuclear facilities and radiation utilization facilities aiming to facilitate common understanding among the nuclear community (industry and regulators), the media and the public. The INES was introduced in March 1990 as a trial use for the events occurring at the nuclear power plants in the individual member countries. Afterwards, the scale was refined in the light of experience gained through the trial use and extended to be applicable to any event associated with radioactive materials and/or radiation, including the transport of radioactive materials. In March 1992, the official operation of INES was commenced. Japan formally introduced the INES in August 1992 and has been operating it since then. Since January 2002, the original reports are available at NEWS (Nuclear Event Web-based System) on the IAEA website[15], and their translated versions can be found at the website of the Japan Nuclear Energy Safety Organization (JNES)[16]

The scale can be applied to any event occurring at all types of civilian nuclear facilities (power reactors, research reactors, testing reactors, uranium mining and refining facilities, uranium enrichment facilities, fuel fabrication facilities, spent fuel storage and reprocessing facilities, waste treatment/storage/disposal facilities), transport of radioactive material between these facilities, and irradiation equipment/devices using radioactive sources (for medical or industrial uses). The INES is also applicable to the lost or stolen radioactive sources and the discovery of orphan sources at a place where they should not be. In addition, each event reported to the INES is accompanied by both the event summary that describes where and when it occurred, what kind of event occurred, and how it affected radiologically, and the "scale" that shows the safety significance of the event. On the scale, events are classified at 8 levels from level 0 for an event with no safety significance to level 7 for a major accident that may cause a wide range of health and environmental effects. These levels are determined according to the following three criteria[14].

(1) Criterion 1 (Impact on People and the Environment): This criterion is to rate an event resulting in the off-site releases of radioactive materials or the actual radiological impacts on workers and members of the public. The scale is set based on the amount of radioactivity actually released to the environment or the exposure dose of the workers or the public members. The rating based on this criterion covers a range from level 7, which corresponds to major release of radioactive materials with widespread health and environmental effects, to level 2, which is an event involving the exposure of a member of the public in excess of 10 mSv or the exposure of a worker in excess of the statutory annual limits.

(2) Criterion 2 (Impact on Radiological Barriers and Controls at Facilities): This criterion applies to facilities handling a large quantities of radioactive materials such as power reactors, reprocessing plants, or large source production facilities and is to rate an event involving significant damage to the primary barriers that prevent a large release of radioactive materials (e.g., reactor core melt or loss of confinement of radioactive materials) or a major spillage of radioactive materials or a significant increase in dose rate. The rating based on this criterion is classified into 4 levels ranging from level 5 to level 2. Specifically, an event resulting in significant damage to the primary barriers or a significant release of radioactive material within an installation is rated as level 4 or 5; an event involving significant contamination

within the facility into an area not expected by design is rated as level 2 or 3.

(3) Criterion 3 (Impact on Defense in Depth): This criterion applies to an event that deteriorates the defense-in-depth concept at a nuclear facility, a radioactive source utilization facility or during the transportation of radioactive material with no or lower levels of actual radiological consequences. All nuclear facilities are designed to avoid any significant on-site or off-site consequence by providing multiple layers of safety provisions based on the "defense-in-depth" concept. As well, the similar concept is applied to the design of radiation source utilization facilities and transportation of radioactive materials. For example, the radioactive materials are confined by using multiple layers. This criterion rates an event as a level from 3 to 1 considering the following two factors: the maximum potential consequence of the facility in the case that all the safety layers are lost, and the number of safety layers remaining their respective integrity (and reliability). For the lost or found radioactive source/device, the rating of such an event is carried out according to its activity. In addition, the rating based on this criterion may be upgraded by one level when the possibility of common cause failure, deficiencies in operating manuals/procedures or lack of safety culture is identified, while it may be downgraded by one level when the time period during which the concerned safety protection layer is lost is very short.

The individual member countries are strongly encouraged to communicate events internationally through the IAEA within 24 h, if possible, according to the criteria which are events at level 2 or higher, or events attracting international public interest. The INES has 61 member countries, including all countries except Taiwan that have nuclear power plants (32 countries). As of December 2012, approximately 850 reports have been submitted to the INES since its trial use was commenced in 1990.

In the nuclear industry, the necessity of a global network for informing events at nuclear power plants and disseminating the lessons learned from them was recognized worldwide after the Chernobyl accident in 1986[17-19], resulting in establishment of the World Association of Nuclear Operators (WANO) in May 1989 to provide a forum for sharing and utilizing operating experience over the world. Currently, more than 440 nuclear power plants in 35 countries take part in the WANO. The WANO analyzes operational events reported by members to identify important safety issues and prepares the reports, according to the safety significance and the number of events, as follows; SOER (Significant Operating Experience Reports), SER (Significant Event Reports), JIT (Just-in-Time Briefings)[20]. In addition, the WANO dispatches an international review team consisting of specialists from the member countries, investigates the performance of plant and personnel through walkdowns and interviews with staff, and makes proposals for improving safety and reliability. Furthermore, the WANO facilitates the exchange of information on good practices in routine works, the exchange of opinions and information on operations through the interchanging of operating staff, and the comparison of operating experience at the individual plants by quantifying performance indicators in nine major areas related to plant safety and reliability.

I.3 Objectives of Present Thesis

Operating experience feedback has been recognized one of the important issues needed for ensuring the public and environmental safety. This can be achieved by a systematic and comprehensive review and analysis of events that actually occurred from various points of view such as generic aspects (generality, commonality, similarity), specific aspects (specificity, uniqueness), mechanical aspects, human aspects, and risk significance.

The present thesis aims at identifying the safety significant events and their trends, and providing insights useful for improving nuclear and/or radiological safety, in particular, on nuclear power plants, through the generic studies, topical studies, and accident sequence precursor studies. As well, this thesis addresses the technical issues to be further examined on the severe accidents at the Fukushima Dai-ichi Nuclear Power Plant (hereafter, the Fukushima accident) and the safety significant insights which should have been taken into account prior to the Fukushima accident so that the lessons learned from the accident can be fed back to the design, construction, operation and management of existing and future nuclear facilities in the world.

I.4 Outline of Present Thesis

The present thesis consists of the following chapters: Chapter II describes the generic studies and comprehensive reviews of operational events to identify safety significant events, that is, events which could threaten the plant safety. These studies and reviews cover the nuclear and radiological events reported to the INES and IRS. As well, addressed are the supporting tools which were developed so that the comprehensive reviews of events can be carried out more efficiently. Chapter III describes the topical studies on the safety significant events identified in the comprehensive review and/or generic study, focusing on three topics; loss of decay heat removal during reactor shutdown at pressurized water reactors, criticality accident in fuel processing facilities and setpoint drift of safety or safety/relief valves at light water reactors. In the topical studies on events at nuclear power plants, the event information is obtained mainly from the USNRC's licensee event reports (LERs). Chapter IV addresses the accident sequence precursor (ASP) studies on steam generator tube rupture, one of safety significant events, and the risk trends at nuclear power plants with newly proposed indicators, and provides the findings and insights from these ASP studies. The analytical approach for the ASP studies is also mentioned. Chapter V delineates the accident scenarios of the Fukushima accident with event trees and identifies the technical issues to be further discussed through the review of the five investigation reports on the Fukushima accident, which focuses on the accident progression and causes. In Chapter VI, the insights and precursors to the Fukushima accident are discussed based on the studies mentioned in Chapters II to IV. The relations of the individual chapters in the present thesis are shown in **Figure I.2**.

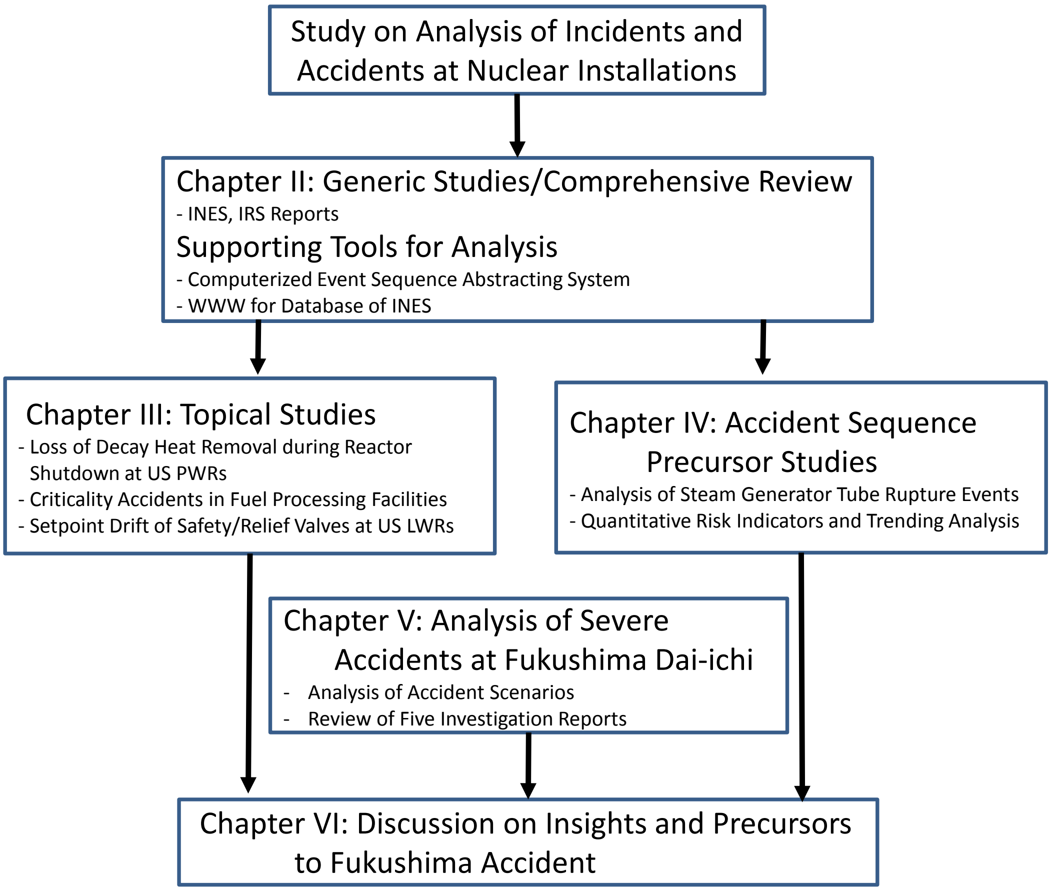


Figure I.2 Outline of Present Thesis

References in Chapter I

(1) Nuclear Safety Commission of Japan, *The uranium processing plant criticality accident investigation committee report*, 1999 (in Japanese).

(2) T. Ohnishi, et al., *Accidents in Nuclear Installations*, JAERI 4052, 1970 (in Japanese).

(3) Nuclear Safety Commission of Japan, Monthly Report, *The 3rd Special Investigation Report on the Accident at the U.S. Nuclear Power Plant,* Vol.33, 1981 (in Japanese).

(4) K. Sato, *Logic of Nuclear Safety (Japanese title: Genshiryoku anzen no ronri),* Nikkan Kogyo Shimbun, Ltd., 1984 (in Japanese).

(5) Nuclear Safety Research Association, *Investigation Report on Three Mile Island Nuclear Power Plant Accident (II) - NRC Special Inquiry Group (The Rogovin Report),* 1981 (in Japanese).

(6) For example, *Rapid Depressurization Occurs*, Nuclear Safety, Vol.19, No.2, pp.241, March-April, 1978.

(7) For example, TEPCO, *Fukushima Nuclear Accident Analysis Report*, 2012.

(8) USNRC, *Licensee Event Reporting System*, NUREG-1022, 1983.

(9) USNRC, *Event Reporting Guidelines 10 CFR 50.72 and 50.73,* NUREG-1022 Revision 1, 1998 and Revision 2, 2000 .

(10) IAEA & OECD/NEA, *IAEA/NEA Incident Reporting System (IRS): Reporting Guidelines*, 1997.

(11) *Web Based Incident Reporting System – WB IRS: "www-ns.iaea.org/home/ni/databases.asp#irs"*. (as of December 2013).

(12) IAEA & OECD/NEA, *Nuclear Power Plant Operating Experience from the IAEA-NEA International Reporting System for Operating Experience 2009-2011*, 2012

(13) IAEA & OECD/NEA, *THE IAEA/NEA INCIDENT REPORTING SYSTEM (IRS) Using Operating Experience to Improve Safety*.

(14) IAEA & OECD/NEA, *The International Nuclear (and Radiological) Event Scale (INES) User's Manual 2001 Edition*, (2001) and *2008 Edition*, (2009).

(15) *NEWS Web-site: "www-news.iaea.org"*. (as of December 2013)

(16) *JNES Website:"www.atomdb.jnes.go.jp/"*. (as of December 2013)

(17) IAEA, *Summary Report on the Post-Accident Review on the Chernobyl Accident,* IAEA Safety Series No.75 – INSAG-1 (1986).

(18) Atomic Energy Bureau of Science and Technology Agency, *Chernobyl – Safety of*

Nuclear Reactors, (1999) (in Japanese).

(19) Nuclear Safety Commission of Japan, *1986 White Paper on Nuclear Safety*, (1987) (in Japanese).

(20) *WANO Website: "www.wano.info/programmes/operating-experience/"*. (as of December 2013)

Chapter II

Generic Studies and Supporting Tools

II.1 Generic Studies on Nuclear Events

Generic studies comprehensively review and analyze event reports to the IAEA-OECD/NEA INES and IRS as well as USNRC's generic communications. The objectives of these studies are to (i) identify events with safety significance or safety implications, which (might) have affected the facility, environment and public health, and (ii) observe the overall trends and characteristics of such safety significant events.

The INES places emphasis on rapid communication of nuclear or radiological events. Therefore, it can be used as an effective tool for promptly understanding what type of event has recently took place and for identifying and/or categorizing events with safety significance, involving radioactive release to the environment or involving overexposure of public members or employees. As well, rough analyses of these event reports may provide some insights useful for determining whether the individual events implicate any important issues from the points of facility and/or radiation safety even though the respective event information is limited. To facilitate common understanding among the nuclear community, the media and the public in Japan, the individual event reports have been translated into Japanese and the Japanese versions have been published through the internet(1) and in the documents(2,3). In addition, the generic study was performed to examine trends and characteristics of approximately 500 events reported to the INES during the period 1990-2000(4).

On the other hand, the IRS aims at ensuring that feedback of operating experience gained from nuclear power plants (NPPs) worldwide on safety related events is widely shared amongst the international nuclear community to help prevent occurrence or recurrence of serious events. As the IRS has to provide sufficient detail to highlight the wider relevance of operating experience to the recipient, an IRS report should provide detailed information on root causes, safety significance, lessons learned, and corrective actions from the technical, organizational, and human factors aspects. As such, the IRS is designed for specialists of the nuclear community as a source of detailed information on analysis and lessons learned from operating experience and information reported is not intended for distribution to the general public. Thus, access to the IRS reports is restricted to encourage openness within nuclear community, including the disclosure of incident details and related plant actions[5,6]. Since 1988, more than 2000 IRS reports have been analyzed. For the first ten years, approximately 700 IRS reports submitted to the OECD/NEA were reviewed and the results were reported to the Nuclear Safety Commission of Japan[7-11]. After the two IRS systems operated by the OECD/NEA and the IAEA separately were officially integrated into one system in 1997, the individual IRS reports have been analyzed and complied in confidential documents on a yearly basis and these documents were disseminated to the regulatory authorities and utilities in Japan to assist them in understanding the events which had occurred in foreign countries[12-27].

Also, the USNRC's generic communications are useful for identifying safety significant events because these are issued for the events with safety implications or generic issues.

In this section, discussed are the generic studies of events reported to the INES and IRS, respectively.

II.1.1 Generic Study on Events Reported to INES – Trending Analysis.

1. Overall Trends

1.1 Number of Reports by Year

Although, according to the criteria of INES, the individual member countries are strongly encouraged to report the events with a rating of Level 2 or higher, as previously mentioned, many events with a rating of lower than Level 2 have been reported from member countries. During the period 1990-2000, there have been reported a total of 488 events, which consist of 409 events at NPPs (237 at pressurized water reactors (PWRs), 61 at boiling water reactors (BWRs), 46 at light water cooled graphite moderated reactors (LWGRs), 51 at pressurized heavy water reactors (PHWRs), 5 at gas cooled reactors (GCRs) and 9 at fast breeder reactors (FBRs)), 28 events in fuel reprocessing or fabrication facilities, 14 events at testing or research reactors and 37 events in other facilities. In the following, the overall trends are discussed for these events.

Figure II.1 shows the number of the INES reports with a rating of Level 2 or higher and that with a rating of lower than Level 2 year by year. The first two years, 1990 and 1991, were the period for a trial use and 3 events and 26 events were reported to the INES during these two years, respectively. Approximately 70 events were reported during the first five years, 1992-1996, after the official use was launched, and more than 50 events of them were rated at lower than Level 2. However, the number of reports decreased to 20-30 events since 1997. While, in particular, the number of events with a rating of lower than Level 2 dramatically decreased, that with Level 2 or higher was almost constant throughout the years.

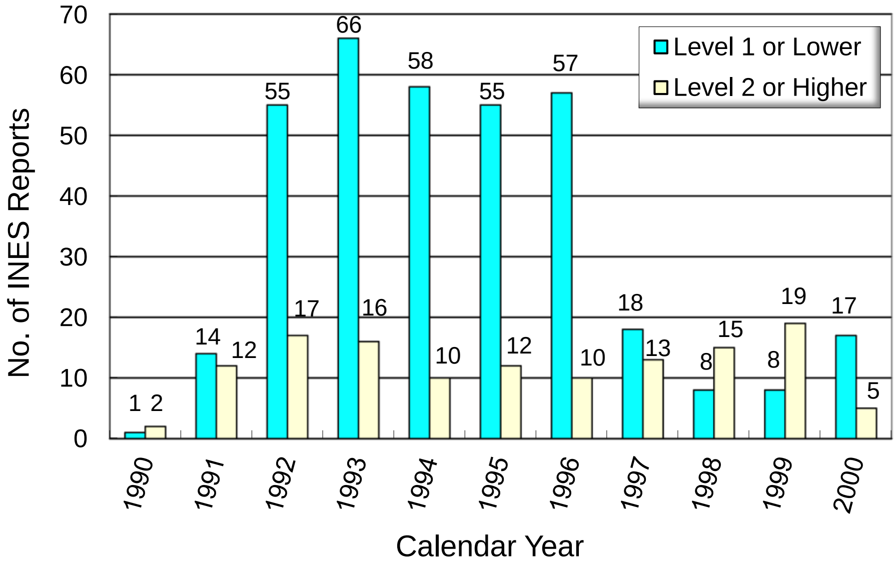


Figure II.1 Number of INES Reports Year by Year

1.2 Characteristics observed in Events with uprating

When rating an event based on the “defense-in-depth” criterion, the uprating may be applied if additional factors were observed; common cause failures (CCFs), procedural inadequacies or lack of safety culture (violation of procedures or limiting conditions for operation (LCOs), quality assurance issues, repetitive human errors, recurring similar events, inadequate control of radioactive materials). The characteristics of such uprated events are discussed below.

There were 58 events uprated, 25 of which were uprated at Level 1, 30 of which were at Level 2, and 3 of which were at Level 3. Focusing on the factors of uprating, 4 events were due to CCFs, 7 events due to procedural inadequacies, and 43 events due to lack of safety culture as shown in **Figure II.2**. As well, 4 events involved both procedural inadequacies and lack of safety culture. The 43 events involving lack of safety culture can be divided into 13 events with violation of procedures or LCOs (4 at Level 1, 7 at Level 2, and 2 at Level 3), 6 events with recurring similar events (2 at Level 1, 3 at Level 2, and 1 at Level 3), 9 events with quality assurance issues (3 at Level 1 and 6 at Level 2) and 15 events with factors unidentified (8 at Level 1 and 7 at Level 2).

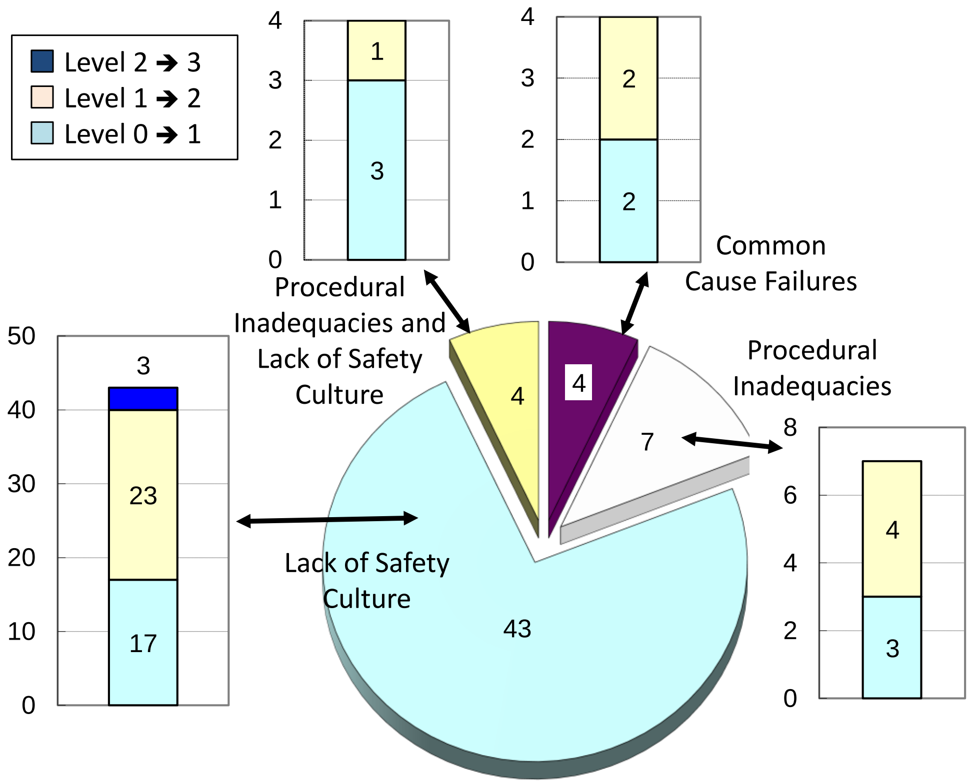


Figure II.2 Number of Events Uprated by Additional Factors

2. Trends on Safety Significant Events

The safety significant event is defined as an event rated at 2 or higher in this study. This subsection describes the trends observed on these events.

2.1 Number of Reports by Level

A total of 131 events were rated at Level 2 or higher. **Figure II.3** shows 114 events at Level 2, 15 at Level 3, and 2 at Level 4.

Being sorted by facility type, 73 of 114 events rated at Level 2 occurred at NPPs, 17 of them in fuel reprocessing or fabrication facilities, and 5 events at testing or research reactors. Sixteen events in other facilities include those in experimental facilities and non-nuclear facilities which used radioactive materials and those involving lost sources. Fifteen events rated at Level 3 consist of 6 events at NPPs, 2 events in fuel reprocessing or fabrication facilities, and 7 events in other facilities. During this period, two events rated at Level 4 were reported to the INES: the JCO criticality accident involving two fatalities in Japan in 1999 and the radiological accident involving two fatalities caused by a lost source in Egypt in 2000. A total of 409 events at NPPs were reported to the INES during this period and only about one-fifth (79 events) were rated at Level 2 or higher, implying that events at NPPs had actively been reported even in the case that it was rated at lower than Level 2.

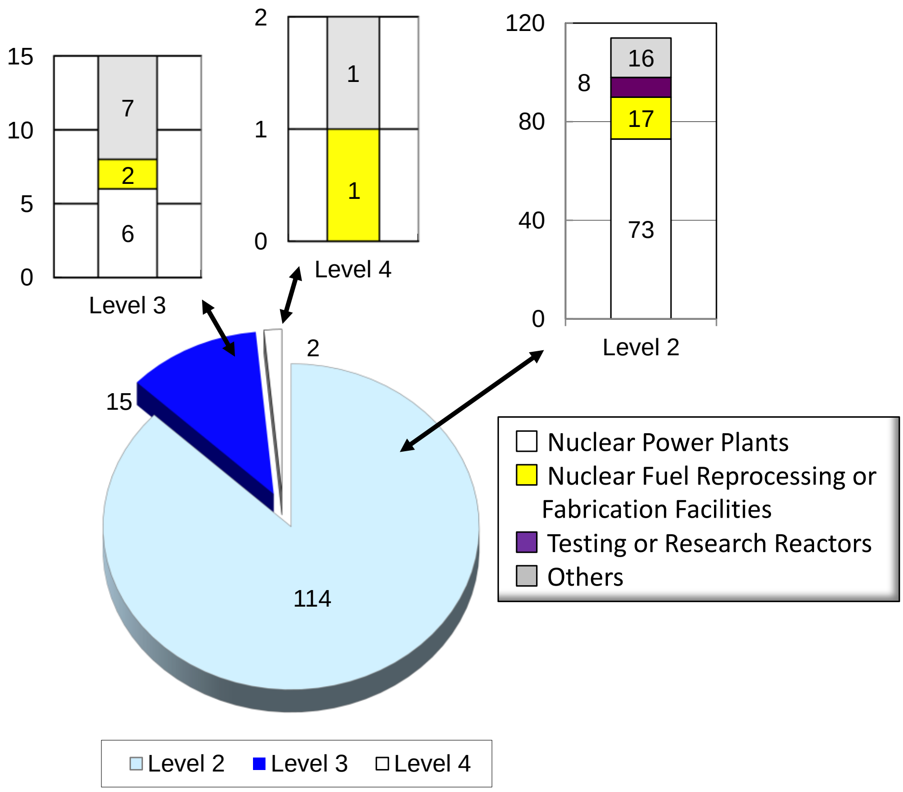


Figure II.3 Number of Reports by Level

2.2 *Number of Reports by Rating Criterion*

The rating of each event is based on three criteria: Impact on People and the Environment (Criterion 1: “offsite consequence” criterion), Impact on Radiological Barriers and Controls at Facilities (Criterion 2: “onsite consequence” criterion), and Impact on Defense in Depth (Criterion 3). From this point of view, the characteristics of events with Level 2 or higher are discussed below.

(a) Events with Level 2

As shown in **Figure II.4**, 27 of 114 events with Level 2 were rated according to the Criterion 2, which consist of 19 events involving overexposure of workers or employees and 8 events associated with radioactive release or contamination inside facilities. The former includes 7 events at NPPs and 7 events in fuel reprocessing or fabrication facilities, most of which attributed to inadequate work control during fuel handling or maintenance activities. For the latter, 6 events in fuel reprocessing or fabrication facilities occurred as a result of radioactive release inside the facilities caused by equipment failures or work control deficiencies but one event at a NPP resulted from a human error. In addition, 83 events were rated based on the Criterion 3, most of which (64 events) took place at NPPs. Thirty-one events were uprated at Level 2. Only 4 events in fuel reprocessing or fabrication facilities were rated according to the Criterion 3. Although there were many events at NPPs caused by equipment failures, none of which involved the radiation exposure or contamination, the events in fuel reprocessing or fabrication facilities resulted in radioactive release inside facilities stemming from equipment failures. The different trends were observed between events at NPPs and those in fuel reprocessing or fabrication facilities. This implies that the difference of multi-layered safety provisions is one of the contributing factors.

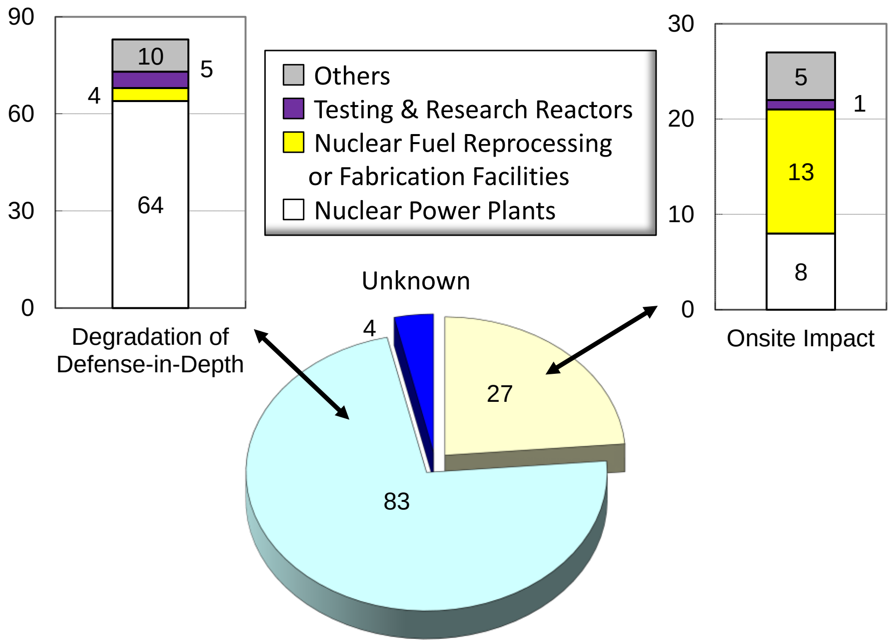


Figure II.4 Number of Reports by Rating Criterion

(b) Events with Level 3

According to the rating criteria, as shown in **Table II.1**, 15 events with Level 3 can be categorized into 2 events based on the Criterion 1, 7 events on the Criterion 2, and 6 events on the Criterion 3. Two events based on the Criterion 1 resulted in overexposure of the public members due to a lost source and in particular, one event involved three fatalities, which is equivalent to the event to be rated at Level 4. At that time, however, the rating criteria were not clear and thus, this event might have been provisionally rated at Lever 3. Four of the 7 events based on the Criterion 2 are associated with overexposure of workers, which took place in the non-nuclear facilities. One event involved high radiation doses to a worker, resulting in amputation of limbs (the Jihua event in 1996). The other three events consist of two involving contamination at NPPs and one involving a lost source. Four of the 6 events based on the Criterion 3 occurred at NPPs and the other two in fuel reprocessing or fabrication facilities. Among them, the event at Smolensk Unit 2 in Russia involved unavailabilities of two emergency core cooling system (ECCS) trains during restart operation and the event at Narora Unit 1 in India involved complete loss of onsite power due to fire in its turbine building. These might have been safety significant events because of the defense in depth being significantly degraded. On the other hand, the event at Kola Units 1 and 2 and the event at the bituminization facility were basically rated at Level 2 and uprated to Level 3 due to lack of safety culture.

Table II.1 Events Rated at Level 3

No.	Facility Name (Type) Country, Event Date	Event Summary	Rating Criterion
1	Bilibino A (LWGR) Russia, 07/10/1991	While transporting cask with fuel fragments, the container struck the pipe located in the upper section of tunnel and fell overboard remaining retained by the safety cable. In this position it was transported to the radwaste repository, resulting in the radioactive contamination of the tunnel surfaces and of the road on the plant territory.	2
2	Smolensk-2 (LWGR) Russia, 07/22/1991	While preparing for restart after maintenance, three of four main safety valves failed to completely close during testing, leading to pressure increase in the sealed room up to the setpoint of emergency protection system. Two of three ECCS subsystems failed to actuate as designed.	3
3	Kola-2 (PWR) Russia, 02/02/1993	While operating at power, transmission lines were damaged due to tornado and as a result, turbine generators tripped and the reactor scrammed. Emergency diesel generators started but tripped due to undervoltage on essential buses. The core was cooled by natural circulation.	3
4	Kola-1 (PWR) Russia, 02/02/1993	While operating at power, transmission lines were damaged due to tornado and as a result, turbine generators tripped and the reactor scrammed. Emergency diesel generators started but tripped due to undervoltage on auxiliary power buses. The core was cooled by natural circulation.	3
5	Narora-1 (PHWR) India, 03/31/1993	While operating at power, a fire occurred in the turbine building. The reactor was tripped manually. Many cables were damaged due to fire, resulting in complete station blackout. Decay heat removal was carried out by natural circulation. Water was fed to the secondary side of steam generators using diesel fire pumps.	3

Table II.1 Events Rated at Level 3 (continued)

No.	Facility Name (Type) Country, Event Date	Event Summary	Rating Criterion
6	Chernobyl-1 (LWGR) Ukraine, 11/27/1995	A considerable contamination of reactor hall floor was detected. Average contamination level was monitored to be 15000-20000 cpm/cm^2 and maximum level monitored 126000 cpm/cm^2. The contamination was caused by a damaged fuel assembly which earlier had been taken out of the reactor.	2
7	Sellafield (Reprocessing) UK, 09/08/1992	Plutonium nitrate leaked from a corroded pipe to the containment cell and accumulated as a crystalline solid mass. There was no release to the operations areas nor the environment.	3
8	PNC Tokai Works (Bituminization Facility) Japan, 03/11/1997	A fire occurred in the cell of the Bituminization Demonstration Facility and ventilation system stopped. Ten hours later, an explosion took place and the confinement of cell and building was degraded.	3
9	Xinshou (Rad. Facility) China, 12/00/1992	One piece of Cobalt-60 source was lost. One person who picked up the lost source and his brother and father died due to overexposure.	1
10	Tianjin (Accelerator) China, 11/21/1995	During the commissioning of high-frequency and high-voltage electron accelerator, five workers entered the irradiation hall to change cooling water pipe. Two of them were injured (burnt) by exposure and treated by skin transgraft.	2
11	Jihua (Chemical Plant) China, 01/05/1996	The Iridium-192 source slipped out from the container of gamma radiograph and dropped on the ground because of the broken safe lock key and the unlocked shutter. A worker picked up the lost source and kept it in the pocket. Due to overexposure, his right leg and left forearm were amputated.	2
12	Treviso (Radiography) Italy, 09/29/1997	Due to a misconnection between the remote control cable and the Cobalt-60 source holder during a non-destructive test, the source was left in the exposure head at the end of irradiation. A worker who collected all the instruments and apparatuses received radiation doses (whole body: 0.98 Sv, hand: 3.56 Sv, eye-lens: 0.89 Sv).	2
13	Not identified (Scrapyard) Turkey, 01/08/1999	Accidental overexposure occurred due to the Cobalt-60 source being broken into pieces by scrap dealers. The five seriously affected patients received an estimated absorbed dose of 3-6 Gy. One of them showed signs of radiation induced burns on two fingers.	2
14	Not identified Turkey, 01/08/1999	An approximately 26 TBq shielded or unshielded Cobalt-60 source was lost. The explanations by the source owner are very confused, leading to the competent authority to consider that the source has been sold and probably sent aboard.	2
15	BECO (Gammagraphy) Peru, 02/20/1999	A radioactive source of Iridium-192 was found out of its container in a construction field of a new hydroelectricity plant. The worker took the source with his hands and put it into his trouser in the pocket. The worker and 5 people around him were exposed. The worker was locally injured severely due to high dose (the local dose was estimated at greater than 50 Gy).	1

(c) Events with Level 4

As indicated in **Table II.2**, two events with Level 4 involved fatalities due to high radiation exposure, one of which is the criticality accident at JCO resulting in two fatalities of workers and the other is due to a lost source in Egypt leading to two deaths of the public members. For the JCO accident, the public members received 10-20 mSv of doses and thus, the event was rated at Level 4 based on the Criteria 1 and 2. Any rating criterion for the event in Egypt was not identified but it seems that this event would have been rated based on the Criterion 1 because of the public fatalities being involved.

Table II.2 Events Rated at Level 4

No.	Facility Name (Type) Country, Event Date	Event Summary	Rating Criterion
16	JCO (Fuel Fabrication) Japan, 09/30/1999	For the homogenization of uranyl nitrate solution, the workers fed several batches of the solution into the precipitation tank using a stainless steel bucket and a funnel. The uranyl nitrate solution in the tank reached a criticality. Three workers were seriously exposed to radiation and two of them died.	1, 2
17	Not identified Egypt, 06/26/2000	The accident resulted from a lost Iridium-192 source found in the way to one of the village's houses. Due to overexposure, the farmer and his son died and the other five family members suffered from radiation sickness.	1

As summary, there are two events based on the Criterion 1 involving the public fatalities due to lost sources. The events rated based on the Criterion 2 involved overexposure of workers or employees, most of which at NPPs and in fuel reprocessing or fabrication facilities attributed to work control issues related to refueling and maintenance activities. For the events based on the Criterion 3, most of them occurred at NPPs, including the safety significant events involving loss of redundancy of ECCS or loss of onsite power supplies.

3. Number of Reports by Facility Type

A total of 131 events with Level 2 or higher can be grouped into 79 events at NPPs, 20 events in fuel reprocessing or fabrication facilities, 8 events at testing or research reactors, and 24 events in the others. The following discusses the characteristics of events by facility type.

3.1 Events at Nuclear Power Plants

As shown in **Figure II.5**, the 79 events at NPPs (73 with Level 2 and 6 with Level 3) consist of 53 at PWRs (51 with Level 2 and 2 with Level 3), 11 at LWGRs (8 with Level 2 and 3 with Level 3), 8 at PHWRs (7 with Level 2 and 1 with Level 3), 4 at BWRs (with Level 2), 2 at GCRs (with Level 2), and 1 at FBR (with Level 2). During the period of years 1990-2000, there were 250-260 PWRs operating per year, which are several times higher than the other types of reactor (BWR: 80-90 units per year, PHWR: 30-40 units per year, GCR: 30-40 units per year, LWGR approximately 20 units per year).

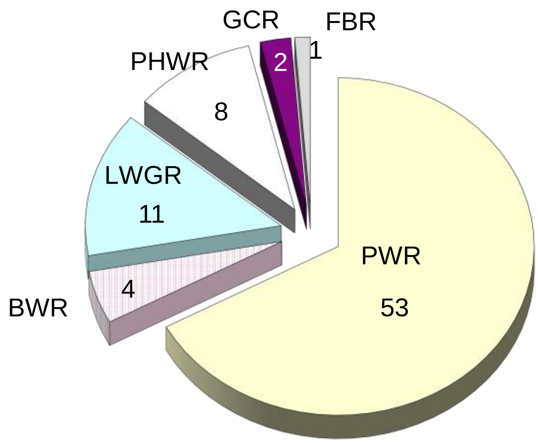


Figure II.5 Number of Reports at Nuclear Power Plants by Reactor Type

Of the 53 events at PWRs, 50 events were rated according to the Criterion 3 and involved failures of safety provisions such as electric power supplies or safety systems, primary coolant leakage, broken feedwater pipe, and loss of decay heat removal or loss of coolant inventory during reactor shutdown. Seven events involved failures in electric power supplies, two of which were initiated by loss of offsite power (LOOP) due to tornadoes followed by inadvertent trips of emergency diesel generators (EDGs) and uprated to Level 3 because lack of safety culture was observed. Five events involved failure to fully insert control rods, two of which occurred at French PWRs and were caused by a structural problem of control rod drive mechanism. This problem was a potentially common issue to all of the same type of French PWRs. Other two events with failure to fully insert control rods occurred at Hungarian PWRs and resulted from intrusion of foreign materials. Three of four events with safety equipment unavailable stemmed from inadequate procedures or violation of rules. The other one involved the internal flooding, which resulted from overflow of the river due to heavy rainstorm, causing plant equipment unavailable. One of three events with the primary coolant leakage took place during reactor shutdown and resulted in a large leakage of 30 m^3/h caused by a crack with 180 mm long in residual heat removal system pipe. Three events involving radiation exposure were caused by work control problems during maintenance such as inadvertent entry into the access limited areas.

Of the four events at BWRs, three involved actual or potential loss of ECCS. In particular, in one event, an assistant operator error shut the valve in component cooling system, making its associated containment spray system, residual heat removal system and EDG unavailable.

Five of the 11 events at LWGRs involved the radiation exposure or contamination (3 exposure events and 2 contamination events). These events took place during fuel handling or maintenance activities and resulted from inadequate work control and violation of procedures. In addition, it should be noted that an event involved loss of two ECCS trains with 3 safety valves unavailable, an event resulted in the turbine building fire due to the leaked hydrogen stemming from the turbine reversing (i.e. motoring) and in another event, the fuel assembly was damaged due to loss of coolant flow in the fuel channel.

Four of the 8 events at PHWRs are associated with the primary coolant leakage, three of which were caused by the failures of relief valve or drain valve. In particular, one event involved a small beak LOCA due to rupture of a relief pipe followed by ECCS actuation. In another event, as well, the turbine building fire resulted in damages of many electric cables, causing a station blackout. This event was rated at Level 3 according to the defense-in-depth criterion.

Two events at GCRs were rated at Level 2, one of which was the partial blockage of fuel channels due to the parts being dislodged from fuel handling machine and the other was two LOOP events with EDG temporarily unavailable.

As mentioned above, at NPPs, there were many events where part of safety systems such as ECCS failed, offsite power or EDG was unavailable, or the primary coolant leaked. The events involving radiation exposure or contamination occurred, during the maintenance or fuel handling, due to work control issues or violations of procedures.

3.2 Events in Fuel Reprocessing or Fabrication Facilities

Sixteen of the 20 events occurred in fuel reprocessing facilities and, as shown in **Figure II.6**, are composed of 6 events involving the contamination due to leakage or release of radioactive materials, 7 events involving the radiation exposure of workers and 3 events with a loss of ventilation due to cable damages, a loss of radioactive confinement due to fire/explosion or inadequate storage of radiation sources in a non-standard transportation cask and a long-term utilization of it.

The remaining four events took place in fuel fabrication facilities, including the JCO accident in 1999. The other three events were the radiation exposure of workers, a long-term loss of ventilation due to electrical system failure and a failure of MOX (mixed oxide) fuel assembly followed by plutonium contamination.

It should be noted that the events involving the radiation exposure of workers and the contamination were mainly caused by work control issues and/or radiation control deficiencies.

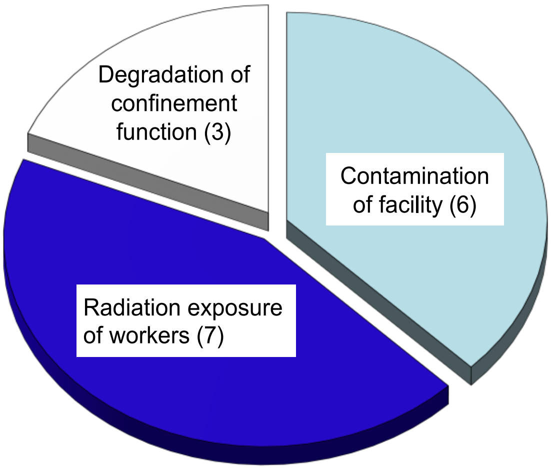


Figure II.6 Event Characteristics in Fuel Reprocessing or Fabrication Facilities

3.3 Events at Testing or Research Reactors

The eight events consist of two involving power excursion due to misoperation, one involving inadvertent radiation exposure of workers, one with the fuel element dropped due to procedural error, one involving thimble failures due to violation of rules, one with coolant leakage due to corrosion, one involving hydrogen explosion followed by trip of ventilation system, and one with control rod control system inadvertently disconnected. The event with thimble failures was uprated to Level 2 because lack of safety culture was observed. Any event at testing or research reactors should be rated at Level 2 or lower based on the defense-in-depth criterion, considering the maximum potential consequence these facilities have, and 7 events were rated at the highest level (Level 2) according to this criterion. As well, it can be observed that most of the events were caused by mistakes or errors of workers.

3.4 Events in Other Facilities

There are 24 events which consist of those in the facilities such as experimental, medical and non-nuclear ones and those involving lost sources. As indicated in **Figure II.7**, more than half of them (15 events) were associated with lost sources and include 6 events involving overexposure of public members. Particularly, two events in 1992 (in Xinshou, China) and 2000 (in Egypt) involved three and two fatalities, respectively. In another event in 1995 (in Jihua, China), the overexposure of a worker was caused by a lost source in the chemical plant construction site, resulting in amputation of his extremities. Eight events associated with lost sources occurred during the transportation, two of which involved overexposure and six of which were rated at Level 1.

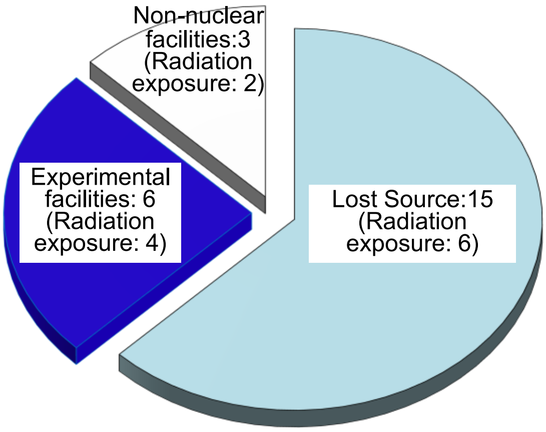


Figure II.7 Event Characteristics in Other Facilities

The other 9 events consist of six at accelerators and experimental facilities and three at non-nuclear facilities. For the former type of event, four involved radiation exposure of workers, in one of which (in Tianjin, China, in 1995), a skin grafting was made for the worker who heavily exposed to radiation due to inadvertent entry to the irradiation chamber. The other two were contamination events, both of which were caused by the release of radioactive wastes into the facilities due to workers' misoperation. For the latter type of event, two involved overexposure of workers when handling the radiation devices due to their errors and the other was the contamination event caused by a broken radiation source in a chemical plant.

The characteristics of such events can be summarized as follows: Events associated with lost sources include those involving fatalities and/or serious physical damages due to overexposure and those involving sources found at inappropriate places such as scrapyards and/or thefts of vehicles with sources. As well, some events with lost sources during transportation have been reported, indicating the importance of enhancing the control of radioactive sources. Events at the accelerators, experimental facilities and non-nuclear facilities involved the overexposure or contamination due to inadvertent workers' actions, implying the importance of improving educational training of workers.

4. Summary

In this generic study, approximately 500 event reports to the INES were analyzed to examine overall trends and characteristics of events rated at Level 2 or higher. The results are summarized as follows:

(1) While approximately 70 events were reported to the INES annually during several years since its official operation was initiated, the number of reports had been decreased afterwards. However, the number of events rated at Level 2 or higher has been remained almost constant (10-20 events per year) throughout the period.
(2) Since additional factors such as common cause failure(s) or lack of safety culture were observed, approximately 60 events were uprated. Lack of safety culture was main reason of uprating and included violations of operating/maintenance procedures or LCOs, quality assurance issues, and administrative control issues.
(3) Two events rated at Level 4 were reported, both of which involved fatalities due to overexposure to radiation. About half of 15 events rated at Level 3 were associated with overexposure resulting from lost sources or radiation devices in non-nuclear facilities, one of which involved a fatality. About two thirds of 114 events with Level 2 occurred at NPPs.
(4) Categorizing the events reported based on the reporting criteria, 5 events including the JCO accident were rated on the “offsite consequence” criterion and resulted in the deaths and/or physical damages of worker(s) or public member(s). The events rated on the “onsite consequence” criterion were dominated by overexposure and/or contamination events in fuel reprocessing or fabrication facilities and overexposure events in experimental or non-nuclear facilities. Although most of the events at NPPs were rated on the “defense-in-depth” criterion, there were only a few such events involving contamination in the plant site and no event involving radioactive releases.
(5) The events at NPPs have different characteristics depending on the reactor types but the degraded functions of ECCS and electric power systems were common issues to all types of reactors. The events in fuel reprocessing or fabrication facilities involving overexposure and contamination were mainly caused by work control and/or radiation control issues during maintenance activities. Although there were a small number of events at testing or research reactors (8 events), those were mainly due to the workers’ errors. As well, the events associated with lost sources had been remarkably increasing, some of which involved fatalities and/or physical damages of public members. Furthermore, the events in the experimental or non-nuclear facilities involved radiation exposures and/or contamination due to workers’ errors.

II.1.2 Comprehensive Reviews of Events Reported to IRS

The IRS is designed to be of value mainly to technical experts working in the nuclear power field and therefore, event information reported is written with technical detail not intended for distribution to the general public because of its proprietary nature. To encourage openness within the nuclear community, as mentioned previously, access to the IRS reports is restricted. Although generic and topical studies have been carried out based on the IRS reports, the results from these studies are also restricted. Thus, only a limited number of publications are available to the general public(28-32). These publications highlighted important lessons learned based on a review of 200-300 event reports submitted to the IRS over the respective three-year periods and provided the short summaries of individual events with safety significance.

1. Outlines of Comprehensive Reviews

The comprehensive reviews have been carried out such that the reported event information can be shared among regulatory authorities and operating organizations to learn from operating experiences in foreign countries and to enhance the safety of NPPs in Japan. In particular, since 1997, the individual IRS reports have been reviewed and the results have been complied in a confidential report on a yearly basis(12-27), which provides their respective event summaries of individual reports and identifies the IRS reports addressing safety significant events or generic issues, and disseminated to the relevant organs in Japan in accordance with the basic principle of IRS.

During the years 1988 to 1996, more than 700 IRS reports submitted to the OECD/NEA were reviewed and approximately 60 reports were identified as ones addressing the events with safety significance. Examples of them are listed in **Table II.3**. Particular attention should be paid to some of them. For example, several reports addressed the control rod insertion problems such as failure of rod assemblies to fully insert and slow insertion time. Other significant reports included those describing the pressure locking of valves, actual or potential ECCS strainer clogging, core shroud cracking, reactor vessel head cracking, turbine fire and missile events, power oscillation events, or LOOP.

Since 1997, more than 1300 IRS reports have been analyzed and approximately 140 reports were identified as safety significant ones. **Figures II.8** and **II.9** show the number of IRS reports reviewed and that of reports identified significant in the individual calendar years, respectively. As well, **Table II.4** lists the safety significant events identified in the individual years.

The 1997 edition of comprehensive reviews covered 99 IRS reports submitted from 20 member countries in the calendar year 1997(12). At that time, a total of 24 countries participated in the IRS. Of these, five IRS reports were identified as significant ones addressing the events with potential or actual degradations of plant equipment due to cold weather conditions, control rod cluster drive mechanism anomaly and so on. The 1998 edition covered 122 IRS reports from 18 member countries(13). Nine of these were identified important, addressing the events associated with water hammer, loss of coolant inventory during reactor shutdown, potential sabotages, LOOP followed by excessive cooldown, or inadvertent containment spray. The 1999 edition analyzed 119 IRS reports from 21 member countries and identified 10 reports describing the events related to adverse effects of fire protection system actuation (internal flooding), inadequate configuration control of safety systems, CCFs, potential degradation of ECCS, or control rod seizure(14). In the 2000 edition, 82 reports from 23 countries were analyzed, 7 of which were identified safety significant. It should be noted that these significant reports addressed the events involving LOOP, external flooding, steam generator tube failure, power oscillation, and weld cracks in reactor coolant system piping(15). The 2001 edition covered 64 reports from 18 countries and identified 6 reports addressing the safety significant events related to CCFs, stress corrosion cracking in safety injection pipes, residual heat removal pump failure during mid-loop operation, or through-wall cracking of reactor vessel head control rod drive mechanism penetration nozzles(16). In the 2002 edition, 55 IRS reports from 16 countries were reviewed and 7 reports were identified safety significant. These reports addressed the events with LOOP, significant degradation of reactor pressure vessel head, primary system leakage, etc(17). The 2003 edition analyzed 70 reports from 20 countries, 9 of which described the safety significant events including pipe rupture due to hydrogen explosion, steam generator tube rupture (SGTR), EDG failures, leakage from reactor vessel bottom mounted instrument nozzles, and potential ECCS sump clogging(18). The 2004 and 2005 editions covered 67 reports from 20 countries and 63 reports from 13 countries, respectively. For both years, 7 reports were identified as significant ones which addressed LOOP, cable penetration fire, damages to fuel assemblies in cleaning tank, pressurizer penetration nozzle cracking, potential ECCS sump clogging, or external flooding by tsunami attack(19,20). The recent editions for the years 2006 to 2012 reviewed approximately 80 reports per year. The reports were submitted by 19, 16, 21, 22, 19, 27 and 24 countries in the individual years. Each edition identified about 10 reports as significant ones. The 2006 edition addressed the significant events involving EDG failures, control rod insertion problem, or foreign material intrusion in ECCS or primary system piping(21). In the 2007, 2008 and 2009 editions, it should be noted that several reports addressed failures of EDGs and their associates. In addition, there were some notable events involving silting up of raw water intake channel of 4 units, potential CCFs of essential service water system (ESWS), safety/relief valve malfunction, or transformer failures(22-24). The 2010 edition identified the significant reports which described the events related to the environmental adverse effects such as fragile icing in cooling water channel, loss of heat sink due to ingress of vegetable materials or seaweeds, or massive ingress of plant debris into raw water pumping station, in addition to the EDG problems(25). The 2011 edition also addressed the events with the EDG problems, pressurizer relief valve failure, or ESWS unavailability due to cold weather condition(26). In the 2012 edition, it should be noted that a station blackout took place during shutdown, resulting in loss of residual heat removal and spent fuel pool cooling. As well, the significant events identified included those involving siphon breaker problems in spent fuel pool and LOOP(27).

Table II.3 Examples of Safety Significant Events Identified during Years 1988-1996

Significant Events Identified	No. of IRS Reports
Control rod insertion problems	4
Scram solenoid pilot valve problems	2
Pressure locking of valves	2
Core shroud cracking	4
Actual or potential loss of decay heat removal during shutdown	8
Power/neutron flux oscillation	4
Actual or potential emergency core cooling system (ECCS) strainer clogging	3
Turbine fire	4
Cracks found on reactor pressure vessel (RPV) head at PWRs	3
Emergency diesel generator (EDG) problems	1
Digital system/component malfunction	3
Loss of offsite power (LOOP)	1

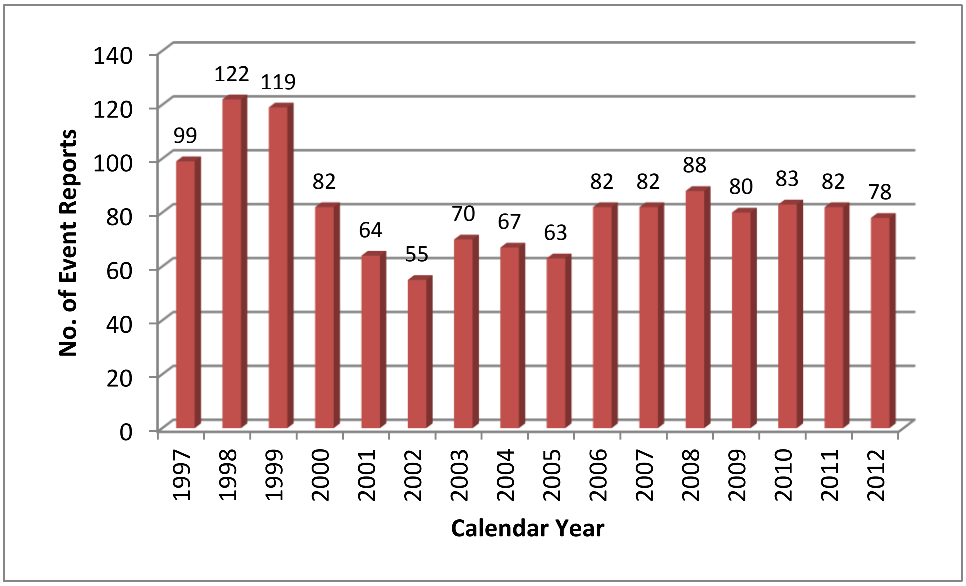


Figure II.8 Number of IRS Reports Reviewed Year by Year

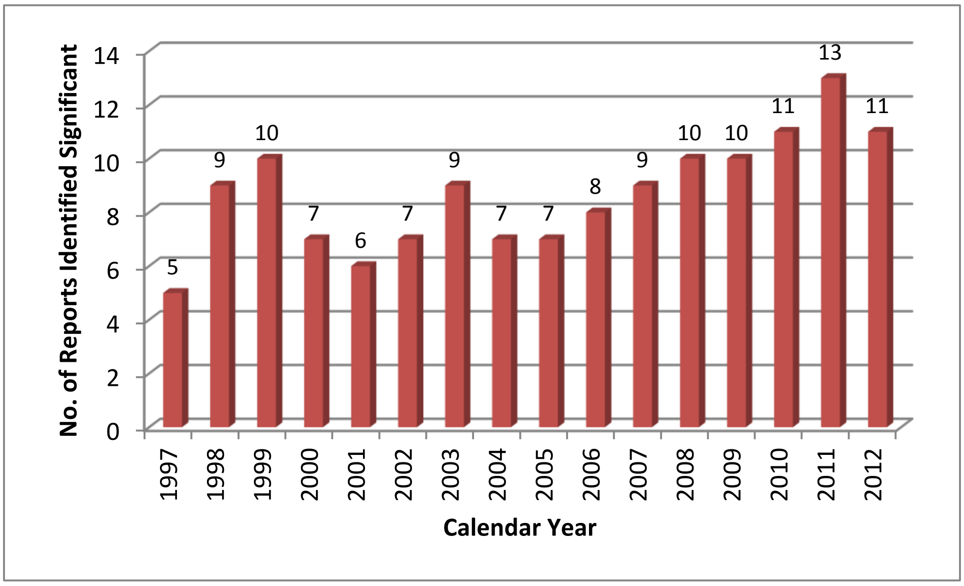


Figure II.9 Number of IRS Reports Identified Significant Year by Year

Table II.4 List of Safety Significant Events Identified during Years 1997-2012

Report Edition	Significant Events Identified	Event Category
1997	- Degradation of cooling water systems due to icing	External hazard
	- Abnormal impact of cold weather on equipment operation	External hazard
	- Localized corrosion in fuels	-
	- Toppling of a new fuel stringer	-
	- Control rod drive mechanism (CRDM) anomaly	Recurring
1998	- Water hammer events (3 reports)	Recurring
	- Loss of coolant inventory during shutdown (3 reports)	Recurring
	- Indications of tampering, vandalism, or malicious mischief	-
	- Excessive cooldown of primary coolant system following loss of offsite power (LOOP)	-
	- Inadvertent containment spray	-
1999	- Adverse effects of fire protection system actuation (3 reports) – Internal flooding (2)	Recurring
	- Configuration control errors (2 reports)	Recurring
	- Common cause failures (CCFs) (3 reports: emergency diesel generators (EDGs), pumps, breakers, valves)	Recurring
	- Potential for degradation of emergency core cooling system (ECCS) and containment spray system due to protective coating deficiencies/foreign material	Recurring
	- Control rod seizure	Recurring
2000	- Cracks in weld area of reactor coolant system piping (2 reports)	Recurring
	- Non-vital bus fault leading to fire and LOOP	External hazard
	- External flooding due to bad weather	External hazard
	- Power/neutron flux oscillation	Recurring
	- Risks in refueling outage	-
	- Steam generator tube failure	Recurring
2001	- CCF of motor operated valves (MOVs)	Recurring
	- Stress corrosion cracking (SCC) in austenite stainless steel components	Recurring
	- Excessive response time from reactor pressure vessel (RPV) level sensors	-

Table II.4 List of Safety Significant Events Identified during Years 1997-2012 (continued)

Report Edition	Significant Events Identified	Event Category
2001	- Potential loss of safety-related equipment due to lack of high-energy line break barriers	-
	- Through-wall cracking of CRDM penetration nozzles	Recurring
	- Defect in residual heat removal (RHR) pump during mid-loop operation	Recurring
2002	- LOOP	Recurring
	- Non-compliances with technical specifications and human-based CCFs	-
	- Primary coolant unisolable leak caused by thermal fatigue	Recurring
	- Refueling error	Recurring
	- Significant degradation of RPV head	New/Unexpected
	- Turbine trip followed by safety injection due to steam generators overfill	Recurring
	- Erroneous design assumptions for refueling water storage and emergency feedwater system tanks	-
2003	- Safety injection (SI) actuated by very low pressurizer pressure protection caused by inappropriate operator manoeuvre	-
	- Pipe rupture due to radiolysis gas explosion	New/Unexpected
	- Failure of reactor core isolation cooling (RCIC) injection valve	Recurring
	- Steam generator tube rupture (SGTR) during plant cooldown	Recurring
	- Degraded 6.6 kV breakers and failure of two EDGs on demand	Recurring
	- Leakage in bottom-mounted instrumentation nozzles (2 reports)	New/Unexpected
	- Emergency sump recirculation blockage issues	Recurring
	- ECCS recirculation valve stuck closed	Recurring
2004	- Fire in secondary electrical penetration	External hazard
	- Loss of bulk electrical system	Recurring
	- Inspection of alloy 82/182/600 materials in pressurizer penetrations & pipe connections	Recurring
	- Damage of fuel assemblies in cleaning tank	New/Unexpected
	- Reactor scram due to loss of non-class uninterruptible power	Recurring
	- LOOP resulting in dual unit scram	Recurring
	- Emergency sump recirculation blockage issues	Recurring
2005	- Reactor shutdown following tsunami strike	External hazard
	- Three-unit trip and LOOP	Recurring
	- Containment sump screen blockage problems (2 reports)	Recurring
	- Tin whisker	New/Unexpected
	- Torus cracking	New/Unexpected
	- Compliance deviation of connection boxes	-
2006	- Loss of 400 kV and failure to start EDGs	Recurring
	- Control rod insertion problem	Recurring
	- Foreign material in ECCS/containment spray piping or primary system (3 reports)	Recurring
	- Insufficient water level in containment spray recirculation sump	Recurring
	- Air entrainment into ECCS/containment spray	Recurring
	- Unavailabilities of low head SI/containment spray pumps due to excessive vibration	Recurring
2007	- Configuration control error	Recurring
	- Silting up of raw water intake channel of 4 units	External hazard
	- Wrong settings of voltage protection relays	Recurring
	- Potential common cause vulnerabilities in essential service water systems	Recurring
	- Double inversion of remote control to safety related valves	New/Unexpected
	- Unavailable containment spray recirculation	Recurring
	- Failure on 6.6 kV emergency switchboard	Recurring
	- EDG problem (2 reports)	Recurring

Table II.4 List of Safety Significant Events Identified during Years 1997-2012 (continued)

Report Edition	Significant Events Identified	Event Category
2008	- EDG problem (2 reports)	Recurring
	- Radioactive particles outside radiation controlled area	New/Unexpected
	- Loss of safety electrical equipment due to generator high voltage	Recurring
	- Unintentional opening of safety/relief valves	Recurring
	- Unavailable RHR and containment cooling system train	Recurring
	- Gas accumulation in ECCS, decay heat removal and containment spray systems	Recurring
	- Cracking on steam generator tube	Recurring
	- Hydrogen hazards	Recurring
	- Failure of containment prestressing cables	New/Unexpected
2009	- Incorrectly installed anchor	-
	- Unavailability of 2 of 3 high head SI lines	Recurring
	- Unavailability of letdown line due to water soluble paper clogging	New/Unexpected
	- EDG problem and LOOP	Recurring
	- Transformer failures	Recurring
	- Excessive drainage of reactor coolant	Recurring
	- EDG inoperable due to painting and cleaning agents	Recurring
	- Configuration control error	Recurring
	- Inadvertent withdrawal of two fuel assemblies	Recurring
	- Transformer fire and intrusion of fire gas into control room	External hazard
2010	- Frazil ice/icing in cooling water system (2 Reports)	External hazard
	- Loss of heat sink due to vegetable materials, seaweed, plant debris (3 reports)	External hazard
	- Power reduction due to external impacts	External hazard
	- Excessive suspended materials in cooling water	External hazard
	- Trip of busbars in switchyard and loss of standby power due to current transformer fire	External hazard
	- EDG problem (2 Reports)	Recurring
	- Submerged electrical cables	New/Unexpected
2011	- Failure on 6.6 kV emergency switchboard	Recurring
	- EDG problems (3 reports)	Recurring
	- Loss of grid supplies to transformer	Recurring
	- 500 kV line trip due to hill fire	External hazard
	- Failure of main transformer resulting in LOOP	Recurring
	- False activation of RPV float level switches and ECCS flow oscillation	-
	- Unavailable reactor protection system containment isolation function	Recurring
	- Failure of pressurizer safety valve to close	Recurring
	- Control rod problem (2 reports)	Recurring
	- Inoperability of essential service water system trains due to cold weather	External hazard
2012	- Complicated events due to fire	External hazard
	- Loss of shutdown cooling due to station blackout	New/Unexpected
	- Dual unit LOOP	Recurring
	- Low voltage safety grade power electronics failure due to lightning overvoltage	Recurring
	- Cracking in RPV bottom head penetration	Recurring
	- Flaw indication in RPV	Recurring
	- Cracking of CRDM housing	Recurring
	- RPV closure head studs de-tensioned	New/Unexpected
	- Flood in containment	New/Unexpected
	- Spend fuel pool syphon breaker problem (2 reports)	New/Unexpected

2. Safety Significant Events Identified by Comprehensive Reviews

As mentioned in the previous subsection, the comprehensive reviews identified a total of approximately 200 IRS reports which addressed safety significant events or generic issues. These can be categorized into three types of events; recurring events, events associated with external hazards, and the events related to new phenomena or unexpected aggravating conditions as shown in Table II.4. In this subsection, the safety significant events identified are summarized focusing on these three types of events.

2.1 Recurring Events

A recurring event is defined as one with actual or potential safety significance that is the same as or similar to the previous events and/or has the same or similar causes as the previous ones. About two-thirds of safety significant events identified during the years 1997 through 2012 (89 of 139 events) belong to this event category. Examples include loss of residual heat removal during reactor shutdown at PWRs, service water degradations due to bio-fouling, power oscillations at BWRs, reactor vessel head stress corrosion cracking at PWRs, and SGTR.

The actual or potential loss of residual heat removal during reactor shutdown was reported from more than 10 PWRs and in particular, several PWRs experienced such events during mid-loop operation, most of which took place during the period 1988 to 1996 as seen from Table II.3. The dominant causes were loss of power to the flow control valves or pumps, procedural errors, inadequate reactor water level instruments. Since 2000, only one event was reported, the cause of which was different from the previous events. In this event, a special type of pump called “canned motor pump” failed due to a lost locking screw for the radial bearing.

At least three events involving an actual or potential STGR were reported, including one at the Mihama Unit 2 in 1991. Both of the other two events also involved the pressurizer level decrease which required the manual actuation of safety injection. In one event, the excessive cooldown was carried out, leading to several conditions that complicated the subsequent event response and delayed the reactor cooling system (RCS) cooldown. As well, the operators were slow to recognize configuration lineup problems that could prevent successful operation of the auxiliary spay system to lower the RCS pressure. The other event occurred during reactor shutdown and the N-16 radiation monitor was out of service, delaying the detection of primary-to-secondary leak (the first indication of leak was at 13 min after the indication of pressurizer level decrease).

During the period 1988 to 1996, several IRS reports addressed the control rod insertion problems where the control rod elements were inserted at a slower rate than usual, or inserted only partly at both PWRs and BWRs. One of dominant causes was that fuel assemblies were found to have been deformed, such that there was not a straight, smooth path for inserting a rod. Other potential causes were as follows: debris (foreign material), control rod swelling, corrosion products, thimble tube bowing, fuel assembly bowing and/or twisting, reduction in thimble tube diameter and/or design tolerances. In addition, the slow rod insertion was observed at several BWRs, which were caused by the degradation of scram solenoid pilot valve diaphragms. Since 1997, as well, control rod insertion problems were observed but the causes were different from the previous ones. For example, at a PWR, 22 of 61 control rods were stuck in the upper position, direct cause of which was 'detention' in the foreheads of the movable and immovable poles of the fixing electromagnet.

Other reports described the core shroud cracking at BWRs, power oscillations at BWRs, actual or potential ECCS strainer clogging at BWRs, reactor vessel head penetration cracking at PWRs, LOOP, EDG failures, and turbine fires. The cracks in the core shroud have been reported from more than 10 plants. These cracks were found within base material or in the heat-affected zone of the welds. The causes of shroud failures were intergranular stress corrosion cracking related to high carbon content in the material, unfavorable heat treatment, and unfavorable welding procedures.

The power oscillation events at BWRs resulted from the operation in the forbidden region (instable region) of the power-flow diagram, skewed power distribution (depending on control rod pattern and fuel loading configuration), and/or low feedwater temperature.

The ECCS clogging events might have resulted from debris generated or materials used/brought inside containment. One event involving strainer clogging took place as a result of piping insulation material being damaged and carried to the suppression pool. Another event involved the strainers being covered with fibers, sludge and corrosion products, potentially leading to their blockage.

The reactor vessel head penetration nozzles were generally made of nickel-based alloys (e.g., Alloy 600), cracking of which was caused by the primary coolant water and the operating conditions through a process called primary water stress corrosion cracking (PWSCC). In particular, at a PWR, a significant cavity in the low-alloy reactor vessel head was found unexpectedly. The cavity was apparently caused by boric acid erosion/corrosion resulting from leakage of reactor coolant from a crack in the control rod drive mechanism nozzle. In this event, the pressure boundary was retained by only the stainless steel cladding with its thickness of 0.3 inch. The susceptibility of vessel head penetrations to PWSCC appears to be strongly linked to the operating time and temperature of the vessel head and thus, the PWSCC-related problem has increased as plants have operated for a longer period of time.

LOOP events occurred in many countries. In some cases, a LOOP was caused by the electrical grid disturbances, which occurred due to equipment failure, overloading, lack of maintenance, human errors, etc. These events are considered risk-significant because they challenge multiple safety systems required to bring the reactor into a safe shutdown condition. In one event, a LOOP was caused by a human error during a protective relay test of the main generator and followed by a EDG failure to start with the other EDG being out of service for maintenance, resulting in station blackout (SBO). The SBO condition lasted for 12 min. The EDG failure was resulted from the air start system failure. In another event, the maintenance activities in the switchyard were not performed according to the applicable procedures by the grid operator and the plant personnel were not informed of the ongoing work in the switchyard. The inadequate maintenance work resulted in a high voltage short circuit. An additional maintenance error at the plant led to a high voltage spark propagating to the uninterruptable power supply system (UPS). As a result, two out of four redundancies of UPS were unavailable for about 20 min, resulting in loss of two 220 VAC bus bars. All four EDGs started automatically but two of them did not connect to their respective bus bars due to loss of power in the 220 VAC grid served by the failed UPSs.

Many events involved the EDG failures caused by problems in the air start system, fuel oil supply system, cooling system, output breakers, lube oil system, voltage regulators, engine itself and so on. In some events, two EDGs failed to start due to a common cause. In such cases, as the likelihood of SBO would increase, a movable or immovable generator such as gas turbine generator has been installed as a backup at plants in European countries and the United States.

At least four major turbine fire events reported resulted from hydrogen leakage, explosion and burn. Two of them involved the turbine blades having broken and penetrated the casing (so-called turbine missiles), leading to the damages to several system pipes including condenser. One was caused by the turbine overspeed due to solenoid valve failures in the turbine trip system and the other was by excessive vibration of turbine. The latter event involved internal flooding due to the damage of tube which circulates cooling water from the lake. In the third event, turbine fire caused the electrical cables to be burned, resulting in an SBO. Although the SBO condition lasted for about 17 h, the core cooling was maintained by the thermosyphoning effect by supplying water to the steam generators with use of diesel driven fire pumps. The fourth turbine fire caused several system failures inducing a significant flooding of reactor building. As a consequence, the core cooling was compromised due to loss of two gas circulators and the regulation problems of feedwater to the main heat exchanger. The fire lasted for 6 h, water extraction from the building was ended 12 h later and feedwater to main heat exchanger was stabilized 19 h after event initiation.

2.2 *External Hazards*

Although nuclear power plant systems, structures and components important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, storms, floods and so on without loss of capability to perform their safety functions, some plants have suffered from natural phenomena, in particular severe weather condition or earthquake induced floods. Nineteen safety significant events during 1997-2012 are placed in this category.

Strong winds and heavy rains resulted in the high waves moving up the river along which the plant is located, causing a partial flooding of two reactor units. Offsite power supplies were lost due to loss of grid and mechanical and electrical penetrations were damaged. The buildings where safety-related components are located were partially flooded, causing portions or all of ESWS, the safety injection system, containment spray system unavailable. The site was momentarily isolated, impairing emergency assistance from the outside. Although the flooding has been taken into account in the plant design, this event revealed that conditions could be encountered where further design improvements are desirable. The disastrous tsunami caused by the massive earthquake off the coast of island hit the site. As a result, seawater entered the pump house through the intake tunnels and water level in the pump house rose up to condenser cooling water (CCW) pump level, causing all the CCW pumps and all the process sea water pumps unavailable. There was seawater inundation of about 0.5 m above the ground/road level up to the east periphery of the turbine building but no seawater entered into the reactor building, turbine building or service building. Thus, the effect of tsunami waves on the plant was minimum and limited to a plant outage only. The tsunami events should be considered in the siting and design of existing and future nuclear facilities.

A riverside plant experienced a serious accumulation of sand in the sand trap. The collapse of the heap of accumulated sand would have blocked the raw water intake tunnel under the river connecting the river water intake and intake channel. Therefore, the site would have totally lost the normal supply of raw cooling water. This situation originated from the weakness of river water intake design, which was very sensitive to environmental condition and resulted from shortcomings in the silting monitoring methods.

At several plants, low temperature challenged the plant safety-related systems such as ESWS. In 2011, particularly, three plants reported the associated events. At one plant, frazil ice was formed in cooling water intake and blocked its protective grid, causing the water level at the entrance of the pumping station below the lowest water level without causing ESWS pumps to malfunction. Another event involving frazil icing resulted in partial blockage of the screen in cooling water channel, causing low water level in the snore pieces of the main seawater pumps. As a consequence, those pumps stopped and pressure in the condensers increased, leading to reactor shutdown. In such cases, ESWS might have been affected, leading to reactor shutdown. As well, at a plant, both independent ESWS trains became inoperable due to the blockage of the valve caused by the freezing water in the valve bonnet.

Several events involved loss of heat sink due to clogging of the water intake structures including trash rack and drum screen by vegetable materials, seaweeds, plant debris and suspended debris. At one plant, a very significant amount of vegetable materials blocked the pumping station intake for two units by clogging the trash rack and drum screens, resulting in one of two ESWS trains being unavailable. Thus, one unit was manually shutdown. After switching to the other ESWS train, its water intake became clogged in turn and as a consequence, a total loss of heat sink occurred at the unit. Another plant experienced significant seaweed ingress to the station cooling water inlet. Both reactors were manually shut down and there was a partial, temporary loss of reactor seawater cooling at one unit. At a riverside plant, three events involving loss of heat sink took place during one month. In each case, the circulating water pumps were tripped on high drum screen head loss due to the fouling of the drum screen by massive ingress of plant debris from the river. The two cases led to the reactor trips of all four units on the site. In the third case, two units tripped.

2.3 *New Phenomena or Unexpected Aggravating Conditions*

The events in this category cover a pipe rupture due to hydrogen burning at BWRs, a fault on a solid state protection system (SSPS) circuit card due to the growth of tin whisker, a failure of prestressing cable of the containment dome, and a syphon breaker problem on spent fuel pool cooling (SFPC) lines. This category includes 17 events identified safety significant.

Two events involved a pipe rupture due to the explosion of radiolysis gases (hydrogen and oxygen) which accumulated during power operation. One event occurred during the surveillance test of high pressure coolant injection system (HPCI), where the steam condensation pipe of residual heat removal system (RHR) ruptured. The other event took place, in which the reactor pressure vessel head spray (RPVHS) line ruptured. In the former event, it was presumed that the noble-metal which was used for suppressing the stress corrosion cracking, would work as a catalyst at the time of ignition. For the latter case, the ignition source had remained unknown but it was assumed that the ignition may have started in the area of the flange in the ruptured pipe.

Tin whiskers are electrically conductive crystalline structure of tin that sometimes grows from surface and would cause short circuits in electronic system by bridging closely spaced circuit element. One event involving tin whisker resulted in an unexpected safety injection and reactor trip caused by a fault on an SSPS circuit card. Tin whiskers appear to have increased following the efforts to remove alloying metals from solder and other circuit card. Tin whiskers have been cited as the cause for various minor component failures in the nuclear industry.

At a plant, control measurement of prestressing forces showed a complete loss of prestress on both ends of one dome prestressing cable. The direct cause of this failure was the exhaustion of wire plastic deformation capacity on the outer part of ring reel surface that led to the violation of strength limit due to the combination of design factors (value of wire stress and wires number, radii of bends, possibility of crossings), manufacturing, assembling and stretching technology factors (radii of reels and pulleys, wire rectification method, cable introduction method, wire tension uniformity), and operational factors (checks by hydraulic lifts, temperature changes).

The syphon breakers are provided to interrupt the excessive drainage of spent fuel pool in the case of a leak, rupture or alignment error in an SFPC line. One plant operator checked the operability of the syphon breakers and found that these devices were not present on the pipes submerging in the pools of two units because they had not drilled at the time of construction. In addition, the diameter of the syphon breakers was less than the specified value at other plants. In another country, it was found that the syphon breaker valve had never been tested functionally and was not included in the inspection program. Both events were identified by the inspection carried out in the wake of the accident at the Fukushima Dai-ichi Nuclear Power Plant.

3. Summary

The comprehensive reviews covered more than 2000 IRS reports during the period 1988 through 2012 and identified approximately 200 safety significant events. Safety significant events can be roughly categorized into several groups. In this section, described are three groups of events: recurring events, events associated with external hazards and the events related to new phenomena or unexpected aggravating conditions.

In particular, the recurring events indicate that the previous corrective actions have not necessarily been effective to prevent the recurrences and thus, the event analysis should place more emphasis on why the actions taken after the first event have failed to prevent the recurrences. As for the external hazards, more attention should be paid for ones beyond the design basis assumptions based on the previous events which challenged the plant safety. The new phenomena or unexpected aggravating conditions have a potential to threaten the plant safety but it seems difficult to detect them prior to their appearances. However, such events might have been observed in other fields of technologies and thus, special attention should be paid to the operating experience in such fields, especially in the design phase.

The results of comprehensive reviews provided inputs to the topical studies described in Chapter III.

