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NUCLEAR ENERGY AGENCY
COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES

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LOSS OF RESIDUAL HEAT REMOVAL (RHR) WHILE AT MID-LOOP CONDITIONS CORRECTIVE ACTIONS

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

EXECUTIVE SUMMARY

This report is the third in a series of the WGOE (Working Group on Operating Experience) of reports on recurring events. The first two reports NEA/CSNI/R(1999)19 and NEA/CSNI/R(2003)13 discussed a variety of recurring events extracted from more than two decades of international operating experience exchange. In addition to this, the WGOE has issued a consolidated Technical Opinion paper (CSNI Technical Opinion Paper - No. 3, Recurring Events) that summarizes the essence of both of the first two reports for in a brief form.

On the basis of the first two reports the WGOE decided to study in more detail the various sorts of corrective actions associated with recurring events. The aim was to learn about effective corrective actions to prevent from them or to mitigate their consequences. Several recurring events were evaluated against specified selection criteria, and on this basis it was decided to study in detail the PWR event Loss of Residual Heat Removal (RHR) While at Mid-Loop conditions. This event had several properties that made it relevant.

The event has reoccurred in a number of member states over the past 25 years and, to a degree, is still occurring in some countries. A number of risk studies have identified this sequence as the highest-risk scenario in shutdown and low-power operations. This is due to that, e.g., at some plants there may not be any reliable method of observing the water level; it is essential to maintain sufficient level in the primary system hot leg to provide suction supply to the RHR pumps; there may be no automatic system that would automatically restore water level and consequently human actions are in a major role; there is in some cases little time, perhaps substantially less than one hour, between loss of RHR and the onset of boiling in the primary system. Finally, not all countries regard the issue as resolved.

The report first presents the criteria for selecting the recurrent event for study of corrective actions, then the event and its features are discussed. The report presents the questionnaire used to collect information from member countries and the replies obtained. An analysis of the replies is provided and the conclusions based on this are finally presented.

The results show that style of corrective actions in WGOE member countries ranges from voluntary (in response to suggestions by the regulatory authority, or the individual operating organization, or owners' groups, or industry organizations) to mandatory, as imposed by the regulatory authority. The conclusion is that voluntary initiatives are not very effective in eliminating the scenario, while the mandatory solutions imposed by the regulator (or, in some cases, solutions jointly reached by the utility and regulator which are then converted into requirements) seem to be quite efficient.

The report was prepared by Dr. Denwood Ross, on the basis of discussions with, and input provided by, the members of the Working Group.

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1. INTRODUCTION

The Committee on Safety of Nuclear Installations (CSNI), through its Working Group on Operating Experience (WGOE)[1], has issued two studies on recurring events [1], [2], and, in cooperation with the World Association of Nuclear Operators (WANO), sponsored a workshop on recurring events [3]. (References are in Appendix A).

The WGOE proposed, at its October 2003 meeting, an extension of the recurring events work into the area of corrective actions. The scope of that task was the review a range of corrective actions that were applied to recurring events, in order to determine the most effective type of actions. This task extension was reviewed and approved by the CSNI during its December 2003 meeting. The specific scope of this effort is detailed in Appendix B.

Corrective actions as applied by the member states come in a variety of forms. The actions might be specific only to the plant that encountered the event of concern, and the response by the licensee and regulatory body might be tailored to that plant. By contrast, the regulatory authority (or, perhaps, and industry body such as the Institute for Nuclear Power Operations, or INPO) might observe plant performance in general and then develop some sort of broad generic guidance. In this case, the guidance might be informal and in the form of a recommendation. In the case where the regulatory body is the party taking the action, there also could be mandatory instructions, such as rule change or order modifying license. All of these types of reaction were observed during this study.

In addition to various sorts and types of regulatory actions, there were also a variety of recurring events that had emerged from the preceding studies [1, 2]. It was decided to focus on one class of recurring event, and then analyze the corrective actions for the selected event for trial use. The recurring event categories considered by WGOE for this study by WGOE were:

- Pipe cracking
- Strainer clogging
- Boric acid corrosion
- BWR power oscillations
- Hydrogen detonations
- Loss of service water
- Loss of RHR while at reduced inventory (so-called mid-loop operation)

[1] The WGOE was transferred from CSNI to CNRA in 2005.