II.2 Supporting Tools for Generic Studies

It is required for the comprehensive reviews to look at many event reports written in English and to identify the events with safety significance or safety implications. Therefore, the reviews take much effort because of a large number of reports and the results should be accumulated so that the significant events can be easily reached if requested. In order to reduce such effort and promote the effective use of review results, two types of software tools were developed as follows:

- Computerized Event Sequence Abstracting System (CESAS) [33-37]
- World Wide Web for Database of Japanese Translation on INES Reports[1]

II.2.1 Development of Computerized Event Sequence Abstracting System

When analyzing event reports, it is necessary to identify occurrences such as component/system failures/unavailabilities and operators' actions from beginning to end of the incident and then, to understand the sequential and causal relationships between occurrences, that is, event sequence. Such an event sequence is useful for determining the causal factors which have the potential to lead to another event, identifying the events which have similarity in terms of plant behaviors, predicting the occurrences which could exacerbate the plant conditions and examining the preventive measures for recurrence of events[38]. For the purpose of more efficiently utilizing event reports, however, it is necessary to systematically extract the event sequences from individual event reports.

On the other hand, most of event reports from foreign countries are generally written in natural language of English and there are a large number of reports. Therefore, it takes much effort, in particular, for Japanese to grasp the gist of event description and to extract the event sequence from these reports. Aiming at reducing such effort, a new computer-based analysis system, named Computerized Event Sequence Abstracting System (CESAS), was developed using the knowledge engineering technique used in the field of natural language processing such as machine translation[33-37]. This system analyzes the narrative description written in English, systematically identifies sentences or clauses representing occurrences and schematically abstract event sequence. To implement such processing on a computer, the following three processes are required; (i) analyzing grammatical structures of individual sentences and clarifying modification relations between words and the subject-predicate relations, (ii) understanding the meaning of each word and identifying the phrases, clauses and sentences representing occurrences, and (iii) recognizing the relations between the phrases, clauses and sentences. As for (i), applied was the existing analysis method which was developed and has been utilized in the machine translations and for (ii) and (iii), the analytical approach was newly developed. In this section, the analysis approach incorporated into CESAS is described and analytical results are shown in comparison with manually-extracted event sequences to verify its feasibility.

1. System Design

1.1 Basic Concepts

Machine translation is generally executed through the analysis of original sentences, the transformation of sentence structures and the generation of target sentences(39-42). The process in the machine translation is processed by using the dictionaries which contain grammatical and semansiological information of words, and the rules which contain the grammars for original and target languages. The machine translation cannot necessarily translate all styles of sentences and thus, the controls are introduced or the modification of original text is required prior to translation(43). Considering such circumstances in the machine translation technique, the system design was performed based on the following concepts.

(a) Since the dictionaries and rules are needed to analyze the natural language, the system is configured based on the knowledge engineering technique, enabling the system to have dictionaries and rules independently from computer programs and to manage the variety of sentence styles by modifying the dictionaries and rules.

(b) The existing analysis method is applied to the process of analyzing grammatically sentences.

(c) The dictionary is newly prepared taking into consideration the frequent appearances of words specific to event reports and/or nuclear power plant systems/components.

(d) The rules are prepared for the basic grammar of English.

(e) A new analytical approach is developed for the processes of understanding the meaning of words and recognizing the relations between phrases, clauses and sentences.

1.2 System Configuration

As shown in **Figure II.10**, CESAS consists of one dictionary, three analysis rules and one model as knowledge-base, and a four-step analytical process. The analytical process is composed of morphemic analysis, syntactic analysis, semantic analysis and syntagmatic analysis, each of which is outlined below, together with analysis rules and a model.

1) Morphemic Analysis

   Morphemic analysis is a pre-process of syntactic analysis to identify the part-of-speech for each word in sentences by referring to the word dictionary.

2) Syntactic Analysis

   Syntactic analysis defines phrases and clauses and determines the grammatical structure of each sentence by collating a row of words (row of parts-of-speech) with the English grammatical rules (syntactic rules).

3) Semantic Analysis

   Semantic analysis consists of two steps; semantic structure analysis and extraction of occurrence expressions. The former clarifies the semasiological relations between words and phrases and identifies the subjects, predicates, objects and so on using the semantic rules. The latter extracts phrases and clauses representing occurrences by applying the semantic model.

4) Syntagmatic Analysis

Syntagmatic analysis recognizes phrases and clauses indicating the identical items and occurrences and then, deduces mutual relations between phrases, clauses and sentences representing occurrences with use of the syntagmatic rules.

The analysis method proposed by Marcus for the machine translation[44] was applied to the morphemic and syntactic analyses. For the semantic structure analysis, the existing approach[45] was modified so that the semasiological relations could be identified.

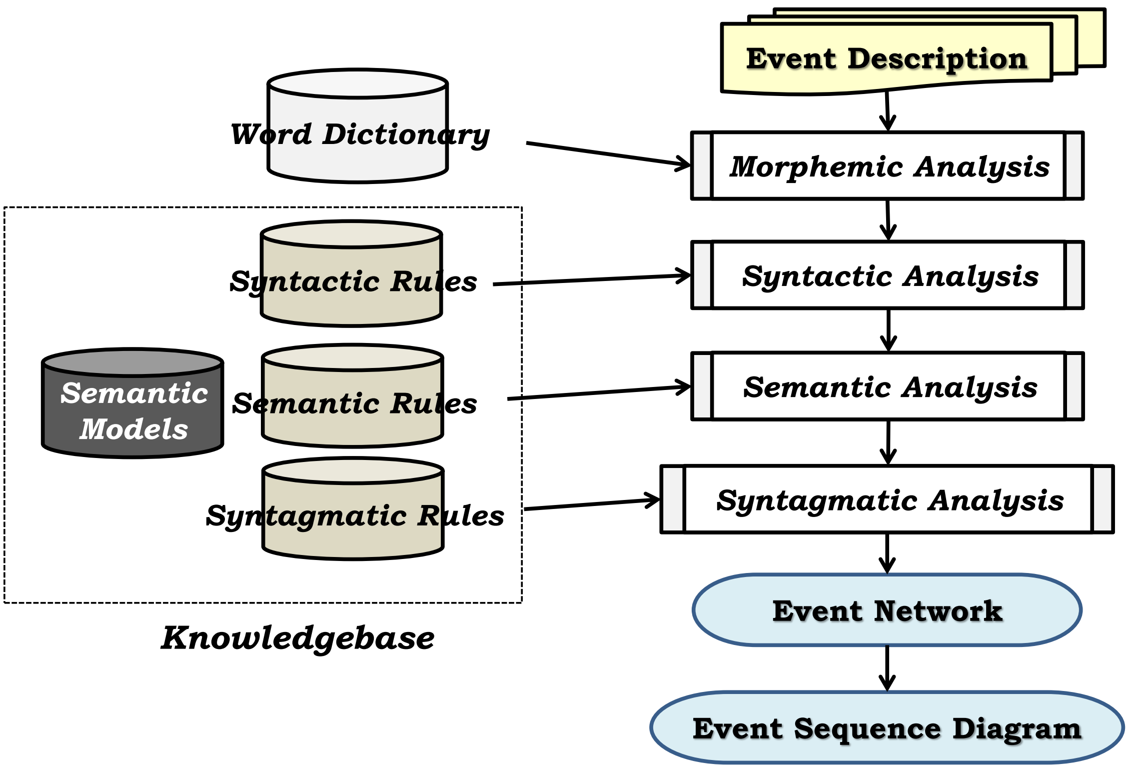


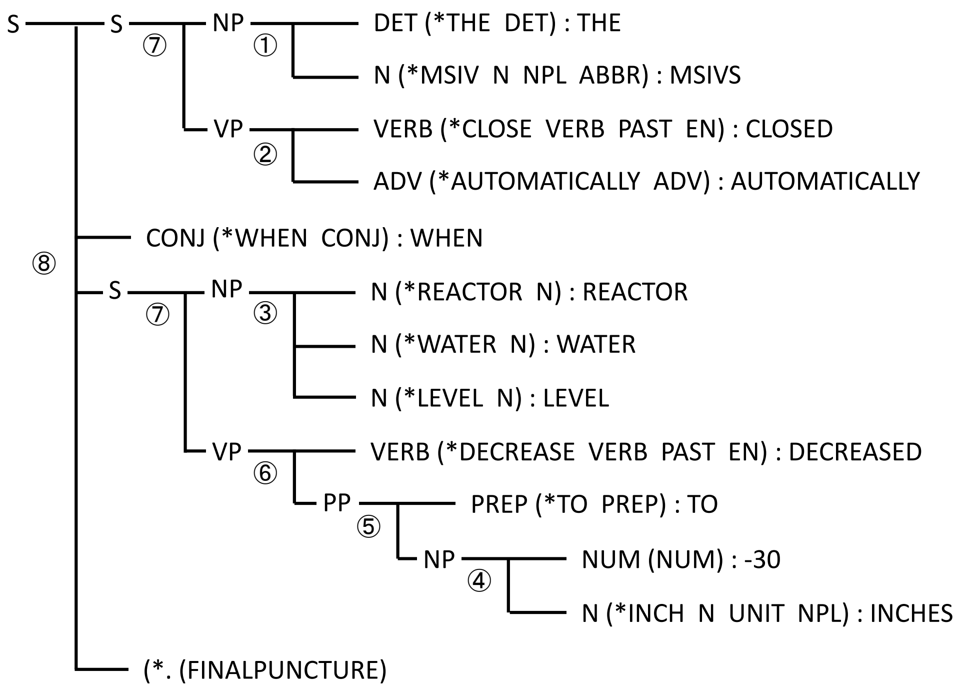
Figure II.10 Structure of CESAS

2. Analytical Approach

2.1 *Morphemic and Syntactic Analyses*

In the morphemic analysis, each sentence in an event description is firstly divided into words and phrases. Next, the part-of-speech for each word or phrase is defined by referring to the word dictionary. The dictionary contains syntactical properties such as part-of-speech and conjugations for words and phrases. Nomenclatures and abbreviations of equipment, systems and components at nuclear power plants are also stored in this dictionary.

In the syntactic analysis, phrases and clauses are determined for each sentence by collating a row of parts-of-speech with the syntactic rules. Then, the grammatical tree structure (syntactic tree) of sentence is constructed according to combinations of the phrases and clauses. The syntactic rules were prepared by quoting from Ref. (44). **Figure II.11** shows an example of syntactic tree for a sentence, *"The MSIVs closed automatically when reactor water level decreased to -30 inches"*. As seen from this figure, several kinds of phrases such as noun-phrase (NP) and verb-phrase (VP) are identified by applying syntactic rules (①～⑥ in Figure II.11) and then, grammatical structure of the sentence is determined with rules such as ⑦ and ⑧.



(NOTE)

DET: demonstrative or definite article	N: noun	NUM: numeral
PREP: preposition	CONJ: conjunction	ADV: adverbial
NPL: plural	PAST: past tense	EN: past particle
ABBR: abbreviation	*: original form	S: sentence
NP: noun phrase	VP: verb phrase	PP: prepositional phrase

(SYNTACTIC RULES APPLIED)

① DET + S = NP	② VERB + ADV = VP	③ N + N + N = NP
④ NUM + N = NP	⑤ PREP + NP = PP	⑥ VERB + PP = VP
⑦ NP + VP = S	⑧ S + CONJ + S = S	

Figure II.11 Example of Syntactic Trees

2.2 Semantic Analysis

It is necessary to understand their respective meanings of words and identify the words and phrases representing occurrences. In the event description, occurrences are often represented in predicative verbs, subjective nouns and objective ones underlined as shown below.

"The reactor tripped automatically."
"The level decrease caused the reactor trip."

Taking into account such features of event description, the semantic analysis is divided into two parts; (i) analyzing semantic structure (semantic structure analysis) and (ii) extracting occurrence expressions (occurrence extraction).

The semantic structure analysis utilizes the existing analysis method[(45)] to identify grammatical parts such as predicates, subjects and objects based on the syntactic trees using the semantic rules. For a simple sentence, the predicative verb is obviously determined. A compound or complex sentence is divided into two simple sentences and then, the predicative verb of each sentence is identified. After that, defined are their respective subjects, objectives and complements of individual simple sentences according to the semantic rules shown in **Table II.5**. As seen from this table, the subjective part is a noun-phrase in front of the predicate and the objective or complementary part is a noun- or adjective-phrase in rear of the predicate. In addition, the noun-phrase is analyzed to determine its main noun and antecedent. The main noun is defined as the word located at the end of noun-phrase and the word located ahead is regarded as an antecedent. For example, in the noun-phrase, *"reactor water level"*, the main noun is defined as the word, *"level"*, and its antecedent is the word, *"water"*. The word, *"reactor"*, is the antecedent of *"water"*.

Table II.5 Semantic Rules

<table>
<tr><th>Syntactic structure</th><th>Five standard patterns</th></tr>
<tr><td>NP + Vi (+ ADV or PP)</td><td>S – V</td></tr>
<tr><td>NP + Vi + ADJ</td><td>S – V – C</td></tr>
<tr><td>NP + Vt + NP (+ ADV or PP)</td><td rowspan="3">S – V – O</td></tr>
<tr><td>NP + Vt + to do</td></tr>
<tr><td>NP + Vt + that-clause</td></tr>
<tr><td>NP + Vt + NP + NP (or that-clause)</td><td>S – V – O – O</td></tr>
<tr><td>NP + Vt + NP + to do</td><td rowspan="2">S – V – O – C</td></tr>
<tr><td>NP + Vt + NP + ADJ</td></tr>
</table>

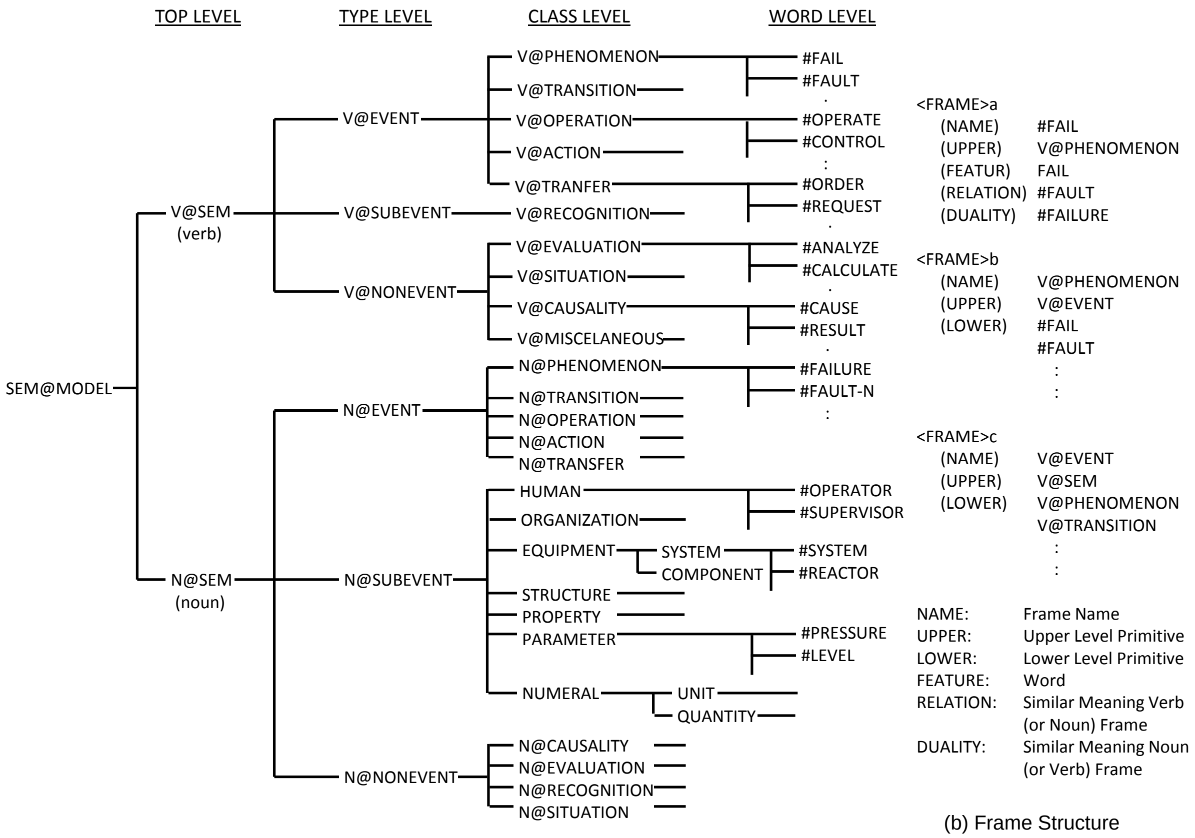
NP: Noun Phrase, Vi: Intransitive Verb, Vt: Transitive Verb,
ADV: Adverbial, PP: Prepositional Phrase, ADJ: Adjective,
S: Subject, V: Predicate, O: Object, C: Complementary

The occurrence extraction process determines whether or not the individual phrases and clauses represent occurrences, such as failure or actuation of plant systems/components, and identifies the noun-phrases representing plant systems/components and plant parameters, such as pressure and water level, to clarify what happened by making out the semantic primitives of nouns or verbs. The semantic primitives are classified into several concepts such as objects, phenomena, operations and so on according to the meanings of nouns or verbs. The semantic primitives are described in the form of *"FRAME"* organized in a hierarchical structure as shown in **Figure II.12**. This hierarchical structure is called "semantic model" here. In this model, as seen from Figure II.12(a), the lowest level (*WORD-LEVEL*) is a node corresponding to each noun or verb and its upper level (*CLASS-LEVEL*) is one standing for a semantic marker common to nouns or verbs which belong to one group and is divided into 26 categories (10 for verbs and 16 for nouns). For example, a node, *V@PHENOMENON*, is a semantic marker for verbs representing failure or anomaly such as *"fail"*, *"fault"* and *"rupture"* and a node, *V@OPERATION*, is one for verbs representing system/component response such as *"open"* and *"operate"*. Each node is an element of the node, *V@EVENT*, which stands for occurrence, in the upper level, *TYPE-LEVEL*. In the process of CESAS, the semantic primitive of each noun or verb is identified by tracing the semantic model from the *WORD-LEVEL* to the *TYPE-LEVEL*. If the *TYPE-LEVEL* for a noun (or verb) is *N@EVENT* (or *V@EVENT*), in other words, it is recognized that the noun (or verb) represents an occurrence. As well, a node, *V@SUBEVENT*, is prepared for verbs such as *"find"* and *"notice"* and the sentence where such a verb is the predicate is regarded as an occurrence if its subject or object represents an occurrence as indicated below.

"The valve <u>failure</u> was found."
"Licensee speculated that the valve <u>failed</u>."

Furthermore, in the semantic model, the synonymous words are defined in the "*FRAME*" as shown in Figure II.12(b). For example, a verb, *"fail"*, is defined as synonymous with a verb, *"fault"*, and a noun, *"failure"*, by linking the node, *#FAIL*, with the nodes, *#FAULT* and *#FAILURE*, via *RELATION* and *DUALITY*. *RELATION* is used to link two verbs (or nouns) and *DUALITY* is to link a verb (or noun) with a noun (verb).

As a result of the semantic analysis, the syntactic tree is transformed into a semantic tree shown in **Figure II.13**, which contains the grammatical relations among the subjects, predicates, objects, and so on together with the semantic information for each noun or verb.



(a) Hierarchical Structure

(b) Frame Structure

Figure II.12 Semantic Model

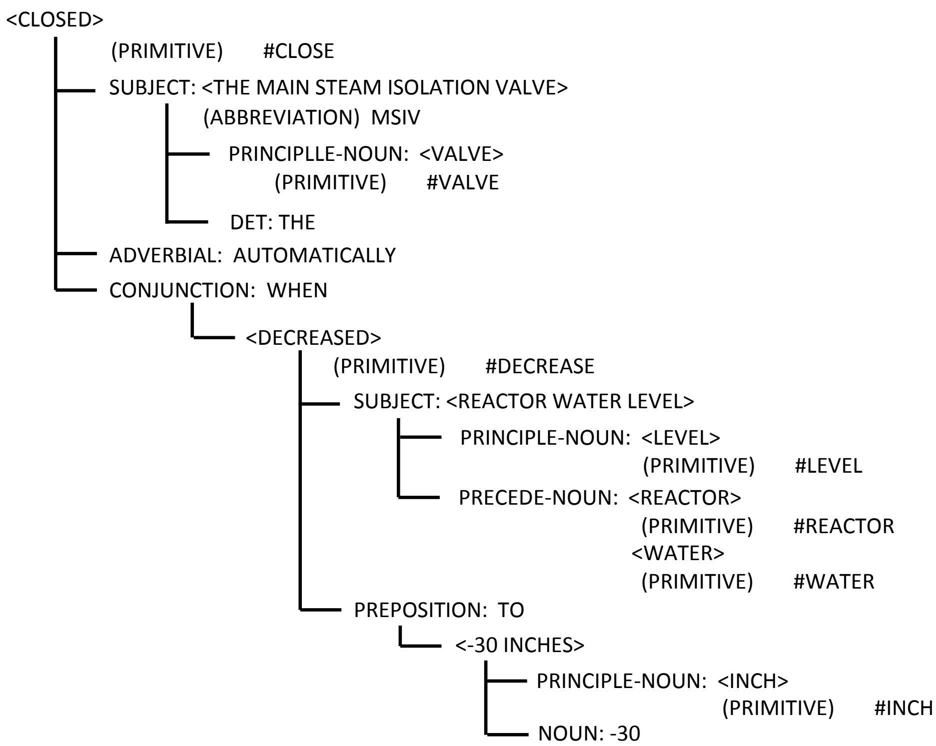


Figure II.13 Example of Semantic Trees

2.3 Syntagmatic Analysis

In the event description, the same systems/components, parameters or occurrences often appear in the different expressions. For example, systems or components are expressed in their respective full names and abbreviations (e.g. *"safety relief valve"* versus *"SRV"*) and a definite article or a demonstrative adjective is used to refer to the items or occurrences previously described (e.g. *"the valve"* or *"this valve"*). As well, some occurrences can be expressed in different grammatical styles; the sentence and the noun-phrase (e.g. *"reactor tripped"* versus *"reactor trip"* or *"reactor scram"*).

The syntagmatic analysis firstly recognizes phrases and clauses representing the identical items such as systems, components, or parameters and the same occurrences, and then defines the sequential and causal relationships between occurrences, using the syntagmatic rules. The sequential and causal relationships are defined focusing on how a specific system/component or operators responded, how the parameter varied, and what caused the occurrence.

In order to recognize the phrases and clauses indicating the identical items or occurrences, the syntagmatic analysis uses three processes as described below.

(a) Correspondence analysis

This process clarifies the correspondence relation among noun-phrases, where their respective main nouns are the same or synonymous, by searching (i) the noun-phrases with the same row of words and (ii) those expressed in the full names and abbreviations and then identifies the same items and occurrences. In the former search process, it is required that all the following rules be met;

Rule 1: antecedent nouns (phrases) are in agreement,
Rule 2: antecedent adjectives are in agreement, and
Rule 3: antecedent quantifiers are in agreement.

Additionally, the above three rules are used to take the correspondence between noun-phrases with the same or synonymous meanings in different expressions. For example, two noun-phrases, *"the failure of the valve"* and *"the valve failure"*, are regarded as the same occurrence based on these rules. As a result of the semantic structure analysis for the former noun-phrase, the noun located ahead of preposition (*"failure"*) is identified the main noun and the noun located behind preposition (*"valve"*) is an antecedent noun. Therefore, its semantic tree structure is the same as that for the latter noun-phrase. Applying the above rules, the noun-phrases with the main noun being the same or synonymous can be identified. As well, the noun-phrases, *"reactor trip"* and *"reactor scram"*, can be recognized identical if the relation between two words, *"trip"* and *"scram"*, is defined as synonymous by *RELATION* in the semantic model shown in Figure II.12. In the search process (ii), surveying the abbreviation dictionary, the individual abbreviations are temporarily replaced by their respective full names and the noun-phrases are recognized identical.

(b) Anaphoric analysis

In this process, the identical systems/components, parameters and occurrences are recognized by identifying phrases, clauses or sentences which a definite article or a demonstrative adjective refers to. The anaphoric relation is determined between the noun-phrase with a deictic (definite article, pronoun, demonstrative adjective) and any preceding noun-phrase which meets the following two rules;

Rule 1: the main noun is the same as or synonymous with that of the phrase with a deictic, and

Rule 2: the antecedent noun or noun-phrase is in agreement or includes that of the phrase with a deictic,

Such a relation is also clarified between the noun-phrase with a deictic and any preceding clause or sentence if it meets the two rules as follows;

Rule 3: the predicate is synonymous with the main noun of the phrase concerned, and

Rule 4: the subject is a noun or noun-phrase which has the same row of words as the phrase concerned or which includes the antecedent noun or noun-phrase of the phrase concerned.

Consider the case that the four sentences appear in the order as follows.

"A turbine-driven feedwater pump tripped." ①
"A motor-driven feedwater pump started." ②
"The pump trip was caused by the stop signal." ③
"The motor-driven pump continued to run." ④

This analysis process recognizes that the noun-phrase, *"the pump trip"*, of the Sentence ③ refers to the Sentence ① using the Rules 3 and 4 and that the noun-phrase, *"the motor-driven pump"*, of the Sentence ④ refers to the subject, *"the motor-driven feedwater pump"*, of the Sentence ② using the Rules 1 and 2.

(c) Relevance analysis

This process searches the noun-phrases with the row of words partially in agreement to recognize the same kind of systems/components and parameters and the similar occurrences. The rules on partial agreement in row of words are;

Rule 1: the rows of two or more consecutive words are in agreement or the row of synonymous words is used, and

Rule 2: neither correspondence nor anaphoric relation is defined.

In this analysis, for example, it is recognized that two noun-phrases, *"turbine-driven feedwater pump"* and *"motor-driven feedwater pump"*, represent the same kind of components and two noun-phrases, *"manual reactor trip"* and *"reactor trip"*, are similar occurrences.

(d) Keyword and Sequence analysis

After the identical systems/components, parameters and occurrences are recognized through the above processes, the sequential and causal relationships are deduced among the occurrences. These relationships are determined by searching the words that represent the sequential order, such as *"after"* and *"before"*, and the causality, such as *"cause"* and *"result in"*. Here, such words are called keywords. Additionally, the appearance order of sentences representing occurrences is used to deduce the sequential relationships. The analysis processes based on the keywords and the appearance order are described below.

i) Keyword analysis

The relationships between occurrences are often indicated using the keywords such as *"before"*, *"after"*, *"cause"*, *"result in"* and so on. In this analysis process, the relationships are deduced by applying the keyword rules as shown in **Table II.6**. In the rules, keywords are classified into two types; one representing a sequential relationship and the other representing a causal relationship. In addition, each type is categorized into several groups and the relationship is specified for each group, according to their respective meanings and parts-of-speech of individual keywords.

The keyword analysis firstly clarifies the relation between two phrases, a phrase and a clause, or two clauses located before and after a keyword. Then, the sequential or causal relationship is deduced if the phrase(s) or clause(s) is recognized as an occurrence. For example, the following sentence includes two the predicates, *"close"* and *"decrease"*, representing occurrences and the keyword *"when"* provides a link called *REVERSE LINK*, that is, the relation from the subordinate clause (or object) to the main one (or subject).

"The MSIV closed when reactor water level decreased."

Thus, the sequential relationship between the two clauses is deduced as follows.

"reactor water level decreased" → *"MSIV closed"*

Table II.6 Example of KEYWORD Rules

Keyword type		Relationship for E1 (Keyword) E2	Keywords
Sequential	Conjunction	E1 → E2 (REGULAR LINK)	before, until, till
	Preposition		before, until, till, prior to
	Conjunction	E2 → E1 (REVERSE LINK)	when, as, while, after
	Preposition		after, during
	Verb	E1 → E2 (REGULAR LINK)	(Active Voice) follow, ensue from, succeed (Passive Voice) precede
		E2 → E1 (REVERSE LINK)	(Active Voice) precede (Passive Voice) follow, ensue, succeed
Causal	Conjunction	E2 → E1 (REVERSE LINK)	because, since
	Preposition		because of, due to, as a result of
	Verb	E1 → E2 (REGULAR LINK)	(Active Voice) cause, result in, lead to
		E2 → E1 (REVERSE LINK)	(Active Voice) result from (Passive Voice) cause

E1, E2: sentence, clause or phrase representing occurrence.

As well, in the sentences shown below, the two noun-phrases representing occurrences, *"the MSIV closure"* and *"reactor water level decrease"*, are linked by the keywords, *"result from"* and *"cause"*, respectively.

"The MSIV closure resulted from reactor water level decrease."

"The reactor water level decrease caused the MSIV closure."

The keyword, *"result from"*, provides a *REVERSE LINK* and the other keyword, *"cause"*, provides a link called *REGULAR LINK*, that means the relation from the main clause (or subject) to the subordinate one (or object). Hence, the above two sentences are converted to the following causal relationship.

"reactor water level decrease" → *"MSIV closure"*

ii) Sequence analysis

Although the keyword analysis deduces the sequential or causal relationship between phrase(s) and clause(s) in one sentence, the sequence analysis determines the sequential relationship among two or more sentences according to their appearance order. Only the sentences written in the past or perfect tense are selected as those to be analyzed. If the keyword is included in the sentence, the relationships between the sentence and the others in anteroposterior positions are defined in consideration of the relationship deduced by the keyword, as shown below.

- In the case that the sentence include the keyword with *REVERSE LINK*:
  anterior sentence → subordinate clause → main clause → posterior sentence
- In the case that the sentence include the keyword with *REGULAR LINK*:
  anterior sentence → main clause → subordinate clause → posterior sentence

Also, this analysis specifies their respective changes/variations of the identical systems, components or parameters according to the appearance order of sentences if the sentence(s) and/or clause(s) meet the following conditions.

Condition 1: the respective main nouns of the subjects represent a component, system or parameter,
Condition 2: the subjects have the correspondence or anaphoric relations, and
Condition 3: the predicates are written in the past tense.

For example, the subjects, *"safety relief vale"* and *"SRV"*, of two sentences shown below are determined to be the same by the correspondence analysis and their respective predicates, *"opened"* and *"closed"*, are regarded occurrences. Thus, by linking these two sentences, it is recognized that the safety relief valve changed from the closed position to the open position and then, from the open position to the closed position.

"The safety relief valve opened."
"The SRV closed successfully."

The results from the syntagmatic analysis are delineated in the form of event network as shown in **Figure II.14**. In this figure, the occurrence expression is denoted in a box of the dotted lines. For instance, the denotation, *<<<#TRIP-N0000>>>*, is a noun-phrase representing the occurrence and the actual expression, *"THE TRIP"*, is indicated in the field of "*DESCRIPTION*". The denotation, *<<<#DECREASE0000>>>*, is a sentence or clause representing the occurrence and its subject, predicate and prepositional phrase are indicated in the field of "*SUBJECT*", "*VERB*", and "*PREP*", respectively. The relationship between occurrences is denoted as a broken arrow (--->) and the reason of each relation is provided in the field of "*FOLLOWED-BY*". For example, *"AFTER"* or *"WHEN"* is the sequential or causal relation defined by keyword analysis, *"SEQUENCE"* is the temporal changes/variations of the identical items by sequence analysis and *"CONTEXT"* is the sequential relation by the appearance order. In addition, the identical systems, components or parameters are given in the field of "*COR-LINK*" (by correspondence analysis) or "*REF-LINK*" (by anaphoric analysis). In the Figure II.14, the denotations, *<<<LEVEL0001>>>*, *<<<LEVEL0002>>>* and *<<<LEVEL0003>>*, are defined as the identical parameter, *"REACTOR WATER LEVEL"*. As well, it is recognized that two occurrences, *"the reactor trip"* and *"HPCI had started"*, are identical with the occurrences previously identified by *"the reactor tripped"* and *"HPCI started automatically"*, respectively.

```
----- <<< #TRIP-N0000 >>> ------------------------------
DESCRPTION               : (THE TRIP)
FOLLOWED-BY              : AFTER                          ----------> (#DECREASE0000)
---------------------------------------------------------
----- <<< #DECREASE0000 >>> ----------------------------<---------- (#TRIP-N0000)
VERB                     : (DECREASED)
SUBJECT                  : #LEVEL0001
PREP                     : (TO #SETPOINT0000)
FOLLOWED-BY              : SEQUENCE                       ----------> (#CONTINUE0000)
---------------------------------------------------------
     <<< #LEVEL0001 >>> : (REACTOR WATER LEVEL)
     <<< #SETPOINT0000 >>>  : (THE LOW LEVEL SETPOINT)
----- <<< # CONTINUE0000 >>> ---------------------------<---------- (#DECREASE0000)
VERB                     : (CONTINUED to DECREASE)
SUBJECT                  : #LEVEL0002
FOLLOWED-BY              : SEQUENCE                       ----------> (#DECREASE0002)
---------------------------------------------------------
     <<< #LEVEL0002 >>> : (REACTOR WATER LEVEL)
         COR-LINK       : #LEVEL0001
----- <<< #DECREASE0002 >>> ----------------------------<---------- (#CONTINUE0000)
VERB                     : (DECREASED)
SUBJECT                  : #LEVEL0003
PREP                     : (TO #INCH0001)
FOLLOWED-BY              : WHEN                           ----------> (#CLOSE0000)
---------------------------------------------------------
     <<< #LEVEL0001 >>> : (REACTOR WATER LEVEL)
         COR-LINK       : #LEVEL0001
     <<< #INCH0001 >>>  : (-30 INCHES)
----- <<< # CLOSE0000 >>> ------------------------------<---------- (#DECREASE0002)
VERB                     : (CLOSED)
SUBJECT                  : #VALVE0000
FOLLOWED-BY              : CONTEXT                        ----------> (#START0000)
---------------------------------------------------------
     <<< #VALVE0000 >>> : (THE MAIN STEAM ISOLATION VALVES)
         ABRIDGMENT     : MSIV
```

Figure II.14 Example of Event Network

3. Application

Using CESAS, several event descriptions were analyzed and the results were compared with the manually-extracted event sequences in terms of (i) occurrences extracted, (ii) the identical occurrences, systems/components or parameters, and (iii) the sequential and causal relationships between occurrences.

Because it is difficult to prepare general rules needed for syntactically analyzing various styles of sentences, the styles which can be analyzed with CESAS are limited to the followings:

- simple sentence which corresponds to either of the five sentence patterns,
- compound sentence which consists of two or three simple sentences, and
- complex sentence which includes one embedded sentence such as that-clause.

Before analyzing event descriptions with CESAS, manual preparatory modifications of some sentences are needed so that CESAS can syntactically analyze them. For example, the event description was modified as shown in **Figure II.15**, where the modified sentences are provided with an asterisk. In this example, a participial construction has to be modified by supplementing a subject, and a complex sentence has to be broken up into analyzable sentence styles shown above. In addition, what a pronoun such as "*this*" or "*them*" refers to should be preliminarily defined in order to precisely recognize the identical occurrences or items. Levels of manual preparations required are grouped into the followings;

- Level 0: no modification (that is, the original sentence).
- Level 1: routine modifications in syntax such as breakup of the sentence and nominalization of the gerund.
- Level 2: partial changes in semantics such as supplementation of the subject for a participial construction and clarification of the pronoun, and
- Level 3: sweeping changes of the sentence into an analyzable style.

While operating at 100% steady state power, a spurious reactor high pressure signal resulted in a reactor scram.
*   While the unit was operating at 100% steady state power, ……...
Reactor pressure was at 1005 psig and the high pressure trip setpoint was at 1035 psig.
The variation in pressure with time is shown in Figure 1.
The licensee speculated that workmen cleaning the floor near the pressure sensing instrumentation may have bumped the racks, causing the high pressure signal.
*   The licensee speculated that 'workmen who were cleaning the floor near the pressure sensing instrumentation may have bumped the racks.   The licensee speculated that 'the bump caused the high pressure signal.'
The A and B reactor protection system channels are located on separate panels, but are physically close to each other.
*   The A and B ………., but the A and B are physically close to each other.
Reactor water level decreased to the low level setpoint of 15 inches soon after the trip and continued to decrease since the main turbine had not yet been tripped off.
*   Reactor water level decreased ………. the trip, and reactor water level continued to decrease since ……...
The main steam isolation valves (MSIVs) closed automatically when reactor water level decreased to -30 inches.
The high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) started automatically.
The recirculation pumps tripped off automatically.
Reactor water quickly recovered and upon reaching the high level setpoints, the HPCI and RCIC tripped off.
*   Reactor water quickly recovered, and the HPCI and RCIC tripped off when reactor water reached the high level setpoints.
Reactor pressure began increasing with MSIVs closed.
*   Reactor began increasing with MSIVs closure.

* Modified sentence for CESAS system

Figure II.15 Example of Event Description

Table II.7 summarizes the number of modified sentences level by level. By making the level 1 modification, the number of sentences which CESAS can analyze turned into 70-80% of the total number of sentences in each description.

After the above modification, event description was analyzed with CESAS. Based on the analytical result given in the form of event network, the event sequence diagram shown in **Figure II.16** was prepared to delineate the CESAS-extracted event sequence more comprehensively and to compare it with manually-extracted one. In this figure, the solid and dotted lines denote the CESAS- and manually-extracted sequences, respectively. Also, the relationships marked with (a) and (b) mean those by the keyword analysis and those by the sequence analysis. For example, the two occurrences, *"main turbine had not tripped"* and *"reactor water level continued to decrease"*, are linked as a causal relationship by the keyword, *"since"*. The following three occurrences are recognized as the temporal variation of the identical parameter, *"reactor water level"*.

"reactor water level decreased to the low level setpoint"
"reactor water level continued to decrease"
"reactor water level decreased to -30 inches"

Table II.7 Number of Modified Sentences

Reference of Reports	Modification Levels				Total
	Level 0	Level 1	Level 2	Level 3	
Information Notice IN 85-50	21	10	5	0	36
Information Notice IN 86-47	24	9	11	0	44
NUREG-0909 (Executive Summary)	39	10	17	1	67
LER 86-028	28	9	9	1	47
Nuclear News, Sep. 1986 (P.24)	17	6	20	3	46
Nuclear News, March 1985 (P.80)	13	12	11	0	36
Nuclear News, May 1985 (P.70)	22	10	6	3	41
IAEA/NEA-IRS 236	12	4	2	0	18
IAEA/NEA-IRS 745-02	34	8	7	1	50
IAEA/NEA-IRS 375	15	11	9	0	35
IAEA/NEA-IRS 327	15	4	2	0	21
NUREG/BR-0051 Vol.9, No.5 (P.4)	27	12	9	2	50

By diagraming the narrative event description as shown in Figure II.16, the occurrences and their relationships can be more easily understood as follows;

(i) *the workmen cleaning the floor bumped the racks → a spurious reactor high pressure signal → reactor scram*

(ii) *reactor scram → reactor water level decreased to low level setpoint → reactor water level continued to decrease → reactor water level decreased to -30 inches → HPCI and RCIC operation → reactor water level recovered → reactor water level reached to high level setpoint → HPCI and RCIC tripped*

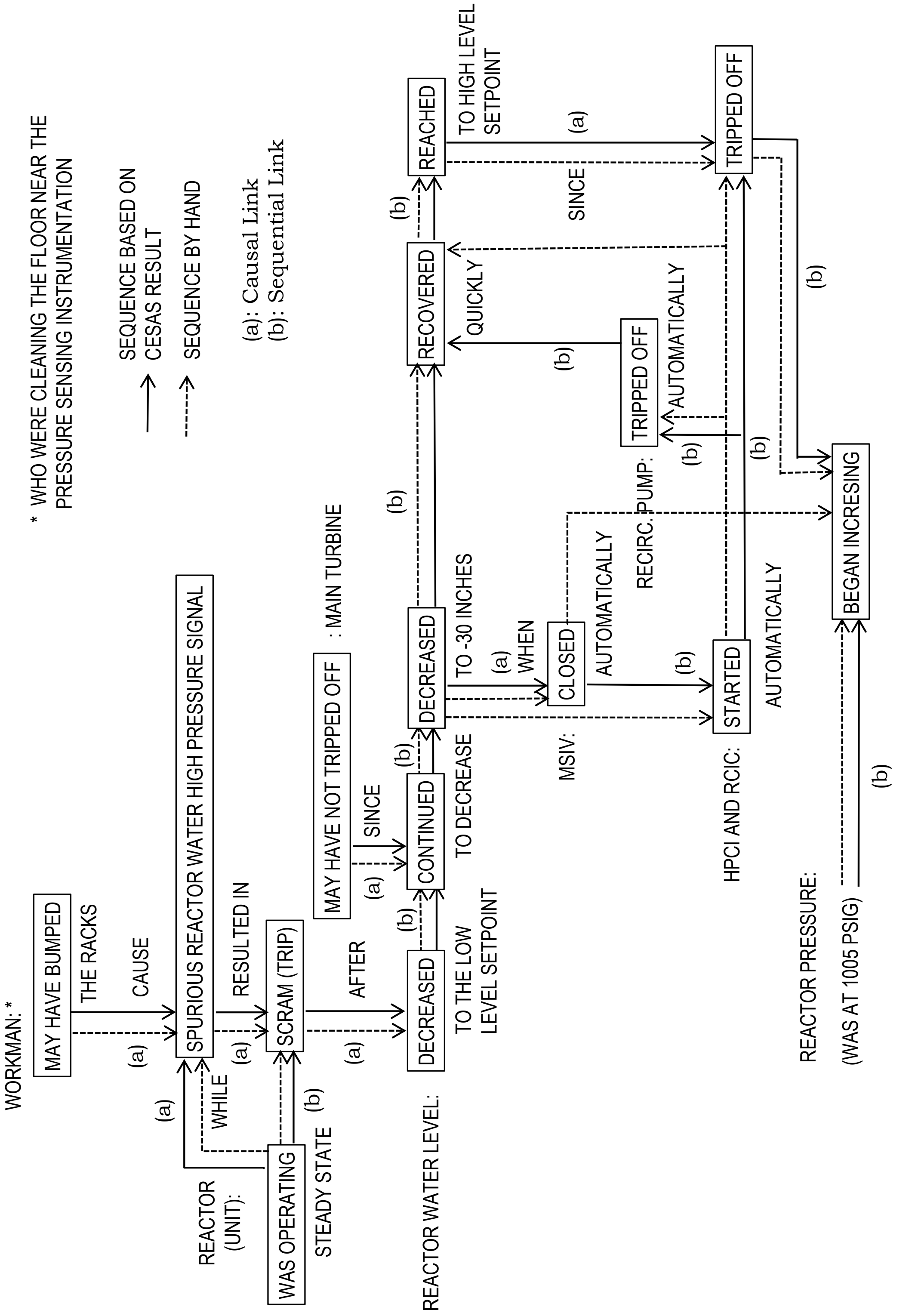


Figure II.16 Event Sequence Diagram

Comparing the analytical results with manually-extracted event sequences, the occurrences identified by CESAS are in good agreement with those obtained by manual work. However, there are differences between the two results for the identical items and the relationships recognized. Although the two noun-phrases, *"reactor water level"* and *"reactor water"*, are identical in actual, the CESAS defines them similar parameters instead of the identical one because they have the same row of words but their respective main nouns are different from each other (*"level"* versus *"water"*). Strictly speaking, however, the original description has an ambiguity on this matter, which results in such a difference. As well, observed are the differences in the relationships as follows: that between the two occurrences, *"reactor pressure began increasing"* and *"MSIVs closed"*; that between the occurrence, *"HPCI and RCIC started"*, and its direct cause; and that between the occurrence, *"reactor water recovered"* and its direct cause. The first difference is due to the difficulty in understanding accurately semantics of preposition, "with". The second and third differences stem from the lack of knowledge on nuclear power plant systems, that is, expertise. Although, in other words, a human analyst can recognize that the occurrence, *"HPCI and RCIC started"*, was due to *"reactor water level decrease"*, and the occurrence, *"reactor water recovered"*, was a result of *"HPCI and RCIC started"* if he/she is familiar with nuclear power plant systems, the CESAS defines the relationships between them according to the appearance order of sentences because of these relations being not explicitly described.

The comparative studies on several event descriptions are summarized in **Table II.8**. Depending on the individual event descriptions, the occurrences recognized by CESAS and manual work are in agreement 90% or more and for the identical items, the two results match to the extent of approximately 80%. The CESAS-extracted event sequences are consistent with the manually-extracted ones about 80%. The differences observed can be summarized as follows.

Table II.8 Comparison of CESAS and Manually-Extracted Event Sequences

Reference of Reports	Comparison Results (Degree of Agreement)		
	Occurrence Extraction	Recognition of Identical Item	Identification of Relationship
Information Notice IN 85-50	90% ~	70% ~ 80%	70% ~ 80%
Information Notice IN 86-47	90% ~	80% ~ 90%	80% ~ 90%
NUREG-0909 (Executive Summary)	~ 90%	80% ~ 90%	80% ~ 90%
LER 86-028	~ 90%	70% ~ 80%	80% ~ 90%
Nuclear News, Sep. 1986 (P.24)	~ 90%	80% ~ 90%	70% ~ 80%
Nuclear News, March 1985 (P.80)	~ 90%	80% ~ 90%	70% ~ 80%
Nuclear News, May 1985 (P.70)	~ 80%	70% ~ 80%	60% ~ 70%
IAEA/NEA-IRS 236	90% ~	90% ~	80% ~ 90%
IAEA/NEA-IRS 745-02	~ 90%	80% ~ 90%	70% ~ 80%
IAEA/NEA-IRS 375	~ 90%	80% ~ 90%	80% ~ 90%
IAEA/NEA-IRS 327	~ 90%	80% ~ 90%	80% ~ 90%
NUREG/BR-0051 Vol.9, No.5 (P.4)	~ 90%	80% ~ 90%	80% ~ 90%

- CESAS cannot recognize the occurrence expression with an adjective in the case that it represents an anomaly such as *“high flux”* and *“an unavailable pump”*. In such cases, the noun-phrases are regarded as a parameter and component, respectively, according to the semantic model.
- As for recognition of the identical occurrences, there are two major differences: one is due to the identification of occurrences repeated by several sentences in the past tense and the other is the interpretation of synonymous expressions. For the former case, CESAS recognizes two sentences as independent occurrences because both of them are written in the past tense despite the identical occurrence. The latter case stems from failure to identify the actual relationship between two or more synonymous phrases. Since the actual meaning of a noun-phrase such as *"the actuation”* or *"the operation”* depends on the context, for example, a noun-phrase, *“the actuation”*, may refer to *"valve closure"* in some cases and *"pump start”* in other cases. As well, CESAS cannot define the correspondence relations between a single word and a noun-phrase, both of which are synonymous, such as *“depressurization”* and *“pressure decrease”*, and anaphoric relations between two noun-phrases where the past participle of verb is used as a premodifier and postmodifier such as *“the failed valve”* and *“the valve failed”*.
- The inconsistencies in sequential and causal relationships are attributed to the lack of expertise and the differences in recognition of the identical occurrences as described above. In addition, CESAS cannot identify the correct sequential or causal relations if a preposition conditionally plays a role of causal keyword or if there are ambiguous expressions stemming from syntactic or semantic vagueness in the description. For example, the preposition, *“on”*, actually represents the causal relationship between the two occurrences, *“reactor trip”* and *“high flux”*, in the sentence, *“the reactivity addition resulted in a reactor trip on high flux”*, but these relationships cannot be defined by CESAS because it is very difficult to model semantics of such preposition. Also, the relation defined by a conjunction may be incorrectly recognized in the case that what the subordinate clause syntactically qualifies is inconsistent with the semantics as seen from the sentence as follows; *"the plant was operating at 100% power with one main feedwater pump in manual control because problems in automatic had been experienced”*. This sentence includes three occurrence expressions underlined. The conjunction, *“because”*, actually denotes a causal relationship between the two occurrences, *"one main feedwater pump in manual control”* and *“problems had been experienced”*. In the syntactical point of view, however, the subordinate clause usually qualifies the main clause. Therefore, CESAS recognizes that the occurrence, *“problems had been experienced"*, causally relates to

the occurrence, *"the plant was operating"*. Since syntactically vague statements and ambiguous meanings, as mentioned above, cannot be precisely analyzed, some modifications of sentences are needed in such cases.

- Furthermore, it is assumed in the CESAS analysis that sentences are chronologically ordered in the event description. This assumption may produce the differences from the actual relationship between occurrences.

Through the analyses of event reports with CESAS, the suitable styles of event description for CESAS were examined. Event descriptions written in the following styles can be easily and automatically analyzed with CESAS to extract event sequences.

- The event description is written in a plain style without use of syntactically and semantically complicated expressions.
- Each sentence is described without using any pronoun so that it could clearly represent occurrences or relations between occurrences, and
- Sentences in the event description are chronologically arranged for plainly representing the sequential relations between occurrences.

4. Summary

For the purpose of efficient utilization of event information, a new computer software package, CESAS, was developed. CESAS is to extract systematically the event sequence, which is sequential and causal relationship between occurrences, from the event description written in natural language of English. This system is based on knowledge engineering technique utilized in the field of natural language treatment. The analytical process in this system consists of following three steps: (i) to analyze each sentence syntactically and semantically, (ii) to identify sentences, clauses or phrases representing occurrences, (iii) to deduce the mutual relationship between occurrences. An existing syntactic analysis approach was applied to the above the step (i). As for the steps (ii) and (iii), a new analytical approach was developed.

Main features of CESAS are summarized as follows:

- Occurrences are systematically extracted from the event description through syntactic and semantic analyses.
- The sequential and causal relationships between occurrences are systematically abstracted according to the expressions in the description.
- Preparing event sequence diagram based on the analytical result makes it easier to understand plant system responses and operators' actions during the event and to

identify the causes of occurrences.

As well, comparative studies showed that CESAS-extracted event sequences generally agreed with manually-extracted ones but identified several difficulties as follows:

- Extraction of occurrence expressions with an adjective such as *"high pressure"* and *"low level"*.
- Recognition of the identical occurrence which is repeatedly represented by several sentences in the past tense
- Identification of the relation defined by a preposition such as *"on"* and *"with"* which conditionally plays a role of the sequential or causal keyword. Additionally, CESAS requires manual modification of sentences to eliminate syntactic and semantic vagueness.

Through the comparative studies, the effectiveness and feasibility of CESAS were demonstrated despite some technical difficulties to be overcome and the technological perspectives were obtained on establishing computer aid tool for analyzing event information.

II.2.2 Development of Web-Based Database for Japanese Translation of INES Reports

The International Nuclear Event Scale (INES) is a means designed for providing prompt, clear and consistent information related to nuclear or radiological events and facilitating communication between the nuclear community, the media and the public. Aiming at achieving the main objectives of INES and utilizing more efficiently the event reports, the individual reports written in English have been promptly translated into Japanese and the Japanese translation versions have been accumulated and published through a world wide web (WWW) based database(1). This section briefly describes the information stored in the database and discusses technical use of the INES reports and the availability/effectiveness of the database.

1. Information Stored in Database

Until 2001, individual events are reported the INES in the form shown in **Figure II.17**. Although the form was changed in the end of 2001, the contents were basically the same as those in the previous form. Therefore, the database was designed to store all the information, except for "contact person", provided in the previous form. Specifically, the following items are stored in the database.

<table>
<tr><td colspan="15">・ THE INTERNATIONAL NUCLEAR EVENT SCALE (INES)</td></tr>
<tr><td colspan="15">EVENT RATING FORM (ERF)
To be sent to INES coordinator, IAEA, WAGRAMERSTRASSE 5, P.O. BOX 100, A-1400 VIENNA, AUSTRIA
FAX: , E-MAIL: , PHONE</td></tr>
<tr><td>EVENT TITLE</td><td colspan="12"></td><td colspan="2">EVENT DATE</td></tr>
<tr><td rowspan="2">RATING</td><td rowspan="2">RATING DATE</td><td rowspan="2">OUT OF SCALE</td><td rowspan="2">BELOW SCALE</td><td colspan="8">ON SCALE</td><td colspan="3">SAFETY ATTRIBUTE</td></tr>
<tr><td>0</td><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>6</td><td>7</td><td colspan="2">Degr. Defence in Depth</td><td></td></tr>
<tr><td>Provisional ☐
Final ☐</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="2">On-site Impact
Off-site Impact</td><td></td></tr>
</table>

COUNTRY		FACILITY NAME		FACILITY TYPE	

ASPECT OF SIGNIFICANCE TO THE PUBLIC	YES	NO
ACCIDENT ☐ INCIDENT ☐ DEVIATION ☐		
RADIOACTIVE RELEASE OFF-SITE	☐	☐
RADIOACTIVE RELEASE ON-SITE	☐	☐
WORKERS INJURED BY RADIATION	☐	☐
WORKERS INJURED PHYSICALLY	☐	☐
PLANT SAFETY IS UNDER CONTROL	☐	☐
THE EVENT REPORTED IS DISCOVERY OF A DEFICIENCY BY ROUTINE SURVEILLANCE	☐	☐
A PRESS RELEASE WAS MADE (IF YES, PLEASE ATTACH IT)	☐	☐

SHORT DESCRIPTION OF THE EVENT:

JUSTIFICATION OF THE RATING:

<table>
<tr><td rowspan="4">CONTACT PERSON FOR FURTHER INFORMATION</td><td>NAME</td><td></td></tr>
<tr><td>ADDRESS</td><td></td></tr>
<tr><td>PHONE</td><td></td></tr>
<tr><td>FAX</td><td></td></tr>
<tr><td colspan="3">Please attach additional information on justification of the event rating and difficulties encountered, if needed.</td></tr>
</table>

Figure II.17 Event Rating Form of INES

- Event title
- Event date
- Rating type: provisional or final
- Rating date
- Rating scale: out of scale, below scale, on scale 1-7
- Safety attributes: degradation of defense in depth, onsite impact, offsite impact
- Country: the place where the event occurred
- Facility name: the facility where the event occurred
- Facility type: nuclear power plant (PWR, BWR, PHWR, GCR, etc.), research/test reactor, radwaste facility, irradiation facility, fuel fabrication facility, reprocessing facility, enrichment facility, mining/milling facility, research/experimental facility, transportation, radiation source
- Type of events: accident, incident, or deviation (below scale)
- Aspects of significance to the public: offsite radioactive releases, onsite radioactive releases, workers injured by radiation, workers injured physically, plant safety is under control, discovery of a deficiency by routine surveillance, a press release
- Short description of the event

All the information above mentioned is retrievable in the database and the results are displayed in Japanese, including event title, country, facility name and type, event date, rating scale, rating criterion, and short description.

2. Availability/Effectiveness of Database

It is not appropriate to use the INES to compare safety performance between facilities, organizations or countries and the statistically small number of events at Level 2 and above, which also varies from year to year, makes it difficult to put forth meaningful international comparisons, according to the INES manual. However, the INES is an international reporting system of events with safety significance or public concerns and places emphasis on the promptness of reporting. Thus, it is useful for nuclear community to promptly grasp what type of event occurred. The events reported can be categorized into those with safety concerns, those involving offsite radioactive release or those involving the overexposures and as well, overall trends in events reported can be examined as shown in Section II.1. The database is an effective tool for doing such analyses. Examples of such analyses are discussed below.

<u>Example 1: Categorization of Events by Level:</u> The INES reports stored in the database

can be easily classified by specifying a scale as a search condition. The search results can indicate the number of INES reports with Level 2, 3, 4, 5, 6 or 7, separately, leading to easy identification of events with safety concerns.

<u>Example 2: Categorization of Events by Rating Criteria:</u> Using the safety attributes in the database, the reports can be categorized into 3 groups of events rated by degradation of defense in depth, onsite impact and offsite impact. Such categorization is useful to identify the events involving actual or potential environmental impact, those with plant contaminated and those involving the safety system failures.

<u>Example 3: Categorization of Events by Aspects of Significance:</u> By searching the database with use of the aspects of significance, the events can be grouped into those involving offsite radioactive releases, onsite radioactive releases, worker injured by radiation, and worker injured physically. Such categorization is useful for identifying the events with actual or potential radioactive consequences.

<u>Example 4: Identification of Events by Keywords:</u> Search by a keyword discriminates the specific events such as fire events, fatal accidents and events related to safety culture problems. Although some events identified are not relevant, generally good results are obtained and thus, this search is helpful in identifying the specific events.

3. Summary

The INES is a means of promptly communicating nuclear or radiological events to the media and the public. Aiming at more efficient utilization of the INES information, which is originally written in English, inside Japan, the author has translated the individual INES reports into Japanese, and developed the web-based database. The database, which contains all of the INES reports translated into Japanese, is an effective tool for the media and public to grasp what type of event occurred in the world and as well, for the nuclear community to examine the overall trending of events by categorizing the reported events. In addition, the database may provide the information useful for identifying the event group to be analyzed in detail.

II.3 Concluding Remarks

The generic study analyzed approximately 500 event reports submitted to the INES during the period 1990 to 2000 focusing on overall trends and characteristics of events rated at Level 2 or higher. The results show that while the number of events reported to the INES indicates the decreasing trend, the number of events rated at Level 2 or higher has been 10-20 events per year throughout the period. Also, approximately 60 events were uprated due to additional factors such as common cause failure(s) or lack of safety culture, which included violations of operating/maintenance procedures or LCOs. The characteristics of events observed are summarized as follows: two events rated at Level 4 involved fatalities due to overexposure to radiation and about half of 15 events at Level 3 were associated with overexposure resulting from lost sources or radiation devices in non-nuclear facilities; 5 events rated on the "offsite consequence" criterion resulted in the deaths and/or physical damages of worker(s) or public member(s), the events on the "onsite consequence" criterion were dominated by overexposure and/or contamination events in fuel processing, experimental or non-nuclear facilities, and most of the events at NPPs were rated on the "defense-in-depth" criterion; the events at NPPs have different characteristics depending on the reactor types but the degraded functions of ECCS and electric power systems were common issues to all types of reactors.

The comprehensive reviews covered more than 2000 IRS reports during the period 1988 through 2012 and identified approximately 200 safety significant events. These events can be roughly categorized into three groups of events: recurring events, events associated with external hazards and the events related to new phenomena or unexpected aggravating conditions. Recurring events indicate that the previous corrective actions have not necessarily been effective to prevent the recurrences and thus, the event analysis should place more emphasis on why the actions taken after the first event have failed to prevent the recurrences. As for the external hazards, more attention should be paid for ones beyond the design basis assumptions based on the previous events. The new phenomena or unexpected aggravating conditions have a potential to threaten the plant safety but it seems difficult to detect them prior to their appearances.

The generic studies including comprehensive reviews have provided the events with safety implications, which (might) have affected the facility, environment and public health and thus, should be further examined and discussed. The results from these studies have been disseminated into the nuclear community so that the event information could have been shared and utilized more efficiently and effectively to improve the facility safety. Through these studies, the fundamental framework for the operating experience feedback process was established. As well, these studies have provided inputs to the topical studies. The generic studies demonstrate described here that the systematic and comprehensive reviews are essential to identify and characterize the safety significant events and to share the lessons learned from the individual events.

For the purpose of efficient utilization of event information, a new computer software package, CESAS, was developed. CESAS is to extract systematically the event sequence from the event description written in natural language of English. This system is based on knowledge engineering technique utilized in the field of natural language treatment. The analytical process in this system consists of three steps: (i) to analyze each sentence syntactically and semantically, (ii) to identify sentences, clauses or phrases representing occurrences, (iii) to deduce the mutual relationship between occurrences. Through the comparative studies, it was shown that CESAS-extracted event sequences generally agreed with manually-extracted ones, demonstrating the effectiveness and feasibility of CESAS, and the technological perspectives were obtained on establishing computer aid event analysis.

The web-based database, which contains the INES reports translated into Japanese, was developed to provide an effective tool for the media and public as well as for the nuclear community. The database may also provide the information useful for identifying the event group to be analyzed in detail.

References in Chapter II

(1) N. Watanabe and M. Hirano, *World Wide Web for Database of Japanese Translation on International Nuclear Event Scale Reports*, Journal of Atomic Energy Society of Japan, Vol.41, No.6, pp.628-638, 1999 (in Japanese).

(2) N. Watanabe, T. Oikawa and M. Hirano, *Compilation of INES (International Nuclear Event Scale) Information: Japanese Translation*, JAERI-Data/Code 98-023, 1998 (in Japanese).

(3) N. Watanabe, *Compilation of INES (International Nuclear Event Scale) Information: Japanese Translation (Vol.2)*, JAERI-Data/Code 2001-002, 2001 (in Japanese).

(4) N. Watanabe, *Trends and Characteristics Observed in Nuclear Events Based on International Nuclear Event Scale Reports*, Journal of Atomic Energy Society of Japan, Vol.43, No.10, pp.1017-1026, 2001 (in Japanese).

(5) IAEA, *IRS Guidelines – Joint IAEA/NEA International Reporting System for Operating Experience*, 2010.

(6) IAEA & OECD/NEA, *THE IAEA/NEA INCIDENT REPORTING SYSTEM (IRS) Using Operating Experience to Improve Safety*.

(7) N. Watanabe, T. Oikawa, M. Hirano, F. Araya, K. Fujiki and K. Namatame, *Analysis and Compilation of Events Reported to OECD/NEA IRS*, 1992 (in Japanese) (Restricted).

(8) T. Oikawa, N. Watanabe, T. Tamakoshi and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (2)*, 1994 (in Japanese) (Restricted).

(9) T. Oikawa, N. Watanabe, T. Tamakoshi and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (3)*, 1996 (in Japanese) (Restricted).

(10) N. Watanabe, T. Oikawa and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (4)*, 1996 (in Japanese) (Restricted).

(11) N. Watanabe, T. Oikawa and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (5)*, 1997 (in Japanese) (Restricted).

(12) N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (1997 Edition)*, 1998 (in Japanese) (Restricted).

(13) N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (1998 Edition)*, 1999 (in Japanese) (Restricted).

(14) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (1999 Edition)*, 2000 (in Japanese) (Restricted).

(15) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2000 Edition)*, 2001 (in Japanese) (Restricted).

(16) N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2001 Edition)*, 2002 (in Japanese) (Restricted).

(17) N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2002 Edition)*, 2003 (in Japanese) (Restricted).

(18) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2003 Edition)*, 2004 (in Japanese) (Restricted).

(19) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2004 Edition)*, 2005 (in Japanese) (Restricted).

(20) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2005 Edition)*, 2006 (in Japanese) (Restricted).

(21) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2006 Edition)*, 2007 (in Japanese) (Restricted).

(22) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2007 Edition)*, 2008 (in Japanese) (Restricted).

(23) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2008 Edition)*, 2009 (in Japanese) (Restricted).

(24) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2009 Edition)*, 2010 (in Japanese) (Restricted).

(25) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2010 Edition)*, 2011 (in Japanese) (Restricted).

(26) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2011 Edition)*, 2012 (in Japanese) (Restricted).

(27) N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2012 Edition)*, 2013 (in Japanese) (Restricted).

(28) OECD/NEA & IAEA, *Nuclear Power Plant Operating Experiences from the IAEA/NEA Incident Reporting System 1996-1999*, 2000.

(29) IAEA & OECD/NEA, *Nuclear Power Plant Operating Experiences from the IAEA/NEA Incident Reporting System 1999-2002*, 2003.

(30) OECD/NEA & IAEA, *Nuclear Power Plant Operating Experiences from the IAEA/NEA Incident Reporting System 2002-2005*, 2006.

(31) IAEA & OECD/NEA, *Nuclear Power Plant Operating Experiences from the IAEA/NEA International Reporting System for Operating Experience 2005-2008*, 2010.

(32) OECD/NEA & IAEA, *Nuclear Power Plant Operating Experiences from the IAEA/NEA International Reporting System for Operating Experience 2009-2011*, 2012.

(33) N. Watanabe, K. Kobayashi, Y. Arise and M. Okuda, *Development of Computerized*

Event Sequence Abstracting System – System Design and Analytical Approach, Journal of the Atomic Energy Society of Japan, Vol.30, No.8, pp. 687-698, 1988 (in Japanese).

(34) N. Watanabe, K. Kobayashi and K. Fujiki, *CESAS: Computerized Event Sequence Abstracting System – Outlines and Applications*, Proc. of the OECD/CSNI-CEC Specialist Meeting on Trend and Pattern Analysis of Operational Data from Nuclear Power Plants, 1989 (Invited Paper).

(35) N. Watanabe, K. Kobayashi, T. Ishigami, M. Hikawa, Y. Arise and M. Okuda, *Development of the Computerized Event Sequence Abstracting System*, Proc. of International ANS/ENS Topical Meeting on Thermal Reactor Safety, 1986.

(36) N. Watanabe, K. Kobayashi, Y. Arise and M. Okuda, *Outlines of Computerized Event Sequence Abstracting System*, The Institute of Electronics, Information and Communication Engineering (IEICE) Technical Reports, Vol.87, No.209, NLC 87-6, 1987 (in Japanese).

(37) N. Watanabe, *Computerized Event Sequence Abstracting System – Applications and Studies on Description Styles*, Extended Synopses of Seminar on the Use of Unusual Event Reports for Improving Nuclear Power Plant Safety, 1990.

(38) T. Nishiyama and Y. Shinohara, *Development of Expert System for Abnormal-Event Recurrence Prevention*, Journal of the Atomic Energy Society of Japan, Vol.29, No.4, 1987 (in Japanese).

(39) J. Nakamura, et. al., *Improvement and Problems of Utilisp in Machine Translation Software GRADE*, Kigo-Shori 33-2, Information Processing Society of Japan (IPSJ), 1985 (in Japanese).

(40) H. Tanaka, et. al., *Semantics based Machine Translation System*, IPSJ Magazine, Vol.26, No.10, 1985 (in Japanese).

(41) M. Nagao, *Kikai-Honnyaku wa dokomade kanoka*, Iwanami-Shoten, 1986 (in Japanese).

(42) K. Tanaka, *Chishiki-Kogaku*, Maruzen, 1984 (in Japanese).

(43) M. Nagao, *Evaluation of Quality of Machine-Translated Sentences and the Control of Language*, IPSJ Magazine, Vol. 26, No.10, 1985 (in Japanese).

(44) M. P. Marcus, *A Theory of Syntactic Recognition for Natural Language*, MIT Press, 1980.

(45) T. Gunji, *Phrase Structure Grammar*, IPSJ Magazine, Vol.27, No.8, 1985 (in Japanese).

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Chapter III

Topical Studies on Safety Significant Events

III.1 Background

Several topical studies were performed on safety significant events to examine the characteristics of events and to obtain the generic insights common to those events and/or the generic lessons which could be useful for improving safety regulations and enhancing the facility safety. In these studies, topics to be analyzed were selected primarily based on the results from the generic studies/comprehensive reviews described in the previous chapter. Selection of topics focused on the events with generic safety implications, such as similar events in several countries, and the events with the public/media interest. However, the IRS reports are restricted and thus, these studies have collected the event information on individual topics mainly from the United States Nuclear Regulatory Commission's (USNRC) publications so that the results from the studies could be shared throughout the nuclear community more widely and utilized more effectively and efficiently.

So far, the following topical studies have been carried out:

- **Loss of decay heat removal (DHR) during reactor shutdown at pressurized water reactors (PWRs)**[1-3]: Many events had taken place in 1980s, particularly at PWRs in the United States. Since 1990, similar events had been reported from several countries. This topical study reviewed a total of 63 events during reactor shutdown with reactor cooling system (RCS) inventory reduced to identify the causes which could prolong the duration of DHR loss and evaluated the times to bulk boiling in the core and to core uncovery after losing DHR. It was revealed that the prolonged loss of DHR was caused by mainly air entrainment into the pumps and the major contributor of air entrainment was the lowering the RCS water level too far, most of which resulted from inaccurate level indication. The bulk boiling and core uncovery were estimated to take place within 1 h and several hours, respectively.

- **Emergency core cooling system (ECCS) strainer clogging at boiling water reactors (BWRs)**[4]: In 1992, the Swedish BWR experienced the ECCS strainer clogging due to the insulation material dislodged by steam from a safety/relief valve that spuriously opened. After this event, similar or potential events occurred at BWRs, especially in the United States and therefore, the ECCS strainer clogging was recognized a safety significant issue worldwide. The review of such events indicated that the strainer clogging would be caused by debris generated by LOCA such as insulation debris, those generated during normal operation such as corrosion products, those stemming from fibrous filters and labeling materials installed in the containment, those stemming from maintenance-type materials such as tools and plastic bags, etc.
- **Criticality accidents in fuel processing facilities**[5,6]: The JCO accident occurred in 1999 and just before this accident, the previous criticality accidents were made public from the Russian Federation. This topical study examined the overall trends observed in 21 criticality accidents, which had taken place in foreign countries, and analyzed the sequences and causes of the accidents in terms of similarities to the JCO accident. The most of them took place when handling uranium or plutonium solutions in the vessels with unfavorable geometry. The common issues identified were the problems related to the operating procedures.
- **Primary water stress corrosion cracking (PWSCC) at PWRs**[7-9]: The operating experience with Alloy 600 degradation in the United States was analyzed by reviewing a total of 45 licensee event reports (LERs) from 1999 to 2005 and the trends of these events were examined focusing on affected components, characteristics of cracking and inspection approaches for detecting PWSCC. It was found that PWSCC was observed on the reactor coolant pressure boundary components exposed to the environment with high temperature such as the reactor vessel head penetration nozzles and pressurizer heater sleeves, and has a tendency to happen for specific manufactures and materials. As well, it was shown that different repair techniques were applied depending on the components affected.
- **Safety or safety/relief valve setpoint drift**[10, 11]: This topical study analyzed the operating experience with setpoint drift in safety/relief valves (SRVs) at BWRs, pressurizer safety valves (PSVs), and main steam safety valves (MSSVs) at PWRs in the United States by reviewing approximately 90 LERs during the years from 2000 to 2006 and examined the trend focusing on causes and drift ranges. It was shown that for SRVs and MSSVs, valve disc-seat bonding was a dominant cause of the setpoint drift high and has a tendency to result in a relatively large deviation of the setpoint. For PSVs, the deviation of setpoints was generally small though its causes were not specified in many instances.

- **Fire events**(12): This topical study provided the results from analysis of the experience with fire at light water reactors (LWRs) in the United States and analyzed several major fire events in foreign countries focusing on turbine fires and cable fires. This study showed that many fires took place in the turbine building and some of them involved turbine missile, hydrogen explosion and/or massive internal flooding, leading to the complicated events. The cable fires have a tendency of spreading to multiple areas. In some fire events, the main control room suffered from ingress of smoke and/or toxic gases. These fire events highlighted the need to take adequate measures such as physical separation and use of flame-retardant materials for preventing the fire spread and mitigating their consequences.

In the following, the topical studies on loss of decay heat removal, criticality accidents, and safety or safety/relief valve setpoint drift are described in detail.

III.2 Analysis of Loss of Decay Heat Removal during Reactor Shutdown at PWRs

The safety function of the DHR system, which is also referred to as residual heat removal (RHR) system or shutdown cooling (SDC) system, is to remove fission product decay heat from the reactor core so that the reactor can be maintained in a safe shutdown condition. An extended loss of DHR could lead to core uncovery and resultant fuel damage.

No serious damage has resulted from the total loss of DHR system. Nevertheless, a large number of events involving loss of DHR during the reactor shutdown condition which occurred so far have increased safety concerns for past years. In the United States, numerous studies have been performed on this subject. For example,

- In 1983, the Nuclear Safety Analysis Center (NSAC) reviewed loss of DHR events during the years 1976 through 1981 and made recommendations to improve the DHR system reliability and reactor safety during shutdown(13) .
- In 1985, the Office of USNRC, Analysis and Evaluation of Operational Data (AEOD), analyzed loss of DHR events which occurred during the years 1976 through 1984 and recommended remedial actions(14).

Besides these studies, the USNRC and industry have issued numerous publications on loss of DHR events including recommendations for improving the situation(15-19).

Though the licensees have been implementing recommendations made by the USNRC and industry, the continued occurrence of loss of DHR events has been experienced at U.S. PWRs and some events with RCS partially drained resulted in boiling in the core. At PWRs, RCS must be drained down to the mid-height of the hot leg piping during shutdown to allow maintenance and inspection activities on steam generators (SGs) and reactor coolant pump (RCP) seal and the DHR systems need to be operated for core cooling under such a reduced inventory condition, which is referred to as “mid-loop operation".

Loss of DHR in such a reduced inventory condition would lead to core boiling within a short time. Should core boiling occur and any recovery actions be not taken, coolant boiloff and the subsequent RCS pressurization would lead to coolant inventory loss via unisolated paths and/or would not allow water flow from the refueling water storage tank (RWST) into RCS by gravity feed, and consequently the core would be uncovered. However, loss of DHR events with the RCS inventory reduced, especially in the early stage of shutdown, afford little time to correct the problem and restore DHR flow before boiling conditions are reached. In addition, during shutdown operations, the operators are not necessarily provided with well thought out procedures for recovery from loss of DHR and may not be fully aware of equipment and/or time available for recovery.

On the other hand, several studies on probabilistic risk/safety assessment showed that the core damage frequency for shutdown condition was estimated at the range of 5×10^{-6} - 5×10^{-5} per reactor year, which was comparable to that for power operation[(20-22)]. These studies also noted that a loss of DHR event during reduced inventory conditions was one of major contributors to the shutdown risks.

In order to analyze the safety implications associated with total loss of DHR systems and provide information useful for improving the plant safety during shutdown operations, especially for preventing recurrence of such events, this topical study reviewed U.S. PWR operating experience involving total loss of DHR systems which occurred during the years 1976 through 1990 and analyzed direct and root causes leading to loss of DHR events. A total of 197 1oss of DHR events were reported to have occurred at U.S. PWRs during this period, which was equivalent to a rate of several per year. Out of these events, 63 loss of DHR events occurred when the RCS inventory was reduced and five events experienced boiling in the core. Highlighting loss of DHR events during reduced inventory conditions, their trends and characteristics were identified for these 63 events to clarify the major causes which could prolong the duration of the DHR loss. In addition, the RCS water heatup rate during the DHR loss and times to bulk boiling in the core and core uncovery after losing DHR were evaluated based on the data obtained from the actual events to examine time available for recovery of loss of DHR events.

1. Outlines of Loss of Decay Heat Removal Events

As shown in **Figure III.1**, a typical DHR system at PWRs is composed of two redundant trains(14). Generally, both trains take suction from the same RCS hot leg and the connecting piping is a “drop" line attached to either the bottom or a lower quadrant of the RCS hot leg. The single suction design makes the DHR system susceptible to total loss of the DHR function due to a single failure of a suction line valve. During reduced inventory conditions as illustrated in **Figure III.2**, lowering the RCS water level too far or increasing the DHR flow can cause vortexing in the hot leg at the nozzle of the suction drop line, air entrainment in the water flowing to the operating DHR pump, and subsequent air binding of the pump. If the standby DHR pump is started in such cases, it will also become air bound, resulting in loss of DHR system. Once air is entrained in the DHR system, it takes a long time for the trapped air to migrate back to RCS or to be vented from the DHR pumps because of long horizontal piping from RCS to the DHR pumps. If any recovery actions of DHR flow would be prolonged, bulk boiling would occur in the core and the subsequent RCS pressurization would lead to coolant inventory loss and/or would not allow water addition by gravity feed, resulting in core uncovery.

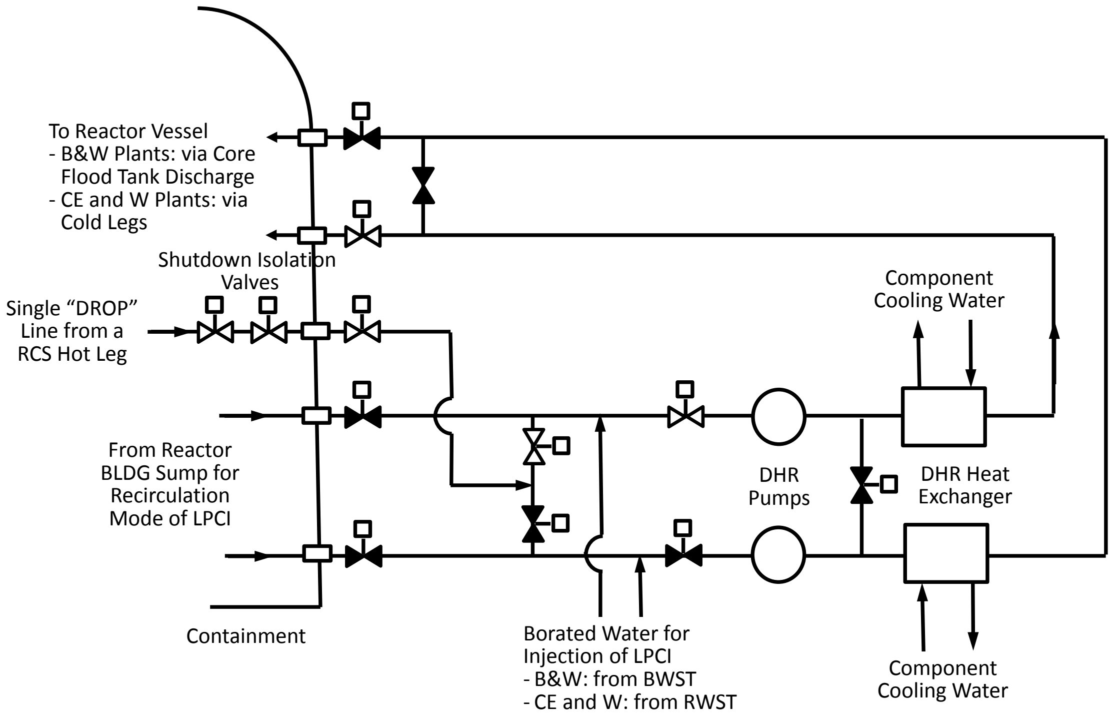

Figure III.1 Schematic Diagram of DHR Systems at U. S. PWRs(14)

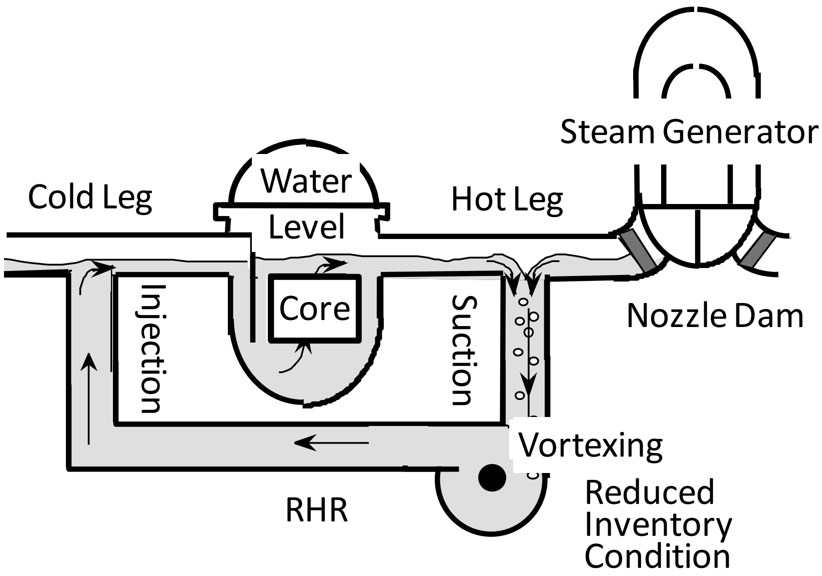

Figure III.2 Reduced Inventory Condition

In this study, a loss of DHR event was defined as an event in which both trains of the DHR system were unable to perform their functions while operating or when required. Loss of DHR events reviewed in this study came from operating experience at U.S. PWRs from 1976 through 1990 (up to the first quarter of 1990). The data sources referred were the following three groups:

- Loss of DHR events from 1976 through 1981 were obtained from NSAC report, NSAC-52(13), which contained 96 events.
- Loss of DHR events from 1982 through 1984 were obtained from USNRC/AEOD report, AEOD-C503(14),which contained approximately 140 events including about 90 events identified in NSAC-52.
- Operating experience for the last five years from 1985 to the first quarter of 1990 was obtained from LERs and USNRC reports.