The WGOE considered the following attributes of recurring events, in general, in order to assist the group in the selection of one recurring event for the study:

- Are multiple countries involved?
- Is there a long period of observation of this event?
- Is it still occurring?
- Is it of high-risk significance?
- Is it considered resolved?

The seven recurring events were analyzed according to these five factors:

1. Pipe cracking. Pipe cracking has occurred in several countries, notably France, USA, and Germany. Reported events date back to at least 1987. Pipe cracking is probably still recurring, although it may be considered resolved by most countries. There are multiple and, for the most part, unrelated causes. Thermal fatigue, one source of pipe cracking, may be resolved.

2. Strainer clogging. This is a concern for most countries, although clogging itself has occurred in only a few. The concern is what might happen for an extreme event such as loss of cooling accident (LOCA). There have been concerns for loose debris in containments, both PWR and BWR, for many years. The term strainer refers to the inlet manifold for the BWRs and the emergency sump screen for the PWRs. It cannot be considered resolved at this time, as many plants are considering modifications. In the midst of changes, it does not seem appropriate to study this event category, as the existing database may have little relevance to the plant after modifications are made.

3. Boric Acid Corrosion. Corrosion of the upper head or primary system piping in PWRs has been reported in the USA, Brazil, and Sweden. Some of these reports go back more than 20 years. It may still be occurring, although the recent severe event at the Davis Besse plant has probably greatly raised the sensitivity of this event by the licensee and regulator alike. It may be considered resolved; certainly the hazard has been illustrated in a graphic manner. It is of very high-risk significance, in that it could lead to a loss of coolant accident. As for strainer clogging, it is concluded that this event is dynamic in that regulatory interest and actions are serving to modify the sample population.

4. BWR Oscillations. BWR instability, illustrated by power oscillations, has occurred in BWRs for more than 20 years. Such oscillations have occurred in the USA, Germany, Switzerland, Spain, Sweden, and Italy. The solution is more or less empirical and may not be solved. In the past such oscillations have not been regulated to any significant degree. Some countries have installed advanced instrumentation and control, which should suffice, but many have not. BWR power oscillation is not considered to be of high-risk significance.

5. Hydrogen detonations. Some BWRs have a small pipeline connecting to the top of the reactor upper head that can, under certain circumstances, have an accumulation of hydrogen, and oxygen, that ignites or detonates. Detonations have recently been observed in Germany and Japan. Similar events occurred more than 15 years ago. There is not enough data to consider that the event is still occurring, or resolved. This event does not seem to affect enough countries, or be sufficiently abundant, to warrant a detailed corrective action study.

6. Loss of Service Water. Loss of service water, whether it be an interruption at the ultimate heat sink itself, or else some common mode failure in the emergency service water system, is risk-important. It has occurred in one form or another for more than 15 years, in a number of member states. The corrective actions for loss of the ultimate heat sink are perhaps resolved, or at least identified, but implementation may be lacking. Firm and prescriptive regulatory action seems to be lacking, also. This event could have served as the selected topic for corrective actions.

7. Loss of RHR While at Mid-loop. During outages it is sometimes necessary to reduce the inventory in a PWR primary system to enable opening of the pressuriser manway or else an inspection hatch in a steam generator. In many cases the reactor core is still in the vessel, thus the RHR must keep running. However, the suction supply for the RHR, which is at the bottom of one of the hot legs, may be starved for water if the level is reduced too much. This interrupts the flow of RHR through the RHR heat exchangers, and also induces cavitation in the RHR pumps (which can lead to pump failure). This loss of RHR while at mid-loop has occurred more than 50 times over the past 25 years. It is considered of high risk-importance. It may be solved in some countries, where specific and mandatory guidance has issued from the regulator. In other cases, the event may not be resolved. (It does appear that the event frequency has diminished).

On the basis of the analyses presented above, the WGOE selected the last topic, Loss of RHR While at Mid-Loop, as the event to be studied for corrective action effectiveness.

It was noticed, as the study progressed, that there were some events occurring with some resemblance to Loss of RHR While at Mid-Loop in both the CANDU and WWER designs. However, the phenomena were sufficiently different to preclude events at these two reactor designs from the general study.

The study process included:

- Issuance of a questionnaire (Chapter 2).

- Discussion of the phenomena of loss of RHR at mid-loop, and review of risk significance (Chapter 3).

- Member state responses to questionnaire (Chapter 4).

- Analysis and Conclusions (Chapter 5).

2. PHENOMENA OF LOSS OF RHR AT MID-LOOP AND RISK SIGNIFICANCE

2.1 Description of the mid-loop sequence

The design of many PWRs requires that, during certain phases of the shutdown period, maintenance operations be performed while the water level in the primary system is lowered. The water level may be reduced to the mid-point of the outlet leg (hot leg) of the primary coolant system, and thus the term mid-loop applies. This reduced inventory phase allows opening of flanges on the steam generator, and sometimes the pressuriser, to permit access for inspection and repair.

In many cases some, or all, of the reactor core may retain in the reactor vessel. It is necessary to provide a continuous heat removal system for the reactor decay heat. Inasmuch as the steam generators can no longer perform this function (there is no forced or natural circulation through the steam generators) the sole heat removal system is the Residual Heat Removal (RHR) system (sometimes termed the Decay Heat Removal system). This system has an inlet pipe connected to the primary system hot leg. Heated water, which exits the reactor vessel, flows down through the hot leg to the inlet of the RHR system. Heat is removed by circulating through heat exchangers, and the cooled water is returned to the cold (inlet) leg of the primary coolant system. Figure 2.1 is an elevation view of the primary coolant system and shows the relative elevations of these important parts of the primary system.

During this heat removal process, it is important to maintain the reduced level within a somewhat narrow range. The level must be low enough to enable opening of the inspection hatches, but not so low as to uncover the pipe leading from the hot leg to the RHR system (this pipe is sometimes referred to the drop line). If the water level is too low, the water supply to the RHR system is disrupted. At first a vortex may form and air may be ingested. A complete interruption of water supply leads to a pump damage mechanism known as cavitation; this can lead to irreversible pump damage.

This sequence has occurred in various forms more than 50 times over the past 25 years, and it is the sequence that is focused on in this report.

2.2 Phenomena of Loss of Shutdown Cooling at Mid-loop

Should the inlet supply to the RHR system be interrupted, the following sequence of events might take place:

- Following a loss of suction supply to the RHR pumps, pump cavitation might occur, with the potential for damage to the pumps, perhaps of an irreversible nature. This would be indicated in the control room by low flow rate on the RHR system, and low RHR pump current.

- A precursor to a full loss of suction supply, when the water level is marginally sufficient in the hot leg supply line, is vortexing. At this point there could be swirling or vortexing in the drop line, and ingestion of air bubbles into the drop line. This would be indicated in the control room by variation in shutdown cooling flow rate, and wide variations in RHR pump current meters.