As a result, 197 1oss of DHR events were identified in this study. This figure is equivalent to a frequency of 0.23 per reactor year, based on about 850 reactor years of commercial U.S. PWR operation. About one-third (63 events) of 197 loss of DHR events occurred during reduced inventory conditions. For these 63 events, **Table III.1** summarizes duration of a DHR loss, coolant heatup during an event, and event category which will be discussed in the following subsection.

Table III.1 Summary of 63 Loss of DHR Events during Reduced Inventory Conditions

No	Plant	Date	Duration (min)	Heatup (°C, °F)	Event Category and Causes	Reference
1	Trojan	05/21/1977	55	Unknown	A1: Inaccurate level indication (Calibration error)	NSAC-52
2	Palisades	01/06/1978	45	47.2(85)	B2: Isolation of RHR HX outlet valve (Controller failure)	NSAC-52
3	Trojan	03/25/1978	10	Unknown	A1: Inaccurate level indication (RCS higher pressure)	NSAC-52
4	Trojan	03/25/1978	10	Unknown	A1: Inaccurate level indication (RCS higher pressure)	NSAC-52
5	Trojan	04/17/1978	Unknown	Unknown	A1: Inaccurate level indication (Incompletely filled ref. leg)	NSAC-52
6	Trojan	04/25/1978	1	Unknown	B3: Reducing RHR flow (Loss of coolant, drain valve opening)	NSAC-52
7	Beaver valley-1	09/04/1978	60	16.7(30)	A4: not known	NSAC-52
8	St. Lucie-1	11/03/1978	222	Unknown	A2: Misalignment of RHR discharge valves (Personnel error)	NSAC-52
9	Millstone-2	03/14/1979	Unknown	32.2(58)	A1: Inaccurate level indication (not known)	NSAC-52
10	Salem-1	06/30/1979	34	Unknown	A1: Inadequate level monitoring (Procedural/personnel error)	NSAC-52
11	Beaver valley-1	01/17/1980	Unknown	Unknown	A4: Inadvertent actuation vessel vent eductor system	NSAC-52
12	Beaver valley-1	04/08/1980	35	Unknown	A3: Increasing RHR pump flow (Procedural error)	NSAC-52
13	Beaver valley-1	04/11/1980	70	3.9(7)	A3: Increasing RHR HX flow (Procedural error)	NSAC-52
14	Davis Besse-1	04/18/1980	29	5.6(10)	A2: Partially opening of RHR discharge valve (Personnel error)	NSAC-52
15	Davis Besse-1	04/19/1980	150	44.4(80)	A4: Inadvertent pump suction transfer (Electrical fault)	NSAC-52
16	Beaver valley-1	03/05/1981	54	38.7(66)	A1: Inaccurate level indication (Evaporation in ref. leg)	NSAC-52
17	Trojan	06/26/1981	72	5.6(10)	A1: Inaccurate level indication (Inadequate check of system)	NSAC-52
18	Palisades	07/18/1981	90	41.4(74)	B2: Isolation of RHR HX outlet valve (Positioner failure)	NSAC-52
19	Millstone-2	12/09/1981	30 – 60	65.6(118)	B1: Spurious trip signal on the running pump (personnel error)	NSAC-52
20	McGuire-1	03/02/1982	50	13.9(25)	A1: Inaccurate level indication (RCS higher pressure)	NUREG-1410
21	North Anna-2	05/20/1982	8	Unknown	A1: Inaccurate level indication (Installation problem)	LER-82026
22	North Anna-2	05/20/1982	26	Unknown	A1: Inaccurate level indication (Installation problem)	LER-82026
23	North Anna-2	05/20/1982	60	Unknown	A1: Inaccurate level indication (Installation problem)	LER-82026
24	North Anna-2	07/17/1982	Unknown	Unknown	A3: Actuation of standby pump (Procedural error)	NUREG-1410
25	North Anna-2	07/30/1982	46	Unknown	A2: RHR pump seal leakage	NUREG-1410
26	North Anna-2	08/02/1982	Unknown	Unknown	B3: Reducing RHR flow (Loss of coolant, pump seal leakage)	NUREG-1410
27	North Anna-1	10/19/1982	36	Unknown	A1: Inaccurate level indication (Installation problem)	LER-82067
28	North Anna-1	10/20/1982	33	Unknown	A1: Inaccurate level indication (Installation problem)	LER-82067
29	McGuire-1	04/05/1983	Unknown	15.6(28)	A1: Inaccurate level indication (Inadequate check of system)	NUREG-1410
30	R.E.Ginna	04/12/1983	12	8.3(15)	A4: Pressurization of SG plenum and nozzle dam leakage	LER-83015
31	North Anna-2	05/03/1983	Unknown	Unknown	A1: Inadequate level monitoring (Personnel error)	LER-83038
32	Surry-1	05/17/1983	81	Unknown	A1: Inaccurate level indication (Inadequate check of system)	NUREG-1410
33	Sequoyah-2	08/06/1983	77	51.1(92)	A1: Inaccurate level indication (Inadequate check of system)	LER-83101
34	Calvert Cliffs-1	10/23/1983	40	36.1(65)	B2: Auto-closure of suction valve (Personnel error)	NUREG-1410
35	McGuire-2	12/31/1983	43	Unknown	A1: Inaccurate level indication (Installation problem)	LER-84001
36	McGuire-2	01/09/1984	62	Unknown	A1: Inaccurate level indication (Installation problem)	LER-84001
37	R.E.Ginna	03/07/1984	Unknown	8.3(15)	A2: Inadequate valve lineup (Personnel error)	NUREG-1410
38	Trojan	05/04/1984	40	53.3(96)	A1: Inaccurate level indication (Blockage in indicating system)	LER-84010
39	D.C.Cook-2	05/21/1984	25	Unknown	A3: Actuation of standby pump (Procedural error)	LER-84014
40	ANO-2	08/29/1984	60	66.1(119)	A1: Inaccurate level indication (RCS higher pressure)	LER-84023
41	Zion-1	09/14/1984	45	20.6(37)	A1: Inaccurate level indication (RCS higher pressure)	LER-84031
42	North Anna-2	10/16/1984	120	19.4(35)	A1: Inaccurate level indication (Clogging in indicating system)	LER-84008
43	Catawba-1	04/22/1985	81	20.6(37)	A1: Inaccurate level indication (Inadequate check of system)	NUREG-1410
44	Sequoyah-1	10/09/1985	43	0.06(0.1)	A3: Actuation of standby pump (Procedural error)	LER-85040
45	Zion-2	12/14/1985	75	8.3(15)	A1: Inaccurate level indication (Inadequate check of system)	NUREG-1410
46	Crystal River-3	02/02/1986	24	18.3(33)	B1: Sheared pump shaft	LER-86003
47	San Onofre-2	03/26/1986	70	53.3(96)	A1: Inaccurate level indication (Installation problem)	LER-86007
48	Waterford-3	07/14/1986	220	52.2(94)	A2: Inadvertent opening of mini-flow valve (Procedural error)	LER-86015
49	Sequoyah-1	01/28/1987	90	11.1(20)	A1: Inaccurate level indication (Blockage in indicating system)	LER-87012
50	Ft. Calhoun-1	03/21/1987	5	Unknown	B1: Loss of power to the running pump (Personnel error)	LER-87008
51	Diablo Canyon-2	04/10/1987	88	73.9(133)	A2: Valve leakage	NUREG-1269
52	McGuire-1	09/16/1987	6	26.1(47)	B1: Loss of power to the running pump (Personnel error)	NUREG-1410
53	Palisades	10/15/1987	29	16.6(30)	B2: Cyclic actuation of flow control valve (Personnel error)	LER-87035
54	Waterford-3	05/12/1988	Unknown	Unknown	A1: Inaccurate level indication (Loop seal in indicating system)	NUREG-1410
55	Sequoyah-1	05/23/1988	Unknown	Unknown	A2: Inadvertent opening of isolation valves (Personnel error)	LER-88021
56	D.C.Cook-2	06/16/1988	Unknown	Unknown	A1: Inadequate level monitoring (Procedural/Personnel error)	NUREG-1410
57	San Onofre-3	09/11/1988	Unknown	Unknown	A1: Inaccurate level indication (not known)	NUREG-1410
58	Oconee-3	09/18/1988	15	8.3(15)	B1: Loss of power to the running pump (Personnel error)	LER-88005
59	Byron-1	09/19/1988	14	Unknown	A1: Inadequate level monitoring (Procedural/personnel error)	LER-88007
60	ANO-1	10/26/1988	23	10.6(19)	B2: Inadvertent closure of flow control valve (Personnel error)	LER-88014
61	Salem-1	05/20/1989	53	16.7(30)	A1: Inadequate level monitoring (Procedural/ Personnel error)	LER-88019
62	Comanche Peak-1	07/18/1989	Unknown	Unknown	B2: Inadvertent opening of flow control valve (Electrical fault)	NUREG-1410
63	Vogtle-1	03/20/1990	36	25.6(48)	B1: Loss of power to the running pump (Personnel error)	NUREG-1410

Note) A1: lowering RCS level too far, A2: loss of coolant inventory, A3: vortexing in RHR pump suction, A4: others,
B1: inadvertent trip of running RHR pump, B2: malfunction of RHR valves, B3: manual pump trip due to decreased flow

2. Analysis of Loss of Decay Heat Removal Events

2.1 Trend Analysis

The loss of DHR events were analyzed to examine if there were any significant trends. One measure of significance of loss of DHR events is the duration that the DHR function was lost. The duration of DHR loss was reported in 149 of 197 events. **Figure III.3** presents a summary of the time durations of DHR losses for these 149 events, discriminating the events during reduced inventory conditions from ones with RCS filled. There were 50 events during reduced inventory conditions, 18 events of which lasted for more than 1 h and 17 events lasted for from 30 min to 1 h. Most of the events with the durations of more than 1 h (17 of 18 events) were caused by air entrainment in the DHR system. As for 99 loss of DHR events which occurred with RCS filled, there were 79 events in which the DHR function was recovered within 30 min. These facts indicate that the DHR losses, especially resulting from air entrainment, during reduced inventory conditions tend to prolong the duration of the DHR loss while it takes a relatively short time to recover the DHR function in the events with RCS filled.

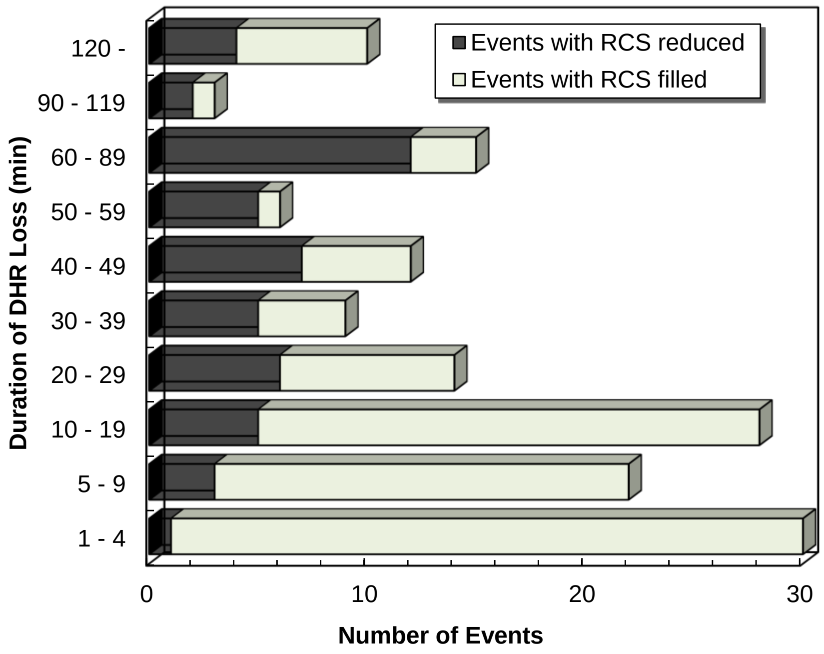

Figure III.3 Duration Times of DHR Losses

Another measurement of significance is the temperature rise in RCS during the DHR losses. Of 50 events in which the temperature rise was reported, there were 34 events during reduced inventory conditions and 16 events with RCS filled. **Figure III.4** provides the number of events in several ranges of the temperature rise by shutdown condition (that is, reduced inventory condition or filled RCS condition). As shown in this figure, all of ten events in which the RCS temperature increased more than 40°C (72°F) occurred during reduced inventory conditions. At least five events listed in **Table III.2** were reported to have experienced bulk or local boiling in the core. Although it was not specified whether or not boiling in the core would have occurred during the events, the RCS temperature was reported to have increased to near or more 93.3°C (200°F) in five events during reduced inventory conditions (this temperature, 93.3°C, is the upper limit for cold shutdown conditions defined in U.S. PWR vendors' standard technical specifications(14)). Since the RCS temperatures were often observed on hot legs or RHR pump exit, as seen in Table III.2, these five events could have experienced core boiling even though the observed temperature was below 93.3°C.

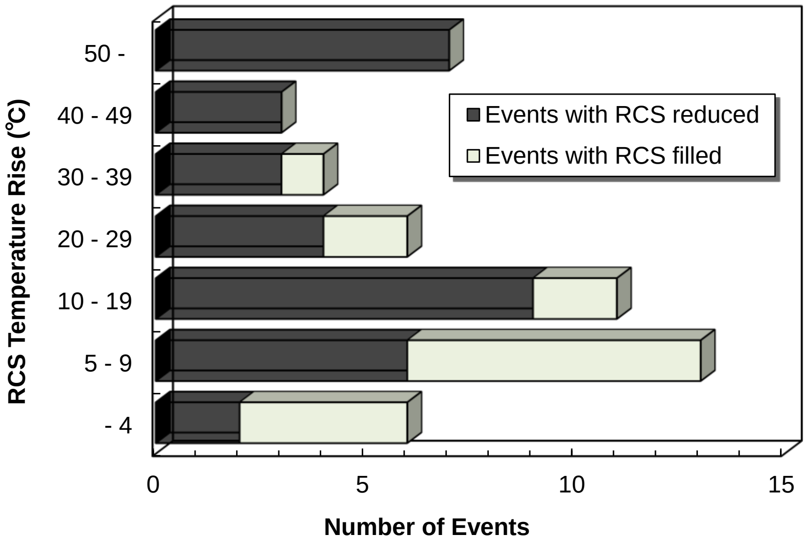

Figure III.4 RCS Temperature Rises during DHR Losses

Table III.2 Core Boiling Events

Plant Name	Event Date	RCS Heatup	Observation Location	Duration Time (min)
Millstone-2	12/09/1981	32.2°C(90°F) → 97.8°C(208°F)	RHR pump exit	30 - 60
ANO-2	08/29/1984	60.0°C(140°F) → 96.1°C(205°F) 60.0°C(140°F) → 126.1°C(259°F)	(RCS bulk average) core exit	30 - 40*1
San Onofre-2	03/26/1986	45.6°C(114°F) → 96.9°C(210°F)	hot leg	70 (49) *2
Waterford-3	07/14/1986	58.9°C(138°F) → 111.1°C(232°F) 58.9°C(138°F) → 112.2°C(234°F)	hot leg core exit	220
Diablo Canyon-2	04/10/1987	30.6°C(87°F) → 104.4°C(220°F)	RHR pump exit	88

*1: The loss of DHR is reported to have lasted for 50-60 min, but during the first 20 min, water has been added to the RCS by gravity feed from RWST and by oscillatory RHR system flow [Ref. 14].

*2: According to LER-86007, the duration of 70 min is represented as a period from the initiation of RHR pump motor current oscillation to the restoration of the DHR flow. However, during the first 21 min, the RHR pump flow has not been lost.

2.2 Categorization of DHR Losses during Reduced Inventory Conditions

As mentioned above, the DHR losses during reduced inventory conditions may have relatively high possibilities of lasting for a long time and/or resulting in boiling in the core. Focusing on events with RCS reduced, this study analyzed direct or root causes leading to the DHR losses and categorized events. As presented in **Figure III.5**, the 63 loss of DHR events which occurred in reduced inventory conditions were grouped into two general categories: one for events involving the cavitation or air binding of DHR pumps due to vortexing or air entrainment and the other for events involving the forced trip of DHR pumps with neither leading to cavitation nor air binding.

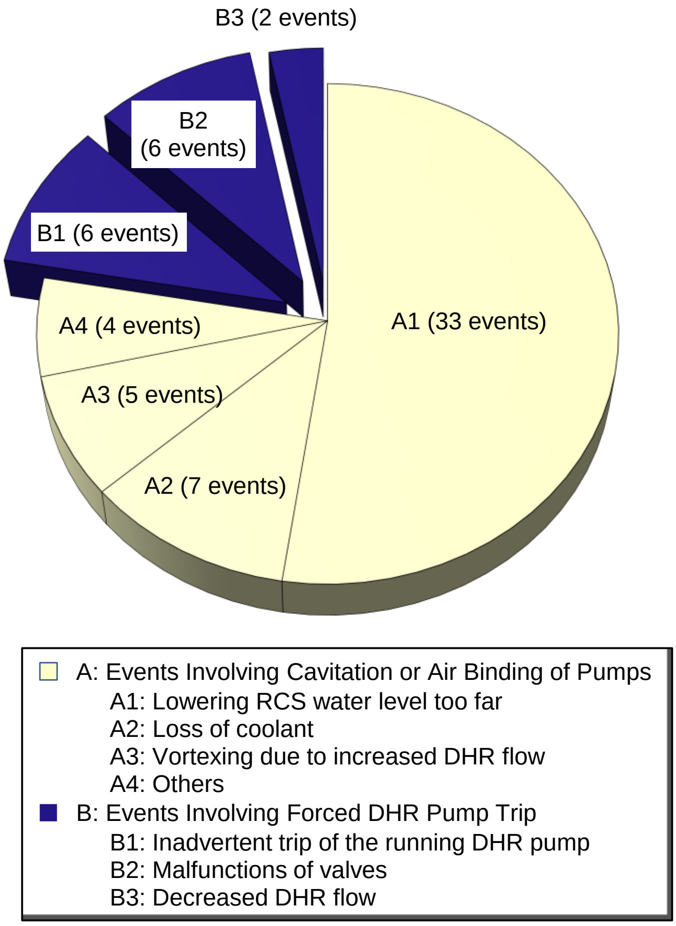

Figure III.5 Categorization of Events during Reduced Inventory Conditions

<u>Category A</u>: Events involving the cavitation or air binding

Forty-nine events included in the first category are further divided into four subcategories. The first subcategory (Al) contains 33 events which resulted from lowering the RCS water level too far. Five events in this subcategory were caused by inadequate RCS level monitoring during drain-down operations. However, most events (28 events) in this subcategory were caused by inaccurate RCS water level indications due to:

- incompletely filled reference legs, or loss of level in reference leg due to leakage or evaporation (two events, e.g. the Beaver Valley-1 event in 1981)
- problems on level indicating system installation (eight events, e.g. the San Onofre-2 event in 1986)
- higher than intended pressure on RCS (five events, e.g. the ANO-2 event in 1984)
- calibration errors or inadequate checks of level indicating system (seven events, e.g. the Sequoyah-2 event in 1983), and
- malfunctions of level indicating system, such as blockage and clogging (four events, e.g. the North Anna-2 event in 1984).

For the other two events (at Millstone-2 in 1979 and San Onofre-3 in 1988), causes of inaccurate level indications were not reported. There were 21 events in which the DHR losses lasted for more than 30 min, including 11 events with the duration of more than 1 h, two of which led to boiling in the core (the ANO-2 and the San Onofre-2 events).

The second subcategory (A2) contains seven events in which loss of coolant inventory caused insufficient pump suction head. In five of these events, loss of coolant inventory was caused by misalignment of valves due to an operator error or procedural deficiencies (e.g. the Waterford-3 event in 1986). The other two events (at Diablo Canyon-2 in 1987 and at North Anna-2 in 1982) in this subcategory resulted from leakage of a valve and a pump seal, respectively. The DHR losses lasted for more than 1 h in three events and boiling in the core occurred in two events (the Waterford-3 and Diablo Canyon-2 events).

The third subcategory (A3) contains five events which involved vortexing in pump suction due to the increased DHR flow. In three events (e.g. the Cook-2 event in 1984), the DHR flow increased when the standby pump was started, resulting in vortexing. The root causes for these events were procedural deficiencies. For the other two events (at Beaver Valley-1 in 1980), causes of the increased DHR flow were unknown. In four events, the durations of DHR losses was reported to be 25-70 min.

The fourth subcategory (A4) contains four events which do not fit in any of the above subcategories. One was caused by automatic transfer of pump suction to the empty emergency sump due to a spurious signal generated by loss of vital instrument buses (at Davis Besse-1 in 1980). In this event, the DHR loss lasted for 2.5 h. The second event (at Beaver Valley-1, 1980) was caused by inadvertent actuation of the vessel vent eductor system, which excessively drew the water from SGs and entrained air into loops. The third event (at Ginna in 1983) was caused by inadvertent SG plenum pressurization and leaky nozzle dam allowing air in SG to pass through the hot leg. The last event (at Beaver Valley-1 in 1978) involved indication of no DHR flow, the cause of which was unknown.

Category B: Events involving forced DHR pump trip
Fourteen events included in this category are further divided into three subcategories. The first subcategory (Bl) contains six events involving inadvertent trip of the running DHR pump due to electrical problems or mechanical failures. Four events were caused by loss of electrical power to the running pump due to personnel errors (e.g. the Vogtle-1 event in 1990), one event by a spurious trip signal due to procedural deficiencies (at Millstone-2 in 1982), and one event by a sheared pump shaft (at Crystal River-3 in 1986). There were two events lasting for more than 30 min, one of which resulted in core boiling (the Minstone-2 event).

The second subcategory (B2) contains six events which were caused by malfunctions of valves. In all cases, the DHR pumps were manually tripped prior to cavitation or air entrainment. Out of these events, four involved loss of DHR flow: one was caused by automatic closure of the suction valves due to a spurious signal during testing (at Calvert Cliffs-1 in 1983), one was caused by inadvertent closure of the flow control valve due to a personnel error (at ANO-1 in 1988), and two resulted from the isolation of the heat exchanger outlet air operated valve because of water accumulation in its control air system (at Palisades in 1978 and 1981). The other two events involved manual trip of DHR systems. In one event (at Comanche Peak-1 in 1989), the flow control valve failed open due to inverter failure and thus the DHR flow exceeded the limit in technical specifications. The other resulted from cyclic actuation of flow control valve due to a personnel error, which caused fluctuation of DHR flow (at Palisades in 1987). In three events in this subcategory, the DHR losses lasted for 20-40 min.

The third subcategory (B3) contains two events involving decreased DHR flow which was caused by loss of coolant inventory. The root causes were leakage from a pump seal for one event (at North Anna-2 in 1982) and effluence due to valve misalignment for the other (at Trojan in 1978). In these events, the running DHR pumps were secured prior to cavitation or air entrainment and hence the DHR losses were recovered within a few minutes.

It should be noted that the DHR losses lasted for 30 min in 30 of 49 events involving cavitation or air binding. As for 14 events not involving air entrainment, on the other hand, only five lasted for more than 30 min. These facts manifest the difficulties in recovery actions, such as removal of trapped air from the DHR system, which may prolong the duration of DHR loss and consequently produce relatively high possibilities of core uncovery.

3. Estimation of Reactor Coolant Heatup and Time to Core Boiling

3.1 Approach

Boiling in the core and the subsequent RCS pressurization would lead to coolant inventory loss via unisolated paths and/or prevent water addition to RCS by gravity feed from RWST, making it more difficult to reestablish the DHR flow. Therefore, the plant personnel should notice the time to boiling following the DHR loss and take some timely actions to restore the DHR flow. However, in boiling events, the plant personnel did not know the time when boiling took place in the core during the event or they were not aware even that boiling had occurred.

In the analysis of the Diablo Canyon-2 event on April 10, 1987[(23)], the time to boiling was estimated with the following heat balance equation on the assumption that the water involved in heatup process would be heated uniformly:

$$\Delta T = \frac{Q * \tau}{V_{ef} * \rho * C_p},$$

where ΔT: coolant temperature rise during event (°C)
Q: decay heat generation rate during event (W)
τ: time after DHR loss (min)
V_{eff}: water volume involved in heatup process (hereinafter, effective water volume) (m^3)
ρ: water density (kg/m^3)
C_p: specific heat at constant pressure of water (kcal/kg°C)

However, that approach might have some difficulties in determining the effective water volume which stemmed from complex and uncertain behavior in RCS during heatup process, for example, how the natural convection would be developed in RCS and how the water in each region of RCS would be heated, and insufficient information on the RCS coolant inventory. Because of these difficulties, such an analysis has not been carried out for most of the events and the time to boiling was simply extrapolated with use of the observed RCS coolant heatup rate for some of them such as the Vogtle-1 event on March 20, 1990(19).

For the calculations of coolant heatup rates and times to boiling with the above equation, the data on the initial RCS coolant temperature, the duration of the DHR loss and the time elapsed from the reactor shutdown are needed. Out of 63 events, identified were the 12 events, including four boiling events, for which these data were described. The data obtained from event reports are summarized in **Table III.3**. Applying the approach used in the analysis of the Diablo Canyon-2 event, the heatup rates of primary system and the times to boiling for these 12 events were evaluated to examine if boiling could have actually occurred and/or when boiling would have occurred.

Table III.3 Summary of Data Obtained from Actual Events

No.	Plant Name (vendor)	Event Date	Rated Power (MWt)	RCS Temperature Ti (before) °C(°F)	RCS Temperature Te (after) °C(°F)	Duration of DHR Loss: τ (min)	Time after Shutdown
1	Diablo Canyon-2 (WH)	04/10/1987	3411	30.8 (87)	104.4 (220)	88 (33.8)*1	7 d
2	ANO-2 (CE)	08/29/1984	2815	60.0 (140)	128.1 (259)	60 (30-40)*2	36 h
3	Millstone-2 (CE)	12/09/1981	2700	32.2 (90)	97.8 (208)	30-60	4 d
4	Sequoyah-2 (WH)	08/06/1983	3411	39.4 (103) 60.0 (140)	90.6 (195) 90.6 (195)	77 35	18 d
5	Davis Besse-1 (B&W)	04/19/1980	2772	32.2 (90)	76.7 (170)	150	12 d
6	Beaver Valley-1 (WH)	03/05/1981	2652	38.9 (102)	75.6 (168)	54	14 d
7	McGuire-1 (WH)	09/16/1987	3411	38.3 (101)	64.4 (148)	6	5 d
8	Vogtle-1 (WH)	03/20/1990	3411	32.2 (90)	57.8 (136)	36	25 d*4
9	Beaver Valley-1 (WH)	09/14/1978	2652	62.8 (145)	79.4 (175)	60	38 d
10	San Onofre-2 (CE)	03/26/1986	3390	45.6 (114)	98.8 (210)	70 (49)*3	11 d
11	Ocinee-3 (B&W)	09/11/1988	2568	32.2 (90)	40.6 (105)	15	32 d*4
12	Palisades (CE)	01/08/1978	2530	54.4 (130) 54.4 (130)	93.3 (200) 101.7 (215)	20 45	7 d

*1: The time to bulk boiling has been estimated by USNRC [Ref.23].
*2: During the first 20 min, water was added to the RCS by gravity feed from RWST and by oscillatory RHR flow [Ref.14].
*3: During the first 21 min, water was added to the RCS by gravity feed from RWST and by oscillatory RHR flow [LER-86007].
*4: The DHR loss occurred after the refueling completed.

In addition, to provide information useful in taking some timely actions to restore the DHR system, the time to boiling was evaluated as a function of the time elapsed from the reactor shutdown to the DHR loss for a typical Westinghouse 4-loop PWR.

In the calculations, the decay heat generation rates *Q* were estimated for the individual events based on representations of the ANSI/ANS (American Nuclear Society/American Nuclear Standards Institute) 5.1 standard of 1979[(24)]. The decay heat power presented here was the combined energy release from fission product decay of ^{235}U, ^{238}U and ^{239}Pu, and actinide decay of ^{239}U and ^{239}Np using the energy release per fission (that is, 200 MeV) and the distribution of fission product sources (that is, 0.8 for ^{235}U, 0.13 for ^{239}Pu, 0.06 for ^{238}U and 0.01 for ^{241}Pu) described in Ref. (24). The calculations used the actual data on the time elapsed from the reactor shutdown, and assumed that the reactors had been operated at rated power for about 10,000 h because of lack of information.

As for the effective water volume V_{eff}, it might be specific to the event and depends on the reactor design and as well the RCS conditions such as water level prior to the event and the water addition during the event. In some events, however, the initial RCS water level was not accurately reported or the amount of water added to RCS during the DHR loss was not known. Such insufficient information makes it impossible to estimate the effective water volume for each event with much accuracy. Therefore, this study aimed at rough estimation of the coolant heatup rates and the time to boiling rather than precise prediction and used 50 m^3 as the effective water volume, which was estimated in the analysis of the Diablo Canyon-2 event[(23)], in the calculations for the 12 events, neglecting the difference in geometrical volume due to the reactor design. Nevertheless, in order to represent ambiguity in the effective water volume stemming from inaccurate water level, its range was defined on the assumption that the RCS water level would have been at the elevations between the top and the mid-loop of the hot leg pipings. The effective water volume used here was in the range of 50-60 m^3. It was considered that the difference in geometrical volume due to the reactor design could not be so large in comparison to this range. The following paragraph describes the estimation of the effective water volume in Ref. (23).

In the analysis of the Diablo Canyon-2 event, the effective water volume was evaluated based on the following assumptions:

- All water inside the core barrel from the elevation at the bottom of the core to the mid-loop level, including the upper plenum water, would be involved,
- 30% of the hot leg water would be involved in the heatup process, taking the distance from the core region into consideration,
- 10% of the downcomer water would be involved in the heatup process, which would cover heat transfer from inside the core barrel to downcomer water,
- The cold leg water would not be involved because the cold legs were far removed from any flow path,

- The lower plenum water would not be involved because there were no direct driving force which caused it to flow into the core region, and
- The internal structure including fuel and cladding would be affected in the heatup process and its heat capacity was about 10,000 Kca1/°C (21,700 Btu/°F).

Consequently, the effective water volume was presented as follows:

$$V_{eff} \ = \ V_{core} + V_{up} + 0.1V_{dc} + 0.3V_{hl} + V_{st} \ ≒ 50\ m^3,$$

where V_{core} : water volume in active core (18.5 m^3)

V_{up}: water volume in upper plenum from the elevation at the top of the core to the mid-loop level (18.3 m^3)

V_{dc}: water volume in downcomer below the mid-loop level (13.6 m^3)

V_{hl}: water volume in hot legs (4.3 m^3)

V_{st}: water volume equivalent to heat capacity of internal structure (10 m^3: heat capacity of which corresponds to 21,700 Btu/°F).

Water volume in each region is illustrated in **Figure III.6**.

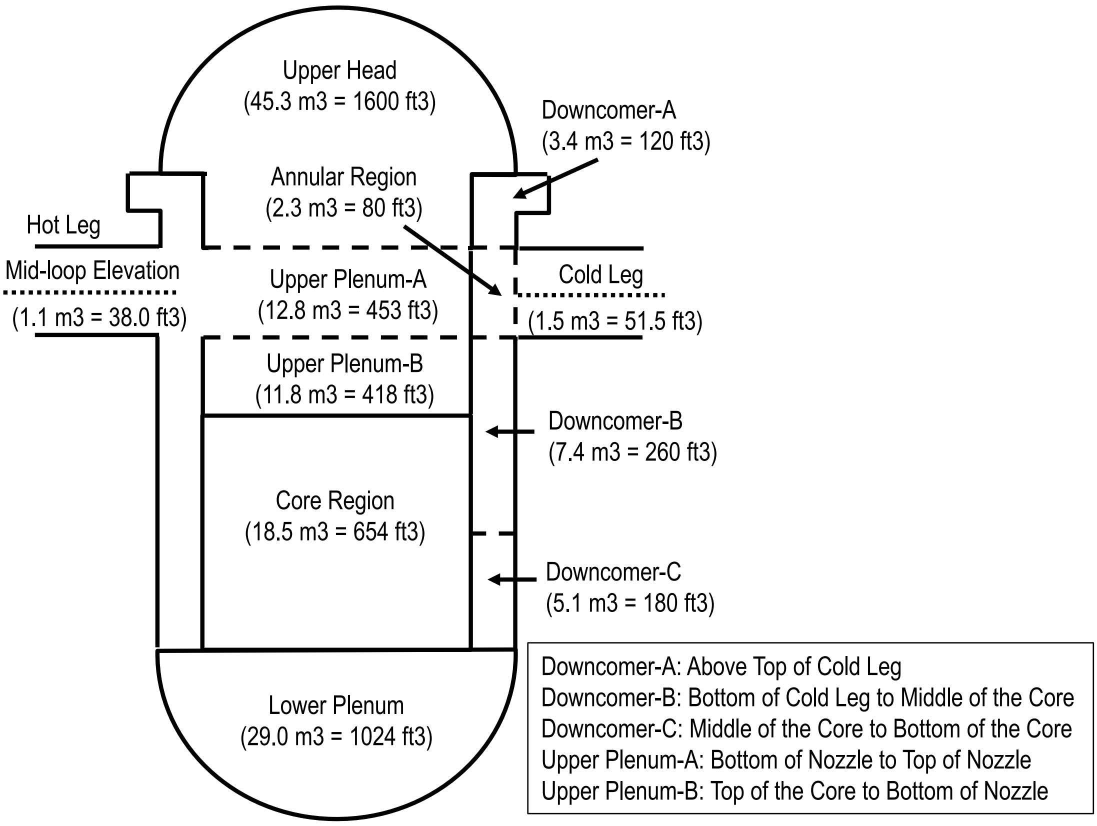

Figure III.6 Water Volume in RCS (All the data are taken from Ref.23)

3.2 Calculated Results and Discussion

For the 12 events listed in Table III.3, the calculated coolant heatup rates are summarized in **Table III.4** and are also shown in **Figure III.7** in comparison with those actually reported.

Table III.4 Summary of Calculated Results

No.	Plant Name (Event Year)	Decay Heat Power (MWt)	Duration of DHR Loss: τ (min)	Coolant Heatup Rate (°C/min) Reported* ($\Delta To/\tau$)	Coolant Heatup Rate (°C/min) Calculated* ($\Delta Tc/\tau$)	Ratio of Heatup Rates R= ΔTc /ΔTo
1	Diablo Canyon-2 (1987)	9.745	33.8	2.18	2.32 – 2.78	1.06 – 1.27
2	ANO-2 (1984)	14.695	30.0	2.21	3.50 – 4.20	1.59 – 1.90
3	Millstone-2 (1981)	9.899	30.0	2.18	2.35 – 2.83	1.08 – 1.29
4	Sequoyah-2 (1983)	6.326	35.0	0.87	1.51 – 1.81	1.73 – 2.07
5	Davis Besse-1 (1980)	6.222	150.0	0.29	1.48 – 1.78	5.00 – 6.00
6	Beaver Valley-1 (1981)	5.539	54.0	0.68	1.32 – 1.58	1.94 – 2.33
7	McGuire-1 (1987)	11.393	6.0	4.35	2.71 – 3.26	0.62 – 0.75
8	Vogtle-1 (1990)	3.071	36.0	0.71	0.73 – 0.88	1.03 – 1.24
9	Beaver Valley-1 (1978)	3.334	60.0	0.28	0.79 – 0.95	2.86 – 3.43
10	San Onofre-2 (1986)	7.922	49.0	1.09	1.89 – 2.26	1.73 – 2.08
11	Ocinee-3 (1988)	1.778	15.0	0.56	0.42 – 0.51	0.76 – 0.91
12	Palisades (1978)	7.228	20.0	1.94	1.72 – 2.07	0.89 – 1.06

*: ΔTo denotes the coolant heatup actually reported.
ΔTc denotes the coolant heatup calculated in this analysis.

In the three boiling events, the calculations used the durations of DHR losses reported as the times to boiling since they were not specified. For the Diablo Canyon-2 event, on the other hand, the heatup rate were calculated with use of the time to boiling estimated by the USNRC[(23)]. As shown in this figure, the calculated coolant heatup rates were in relatively good agreement with the observed ones in the range of factor 2 except for two events (the Davis Besse-1 event in 1980 and the Beaver Valley-l event in 1978). Differences between the calculated and observed heatup rates are considered to stem from the following major uncertainties: uncertainty in the effective water volume, inaccuracy in the reported duration of DHR loss or the time to boiling, and inconsistency of the RCS temperature observation measures. In some events, the loss durations were reported as the time from the initiation of DHR pump motor current oscillation to the DHR flow restoration, while this analysis assumed the DHR flow was completely lost during the durations without any water addition. For example, in the ANO-2 event, it was reported that the DHR was lost for 50-60 min but actually for the first 20 min the water addition to RCS was conducted by gravity feed and by oscillatory DHR pump flow. In some events, core thermocouples were disconnected and thus the RCS temperature rises reported were based on temperatures observed on hot legs or DHR pump exit after the DHR flow restoration. These indications are not representative of temperature rise in the core region when the DHR flow is lost.

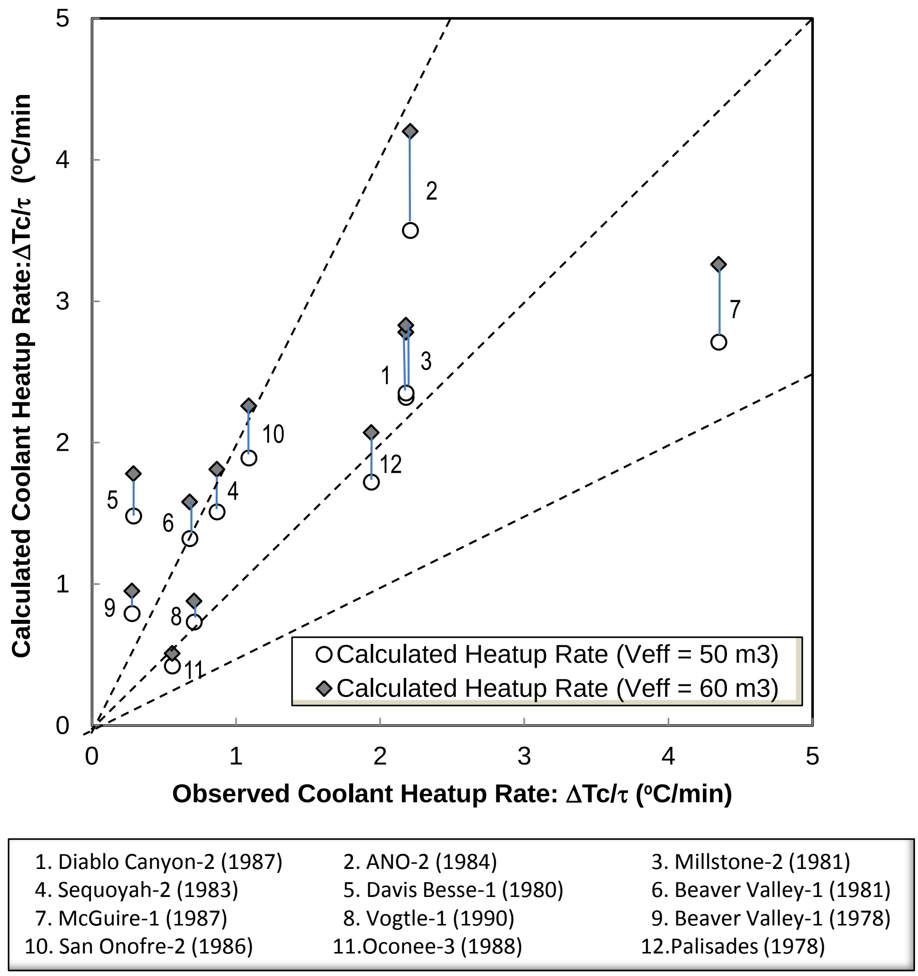

Figure III.7 Comparison of Calculated and Observed Heatup Rates

Figure III.8 shows the predicted times to bulk boiling, along with the reported ones which were simply extrapolated with use of the observed coolant heatup rates. From this figure, it is noted that bulk boiling could have occurred in seven events, including four events which have actually involved core boiling (at Diablo Canyon-2, ANO-2, Millstone-2 and San Onofre-2). For the other three events (one at Davis Besse-1 and two at Beaver Valley-1), core boiling was not identified and/or reported. As for the Beaver Valley-1 event in 1981, it was reported that some makeup water was added to

RCS during 54 min while the DHR lost. Since the amount of the added water was not specified, this calculation neglected the water addition, resulting in higher heatup rate. The causes of discrepancies for the other two events, the Davis Besse-1 event in 1980 and the Beaver Valley-1 event in 1978, remain indeterminate because of lack of information. In Ref. (13), NSAC's calculations of heatup rate in the former event showed that bulk boiling could have taken place even if the water volume being heated was assumed to be all water in the core volume, inside the core barrel, above the core (upper plenum), and in RHR, but the reason of this inconsistency was not referred at all. The latter event occurred 38 d after the reactor shutdown and the RCS temperature increased from 62.8°C (145°F) to 79.4°C (175°F) during 60 min, which was equivalent to the heatup rate of 0.28°C/min (0.5°F/min). It is hard to understand that this value is less than half of the heatup rate of about 0.71°C/min (1.3°F/min) actually observed in the Vogtle-1 event which occurred 25 d after the shutdown with the refueling completed.

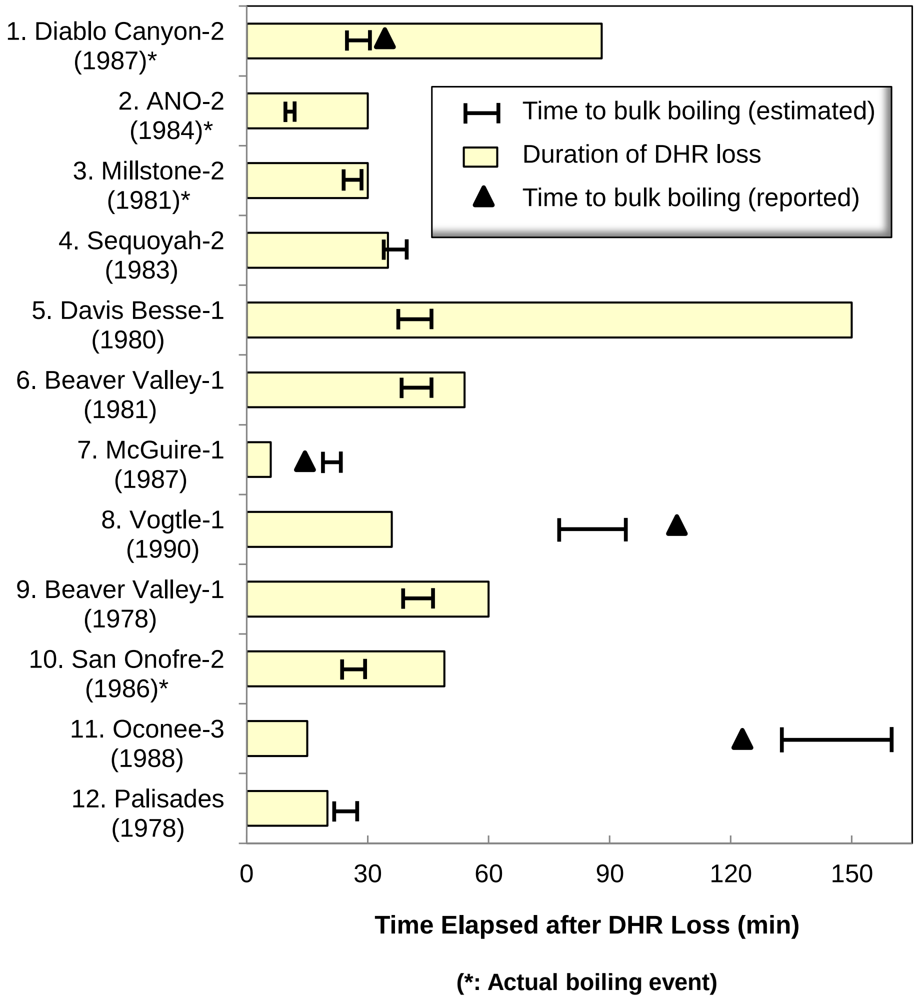

Figure III.8 Comparison of Predicted Time to Bulk Boiling with Actual Duration

Figure III.9 illustrates the calculated time curves, which show how the time to bulk boiling varies as a function of the time elapsed from the reactor trip to the DHR loss. This calculation assumed that the average coolant temperature was 32.2°C (90°F) at the initiation of DHR loss and that the reactor had been operated at the rated power of 3,411 MW thermal (typically, Westinghouse 4-loop PWRs) for 10,000 h prior to DHR loss.

As seen in this figure, bulk boiling in the core could begin within 1 h, in the absence of successful operator action, even if the DHR loss would take place 30 d after the shutdown. In the condition with RCS intact and the reactor vessel head on, RCS would be pressurized, preventing the water addition to RCS by gravity feed, and thus operators should take any corrective actions such as restoration of DHR function, manual actuation of HPI (high pressure injection) and feed and bleed operation with use of HPI and PORVs (power operated relief valves). In addition, the time to core uncovery was also calculated to examine the time margin available to recover the DHR function prior to core uncovery assuming that the water volumes in RCS and above the core were in the range of 100-118 m^3 and 32-50 m^3, respectively (see Figure III.6). The calculated time to core uncovery is shown in Figure III.9, and indicates that in the Diablo Canyon-2 event which occurred 7 d after reactor trip, core uncovery could have begun if any action would not be taken within 3 h although this event actually terminated for about 1.5 h.

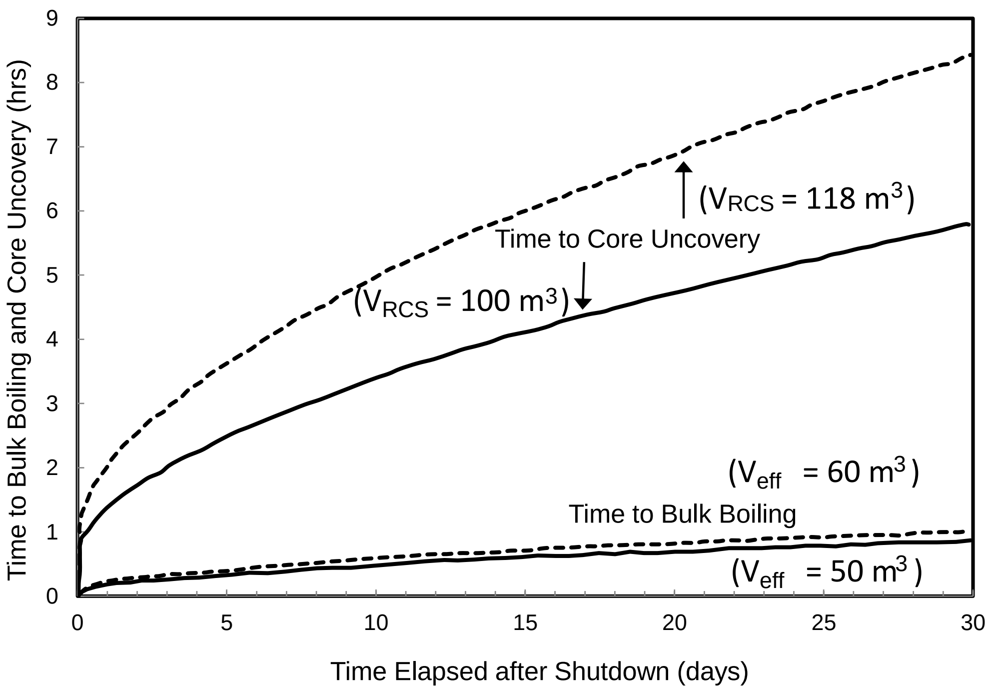

Figure III.9 Time Curves to Bulk Boiling and Core Uncovery

4. Summary

This section describes the analysis of loss of DHR events during reactor shutdown conditions which occurred at U.S. PWRs and estimated the RCS water heatup rates and the times to bulk boiling in the core and core uncovery in reduced inventory conditions.

Between 1976 and 1990, 197 loss of DHR events were reported to have occurred during approximately 850 reactor years of operation and one-third (63 events) occurred in reduced inventory conditions. Analysis of these events indicates that four-fifths (49 events) of DHR losses in reduced inventory conditions were caused by cavitation or air binding of DHR pumps, that is, air entrainment into the DHR pumps. The major contributor of air entrainment (33 of 49 events) is found to be the lowering the RCS water level too far, most of which resulted from inaccurate level indication. Other causes include loss of coolant inventory and increased DHR pump flow.

In many events involving air entrainment, the DHR loss lasted for more than 1 h and the remarkable coolant heatup was observed. Applying the analytical approach used in the Diablo Canyon-2 analysis, the coolant heatup rates and the time to bulk boiling in the core are estimated for 12 events with use of the data obtained from their respective event reports. Although the data used here are not necessarily enough for the analysis, the calculated heatup rates are in reasonably good agreement with the observed ones. As for the four boiling events, it is confirmed that core boiling could have occurred during the DHR losses. The calculated results also show that in the absence of successful operator action, bulk boiling in the core and the subsequent core uncovery would take place within 1 h and several hours, respectively, even if the DHR loss would occur in the late stages of shutdown (for example, 30 d after the reactor trip). Therefore, restoring the DHR function should be required in a short term.

III.3 Analysis of Criticality Accidents in Nuclear Fuel Processing Facilities

On September 30, 1999, a criticality accident occurred at the Tokai-mura uranium processing plant operated by JCO Co. Ltd., which resulted in the first nuclear accident involving fatalities in Japan, and forced the residents in the vicinity of the site to be evacuated and be sheltered indoors. Considering the gravity of the accident, the Nuclear Safety Commission of Japan (NSC) set up the accident investigation committee to identify the causes of the accident and examine the measures for preventing recurrence and its report of the investigation was issued on December 24, 1999(25). The report concluded that the direct cause of the accident was the injection of uranyl nitrate solution with uranium exceeding the critical mass limit into the precipitation tank with unfavorable geometry. As well, the report identified the root causes and contributing factors, and proposed countermeasures.

On the other hand, at least 21 criticality accidents had taken place in foreign fuel processing facilities before the JCO accident. As seen from **Figure III.10**, most of them occurred in 1950's to 1960's. The accidents in the United States and the United Kingdom were described in the reports published by the U.S. Los Alamos National Laboratory (LANL)(26,27), the International Atomic Energy Agency (IAEA)(28) and the Japan Atomic Energy Research Institute (JAERI)(29,30). However, the accidents in Russia had not necessarily been reported in a timely manner. In 1995, twelve accidents were published in the International Conference on Nuclear Criticality Safety (ICNC-95)(31). In 1999, a Russian accident in 1997 was additionally reported and more detailed information on the four accidents was provided in the subsequent ICNC-99 (32-34).

These 21 accidents were analyzed in terms of similarities to the JCO accident, focusing on the accident sequences and causes of accidents, to identify the lessons learned from foreign accidents which should have been fed back(5,6).

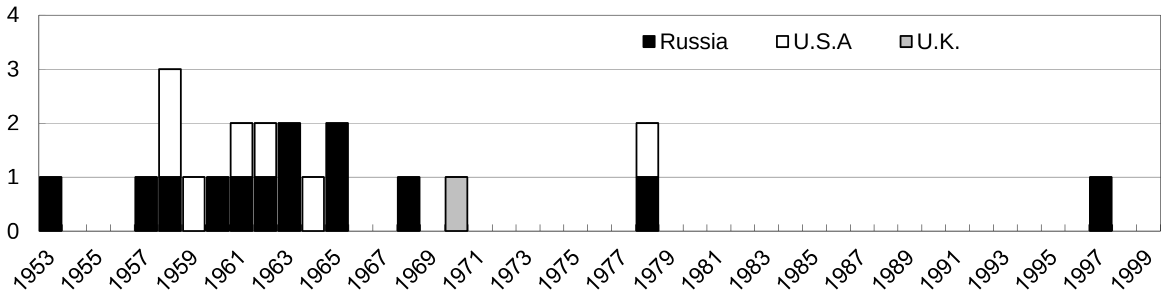

Figure III.10 Number of Criticality Accidents Year by Year

1. Outlines of JCO Accident

1.1 Chronology of Accident

At about 10:35 am, on September 30, 1999, when the volume of uranyl nitrate solution in the precipitation tank (volume: 100 *l*, diameter: 450 mm, height: 600 mm) reached approximately 45 *l*, equivalent to about 16.6 kg of 18.8% enriched uranium, a criticality accident occurred. The nuclear fission reaction in the precipitation tank produced an initial burst of radiation, both neutron and gamma rays. The area gamma monitors detected a high level of radiation and the area alarms sounded. The three workers involved in the operation were evacuated from the building. After the first few minutes, a quasi-steady state fission reaction set in and the radiation emission rates of neutron and gamma rays became essentially stable. As a result, the quasi-steady state fission reaction lasted for more than 20 h until water was drained from the cooling jacket surrounding the precipitation tank. The first attempt to drain water was made by opening the drain valve at about 2:30 am on October 1, 1999, but only a small volume of water was drained, decreasing neutron dose rate somewhat. The water pipe was then broken and cut and finally, at about 6:15 am, argon gas was pumped into the pipe to force out water, terminating the fission reaction. To ensure that the subcritical condition would be maintained, about 17 *l* of borated water with concentration of 25 g/*l* was injected into the tank. At about 8:50 am, the termination of criticality condition was declared. The total number of fissions during the reaction was estimated to be 2.5×10^{18} based on the analysis results of samples from the precipitation tank.

Two of the three workers in that area, that is, one holding the funnel and the other pouring the solution into the funnel, had suffered from severe radiation sickness, resulting in loss of their lives about three and seven months later, respectively. Based on the radiation doses measured by whole body counters, radiation exposures were observed to 56 JCO employees, three firefighters (rescue crews), 24 workers for terminating the criticality, seven public members near the site, and 57 emergency preparedness staff members. In addition, approximately 300 people were supposed to have been exposed to radiation [(35)].

During the accident, evacuation of residents within 350 m of the facility was initiated at about 15:00 on September 30, 1999 and residents within a 10 km radius of the facility were advised to stay indoors at about 22:30 on the same day as a precautionary measure. While the latter recommendation (to stay indoors) was rescinded at about 16:40 on October 1, the evacuation lasted until the next evening (at about 18:30 on October 2).

1.2 Accident Causes

According to the committee's report[(25)], the direct cause of the accident is that uranyl nitrate solution with uranium exceeding the critical mass limit was poured into the precipitation tank with unfavorable geometry, contrary to its originally intended use. The precipitation tank was designed to be used in the process for dissolving, refining and purifying crude U_3O_8 but was not intended for use in the process for redissolving the purified U_3O_8, homogenizing uranyl nitrate solution and producing the final product.

Although the mass limit for criticality safety of the precipitation tank was specified as 2.4 kgU for uranium with enrichment of 16-20%, the amount of actually loaded uranium reached approximately 16.6 kgU. Any concentration limit for criticality safety was not specified in addition to the mass limit. In the previous operations, furthermore, the storage tower with favorable geometry had been used in the process for homogenizing the uranyl nitrate solution with the mass limit being exceeded and no criticality accident had occurred. These might have caused the workers to misunderstand that the precipitation tank could be safely used for the same purpose. The root causes and contribution factors were identified inadequacies/deficiencies in operating process, operation management, technical management, and management control, and problems in licensing process and safety regulations (see **Table III.5**).

Table III.5 Issues Pointed out by JCO Accident Investigation Committee[1]

Issues Pointed out	Description
Issues on Work Processes: - Inadequate work process to redissolve the purified U_3O_8 in nitric acid and to homogenize uranyl nitrate solution	- In the process of producing uranyl nitrate solution, a 10-*l* stainless steel container was used instead of the dissolving tank which was supposed to be used. - The homogenizing process carried out for producing the final product had been unreviewed and unapproved by the competent authority. In order to improve workability for pursuing the efficiency, the homogenizing process had been changed from a cross-blending method with use of 10 containers of 4-*l* capacity to an alternative method by using the storage tower with a favorable geometry, and then, use of the precipitation tank with an unfavorable geometry was introduced instead of the storage tower.
Issues on Operational Control - Exceeded critical mass limit	- The uranium exceeding the critical mass limit was loaded into the precipitation tank in violation of the licensing conditions and technical specifications of facility. - The homogenization of 6-7 batches of uranly nitrate solution was a deviation from the license conditions.
Issues on Technological Control: - No appropriate procedure defined for obtaining the approval of the relevant person when preparing or revising work procedures/instructions.	- There was no system to check whether or not a proposal of using the precipitation tank was valid for homogenization process. - Although the work instruction had not provided any description on use of the precipitation tank and the operating procedures had specified that the storage tower be used for homogenization, the precipitation tank was applied with no approval of a person in charge etc.
Issues on Administrative Control: - Lack of due consideration on characteristics of the work concerned which was the small-scale, non-routine and unique operation compared with the main business	- Since the work to produce uranyl nitrate solution requires several days and has been executed on an infrequent basis, the operators had been inexperienced in the work. - Although there were essential differences in criticality control between the work concerned and the main business with 5% or less enriched uranium being handled, special attention had not been paid to production process and equipment.
Issues on Licensing: - Insufficient description on dissolving process during safety review and design & construction permission review	- The license application did not specify any reason why the dissolving tank was designed and managed based on both mass control and geometry control. - Neither the license application nor the design & construction application had provided any description on the homogenization process. If this process had been added to the work processes after these applications, this change should have been applied at that time as an amendment.
Issues on Safety Regulations: - Ineffective regulatory inspection on compliance with technical specifications	- The competent authority had examined whether or not operations had been conducted in compliance with the technical specifications, by dispatching its inspectors. However, the facility had been operated on an infrequent basis and thus, these inspections had been performed during the facility shutdown.

2. Overall Trends of Criticality Accidents in Foreign Countries

This subsection discusses the 21 criticality accidents in foreign countries on the type of nuclear materials handled, the geometry of vessels/tanks used, the extent of criticality (duration time, fission number), occurrence of recriticality, the mechanism of criticality termination, and the consequence of radiation exposures, in comparison with those of the JCO accident. **Table III.6** summarizes the individual accidents and their respective detailed event descriptions are provided in the Ref.(6).

Table III.6 Summary of 21 Criticality Accidents

No.	Site and Date	Features of Accident	Total Fissions	Duration
1	Mayak Enterprise (Russia) March 15, 1953	plutonium nitrate solution in a receiver tanks; single spike; two significant exposures (1000 rad, 100 rad)	2.5×10^{17}	unknown
2	Mayak Enterprise (Russia) April 21, 1957	uranium precipitate in an oxalate purification chamber; one fatality and five workers with radiation sickness	2×10^{17}	10 min
3	Mayak Enterprise (Russia) January 2, 1958	uranium nitrate solution in a critical parameters measurement tank; single spike; three fatalities and one with loss of eyesight	2.3×10^{17}	short period
4	Mayak Enterprise (Russia) December 5, 1960	plutonium carbonate solution in a vessel with unfavorable geometry; two spikes; several exposures of lower than 5 rad	1×10^{17}	short period
5	Siberian Chemical Combine (Russia) July 14, 1961	uranium hexafluoride accumulation in a vacuum pump oil vessel; two spikes; one significant exposure (approx.200 rad)	1.2×10^{16}	short period
6	Mayak Enterprise (Russia) September 7, 1962	plutonium nitrate solution in a dissolving tank; three spikes; no significant exposure	2×10^{17}	40-50 min
7	Siberian Chemical Combine January 30, 1963	uranium nitrate solution in a tank with unfavorable geometry; eight spikes; four insignificant exposures (6-17 rad)	7.9×10^{17}	10 h
8	Siberian Chemical Combine (Russia) December 2, 1963	uranium nitrate solution accumulation in a trap; sixteen spikes; no significant exposures	1.6×10^{16}	16 h
9	Electrostal Fuel Fabrication Plant (Russia) November 3, 1965	uranium oxide slurry in a vacuum supply tank; single spike; no significant exposure	1×10^{16}	unknown
10	Mayak Enterprise (Russia) December 16, 1965	uranyl nitrate solution in a dissolving tank; eleven spikes; no significant exposure	7×10^{17}	7 h
11	Mayak Enterprise (Russia) December 10, 1968	plutonium nitrate solution in a 60 *l* vessel; two spikes; one fatality and one with both legs amputated	6×10^{16}	unknown
12	Siberian Chemical Combine (Russia) December 13, 1978	plutonium ingots in a storage container; single spike; one significant exposure (250 rad for whole body and 2000 rad for hands) and seven exposures (5-60 rad)	3×10^{15}	short period
13	Novosibirsk Chemical Concentration Plant (Russia) May 15, 1997	uranium dioxide deposition in receiver vessels; six spikes; no significant exposure	5.5×10^{15}	27 h

Table III.6 Summary of 21 Criticality Accidents (continued)

No.	Site and Date	Features of Accident	Total Fissions	Duration
14	Oak Ridge Y-12 Chemical Processing Plant (USA) June 16, 1958	uranyl nitrate solution in a drum for receiving water; two spikes; eight significant exposures (28.8-461 rem)	1.3×10^{18}	18 min
15	Los Alamos Scientific Laboratory (USA) December 30, 1958	plutonium organic solution in a large process vessel; one spike; one fatality and two significant exposures (134 and 53 rem)	1.7×10^{17}	2 sec
16	Idaho Chemical Processing Plant (USA) October 16, 1959	uranyl nitrate solution in a waste receiving tank; one spike; two significant exposures (50 and 32 rem)	4×10^{19}	20 min
17	Idaho Chemical Processing Plant (USA) January 25, 1961	uranyl nitrate solution in a vapor disengagement cylinder; one spike; no significant exposure	6×10^{17}	2-3 min
18	Hanford Works (USA) April 7, 1962	plutonium solution in a transfer tank; multiple spikes; three significant exposures (19, 43 and 110 rem)	8×10^{17}	37.5 h
19	Wood River Junction (USA) July 24, 1964	uranyl nitrate solution in a sodium carbonate makeup tank; two spikes; one fatality and two significant exposures (60-100 rad)	1.5×10^{17}	unknown
20	Windscale Works (UK) August 24, 1970	plutonium solution in a transfer tank; one spike; no significant exposure	1×10^{15}	10 sec
21	Idaho Chemical Processing Plant (USA) October 17, 1978	Uranyl nitrate solution in a lower section of scrubbing column; one spike; no significant exposure	2.7×10^{18}	1.5 h

Note: 100 rad = 1 Gy, 100 rem = 1 Sv

2.1 Type of Nuclear Materials

The nuclear material which was being handled at the time of accident is uranium in 13 of the 21 events and plutonium in 8 events, as shown in **Figure III.11**. In 11 events, uranium enriched to about 90% of ^{235}U was handled and in the other two events, uranium enrichment is 22.6% and 6.5%, respectively. However, in the accident at the Electrostal Fuel Fabrication Plant in 1965, the processing stream had been operated with 2% enriched uranium for one year and then, it was reconfigured and restarted using 6.5% enriched uranium, resulting in criticality accident 12 d after its restart(33).

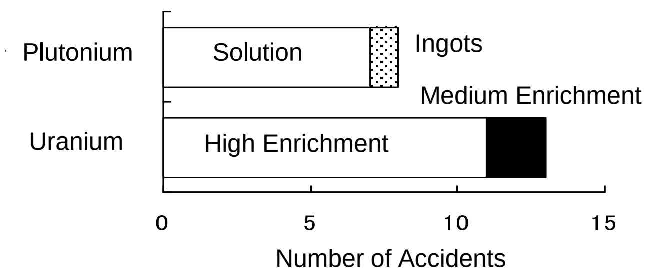

Figure III.11 Types of Nuclear Materials Handled

As for the form of nuclear materials, 20 criticality accidents took place with the fissile material in a liquid and one occurred with metal ingots. However, in one accident (at the Siberian Chemical Combine in 1961), the criticality took place in the tank of vacuum pump because of uncontrollable accumulation of gaseous uranium hexafluoride (HFU) due to the equipment malfunction combined with a human error(32). Also, the Electrostal accident in 1965 was caused by the UO_2 powder having entered the water circuit of the vacuum system and accumulated there due to two operational mistakes associated with filters. The criticality accident at the Siberian plant in 1978 occurred during the loading of one of the plutonium ingots from one container into the other one(31).

The JCO accident took place while processing the unranyl nitrate solution with 18.8% enriched uranium. However, this process stream was essentially different from that for handling low-enriched (4.5-5%) uranium solution. Thus, it can be said that the JCO workers had not been versed in handling of high-enriched uranium solution, similar to the accident at the Electrostal plant in 1965.

2.2 *Geometry of Vessels/Tanks*

Twenty of the 21 accidents occurred in the cylindrical vessels/tanks with unfavorable geometry. However, the criticality took place in a slab vessel designed as favorable geometry at the Novosibirsk Chemical Concentration Plant in 1997(34). Over many years, a large amount of UO_2 was deposited on the vessel walls and as a result, the walls were deformed from the original shape, eventually leading to the criticality. This accident implies that the criticality may not be avoided if a vessel is designed as a favorable geometry.

As mentioned above, in two accidents (at Siberian in 1961 and Electrostal in 1965), the criticality condition was reached in the tank of vacuum system. In addition, at the Siberian plant in 1963, the chain reaction took place in a vacuum trap which had been installed to prevent the uranium solutions from entering the vacuum system(32). As well, the criticality accidents at the Mayak Enterprise in 1958(31) and the Y-12 Chemical Processing Plant in 1958(27) occurred in a tank installed for experimental purpose and a drum intended to receive water for leak test, respectively. It should be noted that in these five accidents, the power excursion took place in the tank/drum, which was not used in the normal process stream.

Two criticality accidents occurred in the vessels, use of which was not expected in the design (at Novosibirsk in 1997 and the Wood River Junction Scrap Recovery Plant in 1964[27,29]). In the JCO accident, the precipitation tank was designed to be used in the refining process of U_3O_8 but was not intended for use in the process for redissolving the purified U_3O_8 and producing the final product. In this point, the JCO accident is similar to these two accidents.

2.3 Duration Time and Fission Number

The duration time is defined, in this study, as the period from the time when the first power excursion/spike was observed to the time when the subcritical condition was confirmed. While the duration time ranges from several seconds to less than 1 h in eight accidents, the critical condition was maintained intermittently or continuously over several hours in five accidents (**Figure III.12**). In the accident at the Hanford Works in 1962, the critical condition continued over 37 h intermittently[27]. Also, six power excursions occurred during 27 h in the Novosibirsk accident in 1997 and 16 power spikes were observed during 16 h in the Siberian accident in 1963. In the JCO accident, the critical condition continued for about 20 h, that is the second longest. Since most of the accidents with the duration time not being specified were terminated due to the solution flowing out of or being splashed from the vessel, it can be assumed that the subcritical condition was reached in a short time for them.

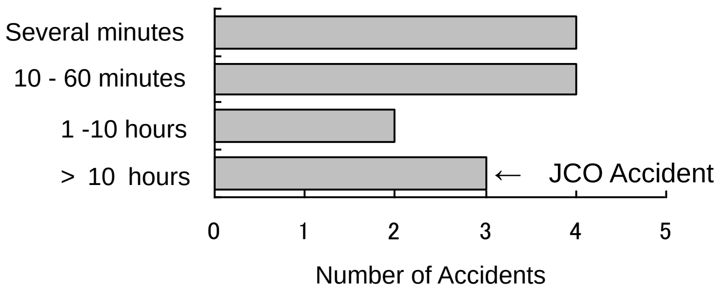

Figure III.12 Duration Time of Criticality

The total fission number ranges from 10^{17} to 10^{18} in 11 accidents, less than 10^{17} in 7 accidents and larger than 10^{18} in three accidents as shown in **Figure III.13**. Compared with these accidents, that in the JCO accident is relatively large (2.5×10^{18}). The largest three fission numbers (4×10^{19}, 2.7×10^{18}, 1.3×10^{18}) were estimated for the accidents at the Idaho Chemical Processing Plant in 1959 and 1978[27] and the Y-12 plant in 1958. In these accidents, the duration times were relatively short (20 min, 18 min and 1.5 h, respectively) and, particularly two of them involved power oscillations in a few minutes shortly after the initial spikes, resulting in the larger number of fissions. On the other hand, in the JCO accident, the critical condition was maintained for a long time because water had been held in the cooling jacket surrounding the precipitation tank. As a consequence, a large number of fissions were produced but the reason of the large fission number is different from the other accidents.

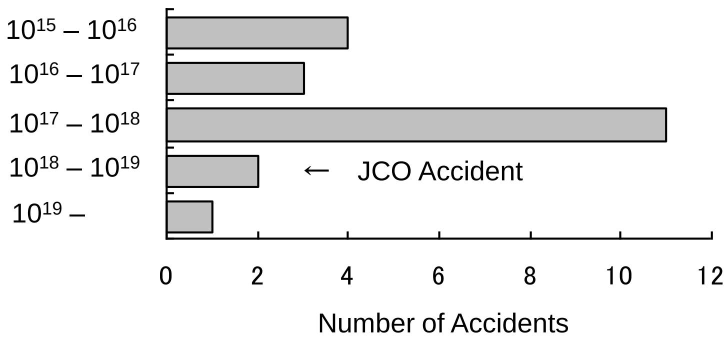

Figure III.13 Fission Number

2.4 Recriticality

Although the first power excursion stopped due to the solution surging into the connecting lines, in the accident at the Mayak Enterprise in 1960, the vacuum system was switched off, resulting in the solution flowing back into the vessel and the subsequent second excursion[(31)]. Also, in another Mayak accident in 1968, the shift supervisor re-entered the processing area and tipped the vessel in order to pour some liquid into the drain, leading to the second spike, after the area was evacuated due to the first criticality[(31)]. In the Wood River Junction accident in 1964, two workers entered the area and turned off the stirrer, creating the geometry change and the subsequent second excursion. It is considered that recriticality in these accidents was caused by lack of workers' knowledge on the criticality. In particular, the second and third accidents mentioned above involved a death of the supervisor/worker. However, the death might have been avoided if he would have enough been trained and understood the hazard and characteristics of criticality well.

As well, in the Novosibirsk accident in 1997, the first critical condition was terminated by injecting the boron solution. After that, nevertheless, several excursions occurred. Finally, lithium chloride solution with much higher solubility was injected into the vessel. This accident implies that the boron solution might not ensure the criticality is terminated completely. Thus, it is necessary to take its solubility into account when selecting the poison to be injected.

2.5 Mechanism of Criticality Termination

Mechanisms to terminate the criticality can be divided into two categories; one is a passive aspect and the other is an active aspect as shown in **Figure III.14**. The former includes the solution flowing/spattering out of the vessel, the solution boiling and the sediment formation, and did not need to take any human action. Eleven accidents were terminated by such mechanisms. The latter consists of injecting the poison, transferring or draining the solution and diluting or stirring the solution, and required taking human actions actually. Three accidents were terminated by injecting the poison: the cadmium solution was used in two of these accidents (at Siberian in 1963 and Mayak in 1965[(31)]) and the lithium chloride solution was used finally after the boron solution was injected in the Novosibirsk accident in 1997. As well, in other four accidents, uranium solution was drained from the vessel, stopping the chain reaction. However, in two of them, the workers were overexposed, indicating lack of the workers' knowledge on the criticality and inadequacies of the decision making when taking actions for terminating the accident. In addition, three accidents were terminated by the solution being stirred and diluted due to the continuous operation of equipment after the criticality occurred. In the JCO accident, applied were the different approach, that is, draining of water from the cooling jacket surrounding the precipitation tank. The boron solution was also injected but this aimed at ensuring the subcritical condition.

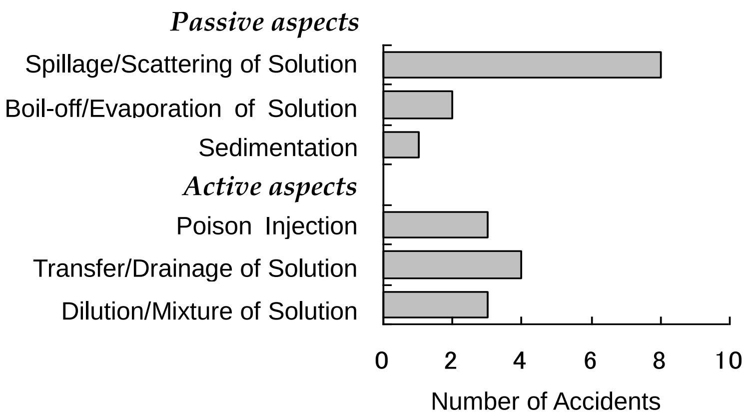

Figure III.14 Mechanism of Stopping Criticality

2.6 Radiation Exposures

Five criticality accidents involved a total of 7 fatalities. In particular, three workers died and one lost his eyesight in the Mayak accident in 1958, resulting in the worst accident from the viewpoint of human damage. Another accident at the Mayak Enterprise in 1968 involved one death and one with both legs being amputated. In the JCO accident, two workers died and one worker was overexposed. Considering that approximately 440 people including general public received radiation doses, it can be said that this accident was the most serious one. Accidents in shielded facilities did not result in direct exposures in excess of occupational limits.

3. Causes of Criticality Accidents in Foreign Countries

In order to understand why or how the criticality accidents took place, their causes of individual accidents were analyzed and organized from the generic point of view. **Figure III.15** shows the number of accidents by causes. Most of the accidents were caused by several factors and thus, a total number exceeds the number of occurrences.

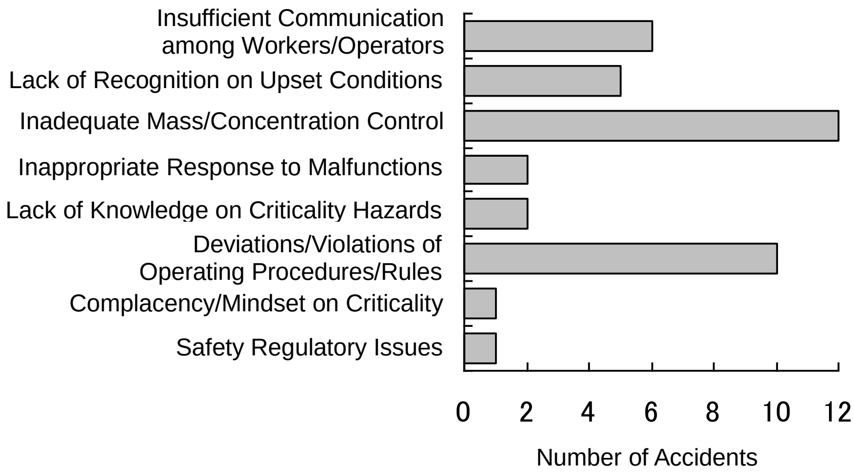

Figure III.15 Causes of Criticality Accidents

(1) Insufficient Communication among Workers/Operators

Lack of sufficient communications among workers/operators was a major contributing factor in six accidents. For example, in the Y12 plant accident in 1958, the operators did not document the fact that the uranium solution had leaked and thus, the coming shift did not notice it and drained the testing water containing the accumulated uranium in order to execute the leak test of process equipment, resulting in the criticality. The Wood River Junction accident in 1964 stemmed from lack of communication between shifts. Concretely, the concentrated uranium solution was drained into bottles identical to those that normally held the very-low-concentration uranium-contaminated trichloroethane (TCE) solution. Since this fact was not transferred to the next shift, a bottle of the concentrated solution was mistaken for the TCE solution and the operator poured it into the tank, resulting in the criticality and his death due to overexposure. Other three accidents were attributable to miscommunication of solution sample concentrations (at Siberian in 1963, Mayak in 1965 and 1968). Particularly, one of them was attributed to the worker having misunderstood the unit on uranium content and underestimated uranium mass (at Siberian in 1963). In the sixth accident (at Idaho in 1961[(27)]), a worker started the pump and opened the suction valve prior to its associated valve being closed by another worker, allowing the high-pressure air to enter the tank and resulting in the criticality. Such communication issues should be disseminated more widely to establish the robust human performance tools.