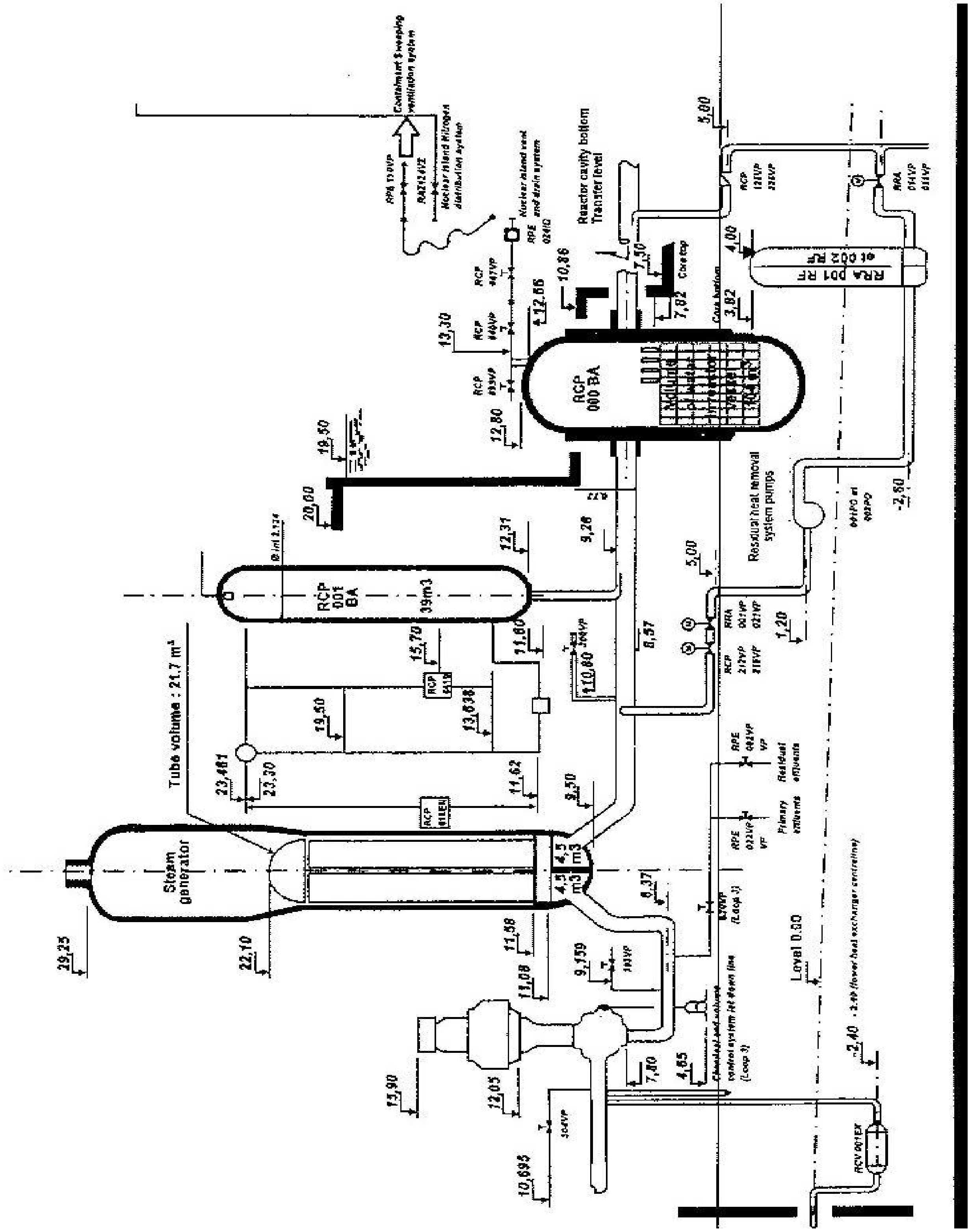

Figure 2.1 Elevation View of Primary Coolant System

- Dependent on the plant-specific instrumentation, there could be an indication (or alarm) on low reactor vessel level. Inaccurate level indications, and/or lack of appropriate alarm signals, have contributed to the direct cause for loss of shutdown cooling at mid-loop.

- Following loss of shutdown cooling, and provided some (or all) of the irradiated core is still in-vessel, there will be a heat up of the water in the reactor vessel. The heat up rate will be dependent on the decay heat level, and the mass of remaining water. In some instances the water was heated to saturation and bulk boiling ensued. This could take place in less than one hour. During this heat up phase there would be an indication of temperature rise on the core exit thermocouples.

- If the reactor happened to be in a closed condition (i.e., no open inspection hatches) then, as heat up started, the reactor pressure would increase. This is significant, as reactor pressure could inhibit injection of water by gravity drain from the large-capacity Residual Water Storage Tank.

- Inasmuch as during this situation there is no liquid in the steam generator primary side, the potential for removing heat by the steam generators is essentially zero.

- If the reactor happened to be open (i.e., some inspection hatches open) then, as the reactor heated up there could be some steam exiting the hatches, and this would possibly be noticed by personnel inside containment.