(2) Lack of Recognition on Upset Conditions

Some of the accidents might have been avoided if the workers involved would have been familiar with and/or have well understood the criticality and the operating processes. For example, in the Mayak accident in 1953[(31)], the operator saw foam when disconnecting the hose from the vessel and thus, he reconnected the hose; however, he did not notice that the criticality took place, resulting in his overexposure. Another accident at the Mayak Enterprise in 1957[(31)] involved a death of worker because he did not notice that the criticality occurred though he observed the gases having been released from the oxalate precipitate. In the third accident at the Mayak Enterprise in 1960, a technician found a discrepancy in the plutonium mass analysis but transferred the solution with no check of the results, leading to the criticality. In the Siberian accident in 1978, an operator extracted ingots manually from the container after the excursion occurred, causing him to receive an excessive dose.

(3) Inadequate Mass/Concentration Control

Twelve accidents were attributed to inadequate mass and/or concentration control. Out of them, six accidents were associated with precipitates or deposits having not been monitored in the facility (at Mayak in 1957, Y-12 in 1958, Siberian in 1961, Electrostal in 1965, Windscale in 1970[(27)], and Novosibirsk in 1997), two cases with no device having been installed for measuring the concentration of solution (both at Siberian in 1963[(31,32)]), and one with the plutonium mass measurement having not been recorded adequately (at Mayak in 1958). In another accident at the Siberian plant in 1978, as well, the ingots were loaded without using instrument to monitor plutonium mass, resulting in the criticality. It can be said that these accidents occurred as a result of inadequate process control. The other two accidents resulted from scrap material having been stored without measurement of its plutonium content and scrap with different uranium content having been stored in one spot, respectively (at Mayak in 1962[(31)] and 1965). These accidents reveal the problems in accounting and storage control.

(4) Inappropriate Response to Equipment Malfunctions

Inappropriate responses to process malfunctions led to the criticality. In the Idaho accident in 1959, the inadvertent initiation of a syphoning action during an air-sparging operation resulted in the transfer of solution from a tank with favorable geometry to another one with unfavorable geometry, leading to the criticality. The air-sparging operation was carried out because the pump normally used for stirring the solution had been inoperable. Another accident at the Idaho plant in 1961 is thought to have been caused by a bubble of high-pressure air forcing uranyl nitrate solution into a cylinder with unfavorable geometry. The high-pressure air was used to correct the line plugged.

(5) Lack of Knowledge on Criticality Hazards

Several accidents were associated with the workers having not been aware of criticality hazards. In the accident at the Siberian plant in 1961, the automatic supply of liquid nitrogen to the main cylinder for condensing gaseous uranium hexafluoride (HFU) was stopped to restrict consumption of liquid nitrogen and also, the manual supply was not performed. The resultant insufficient cooling, combined with processing parameters having not been observed, made a portion of HFU to get into and the uranium to intensively accumulate in the vacuum system, causing the criticality. In the Mayak accident in 1968, although the chief operator and the operator noticed that the liquid was dark-brown indicating high plutonium content, the chief operator poured the first portion of liquid into the vessel. After that, he gave the order to repeat the operation and left the area. When the operator poured the second portion of liquid into the vessel, the criticality condition was reached and all personnel were evacuated. However, the shift supervisor re-entered the area and tipped the vessel to drain the liquid. As a result, the second criticality occurred and the supervisor died. It can be said that lack of knowledge and/or low level of recognition on criticality hazards might have led to the criticality and/or the fatality. These accidents imply the importance of operator training.

For the JCO accident, it was pointed out that the operators had not enough been aware of criticality hazards. For example, the operators might have mistakenly assumed that the precipitation tank with unfavorable geometry could be used for homogenizing uranyl nitrate solution without any problem because the storage tower with favorable geometry had been previously used for homogenization and nothing had happened even though the critical mass limits had been exceeded. The reconversion process at the time of accident was being executed by a special team which had usually been involved in the liquid waste treatment and had lacked the knowledge and experience on the process.

(6) Deviations/Violations of Operating Procedures/Rules

Deviations and/or violations of operating procedures and/or rules contributed to 10 criticality accidents. These can be divided into the cases that operators did not follow the requirements specified and the cases that they took actions deviating or violating the rules and regulations. The former cases include 4 accidents, in which the uranium had gradually accumulated, leading to the criticality, because of failure to conduct periodic cleaning of equipment, sampling of solution, checking of filters, or manual supplying of liquid nitrogen (at Windscale in 1970, Electrostal in 1965, Mayak in 1957, Siberian in 1961). The latter cases contain 6 accidents. The Mayak accident in 1960 was caused by transferring the solution with plutonium mass which exceeded the acceptable error for loading product stipulated in the procedures, and the Los Alamos Scientific Laboratory accident in 1958[27] was due to unexpected plutonium-rich solids being washed into a vessel of dilute aqueous and organic solutions. In addition to these two accidents, the rules and regulations were violated as a result of taking actions to improve the work efficiency, leading to four accidents. Of these, the Mayak accident in 1958 resulted from the tank being tipped to drain the solution faster and the resultant criticality caused three fatalities and one with loss of eyesight. In another accident at the Mayak plant in 1965, the time specified for dissolving scraps were shorten because of a scheduled chamber cleanup. The Wood River Junction accident in 1964 was attributable to the operation being shifted from small bottles with favorable geometry to a sodium carbonate makeup tank with unfavorable geometry, which was not expected in the area. This shift proposed and implemented the processing of an unexpectedly large amount of uranium-contaminated solution without any approval from plant management. The makeup tank was not intended to use for processing the solution. Also, in the Novosibirsk accident in 1997, a total of 6 power excursions occurred in two receiver vessels but the receiver vessels had been used in the process for 13 years without approval from the regulatory authority. These accidents demonstrate that placing a priority on the improvement of work efficiency and/or productivity would lead to less safety consciousness and subsequent violations of rules.

In the JCO accident, the use of precipitation tank was proposed and implemented to improve the work efficiency because a lot of time had been required for homogenizing the solution with use of the storage tower and it was not easy to extract the solution from the tower. However, the homogenization with use of the tower had also been carried out without approval from the regulatory authority. Additionally, the maximum allowable mass for the tower and the critical mass limit for the precipitation tank had been exceeded, and the stainless steel container, the storage tower and the precipitation tank had been used for different purposes from their respective design intention. All of these were also violations of rules and regulations.

(7) Complacency/Mindset on Criticality

The complacency or mindset on criticality is observed in the Novosibirsk accident in 1997. Since all of the vessels used were of favorable geometry (according to design and records), the facility did not have the ability to determine uranium concentration of the solutions generated. The possibility of precipitate formation and uranium deposition in the service piping was also not monitored. Despite the discovery of a solid UO_2 deposit, therefore, the logical search for similar deposits in the service piping, receiver and holding vessels was not initiated. However, the receiver vessels were not favorable geometry for the conditions encountered and the actual internal thickness of receiver vessels was larger than the designed one, significantly affecting the criticality safety margin. Thus, the uranium has gradually accumulated in the vessels, resulting in the criticality.

Also, in the JCO accident, the workers had performed the system/process changes and used the storage tower for homogenizing the solution since the tower was of favorable geometry and the critical condition had not been reached even though 6 or 7 batches were loaded. It was also revealed that the chief engineer for criticality control, who had initially been stationed, was abolished and the licensed chief engineer for handling nuclear fuels had not been involved in the process for developing criticality control criteria and operating procedures. These facts seem associated with complacency/ mindset on criticality in the management level as well as the operators.

(8) Safety Regulatory Issues

The Novosibirsk accident in 1997 was associated with the long term operation without any approval of the competent authority. Although, in this plant, the vessels with favorable geometry had been used in the process different from their original purposes for 13 years, the licensee had not made the application on their use and thus, the regulatory body had not recognized the fact. As well, the JCO plant had used the stainless steel vessel in the dissolving process, implemented the homogenizing process, used the storage tower in the homogenizing process and made the associated system/process modifications for a long time without any approval from the regulatory authority. The authority had not recognized such processes and modifications at all.

4. Summary

The analysis of 21 criticality accidents in foreign countries shows that the most of them took place when handling uranium or plutonium solutions in the vessels with unfavorable geometry and some common issues were identified in the accident scenarios and causes. For example, the operating procedures are proposed and implemented to improve the work efficiency, the vessels were used for the different purpose from their original one, the operating procedures and/or rules were violated due to the priority being given to production and the operating procedures/processes were changed and implemented without any application to the regulatory authority. Lack of understanding of criticality hazards and/or the complacency on criticality contributed to accidents and to exacerbated consequences. It can be said that these are the results from the administrative issues in the company.

While the accidents in the United States and the United Kingdom had been reported at the time of their occurrences, those in Russia had not been published for a long time and thus, any insights from the Russian accidents had not been available. If the lessons learned from the past accidents would have been prevailed worldwide and as a result, for example, the mass and concentration controls would have adequately been performed and the operating rules and regulations have been adhered, similar accidents including the JCO accidents would have not been repeated. Finally, this study points out the need to analyze the past experience and to disseminate the insights obtained from the analysis into the nuclear community worldwide in a timely manner.

III.4 Trend Analysis of Sepoint Drift in Safety or Safety/Relief Valves at U. S. LWRs

Safety valves or safety/relief valves are used to prevent overpressurization of RCS at LWRs. For example, SRVs are installed on the main steam lines between the reactor and inboard main steam isolation valves (MSIVs) at BWRs and the safety valves are on the top of pressurizer at PWRs to relieve pressure by lifting at the required setpoint in the event of load rejection, loss of feedwater and so on. On the other hand, these valves play a role of maintaining the RCS pressure boundary integrity by seating tightly without leakage during the normal operation. At PWRs, MSSVs are also installed on the secondary system to prevent its overpressurization, provide cooling to RCS in the event of MSIV closure prior to reactor scram and remove post-scram decay heat by lifting at the appropriate setpoint, thereby assisting in prevention of the RCS overpressurization.

As well, MSSVs preserve the steam line integrity by seating tightly without leaking.

The setpoints for opening the safety valves or safety/relief valves are determined to have allowable tolerances, taking into account the setpoint drift of valves in service and thus, the in-service testing of valves are carried out routinely. While the setpoint drift low may result in a premature lift of the valve, leading to plant transient, for example, the setpoint drift high may cause the RCS overpressurization, threatening the integrity of pressure boundary. Since the beginning of 1980s, in the United States, there have been many LERs describing setpoint drift in safety or safety/relief valves. The USNRC has issued a lot of generic communications on this issue and the industry has made its efforts to resolve the issue. In the first half of 1990s, the analysis of operating experience involving the safety or safety/relief valve malfunctions was performed and pointed out the importance of feeding such experience back to the testing and maintenance practices(36). As well, the nuclear power industry in United States has analyzed the causes of setpoint drift events and based on the analysis, several corrective actions have been implemented. For example, the guidelines have been prepared for conducting the testing and maintenance of valves and the design modifications have been made to eliminate the causes(37). In the period from January 2001 to August 2006, however, there have been over 70 LERs addressing setpoint drift of safety or safety/relief valves and therefore, the USNRC issued the information notice to alert addressees of operating experience associated with PSVs, MSSVs and SRVs exceeding the lift setpoint tolerance required by technical specifications(38).

As mentioned above, the setpoint drift of these valves has been observed at many plants over the years and corrective actions have been taken but this issue has not yet been resolved. This study examines the trend of the U.S. experience with setpoint drift by reviewing approximately 90 LERs from 2000 to 2006 focusing on causes and setpoint deviation ranges to provide the insights useful for improving the reliability of these valves (10, 11).

1. General Description of Safety and Safety/Relief Valve

1.1 Safety/Relief Valves at BWRs

SRVs can actuate by either of two modes, the safety mode and the relief mode, to provide the RCS overpressure protection and as well, some of them are used by the automatic depressurization system (ADS), one of emergency core cooling systems, and can be manually operated. To provide adequate protection, 11-16 SRVs are mounted depending on the rated power. The design capacity of an SRV is capable of maintaining reactor pressure below 110% of the vessel design pressure at the most severe pressurization transient[39]. In the following, brief descriptions are provided on the structure, operation mechanism, environment and surveillance requirements of the two-stage pilot actuated SRVs generally used at U.S. BWRs.

(1) Structure[36]

SRV consists of a main stage assembly and a pilot stage assembly which causes the main valve disc to move as shown in **Figure III.16**. Principle parts of the main stage are a bonnet, a main disc, a main piston, a main spring, and internal porting for the pilot stage. The pilot stage assembly is composed of a bonnet, a stabilizer disc, a pilot disc, a set pressure spring, and a pneumatic operator.

(2) Operation Mechanism[36]

When the pressure under the stabilizer disc reaches the setpoint of the valve, it will push the pilot disc upward, releasing the pressure behind the main piston into the discharge of SRV. The differential pressure across the piston will lift the main disc, relieving the RCS pressure. The ADS and the manual operation modes are initiated by the pneumatic actuator raising the pilot stem.

(3) Environment[36]

SRV operates in a saturated steam environment at a setpoint pressure which ranges from 7.548 MPa to 8.085 MPa. At the plant equipped with 11 SRVs, for example, these are divided into 3 groups (4 SRVs, 4 SRVs and 3 SRVs), for each of which the lift pressure is set to 7.617 MPa, 7.686 MPa or 7.754 MPa, respectively. The operating pressure at a BWR is about 7.272 MPa.

(4) Surveillance Requirements[39]

The surveillance requirements specify that any SRV lift at the setpoint pressure assumed in the safety analysis. The bench test, which is to be conducted in accordance with the in-service testing program, shall demonstrate that the safety mode lift setpoints of SRVs are maintained within their allowable tolerance during plant shutdown. The allowable tolerance for lift setpoints generally is ±3% for the operability but the valves are reset to ±1% during the surveillance to allow for drift. The surveillance test is to be carried out during plant shutdown and thus, the 18 or 24 month frequency is selected based on the time between refuelings.

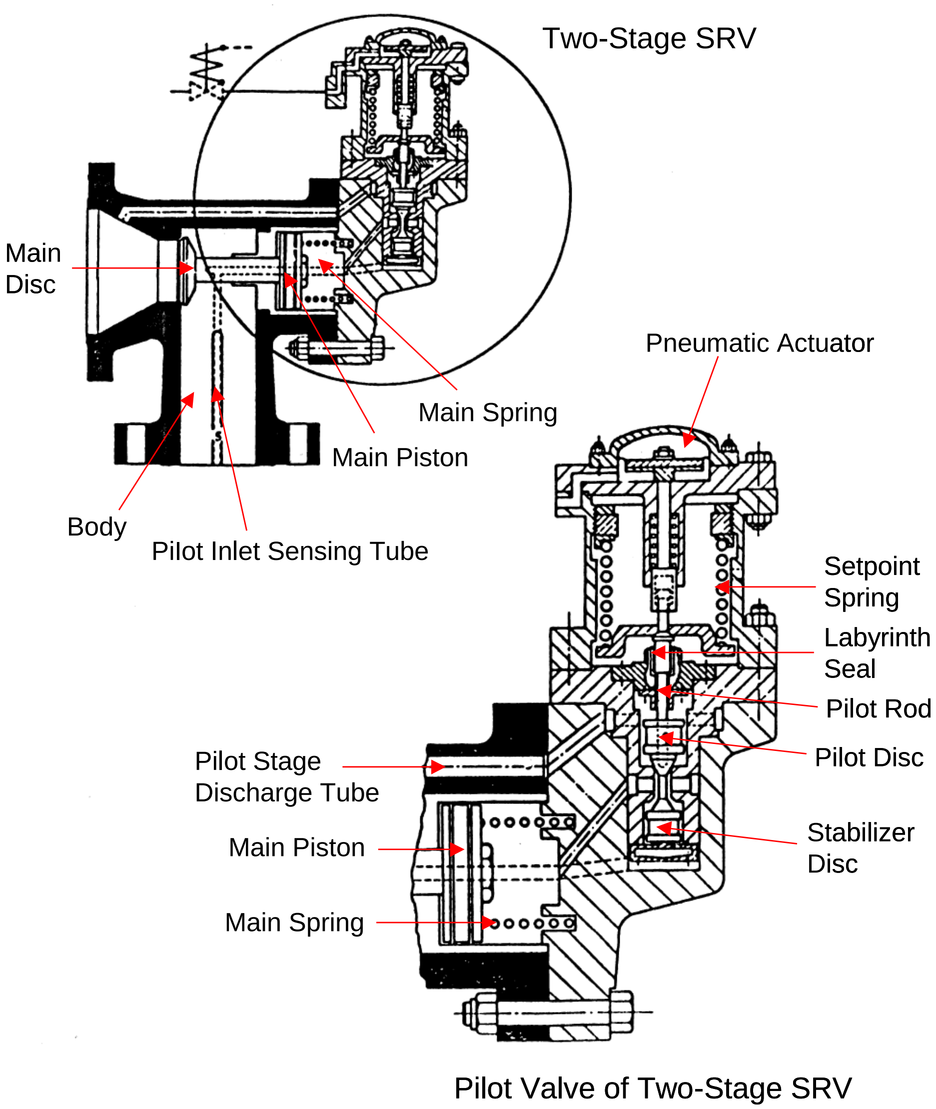

Figure III.16 Structure of Safety/Relief Valve (Two-Stage SRV) (36)

1.2 *Pressurizer Safety Valves and Main Steam Safety Valves at PWRs*

The number of PSVs is different depending on plant designers. In general, a Babcock and Wilcox (B&W) PWR is equipped with 2 PSVs, a Combustion Engineering (CE) PWR is with 2-4 PSVs and a Westinghouse (WH) PWR is with 3 PSVs(40-42). PSVs are designed to prevent the RCS pressure from exceeding 110% of the design pressure based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. The number of MSSVs is dependent on plant designers; in general, 9 at the B&W plant, 8 at the CE plant, and 5 at the WH plant(40-42). The capacity of MSSV also depends on plant designers. While at the B&W and CE PWRs, the rated capacity of MSSV passes the full steam flow at 112% and 102% rated thermal power with a valve fully opened, at WH PWRs, MSSVs are designed to have sufficient capacity to limit the secondary pressure to 110% or less of the steam generator design pressure(40-42). MSSVs provide overpressure protection for the secondary system and as well, play a role of protecting RCS against overpressurization by providing a heat sink for the removal of energy from RCS if the preferred heat sink, provided by the condenser and circulating water system, is not available. In the following, brief descriptions are provided on the structure, operation mechanism, environment and surveillance requirements of the spring-actuated safety valves generally used as PSVs and MSSVs at U.S. PWRs.

(1) Structure(36)

The spring-actuated safety valves consists of a body, a bonnet (or yoke), a spring, a spindle (or stem), and a disc as shown in **Figure III.17**. The spring is connected to the spindle by the setpoint nut and the spindle rests on the disc. The disc sits on the seat which is the upper surface of the nozzle welded or screwed into the body inlet flange. The disc and seat materials in PSV are usually both Stellite or stainless steel to Stellite and those in MSSV are both stainless steel. The disc-seat interface is flat and the pressure boundary

(2) Operation Mechanism(36)

The spring force is transmitted to the disc by the spindle and is equal to the system pressure at which the valve is expected to lift (i.e. the lift setpoint). When the pressure under the disc equals to the setpoint, the disc will begin to lift and the escaping steam assists the lift. The disc will quickly attain full lift at a pressure no greater than 3% above the setpoint and reseat when the system pressure has been reduced to 95% of the setpoint. The characteristics of valve lift and reseat, sharp or tentative, are determined by the guide ring and nozzle ring settings.

(3) Environment(36)

Both PSVs and MSSVs operate in a saturated steam at valve inlet. Operating pressure in the primary system is usually about 15.51 MPa and the setpoint pressure for PSVs is established at 17.23 MPa. Operating pressure in the secondary system is usually about 6.651 MPa and the MSSV setpoints vary from 6.996 MPa to 9.064 MPa.

(4) Surveillance Requirements(40-42)

The surveillance requirements for PSVs are specified in the in-service testing program, in which the frequency and contents of the testing are set forth. PSVs are generally set to open at the RCS design pressure (about 17.33 PMa) with an allowable tolerance of ±1%. At least one PSV is to be tested during every refueling outage and if the test results show that the allowable tolerance was exceeded, it is required that the scope be extended. Since PSVs are removed for testing, the 18 or 24 month frequency is selected. As well, MSSVs shall be tested to verify their respective lift setpoints in accordance with the in-service testing program. It is required that all valves be tested every 5 years and a minimum of 20% of the valves be tested every 24 months according to the ANSI/ASME (American National Standards Institution/American Society of Mechanical Engineers) standard. The allowable tolerance for MSSV setpoints generally is ±3% for the operability but the valves are reset to ±1% during the surveillance to allow for drift. While the test for MSSVs is to be carried out during power operation, the plant is allowed to be brought into a hot standby prior to the testing. MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If MSSVs are not tested at hot conditions, the lift pressure shall be corrected to ambient conditions of the valves at operating temperature and pressure.

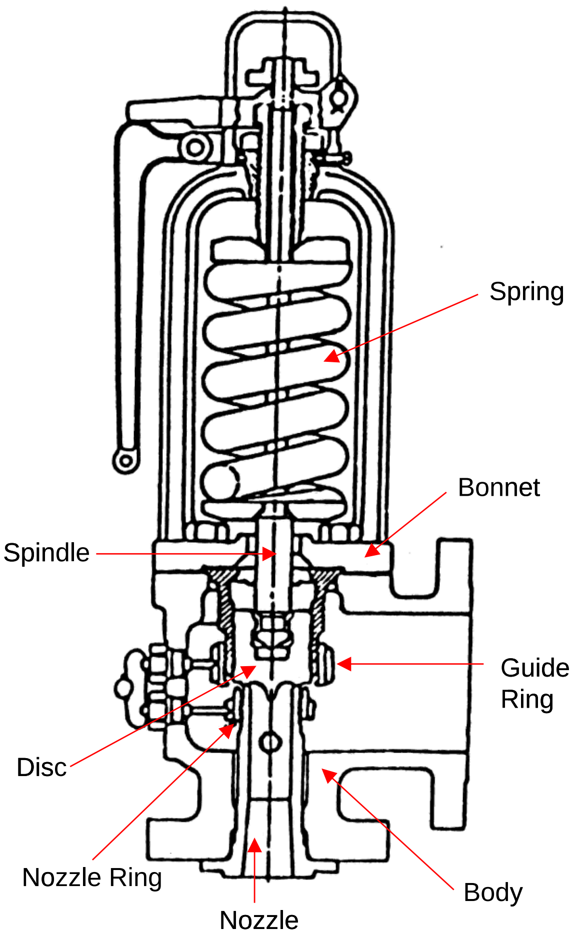

Figure III.17 Structure of Spring-Actuated Safety Valve[36]

2. Trending Analysis

2.1 Overall Trends Observed

This study reviewed 93 events involving the setpoint drift of SRVs, PSVs and MSSVs described in 87 LERs from 54 U.S. Plants during the years 2000 to 2006. **Figure III.18** shows the number of such events year by year. Except the year 2004, more than 10 LERs were submitted every year. While the setpoint drifts of PSVs were observed mainly during the first four years (2000 to 2003), those of SRVs occurred mainly during the last four years (2003 to 2006).

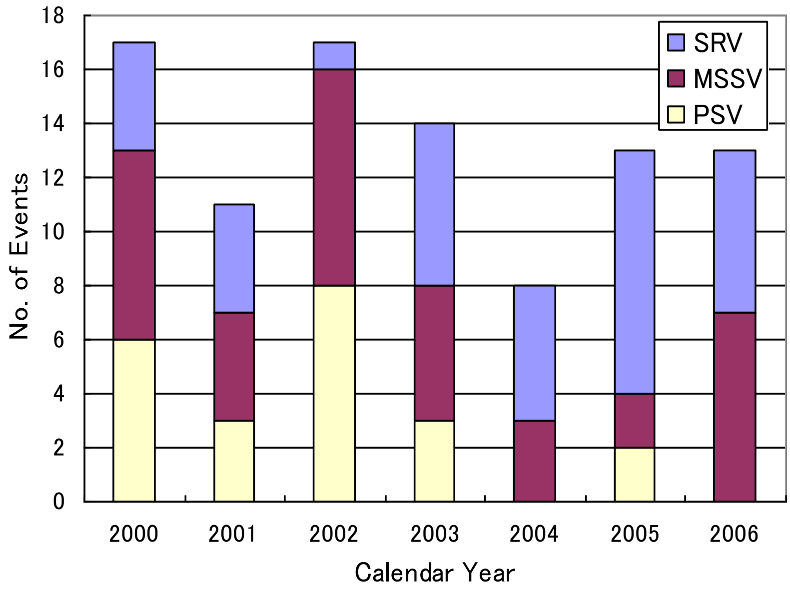

Figure III.18 Number of Setpoint Drift Events Year by Year

Thirty-five BWRs had been in operation during the 7-year period, 18 of which experienced the setpoint drift of SRVs and submitted a total of 31 LERs. At nine BWRs, the setpoint drift events recurred and in particular, two BWRs (Hope Creek and Pilgrim) experienced 5 events and 4 events, respectively. As well, the setpoint drift of PSVs or MSSVs was observed at 36 of 69 operating PWRs (22 of 48 WH plants, 8 of 14 CE plants and 6 of 7 B&W plants) and a total of 56 LERS were submitted (32 LERs at WH plants, 17 at CE plants and 7 at B&W plants). Three WH plants (Byron-1, Salem-1 and Turkey Point-3) experienced three events involving the setpoint drift of MSSVs and also, the PSV setpoint drift occurred at two of them (Byron-1 and Salem-1). The setpoint drift of MSSVs was observed at 6 CE plants, at 4 of which, such events occurred repeatedly. Particularly, the Palo Verde-2 submitted 4 LERs for years 2000 to 2002. Two CE plants (Palo Verde-2 and St Lucie-2) have an experience with the setpoint drift of both PSVs and MSSVs. As for the B&W plants, only one plant (Davis Besse) experienced the recurrence of MSSV setpoint drift.

Ninety-three LERs address the setpoint drift observed in a total of 277 valves, 189 of which involved the setpoint drift high and the rest involved the setpoint drift low. **Figure III.19** indicates the number of affected valves year by year. More than 17 valves experienced the setpoint drift high in every year and particularly, in 2000 and 2006, more than 30 valves were affected. For the setpoint drift low, more than 10 valves were affected in every year except for 2004 and 2006 and in particular, 25 valves experienced setpoint drift low in 2000.

In the following, the trends observed for SRVs, PSVs and MSSVs, respectively, focusing on either the setpoint drift high or low.

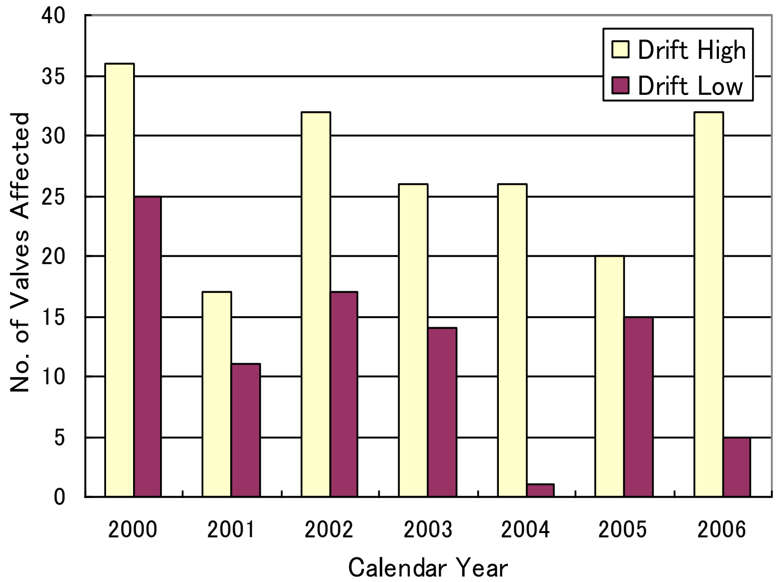

Figure III.19 Number of Valves Affected

2.2 *Safety/Relief Valves at BWRs*

Thirty-one LERs address thirty-five events involving the setpoint drift of SRVs. A total of 125 SRVs were affected. As seen from **Figure III.20**, more than 10 SRVs were affected in each year except 2002. In the years, 2000, 2003 and 2005, more than 20 SRVs experienced setpoit drift. In 2005, the numbers of such events and of the affected valves are larger than those in other years. Although the number of such events in the year 2000 or 2003 is comparable to that in 2001 or 2004, the number of the affected valves in 2000 or 2003 is twice as much as that in 2001 or 2004.

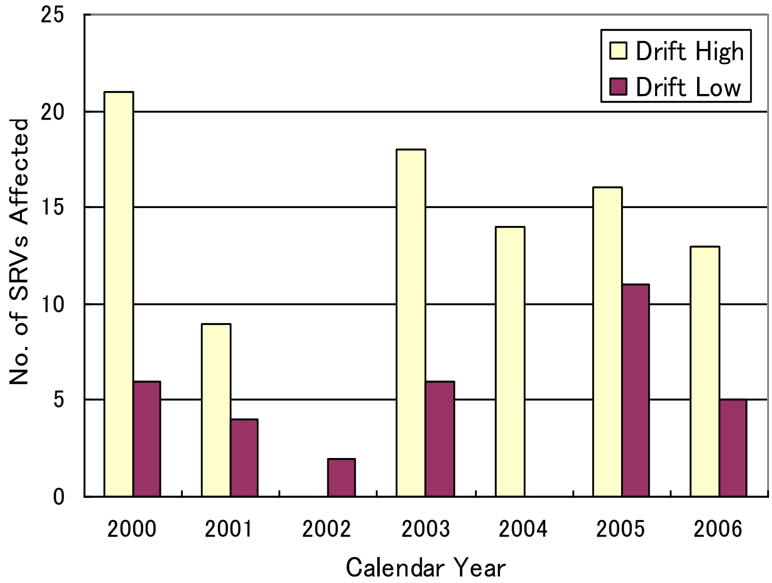

Figure III.20 Number of SRVs Affected

More than 70% of the affected SRVs (91 of 125 SRVs) experienced the setpoint drift high. As seen from Figure III.20, the number of the affected SRVs is more than 10 in every year except for 2001 and 2002 but a slightly decreasing trend is observed since 2000. Analyzing the causes of setpoint drift high, as shown in **Table III.7**, it is revealed that the setpoint drift high in 71 of 91 SRVs was due to disc-to-seat corrosion bonding in the pilot valve. The disc-to-seat corrosion bonding is thought to occur when oxygen in the steam corrodes the disc. More specifically speaking, the radiolytic breakdown of water and/or steam condensation may create the oxygen-rich environment in the pilot assembly, causing the oxygen to combine with the exposed internal metal surface and then forming the metal oxide film, that is, corrosion bonding, during normal operation. With corrosion bonding between the pilot disc and seat, more pressure is needed to lift the pilot disc off the seat and the lift pressure becomes higher. Actually, the resulting setpoint drift due to corrosion bonding ranged from several percent to greater than 10 percent. However, once the valve lifts, the corrosion bonding would be broken and the subsequent lifts would not or less be affected by the corrosion bonding. In many events, the SRVs that failed the initial lift test had successful lift tests within an allowable tolerance subsequently. This phenomenon had been recognized since the beginning of 1980s and the industry had made efforts such as the revision of maintenance procedures and the design modification of valve materials. Although the performance of SRVs had been improved as a result of actions which had taken by the industry until 2000, it can be said that any final resolution to eliminate this problem has not been attained because the corrosion bonding has occurred repeatedly after that. Other causes of setpoint drift high include erroneous settings of pressure switches (4 SRVs in 2 events), poorly controlled maintenance leading to friction on the sliding surface, excessive lapping of pilot disc or misalignment of internal parts of valves (7 SRVs in 4 events), and design deficiencies such as insufficient pressure adjusting spring tolerance specifications and lack of chamfer (2 SRVs in 2 events). The setpoint drift ranges are summarized in **Figure III.21** for the 79 affected SRVs, thirty and twenty-nine of which had the deviations of 3% to 5% and 5% to 10%, respectively. At some plants, the allowable tolerance has been specified as ±1% and thus, the setpoint drift ranges were 1% to 3% in 14 SRVs. Four of them involved the pressure switch problems. The deviations of greater than 10% were observed in 6 SRVs, all of which were caused by the corrosion bonding. The number of SRVs affected by the corrosion bonding was 6, 24, and 23 for the setpoint deviation of 1-3%, 3-5% and 5-10%, respectively, implying that the setpoint drift caused by the corrosion bonding tends to result in a higher deviation from the specified value.

The setpoint drift low was observed for 11 SRVs in 2005 but in the other years, it was for only a few SRVs (see Figure III.20). While the apparent causes of drift low were not identified for 20 of the 34 affected SRVs (8 of 14 events), one of the major causes was the seat leakage of pilot valve due to steam cutting. As seen from Table III.7, the setpoint drift low resulted from the steam cutting for 10 SRVs (4 events). The steam cutting can occur when system pressure reaches greater than 90% of set pressure, which typically occurs during a unit startup. At this point, system pressure is sufficient to allow the relief valve to simmer. If this simmering condition is allowed to exist for an extended period, the steam flow has the potential to erode small grooves into the seating surface. These small grooves create a permanent leak path by which steam continues to erode the disc and nozzle, leading to the seat leakage. As well, the steam cutting may be caused by foreign material lodged between the valve seat and disc, creating flaws on the seating surface and then leading to leakage. Many efforts had also been made to eliminate the seat leak problem but any final solution might have not been found. For the 20 SRVs with no cause being identified, the as-found setpoints were outside technical specifications acceptance criteria of ±1% but most of them (19 SRVs) were within the allowable tolerance of ±3% specified in the ASME standard. Other causes include the inherent characteristics of the valve hardware in conjunction with heating/cooling cycles and the vibration on the valves during service (2 SRVs in 1 event), sticking of air actuator stem preventing the pilot set spring from fully extending (1 SRV in 1 event), and a misalignment of internal parts (1 SRV in 1 event). The air actuator stem sticking was considered a unique case as there was no industry evidence of a previous occurrence. For 24 of the 34 affected SRVs, the drift ranged from -1% to -3% and such deviations are smaller compared with the drift high as shown in Figure III.21. However, in the case of drift low due to seat leakage, 7 of 10 SRVs experienced the drift low of -3% to -5%, indicating that the seat leakage may result in a greater deviation compared with the other cases.

Table III.7 Causes of SRV Setpoint Drift

Causes	No. of Valves (Events)
- SRVs: Drift High -	
Pilot Disc-Seat Corrosion Bonding	71 (20)
Maintenance Problems (ex. misalignment of internal components):	7 (4)
Design or Specification Problems (ex. lack of chamfer)	2 (2)
Pilot Seat Leakage	2 (1)
Error in Changing Pressure Switch Settings	4 (2)
Not Specified	5 (4)
- SRVs: Drift Low -	
Pilot Seat Leakage (ex. due to Steam Cutting)	10 (4)
Heating/Cooling Cycle & Vibration	2 (1)
Misalignment of Internal Components	1 (1)
Air Operator Stem Sticking	1 (1)
Not Specified	20 (8)

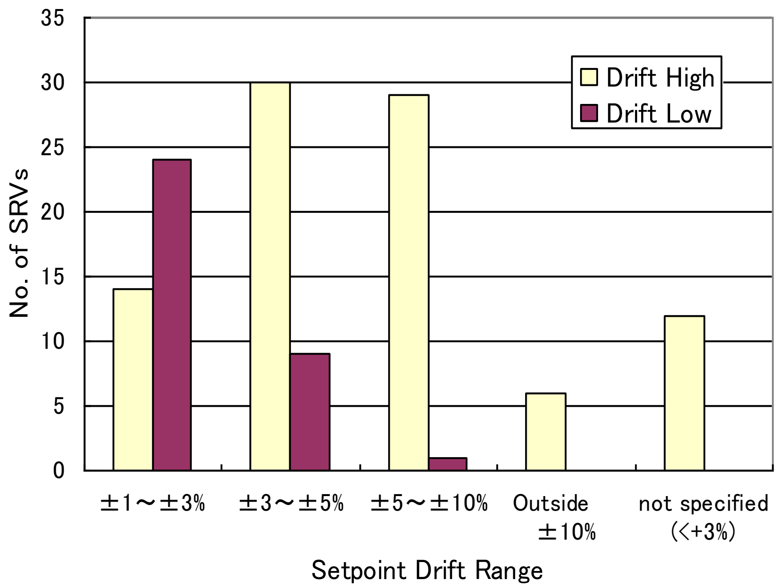

Figure III.21 Number of SRVs by Setpoint Drift Ranges

2.3 Pressurizer Safety Valves at PWRs

A total of 45 PSVs have an experience with setpoint drift, which were reported in 22 LERs. Considering that only 2 to 4 PSVs are mounted on a reactor while 11 to 16 SRVs and 15 to 20 MSSVs are on a reactor, it can be said that the number of the affected PSVs is comparable to that for SRVs or MSSVs. As shown in **Figure III.22**, the setpoint drift low was observed in 12 PSVs (6 events) and 16 PSVs (8 events) in 2000 and 2002, respectively, corresponding to approximately 60% of the total number of occurrences. In addition, during the last three years, only two events were reported in 2005, both of which describe the setpoint drift low observed in four PSVs. This implies that the setpoint drift of PSVs has a decreasing trend.

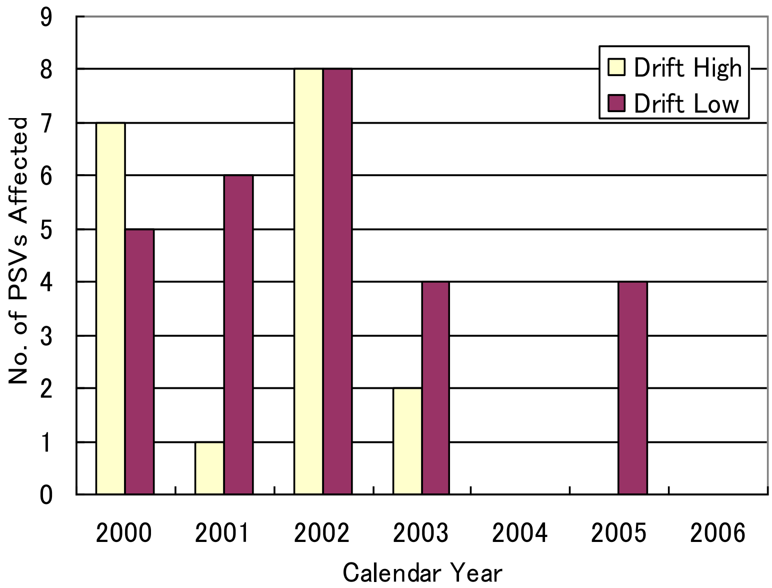

Figure III.22 Number of PSVs Affected

Twenty-seven of the 45 affected PSVs (14 of 22 events) involved the setpoint drift low and the remaining 18 PSVs (15 events) involved the drift high. **Figure III.23** shows the number of PSVs by setpoint drift ranges. As seen from this figure, the deviation range of ±1% to ±2% was observed in more than half of the affected PSVs (drift high in 9 PSVs and drift low in 15 PSVs). The 11 PSVs experienced the deviation of greater than ±3% (drift high in 5 PSVs and low in 6 PSVs) but the greatest deviation was +3.8% for drift high and -4.4% for drift low. Many LERs did not identify a root cause of setpoint drift, some of which defined the random spread as a root cause and the others provided no information on the root cause. As seen from **Table III.8**, apparent root causes of setpoint drift were provided for only 7 of the 45 affected PSVs (4 of 27 PSVs with setpoint drift low and 3 of 18 PSVs with setpoint drift high). The apparent causes of drift low were inadequate lapping of internal parts and the nozzle loading effects (the variation in thermal growth of the valve inlet and outlet piping) inducing stresses and misalignments in the valve (1 PSV in 1 event, 3 PSVs in 1 event, respectively). The setpoint drift high was caused by the spring performance problems such as degradation due to aging and insufficient contact area between spring and washer. However, the deviation range observed in these events was relatively large, exceeding 3% for the drift high (+3.8, +3.7, +3.66) and -2.5% for the drift low (-3.5, -3.18, -2.98, -2.53). Many plants specify an allowable tolerance of ±1% in the technical specifications but some plants specify the value of ±2% or ±3%. As well, the ASME standard defines the tolerance of ±3% for the PSV lift setpoint. Therefore, some plants will apply for the change of the technical specifications allowable tolerance from ±1% to ±3%.

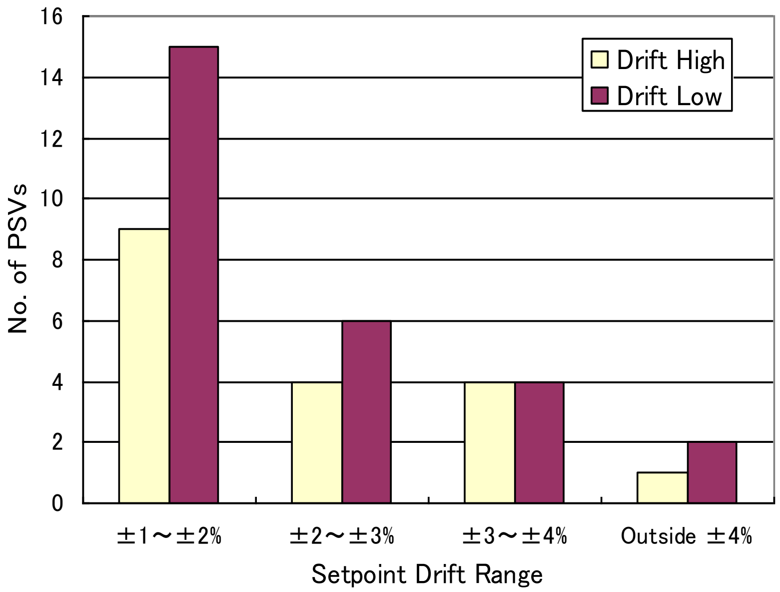

Figure III.23 Number of PSVs by Setpoint Drift Ranges

Table III.8 Causes of PSV Setpoint Drift

Causes	No. of Valves (Events)
- PSVs: Drift High -	
Random Spread	2 (2)
Spring Performance Problems (ex. aging)	3 (3)
Not Specified	13 (10)
- PSVs: Drift Low -	
Random Spread	7 (4)
Nozzle Loading Effects	3 (1)
Manufacturing Problem (Inadequate Lapping)	1 (1)
Not Specified	16 (9)

2.4 Main Steam Safety Valves at PWRs

Thirty-six events involved the setpoint drift of totally 107 MSSVs. **Figure III.24** shows the number of the MSSVs with setpoint drift high and low year by year. The number of the affected MSSVs varies significantly for individual years. While, particularly, 31 MSSVs were affected in 2002 (8 events), there were only 4 MSSVs with setpoint drift in 2005 (2 events).

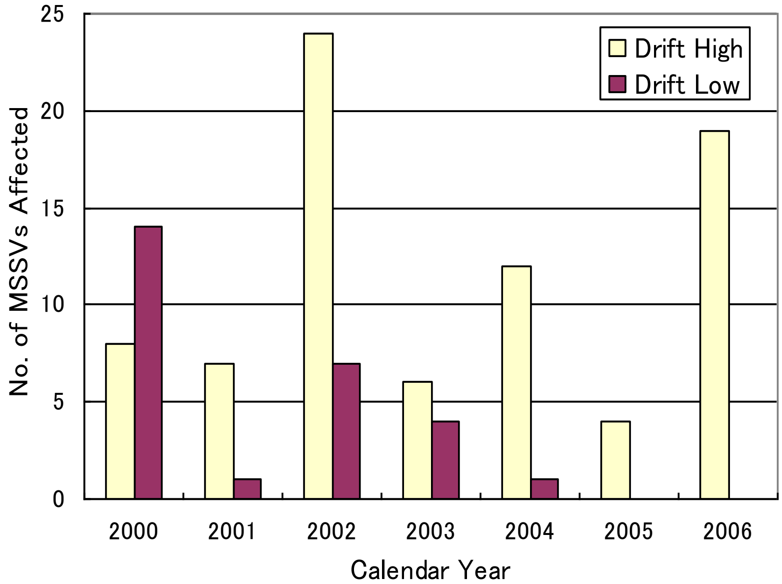

Figure III.24 Number of MSSVs Affected

Eighty of the 107 affected MSSVs (27 of 36 events) have an experience with setpoint drift high. As seen from Figure III.24, there were more than half of the affected MSSVs in two years, 2002 and 2004 (24 MSSVs and 19 MSSVs, respectively) but only several MSSVs were affected in for years, 2000, 2001, 2003, and 2005. The root causes of setpoint drift high were identified for the 70 affected MSSVs (see **Table III.9**). The dominant contributor was the disc/nozzle-seat oxide bonding, which was observed for the 54 MSSVs. The oxide bonding is defined as the physical surface bonding of the disc seating surface and the nozzle or the disc and seating surfaces. During power operations, an aqueous, somewhat alkaline, condition exists between the disc seating surface and the nozzle and promotes buildup of oxide films which over time grow together and form a mechanical bonding. As well, the fusion of the martensitic stainless steel disc to the austenitic stainless steel seat occurs by an oxide film and results in a bonding mechanism. Once any oxide bonds are broken, the valve seat and nozzle do not tend to bond again as long as the oxide layers remain in-place, regardless of the valve materials. This phenomenon had been observed at many plants since the mid-1990s. Hence, the disc material had been changed from stainless steel to Inconel X-750 and a pre-oxidation treatment of the disc/seat had been conducted prior to re-assembly in order to avoid the oxide bonding. These efforts were considered to have mitigated the bonding conditions but did not lead to the final solution. In addition, another bonding mechanism has been recognized as a cause of setpoint drift high. Micro-bonding of nozzle and disc can occur when the harder disc causes microscopic galling of the softer nozzle during heatup, because of the differential thermal expansion between the contact surfaces. These small gall beads cause the disc and nozzle to fuse to a limited degree. This phenomenon was determined for five MSSVs in three events (two in 2003 and one in 2006), all of which occurred at Turkey Point. Although the licensee lapped the valve seats and changed the seat and disc surface finish as corrective actions to address the micro-bonding after the 2002 events, the micro-bonding recurred in 2006. This indicates that the previous corrective actions was helpful in improving resistance to galling but did not eliminate the micro-bonding. As a result, Turley Point replaced all the MSSV stainless steel discs with Inconel discs. The other causes include a buildup of corrosion between the spring and spring washer increasing the compression of the spring (2 MSSVs in 2 events), the aging of valve internals (2 MSSVs in 2 events), mechanical component failure due to the spindle binding (4 MSSVs in 1 event), and inherent test methodology inaccuracies (1 MSSV in 1 event).

Table III.9 Causes of MSSV Setpoint Drift

Causes	No. of Valves (Events)
- MSSVs: Drift High -	
Disc-Nozzle Seat Oxide Bonding	54 (15)
Disc-Nozzle Seat Bonding due to Differential Thermal Expansion	5 (3)
Ageing, Damage or Corrosion of Internal Components	8 (5)
Design Problem: Incorrect Dimension of Spring Washer	1 (1)
Misalignment of Internal Components	1 (1)
Incorrect Test Method	1 (1)
Not Specified	10 (6)
- MSSVs: Drift Low -	
Spring Problems (Relaxation, Insufficient Contact of Spring Caps)	6 (2)
Seat Leakage due to Steam Cutting	3 (2)
Differences in Results by Using Two Test Methods/Devices	5 (2)
Not Specified	13 (4)

The setpoint drift ranges were specified for the 76 affected MSSVs. As shown in **Figure III.25**, the deviations of 3% to 5% were observed for the 46 MSSVs. The 19 MSSVs exhibited the deviations of greater than 5%, for 16 of which the disc bonding was determined to be a cause of drift high (14 MSSVs: oxide bonding, 2 MSSVs: micro-bonding). As a small number of plants specified the allowable tolerance of ±1% for the MSSV setpoints, the drift ranged from 1% to 3% for the 11 MSSVs, 4 of which involved the oxide bonding.

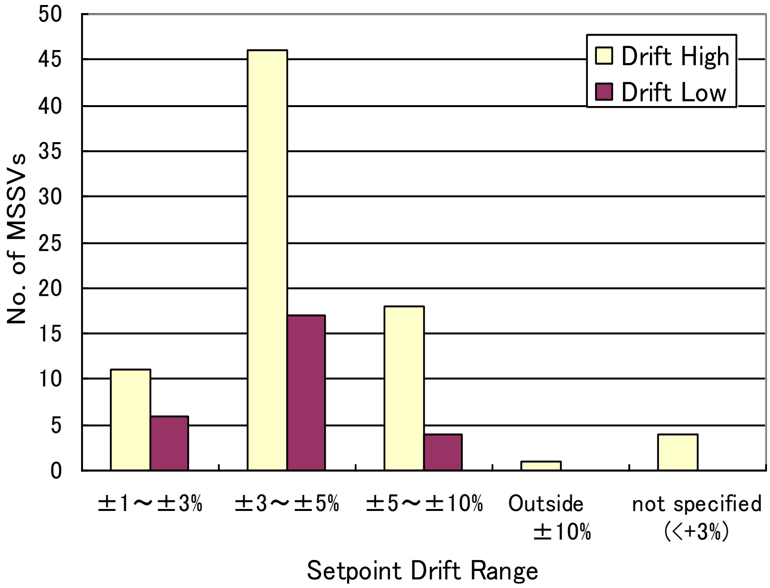

Figure III.25 Number of MSSVs by Setpoint Drift Ranges

The setpoint drift low was observed for 14 MSSVs in 2000 but after that, a decreasing trend was indicated; one in 2001, seven in 2002, four in 2003, one in 2004, and zero in 2005 and 2006, as seen from Figure III.24. For the 14 MSSVs with setpoint drift low, the root causes identified can be classified into three groups. One was associated with the spring problems such as relaxation (5 MSSVs in 1 event) and insufficient contact of spring caps (1 MSSV in 1 event). Another was the different test methodologies applied in the setpoint verification in situ and at the vendor facility (5 MSSVs in 2 events). The other was excessive seat leakage due to steam cutting (3 MSSVs in 2 events, both of which occurred at Salem). As seen from Figure III.25, the setpoint drift ranged from -1% to -3% for 6 MSSVs, from -3% to -5% for 17 MSSVs and from -5% to -10% for 4 MSSVs. Three MSSVs with excessive seat leakage exhibited a relatively slight deviation of -1.2 to -1.3% and were identified at Salem, where the allowable tolerance was specified as ±1% of nominal setpoint in technical specifications. Thus, this plant applied for the change of allowable tolerance to ±3%. The MSSVs with deviations of -3% to -5% include 5 MSSVs with drift low resulting from the different test methodologies and 4 MSSVs associated with the spring problems. For the other 8 MSSVs, the cause of drift low was unknown. Two of the 4 MSSVs with a deviation of

-5% to -10% were stemming from the spring relaxation.

3. Summary

The U. S. nuclear power plants have experienced the setpoint drift of SRVs, PSVs and MSSVs for many years and thus, the USNRC has issued a lot of generic communications to alert the licensees of operating experience involving setpoint drift and insights on this matter. The nuclear industry has also conducted the root cause analyses and implemented corrective actions such as disc material change to avoid this problem. Nevertheless, the setpoint drift has been observed continuously, implying that the definite solution has not been achieved.

SRVs have a tendency of the setpoint drift high due to corrosion bonding of pilot disc and seat and the deviation is relatively large and may exceed 10% of the nominal setpoint, resulting in a potential safety significant issue for overpressure protection. On the other hand, in the case of drift low due to seat leakage, the deviation of setpoint ranges from -3% to -5% and in the cases with cause unknown, the as-found setpoints are outside technical specifications acceptance criteria but are within the allowable tolerance specified in the ASME standard. The deviations due to the drift low are smaller compared with those due to the drift high.

While in many events involving setpoint drift of PSVs, the root causes are unknown, the deviation ranges from 1% to 3%. On the other hand, the drift high due to the spring problem indicates a tendency of relatively large deviations exceeding 3% although there are a small number of affected valves. Many plants specify an allowable tolerance of ±1% in the technical specifications, resulting in a lot of reportable events. The root causes have not been identified for such small deviations in many cases and it is considered difficult to keep the setpoint within a narrow range. Furthermore, the ASME standard defines the tolerance of ±3% for the PSV lift setpoint. Therefore, some plants attempted to apply for the change of the technical specifications allowable tolerance.

The setpoint drift high of MSSVs due to disc/nozzle-seat oxide bonding tends to result in a relatively large deviation with some cases exceeding 10% and may challenge overpressure protection. To cope with the oxide bonding, valve materials had been changed but any final resolution has not been reached. The setpoint drift low indicates a decreasing trend and the deviation range are relatively small. Since many event reports had not identified the root causes of drift low, no apparent trend on causes is observed. However, there is one case with its cause unknown involving a relatively large deviation.

This implies the safety significance of the setpoint drift low with a large deviation because it may lead to premature or spurious lift.

III.5 Concluding Remarks

This chapter discusses three topical studies on loss of decay heat removal during reactor shutdown at PWRs, criticality accidents at nuclear fuel processing facilities, and setpoint drift in safety or safety/relief valves at U.S. LWRs.

Between 1976 and 1990, 197 loss of DHR events were reported to have occurred at U. S. PWRs and one-third (63 events) occurred in reduced inventory conditions. Analysis of these events indicates that four-fifths (49 events) of DHR losses in reduced inventory conditions were caused by air entrainment into the DHR pumps due to lowering the RCS water level too far, loss of coolant inventory, increased DHR pump flow, and so on. In particular, the lowering RCS water level too far resulted from mainly inaccurate level indications, highlighting the need to improve the reliability of water level instruments. In many events involving air entrainment, the DHR loss lasted for more than 1 h and the remarkable coolant heatup was observed. The coolant heatup rates and the times to bulk boiling in the core were estimated for 12 events with use of the data obtained from their respective event reports. As well, the time to core uncovery was estimated. The calculated heatup rates are in reasonably good agreement with the observed ones and it is shown that bulk boiling and core uncovery would take place within 1 h and several hours respectively even if the DHR loss would occur in the late stages of shutdown (for example, 30 d after the reactor trip). This implies that the recovery actions need to be prepared in advance against loss of DHR pumps.

Twenty-one criticality accidents which occurred in foreign countries prior to the JCO accident were analyzed to examine the overall trends and to identify common issues on the sequences and causes of accidents in terms of similarities to the JCO accident. Almost all of them occurred when handling uranium or plutonium solution in vessels/tanks with unfavorable geometry. In some cases the problems similar to those observed in the JCO accident were identified: violations of procedures and/or technical specifications for improving work efficiency and/or productivity, procedural changes without any application to and permission from the regulatory body, lack of understanding of criticality hazards, and complacency that a criticality accident would not occur. In addition, particular attention should be paid to the fact that most of accidents occurred in 1950s to 1960s which might have contributed to the JCO accident. This study underscores the importance of making the lessons learned from the past similar events permeate throughout the workers.

Since the beginning of 1980's, in the United States, there have been many events involving setpoint drift in safety or safety/relief valves. The U.S. experience with setpoint drift in SRVs at BWRs, PSVs and MSSVs at PWRs was analyzed by reviewing approximately 90 LERs from 2000 to 2006 and the trend was examined focusing on causes and setpoint deviation ranges. The results show that for SRVs and MSSVs, disc-seat bonding is a dominant cause of the setpoint drift high and has a tendency to result in a relatively large deviation of the setpoint. This means that disc-seat bonding might be a safety concern from the viewpoint of overpressure protection. To cope with the oxide bonding, valve materials had been changed but any final resolution has not been reached. For PSVs, the deviation of setpoint is generally small, although its causes are not specified in many events. However, the drift high due to the spring problem indicates a tendency of relatively large deviations. This study points out the importance of sharing the insights obtained from the analysis of events involving setpoint drift from the viewpoint of overpressure protection and pressure boundary integrity.

These topical studies identify the common issues such as the same/similar causes applicable to a lot of events and indicate that the lessons learned from past events or accidents have not sufficiently been employed and/or effective measures have not taken. If previous similar events had been recognized throughout the nuclear community, their recurrence might have been avoided. Sharing and exchange of information on operating experience should be enhanced on national and international basis and in particular, the lessons and insights in common should be disseminated into the nuclear community. Also, the licensees or operating organizations should improve their surveillance or testing programs, perform root cause analysis and then, implement effective corrective actions, in particular paying more attention to past events which have the potential risks. The topical studies described here indicate that it is essential to perform such studies focusing on the recurring events in order to derive the generic safety implications.

References in Chapter III

(1) N. Watanabe, M. Hirano and T. Oikawa, *Analysis of Operating Experience data in Nuclear Power Plants – Loss of Decay Heat Removal during Reactor Shutdown*, JAERI-M 91-143, 1991 (in Japanese).

(2) N. Watanabe and M. Hirano, *Analysis of Operating Experience Involving Loss of Decay Heat Removal during Reactor Shutdown in Pressurized Water Reactors*, Journal of Nuclear Science and Technology, Vol.29, No.12, pp.1212-1223, 1992.

(3) N. Watanabe, *Compilation and Review of Loss of Decay Heat Removal during Reactor Shutdown in PWRs*, JAERI-M 94-076, 1994 (in Japanese).

(4) N. Watanabe, *Review of Recent Events and Regulatory Guidances Concerning Emergency Core Cooling System Strainer Blockage at Boiling Water Reactors in the U. S. A.*, JAERI-Review 97-008, 1997 (in Japanese).

(5) N. Watanabe, *Analysis of Causes of Criticality Accidents at Nuclear Fuel Processing Facilities in Foreign Countries – Similarities to the Criticality Accident at JCO's Uranium Processing Plant*, Journal of Atomic Energy Society of Japan Vol. 42, No.11, pp.94-102, 2000 (in Japanese).

(6) N. Watanabe and H. Tamaki, *Review and Compilation of Criticality Accidents in Nuclear Fuel Processing Facilities Outside of Japan*, JAERI-Review 2000-006, 2000, (in Japanese).

(7) N. Watanabe, *Review of Incident Involving Reactor Pressure Vessel Head Degradation at U. S. PWR*, JAERI-Review 2004-015, 2004 (in Japanese).

(8) S. Takahara and N. Watanabe, *Trending Analysis of Incidents Involving Primary Water Stress Corrosion Cracking on Alloy 600 Components at U. S. PWRs*, Transaction of Atomic Energy Society of Japan, Vol.5, No.4, pp.282-291, 2006 (in Japanese).

(9) N. Watanabe and S. Takahara, *Review of Recent Events Involving Degradation of Reactor Coolant System Pressure Boundary due to Primary Water Stress Corrosion Cracking at U. S. PWRs*, JAEA-Review 2006-027, 2006 (in Japanese).

(10) N. Watanabe, *Review of Recent Events Involving Setpoint Drift in Safety or Safety/Relief Valves at U. S. Nuclear Power Plants*, JAEA-Review 2007-043, 2007 (in Japanese).

(11) N. Watanabe, *Trend Analysis of Incidents Involving Setpoint Drift in Safety or Safety/Relief Valves at U. S. LWRs*, Transaction of Atomic Energy Society of Japan, Vol.7, No.1, pp.74-84, 2008 (in Japanese).

(12) N. Watanabe, *Fire Events at Nuclear Power Plants*, Journal of Japan Association

for Fire Science and Engineering, ISSN 0449-9042, Vol.58, No.2, pp.11-16, 2008 (in Japanese).

(13) EPRI, *Residual Heat Removal Experience Review and Safety Analysis – Pressurized Water Reactors*, NSAC-52, 1983.

(14) USNRC, *Decay Heat Removal Problems at U. S. Pressurized Water Reactors*, AEOD/C503, 1985.

(15) USNRC, *Loss of Decay Heat Removal due to Loss of Fluid Level in Reactor Coolant System*, Information Notice 86-101, 1986.

(16) USNRC, *Loss of Decay Heat Removal during Low Reactor Coolant Level Operation*, Information Notice 87-23, 1987.

(17) USNRC, *Loss of Decay Heat Removal while the Reactor Coolant System is Partially Filled*, Generic Letter 87-12, 1987.

(18) USNRC, *Loss of Decay Heat Removal*, Generic Letter 88-17, 1988.

(19) USNRC, *Loss of Vital AC Power and the RHR System during Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990*, NUREG-1410, 1990.

(20) EPRI, *Zion Nuclear Power Plant Residual Heat Removal PRA*, NSAC-84, 1985.

(21) EdF, *Probabilistic Safety Assessment of Reactor Unit 3 in the Paluel Nuclear Power Centre (1,300 MWe)*, 1990.

(22) CEA/IPSN, *A Probabilistic Safety Assessment of the Standard French 900 MWe Pressurized Water Reactor*, 1990.

(23) USNRC, *Loss of Residual Heat Removal System, Diablo Canyon Unit 2, April 10, 1987*, NUREG-1269, 1987.

(24) ANS, *ANS/ANSI 5.l Standard*, 1979.

(25) Nuclear Safety Commission of Japan/Investigation Committee of Uranium Processing Plant Accident, *The Uranium Processing Plant Criticality Accident Investigation Committee Report*, 1999 (in Japanese).

(26) W. R. Stratton, *A Review of Criticality Accidents*, LA-3611, 1967.

(27) W. R. Stratton (revised by D. R. Smith), *A Review of Criticality Accidents*, DOE/NCT-04, 1989.

(28) IAEA, *Significant Incidents in Nuclear Fuel Cycle Facilities*, IAEA-TECDOC-867, 1996.

(29) T. Ohnishi, et al., *Nuclear Accidents in Nuclear Installations (Survey Report)*, JAERI 4052, 1970 (in Japanese).

(30) S. Tachimori and S. Sakurai, *Review of the Criticality Accidents in Nuclear Fuel Facilities*, JAERI-M 84-155, 1985 (in Japanese).

(31) V. V. Frolov, et al., *A Review of Criticality Accidents Which Occurred in the Russian Industry*, Proc. of ICNC-95, 1995.

(32) G. T. Kirillov, *The Criticality Accidents Which Occurred at the Siberian Chemical Combine – Their Reasons and Lessons*, Proc. of ICNC-99, 1999.
(33) A. Romanov, *The Nuclear Criticality Accident That Took Place on 03.11.1965 at "MASHINOSTROITELNY ZAVOD"; Electrostal, Russia*, Proc. of ICNC-99, 1999.
(34) A. G. Ustyugov, *Criticality Accident at the Novosibirsk Chemical Concentration Plant, 15 May, 1997*, Proc. of ICNC-99, 1999.
(35) Science and Technology Agency/Accident Investigation Task Force, *Status of Radiation Exposures due to Criticality Accident at JCO Tokai Site and Future Efforts*, Reported to the Nuclear Safety Commission of Japan, 2000 (in Japanese).
(36) USNRC, *Special Study: Safety and Relief Valve Reliability*, AEOD/S92-02, 1992.
(37) South Texas Project Unit 1, *Pressurizer Safety Valves Setpoints Discovered Outside Required Tolerances*, LER No. 50-498/2001-002-00, 2001.
(38) USNRC, *Recent Operating Experience Associated with Pressurizer and Main Steam Safety/Relief Valve Lift Setpoints*, Information Notice 2006-24, 2006.
(39) USNRC, *Standard Technical Specifications: General Electric Plants BWR/4*, NUREG-1433, Vols.1&2, Rev.3.0, 2004.
(40) USNRC, *Standard Technical Specifications: Babcock and Wilcox Plants*, NUREG-1430, Vols.1&2, Rev.3.0, 2004.
(41) USNRC, *Standard Technical Specifications: Combustion Engineering Plants*, NUREG-1432, Vols.1&2, Rev.3.0, 2004.
(42) USNRC, *Standard Technical Specifications: Westinghouse Plants*, NUREG-1431, Vols.1&2, Rev.3.0, 2004.

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Chapter IV

Accident Sequence Precursor Studies

IV.1 Background

The Accident Sequence Precursor (ASP) Program was established by the United States Nuclear Regulatory Commission (USNRC) in 1979 to provide a structured, probabilistic method of analyzing operational events at light water reactors (LWRs) to determine and assess both known and unrecognized vulnerabilities that could lead to core damage(1). The ASP Program systematically evaluates operating experience at LWRs, using the probabilistic risk/safety assessment (PRA/PSA) technique, to identify, document and rank the operational events most likely to lead to inadequate core cooling and severe core damage, that is, precursors if additional failures had occurred. Therefore, the primary objective of the ASP Program is to identify precursors to potential severe core damage accident sequences at nuclear power plants in terms of the core damage risks. The secondary objectives of the ASP Program are to categorize the precursors by their plant-specific and generic implications, to provide a measure for trending risk at nuclear power plants and to provide a partial check on PSA-predicted dominant core damage scenarios. The significance of a precursor is evaluated by calculating the probability of postulated core damage accident sequences given the failed equipment associated with the particular event, applying the event trees and fault trees generally used in PSA. This probability is called a conditional core damage probability (CCDP). The events with a CCDP of 1×10^{-6} or higher are identified as precursors and in particular, those with a CCDP of 1×10^{-4} or higher are considered significant in the ASP Program(2).

So far, the ASP analysis methods and results were applied to draw generic implications related to nuclear power plant risks by examining the risk significant trends, to characterize risk insights useful for identifying plant vulnerabilities, to feed the lessons learned from the study back to plant operations, and to establish risk indicators for examining the overall trends of precursors. For example, the ASP analyses were carried out for a specific event, that is, steam generator tube rupture (SGTR), with use of the consistent ASP model to identify the risk significant anomalies found during the events and to obtain generic insights useful for enhancing the safety of nuclear power plant operations for SGTR(3). Also, the risk information was derived by reviewing the ASP documents published by the USNRC, aiming at obtaining risk significant trends and characterizing risk-related insights(4-7) and the development of event tree models for providing a more realistic ASP analyses(8).

In this chapter, the ASP analysis of SGTR events(3) and the trending analysis of precursors with newly proposed risk indicators(7) are described.

IV.2 What is Precursor Study

The ASP Program has used an event tree analysis method since the program was initiated. However, the ASP models had been improved several times to reflect the state of the art in PSA techniques and/or plant design changes(9-26). Therefore, different models had been used in the event analyses over the years and may have affected the analysis results. Firstly described are the major changes in the models as background information for better understanding of discussions.

1. Evolution in of Accident Sequence Precursor Models

In the ASP Program, the first phase analysis for 1969-81 events used simplified, standardized event trees to model potential core damage sequences. One set of event trees was used for pressurized water reactors (PWRs) and a separate set was used for boiling water reactors (BWRs). These event trees were developed for four initiating events: loss of feedwater (LOFW), loss of offsite power (LOOP), small break loss of coolant accident (LOCA), and main steam line break (MSLB). They were functional event trees, where only mitigation systems were functionally represented. Success criteria for the mitigation systems varied from plant to plant but these variations were addressed in the probabilities assigned to branches in the event trees.

In 1985, the event tree models were improved to reflect the design differences among the U. S. nuclear power plants. This improvement incorporated eight plant classes, five for

PWRs and three for BWRs. Systemic event trees were reconstructed for three initiating events - a nonspecific reactor trip (TRIP) including LOFW, LOOP and small break LOCA - for each plant class. These systemic event trees were employed in the analysis of 1984-87 events. In these models, in addition to two ordinary end states, “success” and “core damage”, another end state, “core vulnerability” was assigned to sequences in which core protection was expected to be provided, but for which no specific analytical basis was available or which involved non-proceduralized operator actions.

In 1989, two modeling changes were made to the event trees: reassignment of core vulnerability sequences as success or core damage, and explicit representations of electric power recovery and PWR seal LOCA. Other two changes were also made to simplify the BWR event trees: elimination of the top event for standby liquid control (SLC), resulting in assignment of reactor trip failure to the anticipated transient without scram (ATWS) end state, and consideration of the condensate system as a recovery action instead of low pressure systems. Although the revised event tree models were used in the analysis of 1988-93 events, supplemental and plant-specific mitigation systems and/or procedures beyond those included in the basic models were considered in the analysis of 1992-93 events. These models were also employed in the analysis of 1982-83 events.

In 1995, the ASP models were significantly changed by applying the fault tree linking technique to the modeling of safety-related systems on the event trees. In addition, an event tree for SGTR was developed for each PWR plant class. The event trees for the other initiating events were basically the same as those used in the 1988-93 analyses. After that, the fault trees were updated based on the plant design and procedural changes, and has been used in the ASP analysis. The integrated models of fault trees and event trees are called the standardized plant analysis risk (SPAR) models which were prepared for 72 plant types[27].

2. Analytical Approach

As mentioned above, the ASP analysis had been carried out with use of event tree models for several initiating events before 1995. On the other hand, the fault tree linking approach has been used since 1995. In the following, these two analytical approaches are described through the anaysis of an actual event[28].

2.1 Description of Event to Be Analyzed

The event to be analyzed took place at South Texas Project Unit 1 (STP-1) in Janualy 1993, the ASP analysis of which was provided in Ref. (20). In this event, the unit had operated with one of three emergency diesel generators (EDGs) inoperable for approximately 25 d, during 61 h of which a second EDG was removed from service for maintenance. As well, the turbine-driven auxiliary feedwater (TDAFW) pump was inoperable for about 40 d, which encompassed the time period when EDGs were inoperable. Consequently, only one EDG remained available and three trains of motor-driven auxiliary feedwater system (MDAFW) were operable.

2.2 *Analysis with Event Tree Model*

In the ASP analysis perforemed by the USNRC[20], the event was modeled as a potential LOOP for approximately 25 d (a total of 597 h). It was assumed that one EDG and the TDAFW pump were inoperable during this period and a second EDG was inoperable for 61 h of this period. The model was revised to reflect this condition by disabling one or two trains of equipment dependent on emergency power. However, the recovery probability for the AFW was not changed because the failures related to the TDAFW pump were considered recoverable regardless of power availability. For this event, the USNRC's ASP analysis defined two cases: one was analyzed as a LOOP with one EDG and the TDAFW pump inoperable for 536 h (597–61 h) and the other was as a LOOP with two EDGs and the TDAFW pump inoperable for 61 h. In addition, each of the two cases was further decomposed considering that the battery lifetime may be extended by shedding unnecessary loads (cases with and without battery load shedding). Eventually, a total of four cases were calculated in the ASP analysis for this event.

Figure IV.1 shows the event tree used in the USNRC's ASP analysis and **Table IV.1** lists the branch probabilities used for the calculation of conditional core damage probabilities. As seen from this table, it was needed to change the branch probabilities for the initiating event (LOOP) and the safety systems such as electrical power supply (EP), auxiliary feedwater system (AFW), high pressure injection system (HPI) and high pressure recirculation system (HPR), which were inoperable during the event. For example, the occurrence probability of initiating event, $P(IE)$, was obtained by the following equation;

$$P(IE) = [1-exp(-\lambda\, T_{event})]\, P_{rec}(IE) \qquad (1)$$

where, λ : initiating event occurrence probability per hour

T_{event} : duration time of event

$P_{rec}(IE)$: probability of failure to recover the offsite power

Next, the branch probabilities, *P(EP)*, *P(AFW)*, *P(HPI)* and *P(HPR)*, were calculated as follows;

$$P(EP) = P_1(EP)P_2(EP)P_3(EP)P_{rec}(EP) \quad (2)$$

$$P(AFW) = [P_1(MDAFW)P_2(MDAFW)P_3(MDAFW)P_4(TDAFW) + P_c(AFW)]P_{rec}(AFW) \quad (3)$$

$$P(HPI) = P_1(HPI)P_2(HPI)P_3(HPI)P_{rec}(HPI) \quad (4)$$

$$P(HPR) = P_1(HPR)P_2(HPR)P_3(HPR)P_{rec}(HPR)+P_o(HPR) \quad (5)$$

where, P_n: unavailability of train *n*

P_c: unavailability of shared portion of AFW trains

P_{rec}: probability of failure to recover the system

P_o: probability of operator error

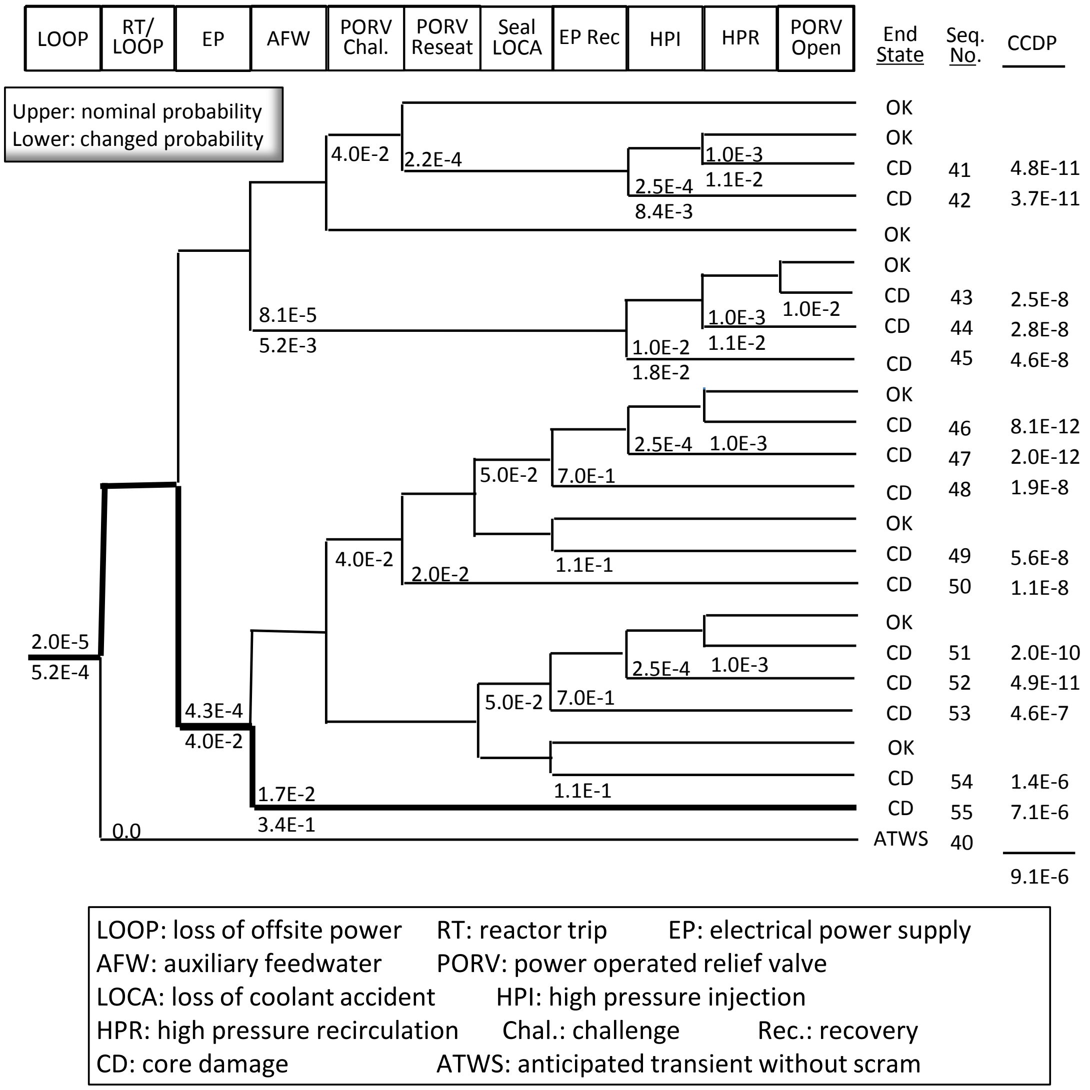

Figure IV.1 Event Tree Model Used for USNRC's ASP Analysis of STP-1 Event[20]

Table IV.1 Branch Probabilities Used in USNRS's ASP Analysis

Branch	System unavailability (*Pn*)	Nonrecovery prob. (*Prec*)	Operator fail prob. (*Po*)	Branch prob. (*P*)
LOOP	$2.0\times10^{-5} \rightarrow 1.2\times10^{-3}$	4.3×10^{-4}		5.2×10^{-4}
RT/LOOP	0.0	1.0		0.0
EP	$5.4\times10^{-4} \rightarrow 5.0\times10^{-2}$	8.0×10^{-1}		4.0×10^{-2}
EP-1 (EDG-1)	5.0×10^{-2}			
EP-2 (EDG-2)	$5.7\times10^{-2} \rightarrow 1.0$			
EP-3 (EDG-3)	$1.9\times10^{-1} \rightarrow 1.0$			
AFW	$3.1\times10^{-4} \rightarrow 2.0\times10^{-2}$	2.6×10^{-1}		5.2×10^{-3}
MDAFW-1	2.0×10^{-2}			
MDAFW-2	$1.0\times10^{-1} \rightarrow 1.0$			
MDAFW-3	$3.0\times10^{-1} \rightarrow 1.0$			
TDAFW	$5.0\times10^{-2} \rightarrow 1.0$			
Shared Components	2.8×10^{-4}			
AFW/EP	$5.0\times10^{-2} \rightarrow 1.0$	3.4×10^{-1}		3.4×10^{-1}
PORV Challenge	4.0×10^{-2}	1.0		4.0×10^{-2}
PORV Reseat	2.0×10^{-2}	1.1×10^{-2}		2.2×10^{-4}
PORV Reseat/EP	2.0×10^{-2}	1.0		2.0×10^{-2}
Seal LOCA	$3.1\times10^{-4} \rightarrow 5.0\times10^{-2}$	1.0		5.0×10^{-2}
EP Rec/Seal LOCA	7.0×10^{-1}	1.0		7.0×10^{-1}
EP Rec/-Seal LOCA	1.1×10^{-1}	1.0		1.1×10^{-1}
HPI	$3.0\times10^{-4} \rightarrow 1.0\times10^{-2}$	8.4×10^{-1}		8.4×10^{-3}
HPI-1	1.0×10^{-2}			
HPI-2	$1.0\times10^{-1} \rightarrow 1.0$			
HPI-3	$3.0\times10^{-1} \rightarrow 1.0$			
HPI (F&B)	$3.0\times10^{-4} \rightarrow 1.0\times10^{-2}$	8.4×10^{-1}	1.0×10^{-2}	1.8×10^{-2}
HPR	$1.5\times10^{-5} \rightarrow 1.0\times10^{-2}$	1.0	1.0×10^{-3}	1.1×10^{-3}
HPI-1	1.0×10^{-2}			
HPI-2	$1.5\times10^{-2} \rightarrow 1.0$			
HPI-3	$1.0\times10^{-1} \rightarrow 1.0$			
PORV Open	1.0×10^{-2}	1.0	4.0×10^{-4}	1.0×10^{-2}

Because two of three EDGs were inoperable in the event, the unavailabilities of them were changed from their respective nominal probabilities to 1.0 but the unavailability of operable EDG was not changed, as shown in Table IV.1. Then, the branch probability for EP was obtained by multiplying the changed EDG unavailability by the probability of failure to recover.

The branch probability for AFW was determined in consideration of relation between AFW and EP. Specifically, two trains of AFW were assumed unavailable because their respective EDGs were inoperable and hence, the calculated branch probabilities for AFW were different depending on the availability of electrical power supply. In the same way, the branch probabilities for HPI and HPR were calculated taking into account the relation between the respective systems and electrical power supply. The event tree in Figure IV.1 was quantified using the calculated branch probabilities. As a result, the CCDP of this event was estimated at 9.1×10^{-6} and the sequence with station blackout (SBO) followed by AFW failure was identified dominant (Sequence No.55 in Figure IV.1) .

2.3 Analysis with Fault Tree Linking Approach

The fault tree linking approach utilizes fault trees as well as event trees. The event trees are reconstructed by modeling the accident management such as containment venting at BWRs and the recovery actions of secondary system at PWRs. An example of the event tree models is illustrated in **Figure IV.2**(29). The fault trees are constructed for individual safety systems and their support systems and model the component failures, failure of backup and recovery actions, common cause failures and dependent failures. An example of fault trees is displayed in **Figure IV.3**(19).

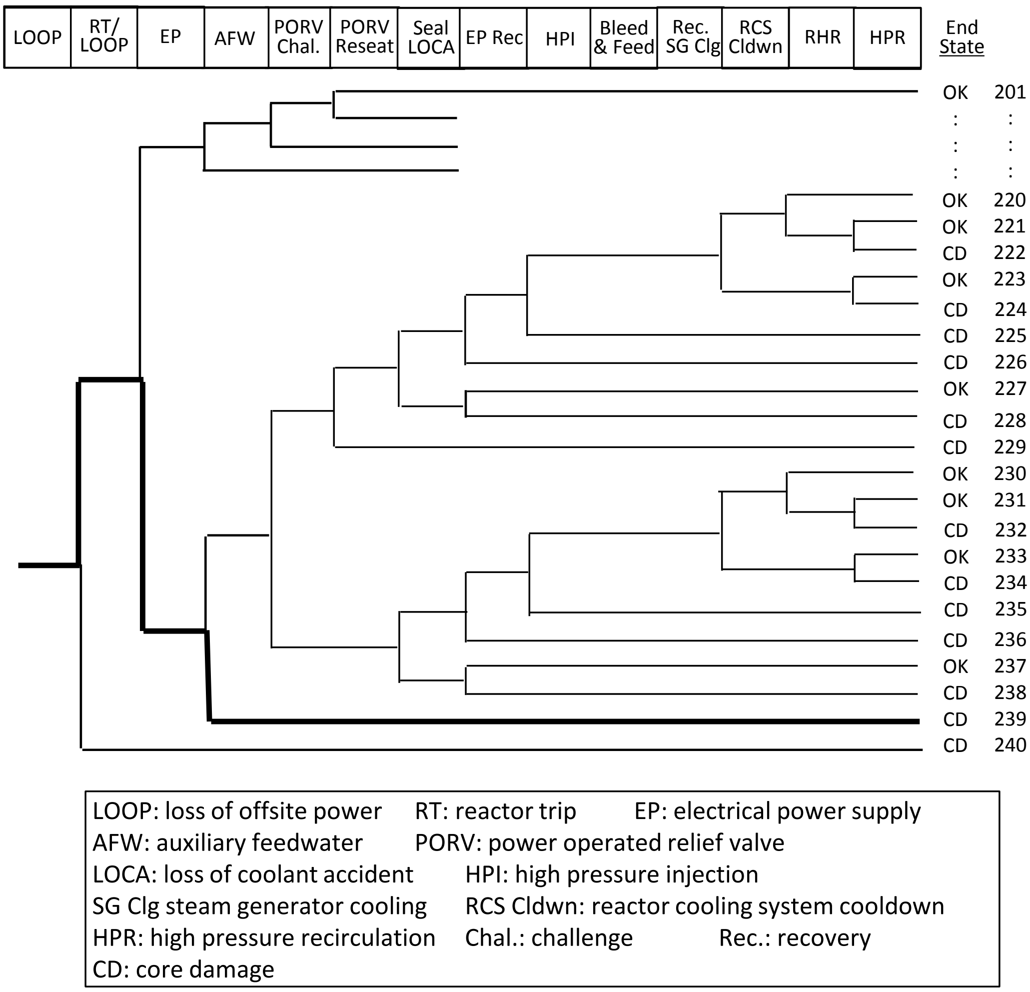

Figure IV.2 Example of ASP Event Tree Models Reconstructed

In this appoach, the initiating event occurrence probability is calculated in the same way as the conventional event tree models mentioned in Subsection 2.2. The event trees are quantified by linking fault trees according to individual sequences delineated in the event trees instead of calculating the branch probabilities. For example, the sequence with SBO followed by AFW failure (Sequence No.239 in Figure IV.2) is quantified by linking fault trees for EP and AFW and multiplying the occurrence probability of LOOP to obtain the CCDP for this sequence. In this process, the failure probabilities of components represented in fault trees are changed to reflect the plant system conditions during the event. For the STP-1 event, the failure probabilities of two EDGs and TDAFW pump are changed from their respective nominal values to 1.0. In addition, the probablities of common cause failures are changed to 0.0 because only one EDG and one AFW train were operable. The similar changes are applied to the fault trees for HPI and HPR. In such a way, the CCDPs for the other sequences are calculated.

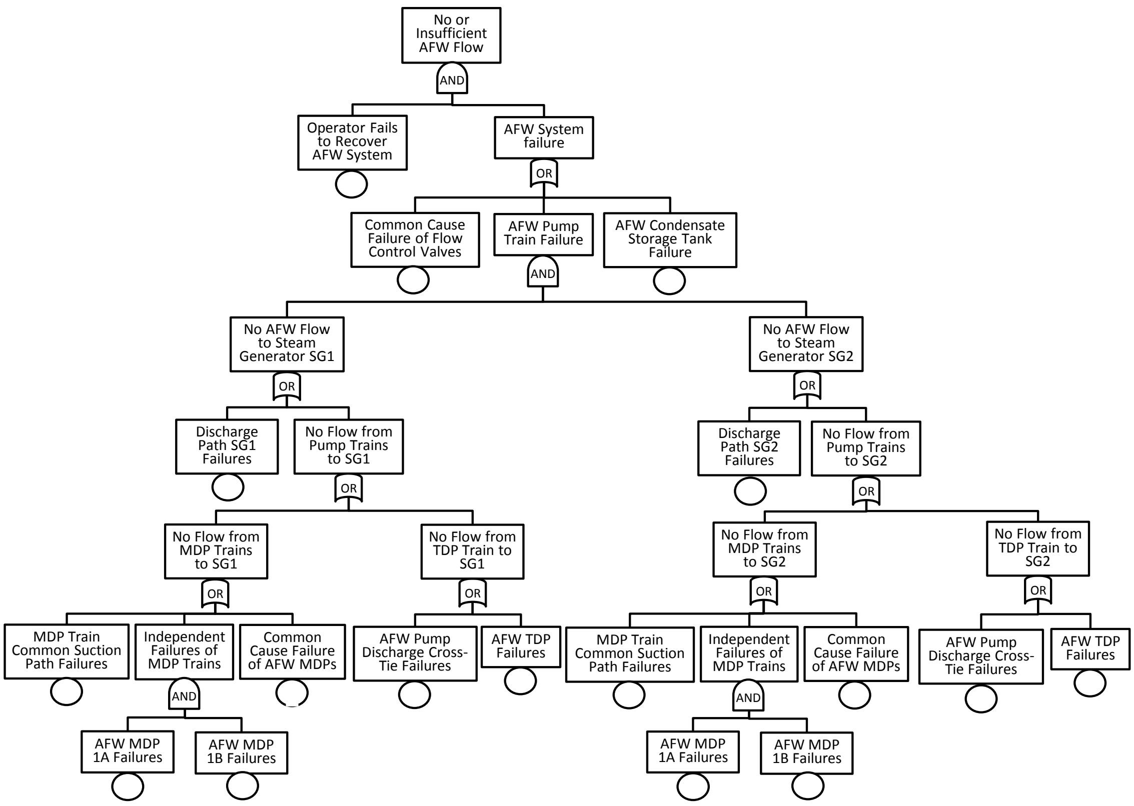

Figure IV.3 Example of ASP Fault Tree Models(19)

IV.3 Accident Sequence Precursor Analyses for Steam Generator Tube Rupture Events

The steam generators (SGs) at PWRs are shell- and -tube heat exchangers, each of which contains several thousand thin-walled tubes. These tubes are designed to confine radioactivity to the primary coolant during normal operation. However, the primary pressure is higher than the secondary pressure, so rupture of the tube can result in radioactivity release to the environment through relief valves in the secondary system.

The PRA/PSA studies for PWRs have shown that, even though SGTRs are small contributors to the total core damage frequency, they are risk significant because radionuclides are likely to bypass the containment through relief valves in the secondary system(30,31). In addition, operating experience indicates that a significant number of the defective tubes have been found and removed from service or repaired each year. Despite appropriate countermeasures and inspections taken, the tube degradation has been still a continuous important issue to nuclear power plant safety. The previous study by MacDonald(31), which integrates and evaluates the relevant information on SG tube failures, indicates that ten tube rupture events have occurred at the rate of about one every two years during 20 years. These rupture events resulted in leak rates ranging from 425 *l*/min to 2,900 *l*/min and complex plant transients, in some of which the operators took a relatively long time to recognize that SGTR had occurred and consequently, isolation of the affected SG was delayed and/or the primary pressure was held higher than the secondary pressure in the affected SG for a relatively long period. Also, some events involved additional malfunctions.

On the other hand, the USNRC's ASP Program analyzed four actual SGTRs and one potential SGTR(9,14,16,20). Direct comparison of the results is not possible, however, because ASP analyses have been performed on a yearly basis and the models used were different from year to year. Therefore, any discussion on the results has not been done so far from the generic point of view.

In this study, all of the ten actual SGTR events identified in Ref. (31) and the potential SGTR event identified in Ref. (16) are systematically analyzed within the framework of the ASP methodology. The primary objectives of the study are to identify risk significant anomalies observed during the events in terms of the potential for core damage and to obtain generic insights useful for examining alternative mitigation measures for SGTR. In order to meet these objectives, this study prepares an ASP model consistently applicable to the above events, which is called the consistent ASP model, based on the latest version of the USNRC's ASP models and evaluates their respective risk significance of the SGTR events on a common basis. The latest ASP models consist of standardized event trees and plant-specific fault trees(22). Although the event trees are available, the fault trees have not been open. This study modifies the standardized event trees to represent proceduralized recovery actions that have not been considered in the USNRC's models, and estimates the branch probabilities needed for the event tree quantification by constructing simplified fault trees. The fault trees are composed of failures of pump trains and/or valves and operator errors in consideration of the plant system configurations based on the USNRC's plant information book(32) and so on. This section describes the model applied and the results obtained, and discusses the significant anomalies observed during the SGTR events from the point of CCDPs.

1. ASP Model and Method Applied

1.1 Event Tree Models

The standardized event trees for the ASP analysis are provided for eight plant classes to reflect the design differences among commercial nuclear power plants in the United States(21). As shown in **Table IV.2**, PWRs are separated into five plant classes: Classes A, B, E, G and H. Classes A and B address Westinghouse plants, Class D represents Babcock & Wilcox plants, and Classes G and H include Combustion Engineering plants. Event trees for each plant class depict core damage sequences for four initiating events: nonspecific reactor trip, LOOP, small break LOCA and SGTR. The SGTR event trees constructed for all the plant classes except for Class H, which are identical to each other, do not model the primary feed and bleed (F&B) operation though it can be utilized as an alternative method for secondary cooling. In this study, the SGTR event tree was reconstructed by incorporating the F&B capability in order to evaluate more realistically their respective significance of the SGTR events as shown in **Figure IV.4**. As for Class H, the event tree provided in Ref. (21), shown in **Figure IV.5**, was applied without any modification.

Table IV.2 PWR Plant Classification in ASP

PWR Class A	Westinghouse plants which require the use of containment spray system for decay heat removal	7 plants: (Ex) North Anna-1&2, Surry-1&2
PWR Class B	Westinghouse plants which can utilize high and low pressure recirculation for decay heat removal	44 plants: (Ex) Davis Besse[a], Ginna, Maine Yankee[b], McGuire-1&2, Point Beach-1&2, Prairie Island-1&2
PWR Class D	Babcock & Wilcox plants which have the capability of the primary feed and bleed without PORVs	6 plants: (Ex) Oconee-1,2&3, TMI-1
PWR Class G	Combustion Engineering plants which have the capability of the primary feed and bleed	8 plants: (Ex) Fort Calhoun, Millstone-2
PWR Class H	Combustion Engineering plants which can utilize the condensate system as an alternative method	6 plants: (Ex) Palo Verde-1,2&3

[a] Davis Besse is a Babcock and Wilcox plant. [b] Maine Yankee is a Combustion Engineering plant.

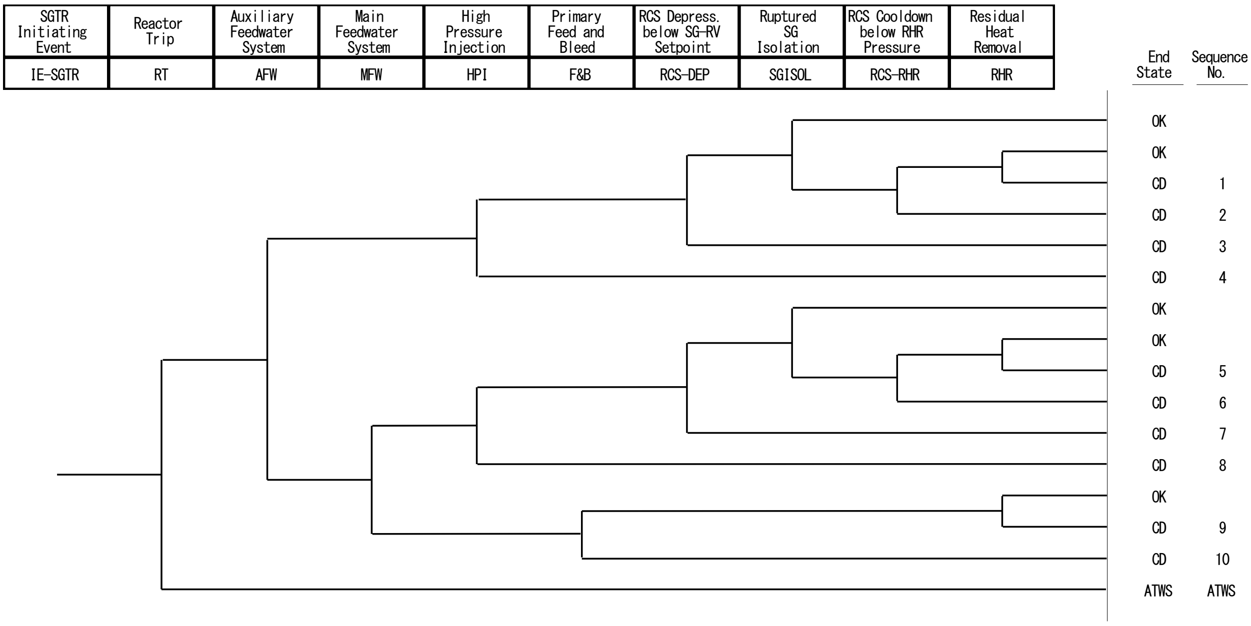

Figure IV.4 SGTR Event Tree Model for PWR Classes A, B, D and G

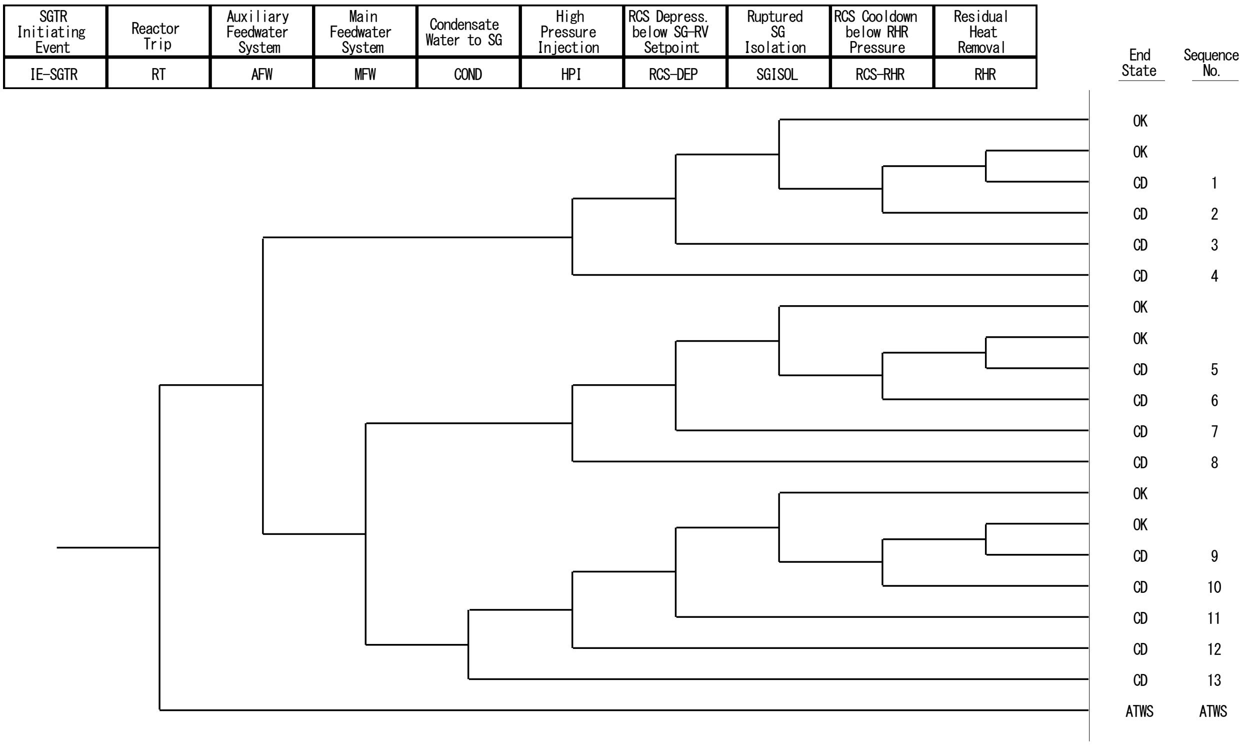

Figure IV.5 SGTR Event Tree Model for PWR Class H

1.2 Fault Tree Models

In this study, simplified train-based fault trees were developed for branches on the event trees by incorporating only failures of major active components, such as pumps, and operator errors including recovery failures. In developing fault trees, system configurations were taken into account by modeling the number of pump trains and pressurizer power-operated relief valves (PORVs) installed in a particular plant. An example of the simplified fault trees is shown in **Figure IV.6**. As indicated in this figure, a likelihood of recovering from system failures was considered for some systems. The probabilities for component failures and operator errors were referred to the USNRC's document, 'Daily Events Evaluation Manual'(29), that provides the event trees and the probabilities of major component failures and operator errors for typical plants to prepare the ASP models.

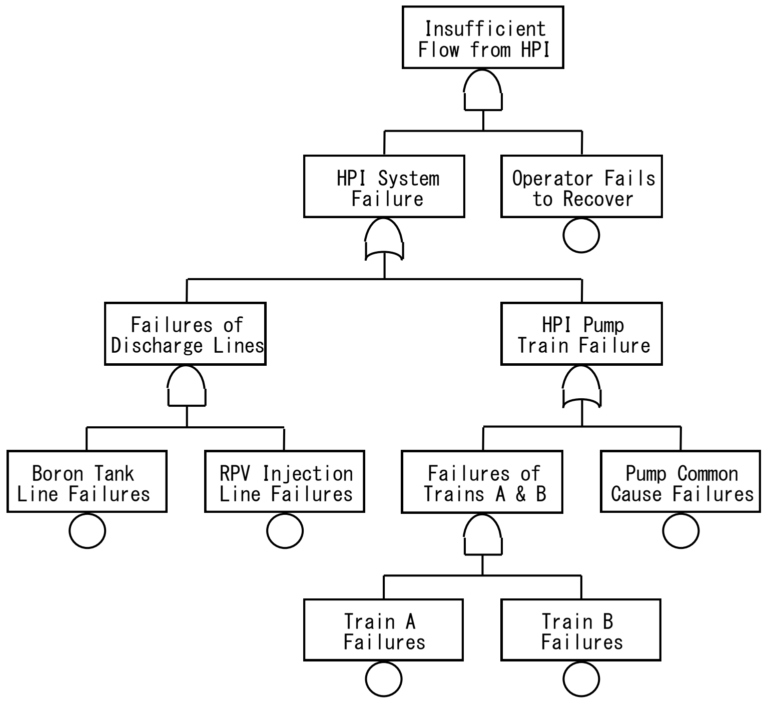

Figure IV.6 Example of Train-based Simplified Fault Tree Models

1.3 Evaluation Method of Event Significance

The method for evaluating risk significance in this study followed the approach used in the USNRC's ASP program, where the risk significance is evaluated by calculating a CCDP of subsequent core damage given the failures observed during a particular event. In the processes of this calculation, the observed failures or unavailabilities are reflected on simplified fault trees by setting their corresponding failure probabilities to 1.0 (failed). On the contrary, a likelihood that the component(s) operable during the event would fail is expressed as the failure probability preset on simplified fault trees, that is called the preset value hereafter. Therefore, branch probabilities for systems/functions observed to be unchallenged or successful during the event are estimated without any change of the preset values.

2. Description of SGTR Events

2.1 Typical Plant Response to SGTR

Typically, following SGTR, automatic reactor trip could occur on either low pressurizer pressure or over-temperature delta-T and subsequently, safety injection (SI) signal would be generated on low pressurizer pressure. Reactor coolant system (RCS) makeup would be performed by high pressure injection (HPI). For smaller tube failures, reactor trip and/or HPI actuation may be done manually. Main feedwater (MFW) is tripped on SI signal and auxiliary feedwater (AFW) is automatically actuated to supply coolant to all SGs. The operators are required to identify the ruptured SG and isolate it by closing the main steam isolation valve (MSIV) and other potential leak paths. In addition, the operators should perform depressurization of RCS by secondary cooling with use of AFW/MFW and steam dump or PORVs to minimize the leak flow. After the RCS depressurization is completed, the operators should secure HPI to prevent or limit the RCS repressurization. Long-term decay heat removal is carried out by secondary cooling, high pressure recirculation (HPR) or residual heat removal (RHR). Prior to the RHR operation, RCS should be cooled down to the RHR initiation pressure.

2.2 Description of Actual Events

To provide easier comparison of the ten actual events and one potential event to be analyzed, plant transient information is summarized in **Table IV.3**. In the following, a brief description of each event is provided.

(1) Point Beach-1 event (26/2/1975)[31,33]

SGTR occurred while operating at full power with one charging pump (CHP) running. The other two CHPs started at 2 and 19 min. The operators determined the leak in SG- B at 28 min and initiated power reduction at 30 min. The reactor was manually tripped at 47 min from 25% power. SI did not occur and was blocked at 54 min. MSIV for SG-B was closed at 48 min but the feedwater to SG-B was not stopped until 58 min. At 51 min, the RCS cooldown was begun by dumping steam from the intact SG to the condenser and was continued. At 108 min, the RCS pressure was reduced to below the SG relief valve setpoint. However, the primary pressure remained slightly above the secondary pressure for 6-7 h. During the cooldown, the RCS inventory was controlled by the intermittent HPI operation.

Table IV.3 Summary of Plant Transient Information

	Point Beach-1 WH-2loop	Surry-2 WH-3loop	Doel-2 ACE/WH-2loop	Prairie Island-1 WH-2loop	Ginna WH-2loop	Fort Calhoun CE-2loop	North Anna-1 WH-3loop	North Anna-1 WH-3loop	McGuire-1 WH-4loop	Mihama-2 MHI/WH-2loop	Palo Verde-2 CE-80(2loop)
Event Date	1975.2.26	1976.9.15	1979.6.25	1979.10.2	1982.1.25	1984.5.16	1987.7.15	1989.2.25	1989.3.7	1991.2.9	1993.3.14
Max. Leak Rate	470 ℓ/min	1250 ℓ/min	510 ℓ/min	1270 ℓ/min	2900 ℓ/min	425 ℓ/min	2410 ℓ/min	290 ℓ/min	1900 ℓ/min	1330 ℓ/min	910 ℓ/min
Plant Status	At power	At power	Startup	At power	At power	Startup	At power	Hot shutdown	At power	At power	At power
SGTR Identified	24-28 min	17-18 min	9 min	18.5 min	< 1 min	32 min	5-16 min	< 20 min	< 1 min	5 min	< 57 min
Type and Time of Scram	Manual 47 min	Manual 10 min	N/A	Automatic 10 min 9 sec	Automatic 3 min 12 sec	N/A	Manual 5 min	N/A	Manual 8 min	Automatic 10 min	Manual 13 min
Type and Time of SI	No (blocked)	Manual 11 min	Automatic 19.2-19.5 min	Automatic 10 min 14 sec	Automatic 3 min 20 sec	No	Automatic 5 min 20 sec	No	No	Automatic 10 min 7 sec	Automatic 13 min 22 sec
Rup. SG Isol.	48 min	17-18min	9.4 min	27 min	15 min	41 min	18 min	< 60 min	8-11 min	22 min	2 h 54 min
Method of RCS Depresurization	- Steam dump (51 min)	- PZR spray - Steam dump (21 min)	- PZR spray (28 min)	- PZR PORV (43 min)	- PZR PORV (42 min) - Steam dump (75 min)	- Steam dump (41 min)	- PZR PORV - Steam dump (19 min)	No Information (79 min)	- Steam dump (14, 25 min) (3.5 hrs)	- Steam dump (22 min)	- Steam dump (167 min)
RCS-SG Press. Equalization	6-7 h	< 60 min	> 70 min	61 min	approx 3 h	a few hrs (> 1 h)	34 min	approx 2 h	(1) 47 min (2)10 h 47 min	68 min	> 3 h
RHR in Service	3 h 5min	approx 11 h	3 h 15 min	16.5 h	21.5 h	3 h 47 min	5 h 49 min	> 5 h	17 h	8.5 h	18 h
Unavailable Components	- Rad. monitors	- Blowdown monitor	- Letdown line valve	- MSIV for intact SG - Rad. monitors	- PZR PORV - SG relief valve	None	- MFW relif valves	- RHR suction valve	- SG relief valve - Letdown isol. valve, etc.	- PZR PORVs - MSIV for ruptured SG	- SGTR alarms

(2) Surry-2 event (15/9/1976) [31,33]

While operating at full power, the maximum charging flow alarm and a sudden decrease in the RCS pressure were indicated. At 5 min, the operators started the second CHP but 2 min later, the CHP suction automatically switched to the refueling water storage tank (RWST) and then, power reduction was commenced. At 10-11 min, the operators tripped the turbine, resulting in automatic reactor trip from 70% power, two AFW pumps started and manual SI was initiated. At 15-16 min, one HPI pump (HPI-A) was stopped and one HPI pump discharge was realigned to the normal charging flow path. At 17-18 min, SGTR was determined to be in SG-A and SG-A was isolated. AFW was automatically tripped and one MFW pump was initiated for the RCS cooldown. At 21 min, HPI-A restarted and the HPI discharge lines were realigned to the SI flow paths. The plant cooldown was done by dumping steam from the intact SGs to the condenser and the RCS pressure was reduced to below the SG relief valve setpoint at 60 min.

(3) Doel-2 event (25/6/1979) [31,33]

While the plant was heated to normal operating condition with the reactor not critical, a rapid decrease of the RCS pressure was indicated. The second CHP was manually started at 1.8 min. At 9.4 min, the operators isolated the ruptured SG (SG-B) and then, the third CHP was started and the CHP suction was aligned to RWST. At 19 min, HPI automatically started on low pressurizer pressure. In attempt to decrease the RCS pressure, at 28 min, pressurizer spray was manually initiated. At 41 min, AFW was automatically actuated and at 50 min, the AFW flow to SG-B was terminated. The RCS cooldown was commenced at 68-88 min and the HPI was stopped.

(4) Prairie Island-1 (2/10/1979) [9,31,33]

SGTR occurred while operating at full power. At 7-8 min, low pressurizer pressure and level were alarmed and power reduction was initiated. The other two CHPs were manually started at 9-10 min. Immediately after that, the reactor was automatically tripped and SI occurred. The operators identified tube rupture in SG-A at 18.5 min and closed its MSIV at 27 min. MSIV for the intact SG was also automatically closed but manually opened immediately. At 42-43 min, one HPI was secured and one PORV was opened intermittently to reduce the RCS pressure. The operators secured the remaining HPI at 52 min since the pressurizer level reached the high level setpoint, but the pressurizer relief tank rupture disk burst. At 61 min, the primary and secondary pressures were equalized. At 96 min, normal cooldown was started.

(5) Ginna (25/1/1982) [31,34,35]

While operating at full power, several alarms indicating low pressurizer pressure and SG-B steam/feedwater flow mismatch were received. At 1.5-2.5 min, power reduction was commenced, the other two CHPs were manually started, and the condenser steam dump automatically initiated. At 3-5 min, the reactor tripped automatically, HPI was actuated, two motor-driven AFW pumps (MDAFWs) and turbine-driven AFW pump (TDAFW) started automatically, and the steam dump stopped. At 7 min, MDAFW feeding SG-B was secured and steam supply to TDAFW from SG-B was terminated. At 15 min, tube rupture was determined to be in SG-B and its MSIV was closed. At 21 min, TDAFW was secured but the SG-B water level continued to rise. At 42 min, the operators opened and closed one PORV in attempt to equalize the primary and secondary pressures, but at 44 min, the PORV stuck open and the operators closed its block valve immediately. At 54 and 63 min, an SG-B safety valve cycled and at 72 min, HPI was secured. Condensate pump was also secured, causing the condenser steam dump unavailable, and the operators began steam dump using an atmospheric relief valve of the intact SG. One HPI was restarted at 102 min. At 132 min, a fifth lift of the SG-B safety valve occurred and HPI was stopped. The safety valve closed but continued to leak water (probably after 55 min). At 167 min, HPI was switched from continuous to intermittent operation. The primary and secondary pressures were equalized at 182 min.

(6) Fort Calhoun Event (16/5/1984) [31]

During plant startup with the reactor being pressurized, the operators noted that the pressurizer level was no longer increasing and the pressurizer pressure was slowly decreasing. The other two CHPs were manually started immediately. At 18 min, the operators switched the CHP suction to the volume control tank (VCT) and increased the charging flow. At 27 min, two CHPs were secured because the VCT level reached 0%. A continuing increase in the SG-B level was indicated at 32 min and then the AFW pump feeding SG-B was secured. At 40 min, MSIV for SG-B was closed and then, the RCS cooldown was commenced using the intact SG and its atmospheric dump valve.

(7) North Anna-1 event (15/7/1987) [14,31,36]

Shortly after the plant reached full power, the pressurizer level and pressure began to rapidly decrease. At 5 min, the reactor was manually tripped and automatic SI occurred due to a low-low pressurizer pressure. At 16-18 min, SG-C was determined to be defected and isolated. At 19 min, the RCS cooldown was commenced by dumping steam from the intact SGs, resulting in the pressurizer level off scale. By controlling the pressurizer spray, the pressurizer level was recovered at 28 min. In order to enhance the depressurization, one PORV was opened at 34 min and then, the primary and secondary pressures in SG-C were roughly equalized. At 48 min, an orderly plant shutdown was initiated. During this event, two relief valves in the MFW system failed to reseat but were manually closed approximately 30 min into the event.

(8) McGuire-1 (7/3/1989) [16,31,37]

While operating at full power, the operators observed decrease in the SG-B feedwater flow and the pressurizer level and immediately recognized SGTR. At 4-5 min, power reduction was initiated and the second CHP was manually started. At 8-9 min, the operators initiated a reactor trip and swapped the CHP suction to RWST. At that time, relief valves for the intact SGs cycled. The operators began immediately to isolate SG-B and initiate the RCS cooldown at 11 min. Turbine bypass valves (TBVs) were opened to dump steam to the condenser at 14 and 25 min. At 21-25 min, SI was manually blocked. At 47 min, the primary and secondary pressures in SG-B were equalized and the break flow was temporarily terminated. However, the SG-B secondary pressure continued to decrease and the break flow resumed. At 3.5 h, further cooldown began using TBVs but the primary pressure remained above the secondary pressure until 10 h. The post-trip review identified several anomalies on SG-B relief valve, letdown isolation valve, SG blowdown sample line monitor and so forth.

(9) Mihama-2 event (9/2/1991)[31,38,39]

SGTR occurred while operating at full power. At 5-7 min, the operators started the third CHP and commenced power reduction. At 10 min, the reactor scrammed automatically on low pressurizer pressure and subsequently, automatic SI occurred on low pressurizer pressure and level. At 12 min, an MDAFW discharge valve to the ruptured SG (SG-A) was closed and at 15 min, the operators attempted to close MSIV for SG-A but it failed to close completely (an operator closed it at the failed valve at 22 min). At 30 min, the operators attempted to open two PORVs for depressurizing RCS but they failed because an air supply valve for both PORVs had been erroneously closed (PORVs were declared inoperable at 45 min). At 39, 49 and 59 min, an SG-A relief valve cycled. During this period, the operators opened a locked closed pressurizer auxiliary spray valve to depressurize RCS and then, HPI were secured after confirming that the pressurizer level was recovered. At 68 min, the primary and secondary pressures were equalized. At 93-94 min, the operators initiated steam dump using TBVs.

(10) Palo Verde-2 event (14/3/1993)[20,31,40]

SGTR occurred while operating at 98% power. At 2 min, the operators started the third CHP. At 13 min, the reactor was manually tripped and then, automatic SI occurred due to low pressurizer pressure. Even though the operators suspected SGTR, a reactor trip was diagnosed using the diagnostic logic tree. However, the entry conditions for reactor trip recovery procedure could not be met because of low pressurizer level. So, the operators entered the functional recovery procedure (FRP) and continued recovery actions per FRP until 114 min although SGTR was confirmed at 57 min. These actions included switchover of the CHP suction from VCT to RWST and restoration of the SG blowdown lines at 46 min. According to the SGTR recovery procedure which was entered at 131 min, the RCS cooldown was restarted at 167 min and the ruptured SG was isolated 7 min later. The use of FRP resulted in significantly delayed isolation of the ruptured SG and depressurization of RCS. In addition, three primary indicator alarms for SGTR were not available, which confused the operators.

(11) Potential SGTR at North Anna-1 (25/2/1989)[16, 41]

The plant automatically tripped from 76% power on SG-C steam/feedwater flow mismatch, caused by closure of the MFW regulating valve, coincident with low SG level. Following the reactor trip, the RCS pressure and temperature decreased and the AFW pumps started automatically. At 20 min, primary to secondary leakage was indicated and after locating the leak, the SG-C steam supply valve to TDAFW was manually isolated. The leak rate was within the capacity of one CHP flow. The RCS cooldown commenced at 79 min and the RCS pressure was reduced to below SG-C pressure at about 2 h. While placing the RHR system in service, its suction isolation valve failed to remain open. RHR could not be placed in operation for more than 3 h.

3. Modelling Assumptions

3.1 Success Criteria for Fault Trees

In constructing simplified fault trees for systems represented on the ASP event tree models for SGTR, the success criteria should be defined in consideration of the plant system configurations. As shown in Figures IV.4 and IV.5 the event trees for SGTR consist of nine top events excluding the initiating event. For four of them (MFW, RCS depressurization to below SG relief valve, ruptured SG isolation and RCS cooldown to below RHR initiation pressure), generic simplified fault trees applicable to all the plants of interest here were constructed because their functionabilities should be heavily dependent on the operators' actions and the contributions due to differences in hardware appeared small. The fault trees for these top events constructed in this study can be represented in the Boolean algebra equations as follows:

Top(MFW)=(MFW hardware failures)
　　　　∩(failure to recover MFW),
Top(RCS Depress.)=(hardware failures)

$\cup$*(failure to initiate depressurization),*

Top(RCS Cooldown)=(hardware failures)

$\cup$*(failure to initiate cooldown),*

Top(SG Isolation)=(failure to timely isolate due to valve failures/operator error)

$\cap$*(failure to recover before RWST depletion)*

where '$\cap$' and '$\cup$' denote 'AND-gate' and 'OR-gate' in the fault tree, respectively.

Also, as seen in many PSA studies[30], the fault tree for reactor trip expressed by the following Boolean algebra equation was applied in common to all the plants.

Top(Reactor Trip)=(trip system failure) $\cap$*(manual trip failure).*

For the top event 'condensate water to SG' on the event tree for Palo Verde (Class H), the branch probability was derived from Ref. (29) instead of constructing a fault tree.

As for the other top events (AFW, HPI, F&B and RHR/HPR), their success criteria were defined for individual plants. **Table IV.4** summarizes the success criteria defined in this study. Along with this table, the success criteria are described below.

(1) Auxiliary feedwater (AFW)

Five of the ten plants (Surry, Doel, North Anna, McGuire and Mihama) have the three-train system which consists of two MDAFWs and one TDAFW and thus, their success criteria were defined as one-out-of-three trains. Although other two plants, Point Beach and Prairie Island, also have the three-train system, one MDAFW is shared by their respective adjacent units and additional operators' actions were needed for using this train. Therefore, the success criteria of AFW for these plants were defined as one-out-of-two trains. The two plants, Fort Calhoun and Palo Verde, have the two-train system (one MDAFW and one TDAFW) and their success criteria were defined as one-out-of-two trains. The Ginna plant has the uniquely designed system which consists of a main three-train system and a standby two-train system. The main system has two MDAFWs and one TDAFW, and the standby system has two MDAFWs with 50% capacity per each. Hence, the success criteria were defined as one-out-of-three main AFW trains or two-out-of-two standby AFW trains.

(2) High pressure injection (HPI)

The four plants (Point Beach, Prairie Island, Mihama and Palo Verde) have the two-train system and thus, their success criteria were defined as one-out-of-two trains. At the two 3-loop plants (Surry and North Anna), all of three HPI pumps are used as charging pumps. The success criteria of HPI were defined as one-out-of-three trains in consideration with one charging pump operating. Ginna and Fort Calhoun have the three-train system and it was assumed that one-out-of-three trains was needed for SGTR. At McGuire, in addition to two-train HPI system, two charging pumps can be used as the SI pumps. The success criteria were defined as one-out-of-two HPI trains or two-out-of-two charging pump trains. Doel has the four-train HPI system and two-out-of-four trains were assumed as success criteria.

(3) Primary feed and bleed (F&B)

The F&B Operation requires the HPI operation in conjunction with opening of PORVs. The success criteria of HPI were as mentioned above. For all the plants except Palo Verde which has no capability of this function, it was assumed that opening of all PORVs was required even though there is a difference in the number of PORVs installed.

(4) Residual heat removal (RHR)

The success criteria of RHR were defined as one-out-of-two trains for all the plants that have the two-train RHR system installed. Because only limited information on Doel is available, as well, the two-train RHR system was assumed and the above success criteria were used. In the event where the F&B operation would be used, HPR in conjunction with RHR is required for long-term decay heat removal at Class B plants such as Point Beach, and HPR and containment spray recirculation (CSR) are required at Class A plants such as Surry and at Class G plant (Fort Calhoun). Therefore, the success criteria for such an event were assumed as follows: One HPI train and one RHR train including a heat exchanger for Class B plants, one HPI train and one CSR train including a heat exchanger for Class A and G plants.

3.2 Failure Probabilities Applied

The respective branch probabilities for top events were obtained from the simplified fault trees constructed based on the above assumptions. In calculating branch probabilities, actual and potential component failures and/or operator errors observed during individual events were reflected by changing failure probabilities from preset values to event-specific values. On the other hand, branch probabilities for the system successfully operated or unchallenged during the events were given by quantifying the simplified fault trees with use of the preset values. All of the individual events involved successful actuation or operable condition of the reactor trip, AFW, HPI and the RCS cooldown to the RHR pressure, and in the Palo Verde event, the condensate system was also maintained operable. Hence, no failure probabilities were changed for these fault trees. In the following, the changed failure probabilities are described.

Table IV.4 Plant Safety Features and Success Criteria Concerning SGTR

	Point Beach-1 Class B, 2loop	Surry-2 Class A, 3loop	Doel-2 Class B, 2loop	Prairie Island-1 Class B, 2loop	Ginna Class B, 2loop	Fort Calhoun Class G, 2loop	North Anna-1 Class A, 3loop	McGuire-1 Class B, 4loop	Mihama-2 Class B, 2loop	Palo Verde-2 Class H, 2loop
AFW	1 MDP, 1 TDP 1 of 2 trains	2 MDPs, 1 TDP 1 of 3 trains	2 MDPs, 1 TDP 1 of 3 trains	1 MDP, 1 TDP 1 of 2 trains	2 MDPs, 1 TDP, 2 SAFW pumps 1 of 3 trains or 2 of 2 SAFW	1 MDP, 1 TDP 1 of 2 trains	2 MDPs, 1 TDP 1 of 3 trains	2 MDPs, 1 TDP 1 of 3 trains	2 MDPs, 1 TDP 1 of 3 trains	1 MDP, 1 TDP 1 of 2 trains
MFW (needs operators' action)	2 MDPs 1 of 2 pumps	2 MDPs 1 of 2 pumps	2 MDPs 1 of 2 pumps[a]	2 MDPs 1 of 2 pumps	2 MDPs 1 of 2 pumps	3 MDPs 1 of 3 pumps	3 MDPs 1 of 3 pumps	2 TDPs 1 of 2 pumps	3 MDPs 1 of 3 pumps	2 TDPs 1 of 2 pumps
HPI	2 SIPs 1 of 2 trains	3 SIPs (CHPs) 1 of 3 trains	4 SIPs 2 of 4 trains	2 SIPs 1 of 2 trains	3 SIPs 1 of 3 trains	3 SIPs 1 of 3 trains	3 SIPs (CHPs) 1 of 3 trains	2 SIPs, 2 CHPs 1 of 2 SIPs or 2 of 2 CHPs	2 SIPs 1 of 2 trains	2 SIPs 1 of 2 trains
Condensate Sys.	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	1 of 3 pumps
Primary Feed & Bleed Operation	HPI + 2/2 PORVs	HPI + 2/2 PORVs	HPI + 2/2 PORVs[a]	HPI + 2/2 PORVs	HPI + 2/2 PORVs	HPI + 2/2 PORVs	HPI + 2/2 PORVs	HPI + 3/3 PORVs	HPI + 2/2 PORVs	N/A
RHR (with RCS depressurized)	1 of 2 trains	1 of 2 trains	1 of 2 trains	1 of 2 trains	1 of 2 trains	1 of 2 trains	1 of 2 trains	1 of 2 trains	1 of 2 trains	1 of 2 trains
HPR (following successful F&B)	1/2 SIPs + 1/2 RHR	1/3 SIPs + 1/2 CSR (1/2 HX, 1/2 pumps)	1/2 SIPs + 1/2 RHR[a]	1/2 SIPs + 1/2 RHR	1/3 SIPs + 1/2 RHR	1/3 SIPs + 1/2 CSR (1/2 HX, 1/3 pumps)	1/3 SIPs + 1/2 CSR (1/2 HX, 2/4 pumps)	1/2 SIPs + 1/2 RHR	1/2 SIPs + 1/2 RHR	N/A

(1) Initiating event

The initiating event probabilities were set to 1.0 except for the North Anna event in 1989, that is the potential SGTR event. The leak size observed in this event was small and the occurrence probability of a large tube rupture was assumed to be 0.1, that was employed in the USNRC's analysis in Ref. (16).

(2) Main feedwater (MFW)

The two events at North Anna involved the malfunctions of the valves: one was on the regulating valve and the other was on two relief valves. For these events, the probability of 1.0 was set to the basic event 'MFW hardware failures'.

(3) Primary feed and bleed (F&B)

In the Ginna event, one PORV stuck open and its block valve was closed. The stuck open PORV might not prevent the RCS depressurization but this study assumed that the F&B operation could not be done due to the PORV inoperability. The Mihama event experienced failure to open of both PORVs. Therefore, the branch probability of 1.0 was applied to these two events.

(4) RCS depressurization to below SG relief valve setpoint

In the five events at Point Beach, Ginna, Fort Calhoun, McGuire and Palo Verde, it took considerably longer times (more than a few hours) to equalize the primary and secondary pressures. Therefore, it is considered that these events would have the potential for serious conditions if any operator actions would have not been taken. This study examined whether or not such a delayed action might be risk significant from the point of the potential for core damage. As can be seen from the Boolean algebra equation in the previous subsection, however, the fault tree constructed there does not model such a delayed action explicitly mainly because such a delayed action is generally regarded as one of operator errors or not considered in fault trees. In order to represent the delayed action observed during these events, in this study, the basic event 'failure to initiate depressurization' on the relevant fault tree was divided into 'failure to timely initiate' and 'failure to recover from the initial error'. On the other hand, the USNRC's analysis for the Palo Verde event in Ref. (20) took into account failure to depressurize RCS and determined its probability by applying a probability of 0.12 to failure to timely initiate the RCS depressurization and a probability of 0.34 to failure to recover from the errors during the initial depressurization. Therefore, this study determined the probability of 'failure to initiate depressurization' on the relevant fault tree by multiplying these two probabilities and applied it to the above five events in common even though their respective situations were actually different from each other. In the Mihama event, PORVs failed to depressurize RCS and the pressure equalization was performed by using the pressurizer auxiliary spray and securing the HPI pumps based on the emergency operating procedure[(38)]. In order to address the PORV failure, this study assigned a probability of 1.0 to the basic event ‘hardware failures’ on the relevant fault tree. As well, the use of pressurizer auxiliary spray was regarded as a recovery action and a probability of 4.0×10^{-2}, which was derived from Ref. (19), was applied to its failure. In Ref. (19), four recovery classes are defined and a non-recovery probability is given for each class. The probability applied here is that for the failure recoverable in the required period on a procedural basis and the lowest one among them. However, other possible recovery actions were not taken into account for this event.

(5) Isolation of ruptured SG

The ruptured SG isolation was slowly done (it took more than 40 min) in the four events at Point Beach, Fort Calhoun, North Anna in 1989 and Palo Verde. In order to address the delayed isolation, a basic event ‘failure to timely isolate due to operator error’ was taken into account instead of the basic event ‘failure to timely isolate due to valve failures’ on the relevant fault tree. It was assumed that a probability for the former basic event depended on the leak rate observed but was independent from the other basic event ‘failure to recover’ on the fault tree because any methods or data for treating dependencies between the actually observed failures and the successful operations (that is, the potential failures) were not available. Considering the fact that the first three events experienced a relatively small leak size of approximately 300 *l*/min to 500 *l*/min and the Palo Verde event had a leak rate of more than 900 *l*/min, a probability for the basic event ‘failure to timely isolate’ was assumed to be 0.1 for the first three events and that for the Palo Verde event was assumed to be 1.0. As well, the Mihama event involved the MSIV malfunction (failure to close completely) and the operators took the recovery action successfully at the failed valve. A failure probability of 1.0 was applied to the basic event ‘failure to timely isolate due to valve failures’ on the fault tree. As a probability for failure to recover, the preset value was employed for these events.

(6) Residual heat removal

Only the North Anna event in 1989 involved the RHR component failure (suction isolation valve failure to remain open). The probability of 1.0 was assigned to the RIIR system unavailability.

4. Analysis Results and Discussions

4.1 Branch Probabilities

The branch probabilities calculated are indicated in **Table IV.5** for individual events.

Table IV.5 Branch Probabilities on ASP Event Tree Models for SGTR

	Point Beach-1 (26/2/1975)	Surry-2 (15/9/1976)	Doel-2 (25/6/1979)	Prairie Island-1 (2/10/1979)	Ginna (25/1/1982)	Fort Calhoun (16/5/1984)	North Anna-1 (15/7/1987)	North Anna-1 (25/2/1989)	McGuire-1 (7/3/1989)	Mihama-2 (9/2/1991)	Palo Verde-2 (14/3/1993)
SGTR	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.1	1.0	1.0	1.0
Reactor Trip	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
AFW	9.6×10^{-5}	7.3×10^{-5}	7.3×10^{-5}	9.6×10^{-5}	1.6×10^{-6}	9.6×10^{-5}	7.3×10^{-5}	7.3×10^{-5}	7.3×10^{-5}	7.3×10^{-5}	9.6×10^{-5}
MFW	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}	0.34	0.34	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}
HPI	5.4×10^{-4}	4.6×10^{-4}	7.6×10^{-4}	5.4×10^{-4}	2.5×10^{-4}	2.5×10^{-4}	1.2×10^{-3}	1.2×10^{-3}	1.9×10^{-6}	5.4×10^{-4}	5.4×10^{-4}
Condensate Sys.	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.35
Primary Feed & Bleed Operation	1.45×10^{-2}	1.45×10^{-2}	1.48×10^{-2}	1.45×10^{-2}	1.0	1.43×10^{-2}	1.52×10^{-2}	1.52×10^{-2}	1.60×10^{-2}	1.0	N/A
RCS Cooldown < SG RV Setpoint	4.1×10^{-2}	4.1×10^{-4}	4.1×10^{-4}	4.1×10^{-4}	4.1×10^{-2}	4.1×10^{-2}	4.1×10^{-4}	4.1×10^{-4}	4.1×10^{-2}	4.0×10^{-2}	4.1×10^{-2}
Isolation of Ruptured SG	1.0×10^{-2}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-2}	1.0×10^{-3}	1.0×10^{-2}	1.0×10^{-3}	0.1	0.1
RCS Cooldown < RHR Pressure	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}
RHR (with RCS depressurized)	1.5×10^{-3}	7.7×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	7.7×10^{-3}	7.7×10^{-3}	0.12	1.5×10^{-3}	1.5×10^{-3}	1.8×10^{-3}
HPR (following successful F&B)	2.5×10^{-3}	2.3×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	1.9×10^{-3}	2.3×10^{-3}	2.3×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	N/A

As shown in this table, their respective branch probabilities for the two top events, the reactor trip and the RCS cooldown to the RHR pressure, were the same for all of the events because the generic fault tree models were employed and these functions were successful (that for the reactor trip was estimated at 1.0×10^{-5} and that for the RCS cooldown was at 4.0×10^{-3}). However, the branch probabilities obtained for AFW and HPI ranged from 1.6×10^{-6} to 9.6×10^{-5} and 1.9×10^{-6} to 1.2×10^{-3}, respectively, according to a variety of their success criteria.

As for the other top events, the specific branch probabilities were obtained for the events involving additional failure(s). In the following, a brief description for each of them is provided.

(1) Main feedwater (MFW)

The branch probability obtained for the two North Anna events with the MFW valves failed was 0.34 while that for the other events was estimated at 6.8×10^{-2} from the generic fault tree model using the preset values.

(2) Primary feed and bleed (F&B)

Since PORV was inoperable in the Ginna and Mihama events, the branch probability for F&B was set to 1.0. For the other events, the branch probabilities of approximately 1.5×10^{-3} were obtained from the plant-specific simplified fault trees with use of the preset values.

(3) RCS depressurization to below SG relief valve setpoint

For the events involving the delayed equalization of the primary and secondary pressures (at Point Beach, Ginna, Fort Calhoun, McGuire and Palo Verde), the branch probability was given as a product of the probability for failure to timely initiate the RCS depressurization (0.12) and that for failure to recover (0.34) since no hardware failure was observed. In respect of the Mihama event, the branch probability was set to the non-recovery probability of 4.0×10^{-2} because the RCS depressurization using PORVs actually failed. For the other events, the branch probability of 4.1×10^{-4} was obtained from the generic fault tree using the preset values.

(4) Isolation of ruptured SG

For the three events with the SG isolation delayed (at Point Beach, Fort Calhoun and North Anna in 1989), the branch probability of 0.01 was obtained by multiplying the probability assigned to ‘failure to timely isolate’ and the preset probability for ‘failure to recover’. For the Palo Verde event involving the retarded SG isolation and the Mihama event involving the MSIV failure, the non-recovery probability of 0.1 was set to the branch probability. For the other events, the branch probability was estimated at 1.0×10^{-3} using the preset values.

(5) Residual heat removal (RHR)

Since the North Anna event in 1989 involved the valve failure, the branch probability for RHR was set to the non-recovery probability of 0.12 (which is given for the failure recoverable in the required period with substantial operator burden(19)). Because HPR was available throughout the event, however, the branch probability of 2.3×10^{-3} for HPR was obtained using the preset values. For the other events, the branch probabilities for RHR and those for HPR were calculated at 1.5×10^{-3} to 7.7×10^{-3} and 1.9×10^{-3} to 2.5×10^{-3} by applying the preset values to the plant-specific fault trees.

4.2 Conditional Core Damage Probabilities

The event trees were quantified using the branch probabilities mentioned above. As seen from **Figure IV.7**, the CCDPs estimated in the study ranged from 2.8×10^{-4} to 4.2×10^{-2}. This range is primarily due to modeling of additional anomalies observed during the particular event. In the five events with a CCDP of 4×10^{-2} or higher (at Point Beach, Ginna, Fort Calhoun, McGuire and Palo Verde), for example, it took a relatively longer time to depressurize RCS and/or to equalize the primary and secondary pressures. To represent such conditions in the study, a high likelihood of failure to depressurize RCS to below SG relief valve setpoint was applied to these events. Consequently, high CCDPs were obtained and as shown in **Figure IV.8**, a sequence involving failure to depressurize RCS was identified as the dominant one for these events. This is depicted by the sequence No. 3 in the event trees displayed in Figures IV.4 and IV.5. Three of them, at Point Beach, Fort Calhoun and Palo Verde, also involved delayed isolation of the ruptured SG. Although a higher branch probability than the preset value was applied to the top event for the ruptured SG isolation, the CCDPs obtained for the sequences involving failure to isolate the ruptured SG were relatively low (approximately 5×10^{-5} for the Point Beach event, approximately 1×10^{-4} for the Fort Calhoun event and approximately 5×10^{-4} for the Palo Verde event) and had only small contributions to their respective total CCDPs because of a high likelihood of recovering from these sequences by the RCS cooldown and the subsequent RHR operation. In the Ginna event, a PORV anomaly (stuck open) was observed and therefore, the F&B operation was assumed to be disabled during the event. However, the CCDP of 1.1×10^{-7} estimated for the sequence involving the F&B failure was found to have a small contribution to the total CCDP because of a low probability of the AFW failure (1.6×10^{-6}). As for the Mihama event with a CCDP of 4.1×10^{-2}, on the other hand, the PORV failure to open had a large contribution to the CCDP because this failure would lead to failure to depressurize RCS which was involved in the dominant sequence while the contribution of the sequence involving failure of F&B to the total CCDP was found to be negligible (a CCDP for this sequence was estimated at approximately 5×10^{-6}). In addition, the Mihama event involved failure of MSIV for the ruptured SG to completely close but the sequences involving failure to isolate the ruptured SG had relatively small contributions (CCDPs for these sequences were estimated at 4×10^{-4} or lower) due to the same reason as above.

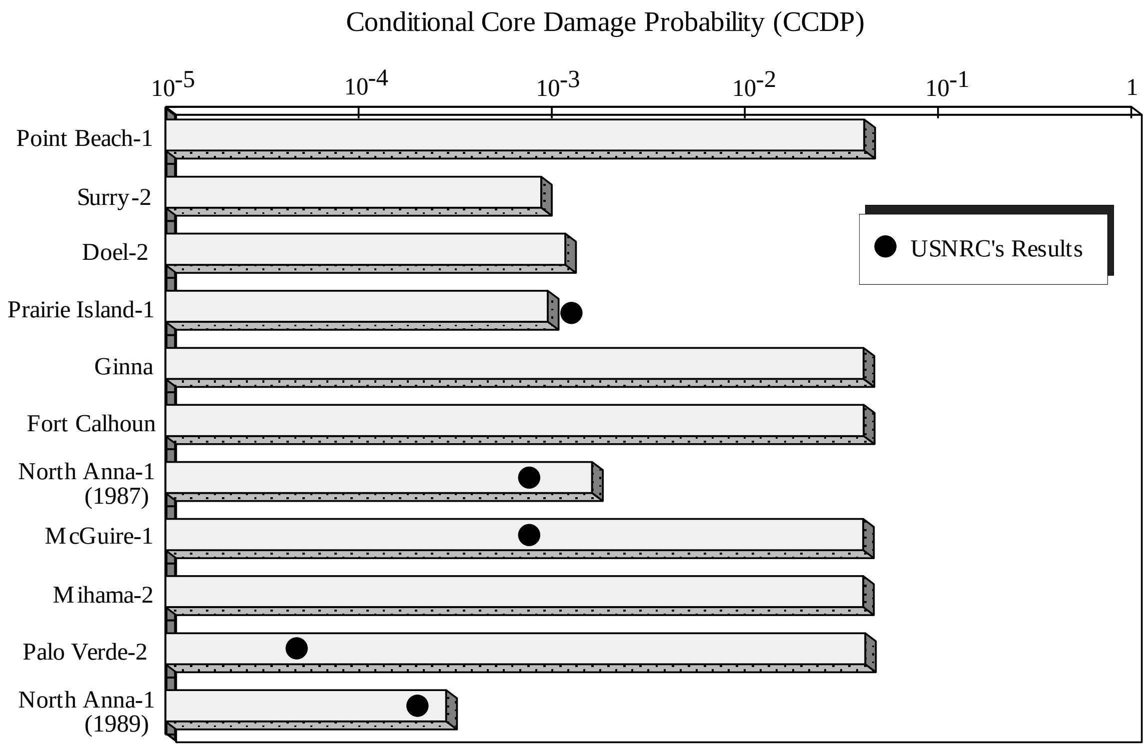

Figure IV.7 Conditional Core Damage Probability Calculated for Each SGTR Event

For all the remaining events except for the North Anna event in 1989, the CCDPs were estimated at approximately 1.0×10^{-3}, which were dominated by two sequences: one involving failure of HPI and the other involving failure to depressurize RCS to below SG relief valve setpoint. Of these four events, the North Anna event in 1987 involved two stuck open MFW relief valves that might cause the MFW unavailability higher but the CCDPs for the sequences involving the MFW failure were estimated at approximately 10^{-7}-10^{-8} due to a high reliability of AFW and consideration of F&B as an alternative method. Although the other three events (at Surry, Doel and Prairie Island) also involved additional anomalies, they did not contribute to the total CCDPs because they were not related to mitigation systems for SGTR.

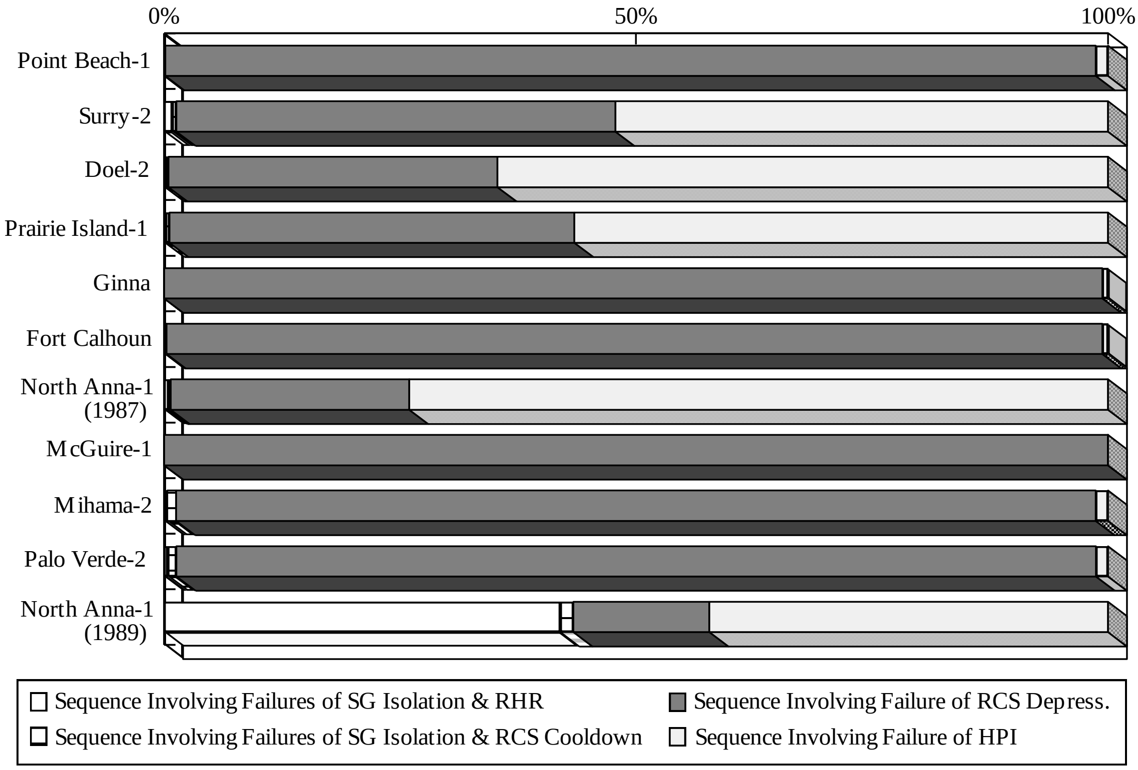

Figure IV.8 Contributions of Sequences for Each SGTR Event

As for the potential SGTR event at North Anna in 1989, the CCDP was estimated at 2.8×10^{-4}, which was dominated by two sequences: one involving the HPI failure and the other involving failure to isolate the ruptured SG and the RHR failure. Of the three malfunctions observed during this event, delayed isolation of the ruptured SG and failure of the RHR suction valve had a large contribution to the total CCDP but the other malfunction, the MFW regulating valve failure, did not contribute substantially because of a high diversity of its relevant function (AFW and F&B).

4.3 Discussion

(1) Effects of ASP models and data applied

As mentioned above, this study indicates that in general, SGTR would be a significant precursor with a CCDP of 1×10^{-4} or higher. Also, it is shown that the sequence involving failure to depressurize RCS to below SG relief valve setpoint has a large contribution to the total CCDPs for all the actual events. For the events with a CCDP of approximately 4×10^{-2}, in particular, this sequence contributes more than 95% of the total CCDP because the relatively high probabilities were applied to failure to depressurize RCS. In order to examine the adequacy of applying such high probabilities, additional calculation was carried out by using the preset value. As shown in **Figures IV.9 and IV.10**, CCDPs are reduced to approximately 1×10^{-3} but the sequence mentioned above has a relatively large contribution and remains dominant. In the ASP method, probabilities higher than the preset values should be used to address the anomalies observed and thus, a CCDP for SGTR involving failure to timely depressurize RCS would become higher than those obtained by using the preset value. This means that failure to timely depressurize RCS during SGTR may be a risk significant factor and recovery actions should be studied for this failure. However, there would be an ample time to take some alternative measures, such as a water addition to RWST and continuous HPI operation, for recovering from such sequences. Taking these measures into consideration might reduce the CCDPs.

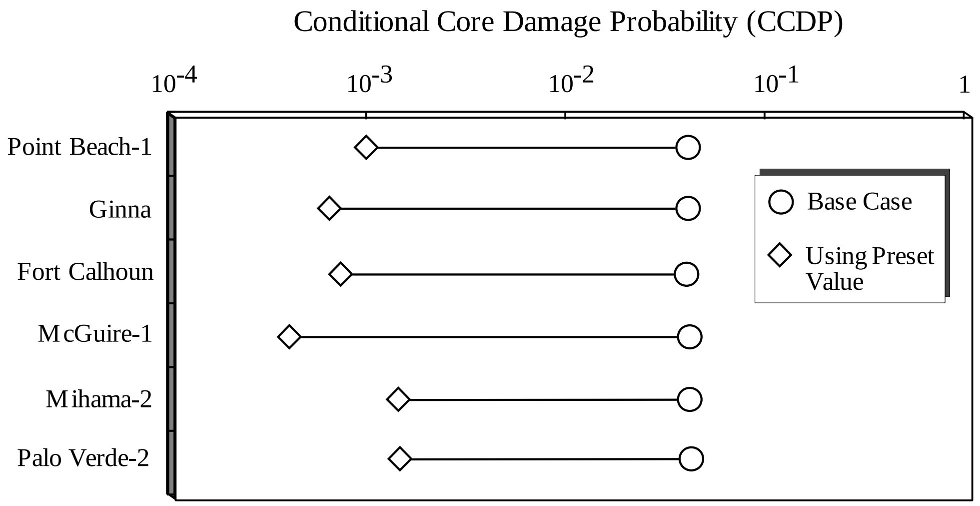

Figure IV.9 Comparison of CCDPs for Base Case and Using Preset Value

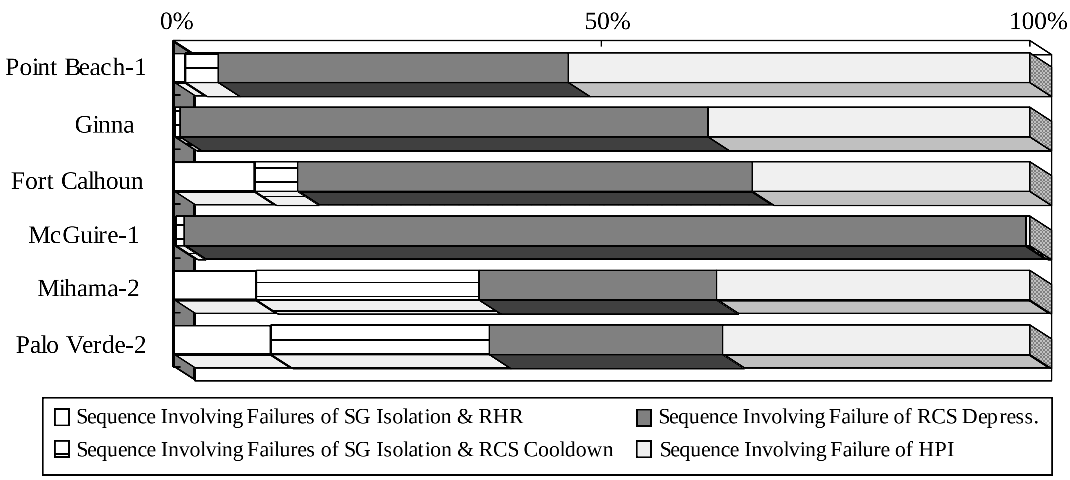

Figure IV.10 Sequence Contributions for Using Preset Value

On the other hand, a sequence involving the HPI failure is also identified dominant for the other six events. The event tree models employed in this study assumed that the HPI failure would lead to core damage regardless of success or failure of AFW/MFW. In a case that AFW/MFW is successful, however, the timely RCS depressurization could terminate the primary-to-secondary break flow and subsequently, the plant would be brought into cold shutdown using RHR. Considering such an alternative method might reduce the CCDPs for these events.

(2) Contributions of additional anomalies observed

In addition to failure to timely depressurize RCS, several anomalies related to mitigation systems observed were taken into account in this study. The failure modes of PORVs identified in the Ginna and Mihama events are different from each other, failure to reseat and failure to open, respectively. Both failure modes are considered important to safety from the separate points of view and their respective effects on SGTR are found to be remarkably different. In other words, the former would create another leak path recovered by closure of its block valve and might disable the F&B capability, which does not substantially contribute to the total CCDP. On the other hand, the latter was not restored during the event and might not only make F&B inoperable but also degrade the functionability of the RCS depressurization, which has a large contribution to the total CCDP. Also, as seen from the results for the two North Anna events, some failures related to mitigation systems such as the RHR suction valve failure would have relatively large contributions to the total CCDPs but some failures such as the MFW valve failure would not contribute. Consequently, their respective contributions of failures in mitigation systems might be dependent on the degree of diversity or redundancy in their relevant functions.

(3) Comparison with USNRC's analysis results

The five SGTR events at Prairie Island, North Anna in 1987 and 1989, McGuire and Palo Verde had been analyzed in the USNRC's ASP Program. In the following, the analysis results obtained in this study are compared with those obtained from the USNRC's analyses and the causes of differences are discussed.

For the events at Prairie Island and two events at North Anna in 1987 and 1989, the results from this study are comparable to those from the USNRC's analyses[9,14,16] as seen from Figure IV.7. Slight differences in CCDPs are mainly due to the different branch probabilities used for the HPI failure. As for the other events at McGuire and Palo Verde, the CCDPs obtained from the USNRC's analyses (7.7×10^{-4} and 4.7×10^{-5}, respectively) and this study are significantly different. This is mainly because this study applied a relatively high probability to failure to depressurize RCS in order to reflect the condition that the RCS pressure remained above the secondary pressure for a long time. In the USNRC's analysis for the McGuire event[16], this condition was not taken into account and all the systems/functions needed for mitigating SGTR were assumed successful. The USNRC's analysis for the Palo Verde event[20] developed the event specific model that incorporated three operators' actions – the RCS depressurization and subsequent low pressure injection (LPI), the SGTR identification, and the RWST refill - and assigned relatively high branch probabilities to the first and second actions (4.1×10^{-2} and 4.0×10^{-2}) to reflect the operators' delayed actions. However, this model assumed that any failure of these actions alone would not lead to core damage and thus, the sequences involving such a failure had relatively low CCDPs. For example, because the first action was assumed to be taken for the HPI failure, a core damage sequence involving failure of this action also involved the HPI failure, resulting in a relatively low CCDP (3.4×10^{-5}). As well, even if the SGTR identification would fail, the sequences involving successful HPI and AFW were assumed recoverable by the RWST refill. As a result, the CCDP for the Palo Verde event was lower, even compared with those from the USNRC's analyses for the other SGTR events where no anomalies in mitigation systems were observed. This implies that the results from the year-by-year analyses might heavily depend on the models and branch probabilities used and thus, comparison of the results would need applying a set of models and data consistently to the ASP analyses.

5. Summary

The ASP analyses were performed for ten actual SGTR events and one potential SGTR event, which occurred during 20 years, using the consistent ASP model. As a result, the CCDPs estimated in the study range from 2.8×10^{-4} to 4.2×10^{-2}. As well, it is shown that five of the ten actual SGTR events, where it took a longer time to identify SGTR or to depressurize RCS, and one actual event with PORVs failed have relatively high CCDPs (4×10^{-2} or higher) which are dominated by the sequence involving failure to depressurize RCS to below SG relief valve setpoint. The remaining four actual events are found to have relatively low CCDPs dominated by two sequences – one is the same as above and the other involving the HPI failure – and additional anomalies observed do not contribute to the CCDPs. The potential event, where component failures were identified in mitigation systems, also has relatively low CCDPs but these failures are found in the dominant sequences.

Through the analyses, it is clearly shown that the SGTR event which may confuse the operators and prevent them from taking their timely actions needed for mitigating the event might generally be a significant precursor. This implies the importance of providing adequate operating procedures for SGTR to the operators and/or of improving instrument devices to detect SGTR earlier and more accurately. As well, some of the additional anomalies observed are found to be significant and thus, alternative measures for recovering from such conditions should be examined and incorporated into the procedures. It should be noted, however, that the CCDPs may have uncertainties since they heavily depend on the models and failure probabilities applied. In particular, modeling and generic failure probabilities applied to the delayed RCS depressurization might be further examined considering actual situations observed in individual events.

IV.4 Quantitative Risk Trends with Newly Proposed Risk Indicators

Since the first report of the ASP Program was issued in 1982, eighteen reports(9-26) describing the results of the ASP Program covering the period from 1969 through 1998 had been published and identified approximately 600 precursors with their respective CCDPs of 10^{-6} or higher. However, the report was subsequently withdrawn from distribution because of the USNRC's heightened awareness of the release of sensitive information to the public following the terrorist attacks on September 11, 2001(27).

The ASP analysis has been carried out on a yearly basis and the precursor events has been accumulated. As the results from the ASP Program include valuable information that could be useful for obtaining and characterizing risk significant insights and for monitoring risk trends in nuclear power industry, trending analyses of the ASP results has been recognized as one of indications of industry risk in the United States. Nevertheless, there are only a few attempts to determine and develop the trends using the ASP results(42, 43) and of them, the occurrence rate of precursors has been employed as an industry-level indicator(27) for monitoring industry risk trends.

In order to more effectively and widely use the ASP results for discussing and/or monitoring industry risk trends, this study proposes new quantitative risk indicators in the industry level, that is, occurrence frequency of precursor events and annual core damage probability deriving from the results of the ASP analyses, and examines the trends in core damage risk at nuclear power plants using these two indicators.

1. Survey of Results from ASP Program

In the ASP Program, approximately 600 precursors had been identified in the NUREG/CR series covering the years 1969 through 1998: 394 events at PWRs and 213 events at BWRs. As discussed in Refs. (4), (5) and (43), trends can be qualitatively discussed from various points of view, such as initiating events, unavailable systems and dominant sequences, by surveying the analysis results provided in NUREG/CR reports. To provide a basis for discussions on quantitative risk trends with the proposed indicators in the subsequent subsections, at first, it was examined whether any trends exist in the number and CCDPs of precursors, both of which are closely related to the proposed indicators, by surveying the results from the ASP Program [4,5]. In the following, the trends observed are discussed taking into account the model changes.

1.1 Trends in Number of Precursors

Figure IV.11 shows the historical records on the reactor years and the number of precursors identified in the ASP Program by calendar year. From this figure, the number of reactor years has increased over the years 1969 to 1991 and remained almost constant after that. It can be also seen that during the period 1969 to 1981, the number of precursors has gradually increased as the reactor years have increased but after 1984, the number of precursors has decreased even though the reactor years have increased. In the years after 1993 with the reactor years being almost constant, as well, the number of precursors has significantly decreased. It should be noted that the number of precursors significantly increased in the years 1982 to 1983. The analyses of the events in these two years were performed in the middle of 1990s primarily for the historical purposes to complete the ASP analysis and documentation for all events and there were some differences in the availability of information and the modeling assumptions used in the 1982-83 event analyses compared with those in the analyses for the other years[23]. For example, the LER reporting requirements for the 1982-83 events were different from those for 1984 and later. Although plant trip information is important from the ASP perspective because one of the categories of events analyzed in the ASP Program includes plant trips with safety systems degraded, LERs prior to 1984 were not required to link plant trip information to reportable events. This is the most important difference, resulting in assumptions made about the relationship between a trip and potentially unavailable equipment. Because the link between trips and events was not described in LERs, it was often impossible to determine whether the component was actually unavailable during the trip or whether it was demanded during the trip. In order to avoid missing any important precursors, as a result, it was conservatively assumed that system unavailabilities reported in LERs could have occurred concurrently with plant trips, leading to the increase in the number of precursors. In addition, the actual likelihood of failing recovery from an event at a particular plant during the years 1982 and 1983 is difficult to assess and may vary substantially from the values used in the analyses for the later years.

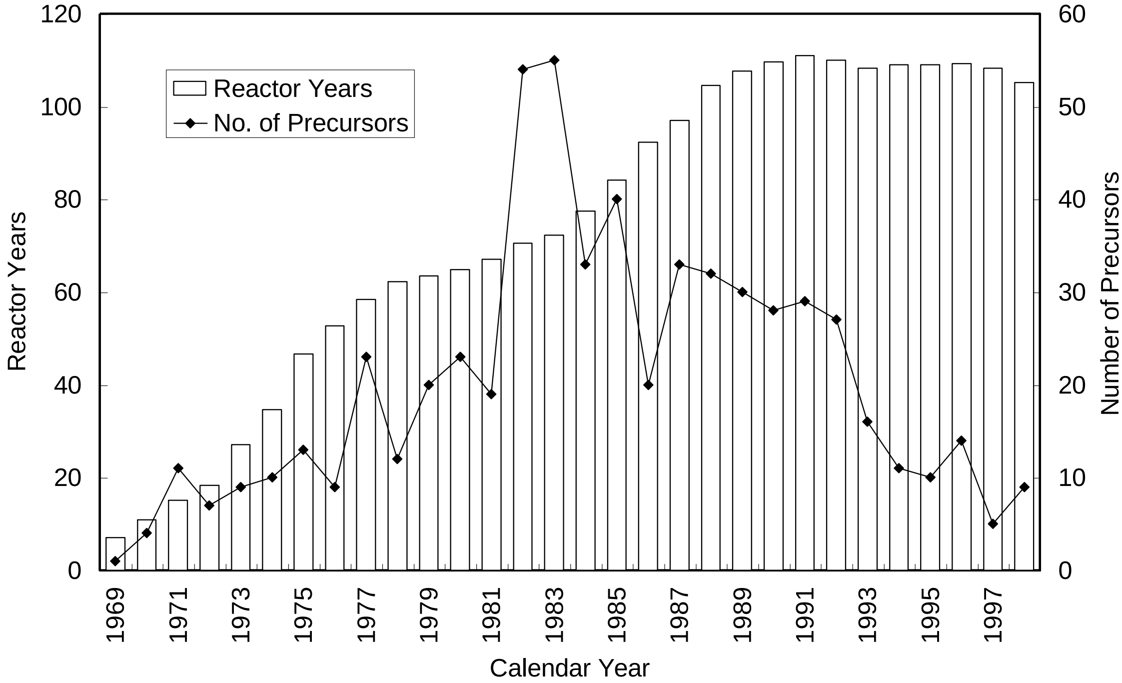

Figure IV.11 Reactor Years and Number of Precursors by Calendar Year

Figure IV.12 compares the annual distributions of precursors at PWRs and BWRs, separately. It can be seen from this figure that their respective numbers and historical trends are slightly different from each other. Although the number of precursors changes drastically before and after the period 1982 to 1983 at both PWRs and BWRs, the number at PWRs is larger than that at BWRs over the years. In particular, this trend can be clearly seen after 1986. At PWRs, twenty or more precursors are identified during the years 1982 to 1992, excluding three years (1984, 1986 and 1990), and the number of precursors in these eight years is up to 195 that is equivalent to about 50% of the total number (394 events) of the precursors over the years 1969 through 1998. On the other hand, at BWRs, there are only five years in which ten or more precursors are identified (1982 to 1985 and 1987) but the number of the precursors in these years is equivalent to almost a half of the total number of BWR precursors (95 of 213 events). As well, for the PWR precursors, an increasing trend is observed during the years 1969 to 1983 and a decreasing trend exists after that. On the other hand, for the BWR precursors, such an increasing trend does not appear while a more remarkable decreasing trend is observed after 1982.

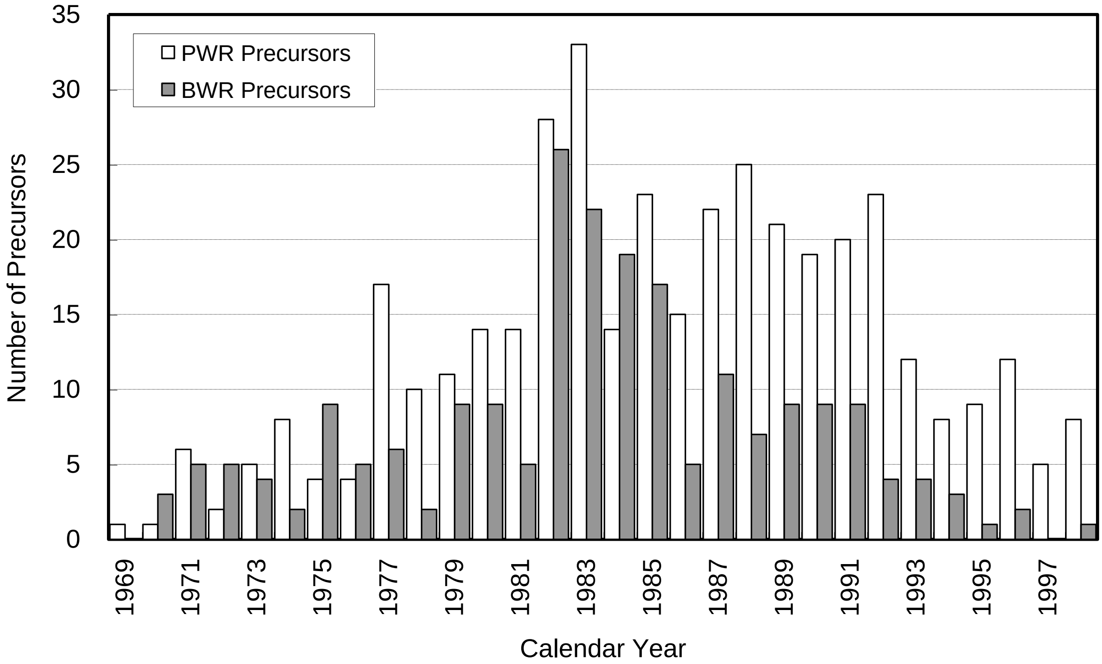

Figure IV.12 Annual Distribution of Precursors at PWRs and BWRs

The precursors identified in the ASP Program can be classified into two general categories in terms of initiating events: precursors involving initiator where an initiating event actually occurred and those involving unavailabilities where safety-related system(s) had been rendered unavailable or experienced degraded conditions with no occurrence of any initiating event. **Figure IV.13** shows the annual distributions of precursors by these two categories. Comparing these two categories, the number of precursors involving initiator is larger than that of precursors involving unavailabilities until 1987, in particular significantly during the years 1982 to 1987. However, in later years, the numbers of these two categories have been comparable. A similar trend is observed for the PWR precursors but for the BWR precursors, the number of precursors involving initiator is larger than that of precursors involving unavailabilities over the years 1969 through 1998. As seen from this figure, in addition, an increasing trend is observed during the years 1969 to 1983 and a decreasing trend exists after that for the precursors involving initiator. Such an increasing trend during the years 1969 to 1983 can be seen at PWRs but not observed for the BWR precursors involving initiator. The decreasing trend after 1983 exists more remarkably at BWRs compared with PWRs. On the other hand, the precursors involving unavailabilities have a different trend. During the years 1974 to 1987 excluding 1982, the number of such precursors is almost constant, that is, less than ten, but after that, a slight decreasing trend is observed. While a similar trend can be seen for the PWR precursors involving unavailabilities, the number of such BWR precursors is almost constant, that is, less than five, over the years except 1982.

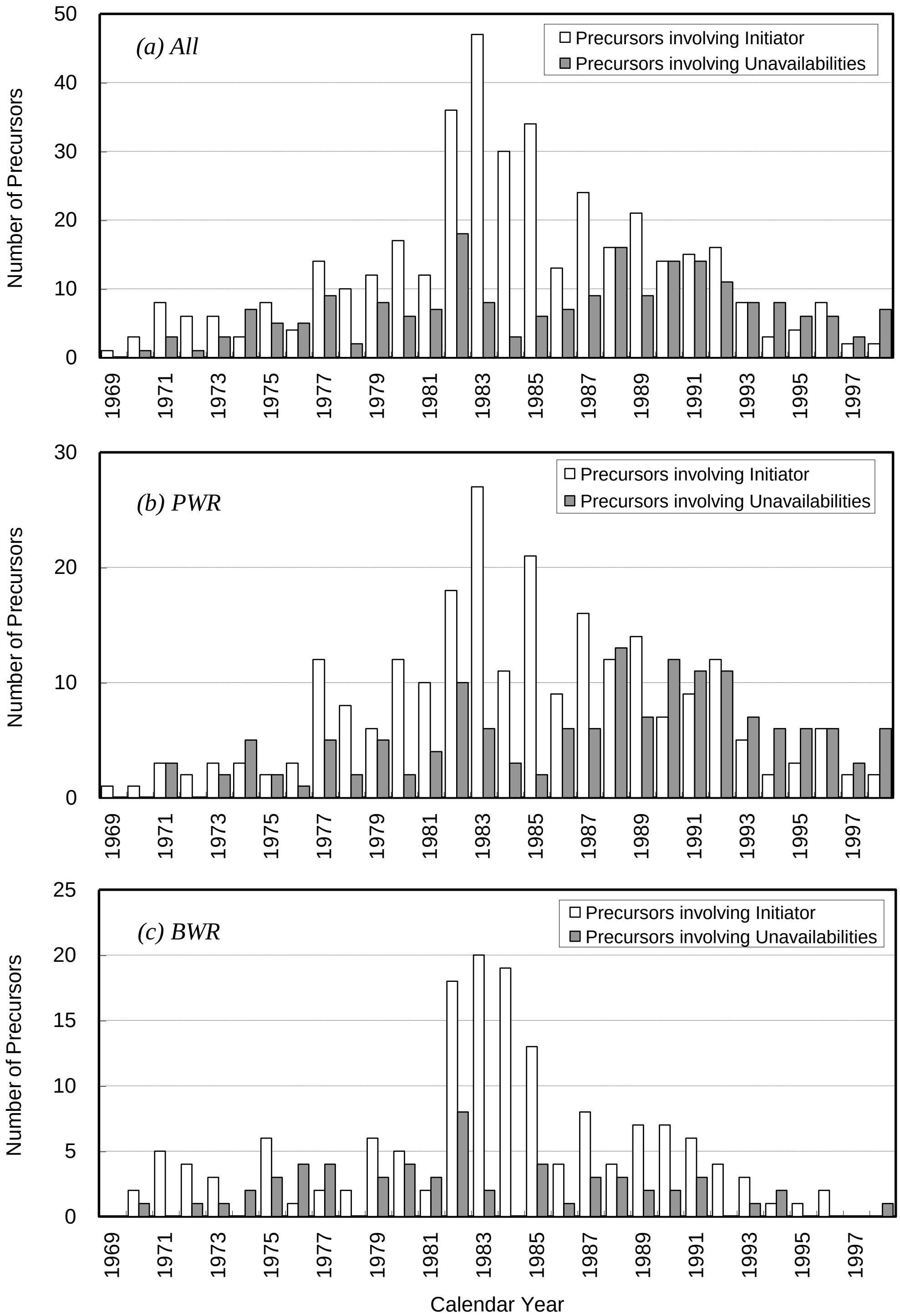

Figure IV.13 Annual Distributions of Precursors by Categories

1.2 Trends in Conditional Core Damage Probabilities of Precursors

Figure IV.14 displays the number of precursors at PWRs and BWRs, separately, according to probability bin (decades of CCDPs: 10^{-6} to 10^{-5}, 10^{-5} to 10^{-4}, 10^{-4} to 10^{-3}, 10^{-3} to 10^{-2}, and higher than 10^{-2}) for the five time periods corresponding to the times of ASP models changed. From this figure, it can be seen that the distributions of precursors from the CCDP point of view vary according to the individual periods. During the period 1969 to 1981, more than a half of precursors have a CCDP of 10^{-4} or higher, approximately 30% of which are in the highest two CCDP bins. This trend can be seen more remarkably for the PWR precursors. At BWRs, however, more than 60% of the precursors are in the lowest two CCDP bins. These trends mean that the PWR precursors in this period have a relatively higher CCDP compared with those at BWRs. Seven of eight precursors with a CCDP of 10^{-2} or higher (six at PWRs and one at BWR) are identified in this period. In the period 1982 to 1983, more than a half of precursors are in the lowest CCDP bin. However, different trends are observed for the PWR and BWR precursors. At PWRs, about 60% have a CCDP of 10^{-6} to 10^{-5} while the BWR precursors consist of about 40% in the lowest CCDP bin, about 40 % in the range 10^{-5} to 10^{-4} and about 20% in the range 10^{-4} to 10^{-3}. Thus, the precursors at BWRs have a slightly higher than those at PWRs in this period contrary to the previous period. For the period 1984 to 1987, the precursors are distributed almost evenly in three CCDP bins, 10^{-6} to 10^{-3}. A similar trend can be seen for the PWR precursors and the BWR precursors even with a slight difference between them. Similar to the previous period, the precursors during the period 1988 to 1993 are almost evenly in three CCDP bins. While this trend can be seen for the PWR precursors more clearly, almost a half of the BWR precursors are in the range 10^{-5} to 10^{-4}. The number of occurrences at PWRs is significantly larger than that at BWRs and hence, the trend observed for PWRs has a large contribution to that for all precursors. In the latest period 1994 to 1998, more than a half of precursors have a CCDP of 10^{-5} to 10^{-4} and about one fourth are in the lowest CCDP bin. A similar distribution is observed for the PWR precursors. While the distribution of BWR precursors is different from that of PWRs, however, their number is significantly small compared with PWRs and hence, their distribution has almost no effect on the overall trend in distribution of precursors.

As mentioned above, significant differences can be observed in the distributions of precursors by CCDP bins for five time periods. These differences seem due to mainly the ASP model changes. For example, a relatively large number of precursors with a high CCDP have been identified for the period 1969 to 1981 due to a set of generic or standardized event trees being applied in common to all PWR events or all BWR events and no recovery action being considered in the analyses. For the other four periods, on the other hand, the contributions of such precursors have been lowered because the event tree models improved by reflecting the plant design difference in 1985 were employed in the analyses of 1984-87 events, recovery actions were incorporated into the models used in the analyses of 1982-83 events as well as 1988-93 events, and the fault trees representing plant-specific design and procedures were applied in the analyses of 1994-98 events.

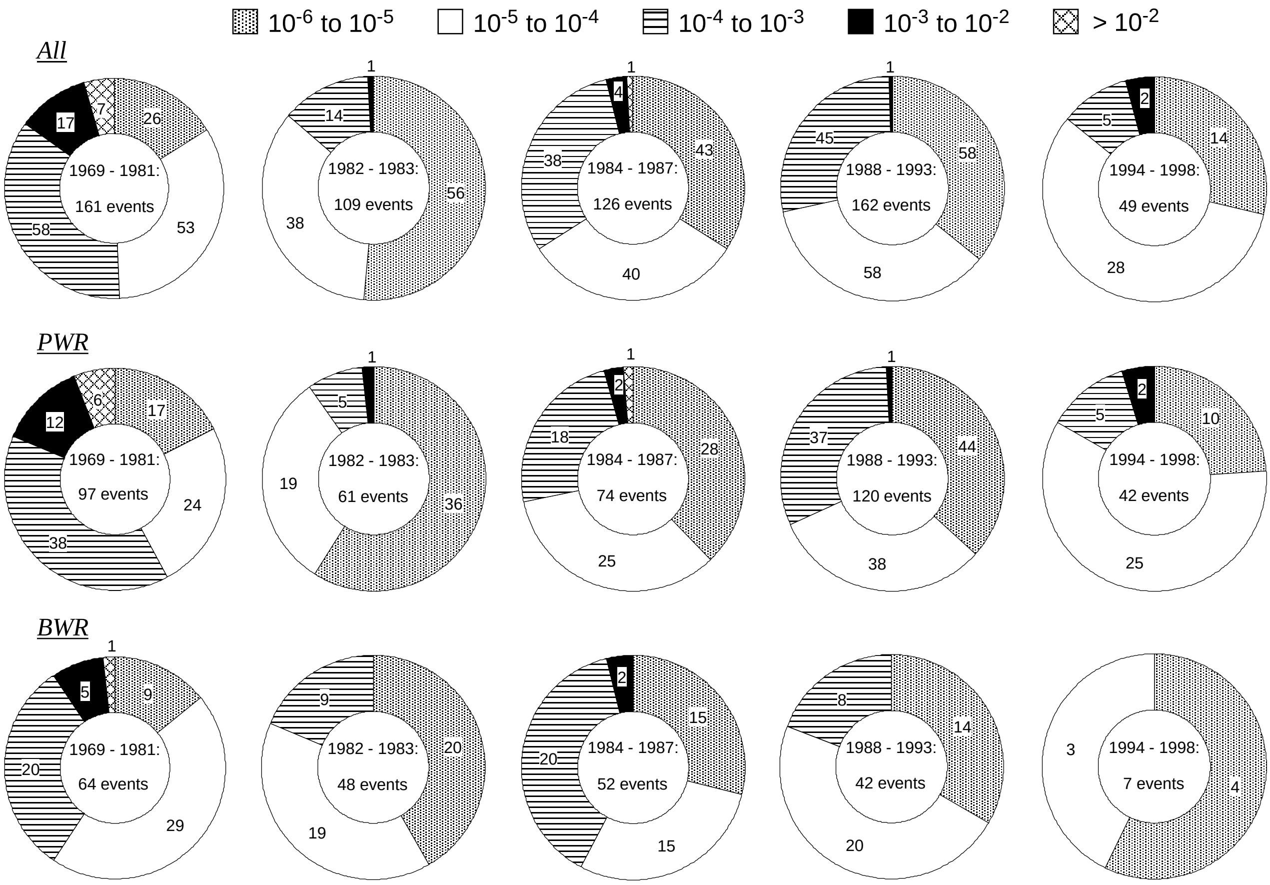

Figure IV.14 Distributions of Precursors by CCDP Bins

2. Definition of New Risk Indicators

The USNRC has been using the occurrence rate of precursors that is defined by dividing the number of precursors by the total reactor years for each calendar year as one of the indicators to assess the industry performance(27). As discussed in the previous subsection, in addition, the trends of precursors can be examined from various points of view such as the number of precursors identified and their CCDPs. These trends are related to core damage risk at nuclear power plants but provide only qualitative views. In order to examine quantitatively the trends in nuclear power plant risk from different points of view, this study newly proposes two quantitative risk indicators, occurrence frequency of precursors and annual core damage probability (annual CDP), and applies these indicators to evaluate the historical trends based on the ASP analysis results.

The occurrence frequency of precursors depicts the trends on how the likelihood of risk significant events have changed at PWR plants and BWR plants, respectively, as well as overall nuclear power industry. This indicator, occurrence frequency, is defined, as shown in Equation (1), by dividing the integrated number of precursors by the integrated reactor years.

$$F_N(t \le t_n) = \frac{N(t \le t_n)}{\sum_{m=1}^{M} \Delta t_m} \tag{1}$$

where,

$F_N(t \le t_n)$: occurrence frequency when the n^{th} precursor event occurred

t: date (month/year)

t_n: date when the n^{th} precursor event occurred

Δt_m: operating period (years) of plant m

$N(t \le t_n)$: number of precursor events prior to t_n

M: number of operating plants when the n^{th} precursor event occurred

The operating period of a plant is defined as a period from the date when its commercial operation commenced to the date of event occurrence.

On the other hand, the annual CDP indicates the trends on how the potential risk has changed at PWR plants and BWR plants as well as in overall nuclear power industry. The indicator, annual CDP is defined, as shown in Equation (2), by dividing the sum of conditional core damage probabilities of the precursors identified by the number of operating reactors for each calendar year.

$$P_a = \frac{\sum_{i=1}^{N} P_i}{M} \tag{2}$$

where,

P_a: annual core damage probability per reactor

P_i: conditional core damage probability estimated for precursor event i

N: number of precursor events for each calendar year

M: number of operating reactors for each calendar year

3. Trends Observed by New Risk Indicators

Based on the results from the ASP Program, this study determines whether any risk related trends are observed at PWRs and BWRs, respectively by applying the risk indicators proposed in the previous subsection.

3.1 Occurrence Frequency of Precursors

As mentioned above, the precursors identified in the ASP Program can be classified into two general categories: precursors involving initiator and those involving unavailabilities. In the following, discussed are the results obtained by applying one of the risk indicators proposed, that is, occurrence frequency of precursor events, for these two categories.

Using the equation (1) defined in the previous subsection, their respective occurrence frequencies were estimated for the precursors involving initiator and those involving unavailabilities at PWRs and BWRs. The estimated occurrence frequencies are shown in **Figures IV.15**, **16**, and **17** for all precursors at both PWRs and BWRs (607 events), the PWR precursors (394 events), and the BWR precursors (213 events), respectively.

From Figure IV.15, it can be seen that the occurrence frequency of precursors is almost constant at around 0.26-0.30 events per reactor year (EPRY) before 1982 (approximately 560 reactor years) and a remarkable increasing trend exists during the years 1982 and 1983 (approximately 560 to 700 reactor years). This increasing trend is because a significant number of precursors have been identified in 1982 and 1983 compared with those in the previous years. The analysis of events in these two years were carried out in the middle-1990s by applying the ASP models different from those used in the previous years and a couple of related events were regarded as one event in the light of new reporting requirements with promulgation of the LER rule in 1984. While the increasing trend continues till the year 1985 (approximately 870 reactor years) because a relatively large number of precursors were identified in the years 1984 and 1985, its gradient is smaller than the previous years. At this time, the occurrence frequency reaches its highest value, 0.40 EPRY. After 1985, the occurrence frequency decreases gradually and linearly to 0.28 EPRY in 1998. These trends can be seen for the 397 precursors involving initiator even though the occurrence frequency is lower. However, the occurrence frequency of the 210 precursors involving unavailabilities remains at almost constant value, 0.10 EPRY, over the years 1975 through 1998 while in the years 1982 and 1983 a slight increasing trend is observed.

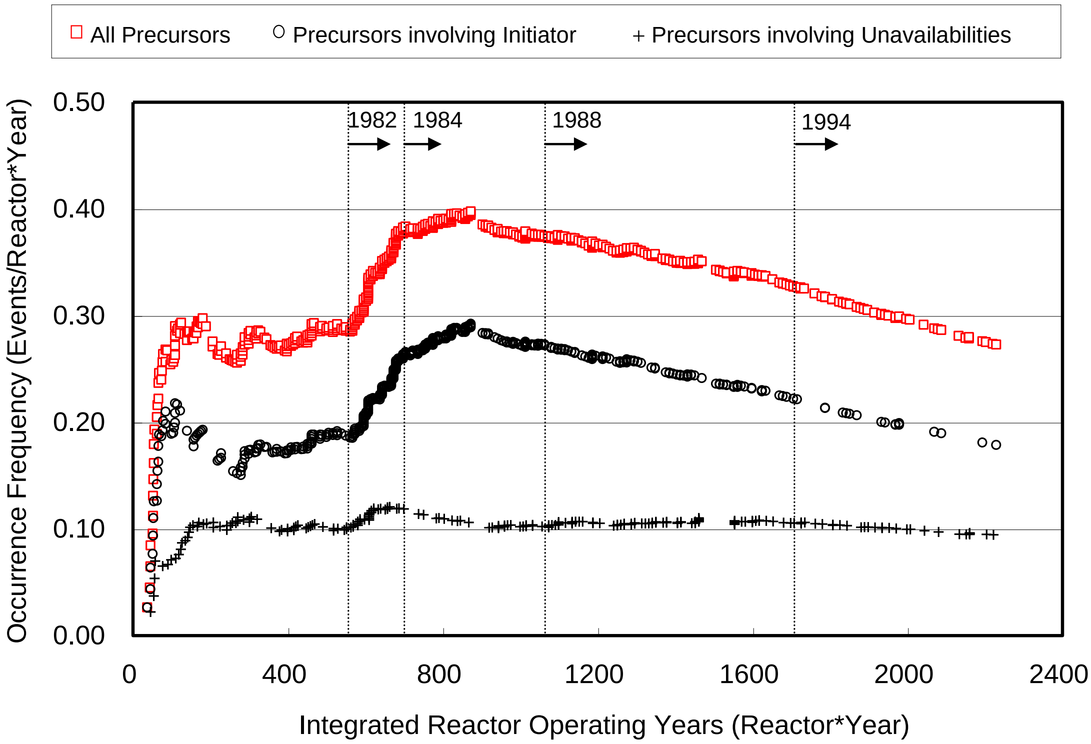

Figure IV.15 Occurrence Frequency of All Precursors

For the PWR precursors, as shown in Figure IV.16, the occurrence frequency increases up to 0.34 EPRY until 1973 (approximately 50 reactor years) and after that, it decreases significantly and reaches the value of 0.24 EPRY in 1976 (approximately 130 reactor years). In 1977, the occurrence frequency increases again up to 0.30 EPRY and remains at around this value for the subsequent years 1978 to 1981. During the years 1982 and 1983, a remarkable increasing trend is observed and subsequently, the occurrence frequency remains at an almost constant rate, 0.38 EPRY in the years 1984 and 1985. After that, a slight decreasing trend exists over the years and the occurrence frequency reaches 0.28 EPRY in 1998, but its gradient is smaller compared with that for all precursors shown in Figure IV.15. A similar trend is observed for the 242 PWR precursors involving initiator but the decreasing trend after 1985 (approximately 520 reactor years) is more remarkable compared with that for all of the PWR precursors. On the other hand, for the 152 PWR precursors involving unavailabilities, the occurrence frequency is in the range of 0.10 to 0.12 EPRY over the years after 1975 (approximately 100 reactor years). Compared with the trend observed for all precursors involving unavailabilities, a slight difference can be seen in the trends before 1976 and after 1988. While the occurrence frequency of the PWR precursors increases up to 0.15 EPRY in 1974 (approximately 70 reactor years) and decreases to 0.10 EPRY in 1976 (approximately 130 reactor years), that of all precursors gradually increases to 0.10 EPRY in this period. As well, the occurrence frequency of the PWR precursors increases slightly from 0.10 to 0.12 EPRY during the years 1988 through 1993 but that of all precursors is almost constant in this period.

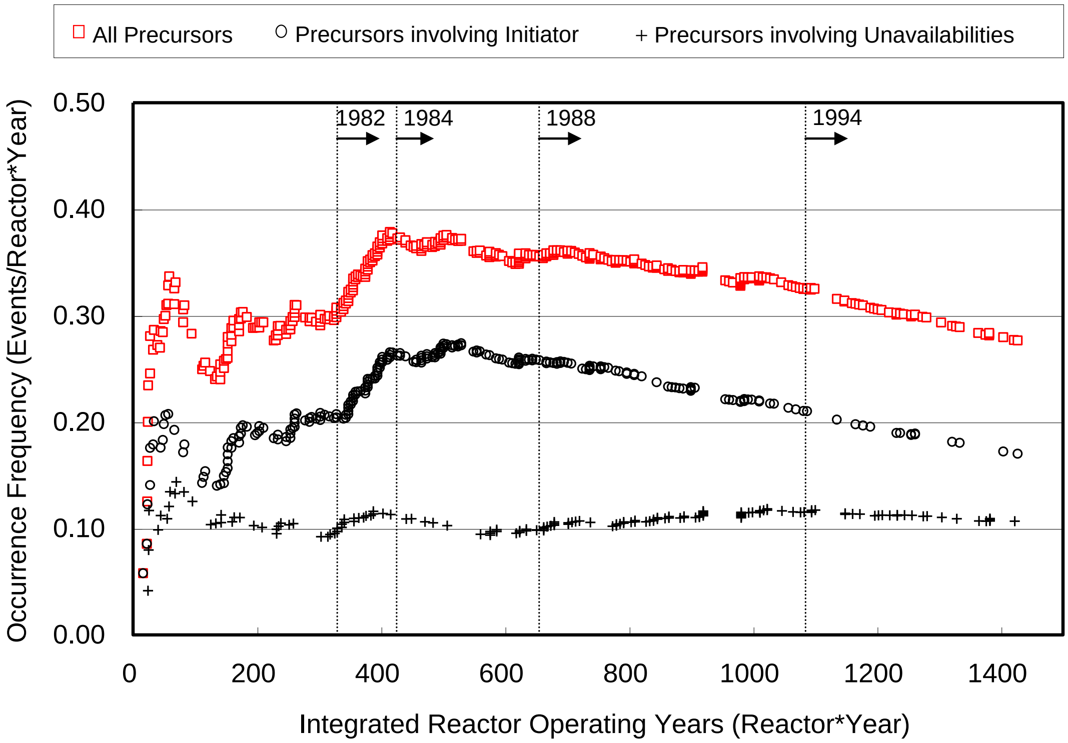

Figure IV.16 Occurrence Frequency of PWR Precursors

For the BWR precursors, as seen from Figure IV.17, the occurrence frequency of the precursors increases up to 0.30 EPRY until 1975 (approximately 100 reactor years) and in the subsequent years 1976 to 1981 (approximately 120 to 240 reactor years), slightly decreases and remains at around 0.25-0.27 EPRY. During the years 1982 through 1985 (approximately 240 to 340 reactor years), a remarkable increasing trend is observed and in 1985, the frequency reaches the highest value of 0.43 EPRY. This increasing trend is due to mainly the relatively large number of precursors identified in four years 1982 to 1985 compared with those in other years. The occurrence frequency gradually decreases over the years after 1985 and its gradient is larger than those of all precursors and the PWR precursors. The quite similar trend is observed for the 155 precursors involving initiators. However, the occurrence frequency of the 58 precursors involving unavailabilities indicates different trends. It gradually increases to 0.12 EPRY until 1977 (approximately 140 reactor years) and remains almost constant subsequently until 1982 (approximately 260 reactor years). In 1982, it rapidly rises to 0.14 EPRY due to a larger number of precursors being identified in

1982 but after that, a decreasing trend is observed because of a smaller number of precursors identified in 1983 compared with those in other years. In particular, no precursor involving unavailabilities was identified in 1984.

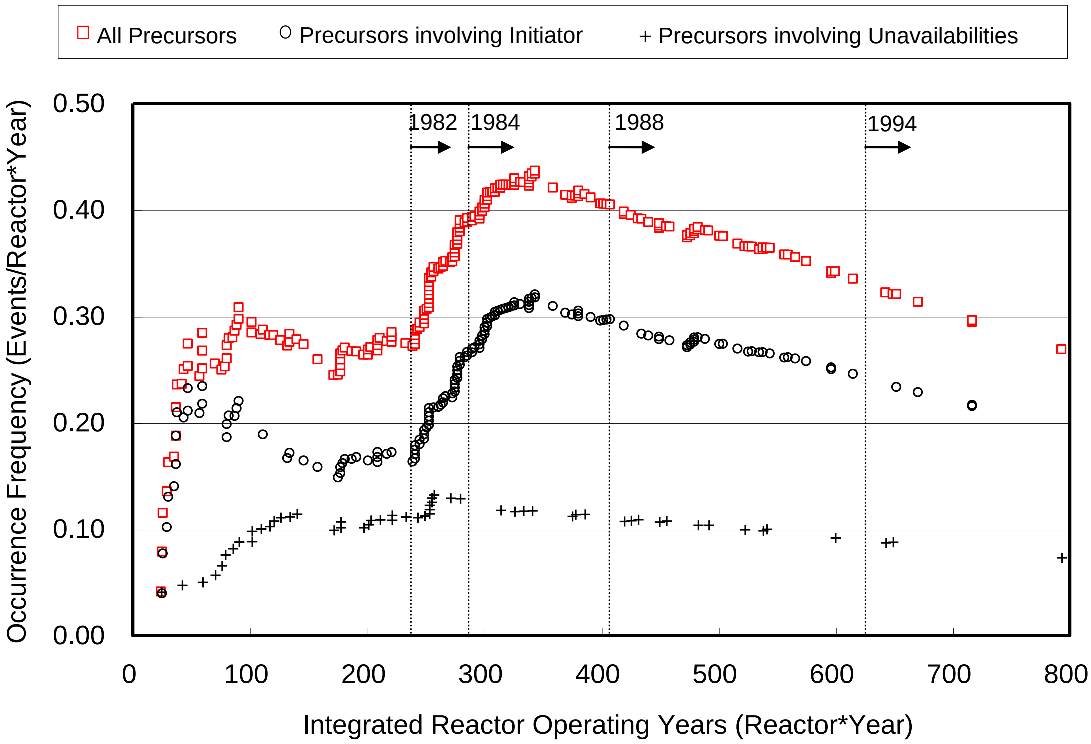

Figure IV.17 Occurrence Frequency of BWR Precursors

Comparing the occurrence frequencies of the PWR and BWR precursors, slightly different trends are observed. In particular, the frequency of BWR precursors has a more remarkable increasing trend during the years 1982 to 1985. While the frequency increases from 0.28 to 0.43 EPRY in this period at BWRs, that increases from 0.30 to 0.37 EPRY during the first two years and after that remains almost constant at PWRs. As well, the decreasing trends over the years after 1986 are different from each other. The frequency at BWRs decreases from 0.43 to 0.27 EPRY but that at PWRs decreases from 0.37 to 0.28 in the same period, indicating a larger decreasing rate at BWRs. During the period 1969 to 1981, the frequency of the PWR precursors is higher than that of the BWR ones even though similar trends are observed at PWRs and BWRs. Such differences can be seen for the precursors involving initiator. In addition, there is a difference in the frequencies of precursors involving unavailabilities. The frequency at PWRs decreases slightly in the years 1984 to 1988, increases slightly in the subsequent years until 1994 and remains almost constant after that while the frequency at BWRs decreases gradually over the years after 1983. Since much more precursors were identified at PWRs than that at BWRs, the trend observed for PWRs has a larger contribution to that for all precursors.

The results above mentioned show that the occurrence frequency of precursors totally has been decreasing over the years but contributions of precursors involving unavailabilities have been increasing. While the precursors involving initiator could be easily recognized as a result of reactor trip, the precursors involving unavailabilities are latent/potential failure and/or degradation of the safety-related equipment that would not be identified during power operation and would be found in the testing or inspection activities. The core damage risk induced by such precursors might increase if additional malfunction(s) would occur and/or potential/latent failure would not be identified for a longer time. Thus, more efforts are needed to lower the occurrence frequency of such precursors by, for instance, earlier detection of potential/latent failure.

3.2 Annual Core Damage Probability

The annual CDP is obtained by dividing the total CCDPs by the reactor years for each calendar year as shown in Equation (2) defined in the previous subsection. **Figure IV.18** shows the annual CDP by calendar years for all precursors, and **Figure IV.19** provides the comparison of annual CDPs for the PWR and BWR precursors.

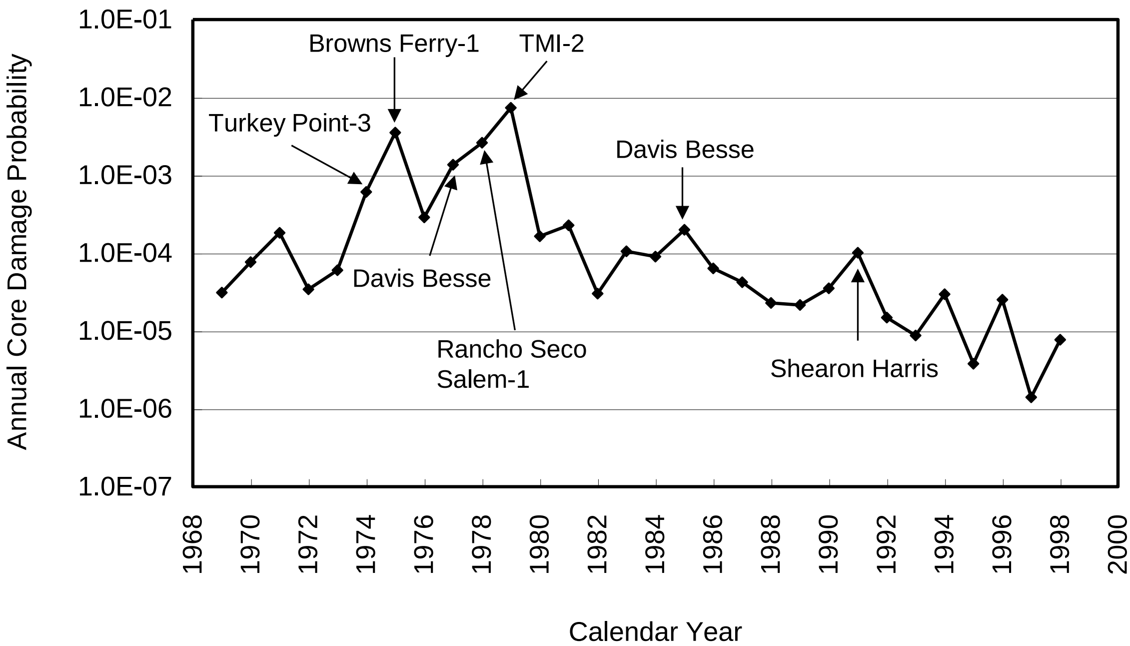

Figure IV.18 Annual Core Damage Probabilities for All Precursors

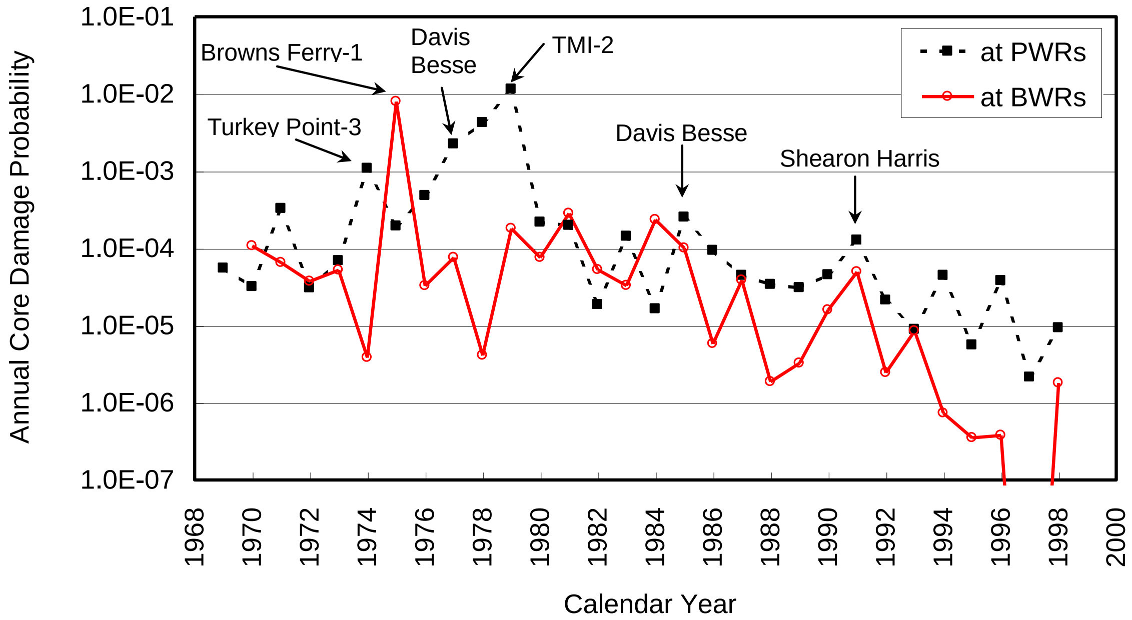

Figure IV.19 Annual Core Damage Probabilities for PWR and BWR Precursors

For all precursors, it can be seen from Figure IV.18 that the annual CDP is relatively high until 1979 because generic models were applied in common to all PWR events or all BWR events and any recovery actions were not considered in the analysis, resulting in conservative evaluations. In particular, a high annual CDP, that is, in the range 10^{-3} to 10^{-2} is observed during the years 1975 to 1979 due to four events with a high CCDP: fire at Browns Ferry Unit 1 (BWR) in 1975 (1.5×10^{-1}), and loss of main and auxiliary feedwater at three B&W PWRs (Davis Besse in 1977 with 7.0×10^{-2}, Rancho Seco in 1978 with 1.4×10^{-1}, and Three Mile Island Unit 2 (TMI-2) in 1979 with 4.6×10^{-1}) as listed in **Table IV.6**. On the other hand, the annual CDP has changed slightly in the range of 10^{-5} to 10^{-4} after 1982 and indicates a decreasing trend over the years. It seems that this drastic change of annual CDP might has been due to lessons learned from the core melt accident at the TMI-2 in 1979 being implemented into the plant design and operations and the improved ASP models being used in the analyses after 1984. In particular, the CCDPs estimated for the events after 1992 are lower than for the equivalent events in earlier years because practically available equipment or procedures were added to the basic models previously used. A relatively higher annual CDP is observed in 1985 and 1991 due to loss of feedwater at Davis Besse (1.1×10^{-2}) and potential unavailability of high pressure injection at Shearon Harris (6.3×10^{-3}), respectively. As shown in Figure IV.19, the annual CDP at PWR is larger than that at BWRs over the years excluding 1975 and 1984. As mentioned above, in 1975, the fire event at Browns Ferry Unit 1 has a large contribution to the annual CDP not only for the BWR precursors but also for all precursors. As well, in 1984, 13 of the 19 BWR precursors have a CCDP of 10^{-4} or higher, resulting in a higher annual CDP while at PWRs, 10 of the 14 precursors have a CCDP of lower than 10^{-4}. Therefore, the trend of annual CDP for all precursors is quite similar to that for the PWR precursors over the years except these two years (1975 and 1984). At BWRs, the annual CDP is approximately 10^{-4} or lower over the years 1970 through 1993 excluding 1975 and after 1994, about 10^{-6} or lower. These trends in annual CDPs imply that in general, PWRs have a relatively higher possibility of leading to core damage compared with BWRs, equivalent to the PSA results in NUREG-1150[30].

Table IV.6 Precursors with CCDP of 10^{-2} or Higher

Plant	*Event Date*	*Event Summary*	*CCDP*
PWR			
Turkey Point Unit 3	May 8, 1974	failure of all three auxiliary feedwater (AFW) pumps in test during power operation	1.6×10^{-2}
Millstone Unit 2	August 3, 1976	inadvertent load shedding on safety-related busses	1.4×10^{-2}
Davis Besse	September 24, 1977	loss of cooling water to steam generators (SG), resulting in SG dryout and a subsequent stuck open pressurizer relief valve	7.0×10^{-2}
Rancho Seco	April 23, 1978	loss of feedwater and subsequent failure of AFW injection valves to open	1.4×10^{-1}
Salem Unit 1	December 11, 1978	loss of a vital instrument bus causing failure of AFW pump to start	1.4×10^{-2}
Three Mile Island Unit 2	March 28, 1979	loss of feedwater, stuck open pressurizer relief valve and loss of high pressure injection (HPI) resulting in core damage	4.6×10^{-1}
Davis Besse	June 9, 1985	loss of feedwater and subsequent stuck open pressurizer relief valve	1.1×10^{-2}
BWR			
Browns Ferry Unit 1	March 22, 1975	extensive cable fire involving loss of high pressure injection	1.5×10^{-1}

Figure IV.20 provides the annual CDP trends of all precursors, the PWR precursors and the BWR precursors by precursor category. As seen from Figure IV.20(a), the annual CDP of precursors involving initiator is generally higher than that of the precursors involving unavailabilities. In particular, this trend can be seen remarkably during two periods, 1975 to 1980 and 1983 to 1986. During the former period, the annual CDP of precursors involving initiator is approximately 10^{-3} or higher while that of precursors involving unavailabilities is in the range 10^{-5} to 10^{-4}. The major reason of such significant differences in the annual CDPs is that one or more precursors involving initiator with a high CCDP (10^{-2} or higher), listed in Table IV.6, took place in each of these years. During the latter period, the annual CDP of precursors involving initiator is approximately 10^{-4} but that of precursor involving unavailabilities is lower than 10^{-5}, and in particular, lower than 10^{-6} in two years 1984 and 1985 because of a small number of such precursors being identified in these two years (three in 1984 and six in 1985). During this period, as well, there are one or two precursors involving initiator with a relatively high CCDP (10^{-3} or higher) in each year except 1983, leading to the higher annual CDP: reactor trip with reactor core isolation cooling and residual heat removal (RHR) unavailable at LaSalle Unit 1 in 1984 (2.3×10^{-3}), loss of feedwater at Davis Besse in 1985 (Table IV.6), inadvertent opening of SG relief valve resulting in reactor trip with HPI unavailable at Catawba Unit 1 in 1986 (3.3×10^{-3}), and reactor trip with a stuck open pressurizer relief valve at Turkey Point Unit 3 in 1986 (1.3×10^{-3}). In 1983, however, about 50 precursors involving initiator are identified while there are less than ten precursors involving unavailabilities, resulting in the significant difference in annual CDPs. On the other hand, in 1974 and 1991, the annual CDP of precursors involving unavailabilities is higher than that of precursors involving initiator due to potential failure of three AFW pumps at Turkey Point Unit 3 in 1974 (Table IV.6), and the potential failure of HPI at Shearon Harris in 1991 (6.3×10^{-3}).

For the PWR precursors, it can be seen from Figure IV.20(b) that the trends in the annual CDPs are similar to those mentioned above excluding the years before 1975 and in 1981. The difference between the annual CDPs of all precursors and PWR precursors in the years before 1975 is mainly due to no precursor involving unavailabilities being identified and the precursors involving initiator having a relatively low CCDP. Although two of ten precursors involving initiator have a CCDP higher than 10^{-3} and largely contribute to the annual CDP in 1981, there are only four precursors involving unavailabilities and their respective CCDPs are relatively low, resulting in the lower annual CDP of such precursors. Also at BWRs, shown in Figure IV.20(c), the annual CDP of precursors involving initiator is generally higher than that of precursors involving unavailabilities. In three years 1974, 1981, and 1998, however, the annual CDP of precursors involving unavailabilities is significantly higher because of an event at Brunswick Unit 1 in 1981 (RHR heat exchanger damage due to oysters: CCDP of 6.7×10^{-3}) and no precursor involving initiator being identified in 1974 and 1998. The annual CDP of precursors involving initiator has changed in the range 10^{-6} to 10^{-4} until 1993 except 1974 and 1975, in which no such precursor is identified and the fire event took place at Browns Ferry Unit 1, respectively, and after 1994, it has decreased to lower than 10^{-6}. For the precursors involving unavailabilities, the annual CDP is almost 10^{-5} or lower excluding 1981 in which the Brunswick event mentioned above occurred and in particular, it has remained at about 10^{-5} or lower during the years after 1984.

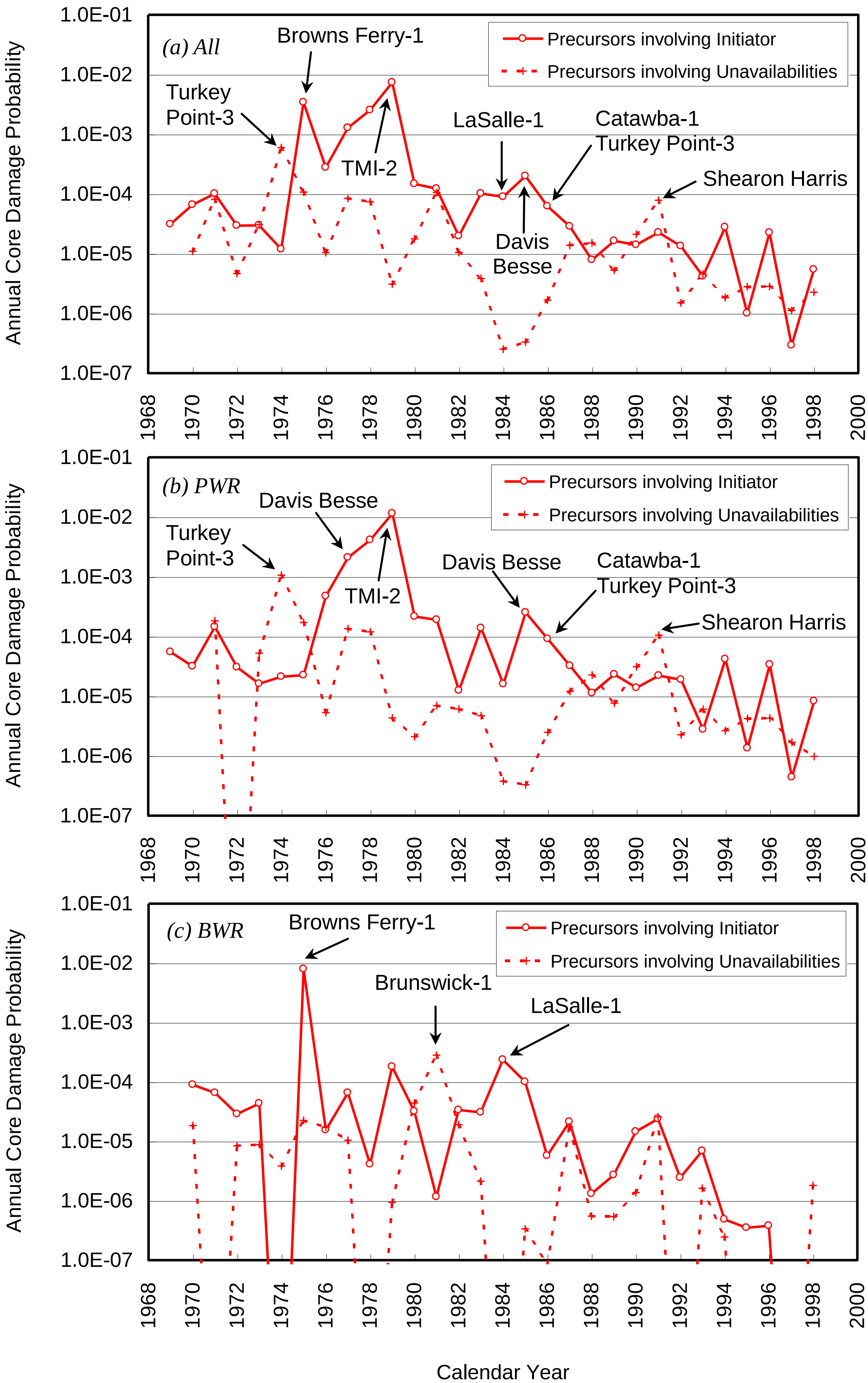

Figure IV.20 Annual Core Damage Probabilities by Precursor Categories

3.3 Overall Risk Trends

The results obtained from application of the newly proposed risk indicators, occurrence frequency of precursors and annual CDP, are summarized as follows:

- The occurrence frequency estimated for all precursors at both PWRs and BWRs changes in the range of 0.26 to 0.30 EPRY during the years until 1981 and significantly increases to 0.40 EPRY in the years 1982 to 1985 but after that, gradually decreases to 0.28 EPRY in 1998. Even though the frequency itself is different, a quite similar trend can be seen for the precursors involving initiator because such precursors have a large contribution to the total number of precursors. However, the trend observed for precursors involving unavailabilities is different and the frequency has remained at about 0.10 EPRY over the years.
- Comparing the occurrence frequencies for the PWR and BWR precursors, slightly different trends are observed. At PWRs, the frequency remains at around 0.30 EPRY during the years until 1981 with its changes in the range 0.24 to 0.34 EPRY in the first several years (up to approximately 130 reactor years) and significantly increases to 0.38 EPRY during two years 1982 and 1983 but after that, gradually decreases to 0.28 EPRY in 1998. At BWRs, the frequency until 1981 is lower (at around 0.27 EPRY) but after that, a remarkable increasing trend is observed (from 0.27 to 0.43 EPRY). Also, the decreasing trend after 1985 is slightly more significant compared with that at PWRs. This means that likelihood of risk significant events has been decreasing at both PWRs and BWRs but in recent years, it might be higher at PWRs than at BWRs.
- The trends observed for the occurrence frequency indicate that the likelihood of risk significant events has been decreasing in total at both PWRs and BWRs but the significance of potential/latent failures has been getting larger. The precursors involving unavailabilities could have a higher CCDP if the time to detect potential/latent failures would be longer. Therefore, it is important that additional efforts should be taken for eliminating such events by, for instance, earlier detection of potential/latent failures.
- The annual CDP increases to 10^{-2} during the years 1969 to 1979 due to several events with a high CCDP such as the fire event at Browns Ferry Unit 1 in 1975 and the TMI-2 accident in 1979 and after that it has changed in the range of 10^{-5} to 10^{-4} with the decreasing trend. This implies that the lessons learned from the TMI-2 accident would have been implemented into the plant design and/or operation practices and the likelihood of risk significant events has been lowered.
- The annual CDP at PWRs is generally higher than that at BWRs over the years, in

particular, significantly during the years 1975 to 1979. This observation is equivalent to the PSA results for PWRs and BWRs.

- Comparing the precursors involving initiator and unavailabilities, the annual CDP of precursors involving initiator is generally higher than that of precursors involving unavailabilities and has a larger contribution to the total annual CDP. This means that the precursors involving initiator would be more risk significant events. As the number of events with a relatively high CCDP becomes smaller, as well, the annual CDP has been getting lower. This implies that lowering the number of precursors involving initiators could result in a lower annual CDP.
- The recent trends observed in the two indicators, the occurrence frequency and the annual CDP, show that risk significant events have not taken place, which implies the nuclear power plant safety would have been improved.

It should be noted that in the ASP Program, the models used in the analyses have been changed over the years and do not explicitly cover all core damage sequences, in particular, those induced by external events such as fire and earthquake and therefore, the trends described above indicate limited observations. Despite such a limitation, the two indicators proposed here can provide the trends on core damage risk in industry-level useful for confirming that the safety of operating nuclear power plants is being maintained.

4. Summary

This study proposes new quantitative risk indicators, that is, the occurrence frequency of precursors and the annual CDP deriving from the results of the ASP analyses carried out by the USNRC and discusses their respective application results. As described above, the trends on how the likelihood of risk significant events and the potential of core damage risk in nuclear power industry have changed are indicated by estimating the occurrence frequency of precursors and the annual CDP. These trends are based on the ASP analysis results for the events that actually occurred and thus, may provide more empirical and/or realistic indications. Specifically, the core damage risks at U.S. nuclear power plants have been lowered and the likelihood of risk significant events has been remarkably decreasing, implying that plant safety in the United States has been improved. Through the applications, it is concluded that the effectiveness of the proposed indicators is demonstrated because these indicators can provide the quantitative information useful for:

- determining the likelihood of risk significant events,

- monitoring and/or predicting the risk level at nuclear power plants, and
- examining the industry risk trends.

Although, in this study, trends on the occurrence frequency of precursors and the annual CDP are displayed and discussed for overall nuclear power industry, PWR plants and BWR plants, it is considered that the proposed indicators can be employed as one of the performance indicators for the specific plant category (for example, category by operating periods) and/or in the utility-level (that is, for individual utilities) if much more events will be assessed using the ASP analysis approach in the future. It is also expected that the relevant data will be accumulated and more widely discussed in establishing risk indicators applicable to the individual plants.

IV.5 Concluding Remarks

The ASP analysis has been developed and used in the United States to identify and categorize the operational events as precursors to potential severe core damage accident sequences, that are conceptually similar to a “near miss” for a core damage, by calculating a probability of core damage given the failed equipment associated with the particular event. The author has been carrying out the ASP analyses of specific events that have the potential of risk significance, trending analysis based on the ASP documents published by the USNRC and development of the event tree models for providing a more realistic ASP analyses to obtain the risk significant trends, to characterize risk insights useful for identifying plant vulnerabilities, to feed the lessons learned from the study back to plant operations, and to establish risk indicators for event assessment. This chapter describes the outlines of the analysis approach, the analysis of STGR events and trending analysis of precursors with newly proposed risk indicators.

Ten actual and one potential STGR events were analyzed with use of a newly developed consistent ASP model to identify the risk significant anomalies observed during the events in terms of the potential for core damage and to obtain generic insights useful for examining alternative mitigation measures for STGR. The analysis results show that the CCDPs for SGTR events range from 10^{-4} to 10^{-2} and in particular, those involving the delayed identification of tube rupture or failure to timely depressurize the reactor could have a relatively high possibility of leading to core damage. This means that such failures are generic safety issues for SGTR, implying the importance of improving the capability to detect SGTR and the operating procedures. It is also shown that some of the other anomalies observed would largely contribute to the possibility of core damage, which points out the need of examining alternative measures for recovering from such conditions.

Quantitative risk trends were examined using newly proposed indicators, that is, the occurrence frequency of precursors and the annual CDP deriving from the results of the ASP analyses to more effectively and widely use the ASP results for discussing and/or monitoring industry risk trends. Through the applications, it is shown that the core damage risks at U.S. nuclear power plants have been lowered and the likelihood of risk significant events has been remarkably decreasing. This implies that plant safety in the Unites States has been improved. Since these trends are based on the events that actually occurred, the proposed risk indicators may provide more empirical and realistic observations useful for examining and monitoring the risk trends and/or risk characteristics in nuclear power industry. As well, this study underlines the need to accumulate the ASP analysis results and to employ the proposed indicators as one of the performance indicators for the specific plant category (for example, category by operating periods) and/or in the utility-level (that is, for individual utilities).

The ASP analysis of specific events can draw generic safety implications useful for identifying plant vulnerabilities, and the proposed risk indicators can examine the overall picture of risks at industry and individual plant levels. Therefore, such ASP studies should be actively carried out to obtain the insights for improving the plant safety. So far, the ASP study has not been conducted systematically for the operational events in Japan and hence, it is expected that the ASP analyses will be carried out and the results will be accumulated so that the nuclear power plant risks could be grasped and discussed more widely and effectively.

References in Chapter IV

(1) USNRC, *Status Report on Accident Sequence Precursor Program and Related Initiatives*, SECY-99-289, 1999.

(2) L. N.Vanden Heuvel, J. W. Cletcher, D. A. Copinger, J. W. Minarick, B. W. Dolan, and P. D. O'Reilly, *1993 Accident Sequence Precursor (ASP) Program Results*, Nuclear Safety, 352, pp.328-338, 1994.

(3) N. Watanabe, *Accident Sequence Precursor Analyses for Steam Generator Tube Rupture Events That Actually Occurred*, Reliability Engineering and System Safety, 57, pp.281-297, 1997.

(4) N. Watanabe, *Trends in Initiating Events and Unavailable Systems for Precursor Events*, Proc. of the 4th International Conference on Probabilistic Safety Assessment and Management (PSAM-4), 1998.

(5) N. Watanabe, *Trending Analysis of Precursor Events*, JAERI-Tech 97-067, 1998.

(6) N. Watanabe, K. Muramatsu, K. Ogura and J. Mori *Risk Insights for PWRs Derived from Accident Sequence Precursor Analysis Results*, Proc. of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM-5), 2000.

(7) N. Watanabe, *Quantitative Risk Trends Deriving from PSA-based Event Analyses – Analysis of Results from U.S.NRC's Accident Sequence Precursor Program*, Transaction of Atomic Energy Society of Japan, Vol.3 No.4, pp.396-406, 2004 (in Japanese).

(8) N. Watanabe, M. Hirano and H. Takahashi, *Sophistication of SGTR Event Tree for Accident Sequence Precursor Analysis*, Proc. of the International Topical Meeting on Probabilistic Safety Assessment (PSA'99), 1999.

(9) J. W. Minarick and C. A. Kukielka, *Precursors to Potential Severe Core Damage Accidents: 1969-1979*, NUREG/CR-2479, 1982.

(10) W. B. Cotttrell et al., *Precursors to Potential Severe Core Damage Accidents: 1980-1981*" NUREG/CR-3591, Vols.1&2, 1984.

(11) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1984*, NUREG/CR-4674, Vols.1&2, 1986.

(12) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1985*, NUREG/CR-4674, Vols.3&4, 1987.

(13) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1986*, NUREG/CR-4674, Vols.5&6, 1988.

(14) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1987*, NUREG/CR-4674, Vols.7&8, 1989.

(15) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1988*, NUREG/CR-4674, Vols.9&10, 1990.

(16) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1989*, NUREG/CR-4674, Vols.11&12, 1990.

(17) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1990*, NUREG/CR-4674, Vols.13&14, 1991.

(18) J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents: 1991*, NUREG/CR-4674, Vols.15&16, 1992.

(19) D. A. Copinger et al., *Precursors to Potential Severe Core Damage Accidents: 1992*, NUREG/CR-4674, Vols.17&18, 1993.

(20) L. N. Vanden Heuvel et al., *Precursors to Potential Severe Core Damage Accidents: 1993*, NUREG/CR-4674, Vols.19&20, 1994.

(21) R. J. Bells et al., *Precursors to Potential Severe Core Damage Accidents: 1994*, NUREG/CR-4674, Vols.21&22, 1995.

(22) R. J. Bells et al., *Precursors to Potential Severe Core Damage Accidents: 1995*, NUREG/CR-4674, Vol.23, 1997.

(23) J. A. Forester et al., *Precursors to Potential Severe Core Damage Accidents: 1982-83*, NUREG/CR-4674, Vol.24, 1997.

(24) R. J. Bells et al., *Precursors to Potential Severe Core Damage Accidents: 1996*, NUREG/CR-4674, Vol.25, 1997.

(25) R. J. Bells et al., *Precursors to Potential Severe Core Damage Accidents: 1997*, NUREG/CR-4674, Vol.26, 1998.

(26) R. J. Bells et al., *Precursors to Potential Severe Core Damage Accidents: 1998*, NUREG/CR-4674, Vol.27, 2000.

(27) USNRC, *Status of Accident Sequence Precursor and SPAR Model Development Programs*, SECY-02-0041, 2002.

(28) N. Watanabe, *Current Status of Accident Sequence Precursor Program for Operational Events at Nuclear Power Plants*, Journal of Atomic Energy Society of Japan, Vol.38, No.4, pp.252-261, 1996 (in Japanese).

(29) Science Applications International Corporation (SAIC), *Daily Events Evaluation Manual*, 1-275-03-336-01, 1992.

(30) USNRC, *Severe Accident Risks: An Assessment of Five US Nuclear Power Plants*, NUREG-1150, 1989.

(31) P. E. MacDonald, V. N. Shah, L. W. Ward and P. G. Ellison, *Steam Generator Tube Failures*, NUREG/CR-6365, 1996.

(32) USNRC, *Nuclear Plant Information Books (www.nrc.gov/ AEOD/pib/states.html (or disclaimer.html))*. (as of August 1996)

(33) L. B. Marsh, *Evaluation of Steam Generator Tube Rupture Events*, NUREG-0651,1980.

(34) USNRC, *NRC Report on the January 25, 1982, Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant*, NURIEG-0909, 1982.

(35) INPO, *A thermal-hydraulic Analysis of the Ginna Steam Generator Tube Rupture Event of January 25,1982*, INPO 84-009, 1984.

(36) Virginia Electric Power Company, *Steam generator tube rupture*, LER 338/87-017, Rev.1, 1987.

(37) Duke Power Company, *A Steam Generator Tube Rupture Occurred on 7 March 1989 and Resulted in An Alert Being Declared and An Unplanned Release of radioactivity,* LER 369/89-004, 1989.

(38) MITI, *Report on Mihama unit 2 steam generator tube rupture*, 1991 (in Japanese).

(39) S. Kondo, *Lessons Learned for PSA from the SGTR Incident at Mihama, Unit 2, in 1991*, Reliability Engineering and System Safety, 45, pp.57-65, 1994.

(40) Arizona Public Service, *Manual Reactor Trip following A Generator Tube Rupture*, LER 529/93-001, 1993.

(41) Virginia Electric Power Company, *Reactor trip due to main feedwater reg. valve closure and subsequent S/G tube leak*, LER 338/89-005, 1989.

(42) D. Rasmuson and P. O'Reilly, *Analysis of Annual Accident Sequence Precursor Occurrence Rates for 1984-1994*, Proc. of the International Topical Meeting on Probabilistic Safety Assessment (PSA'96), 1996.

(43) M. Muhlheim and R. Belles, *A Qualitative Review of the Accident Sequence Precursor Program*, Proc. of the 4th International Conference on Probabilistic Safety Assessment and Management (PSAM-4), 1998.

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Chapter V

Analysis of Severe Accidents at Fukushima Dai-ichi Nuclear Power Plant

V.1 Background

On March 11, 2011, the Tohoku District-off the Pacific Ocean Earthquake and the subsequent tsunami resulted in the severe core damage at Tokyo Electric Power Company's (TEPCO) Fukushima Dai-ichi Nuclear Power Plant Units 1 to 3, involving hydrogen explosions at Units 1, 3, and 4 and the large release of radioactive materials to the environment.

Four independent committees were established by TEPCO, the Japanese Government, the Diet of Japan, and the Rebuild Japan Initiative Foundation (RJIF) to investigate the accident and published their respective reports. TEPCO issued the interim report in December 2011 and the final report in June 2012 (hereinafter, the TEPCO report)(1,2). The Government's committee made public its interim and final reports in December 2011 and July 2012, respectively (hereinafter, the Government report)(3,4). The Diet committee and the RJIF committee opened their respective investigation reports in June 2012 and March 2012 (hereinafter, the Diet report and the RJIF report, respectively)(5,6). Also, the Nuclear and Industrial Safety Agency (NISA; the former nuclear regulator in Japan) carried out an analysis of accident causes to obtain the lessons learned from the accident and made its report public (hereinafter, the NISA report)(7).

This chapter delineates, at first, the severe accident scenarios at Fukushima Dai-ichi Units 1 to 3 with use of event trees for better understanding the accident(8). Next, the differences in their respective positions are clarified by reviewing the five reports from the technological point of view, focusing on the accident progression and causes to specify the issues to be further examined(8). Moreover, the undiscussed issues are identified to provide insights useful for the near-term regulatory activities including accident investigation by the Nuclear Regulation Authority[8].

V.2 Analysis of Accident Scenarios and Measures to Avoid Core Damage

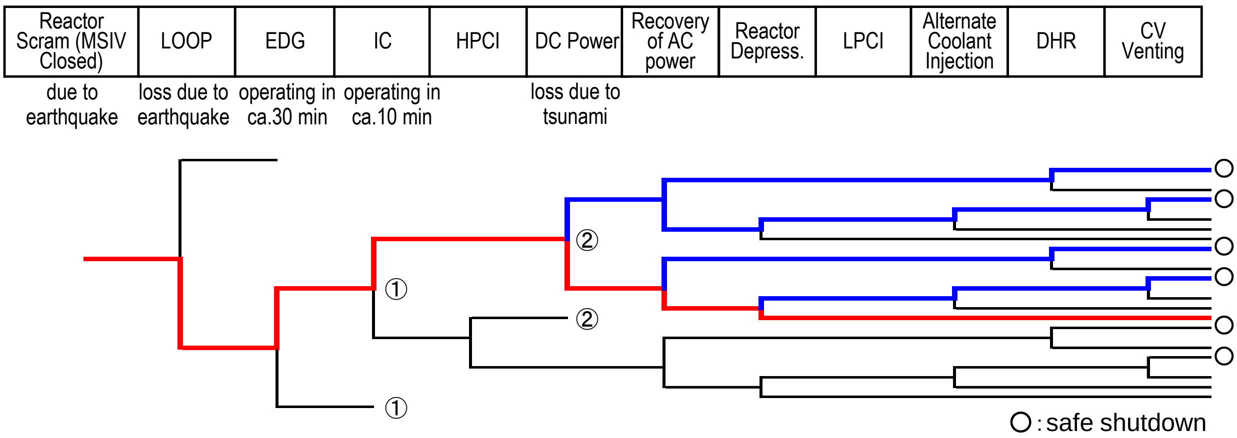
Fukushima Dai-ichi Units 1 to 3 experienced almost the same accident sequences that involved the long-term station blackout (SBO) initiated by the earthquake and the tsunami, leading to severe core damage and subsequent radioactive release. However, the individual sequences were chronologically different from each other. This section analyzes the core damage sequences at Units 1 to 3 and depicts them on event trees initiated by earthquake to clarify the differences among accident sequences at Units 1 to 3. As well, the actual responses to avoid the severe accidents are discussed using the event trees. **Figures V.1** to **V.3** show event trees developed for Units 1 to 3, respectively. In these event trees, the "red" path indicates the actual accident sequence and the "blue" paths mean the ones that could avoid the reactor core from damaging.

The representative accident sequences at Units 1 to 3 were in common as follows: The reactor automatically scrammed due to the earthquake, followed by loss of offsite power (LOOP), auto-start of emergency diesel generators (EDGs), actuation of isolation condensers (ICs) at Unit 1 and reactor core isolation cooling (RCIC) at Units 2 and 3. Afterwards, both DC and AC powers onsite were lost due to flooding of power centers (low voltage switchboards) and metal clads (high/medium voltage switchboards) by the tsunami and the offsite power was not recovered, resulting in the long-term SBO though the high pressure coolant injection (HPCI) actuated at Unit 3. As a result, the reactor was not depressurized, preventing alternative water injection. Provided that DC power would have been available and offsite power would have been restored, the core cooling could have been carried out with RCIC and/or HPCI and as well, the subsequent decay heat removal might have been performed, bringing the reactor into cold shutdown. Even in the case of no AC power restoration, the reactor depressurization and the subsequent alternative water injection with containment venting might have prevent the core from damaging if DC power would be available. As well, timely recovery of electric power supplies could have made the decay heat removal capability available, avoiding the core damage. In such a way, core damage might have been avoided by using some cooling measures, including alternative water injection, if DC power would have been available and AC power would have been restored.

1. UNIT 1

The reactor scrammed due to the earthquake, followed by LOOP, turbine trip, closure of main steam isolation valves (MSIVs), and so on. The closure of MSIVs caused the reactor pressure to rise and ICs automatically actuated. As well, EDGs automatically started and supplied the electric power to onsite loads. ICs operated for approximately 10 min and then, the operators manually tripped ICs. EDGs operated for approximately 30 min until a series of tsunami waves attacked the site. Afterwards, the electrical switchboards were flooded by the tsunami, resulting in loss of DC power. As a result, ICs and HPCI could not start both automatically and manually and the indicators of reactor parameters such as water level were lost in the control room.

At that time, Unit 1 was brought into SBO with DC power lost. Since no electric power supply was restored after that, the accident progressed to core damage. In order to recover from such accident sequences and avoid the core from damaging, the accident managements were required as follows: restoring electric power supplies and subsequently, establishing decay heat removal or assuring alternative water injection with the reactor depressurized and the containment vented. Considering that the core damage might have occurred within 2-3 h after the IC trip, however, it would have been highly unlikely to execute these actions successfully in a timely manner. Actually, such efforts were unsuccessful. The accident progression at Unit 1 seems one of the typical SBO sequences defined in the past probabilistic risk/safety assessment (PRA/PSA).



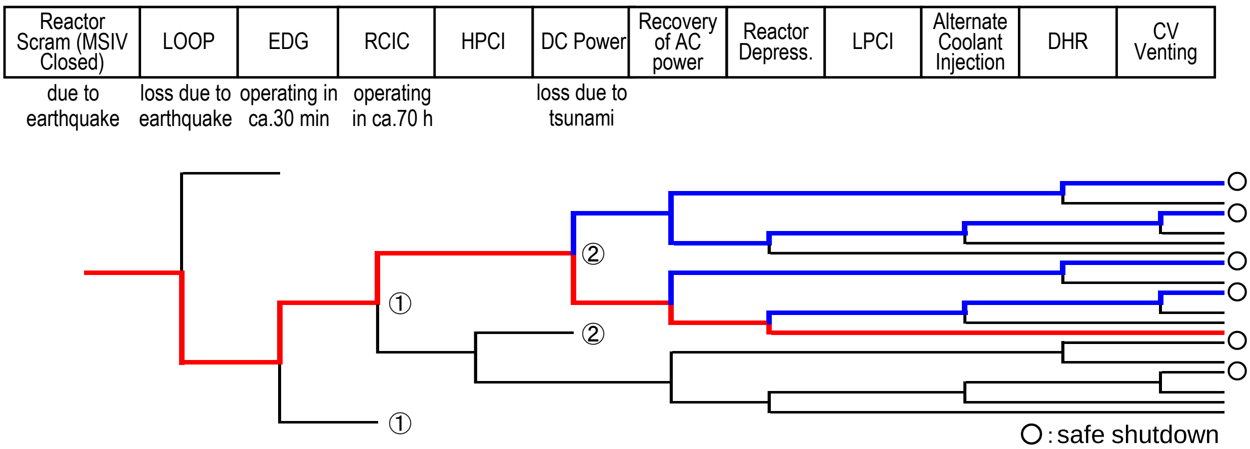
LOOP: loss of offsite power, EDG: emergency diesel generators, IC: isolation condensers, HPCI: high pressure coolant injection
DC: direct current, AC: alternate current, LPCI: low pressure coolant injection, DHR: decay heat removal, CV: containment vessel

Figure V.1 Event Tree and Accident Sequences for Unit 1

2. Unit 2

The reactor scrammed due to the earthquake, followed by LOOP, turbine trip, closure of MSIVs, and so on. The MSIV closure caused the reactor pressure to rise and safety/relief valves (SRVs) cyclically operated to control the reactor pressure. As well, EDGs automatically started and supplied the electric power to onsite loads. Since RCIC was manually started in accordance with the operating procedures for transients with MSIV closed[9] before the electric power supplies were lost by the tsunami, it could continue to run for 70 h despite loss of DC power. After the trip of RCIC, the efforts to restore the electric power supplies and to depressurize the reactor were unsuccessful, resulting in SBO with DC power lost followed by core damage.

In order to recover from such accident sequences and avoid the core from damaging, the following accident managements were required: restoring electric power supplies and subsequently, establishing decay heat removal or assuring alternative water injection with the reactor depressurized and the containment vented. Compared with the sequence at Unit 1, the time available for doing that was longer and thus, the likelihood of having restored electric power supplies and assured alternative water injection is relatively higher. Since the electric power supplies were not restored in a timely manner and the reactor was not depressurized, however, alternative water injection was not carried out. Although the accident sequence at Unit 2 was SBO with DC power lost, RCIC continued to run for a long time without its control and the switchover of water source was performed. Therefore, it can be considered that the sequence at Unit 2 was, in actual, loss of decay heat removal.



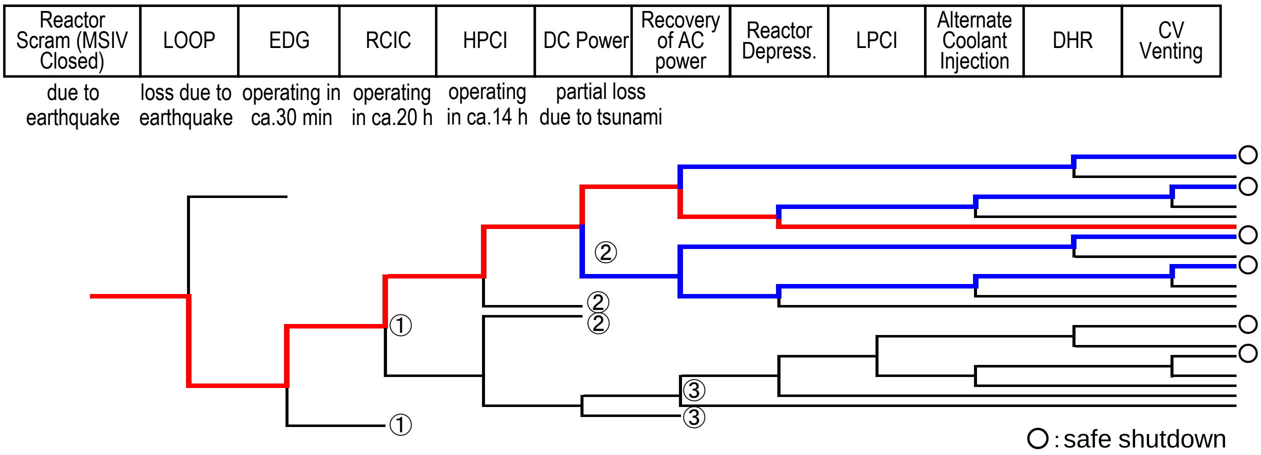
LOOP: loss of offsite power, EDG: emergency diesel generators, RCIC: reactor core isolation cooling, HPCI: high pressure coolant injection
DC: direct current, AC: alternate current, LPCI: low pressure coolant injection, DHR: decay heat removal, CV: containment vessel

Figure V.2 Event Tree and Accident Sequences for Unit 2

3. Unit 3

The reactor scrammed due to the earthquake, followed by LOOP, turbine trip, closure of MSIVs, and so on. The MSIV closure caused the reactor pressure to rise and SRVs cyclically operated to control the reactor pressure. As well, EDGs automatically started and supplied the electric power to onsite loads. The operators manually started RCIC according to the operating procedures for transients with MSIV closed[10]. Although the sequence at Unit 3 was the same one at Unit 2 until the tsunami hit the site, some DC power centers were not affected by the tsunami and thus, control power supplies and HPCI were also available. RCIC automatically tripped due to malfunction of a valve after running for approximately 20 h. Then, HPCI automatically actuated and the reactor water level was recovered and maintained. The operators switched over HPCI to the test line operation to depressurize the reactor. However, HPCI was manually tripped, considering the possibilities of pipe ruptures due to vibrations of the HPCI turbine caused by a long time operation of HPCI under the low pressure conditions. Since SRVs could not be operated at all, afterwards, the efforts to restart HPCI were unsuccessful due to depletion of batteries, resulting in SBO with DC power lost followed by core damage.

In order to recover from such accident sequences and avoid the core from damaging, the following accident managements were required: restoring electric power supplies and subsequently, establishing decay heat removal or assuring alternative water injection with the reactor depressurized and the containment vented. Although the time available for taking some measures was longer, considering that RCIC and HPCI operated for totally 35 h, no accident management measure was successful. In particular, the reactor pressure had been maintained at 1 MPa or lower for 8 h, during which HPCI was operating (0.58 MPa at the time of HPCI trip), and hence, alternative water supply with fire engines might have been allowed during this period. However, no water supply was performed in a timely manner since plant personnel was forced to interrupt their work for preparing the alternative water supply by the hydrogen explosion at Unit 1 and high radiation on the site. At Unit 3, DC power for controlling RCIC and HPCI remained available and the cooling water was supplied by RCIC and HPCI even in the limited time. Therefore, the sequence at Unit 3 was a typical one of the SBO sequences but its progression was longer than that generally assumed in the past PSA. Actually, the operators switched over HPCI to test line operation to depressurize the reactor but after the manual trip of HPCI, the reactor pressure increased again, preventing the alternative water supply. Hence, the HPCI test line operation was not considered as a depressurization measure in the event tree.



LOOP: loss of offsite power, EDG: emergency diesel generators, RCIC: reactor core isolation cooling, HPCI: high pressure coolant injection
DC: direct current, AC: alternate current, LPCI: low pressure coolant injection, DHR: decay heat removal, CV: containment vessel

Figure V.3 Event Tree and Accident Sequences for Unit 3

As mentioned above, the accident sequences can be delineated with event tress. Since the current event tree approach is intended to represent the accident sequences for a single unit, however, the adverse effects from the neighboring unit, such as the impact of hydrogen explosion at Unit 1 on the recovery actions at Unit 3, are not usually considered in event trees. Also, the sequences delineated in event trees at Units 1 and 2 are apparently the same although they are chronologically different from each other. Therefore, it seems essential to improve the event tree approach so that chronologically time-dependent sequences can be explicitly delineated and the circumferences of neighboring unit(s) can be modeled to provide the accident sequences more realistically.

V.3 Review of Accident Investigation Reports

Their respective positions in five reports are not necessarily in agreement. Particularly, the TEPCO report, the Government report and the Diet report express significantly different positions on some matters. This section discusses the differences in their positions.

1. Electric Power Supplies

1.1 Transmission Lines of Offsite Power Supplies

The TEPCO report, the Government report and the NISA report indicate no significant differences in their respective positions on causes of LOOP. On the other hand, the Diet report focuses on the fact that 6 out of 7 transmission lines had been connected to the same substation and points out insufficient diversity and independence of offsite power against external hazards such as earthquake and inadequate aseismic design of substation as one of the causes of LOOP. The NISA report emphasizes the need to improve the reliability of offsite power, considering the fact that the nuclear power plant (NPP) safety had heavily been affected by unavailability of AC power sources, while the report insists that the plant safety should not rely on offsite power sources excessively.

The regulations of NPPs in Japan require the individual reactors have more than two transmission lines from offsite power but do not specify these lines be connected to one or more substations. On the other hand, Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants", in the United States specify that the electric power from the transmission network to the onsite electric distribution system be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions(11). Although this requirement is more stringent than that in Japan, it is not clear whether or not offsite substations are regulated. It is necessary to study how the regulatory coverage should be defined to improve the reliability of offsite power. For reference, in Japan, some sites such as Onagawa Nuclear Power Plant are supplied with offsite power from separate transmission lines connected to two or more independent substations.

1.2 Layout Designs of Electrical Switchgears for Onsite Power Supplies

While the TEPCO report describes only the damage situation of electrical switchgears for onsite power supplies, such as metal clads (MCs), power centers (PCs), and DC power distribution systems, and EDGs, other three reports address the causes of loss of onsite power supplies as follows.

The Government report states that many switchgears for onsite power supplies were flooded by the tsunami, resulting in loss of their respective functions, and raises concerns about lack of physical separation for electrical switchgears and EDGs. In addition, it points out that TEPCO had not implemented any measures to cope with the case where two or more units would be affected simultaneously by external events such as natural disasters and any unit might not be provided with power from its neighboring unit(s) because the TEPCO's accident management against SBO was incorporated based on the assumption that at least one of neighboring units might not be affected.

The Diet report introduces the fact that all normal MCs and emergency MCs, and normal PCs for Unit 1 were located on the first floor of the turbine building and points to the vulnerability of power supply system to external events such as flooding and fire as well as threats from malicious and intentional acts. As well, it states that SBO could have occurred even if only one specific area was damaged.

The NISA report places emphasis on the fact that devices and components located on the same floor of building lost their functions due to a common cause, that is, flooding by the tsunami and thus, highlights the need to enhance the physical separation and independence of onsite power supplies.

However, these reports do not address and analyze reasons why the electrical switchgears and EDGs had been installed on the first floor or basement level of building. The Safety Design Review Guides established by the Nuclear Safety Commission of Japan requires the redundancy or diversity and independence for emergency power supplies(12) but does not specify the requirement on physical separation. This means lack of regulatory rules and is supposed one of root causes of the accidents. More important things are the licensee's thoughts about layout of electrical switchgears and EDGs. Although the layout design at PWRs and BWRs in Japan has been defined focusing on aseismic natures, maintainability, connectivity with the control room, and so on(13, 14), as shown in **Table V.1**, no specific guidelines have been prepared and thus, the layout design has relied on the licensees' discretion, which might have led to the fact that electrical switchgears and EDGs had been installed in some specific areas.

Table V.1 Basic Concepts of Layout Design for Electrical Components – (a) BWRs

Early phase of introduction (Fukushima Dai-ichi NPP Units 1-5)
・ Layout design followed that of U.S. BWRs.
・ Seismic Design: EDGs were installed in the building, which was founded directly on bedrock, and placed on the foundation of basement (the lowest level) considering their heavy weights and vibration mitigation. Switchgears (metal clad and power center) were basically placed on locations near their loads such as pumps.
Middle phase of introduction (Fukushima Dai-ichi NPP Unit 6, Fukushima Dai-ni NPP Units 1-4)
・ EDGs and emergency electric components belonging to the high aseismic category were installed in the accessory building attached to the secondary containment building (reactor building). The accessory building was constructed by enhancing the ground contact area (relief limit of building foundation).
・ EDGs were placed on the foundation of basement considering their heavy weights and vibration mitigation.
Later phase of introduction (Kashiwazaki-Kariwa NPP Units 1-7)
・ Because of deeper bedrock level compared with the preceding sites, the reactor building needed to be deeply buried in the ground.
・ EDGs were placed on the first basement level or the first level, which is near the ground leveling surface of the accessory building, emphasizing maintainability.
・ EDGs were laid out so that their vibrations can be coped with by the building structure.
・ After that, this design concept was followed and the installation of EDGs on the first floor became the standardized layout design.

Table V.1 Basic Concepts of Layout Design for Electrical Components – (b) PWRs

- Basic Concept: The height of buildings and framework were determined in consideration of quake resistance, and the heavy loads were placed on the lower floors and the light loads were on the upper floors.
- Relays, metal clad switchgears and batteries were placed on the locations near the main control room inside the reactor building in consideration of physical connectivity.
- In many plants, relays, metal clad switchgears and battery rooms were located on the floor below the main control room.
- EDGs were placed on the ground level in consideration of their maintainability.

(Note: At PWRs, there is almost no difference in layout design by age and type of reactors.)

1.3 Trip of Emergency Diesel Generators

Although all but the Diet report assume that the cause of EDG tripping was the flooding of EDGs, onsite power distribution system and/or DC power supply system by the tsunami, the Diet report takes different positions on the EDG trips. The Diet report estimates the arrival times of the first and second waves and, based on this estimation, expresses its views that it is impossible for the tsunami to have been the cause of loss of emergency AC power supply from EDG-A at Unit 1, EDG-B (air-cooled) at Unit 2 and EDG-B (air cooled) at Unit 4 unless the second wave had arrived before AC power was lost. Also, the Diet report raises a question about other EDGs (EDG-B at Unit 1, EDG-A Unit 2, EDG-A and B at Unit 3) having tripped by the tsunami and assumes one case that the cooling line or fuel oil supply line of EDG might have been broken by the earthquake and consequently, EDGs might have tripped due to being overheated or supplied with no fuel oil, and the other case that some parts of EDGs might have been deformed or dislodged by the earthquake and the EDG shafts or bearings might have been out of alignment, causing EDGs to trip due to being overheated or thermally bound.

As pointed out by the Diet report, it is likely that piping or mechanical parts could have been broken due to the earthquake, resulting in EDGs having stopped. However, such conditions would have occurred in a random manner. Considering the fact that EDGs had automatically started after the earthquake and subsequently, tripped almost simultaneously, it does not seem realistic to assume that simultaneous tripping of EDGs would have been mainly attributed to the mechanical failures caused by the earthquake. On the other hand, operating experience shows failures to continuously run of EDGs and EDGs might have tripped if random failures would have coincided with defects in maintenance and thus, the views of the Diet report cannot be dismissed.

2. Core Cooling

2.1 Isolation Condenser at Unit 1

All the investigation reports discuss the IC issues at Unit 1 as an important matter. However, their respective positions are different from each other on the manual trip of ICs, actuation of their isolation valves, including logic for closing the valves, and operating procedures.

(a) Reasons of manual trip

Both ICs automatically started on the reactor high pressure signal but were manually tripped approximately 10 min later. The TEPCO report makes the case that the manual trip of ICs was consistent with the relevant operating procedures because the operators recognized that the reactor cooldown rate limit of 55^{o}C/h would have been exceeded. Although the Government report agrees with the TEPCO's position, the Diet report opposes to that and points out the possibility of the broken IC piping. The Diet report states as follows: TEPCO should have well known how the reactor pressure and coolant temperature would vary when both ICs automatically actuate at the same time and should have set the IC auto-start conditions; if the operators manually tripped ICs since the cooldown rate could not be maintained below the limit of 55^{o}C/h, nevertheless, it is likely that the cooling capability of ICs was too high to be used actually or the IC piping rupture would have led to the cooldown rate limit being exceeded. Then, the report points out that the TEPCO's position is irrational and self-contradictory. As well, the Diet report stresses the possibility of IC pipe failures due to the earthquake based on the following two reasons; i) the Unit 1 plant personnel stated that he tripped both ICs to confirm whether or not the coolant was leaking from the IC piping or other pipes because of the rapid reactor pressure drop, to control the reactor pressure and then, to bring the reactor into cold shutdown according to the operating procedures, and ii) ICs were working for 11 min, resulting in the reactor pressure decrease from approximately 6.8 MPa to 4.5 MPa. Furthermore, the Diet report describes that the possibility of a small-break LOCA with a leakage of 0.3 cm^2 or less cannot be excluded referring to the analytical results with RELAP5 by JNES.

However, the manual startup and trip of ICs for controlling the reactor pressure is consistent with the operating procedures and actually, a similar operation was executed in the events at Tsuruga Unit 1 in 2003 and 2004(15). In addition, as pointed out by the Diet report, TEPCO had no experience of using ICs and had not provided the operating staff with any simulator training. Therefore, the plant personnel might not have known how the reactor pressure decreases when ICs work. It is no wonder that they have thought of several possibilities including leakage from the IC piping because the cooldown rate was faster than that observed in the normal shutdown process. Also, the analysis performed by JNES shows that the reactor pressure and temperature calculated for the case of no leakage assumed were in good agreement with those measured. It is not necessarily appropriate to point out that the reactor pressure had decreased too rapidly. In addition, seat leak in valves and/or seal leak in pumps might have been enlarged to the leakage of about 0.3 cm^2 due to unidentified causes. Therefore, the rapid decrease of reactor pressure is not a decisive factor in determining that the damage to piping took place due to the earthquake.

(b) Logic for closing isolation valves

The Diet report makes an objection to the terminology “fail safe” being used to the logic for closing the isolation valves of ICs in the TEPCO and the Government reports. This is a dispute concerning whether or not the concept and use of “fail safe” are appropriate to the logic for closing the isolation valves and is not related directly to the accident progression and causes. Instead, it should be verified whether the logic had been designed as the result of having examined to which priority should be given between the ICs’ function and the isolation of steam supply line break. According to the NISA report, in the logic for RCIC or HPCI, any signal for isolating valves in its turbine steam supply line is not generated when the electrical power is lost for break detection circuits. Additionally, the Diet report states that at a U.S. BWR, Oyster Creek, the isolation valves of ICs are not closed on loss of electric power. It is necessary to analyze and/or examine the reasons why the basic concepts are different in the design of equipment or systems which have a similar function.

The NISA report points out as follow; although the interlock for closing isolation valves had been designed so that ICs might lose their function by activating the interlock in the case of loss of DC power to the control circuits of valves, such a design concept was not correctly recognized by TEPCO, contributing to the accident progression.

(c) Operating procedures

The Government report expresses its view as follows; it is not unnatural that the shift operators were attempting to gradually reduce the reactor pressure with the cooldown rate limit of 55^{o}C/h not being exceeded and finally to bring the reactor into cold shutdown. As well, the TEPCO report claims that the operators were controlling the reactor pressure by opening and closing the valves of ICs to bring the reactor into cold shutdown and there was no particular problem because these actions were similar ways in the training programs. The NISA report states that the reactor pressure decreased to the value lower than that specified in the operating procedures due to the auto-start of ICs but the operator action for controlling the reactor pressure seems to have been properly executed by opening and closing the valves of ICs through the comparison with the pressure trends observed at Tsuruga Unit 1. This implies that no procedural problem was involved in these operations. However, the operating procedures specify that the reactor pressure be maintained at 6-7 MPa with use of ICs. Such an IC operation is considered a kind of temporary actions until reopening MSIVs is allowed. In this accident, the unprecedented large earthquake occurred, resulting in automatic reactor scram and loss of offsite power. Hence, the operators should have considered the possibilities of strong aftershocks and have attempted to bring the reactor into cold shutdown as soon as possible instead of maintaining the reactor pressure using ICs. Actually, the operating procedures for the reactor scram with MSIV closed do not assume the auto-start of ICs and specify that the reactor be depressurized with SRVs, HPCI test line operation without injection or manual operation of ICs(16). In the original design, ICs are intended to maintain the reactor in a hot shutdown condition. Provided that the operators would have fully recognized the original design concept of ICs, it is unnatural that they attempted to place the reactor into cold shutdown using ICs in the case that the reactor automatically scrams due to earthquake like this time. Furthermore, the TEPCO report states that only the operation (opening/closing) of IC valves had been executed without steam flowing, as well as the tabletop exercises, in the training program, implying that the operators could not have understood the reactor pressure behaviors when using ICs. Nonetheless, the report claims that proper operations towards cold shutdown were carried out in the control room as per training. As pointed out in the Government and Diet reports, it is a very important point that there is no experience of using ICs and no simulator training had been performed at all. It should be noted that Fukushima Dai-ichi Unit 1 had been operated without experience and adequate training on safety-related systems which are expected to actuate when a transient occurs (in other words, the safety culture had been lacking over many years) and thus, the individual licensees should confirm that such situations are not applicable to their own plants. Although the NISA report refers to the Tsuruga Unit 1 event in which manual startup of ICs was carried out prior to its auto-start, no further discussion is provided on difference of the IC operations at Fukushima Dai-ichi Unit 1 and Tsuruga Unit 1.

It is noteworthy that none of the five reports discuss the adequacy of the operating procedures applied immediately after the earthquake, including the IC operations. Thus, Section V.4 will address the technical issues on this matter.

Regarding the fact that TEPCO made wrong assumptions that ICs had still been operating based on the temporary readings of water level gauge, the Government and Diet reports note that both the TEPCO headquarters and the site personnel incorrectly assessed high radiation being detected near the Unit 1 reactor building at approximately 17:50 on March 11, 2011. Taking the high radiation level into account, TEPCO should have suspected that ICs had not operated anymore and the core had already been uncovered.

2.2 Continuous Operation of Reactor Core Isolation Cooling (RCIC) and Alternative Water Injection at Unit 2

Although DC power was lost after the tsunami hit the site, at Unit 2, RCIC had been operating for approximately 70 h. About 35 h after loss of DC power, the water source of RCIC was manually switched over from the condensate storage tank (CST) to the suppression pool. On this switchover, the Government report describes that the suppression pool might have gradually lost its capability of pressure suppression and steam condensation, leading to less steam being released from the RCIC turbine and the subsequent decrease of steam flow rate and rotation speed of the turbine. Since CST still had enough water volume at that time, the switchover of water sources is open to question.

In addition, the Government report assumes that the plant personnel interpreted the Unit 2 situation more optimistically based on the fact that RCIC had continuously been operating and did not recognize the needs to monitor and appropriately assess the water temperature and pressure in the suppression pool. As a result, the report points to the likelihood of having carried out containment venting, reactor depressurization and subsequent alternative water injection by fire engines prior to the explosion in the Unit 3 reactor building. In the following paragraph, the TEPCO's responses will be discussed on the continuous operation of RCIC and alternative water injection from a different point of view from the Government report.

It seems that the operators thought that RCIC had been unavailable due to loss of its control power because DC power supply had been lost due to the tsunami. In fact, the plant personnel confirmed, on the spot, that RCIC was operating after they observed the reactor water level being above the top of active fuel (TAF) by connecting the temporary batteries to the instruments. As indicated in the NISA report, however, RCIC is designed to keep the turbine stop valve as is in the case that its control power is lost (that is, "fail as is") and thus, the drive steam is continuously supplied to the turbine even in the case of loss of DC power supply. In addition, the governor valve of RCIC is hydraulically controlled, which is designed to fully open by its spring force in the case that its control power is lost, and the hydraulic pressure is ensured as long as RCIC is operating since the pressure is raised by the turbine. The operators should have checked the operability of RCIC in the earlier stage after loss of DC power supply if such design specifications would have fully been understood. Provided that this check would have been made earlier, the operators could have recognized the conditions where RCIC was operating, have expected the inevitable trip of RCIC, and then have aligned alternative water injection earlier. According to the analytical results, it is highly likely that the core would have been avoided from damaging if the reactor depressurization and subsequent alternative water injection would have been executed within 13,000 sec (approximately 3.6 h) after RCIC tripped[17]. This implies that the operators would have had time to take actions needed for avoiding core damage.

While the TEPCO report does not state the reason why RCIC tripped, the Government report addresses the possibility of the reactor pressure exceeding the pump discharge head. As indicated in the Government report, however, the water temperature and pressure in the suppression pool was 149.3^{o}C and 0.486 MPa respectively about 1 h before the RCIC trip was judged and the pool was in a mostly saturated condition. Consequently, it can be thought that the RCIC pump would have tripped due to cavitation.

2.3 *Trip of Reactor Core Isolation Cooling (RCIC) at Unit 3*

Some of DC power supplies were available at Unit 3. RCIC was aligned so that cooling water could pass through both its injection and test lines. This alignment aimed at avoiding RCIC from automatically tripping on reactor high water level to reduce the drain on batteries due to the RCIC startup and trip and to maintain reactor water level. Then, the operators set the flow rate to allow reactor water level to vary slowly. However, RCIC tripped automatically approximately 20 h after manual startup.

The TEPCO report attributes the cause of trip to a valve malfunction but does not provide its details. On the other hand, the Government report describes that the cause was the latch being dislodged in the valve. If this latch is considered to be the latch mechanism in the turbine stop valve, such failures had previously been observed at Fukushima Dai-ni Units 1 and 2 on September 23, 2003 and July 15, 2008, respectively, the causes of which had been identified mechanical degradations due to inadequate maintenance[18]. In another event at Fukushima Dai-ichi Unit 5 on October 28, 2007, the ill-maintained trip mechanism had caused the latch failure[18]. Therefore, it is necessary to examine how the latches had been maintained and what kinds of corrective actions had been applied to prevent the latches from being dislodged.

2.4 Trip of High Pressure Coolant Injection (HPCI) and Reactor Depressurization with Safety/Relief Valves (SRVs)

Although HPCI operated over approximately 14 h, it was manually tripped with concern that the decreased turbine rotation speed would cause its accessories such as pipes to damage due to vibrations and that the resultant steam leakage would take place. After that, the operators made efforts to restart RCIC and HPCI since SRVs could not be opened. However, RCIC and HPCI were not started because of the valve malfunction and the battery depletion, respectively.

The Government report expresses the critical views on the manual trip of HPCI and the subsequent attempts to restart RCIC and HPCI as follows; (i) the operators should have expected that the batteries had been expended by RCIC and HPCI having operated for approximately 20 h and over 14 h respectively and that the batteries might not have had the capacity sufficient for manually opening SRVs, and (ii) they should have recognized that HPCI would not be restarted in the case of unsuccessful attempts to open SRVs once HPCI had been tripped because they had been aware that the batteries might be exhausted for restarting HPCI. The report also states that the sharing of information on the manual trip of HPCI was delayed, the responses to serious situations fell behind, and as a result, these delayed actions aggravated the plant conditions and the work environment, making it more difficult to depressurize the reactor and to execute alternative water injection. Furthermore, the report addresses that the operators should have attempted to depressurize the reactor using SRVs by aligning alternative water injection lines in advance prior to the manual trip of HPCI. On the other hand, the Diet report points out that the severe accident might have been avoided by quickly depressurizing the reactor to the pressure below the pump head and instantly injecting water to reflood the core, instead of sticking to maintaining water level. The report also raises a question about the prolonged test line operation of HPCI.

These reports are to the point in terms of the timing of manual reactor depressurization. In particular, it is supposed that there was a great opportunity of executing the reactor depressurization and the subsequent water injection with use of fire engines in the period from the afternoon on March 12, 2011 through the predawn on March 13, 2011 since the reactor pressure had been below 1 MPa. Considering that the containment venting was successful several times after that, additionally, the severe core damage might have been avoided if the core could have been reflooded in this timing. As well, a question arises about the fact that the reactor pressure had been maintained low over the long time by the test line operation of HPCI if the operators would have well recognized that SRVs can be manually opened only when the reactor pressure is higher than 0.686 MPa. In any case, it is a very important point that the reactor had not been depressurized using SRVs with alternative water injection aligned while the batteries had been available. Therefore, the detailed survey and analysis are needed on this matter.

The Government report states that it was found that the operators opened SRVs at about 9:50am on March 13, 2011 and raises a question about whether or not the SRVs had been maintained at their open positions in the process of the reactor having been depressurized to 0.460 MPa at about 9:10am on that day. This indication differs from the TEPCO's position as follows: the reactor pressure began to decrease when SRVs had manually been opened at about 9:08am and after that, the reactor pressure had been maintained low with an SRV opened by connecting the batteries to control panels. This is a very important issue for understanding the accident progression and thus, further investigation needs to be carried out to clarify the fact.

3. Reactor Depressurization

3.1 Safety/Relief Valves (SRVs) at Unit 1

Since the measured reactor pressure had not been available after ICs were disabled under the SBO situation, the TEPCO report assumes the scenario as follows; the reactor pressure was controlled by cyclic operation of SRVs, the reactor coolant flowed into the suppression pool through SRVs, the reactor water level gradually decreased, resulting in core uncovery and subsequently core melt, and the SRV nozzle gasket(s) and/or in-core instrument tube(s) failed due to extremely high temperature in the reactor pressure vessel (RPV) in the course of core melting, depressurizing the reactor.

On the other hand, the Diet report points out that at Unit 1, the SRVs might have never (or almost never) actuated during the accident progression because no recorder of SRV operations had been installed and nobody heard any sound stemming from the SRV operations at Unit 1. The report also assumes the scenario that a certain piping connected to RPV might have failed due to the earthquake, leading to a small break LOCA and subsequent core damage. Although the Diet report addresses the possibility of a small break with 0.3 cm^2 based on the analysis with RELAP5 performed by JNES, such a leakage brings almost no change in the reactor pressure and water level and thus, is inconsistent with the scenario the report assumes above. Other two scenarios can be supposed; one that certain piping connected to RPV was cracked due to the earthquake and then the material strength was lowered by being subjected to high temperature after the core uncovery, resulting in the crack growth, and the other that the leakage of SRV nozzle gasket developed into its failure due to being exposed to high temperature conditions. If the RPV melt-through would occur under the high pressure condition, the containment atmosphere might be directly heated by molten fuel particulates released from RPV, resulting in the containment failure. This phenomenon is so called “direct containment heating due to high pressure melt ejection”. Therefore, it is important to determine whether or not such a phenomenon had taken place in the accident in order to better understand the accident progression and obtain the insights on plant behaviors. The further analysis and investigation are needed on how the reactor was depressurized after the initiation of core melt.

3.2 Safety/Relief Valves (SRVs) at Unit 3

Concerning the reactor depressurization with use of SRVs after the manual trip of HPCI, the Government report indicates that the operators had not been trained on how to depressurize the reactor using SRVs when the reactor pressure is lower than 1 MPa though SRVs are designed to be manually operated under the reactor pressure of 0.686 MPa or higher and be maintained at their open position under the pressure of higher than 0.344 MPa. Then, the report addresses that the risk of failure to depressurize the reactor should have been taken into account when the operators tripped HPCI. However, it is considered necessary to examine whether or not the contents of training had been enough for operators to understand the design specifications of such equipment and components.

4. Containment Integrity and Venting

4.1 Operator Actions for Containment Venting

The TEPCO report states that the plant personnel looked at relevant diagrams and consulted with contractors in order to check types and structures of valves on containment venting lines and to determine whether or not those valves can be manually opened. Also, the Government report shows that the plant personnel had to identify the valves to be opened, to confirm their locations and to check if they can be operated manually, one by one. The Diet report points to the problems concretely as follows; (i) the containment venting was one of the simplest actions of operating a switch for the valve based on the premise that the power would be available and the operation would be executed in the control room, and thus the operating procedures had not specified the manual operation for containment venting in the case of loss of power, (ii) manual operation of valves were very difficult due to lack of the relevant diagrams and poor maintenance of manually operable parts, (iii) the training of accident management (AM) including containment venting had been conducted by using the personal computer screen, which simulated the control panel for AM, and clicking a mouse. The positions and indications in these reports imply the problem that no as-built inspection had been performed for the containment venting components, showing the need of checking if licensees themselves can correctly perceive the as-built situations of equipment for ensuring the safety of nuclear reactor installations. Since the AM systems such as containment venting are intended to be used in the case that safety-related systems have already been unavailable, in particular, it should be raised as an issue that the operating procedures and operator training had not been provided based on such on-the-spot actions.

4.2 Containment Integrity at Unit 2

(a) Pressure drop in suppression pool

The suppression pool pressure gauge went off the scale low at about 6:14am on March 15, 2011. The TEPCO report estimates that the pressure drop was caused by a failure of pressure instruments. On the other hand, the Diet report assumes that the pool water temperature was too high to condensate vapor, steam bubbles were formed on the water surface, and then intermittent or continuous vibrations occurred, resulting in a burst or large scale damage in the suppression pool. Since, however, the drywell pressure was increasing afterwards, it is unlikely to assume that the burst would have occurred in the suppression pool as pointed out in the Diet report, and it is reasonable to presume the pressure instrument would have failed as indicated in the TEPCO report.

(b) Implementation of containment venting

The TEPCO report states that several attempts on containment venting were unsuccessful (at about 11:00 on March 13, 16:00 on March 14 and 23:30 on March 14). The Government and Diet reports also indicate the same positions. On the contrary, the NISA report says that a small amount of gases might have been released in a short time through the small venting valve with no change of the containment pressure, according to the rapid increase of dose rate on the site at about 21:00 on March 14 and the subsequent containment pressure rise, though the venting would not have fully functioned. The report also indicates that the rupture disk on the drywell venting line might have opened, leading to a large amount of radioactive materials being released at 0:00 on March 15. However, in the report, the venting is considered one of the possible pathways, through which the radioactive materials were released, and it is likely that the containment leakage might have occurred.

As mentioned above, there are some differences in their respective positions about whether or not the containment venting was successful but it is thought that these differences have not affected the accident progression so much. The RJIF report states that although it is not clear whether or not the containment venting was carried out, the heat could have been released from the containment to the environment earlier if the venting line would not have been equipped with any rupture disk. Since rupture disks are passive components and in general, their reliability and leak-tightness are higher than those of active ones, they are considered suitable for ensuring the containment venting on demand without erroneous operation.

(c) Containment pressure trends while RCIC operating

For the containment pressure behaviors while RCIC operating, the TEPCO report estimates a scenario in which seawater might have entered the torus room and removed heat from the containment due to heat transfer through the suppression pool wall, leading to the gradual containment pressure rise. The estimation is based on the fact that the torus room at Unit 4, the structure of which is similar to the torus at Unit 2, was almost half flooded. The NISA report supports this position. On the other hand, the Government report presumes another scenario in which the containment might have leaked due to overtemperature in the early stage of accident and then the leakage might have gradually been enlarged. In addition, the report expresses its objection to the TEPCO's estimation on seawater ingress into the torus room for the following reasons; the RCIC room located at the first basement of Unit 3 reactor building had not been submerged and it is unlikely that the RCIC room of Unit 2 had been flooded to the same extent because RCIC has been operating continuously. There is no evidence and no data to determine if seawater entered the torus room or if the containment leakage was enlarged and therefore, the further and in-depth survey and examination are required on this matter.

5. Hydrogen Explosion (Hydrogen Flow Paths into Reactor Building)

The TEPCO report describes as follows on hydrogen explosions at Units 1, 3 and 4.

(a) At Unit 1: Hydrogen was generated by the Zr-H_2O reaction in the process of core melting and then, transferred into the containment. In the containment, seals in flanges of top head, equipment and/or personnel hatches and those in electrical penetration assemblies might have failed due to overtemperature, deteriorating the containment integrity. Hydrogen passed through the failed seals into the reactor building. Such a direct leak path is highly likely though another pathway is supposed as the leakage from the containment through the standby gas treatment system (SGTS) lines when venting.

(b) At Unit 3: As well as at Unit 1, two pathways, that is, the direct leak path and the leakage through the SGTS lines are assumed. It is thought that the former is a main pathway and the latter's contribution is limited because (i) the SGTS inlet valve, which connects the venting line to SGTS, is a normally closed isolation valve and is designed to close on loss of power ("fail closed" design) and as well, it is unlikely that venting gases leaked into the SGTS lines according to the dose rates measured upstream and downstream of the inlet valve, (ii) while the containment pressure remained high for a long time, the venting was carried out in a short time and so, the venting gases were allowed to flow into the reactor building through the STGS lines for the limited time, and (iii) radioactive materials could have been removed from the venting gases due to scrubbing effects in the suppression pool but high radiation doses were observed near the reactor building after the explosion.

(c) At Unit 4: Their respective venting lines of Units 3 and 4 are joined just before the inlet of stack. The radiation doses were measured on the SGTS filters (see **Figure V.4**). The results showed that the highest dose was observed on the filter train outlet (downstream) and the dose gradually decreased towards the filter train inlet (upstream). This means that the contaminated gases passed through this line from the downstream side to the upstream one. Thus, it is highly likely that the venting gases flowed from the Unit 3 reactor building through the SGTS lines into the Unit 4 reactor building.

The Government and NISA reports basically agree with the TEPCO's positions above mentioned though the Government report does not exclude the possibility of reverse gas flow (from the containment through the SGTS lines into the reactor building) when venting at Unit 1. For the explosion at Unit 4, on the other hand, the Diet report addresses the contribution of hydrogen generated by radiolysis and released by vaporization in the spent fuel pool (SFP), in addition to the contribution of the leak flow from Unit 3.

Their views that the hydrogen having leaked directly from the containment is a dominant contributor to the explosion at Unit 1 are acceptable, considering the fact that the containment had been maintained at the pressure of about 0.75 MPa until the explosion occurred at 15:35 on March 12. However, it should be noted that the containment venting had been executed one hour before the explosion and hence, the possibility of reverse flow should not be excluded.

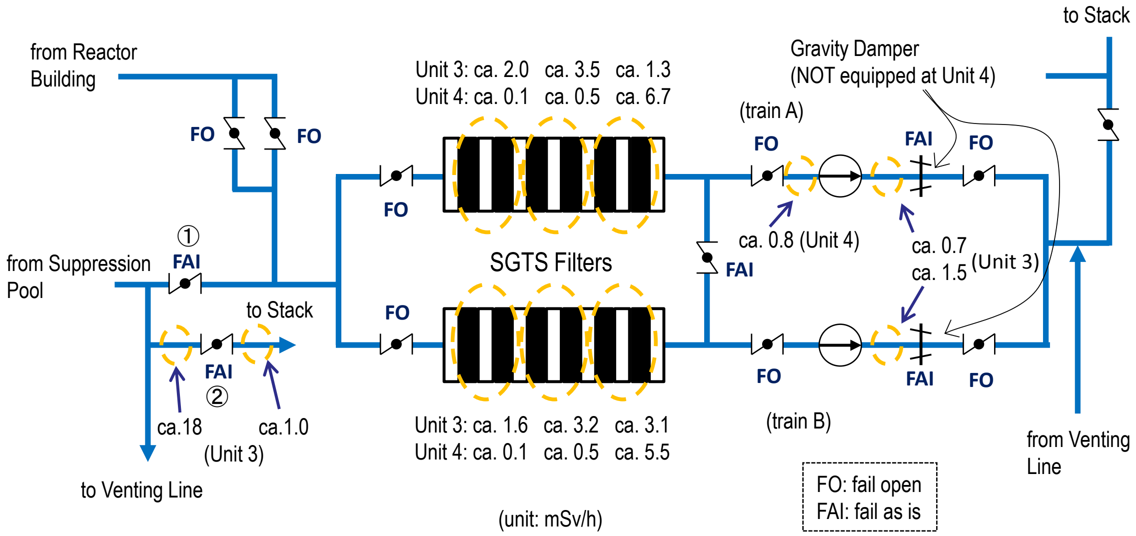


Figure V.4 Dosimetries at Standby Gas Treatment System Filters for Units 3 and 4

(This figure is prepared by modifying the figure provided in Ref.(2).)

Since the hydrogen explosion occurred at Unit 3 immediately after the fourth containment venting was conducted, it is likely that the hydrogen might have flowed back into the reactor building when venting and contributed to the explosion. Based on the radiation doses measured on the SGTS filters at Unit 3, the TEPCO report indicates as follows: different trends of doses were observed in two filter trains; the doses were lower than those measured on the filters at Unit 4; the doses measured downstream of the valve connected to the stack (② in Figure V.4) were significantly low, indicating the valve leak-tightness would have been maintained; and the valve connected a venting line to the SGTS inlet (upstream) (① in Figure V.4) is the same design as this valve. Thus, the report assumes a low likelihood of leakage from this line. However, it is thought that the basis of the above TEPCO's assumption is not necessarily enough because (i) the dose on the filter in the center was the highest, (ii) a total amount of doses at Unit 3 were higher than those at Unit 4, and (iii) as described above (b), it is stated that radiation dosimetry was performed for the valve connected a venting line to the SGTS inlet (① in Figure V.4) but this statement is inconsistent with Figure V.4 (in this figure, no measured dose is indicated for the valve ①). The Government report expresses its view that it is not unnatural that the radiation doses on the SGTS filter at Unit 4 were comparable to those at Unit 3 since the radioactive materials adhered and deposited on the inner surface of piping by condensation while passing through the relatively long SGTS piping from

Unit 3 to Unit 4. However, this view does not deny the possibility of a large amount of hydrogen flowing back to the Unit 3 reactor building. Considering the fact that the containment pressure has been maintained at about 0.4-0.5 MPa, excluding the short period after the SRV was opened (during this period, the containment pressure exceeded 0.8 MPa), the containment pressure did not necessarily exceeded the design pressure for a long time and this is inconsistent with the TEPCO's position that the containment pressure has remained high, as mentioned above (b). As well, the containment pressure increased and decreased before and after the venting, indicating that no large containment leakage occurred any time. Based on these containment pressure behaviors, it seems non-negligible that the hydrogen which flowed back when venting might have contributed to the explosion in the Unit 3 reactor building.

Although, on the other hand, the Diet report addresses the possibility that hydrogen generated in the Unit 4 SFP might have been a dominant contributor to the explosion at Unit 4, this view is not consistent with the fact that the third and fourth floors of the building deformed downwards, the fifth floor was damaged, and its reinforcing steel was bent up. Thus, the possibility seems highly unlikely.

V.4 Undiscussed Issues

This section discusses the technical issues with which none of the five investigation reports dealt. These are related to the design concepts and operating procedures.

(1) Adequacy of operating procedures applied at Unit 1

The TEPCO report clearly states that the operating procedures for transients with MSIVs closed were applied to the operations of Units 2 and 3, but for Unit 1, just says that the operation was executed according to procedures, without specifying operating procedures applied. On the contrary, the TEPCO's submittals to NISA show that the operating procedures for transients with MSIVs closed were applied to the Unit 1 operation and insist that the operation was adequately executed in accordance with those procedures without any problem, as well as Units 2 and 3[9,10,16]. However, those procedures specify that when any MSIV cannot be reopened, the operators use SRV(s) or IC(s) to depressurize the reactor and bring it into cold shutdown, consuming steam by operating HPCI in test-line mode. Concretely, the operators are required to depressurize the reactor by (i) manually opening a SRV, (ii) manually starting the HPCI test-line operation (if water injection is not needed), or (iii) using ICs (see **Table V.2**). In the event at

Tsuruga Unit 1, on the other hand, one IC train was manually started to depressurize the reactor in accordance with the operating procedures(15). Although ICs automatically started six min after the earthquake took place in Fukushima Dai-ichi Unit 1, the auto-start of ICs is not consistent with the operating procedures and also different from those of Tsuruga Unit 1. Considering that RCIC was manually started at Unit 2 four min after the earthquake, the operators might have manually started IC(s) according to the procedures. In addition, the HPCI test-line operation causes the reactor to be rapidly depressurized, as actually observed at Unit 3, by releasing steam to the turbine and the opening of SRV(s) also leads to rapid reactor depressurization. The reactor pressure behaviors by these operations are completely different from the pressure being maintained at 6-7 MPa by the IC operation. Therefore, there seems no consistency of basic concepts in preparing the operating procedures and these procedures should be validated and verified through the comparison with those of Tsuruga Unit 1. Furthermore, for the other procedures, it seems necessary to check and examine their basic concepts by comparing with those of other plants.

Table V.2 Outlines of Accident Operating Procedures for Reactor Scram with MSIVs Closed (Excerpt)

(1)	"1. Outlines of Accident", "(2) In the case that any MSIV cannot be reopened": Depressurize the reactor by using SRV(s) or IC(s), consuming main steam by operating HPCI in test mode, to bring the reactor into cold shutdown.
(2)	"2. Points of Operation": (6) In the case that the condenser hot well level decreased by the MSIV closure, operate HPCI manually to prevent the condensate water system from tripping and to consume main steam.*1 (8) In the case that any MSIV cannot be reopened, depressurize the reactor and cool the core by using SRV(s) or IC(s).
(3)	"4. Flowchart" *2: Control the reactor pressure by using SRV(s). Operate HPCI in the case that the condenser hot well level decreased.
(4)	"6. Reactor Pressure Control": "9. When the reactor pressure increased, open SRV(s) manually or use IC(s) to maintain the reactor pressure in the range of 6.27-7.06 MPa." "10. If the condenser hot well level decreased, operate HPCI manually to maintain the reactor water level."
(5)	"In the case that any MSIV cannot be reopened" (including the case that the condenser vacuum is 67.4 kPa abs or higher and the case that the condenser vacuum was lost), "12. Reactor Depressurization" *3: "3. Depressurize the reactor by (1) opening SRV(s) manually, (2) operating HPCI in test mode (in the case that no coolant injection is required) or (3) using IC(s)."

*1 It seems that the different operator response is required because the condensate water system trips in the case of loss of offsite power.

*2 The operation of ICs is not specified in the flowchart of operating procedures.

*3 While it is specified that the reactor cooldown rate limit of 55°C/h be met, this section of operating procedures does not require the operators to maintain the reactor pressure at 6.27–7.06 MPa.

While the TEPCO report states that the operating procedures had been prepared for the case of reactor scram due to huge earthquake (earthquake-induced reactor scram) and the case of LOOP, these procedures are not referred to at all and are assumed to have not been applied. In this accident, the offsite power was lost due to the earthquake and the actions to be taken might have been different from those for transients with MSIVs closed. The adequacy of applying the procedures for transients with MSIVs closed should be examined and it should be determined whether or not the procedures for huge earthquake or LOOP were applied. If not, the reasons should be clarified. In addition, the operating procedures for the case of earthquake-induced reactor scram with LOOP had already been prepared and the procedures specify that the operators manually initiate the HPCI pump to maintain the reactor water level and manually open and close SRV(s) to control the reactor pressure[19]. However, the TEPCO report does not refer to the procedures at all. It is also necessary to clarify why the procedures were not applied.

The operating procedures for transients with MSIVs closed require the operators to confirm that all of MSIVs fully closed and then to check and report that SRV(s) operated successfully. This means that the procedures presume the operation of SRV(s). However, both the actual reactor pressure trends and the analytical results with RELAP5 by JNES show that after the closure of MSIVs, ICs automatically started before the reactor pressure reached the setpoint for opening SRVs. This pressure behavior is different from that assumed in the procedures. On the contrary, the analytical results with evaluation models (EMs) shown in the safety analysis report of Unit 1[20] indicate that after the closure of MSIVs, the reactor pressure increases, SRV(s) opens, the reactor pressure is suppressed to 7.7 MPa or lower, and then, the pressure is controlled by cyclic operation of SRV(s). Based on these results, it can be assumed that the operating procedures for transients with MSIVs closed might have been prepared by referring to the analytical results with EMs. Originally, the analyses with EMs are intended to provide conservative results and hence, do not simulate actual plant behaviors. Licensees should reference to the analytical results with best estimate codes, which can provide more realistic plant behaviors, when preparing the operating procedures. It is considered necessary to check if any difference exists between assumptions in operating procedures and actual behaviors for all postulated transients and accidents at nuclear power plants.

It is thought that the operators should have manually started one IC train instead of waiting for auto-start of ICs, like the event at Tsuruga Unit 1. At Tsuruga Unit 1, the manual startup of IC is supposed to be executed at the reactor pressure of 6.8 MPa, which is lower than the setpoint for its auto-start (7.2 MPa)[15]. As well, the operating procedures for transients with MSIVs closed do not give a priority to use of ICs at Fukushima Dai-ichi Unit 1 and the operational concept on use of ICs is different from that at Tsuruga Unit 1. It is needed to identify the reasons of such a difference. Moreover, at Tsuruga Unit 1, the operator training has been carried out using a simulator of ICs. Practices of reflecting the simulator training into the procedures should be established, particularly for passive systems such as ICs, since it is very difficult or impossible for operators to understand the plant behaviors at the time of their operation, by a walkdown, on-the-job training and/or routine testing of valve operation. Although Unit 1 experienced at least three incidents involving the MSIV closure in 1980s(18), ICs had not been used at all since its commercial operation was commenced according to the Diet report.

(2) Design concepts of valves in standby gas treatment system (SGTS) on loss of driving forces

At Units 1-4, all the inlet and outlet valves in SGTS are air-operated valves (AOVs) and are designed to open on loss of compressed air in the case of LOOP (so-called “fail open” design). However, this is not always a standard design. For example, at Shika Unit 2, motor-operated valves (MOVs) are used, which are designed to remain as is when their driving forces are lost (so-called “fail as is” design)(21). Because SGTS is one of the safety systems and its driving force is supplied from emergency power systems, both the inlet and outlet valves are automatically opened on demand even in the case of LOOP and the subsequent loss of emergency power makes these valves to remain open, resulting in similar conditions to those at Fukushima Dai-ichi Units 1-4. However, the design concepts on loss of driving force are completely different between “fail open” and “fail as is” and thus, such design concepts for these valves should be clarified by examining the actual designs at the other plants.

(3) Design concepts of valve configuration on containment venting lines

The containment venting line is equipped with a normally-closed MOV and a normally-closed AOV in series outside the containment boundary as well as a rupture disk, setpoint of which is higher than the design pressure of containment. These valves and the rupture disk isolate the containment during normal operation. It seems that the reason why two normally-closed valves and the rupture disk were equipped in series is to meet the requirement on containment isolation specified in the Safety Design Review Guides(12). Considering the design feature of rupture disk, the requirement might have been met by equipping one normally-closed valve and the rupture disk. If the requirement on containment isolation would have been more scrutinized when designing the containment venting lines, in other words, the venting operation might have been carried out in a more effortless way.

Additionally, the venting line is connected to the SGTS inlet and joins the venting line of the neighboring unit just before the stack. Although it was necessary to ensure that the

SGTS would not be adversely affected by connecting the venting line, the "fail safe" design of the SGTS valves (i.e., the design that the valves automatically open in the case of LOOP) was not sufficiently examined because SBO and loss of DC power had not been postulated in the design of SGTS. As a result, lack of such considerations might have led to the hydrogen explosion at Unit 4 and contributed to those at Units 1 and 3.

If the rupture disk would be placed inboard and the normally-closed valve would be outboard of the containment, the manual operability of valve is expected to be improved. It seems essential to check if any consideration had been taken in configuring venting lines.

(4) Operating basis on startup of containment cooling systems

During the period from the reactor scram to SBO, the containment spray or the suppression pool cooling by residual heat removal system (RHR) was started up at Units 1 and 2 to respond to continuously gradual increase of containment pressure and temperature since the containment ventilation systems tripped due to loss of normal power. On the other hand, at Unit 3, no suppression pool cooling was carried out even though the containment pressure and temperature have gradually been increasing due to steam inflow through cyclic operation of SRVs and the RCIC operation. According to the Government report, the operators at Unit 3 were concerned that the RHR seawater pumps would be damaged by the kataseism (dilatational wave) because the large tsunami warning had been announced. As a result, the suppression pool cooling had not been conducted at all. Considering the differences in operators' responses at Units 1, 2 and 3, the adequacy and reasonableness, including compliance of operating procedures, should be examined and the operating procedures should specify, in advance, actions to be taken when the tsunami may hit the site.

V.5 Concluding Remarks

This chapter delineates, at first, the severe accident scenarios at Fukushima Dai-ichi Units 1 to 3 with use of event trees to clarify the differences among accident sequences at Units 1 to 3 and discusses the actual responses to avoid the severe accidents using the event trees. Next, the differences in their respective positions are clarified by reviewing the five investigation reports from the technological point of view, focusing on the accident progression and causes to specify the issues to be further examined. Furthermore, identified are the technical issues which are not discussed in these reports.

In the following, the different positions among the reports are summarized.

(1) Electric Power Supplies: The independence of offsite power transmissions and the aseismic design of substations are disputed as a problem, but the more important thing is how these substations should be taken into account in the nuclear safety regulations. For the onsite electric power systems, a problem that many components had been laid at the specific locations is pointed out and thus, the basic concept of such layouts needs to be discussed in detail. In addition, it is indicated that the tripping of EDGs might have not been caused by the tsunami-induced flooding, based on the arrival time of tsunami. Considering the fact that two or more EDGs tripped almost simultaneously, it is natural to regard the tsunami-induced flooding as the direct cause.

(2) Core Cooling: The possibility of the IC pipe leakage is addressed as the reason of manual trip of ICs, but its bases are not necessarily apparent and not enough to claim that the view that ICs had been manually tripped to observe the operational limit on cooldown rate is unreasonable. As well, it should be noted that the design concepts of control logic for closing isolation valves on the steam supply lines are different between ICs and HPCI/RCIC, instead of disputing the adequacy of applying the terminology, "fail safe", to the control logic. The operations after the scram due to the earthquake, particularly the IC operation, should be further analyzed to determine their adequacies. As for the RCIC operation at Unit 2, it is necessary to analyze how the operators understood the continuous operation of RCIC in the case of loss of DC power, in addition to the fact that the operators did not fully monitor the suppression pool temperature and pressure after the switchover of water source. If RCIC might have tripped due to the dislodged latch in the valve at Unit 3, as pointed out, it is also necessary to review the past maintenance records in detail and to check if any deficiencies were involved in maintenance practices. Indicated is the critical view that alternative water injection should have been configured prior to manual trip of HPCI and then SRVs should have been opened at Unit 3. Considering the fact that the containment venting was successful afterwards, further discussions should be made on the pros and cons of the long-term HPCI operation in test-line mode.

(3) Reactor Depressurization: Although pointed out is the possibility that SRVs had not been operated at all and the coolant flowed out from the broken pipe(s) at Unit 1, its bases are not necessarily apparent. For SRVs at Unit 3, the fact that any operator training had not been conducted on the operation of SRVs under low reactor pressure is acknowledged as a problem but it is more noteworthy what kind of training programs the operating staff had received on specifications and functional restrictions of SRVs.

(4) Containment Integrity and Venting Operation: As indicated, only tabletop exercises had been conducted on venting operation. As well as such training practices, it should be noted that any as-built inspection and/or check had not been carried out. While a possibility of the burst having caused the pressure drop in the Unit 2 suppression pool is pointed out, it is highly unlikely that the burst took place because the drywell pressure increased afterwards. For the containment pressure behavior at Unit 2, two possibilities are assumed; one is that seawater flowed into the torus room and the other is the containment leakage. These two possibilities should be further examined.

(5) Hydrogen Explosions: The hydrogen explosions at Units 1 and 3 are presumed to have occurred due to mainly hydrogen leaking directly from their respective containments and the explosion at Unit 4 is assumed to be caused by hydrogen reversely flowing from the Unit 3 containment when venting. Based on the fact that the containment venting was executed shortly before the explosions at Units 1 and 3, however, the contribution of venting gases should be further discussed.

Also, the following undiscussed issues are identified.

(1) Adequacy of Operating Procedures Applied at Unit 1: The operating procedures applied at Unit 1 assume that SRVs would open and close automatically after the reactor scram and thus, the scenario is different from the actual plant behavior. Also, the procedures do not specify that ICs be positively used. In-depth analyses are required on the basic concepts for preparing the operating procedures. In addition, the procedures for the earthquake-induced reactor scram had been prepared but it is unknown whether those procedures were used. If not, the reason should be clarified.

(2) Design Concepts of SGTS Valves on Loss of Driving Forces: The SGTS inlet and outlet valves connected to the containment venting lines are designed to open when losing their driving forces (so-called "fail open" design). It is necessary to review the design concepts for these SGTS valves in detail.

(3) Design Concepts of Valve Configuration on Containment Venting Lines: The design concepts should be examined about the configuration of venting lines consisting of one rupture disk and two normally-closed valves placed in series upstream of the disk.

(4) Operating Basis of Startup of Containment Cooling Systems: While the suppression pool cooling was initiated after the earthquake occurred at Units 1 and 2, RHR was not started up in consideration of the damages to pumps due to tsunami-induced kataseism at Unit 3. It is necessary to identify the causes of such a difference in operations.

A couple of investigation reports underline the importance of clarifying the effects of the earthquake on the plant facilities in the future. In order to obtain better understandings of the severe accident progression, as well, it is important to investigate the phenomena which influenced the plant behaviors, to identify the damages which might have affected the accident progression and to clarify the conditions for water injection and reactor depressurization by conducting onsite surveys, if possible.

This study highlights the need to further discuss and examine the technical safety issues so that the insights and lessons learned from the Fukushima accident can be fed back to other nuclear facilities as well as power plants domestically and worldwide.

References in Chapter V

(1) Tokyo Electric Power Company, Inc., *Fukushima Nuclear Accident Analysis Report*, 2011 (*www.tepco.co.jp/en/nu/fukushima-np/interim/index-e.html*). (as of December 2013)

(2) Tokyo Electric Power Company, Inc., *Fukushima Nuclear Accident Analysis Report*, 2012 (*www.tepco.co.jp/en/nu/fukushima-np/interim/index-e.html*). (as of December 2013)

(3) Investigation Committee on the Accident at the Fukushima Nuclear Power Stations, *Interim Report*, 2011 (*www.cas.go.jp/jp/seisaku/icanps/eng/interim-report.html*). (as of December 2013)

(4) Investigation Committee on the Accident at the Fukushima Nuclear Power Stations, *Final Report*, 2012 (*www.cas.go.jp/jp/seisaku/icanps/eng/final-report.html*). (as of December 2013)

(5) The National Diet of Japan, *The official report of The Fukushima Nuclear Accident Independent Investigation Commission*, 2012 (*warp.da.ndl.go.jp/info:ndljp/pid/3856371/naiic.go.jp/en/report/*). (as of December 2013)

(6) Rebuild Japan Initiative Foundation (RJIF), *The Report of The Independent Investigation Commission on the Fukushima Nuclear Accident*, 2012 (in Japanese).

(7) Nuclear and Industrial Safety Agency, *Technical Knowledge of the Accident at Fukushima Dai-ichi Nuclear Power Station of Tokyo Electric Power Co., Inc. (Provisional Translation)*, 2012.

(8) N. Watanabe, T. Yonomoto, H. Tamaki, T. Nakamura and Y. Maruyama, *Review of Five Investigation Committees' Reports on the Fukushima Dai-ichi Nuclear Power Plant Severe Accident – Focusing on Accident Progression and Causes*, Transaction of Atomic Energy Society of Japan, Vol.12, No.2, pp.113-127, 2013, (in Japanese).

(9) Tokyo Electric Power Company, Inc., *Application of Accident Operating Procedures at Fukushima Dai-ichi Unit 2 after the Tohoku District-off the Pacific Ocean Earthquake*, submitted to NISA on October 28, 2011 (in Japanese), posted on Web-site (*www.meti.go.jp/press/2011/12/20111208002/20111208002.htm*) on December 20, 2011. (as of December 2013)

(10) Tokyo Electric Power Company, Inc., *Application of Accident Operating Procedures at Fukushima Dai-ichi Unit 3 after the Tohoku District-off the Pacific Ocean Earthquake*, submitted to NISA on October 28, 2011 (in Japanese), posted on Web-site (*www.meti.go.jp/press/2011/12/20111208002/20111208002.htm*) on

December 20, 2011. (as of December 2013)

(11) USNRC Regulations, *Title 10 Code of Federal Regulations Part 50 Appendix A "General Design Criteria for Nuclear Power Plants,"* 1971 (last amended in 2007).

(12) Nuclear Safety Commission of Japan, *Safety Design Review Guides for Light Water Power Reactors*, 1990 (last amended in 2001), (in Japanese).

(13) The Federation of Electric Power Companies of Japan, *Layout Design of Emergency Power Boards at BWRs in Japan*, No.3-1-1, 3rd Meeting of Subcommitte on Safety Design Review Guides under Standing Committee on Nucelar Safety Standards and Guides (held on August 23, 2011), 2011 (in Japanese).

(14) The Federation of Electric Power Companies of Japan, *Layout Design of Emergency Power Boards at PWRs in Japan*, No.3-1-2, 3rd Meeting of Subcommitte on Safety Design Review Guides under Standing Committee on Nucelar Safety Standards and Guides (held on August 23, 2011), 2011 (in Japanese).

(15) The Japan Atomic Power Company, *Response to "Request for Submitting the Operation Records on Actuation of Isolation Condensers at Tsuruga Unit 1,"* submitted to NISA on November 18, 2011 (in Japanese).

(16) Tokyo Electric Power Company, Inc., *Application of Accident Operating Procedures at Fukushima Dai-ichi Unit 1 after the Tohoku District-off the Pacific Ocean Earthquake*, submitted to NISA on October 21, 2011 (in Japanese).

(17) T. Watanabe, et al., *Analysis of BWR Station Blackout Accident*, Journal of Atomic Energy Society of Japan, Vol.10, No.4, pp.240-244, 2011, (in Japanese).

(18) Japan Nuclear Safety Institute, *NUCIA (www.nucia.jp)*, (in Japanese). (as of December 2013)

(19) Tokyo Electric Power Company, Inc., *Accident Operating Procedures for Reactor Scram due to Huge Earthquake at Fukushima Dai-ichi Unit 1*, posted on Web-site (*www.nsr.go.jp/archive/nisa/earthquake/manual/manual_index.html*) on December 20, 2011 (in Japanese). (as of December 2013)

(20) Tokyo Electric Power Company, Inc., *Safety Analysis Report for Applying Design Modification of Fukushima Dai-ichi NPP*, 1991 (in Japanese).

(21) Hokuriku Electric Power Company, *Response to Questions from Members of Stress Test Committee in NISA*, No. ST 12-1-2, 12th Stress Test Committee of NISA (held on March 29, 2012), 2012 (in Japanese).

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Chapter VI

Discussion on Insights and Precursors to Fukushima Accident

This chapter discusses the insights obtained through the analyses described in the previous chapters in terms of their commonalities in the Fukushima accident, identifies precursors to the Fukushima accident, and addresses generic issues to be resolved.

VI.1 Insights from Analysis of Operating Experience

The generic study on the INES report indicates that most of the events at nuclear power plants were rated on the "defense-in-depth" criterion and that these events have different characteristics depending on the reactor types but the degraded functions of emergency core cooling system (ECCS) and electric power systems were common issues to all types of reactors(1). This means that the plant safety have been actually challenged by the degradation of defense-in-depth, in particular ECCS and electrical power supplies. Considering that the Fukushima accident involved loss of AC/DC power and loss of core cooling, the defense-in-depth concept should be incorporated into the plant operation as well as design more strictly to ensure the plant safety.

The comprehensive reviews of IRS reports identified a various types of safety significant events including loss of offsite power (LOOP), failures of emergency diesel generators (EDGs), loss of heat sink, etc. In particular, many recurring events were identified, indicating that the previous corrective actions have not necessarily been effective to prevent the recurrences. This implies the necessity of examining the robust approach to take effective measures. As well, some events are considered precursors to the Fukushima accident, which will be discussed in the following section. As for the external hazards, some plants experienced the events beyond the design basis and thus, more attention should have been paid to those taking into account the challenges they pose and the unexpected conditions they may cause.

The topical study on loss of decay heat removal demonstrates that the decay heat removal function should be restored in a short term during shutdown. In the Fukushima accident, decay heat removal was lost and not recovered at Units 1-3, resulting in severe core damage. Fortunately, at Units 5 and 6 in shutdown, one EDG was operable and the condensate transfer pumps were used for core cooling in a timely manner, bringing these units into safe shutdown conditions. This underscores the necessity of enhancing the diversity of decay heat removal as well as its driving force and of recovering the heat removal within a short time. Another topical study on safety/relief valve (SRV) setpoint drift shows the potential failure to open of SRVs, that is, one of the safety significant issues for overpressure protection. In the Fukushima accident, manual opening of SRVs could not be done because of no driving force (compressed air). Since SRVs play an important role of reactor depressurization to allow low pressure coolant injection to operate, their functionabilities should be maintained even though the driving forces are lost. The topical study on the criticality accidents indicates the common issues/problems among the accidents: the violation of procedures/rules, the changes of procedures without regulatory authorization, unexpected use of equipment, etc. These were mainly caused by the priority being given to the production or work efficiency. In addition, lack of understanding of criticality hazards was also one of contributing factors. For the Fukushima accident, management issues such as cost-reduction and capacity factor enhancement were considered one of the contributors. As well, it was pointed out that the results from probabilistic risk/safety assessment (PRA/PSA) showed lower core damage frequencies compared with those in foreign countries and hence, safety measures had been perceived as being already enough to prevent severe accident. As a result, further improvements to accident management had not been taken into account(2). This implies that the risks of severe accidents had been downplayed. Although tsunami-induced core damage risks had been recognized, no PSA study for tsunami had been carried out and thus, any countermeasures had not been taken based on the literature survey results which indicated that no tsunamigenic earthquake would occur offshore of Fukushima. This also means the lack of understanding of tsunami risks. The common or similar issues identified in the JCO and Fukushima accidents may highlight the importance of learning the lessons from past operating experience at other types of facilities.

The accident sequence precursor (ASP) analysis of steam generator tube rupture (SGTR) events reveals that reactor depressurization is one of dominant factors to core damage sequences. Although the reactor type and initiating event are different, this ASP study and the Fukushima accident imply that the reactor depressurization might be one of key factors to place the reactor to a safe shutdown condition, in particular, in the case of transients with power conversion system unavailable. While manual depressurization of the reactor for such transients can be done by using the pressurizer relief valves and the main steam relief valves at PWRs, only the use of SRVs is available at BWRs. Therefore, alternate measures for manual depressurization should be considered at BWRs.

It is questionable whether or not the insights and lessons obtained from operating experience in foreign countries have adequately been fed back to the plant designs and operations and any additional measures have been taken in Japan. This implies that lack of attitude of learning from the past operating experience widely might have led to one of contributors to the Fukushima accident.

VI.2 Precursors to Fukushima Accident

1. Major Occurrences Contributing to Fukushima Accident

The Fukushima accident was initiated by earthquake and flooding due to subsequent tsunami. Although the earthquake itself might not have led to any significant damage to safety-related systems or structures, it caused the breakdown of external electric power grid in wide areas, resulting in LOOP at the Fukushima Dai-ichi Nuclear Power Plant. As a consequence, all the six units including three units in shutdown lost their offsite power.

The tsunami caused flooding of the site far beyond the design basis assumptions. This affected the units in a slightly different manner(3). At Unit 1, EDGs and DC batteries (or their associated electrical boards) were flooded and unavailable after the tsunami. The extended station blackout (SBO) with DC power unavailable resulted in loss of safety functions such as core cooling, pressure control, residual heat removal and containment cooling. At Unit 2, as well as Unit 1, EDGs and DC batteries were lost due to the flooding but the reactor core isolation cooling (RCIC) continued to run without its control for approximately 70 h. Unit 2 also experienced the extended SBO and failures of DC batteries, leading to loss of core cooling, loss of pressure control, loss of residual heat removal, and loss of containment cooling. On the other hand, at Unit 3, a part of DC batteries were available, allowing RCIC to be controlled and the high pressure coolant injection (HPCI) to actuate. Therefore, RCIC and HPCI operated for approximately 20 h and 14 h, respectively. After the batteries depleted, Unit 3 also lost the core cooling, pressure control, residual heat removal, and containment cooling because of the prolonged SBO. As Unit 4 was in shutdown, all the fuel assemblies had been placed in the spent fuel pool. Due to SBO and loss of DC power, the pool cooling was unavailable. At Units 1 to 4, the operators were forced to carry out the blind operation because all the indications of plant parameters in the main control room (MCR) were lost due to loss of DC power. Although Units 5 and 6 also was in shutdown, all the fuel assemblies have been loaded in the core. Since one EDG and DC power supply remained available at Unit 6 after the tsunami hit, the electric power was supplied to both Units 5 and 6, alternatively, leading to the core cooling with cyclic operation of SRVs and manual operation of condensate transfer pumps.

As a consequence, Units 1 to 3 at the Fukushima Daiichi Nuclear Power Plant experienced the following types of occurrences, which are major ones contributing to the severe core damage and characterized it, as well as the earthquake and tsunami exceeding design basis.

- LOOP (caused by earthquake)
- Flooding (resulting from tsunami)
- Loss of onsite power, unavailabilities of EDGs (caused by flooding)
- Long-term SBO
- Loss of DC power (caused by flooding)
- Loss of MCR (mainly caused by loss of DC power)
- Loss of core cooling, in particular at Unit 1 (caused by loss of DC power)
- Loss of ultimate heat sink, residual heat removal (mainly caused by tsunami)
- Loss of containment function (caused by long term loss of residual heat removal)

The relations among these occurrences are illustrated in **Figure VI.1**.

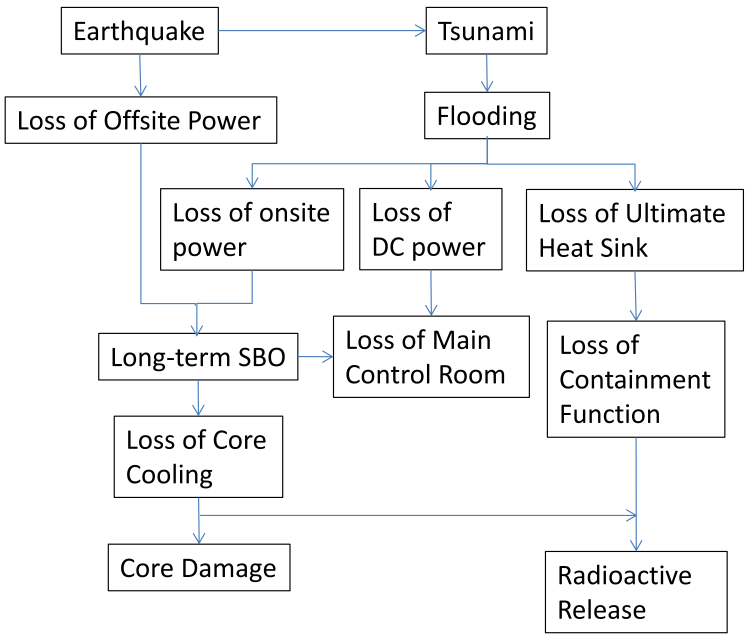


Figure VI.1 Relations among Major Occurrences in Fukushima Accident

Most of these occurrences had previously taken place at some plants and were identified as safety significant events in Chapter II even though the causes of events were different from those of the Fukushima accident. In the following, precursors to the Fukushima accident are discussed focusing on these occurrences and the beyond design basis events in the past.

2. Short Description of Selected Events

Of approximately 200 events identified as safety significant, six events were selected as precursors to the Fukushima accident listed below. These precursors were identified as events involving beyond design basis initiator and/or one or more occurrences which contributed to the severe accident at Units 1 to 3 of the Fukushima Dai-ichi Nuclear Power Plant.

- Event 1: fire, massive internal flooding and loss of safety systems
- Event 2: fire, prolonged station blackout involving loss of control room and loss of residual heat removal
- Event 3: external flooding involving partial loss of safety systems and multi-unit site issue
- Event 4: loss of offsite power with one of two emergency diesel generators inoperable
- Event 5: tsunami-induced external flooding
- Event 6: common-mode loss of instrument power

It should be noted that there may be more safety significant events and the selection is therefore not intended to be complete. Nevertheless, the selected events are good examples to discuss the precursors to the Fukushima accident. The following event descriptions are basically extracted from their respective IRS reports[4-9] but some of them can be seen also in published documents[10-12].

(1) Event-1

The major occurrences in this event are provided in **Figure VI.2**. The turbine blades failed, resulting in significant vibrations of the turbine. This led to an automatic trip of the No. 2 turbine generator (TG-2). Shortly afterwards, the reactor scrammed manually, TG-1 tripped and fires and explosion occurred in TG-2. The fire progressed due to lube oil spread and affected numerous cable trays, causing the unavailability of several safety equipment and/or their support systems. Also, smoke entered the electrical building and MCR, which forced the use of autonomous equipment.

The reactor building was flooded due to large leakage of water from several sources such as seawater from the TG-2 condenser cooling system, overfilling of the component cooling demineralized water tanks and water from the fire protection system. The flooding affected the equipment important for recovering the plant such as shutdown heat exchanger pumps, shutdown ventilation system compressors and spent fuel cooling pumps.

This event did not have any radiological consequences and the core was continuously cooled. However, the event constituted a serious threat to the plant safety since several safety systems failed, affecting the adequate coolability of the core. Plant design was insufficient to cope with common mode failures (mainly due to fire and flooding) and operating procedures were not available to manage such a complex event. In particular, there was neither level instrumentation in the reactor building nor other preventive measures as physical barriers or pedestals for safety equipment.

As summary, this event involved fire, flooding, loss of 48 VDC control busses causing loss of control from MCR, and partial loss of safety systems.

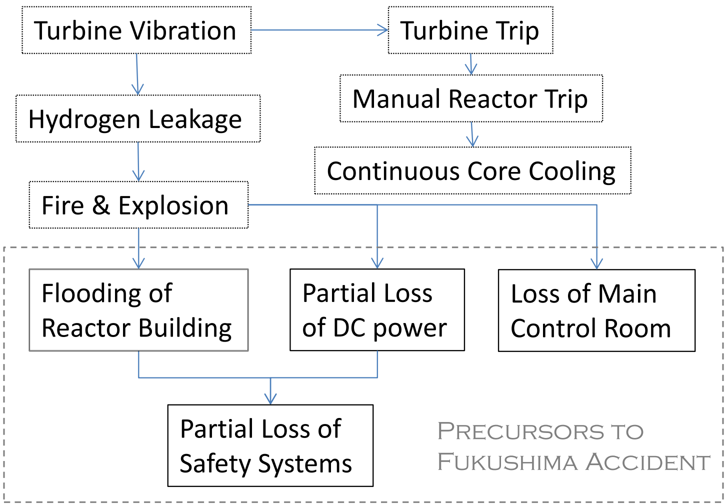


Figure VI.2 Major Occurrences in Event-1

(2) Event-2

The major occurrences in this event are shown in **Figure VI.3**. During the power operation, the turbine blades failed, resulting in turbine vibration and subsequent hydrogen leakage, and a fire took place in the turbine building, accompanied by a hydrogen explosion. The turbine generator tripped automatically and the reactor was immediately shut down. Due to the fire, electrical cables have been burned, resulting in complete loss of power supply in the unit. The core cooling was maintained by thermosyphon effects. As well, the diesel-driven fire pumps were started to inject water into the secondary side of steam generators (SGs). The atmospheric steam discharge valves (ASDVs) were kept open to reduce the pressure in SGs. The operators were forced to leave MCR because a large amount of smoke entered MCR and thus, an attempt was made to take charge of the situation from the emergency control room but it was impossible due to no indication on the unit panel available stemming from loss of power supply. MCR could be reoccupied after about 13 h.

During the incident, two EDGs automatically started but tripped due to loss of control power. The third diesel generator, common to two units, was started and after 6 h one of the busses could be charged. One of the shutdown cooling was started after 17 h. Thus, SBO can be considered to have lasted for about 17 h.

The prolonged SBO condition and consequent degradation of several safety systems were caused by cable fire and lack of proper fire barriers/fire retarding provisions together with physical separation in redundant safety-related cables.

As summary, the incident was a "Beyond Design Basis Accident", as an SBO was not considered during the design stage. Also, this event involved loss of MCR and no indications were available in the emergency control room. Therefore, the important parameters had to be measured directly in the field, resulting in the blind operation of the plant.

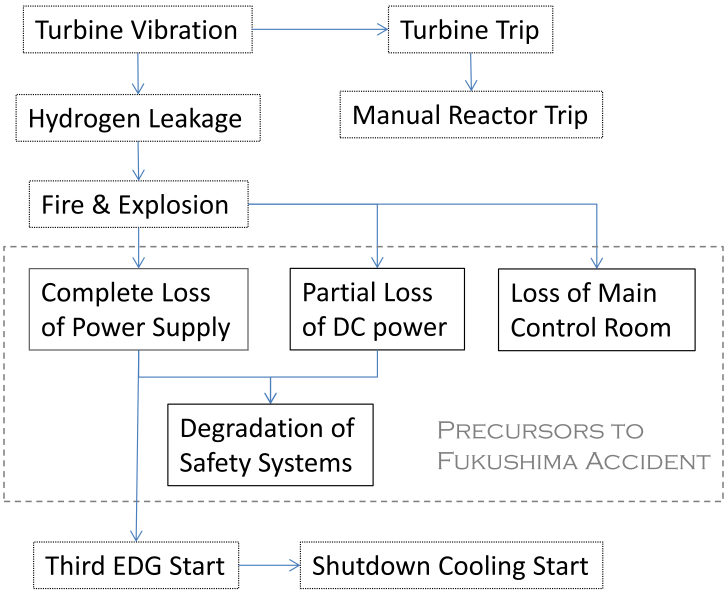


Figure VI.3 Major Occurrences in Event-2

(3) Event-3

Figure VI.4 displays the major occurrences in this event. Exceptionally bad weather (combination of very strong winds and precipitation with tide) resulted in the formation of waves moving up the river, causing partial flooding of four-unit site. At the same time, the 400 kV offsite power supplies to two units were lost for several hours and the 225 kV auxiliary power supply to all the units was lost for about 24 h. Several rooms were flooded because the cable way penetration and fire-door were damaged with water. Due to the flooding, the essential service water system (ESWS) train A pumps were lost as a result of immersion of their motors. In addition, the isolation faults occurred in the 48 VDC switchboards and the low pressure injection and containment spray pumps were declared unavailable because of their pits being flooded. Consequently, two units were shut down and placed to shutdown state by using SGs to cool the primary coolant.

During the first hours of the incident, the arrival of additional teams outside the plant was impossible due to the damage resulting from the tempest (flooding of the access routes, many tree falls, etc.). The plant staff initiated pumping operations to unwater the buildings.

This event highlighted inadequate protection measures against flooding. Hence, the major changes consisted of heightening the protection dyke facing the river and an anti-swell wall was built above the dyke.

As a result of this incident, the condition of all power plants was re-assessed in order to check the compliance with the existing baseline and incorporate rapidly the event feedback. The power plant protection methodology was also revised which led to develop a new methodology for each site. This approach was used to define and implement protection modifications when required.

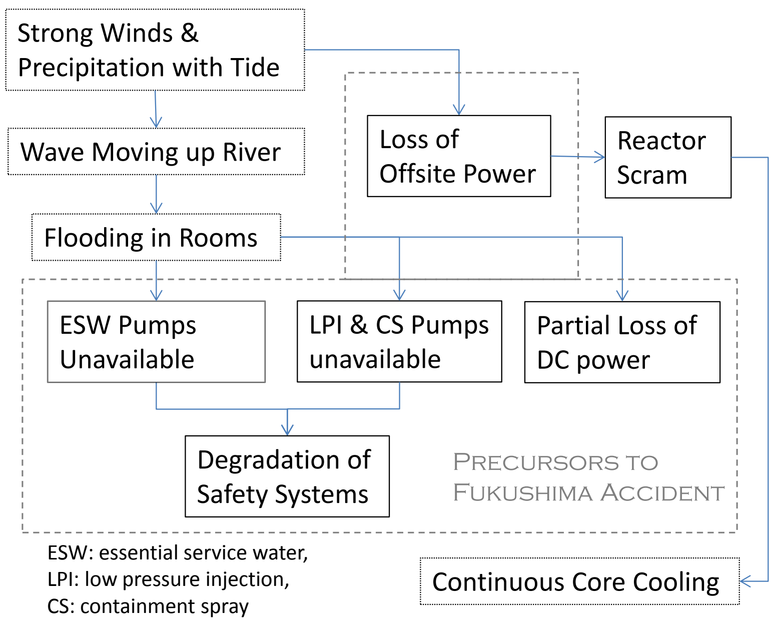


Figure VI.4 Major Occurrences in Event-3

(4) Event-4

As shown in **Figure VI.5**, in this event, two independent failures led to the unavailability of one of two EDGs and the loss of offsite power. The event was initiated by a relay failure on one of the two safety switchboards, resulting in being impossible to connect the EDG. During this incident, another fault on a generator breaker took place and as a result, the line breaker opened, disconnecting the reactor from the 400 kV main offsite power line. In addition, the instrumentation and control device used to switch over to the auxiliary offsite power supply had been cut, according to the required operating procedures. Consequently, the offsite power was totally lost.

The loss of offsite power led to a reactor scram and reactor coolant pump shutdown. As well, the remaining EDG automatically started up and supplied the electric power to the corresponding safety systems. Subsequently, the reactor core was continuously cooled by circulating reactor coolant through the secondary cooling system in thermosyphon mode (natural circulation). At the same time, preparations were started to connect the "Station Blackout EDG" to emergency switchboard. When offsite power was restored several hours after the event initiation, one of the reactor coolant pumps could be started up and forced circulation of coolant in the reactor coolant system became available, bringing the reactor unit to a safe state.

It should be noted that individual plants in this country underwent design changes in the 1980s to deal with a total loss of offsite and onsite power: installation of the "Station Blackout EDG" (mobile generator), together with an emergency turbine generator. This equipment would have guaranteed control of a more severe situation.

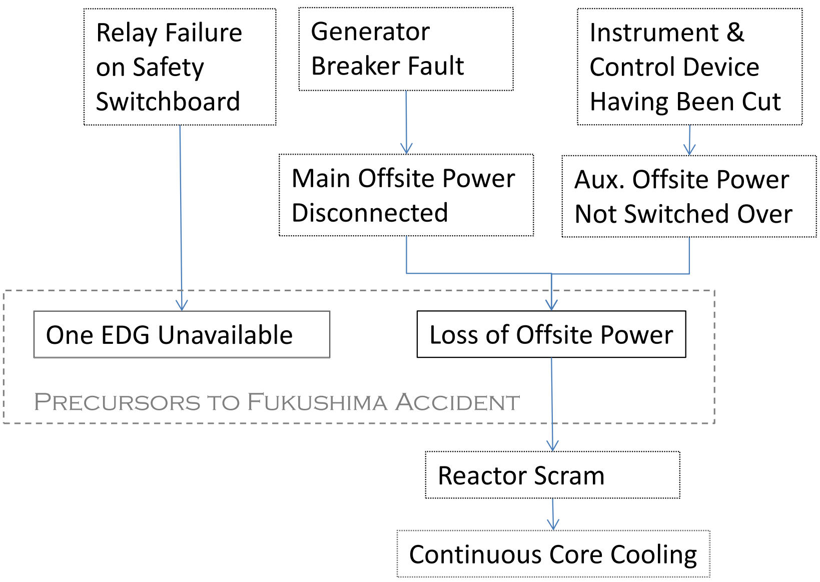


Figure VI.5 Major Occurrences in Event-4

(5) Event-5

Figure VI.6 shows the major occurrences in this event. A nuclear power plant site was flooded by the unexpected high waves, which was induced by the tsunami originating from a huge earthquake off the coast of island, thousands of kilometers from the site. As a result, seawater entered the pump house through the intake tunnel and the water level in the pump house increased up to the condenser cooling water (CCW) pump stool level, causing the tripping of these pumps. The MCR operator tripped the turbine and consequently the reactor tripped. The primary system was cooled down by opening the atmospheric steam discharge valves. Increase in water level in the pump house caused all the CCW pumps and all the process seawater (PSW) pumps unavailable except for one PSW pump. Additionally, the emergency process seawater (EPSW) pumps became unavailable because they got submerged in seawater. Offsite power remained available and the reactor was brought to a safe shutdown state. The sampling tank for collecting the liquid effluents got dislodged and the cement brick enclosure was washed away up to a length of about 60 m from the jetty entrance towards the sea.

As summary, in this incident, the plant safety was not affected because no water entered the reactor building, turbine building and the service building while the process water systems were affected by the tsunami. It should be noted that the site was partly inundated with the tsunami wave, resulting in non-safety structures being dislodged.

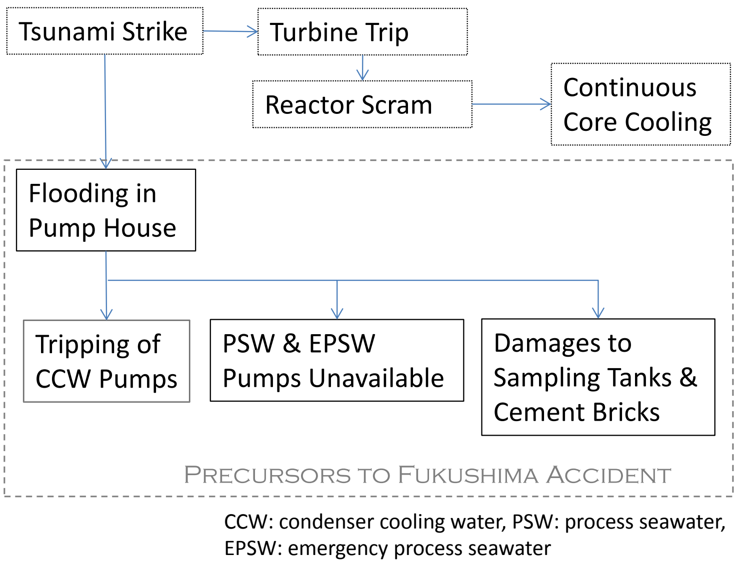


Figure VI.6 Major Occurrences in Event-5

(6) Event-6

The major occurrences in this event are shown in **Figure VI.7**. While operating at full power, an internal failure in the main transformer caused a generator, turbine and reactor trips. Following the generator trip, a momentary voltage decrease occurred on the in-plant electrical distribution system. The degraded voltage resulted in a simultaneous common-mode loss of five of eight non-safety related uninterruptible power supplies (UPSs) that powered important instruments in MCR and other plant equipment. Although the two safety-related UPSs were not affected, all indications of reactor control rod position were lost, resulting in the operators' inability to verify that the reactor would remain shutdown. Also, all MCR annunciators (alarms) were lost, hampering the operators' ability to monitor post-scram operation of the plant, and both the in-plant radios and the page telephone communications systems were unavailable, limiting the MCR communications with in-plant personnel. Other non-safety related systems/components which were lost included safety parameter display system, process computer and some plant lighting. These UPSs have internal continuously charged backup batteries to prevent a loss of control logic power but the backup batteries were past their useful life and were discharged due to inappropriate preventive maintenance. The UPS power was restored in about one-half hour and a normal cooldown was conducted. Internal deficiencies, common to all five UPSs but unknown to the plant staff, had made the power supplies susceptible to failure initiated by degraded voltage.

As summary, this incident did not pose any threat to the plant safety because automatic reactor protection system functioned properly and all engineered safety features were available and used as needed. The difficulty experienced by the operators because of loss of many normally available plant status indications underscored the importance of the lost UPSs.

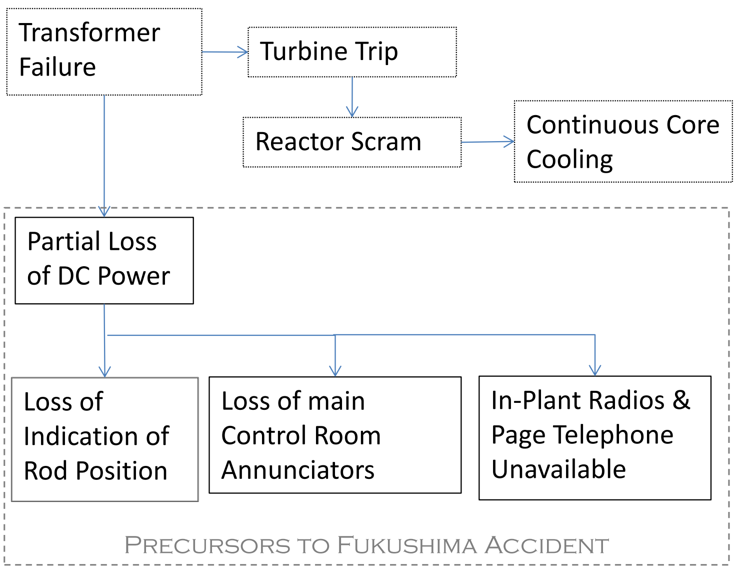


Figure VI.7 Major Occurrences in Event-6

3. Main Features on Effective Safety Layers or Provisions

The consequences of the six events selected are much lower than those from the Fukushima accident because the core cooling remained available or was recovered before the event sequences progressed to core damage. The differences of the selected events are summarized from the viewpoint of the safety layers or provisions that have been effective. **Table VI.1** shows the main features of events selected. The effective safety layers or provisions for the events are briefly described below:

Table VI.1 Main Features of Safety Layers or Provisions in Events Selected

Event No.	Offsite Power	Onsite AC Power (EDG)	DC Power	Heat Removal (Core Cooling)	Main Control Room
1	not lost	not lost	partly lost	partly lost	partly lost
2	completely lost	completely lost	completely lost	shortly lost	lost (evacuated)
3	partly lost	not lost	not lost	partly lost	not lost
4	completely lost	partly lost	not lost	partly lost	not lost
5	not lost	not lost	not lost	not lost	not lost
6	not lost	not lost	partly lost	not lost	partly lost

Event 1: Fire, massive internal flooding and loss of safety systems
Event 2: Fire, prolonged SBO involving loss of control room and loss of residual heat removal
Event 3: External flooding involving partial loss of safety systems and multi-unit site issue
Event 4: Loss of offsite power with one EDG inoperable
Event 5: Tsunami-induced external flooding
Event 6: Common-mode loss of instrument power

(1) Event-1

Core cooling was never lost completely but the equipment important for recovering the plant, such as shutdown heat exchanger pumps, was affected by flooding. Electrical power was available on the site throughout the event, though DC power was partly lost due to cables being affected by fire and the MCR habitability was threatened by smoke. Insufficient design features were identified against common mode failures due to mainly fire and flooding.

(2) Event-2

Electrical power supply was completely lost due to the burnt cables and shutdown cooling was lost though core cooling was carried out by natural circulation with use of the emergency measures including manual actuation of diesel-driven fire pumps and opening of ASDVs. As well, MCR was evacuated due to smoke ingress and the emergency control room had no indications. As a result, the operators were forced to take the blind operation of plant.

(3) Event-3

Offsite power supplies were lost at multiple units, due to severe weather condition, but electrical power was available at the individual units throughout the event. Since heat removal could be ensured by SGs, core cooling was never lost. However, essential service water was partly lost due to flooding and the low pressure injection and containment spray pumps were declared unavailable. The flooded areas were recovered by pumping operation. This event highlighted inadequate protection measures against flooding.

(4) Event-4

The event involved loss of offsite power and failure to connect one of two EDGs due to two independent electrical faults. The unit lost the switchboard on one train as well as offsite power sources. Power supply to the safety equipment was therefore only provided by the remaining EDG. Heat removal was not lost and DC power remained available. The mobile generator was connected on the switchboard in anticipation that the situations would have been aggravated.

(5) Event-5

The event was caused by the earthquake-induced tsunami and only non-safety equipment was affected. Offsite power was available, core cooling and heat removal remained operable and DC power was supplied to essential buses throughout the event because of no water ingress into the reactor, turbine and service buildings. Thus, no threat was given to the plant safety. However, the higher water level in the pump house made the regulatory requirements to be revised.

(6) Event-6

The event involved failures of five non-safety related UPSs, resulting in loss of indications of control rod position and loss of MCR annunciators. Thus, the operators faced the difficulties in monitoring the post-scram plant condition. Because the safety-related UPSs were not affected and offsite power supplies remained available, however, all safety functions were available and the reactor was taken to cold shutdown according to the emergency operating procedures. Plant personnel manually restored power output from UPSs by using an alternate power source.

4. Lessons Learned from Precursor Events

Based on the safety relevance of the events selected, lessons learned were described in their respective IRS reports. Although these lessons could help avoid recurring events at other nuclear power plants or at least reduce their potential consequences, they are specific to the individual events and not necessarily applicable to the Fukushima accident. In the following, the generic lessons are derived and discussed focusing on the relations to the Fukushima accident.

(1) Event-1

In this event, two initiators which caused common mode failures, fire and flooding, seriously compromised the plant safety. The consequences of fire were very significant as follows: all cables in the turbine generator area were burnt resulting in partial loss of core cooling, the compressed air system was severely damaged leading to difficulties to control the feedwater supply and affecting the air-operated isolation valves, and loss of 48 VDC control buses disabled control of important functions from MCR and position changes of certain valves. The significant flooding of reactor building affected equipment very important for recovering the plant. Although the exhaust pumps had been installed in the reactor building, they could not operate due to loss of power supply caused by fire. The plant was not designed to cope with such a large-scale flooding and thus, there was neither level instrumentation in the reactor building nor other preventive measures as physical barriers or pedestals for safety equipment. The root causes were inadequate compartmentation or physical separation of redundancies, buildings and fire zones. As well, there were deficiencies in independence of safety systems.

This event demonstrated that fire might induce internal flooding which could aggravate the situations and hence, measures against both fire and flooding should be taken into account in the plant design and layouts to prevent common mode failures. In particular, special attention should be paid to routing of power cables and the electrical power system for the safety system should be located at physically separated areas train by train. If necessary, additional defenses should be provided For example, fire retardant materials should be used for cables and physical barriers should be added between components.

The loss of control of important functions from MCR forced the operators to execute the manual actions locally, implying the importance of an emergency control room independent from MCR.

(2) Event-2

This event resulted in a complete SBO which lasted for about 17 h. The causes of extended SBO and consequent degradation of several safety systems were the cable fire and lack of proper fire barriers/fire retarding provisions together with inadequate physical separation in redundant safety-related cables. As well, in this event, MCR had to be evacuated due to ingress of smoke and no indications were available in emergency control room due to loss of control power supply. Therefore, some of important parameters had to be locally measured. As a consequence, the operators were forced to take the blind operation of plant. It should be noted that this plant survived the extended SBO and the core cooling was maintained by establishing the natural circulation with use of fire fighting system and ASDVs. This seems a good example of successful accident management.

To prevent recurrence of such an event, the licensee and the regulatory authority obtained the generic lessons as follows: in-depth review of physical separation and fire protection provision for power and control cables should be performed to avoid common mode failures; the control room habitability should be ensured under adverse conditions outside; the capability to cope with extended SBO should be reviewed along with the duration of SBO; and adequacy and reliability of water supply from fire fighting system to cater to simultaneous needs of fire fighting and supply to SGs should be looked into.

(3) Event-3

In this event, exceptionally severe weather conditions combined with inadequate protection measures caused a flooding of the reactor building and the simultaneous failures of safety-related systems at two units. As well, the arrival of additional teams from outside the plant was hampered by the flooding of access routes, many tree falls, and so on. Because electrical power remained available, these units were brought to a shutdown state by using SGs.

The flooding revealed some weaknesses in the site protection against external flooding. The water infiltrated into the duct cover slabs, flooding the sub-levels of the administrative buildings and common auxiliaries building, and then, propagated into the rooms of two units through doors and cableway penetration that was damaged with the water entering, reaching the sub-levels of the electrical buildings and the water pumping station. This event underlined the importance of identifying all the paths by which water might enter the site and/or buildings and eliminating these paths if necessary.

It should be noted, as well, that several defense lines were reinforced at this plant after the event. For example, the protection dyke facing the river was heightened and an anti-swell wall was built above this dyke. The warning system level was changed by introducing a pre-alert and an alert thresholds for average wind speeds aiming at enabling the early arrival of the additional teams.

(4) Event-4

During this event, the unit lost the switchboard on one train as well as offsite power sources. Power supply to the safety equipment was therefore provided by only one EDG. However, the mobile generator was connected to the safety busbar so that the reactor could cope with the subsequent aggravating situation. As well, the emergency turbine generator that runs on the steam produced by the secondary cooling system was ready to operate in the event of simultaneous loss of both offsite power sources and both EDGs. This event highlighted that diverse measures should be implemented to re-establish the power supply to the systems required for heat removal in a short time even in the station blackout. It should also be borne in mind that a number of changes had been made during 1980s to deal with a total loss of offsite and onsite power.

In addition, the switchboard failure was due to a malfunction on an overcurrent relay, the cause of which was the formation of zinc filaments similar to tin whisker. This phenomenon had been observed in other countries but never at the plant. Since such an unexpected phenomenon or event might occur and affect the plant safety, special attention should be paid to the introduction of new technologies and design changes. As well, the event revealed that a measure aiming at preventing emergency battery consumption might make managing the consequences of two electrical failures occurring one after the other more difficult. The lessons from this observation should be taken into account when the practices are changed.

(5) Event-5

Although the tsunami waves hit the plant site, resulting in the flooding of pump house, no seawater entered the vital areas such as the reactor and turbine buildings and thus, the plant safety was not affected. However, some non-safety systems and components were damaged or rendered unavailable, causing the plant outage. After the event, a new alarm was provided in MCR to indicate high level in the pump house.

The highest water level recorded in the pump house was 10.56 m, which was 4.56 m higher than the normal operating level. As per the regulatory requirement, tsunami is one of the external events to be considered for siting and design of plants. Since the requirements at that time were based on information available about 15 years ago, the revision of regulatory requirements was studied taking into account the experience of the tsunami. This event implies the needs to consider the impact of tsunami induced by the teleseismic earthquake in the plant design and/or regulatory requirements.

(6) Event-6

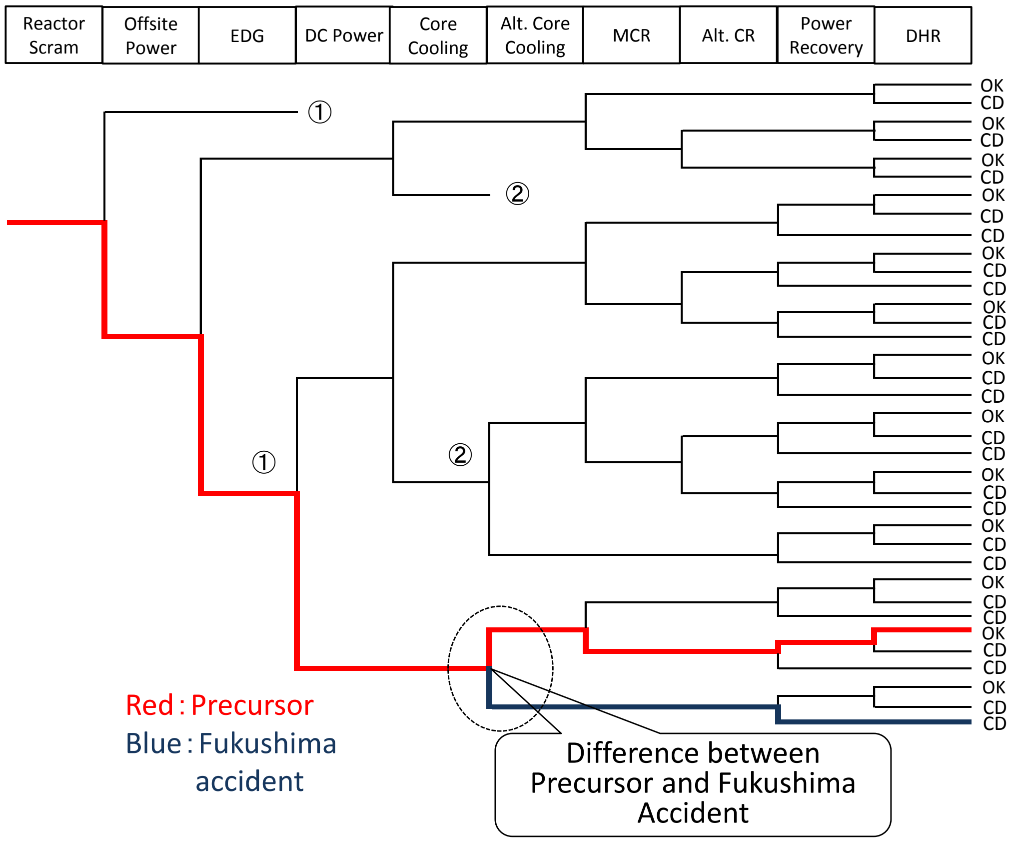
The simultaneous tripping of UPSs was caused by a design deficiency of control logic battery in each UPS and inappropriate preventive maintenance of its backup battery. Under degraded voltage conditions, the UPS logic power supply switching circuit does not actuate until the supply voltage has decreased well below the level that will cause the trip. However, during this event, the backup batteries were discharged due to failure to perform adequate maintenance. Had either deficiency been corrected, the simultaneous loss of UPSs would have been avoided. In this event, the control rod position indications were lost and thus, the operators could not verify that the reactor would remain shut down. This means the needs to prepare the procedures that specify the alternative measures when the parameters important to safety are not monitored in MCR.

This event highlighted the common-mode vulnerability of the multiple non-safety related systems. Plant personnel should assure that design considerations even for such non-safety related systems and components include analysis for determining failures that could degrade the plant safety. Particular attention should be paid to batteries that supply backup power to control logic since their use in such an application may not be well documented in equipment literature. Such non-safety systems/components may be used for accident management measures and therefore, plant personnel should take into account latent failures of them which could challenge the plant safety.

5. Event Tree for Precursor Event and Fukushima Accident

For one of the six precursor events (Event-2) which is the most notable near-miss to the Fukushima accident, the event tree was developed, as shown in **Figure VI.8**, to delineate its accident sequences and to provide an easier comparison of those with the Fukushima accident. In this event tree, the red line denotes the actual sequence in the precursor and the blue line shows the sequence in the Fukushima accident. While the Fukushima accident was initiated by the earthquake and the subsequent tsunami, the precursor was by internal fire. However, in both events, their respective plant system behaviors were similar to each other as follows: the reactor scrammed; offsite power lost, EDGs automatically started but tripped, resulting in extended SBO occurred; DC power became (partly) unavailable; emergency core cooling was unavailable due to loss of power; MCR lost its function, leading to a blind plant operation.

Fortunately, in the precursor, the alternative core cooling was established by using diesel-driven fire pumps to supply water to the secondary side of SGs and thus, the core was cooled by natural circulation. As well, the power was recovered approximately 17 h after the event initiated. In the Fukushima accident, the alternative core cooling was not done and electrical power supplies were not recovered in a timely manner, resulting in severe core damage and the subsequent large release of radioactive materials. The critical difference between these events is whether or not alternative core cooling was successful or not as shown in Figure VI.8. The Fukushima Dai-ichi Nuclear Power Plant also has diesel-driven fire pumps as alternative measures to supply cooling water into the reactor and in particular, at Unit 1, these pumps can provide water to the secondary side of ICs. Nevertheless, at the time of accident, these pumps could not supply water to the reactor because the reactor pressure could not be decreased below the pump head. At Unit 1, ICs were manually tripped before the tsunami attack and became unavailable after that. As a result, diesel-driven fire pumps were no longer able to play a role of alternative water supply. In the precursor event, the SG pressure was decreased by keeping ASDVs open so that the diesel-driven pumps could supply water to SGs continuously.



Note) EDG: emergency diesel generator, MCR: main control room, Alt. CR: alternative control room, DHR: decay heat removal

Figure VI.8 Event Tree for Precursor (Event-2) and Fukushima Accident

Considering these facts observed, it would not be an exaggeration to say that the success or failure of timely depressurization has control of the plant destiny. In other words, this means that both the precursor and the Fukushima accident underscore the importance of preparing the robust depressurization measures for executing alternative water supply in a timely way when safety-related systems are unavailable to cool the core.

VI.3 Generic Issues to Be Resolved

The Fukushima accident was severe core damages at three units and large releases of radioactive materials as a result of prolonged SBO, which was caused by mainly the earthquake and subsequent tsunami. The earthquake damaged breakers and distribution towers, causing a loss of all offsite power sources to the site. As well, the tsunami caused extensive damage to site buildings and flooding of the turbine and reactor buildings. Intake structures were unavailable because the pumps, strainers and equipment were heavily damaged by the tsunami and debris, resulting in a loss of the ultimate heat sink. The tsunami also flooded some of EDGs and the electrical switchgear rooms. As a consequence, all AC power was lost at Units 1-5 and all DC power was lost at Units 1 and 2 (one EDG at Unit 6 continued to function and a part of DC power from batteries remained available at Unit 3).

In such a way, the Fukushima accident experienced the tsunami-induced common cause failures (CCFs) as follows: EDG failures, electrical switchgear failures, loss of ultimate heat sink, loss of core cooling, loss of MCR lighting, loss of MCR instrument, loss of reactor pressure control (inoperable SRVs), and loss of decay heat removal. Eventually, the Fukushima accident highlighted the vulnerabilities to CCFs due to flooding. In other words, it was revealed that there had been deficiencies in layout design and physical separation of electrical components. Since the natural phenomena such as tsunami are inevitable, the plant design is required to assume the design basis and implement the measures against them. At the Fukushima Dai-ichi Nuclear Power Plant, the breakwater had been installed against tsunami(3,13). In addition, the elevation of seawater pump motors had been raised and some measures had been taken for building penetrations and pump seals to prevent inundation(3,13). However, neither layout change nor improvement of physical separation had been carried out until the accident occurred. As a result, the tsunami which exceeded significantly the design basis hit the Fukushima Dai-ichi Nuclear Power Plant site, resulting in those measures having been disabled. This shows the difficulty of coping with the tsunami beyond design basis.

To overcome this difficulty, in principle, it is considered effective that the defense-in-depth concept is incorporated into the plant design. At first, the coastal levee should be built against tsunami beyond design basis to protect a plant site from being inundated. However, it seems very difficult or impossible to determine how high it is enough because the height of such a tsunami wave has a large uncertainty and cannot be evaluated deterministically. Next, the vital areas should be waterproofed to protect safety-related equipment from being submerged, assuming that seawater would ingress into the site. In addition, physical separation, in particular, at different levels is considered one of measures effective to reduce the possibility of common cause failures due to flooding. It is also difficult to assure the water tightness of vital areas because there might be many penetrations for cables and/or pipes. In the third place, assuming that the vital areas would be flooded, the equipment/devices and procedures should be prepared to conduct the pumping operation for removing seawater from those areas. Actually, such an operation was demonstrated to be effective in the past event as mentioned in the previous section (Event-3). However, the pumping operation may depend on the site situation and thus, there are some difficulties in demonstrating its effectiveness. When implementing countermeasures against tsunami or flooding, these difficulties should be overcome by establishing the plausible premises, in particular the design basis assumptions.

Although it has not been determined how the earthquake adversely affected the plant equipment, particularly safety-related equipment in the Fukushima accident, the earthquake beyond design basis should be taken into account in the design. However, the conventional design approach against earthquakes generally has been based on aseismic standards and thus, the safety significant equipment has been designed to withstand the design basis earthquake. In this approach, it seems difficult to avoid the common cause failures of safety systems/components which belong to different layers of defense-in-depth in the case that the earthquake beyond design basis would hit the plant site because those systems/components are usually designed according to the same design standards or equivalent ones. Therefore, when the aseismic design approach is applied against the earthquake beyond design basis, it is essential to clarify how the aseismic standards should be applied to individual layers of defense-in-depth concept and/or to incorporate a new concept such as structural diversity by combining the seismic resistant structure, seismic isolation structure and seismic response control structure into the design.

The Fukushima accident also underlined vulnerabilities to loss of DC power, which led to failure to manually depressurize the reactor and loss of MCR indications. In the accident, the reactor could not be depressurized and as consequence, alternative water injection was not carried out. At BWRs, manual reactor depressurization in the case of transient relies on SRVs, however, the manual opening of SRVs usually needs compressed air to their actuators and DC power to their electromagnetic valves which supply compressed air. Therefore, SRVs become inoperable if either compressed air or DC power is lost. In addition, the alternative water injection has been prepared based on the assumption that the reactor would be depressurized successfully. In light of such current situations, more robust measures should be studied and implemented to ensure the reactor depressurization and subsequent alternative water injection. For example, the diversity should be incorporated into the design of SRVs by using different drive mechanisms. In actual, four different types of SRVs have been applied at U. S. BWRs; one type is direct-acting SRVs and the other three types are pilot-actuated SRVs[(14)]. Direct-acting SRVs use an attached actuator to overcome the spring tension in the main part of SRV to open the valve (see **Figure VI.9**[(14)]) and are the same or similar design to those at BWRs in Japan including the Fukushima Dai-ichi Nuclear Power Plant. Pilot-actuated SRVs use the air actuator to which air pressure is applied by energizing the solenoid-operated valve. For one type of them (two-stage SRV, see Figure III.16 in Chapter III), the air operator directly operates the pilot piston to open the main valve. For another type, the air operator directly opens the second stage disc by mechanically depressurizing the second stage piston to open the main valve (see **Figure VI.10**[(14)]). For the third type (three-stage SRVs, see **Figure VI.11**[(14)]), a second pilot valve is actuated that is independent of the primary pilot valve. It seems that the appropriate combined use of these types is effective to improve the reliability of manual reactor depressurization with SRVs.

Loss of MCR should be noted as one of important issues to be resolved. As mentioned previously, one plant had experienced the blind operation due to loss of MCR but the lessons learned from this event had been fed back to neither the plant design nor operation in Japan. Eventually, in the Fukushima accident, almost all indications in MCRs were lost and thus, the operators could not assess the situations where the individual units were faced. Also, the operators misunderstood the reactor water level at Unit 1 based on the false indication from the level instrument, which was affected by the pressure inside containment. Considering the facts from the precursor and the Fukushima accident, MCR should be backed up with the dedicated DC power or the emergency control room should be improved to avoid the blind operation. Also, the level instrument should be redesigned to provide accurate indications without being affected by its surroundings. At a U.S. BWR, the reactor vessel pressure and level can be monitored in the reactor building by using mechanical instruments which require no electric power to function[(15)]. Addition of such local instruments should be studied so that the plant condition can be grasped even in the case that all DC power would be lost.

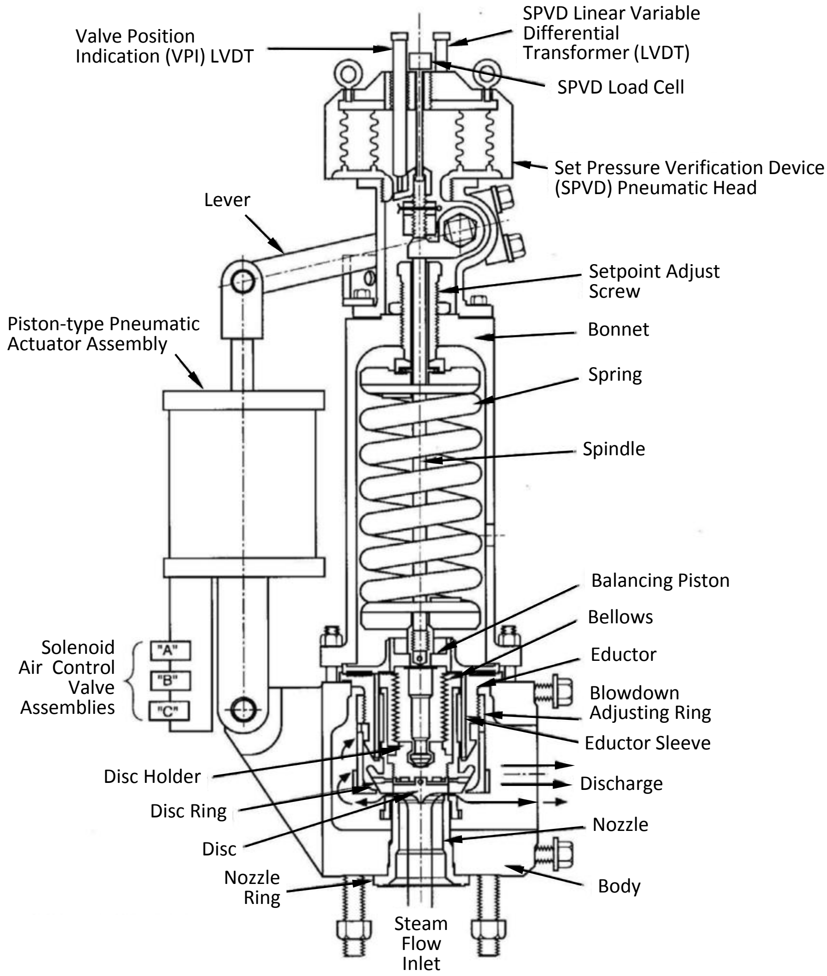


Figure VI.9 Direct Acting SRV(14)

Furthermore, the Fukushima accident revealed some multi-unit site issues. Specifically, all of the six units were affected by the earthquake-induced loss of offsite power and four units suffered from the tsunami-induced loss of DC power. In the past, some multi-unit plants had undergone similar situations. As mentioned previously, at a plant site, two units lost the offsite power and suffered the flooding of buildings due to severe weather condition. At another plant, two units tripped simultaneously due to loss of offsite power as a result of incorrect switchyard protective relay tap settings(16). These incidents show that events affecting two or more units on a site might result in more complicated situations if some systems are shared with those units. Although, in Japan, the safety-related systems are independent from each other between the individual units on a plant site, MCR is shared with two units on many sites as well as the Fukushima Dai-ichi Nuclear Power Plant. It seems very difficult or impossible to separate MCR at the existing plants. Thus, the functions in emergency control rooms should be enhanced so that the operators can monitor the plant conditions and take actions required for placing the reactor to a safe shutdown state even in the case that MCR lost its function. Another multi-unit site issue is insufficient accident management measures. The accident management measures had been taken assuming that the severe accident would occur at a single unit because mainly the internal events had been considered. Although all the nuclear power plants in Japan had implemented onsite interchange power supplies between two units as one of accident management measures, for example, this measure was not effective in the Fukushima accident. In one precursor event, onsite power was recovered by starting up an EDG shared with the neighboring unit which was not affected. If the neighboring unit would have also been involved in the event, however, the plant situation would have been more complicated and aggravated. It seems that this precursor provided an opportunity of reconsidering accident management but the chance was missed. Accident management, in particular, alternative power supplies should be implemented for individual units on a site assuming the case that two or more units would be involved.

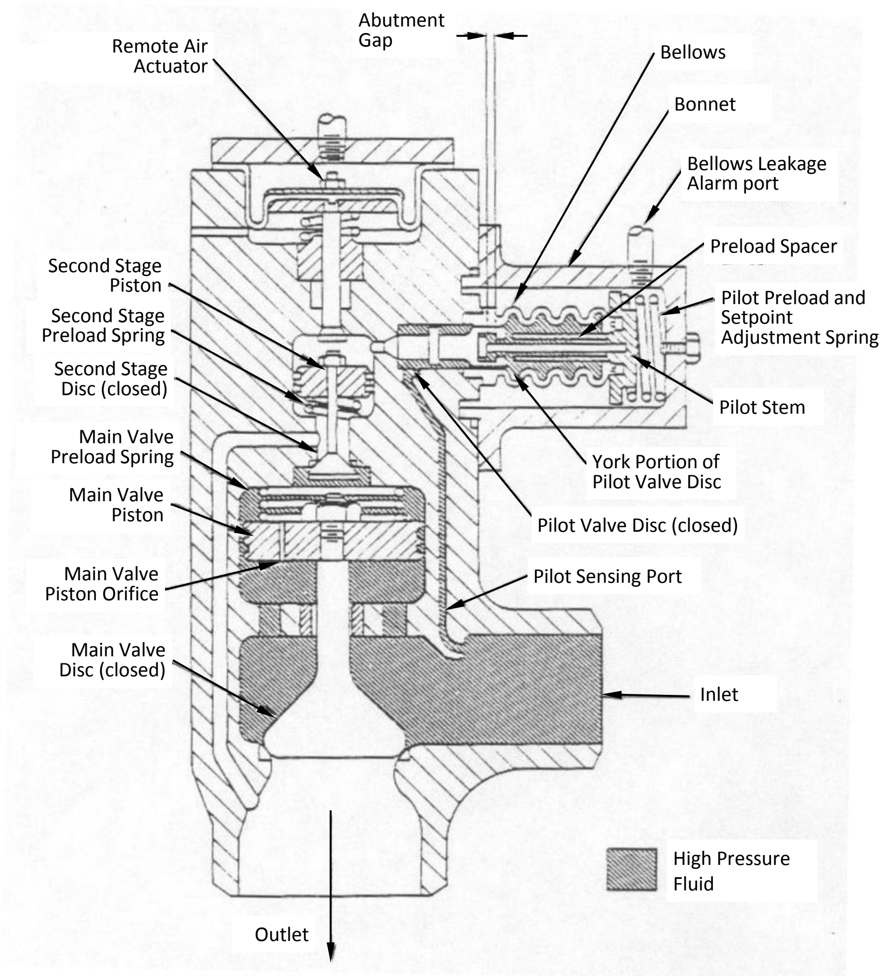


Figure VI.10 Pilot-Actuated SRV – actuation function not dependent on the pilot[14]

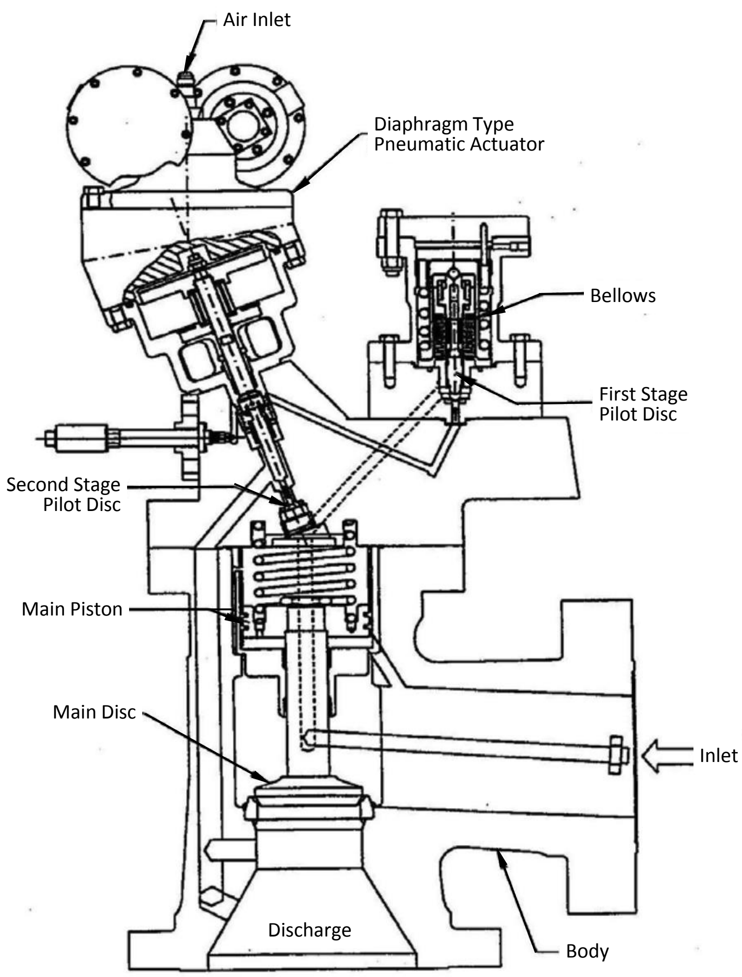


Figure VI.11 Three-Stage Pilot-Actuated SRV – actuation function not dependent on the pilot(14)

For the plant system design, the operators had not recognized types and structures of valves on containment venting lines and had to check whether or not those valves could be manually opened in the Fukushima accident. This implies that the "as-built" inspection by licensees is essential, in particular for the systems and components infrequently used. In addition, the IC valves were designed to close automatically due to loss of DC power at the Fukushima Dai-ichi Nuclear Power Plant, however, this design does not seem the standard one because the different design concept, "fail as is", is applied at the U.S BWRs(2) and the valves on HPCI/RCIC turbine steam supply line do not close due to loss of DC power(17). Furthermore, the SGTS valves are designed to open automatically when their driving forces are lost at Fukushima Dai-ichi Nuclear Power Plant but other some Japanese BWRs incorporates the different concept, "fail as is"(18). These facts indicate that the design concept of either pen/close" or "fail as is" depends on the plant or licensee. The reasons of such differences should be examined to clarify the adequacy and applicability of design concept.

The above mentioned problems are considered generic issues to be resolved to further improve the plant safety.

VI.4 Concluding Remarks

Through the analysis of operating experience, safety significant insights were obtained for nuclear power plants as follows.

- The degraded functions of ECCS and electric power systems were common issues at nuclear power plants and the defense-in-depth concept should be incorporated into the plant operation as well as design more strictly to ensure the plant safety.
- Various types of safety significant events were identified, including loss of offsite power, failures of emergency diesel generators, loss of heat sink, etc., some of which are considered precursors to the Fukushima accident.
- Some plants experienced the events beyond the design basis, showing that more attention should be paid to those taking into account the challenges they pose.
- Identified were several factors which might be applicable to the Fukushima accident. For example, one topical study indicates that loss of decay heat removal might lead to the core uncovery within several hours even in the reactor shutdown state. This underscores the need to recover the decay heat removal function in a short time. Another topical study on safety/relief valves shows that the potential failure to open of them is one of the safety significant issues for overpressure protection, implying the importance of maintaining their functionabilities adequately. The topical study on criticality accidents reveals that it had been recognized that such accidents would not occur before the JCO accident. This is the case with the Fukushima accident. Risks of severe accidents had been downplayed because of lower core damage frequencies compared with those in foreign countries and TEPCO had been convinced that no tsunami earthquake would occur offshore of Fukushima according to the documentary records, implicating lack of understanding of tsunami risks. This means that it is important to learn the lessons from the past operating experience in other types of facilities.
- The ASP analysis of SGTR events and the Fukushima accident imply that reactor depressurization might have been one of key factors to bring the reactor into a safe shutdown state. While manual reactor depressurization is able to be done by using the pressurizer relief valves and the main steam relief valves at PWRs, only the use of

SRVs is available at BWRs. Therefore, alternate measures and diversity should be taken into account for manual reactor depressurization at BWRs.

These insights and lessons should have been fed back to the plant designs and operations adequately. It should be recognized that lack of attitude of learning from the past operating experience might have contributed to the Fukushima accident.

Of approximately 200 events identified as safety significant, six events were selected as precursors to the Fukushima accident: 1) fire, massive internal flooding and loss of safety systems, 2) fire, prolonged station blackout involving loss of MCR and loss of residual heat removal, 3) external flooding involving loss of safety systems and multi-unit site issue, 4) loss of offsite power with one of two emergency diesel generators inoperable, 5) tsunami-induced external flooding and 6) common-mode loss of instrument power. The first event demonstrates that fire might induce internal flooding, which could aggravate the situations, and suggests that the electrical power system for the safety systems be located at physically separated areas and special attention be paid to layout of power cables and protection against common mode failure such as physical barriers. The second event highlights the importance of performing in-depth review of physical separation for electrical cables to protect safety systems against common mode failures, ensuring the control room habitability under adverse conditions outside, and reviewing the capability to cope with extended SBO. The third event reveals some weaknesses in the site protection against external flooding and underlines the possibility that the flooding-induced degradation might affect simultaneously all the units on a site. The fourth event did not lead to SBO but reveals the needs for installing additional electric power sources, such as a mobile generator and emergency turbine generator, to cope with a more severe situation. As well, this event demonstrates that two or more units on a site might be simultaneously affected by external events. The fifth event involved the tsunami-induced flooding, which did not threaten the plant safety, but implies the need to consider the impact of tsunami in the plant design and/or regulatory requirements. After this event, the revision of regulatory requirements was studied taking into account the experience of the tsunami. In the sixth event, the common mode failures of non-safety related UPSs resulted in an unexpected loss of indication of control rod position and MCR annunciators. This event underscores the needs to perform the analysis of their designs for determining common mode failures that have the potential of threatening the plant safety. Some of these events might have induced new phenomena or unexpected aggravating conditions and special attention should be paid to the plant design and accident management measures. In particular, the second event is the most notable near-miss to the Fukushima accident and the timely depressurization and subsequent alternative water supply could avoid the plant from severe accident. This event emphasizes the importance of preparing the robust depressurization measures for executing alternative water supply in a timely way.

The Fukushima accident highlights the vulnerabilities to the flooding. Although the Fukushima Dai-ichi Nuclear Power Plant had taken some measures against tsunami such as the installed breakwater and the elevated seawater pump motors, the tsunami exceeding the design basis hit the site and rendered those measures disabled because neither layout change nor improvement of physical separation had been carried out. To cope with the tsunami beyond design basis, it is considered effective that the defense-in-depth concept is incorporated into the plant design; 1) coastal levee against tsunami beyond design basis, 2) waterproofed vital areas and physical separation, and 3) pumping operation to unwater the areas. However, it seems very difficult or impossible to determine how high the levee is enough and to assure the water tightness of vital areas. When implementing countermeasures, these difficulties should be overcome by establishing the plausible premises, in particular the design basis assumptions. In addition, the Fukushima accident underlines the vulnerabilities to loss of DC power including failure of manual reactor depressurization and loss of MCR, insufficient accident management measures, lack of knowledge on “as-built” or “as is” plant system design, poor quality of accident/emergency operating procedures, and inconsistent design concepts on “fail open/close” or “fail as is”. Manual reactor depressurization depends on SRVs in emergencies and thus, more robust approaches should be implemented in the plant design by applying different drive mechanisms to SRVs. MCR should be backed up with the dedicated DC power or the emergency control room should be improved to avoid the blind operation. Furthermore, the “as-built” inspection by licensees should be performed for the systems and components such as containment venting system which is infrequently used. As well, the reasons of “fail close” design of IC valves and “fail open” design of SGTS valves at the Fukushima Dai-ichi Nuclear Power Plant should be examined to clarify the adequacy and applicability of design concept through the comparison with the designs at other plants.

This study manifests the importance of analyzing the Fukushima accident more comprehensively and in detail in consideration of the insights and lessons gained from past events.

References in Chapter VI

(1) N. Watanabe, *Trends and Characteristics Observed in Nuclear Events Based on International Nuclear Event Scale Reports*, Journal of Atomic Energy Society of Japan, Vol.43, No.10, pp.1017-1026, 2001 (in Japanese).

(2) The National Diet of Japan, *The official report of The Fukushima Nuclear Accident Independent Investigation Commission*, 2012 (*warp.da.ndl.go.jp/info:ndljp/pid/3856371/naiic.go.jp/en/report/*). (as of December 2013)

(3) Tokyo Electric Power Company, Inc., *Fukushima Nuclear Accident Analysis Report*, 2012 (*www.tepco.co.jp/en/nu/fukushima-np/interim/index-e.html*). (as of December 2013)

(4) IRS Report, *Fire in One Turbine Generator Group and Subsequent Failure of Safety Systems* (restricted).

(5) IRS Report, *Complete Station Blackout due to Fire in Turbine Generator* (restricted).

(6) IRS Report, *Partial Loss of Safeguard Systems as a Result of External Flooding* (restricted).

(7) IRS Report, *Reactor Unit Shutdown due to a Failure on a 6.6 kV Emergency Switchboard* (restricted).

(8) IRS Report, *Safe Shutdown of Reactor Following Tsunami Strike* (restricted).

(9) IRS Report, *Reactor Scram following Loss of Five Uninterruptible Power Supplies and Partial Loss of Control Room Instrumentation* (restricted)

(10) *Vandellos fire may lead to closure*, Nuclear Engineering International, Vol.34, No.425, pp.3, 1989.

(11) *Narora 1 turbine hall gutted by fire*, Nuclear Engineering International, Vol.38, No.467, 1993..

(12) USNRC, *Transformer Failure and Common-Mode Loss of Instrument Power*, NUREG-1455, 1991.

(13) INPO, *Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Plan*t, INPO 11-005, 2011.

(14) T. E. Wieman, C. D. Gentillon and B. M. Brady, *Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007*, NUREG/CR-7037, 2011.

(15) Exelon Generation, *DNPS (Dresden Nuclear Power Station) Response to Additional NRC Questions,* ADAMS Accession No. ML13037A045, 2013.

(16) USNRC, *Loss of Offsite Power and Dual-Unit Trip at Catawba Nuclear Generating Station*, Information Notice 2007-14, 2007.

(17) Nuclear and Industrial Safety Agency, *Technical Knowledge of the Accident at Fukushima Dai-ichi Nuclear Power Station of Tokyo Electric Power Co., Inc. (Provisional Translation)*, 2012.

(18) Hokuriku Electric Power Company, *Response to Questions from Members of Stress Test Committee in NISA*, No. ST 12-1-2, 12th Stress Test Committee of NISA (held on March 29, 2012), 2012 (in Japanese).

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Chapter VII

General Conclusions

VII.1 Concluding Remarks

Since the TMI-2 accident, it has been worldwide recognized an important and effective means to obtain the lessons and insights through the analysis of the incidents and accidents that actually occurred and to feed them back into the design, construction, operation and management of facilities. Licensees or operators of nuclear facilities are required to report the events at their own plants to the regulatory authorities immediately, analyze their causes, and implement corrective measures. As well, the international reporting systems have been operated to exchange event information among member countries. Using such information, the activity called operating experience feedback (OEF) has been carried out on a national or international basis. This activity can be achieved by a systematic and comprehensive review and analysis of events from various points of view such as generic aspects (generality, commonality, similarity), specific aspects (specificity, uniqueness), and risk significance aspects.

Aiming at providing insights and technical issues useful for regulating and enhancing the safety of nuclear facilities, the present thesis addresses a series of studies of analysis of nuclear and radiological events as follows: the generic studies (including comprehensive reviews) of a large number of events, topical studies on safety significant events, accident sequence precursor (ASP) studies, and review of the five investigation reports on the Fukushima accident. As well, described are new computer software tools developed to assist in the comprehensive reviews. Finally, insights from these studies and precursors to the Fukushima accident are discussed to clarify the generic safety issues to be resolved.

(1) Chapter II

Chapter II describes the generic studies on INES reports and IRS reports, and computer software tools developed for supporting generic studies.

The generic study covered approximately 500 events reported to INES during the years 1990 to 2000 and examined overall trends and characteristics of events rated at Level 2 or higher. The main observations obtained from this study are as follows: 1) the number of such events has been remained at 10-20 events per year throughout the period; 2) two events rated at Level 4 involved fatalities due to overexposure and about half of 15 events at Level 3 were associated with overexposure resulting from lost sources or radiation devices; 3) the degraded functions of ECCS and electric power systems were common issues to nuclear power plants. The other generic study analyzed more than 2000 IRS reports during the period 1988 through 2012 and identified approximately 200 safety significant events. These significant events can be roughly categorized into three groups: recurring events, events associated with external hazards and the events related to new phenomena or unexpected aggravating conditions. Recurring events indicate that the previous corrective actions have not necessarily been effective to prevent the recurrences. As for the external hazards, more attention should be paid to those beyond the design basis assumptions based on the past events. The new phenomena or unexpected aggravating conditions have a potential to threaten the plant safety but it seems difficult to detect them prior to their appearances. The safety significant events identified (might) have affected the facility, environment and public health and thus, should be further examined and discussed. The results from these studies have been disseminated into the nuclear community, contributing to the sharing and more effective utilization of event information. These studies underline the needs to perform systematic and comprehensive reviews of various events to gain the safety significant insights.

As a supporting tool for comprehensive review, a new computer software package CESAS was developed to extract the event sequence from the event description written in English. The analytical process consists of three steps: (i) to analyze each sentence syntactically and semantically, (ii) to identify sentences, clauses or phrases representing occurrences, (iii) to deduce the mutual relationship between occurrences. Through the application, it is shown that CESAS-extracted event sequences generally agreed with manually-extracted ones, demonstrating the effectiveness and feasibility of CESAS. The web-based database, which contains the INES reports translated into Japanese, was developed to provide an effective tool for the media and public as well as for the nuclear community. These tools may make the generic studies more efficient.

(2) Chapter III

Of topical studies performed on safety significant events identified in the generic studies, this chapter discusses three topical studies on loss of DHR during reactor shutdown at PWRs, criticality accidents at nuclear fuel processing facilities, and setpoint drift in safety or safety/relief valves at U.S. LWRs.

The first study reviewed a total 63 loss of DHR events under reduced inventory conditions during the years 1976 through 1990 at U.S. PWRs to identify the trends and characteristics and to clarify the major causes which could prolong the duration of the DHR loss. It is found that four-fifths of them were caused by cavitation or air binding of DHR pumps, that is, air entrainment into the DHR pumps and the major contributor of air entrainment was the lowering the RCS water level too far, mostly resulting from inaccurate level indication. This highlights the need to improve the reliability of water level instruments. In many events involving air entrainment, the DHR loss lasted for more than 1 h and the remarkable coolant heatup was observed. The heatup rates were evaluated with use of the data obtained from the event reports. The calculated heatup rates are in reasonably good agreement with the observed ones. In addition, the times to bulk boiling in the core and core uncovery were estimated and the results show that bulk boiling and core uncovery would take place within 1 h and several hours respectively even if the DHR loss would occur in the late stages of shutdown, for example, 30 d after the reactor trip. This implies that the recovery actions need to be prepared in advance against loss of DHR pumps.

In the second study, 21 criticality accidents were analyzed in terms of similarities to the JCO accident, focusing on the accident sequences and causes, to identify the lessons which should have been fed back. The results show that almost all of them occurred when handling uranium or plutonium solution in vessels/tanks with unfavorable geometry. Common issues identified in the accident scenarios and causes include violations of procedures and/or technical specifications, unexpected use of vessels/tanks for improving work efficiencies, procedural changes without any application to and permission from the regulatory body, lack of understanding of criticality hazards, and complacency that a criticality accident would not occur. In addition, the fact that most of accidents occurred in 1950s to 1960s might have contributed to the JCO accident. This study underscores the importance of making the lessons learned from the past similar events permeate throughout the workers.

The third topical study analyzed events involving setpoint drift in safety or safety/relief valves at U.S. LWRs (SRVs at BWRs, PSVs, and MSSVs at PWRs) by reviewing approximately 90 LERs from 2000 to 2006, focusing on causes and setpoint deviation ranges, to provide the insights useful for improving the reliability of these valves. The results show that for SRVs and MSSVs, disc-seat bonding is a dominant cause of the setpoint drift high and has a tendency to cause a relatively large deviation of the setpoint. This means that disc-seat bonding might be a safety concern from the viewpoint of overpressure protection. The deviations due to the drift low are smaller compared with those due to the drift high. For PSVs, the deviation of setpoint is generally small, although its causes are not specified in many instances. It is important to share the insights obtained from the analysis of events involving setpoint drift in terms of overpressure protection and pressure boundary integrity.

The topical studies derive the generic safety issues and insights useful for examining the measures against individual topics and show that the lessons learned from past events have not sufficiently been employed and/or effective measures have not taken. This means that sharing and exchange of information on operating experience should be enhanced on a national and international basis. In particular, the lessons learned and insights obtained should be disseminated into the nuclear community. Also, the licensees should perform root cause analysis and then, implement effective corrective actions with particular attention paid to such recurring events. Highlighted is the importance of topical studies focusing on the recurring events to derive the generic safety implications.

(3) Chapter IV

Together with the outlines of ASP analysis approach, described are the ASP analyses of specific events that have the potential of risk significance and trending analysis based on the ASP documents published by the USNRC.

The ASP analysis was carried out for ten actual and one potential STGR events, with use of a consistent event tree model newly developed, to identify risk significant anomalies observed during the events in terms of the potential for core damage and to derive generic insights useful for examining alternative mitigation measures for SGTR. The estimated CCDPs range from 10^{-4} to 10^{-2}. It is also shown that five SGTR events have relatively high CCDPs which are dominated by the sequence involving failure to depressurize RCS to below SG relief valve setpoint. This means that the delayed identification of tube rupture or failure to timely depressurize the reactor is a generic safety issue for SGTR. The analysis underscores the importance of providing more adequate operating procedures and improving the capability to detect SGTR. It is also pointed out that alternative measures need to be examined for the case of failure to depressurize.

Quantitative risk trends were examined using newly proposed risk indicators, that is, occurrence frequency of precursors and annual core damage probability, to grasp the overall picture of nuclear power industry risks based on approximately 600 precursors identified in the USNRC's ASP Program. It is revealed that the core damage risks at U.S. nuclear power plants have been lowered and the likelihood of risk significant events has been remarkably decreasing. Since these trends are based on the events that actually occurred, the proposed risk indicators may provide more empirical and realistic observations. As well, it is demonstrated that the proposed indicators could be useful for determining the risk characteristics of events, monitoring the risk level at nuclear power plants, and examining the industry risk trends. Furthermore, this study underlines the need to accumulate the ASP analysis results and to employ the proposed indicators as one of the performance indicators for the specific plant category and/or in the utility-level (that is, for individual utilities).

These studies show that the ASP analysis of specific events can draw generic safety implications useful for identifying plant vulnerabilities, and the proposed risk indicators can examine the overall picture of risks at industry and individual plant levels. Therefore, such ASP studies should be actively carried out to obtain the insights for improving the plant safety.

(4) Chapter V

This chapter delineates, at first, the severe accident scenarios at the Fukushima Dai-ich Nuclear Power Plant with use of event trees to clarify the differences among accident sequences at Units 1 to 3 and discusses the actual responses to avoid the severe accidents using the event trees. Next, the differences in their respective positions are clarified by reviewing five investigation reports by the Government, Diet, TEPCO, RJIF and NISA, focusing on the accident progression and causes to specify the issues to be further examined. Moreover, the undiscussed issues are identified to provide insights useful for the near-term regulatory activities including accident investigation by the Nuclear Regulation Authority. The different points clarified are related to; layout design concepts of electrical components such as switchgears, causes of EDG tripping, possibility of IC pipe leakage, reason of manual trip of ICs, design concepts on control logic for closing isolation valves of ICs, validity of manual tripping of HPCI at Unit 3, operator training on manual opening of SRVs, operators training on containment venting, integrity of Unit 2 suppression pool (causes of its pressure drop), hydrogen leak path to the reactor building, etc. The undiscussed issues identified include adequacy of operating procedures applied at Unit 1, design concepts of SGTS valves on "fail open", design concepts on valve configuration in venting lines, and operating basis of startup of containment cooling systems when the tsunami is expected. This review highlights the need to further discuss and examine the technical safety issues so that effective OEF can be performed at other nuclear facilities domestically and worldwide.

(5) Chapter VI

This chapter discusses the insights obtained through the analyses of operating experience in terms of their commonalities in the Fukushima accident, identifies precursors to the Fukushima accident, and addresses generic issues to be resolved. The insights obtained are as follows: 1) the degraded functions of ECCS and electric power systems are common issues at nuclear power plants and the defense-in-depth concept should be incorporated into the plant design and operation more strictly, 2) various safety significant events are identified, some of which are considered precursors to the Fukushima accident, 3) some plants experienced the events beyond the design basis, showing that more attention should be paid to those taking into account the challenges they pose and 4) several factors applicable to the Fukushima accident are identified and the lessons learned should be studied and implemented appropriately.

Six events were selected as precursors to the Fukushima accident, 1) fire-induced massive internal flooding and loss of safety systems, 2) fire-induced and prolonged station blackout involving loss of control room and loss of residual heat removal, 3) external flooding involving loss of safety systems and multi-unit site issue, 4) loss of offsite power with one of two emergency diesel generators inoperable, 5) tsunami-induced external flooding and 6) common-mode loss of instrument power. These events underline the importance of physical separation, the necessity of paying special attention to protection against common cause failures, the need to protect the site against external hazards and the importance of considering new phenomena or unexpected aggravating conditions. As well, highlighted is the importance of robust depressurization measures for executing alternative water supply.

The Fukushima accident brings to light the vulnerabilities to the flooding. To cope with the tsunami beyond design basis, it is considered effective that the defense-in-depth concept is incorporated into the plant design; 1) installation of coastal levee against tsunami beyond design basis, 2) waterproofed vital areas and physical separation, and 3) pumping operation to unwater those areas. In addition, the Fukushima accident exposes the vulnerabilities to loss of DC power disabling manual reactor depressurization and the main control room, insufficient accident management measures, lack of knowledge on "as-built" plant equipment, and so on. These issues might be applicable to other plants and thus, should be resolved on a national and international basis.

This study manifests the importance of analyzing the Fukushima accident more comprehensively and in detail considering the insights gained from past events.

VII.2 Future Studies Required

The operating experience feedback has been recognized important to improve the plant safety. Eventually, as mentioned in Chapter VI, there have been several events which are regarded as precursors to the Fukushima accident and a lot of lessons learned should have been fed back to the design and operation of plants. Therefore, the analysis of operating experience is continuously required to obtain the insights from past events by identifying the event causes and corrective actions taken. In particular, it is important to determine whether the individual events potentially may lead to the recurrence with safety significance and examine the adequate preventive measures to eliminate the direct and/or root causes. As well, the precursor studies should be performed for safety significant events involving safety-related system failure(s) with use of the PRA/PSA technique to predict the event sequences which have the potential for severe accidents.

In order to establish the better operating experience feedback process, several types of event analyses need to be carried out. For example, a systematic and comprehensive review of events is essential to identify the potentially safety significant. The topical study on safety significant events is necessary to examine their trends and characteristics. In such a study, the event reports should be collected and analyzed focusing on the similarities of the direct/root causes of events and the corrective actions taken. Among others, the recurring events should be addressed since the previous corrective actions might have been ineffective or inadequate. As well, particular events should be analyzed, assuming additional failure(s), to examine whether those could lead to serious plant conditions. Finally, the insights obtained from these studies should be disseminated throughout the nuclear community in a timely manner so that the plant licensees/operators and regulatory authorities can feed them back to the plant design, operation and/or regulations. Thus, it is quite important that these studies will be performed domestically and internationally.

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Appendix

List of Publications

A.1 Journal Papers

1-1 K. Abe, M. Nishi, N. Watanabe and K. Kudo, *Development of Computer Code System THALES for Thermal Hydraulic Analysis of Core Meltdown Accident (I) Outlines of Code System and Analytical Models*, Journal of the Atomic Energy Society of Japan, Vol.27, No.11, pp.1035-1046, 1985 (in Japanese).

1-2 K. Suzuki, Y. Kameyama, H. Sayama, N. Watanabe, T. Oikawa and K. Abe, *Synthesis of Fault Trees in Sequential Control Systems – Basic Algorithm and its Application*, Journal of System and Control, Vol.30, No.2, pp.109-119, 1986 (in Japanese).

1-3 N. Watanabe, K. Kobayashi, Y. Arise and M. Okuda, *Development of Computerized Event Sequence Abstracting System – System Design and Analytical Approach*, Journal of the Atomic Energy Society of Japan, Vol.30, No.8, pp. 687-698, 1988 (in Japanese).

1-4 T. Harami, N. Watanabe and T. Takano, *Analytical Studies of Thermal-Hydraulic and Radionuclide Behavior in Severe Accident at Boiling Water Reactor*, Journal of Nuclear Science and Technology, Vol.27, No.2, pp 1212-1223, 1990.

1-5 N. Watanabe and M. Hirano, *Analysis of Operating Experience Involving Loss of Decay Heat Removal during Reactor Shutdown in Pressurized Water Reactors*, Journal of Nuclear Science and Technology, Vol.29, No.12, pp.1212-1223, 1992.

1-6 N. Watanabe, *Current Status of Accident Sequence Precursor Program for Operational Events at Nuclear Power Plants*, Journal of Atomic Energy Society of Japan, Vol.38, No.4, pp.252-261, 1996 (in Japanese).

1-7 N. Watanabe, *Accident Sequence Precursor Analyses for Steam Generator Tube Rupture Events That Actually Occurred*, Reliability Engineering and System Safety, 57, pp.281-297, 1997.

1-8 N. Watanabe and M. Hirano, *World Wide Web for Database of Japanese Translation on International Nuclear Event Scale Reports*, Journal of Atomic Energy Society of Japan, Vol.41, No.6, pp.628-638, 1999 (in Japanese).

1-9 N. Watanabe, *Analysis of Causes of Criticality Accidents at Nuclear Fuel Processing Facilities in Foreign Countries – Similarities to the Criticality Accident at JCO's Uranium Processing Plant*, Journal of Atomic Energy Society of Japan Vol.42, No.11, pp.94-102, 2000 (in Japanese).

1-10 N. Watanabe, *Trends and Characteristics Observed in Nuclear Events Based on International Nuclear Event Scale Reports*, Journal of Atomic Energy Society of Japan, Vol.43, No.10, pp.1017-1026, 2001 (in Japanese).

1-11 T. Kishi, H. Kikuchi, S. Miura, M. Fukuda, M. Hirano and N. Watanabe, *Application of Probabilistic Safety Assessment to the Pipe Break Rupture Incident at Hamaoka Unit-1*, Journal of Nuclear Science and Technology, Vol.41, No.1, pp 77-85, 2004.

1-12 N. Watanabe, *Quantitative Risk Trends Deriving from PSA-based Event Analyses - Analysis of Results from U.S.NRC's Accident Sequence Precursor Program -*, Transaction of Atomic Energy Society of Japan, Vol.3 No.4, pp.396-406, 2004 (in Japanese).

1-13 H. Tamaki, K. Yoshida, N. Watanabe and K. Muramatsu, *Development of Probabilistic Safety Assessment Method for Mixed Oxide Fuel Fabrication Facilities*, Transaction of Atomic Energy Society of Japan, Vol.5, No.2, pp.125-135, 2006 (in Japanese).

1-14 S. Takahara and N. Watanabe, *Trending Analysis of Incidents Involving Primary Water Stress Corrosion Cracking on Alloy 600 Components at U. S. PWRs*, Transaction of Atomic Energy Society of Japan, Vol.5, No.4, pp.282-291, 2006 (in Japanese).

1-15 J. Ishikawa, K. Muramatsu and N.Watanabe, *Uncertainty Evaluation of Source Terms under Sever Accident Conditions at LWRs*, Transaction of Atomic Energy Society of Japan, Vol.5, No.4, pp.305-315, 2006 (in Japanese).

1-16 N. Watanabe, *Trend Analysis of Incidents Involving Setpoint Drift in Safety or Safety/Relief Valves at U. S. LWRs*, Transaction of Atomic Energy Society of Japan, Vol.7, No.1, pp.74-84, 2008 (in Japanese).

1-17 N. Watanabe, *Fire Events at Nuclear Power Plants*, Journal of Japan Association for Fire Science and Engineering, ISSN 0449-9042, Vol.58, No.2, pp.11-16, 2008 (in Japanese).

1-18 H. Ohashi, S. Narumiya, K. Miyata and N. Watanabe, Basics of Seismic Safety for Nuclear Power Plants, Journal of Japan Society for Safety Engineering, Vol.49, No.6, pp.402-405, 2010 (in Japanese).

1-19 M. Hirano, T. Yonomoto, M. Ishigaki, N. Watanabe, Y. Maruyama, Y. Sibamoto, T. Watanabe and K. Moriyama, *Insights from review and analysis of the Fukushima Dai-ichi accident*, Journal of Nuclear Science and Technology, Vol.49, No.1, pp 1-17, 2012

1-20 Q. Liu, J. Ishikawa, Y. Maruyama and N. Watanabe, *A simple method for estimating the structure temperatures and the cesium revaporization inside the reactor pressure vessel - I: Basic concepts and model descriptions for the Fukushima Daiichi Nuclear Power Plant*, Journal of Nuclear Science and Technology, Vol.49, No.5, pp 486-495, 2012.

1-21 Q. Liu, J. Ishikawa, Y. Maruyama and N. Watanabe, *A simple method for estimating the structure temperatures and the cesium revaporization inside the reactor pressure vessel - II: Feasibility study for the Fukushima Daiichi Nuclear Power Plant*, Journal of Nuclear Science and Technology, Vol.49, No.5, pp 479-485, 2012.

1-22 N. Watanabe, T. Yonomoto, H. Tamaki, T. Nakamura and Y. Maruyama, *Review of Five Investigation Committees' Reports on the Fukushima Dai-ichi Nuclear Power Plant Severe Accident – Focusing on Accident Progression and Causes*, Transaction of Atomic Energy Society of Japan, Vol.12, No.2, pp.113-127, 2013, (in Japanese).

A.2 Proceedings

2-1 N. Watanabe, T. Oikawa, T. Tobioka, N. Takama and A. Kinoshita, *Large-scale Reliability Analysis by Decomposition and Integration*, Proc. of ANS/ENS International Topical Meeting on Probabilistic Safety Methods and Applications, San Francisco, 1985.

2-2 K. Abe, M. Nishi, N. Watanabe, K. Kobayashi and K. Kudo, *PWR S2D Sequence Analysis with THALES Code System*, Proc. of ANS/ENS International Topical Meeting on Probabilistic Safety Methods and Applications, San Francisco, 1985.

2-3 N. Watanabe, K. Kobayashi, T. Ishigami, M. Hikawa, Y. Arise and M. Okuda, *Development of the Computerized Event Sequence Abstracting System*, Proc. of International ANS/ENS Topical Meeting on Thermal Reactor Safety, San Diego, 1986.

2-4 K. Abe, N. Watanabe, M. Ida, M. Nishi and T. Noguchi, *Overview of Development and Application of the THALES Code System for Analyzing Progression of Core Meltdown Accident of LWRs*, Proc. of 2nd International Topical Meeting on Nuclear Power Plant Thermal Hydraulics and Operations, Tokyo, 1986.

2-5 K. Abe, N. Watanabe, T. Oikawa, M. Nishi and T. Noguchi, *Development of THALES-CV2 Code for Analyzing Containment Temperature and Pressure Response During Core Meltdown Accident*, Proc. of International ANS/ENS Topical Meeting on Thermal Reactor Safety, San Diego, 1986.

2-6 K. Abe, M. Ida, N. Watanabe, T. Oikawa and M. Nishi, *A Study on the Most Influential Parameters on the Progression of PWR Core Meltdown Accident*, Proc. of International ANS/ENS Topical Meeting on Thermal Reactor Safety, San Diego, 1986.

2-7 N. Takama, A. Kinoshita, N. Watanabe, T. Oikawa and T. Tobioka, *Reliability Analysis by a Decomposition and Integration Method*, Proc. of World Congress III of Chemical Engineering, Tokyo, 1986.

2-8 N. Watanabe, K. Kobayashi, Y. Arise and M. Okuda, *Outlines of Computerized Event Sequence Abstracting System*, The Institute of Electronics, Information and Communication Engineering (IEICE) Technical Reports, Vol.87, No.209, NLC 87-6, 1987 (in Japanese).

2-9 K. Abe, N. Watanabe, M. Ida, T. Sakamoto, T. Ishigami and T. Harami, *Sensitivity Study on PWR Source Terms with THALES/ART Code Package and Effects of In-vessel Thermal-hydraulic Models*, Proc. of Probabilistic Safety Assessment and Risk Management (PSA'87), Zurich, 1987.

2-10 K. Muramatsu, K. Abe, M. Ida, N. Watanabe, T. Harami, N. Yamao and K. Soda, *Sensitivity Study on BWR Source Terms Using the THALES/ART and REMOVAL Codes*, Proc. of the International ENS/ANS Conference on the Thermal Reactor Safety "NUCSAFE88", Avignon, 1988.

2-11 N. Watanabe, K. Kobayashi and K. Fujiki, *CESAS: Computerized Event Sequence Abstracting System – Outlines and Applications*, Proc. of the OECD/CSNI-CEC Specialist Meeting on Trend and Pattern Analysis of Operational Data from Nuclear Power Plants, Rome, 1989 (Invited Paper).

2-12 N. Watanabe, *Computerized Event Sequence Abstracting System – Applications and Studies on Description Styles*, Extended Synopses of Seminar on the Use of

Unusual Event Reports for Improving Nuclear Power Plant Safety, Vienna, 1990.

2-13 N. Watanabe, M. Kajimoto and K. Muramatsu, *A New Modelling Approach for Containment Event Tree Construction – Accident Progression Stage Event Tree Method*, Proc. of the 2nd International Conference on Containment Design and Operation, Toronto, 1990.

2-14 M. Kajimoto, N. Watanabe, M. Funasako and K. Muramatsu, *Analysis of Aerosol Behavior in Containment Overpressure Failure Scenarios in BWR MARK-II Plant*, Proc. of Workshop on Aerosol Behavior and Thermal-hydraulics in the Containment (CSNI Report No. 176), Fontenay-aux-Roses, 1990.

2-15 N. Watanabe, M. Kondo and K. Abe, *Japanese Benchmark Exercise on Fault Tree Analysis – Current Status*, Proc. of Use of PSA for Operational Safety (PSA'91), IAEA-SM-321/6, Vienna, 1991.

2-16 M. Kajimoto, K. Muramatsu and N. Watanabe, *Development of THALES-2, A Computer Code for Coupled Thermal-Hydraulics and Fission Product Transport Analysis for Severe Accident at LWRs and Its Application to Analysis of Fission Product Revaporization Phenomena*, Proc. of International Topical Meeting on Safety of Thermal Reactors, Portland, 1991.

2-17 M. Kajimoto, K. Muramatsu and N. Watanabe, *Analysis of Source Term Uncertainty Issues for LWRs*, Proc. of the 2nd International Conference on Probabilistic Safety Assessment and Management (PSAM-II), San Diego, 1994.

2-18 T. Oikawa, N. Watanabe and M. Hirano, *Evaluation of Occurrence Frequencies of Transients at Nuclear Power Plants in Japan*, Proc. of International Conference on Advances in Operational Safety of Nuclear Power Plants, Vienna, 1995.

2-19 N. Watanabe, T. Tamakoshi, H. Takahashi and M. Hirano, *Analytical Study on Effects of Reactor Depressurization Measures during LOCA Sequences Followed by Loss of HPI in PWRs*, Proc. of 3rd JSME/ASME Joint International Conference on Nuclear Engineering (ICONE-3), Kyoto, 1995.

2-20 N. Watanabe, M. Kajimoto and K. Muramatsu, *Categorization of Core-Damage Sequences by Containment Event Tree Analysis for Boiling Water Reactor with MARK-II Containment*, Proc. of the 3rd International Conference on Containment Design and Operation, Toronto, 1995.

2-21 M. Hirano, N. Watanabe, I. Tatewaki and H. Akiba, *Fracture Mechanics Analysis Focusing on Analysis Method*, Transaction of the 13th International Conference on Structural Mechanics in Reactor Technology (SMiRT-13), Porto Alegre, 1995.

2-22 N. Watanabe, *Trends in Initiating Events and Unavailable Systems for Precursor Events*, Proc. of the 4th International Conference on Probabilistic Safety Assessment and Management (PSAM-4), New York, 1998.

2-23 N. Watanabe, M. Hirano, K. Yoshida and M. Akimoto, *JAERI's Practice for Analysis of Human Behavior during Abnormal Occurrences at Research Facilities*, Proc. of ICNCA Nuclear Safety Culture Workshop, Sydney, 1998.

2-24 N. Watanabe, M. Hirano and H. Takahashi, *Sophistication of SGTR Event Tree for Accident Sequence Precursor Analysis*, Proc. of the International Topical Meeting on Probabilistic Safety Assessment (PSA'99), Washington DC, 1999.

2-25 M. Nozawa and N. Watanabe, *Application of IAEA's International Nuclear*

Event Scale to Events at Testing/Research Reactors in Japan, Proc. of the 6th Asian Symposium on Research Reactors (ASRR-VI), Mito, 1999.

2-26 N. Watanabe, K. Muramatsu, K. Ogura and J. Mori, *Risk Insights for PWRs Derived from Accident Sequence Precursor Analysis Results*, Proc. of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM-5), Osaka, 2000.

2-27 M. Nozawa, and N. Watanabe, *Status of Japanese Response to the JCO Accident*, Proc. of A U.S.-Japan Workshop on the Role of Nuclear Energy in a New Environment of Deregulation and Climate Change, Washington DC, 2000.

2-28 N. Watanabe, *Research Activities on Accident Sequence Precursor Study at JAERI*, Proc. of OECD/NEA-CSNI Workshop on Precursor Analysis, Brussels (2001).

2-29 H. Tamaki, N. Watanabe and K. Muramatsu, *Hazard Identification of Criticality Accidents at the JCO Facility*, Proc. of 2001 Topical Meeting on Practical Implementation of Nuclear Criticality Safety, Reno, 2001.

A.3 Reports

3-1 N. Watanabe, *Fault Tree Analysis on BWR Core Spray System*, JAERI-M 82-060, 1982 (in Japanese).

3-2 N. Watanabe, T. Ishigami, M. Hikawa, H. Horii and T. Tobioka, *Comparative Study: Top Event Unavailability by Point Estimates and Uncertainty Propagation*, JAERI-M 82-125, 1983 (in Japanese).

3-3 K. Abe, M. Nishi, N. Watanabe, J. Ishiguro, K. Kudo and A. Nakamura, *Description of THALES-CV1: A Computer Code for Evaluating Containment Temperature and Pressure during Core Meltdown Accident*, JAERI-M 83-037, 1983 (in Japanese).

3-4 T. Ishigami, N. Watanabe, M. Hikawa, H. Kaneki, H. Horii and S. Sasaki, *Users' Manual for the FTA-J (Fault Tree Analysis – JAERI) Code System*, JAERI-M 83-169, 1983 (in Japanese).

3-5 N. Watanabe, T. Chiba, T. Oikawa and K. Abe, *Users' Manual of the LOGIC1 Program Library for Modeling of Instrument and Control System*, JAERI-M 84-173, 1984 (in Japanese).

3-6 N. Watanabe, M. Kondo, K. Uno, T. Chigusa and T. Harami, *LSOP-initiated Event Tree Analysis for BWR*, JAERI-M 89-025, 1989 (in Japanese).

3-7 N. Watanabe and S. Higuchi, *ETAP Users' Manual*, JAERI-M 90-193, 1990 (in Japanese).

3-8 N. Watanabe, M. Hirano and T. Oikawa, *Analysis of Operating Experience data in Nuclear Power Plants - Loss of Decay Heat Removal during Reactor Shutdown* -, JAERI-M 91-143, 1991 (in Japanese).

3-9 N. Watanabe and M. Kiyota, *Users' Manual for Fault Tree Analysis Code: CUT-TD*, JAERI-M 92-089, 1992 (in Japanese).

3-10 N. Watanabe and K. Tamura, *Users' Guide for an Interactive Fault Tree Editor: PC-CREFTS*, JAERI-M 93-120, 1993 (in Japanese).

3-11 N. Watanabe, *Review and Comparison of Dominant Core Damage Sequences in PWRs Identified in NUREG-1150*, JAERI-M 93-164, 1993 (in Japanese).

3-12 N. Watanabe, *Compilation and Review of Loss of Decay Heat Removal during Reactor Shutdown in PWRs*, JAERI-M 94-076, 1994 (in Japanese).

3-13 N. Watanabe, *Review of Recent Events and Regulatory Guidances Concerning Emergency Core Cooling System Strainer Blockage at Boiling Water Reactors in the U. S. A.*, JAERI-Review 97-008, 1997 (in Japanese).

3-14 N. Watanabe and M. Hirano, *IAEA-ASSET's Root Cause Analysis Method Applied to Sodium Leakage Incident at Monju*, JAERI-Tech 97-036, 1997 (in Japanese).

3-15 N. Watanabe, *Trending Analysis of Precursor Events*, JAERI-Tech 97-067, 1998.

3-16 N. Watanabe and T. Suzudo, *Review Report: Safety and Reliability Issues on Digital Instrumentation and Control Systems in Nuclear Power Plants and United States Nuclear Regulatory Commission's Dispositions*, JAERI-Review 98-013, 1998 (in Japanese).

3-17 N. Watanabe, T. Oikawa and M. Hirano, *Compilation of INES (International Nuclear Event Scale Information: Japanese Translation*, JAERI-Data/Code 98-023, 1998 (in Japanese).

3-18 T. Suzudo and N. Watanabe, *Theory and State-of-the-Art Technology of Software Reliability*, JAERI-Review 99-027, 1999 (in Japanese).

3-19 N. Watanabe and H. Tamaki, *Review and Compilation of Criticality Accidents in Nuclear Fuel Processing Facilities Outside of Japan*, JAERI-Review 2000-006, 2000, (in Japanese).

3-20 N. Watanabe, *Compilation of INES (International Nuclear Event Scale Information: Japanese Translation (Vol.2)*, JAERI-Data/Code 2001-002, 2001 (in Japanese).

3-21 Nuclear and Industrial Safety Agency, *Investigation Report on Pipe Rupture at Hamaoka Nuclear Power Station Unit-1*, 2002.

3-22 N. Watanabe, *Review of Incident Involving Reactor Pressure Vessel Head Degradation at U. S. PWR*, JAERI-Review 2004-015, 2004 (in Japanese).

3-23 N. Watanabe and S. Takahara, *Review of Recent Events Involving Dgradation of Reactor Coolant System Pressure Boundary due to Primary Water Stress Corrosion Cracking at U. S. PWRs*, JAEA-Review 2006-027, 2006 (in Japanese).

3-24 N. Watanabe, *Analysis of Nuclear and Radiological Events*, JAEA-Review 2006-040, 2006 (in Japanese).

3-25 N. Watanabe, *Probabilistic Safety Assessment for Nuclear Power Plants: Level 1 PSA for Internal Events*, JAEA-Review 2006-041, 2006 (in Japanese).

3-26 N. Watanabe, *Review of Recent Events Involving Setpoint Drift in Safety or Safety/Relief Valves at U. S. Nuclear Power Plants*, JAEA-Review 2007-043, 2007 (in Japanese).

3-27 M. Kimura, N. Watanabe and T. Homma, *Survey on Emergency Action Levels in the United States of America*, JAEA-Review 2010-021, 2010 (in Japanese).

A.4 Restricted Reports

4-1 N. Watanabe, T. Oikawa, M. Hirano, F. Araya, K. Fujiki and K. Namatame, *Analysis and Compilation of Events Reported to OECD/NEA IRS*, 1992 (in

Japanese).
4-2 T. Oikawa, N. Watanabe, T. Tamakoshi and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (2)*, 1994 (in Japanese).
4-3 T. Oikawa, N. Watanabe, T. Tamakoshi and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (3)*, 1996 (in Japanese).
4-4 N. Watanabe, T. Oikawa and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (4)*, 1996 (in Japanese).
4-5 N. Watanabe, T. Oikawa and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA IRS (5)*, 1997 (in Japanese).
4-6 N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (1997 Edition)*, 1998 (in Japanese).
4-7 N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (1998 Edition)*, 1999 (in Japanese).
4-8 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (1999 Edition)*, 2000 (in Japanese).
4-9 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2000 Edition)*, 2001 (in Japanese).
4-10 N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2001 Edition)*, 2002 (in Japanese).
4-11 N. Watanabe and M. Hirano, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2002 Edition)*, 2003 (in Japanese).
4-12 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2003 Edition)*, 2004 (in Japanese).
4-13 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2004 Edition)*, 2005 (in Japanese).
4-14 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2005 Edition)*, 2006 (in Japanese).
4-15 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2006 Edition)*, 2007 (in Japanese).
4-16 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2007 Edition)*, 2008 (in Japanese).
4-17 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2008 Edition)*, 2009 (in Japanese).
4-18 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2009 Edition)*, 2010 (in Japanese).
4-19 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2010Edition)*, 2011(in Japanese).
4-20 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2011 Edition)*, 2012 (in Japanese).
4-21 N. Watanabe, *Analysis and Compilation of Events Reported to OECD/NEA-IAEA IRS (2012 Edition)*, 2013 (in Japanese).

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