- Potential sources of water makeup include emergency core cooling system high-pressure injection pumps. The low-pressure pumps are also the RHR pumps, and thus might not be available). Use of these pumps would be by operator action, as automatic action is not available in the shutdown mode.

2.3 Risk perspectives of loss of RHR coolant while at mid-loop

The risk from Low Power and Shutdown (LPSD) conditions has been recognized for some time as somewhat comparable to full power risk, and among the LPSD sequences the Loss of Shutdown Cooling while at mid-loop conditions is among the highest. The basis for this statement comes from an examination of information from the following references.

- A USA Advisory Committee on Reactor Safeguards (ACRS) stated in a letter [4] that the risk during LSD operations has been estimated as comparable to that of full power operations.

- Switzerland noted [5] that recent PSA studies have generally found that shutdown risk is a large contributor to overall risk and that the probability of core damage, given an event during shutdown, can be large. This is due, in part by a smaller primary system inventory, and is also affected by the fact that some safety systems may have been disconnected.

- Lanore, in [6], observed that, for reactors in France, the risk of core melt when the reactor is in a shutdown conditions is significant, and even higher than during full power, for particular plant configurations. In the same reference, Muller-Ecker repeated this conclusion for German reactors. Further, for both countries, it was stated that: *"The core melt frequency was in all studies particularly high for a loss of residual heat removal system during mid-loop operation because only a short time is available for the operator to take any action, due to the low primary coolant inventory"*.

- The CSNI issued a compendium of practices on safety improvements in low-power and shutdown operating modes [7]. Some relevant findings of that report were:

  a) For a PWR, the single risk-dominant configuration is the mid-loop operation.

  b) The main concern during mid-loop is the loss of heat removal (due to the coolant level reduced below the pump suction, followed by rapid boiling in the core.

  c) Human interactions are significant contributors to the overall risk level at shutdown and low power, as most of the automatic safety signals are bypassed or disconnected in these operating modes. The safety concept during operation of having a series of redundant and diversified safety systems to cope with initiating events is not present in low power and shutdown modes. Automatic operation is blocked. Systems are out of service because of maintenance. Operators have less instrumentation operable, and some of the key variables are not monitored.

  d) Hardware improvements are an important contributor to safety in low power and shutdown modes, but must be combined with administrative and procedural improvements, in order to be effective.

- The US performed an operational overview of 19 refuelling outages in order to gain a comprehensive understanding of the overall risk [8]. A wide range of values of outage risk, both in the predicted and observed, was seen across the spectrum of plants. In a majority of the PWR outages examined, the facility went to a mid-loop condition (or some sort of reduced inventory) with a concurrent unavailability of either an emergency generator or some sort of significant switchyard maintenance, or sometimes both. Ad hoc contingency measures and controls were in place. For many of these instances the time for the primary coolant to reach saturation temperature, for a loss of shutdown cooling, was less than 30 minutes. This configuration is believed to represent the highest risk state for shutdown conditions.

- Some added insight on risk was provided in NRC Commission paper SECY-00-0007 [9]. The purpose of this paper was to inform the Commission of a plan to further develop low power and shutdown risk information. Here the NRC staff observed that:

  a) There remains a broad consensus that the frequencies of core damage accidents initiated during low power and shutdown (LPSD) operations can be significant;

  b) Numerous events that are potentially risk-significant have occurred at US plants during LPSD conditions;

  c) Licensees have developed qualitative and quantitative methods and tools for managing safety during LPSD conditions;

  d) Current methods provide a strong foundation for considering LPSD accident risks in regulatory activities;

  e) There is a strong need to develop better guidance for licensees and NRC staff on how to use current LPSD risk analysis methods in risk-informed regulatory activities;

  f) In selected areas there is also a need to improve methods and tools for assessing LPSD accident risk.

3. QUESTIONNAIRE ISSUED TO MEMBER STATES

In other studies it has been found that not all events of interest are archived in the Advanced Incident Reporting System (AIRS). Although AIRS did provide a substantial number of events, a questionnaire process was formulated and sent to the member states (Appendix C). The database thus consisted of:

- AIRS.
- Response to questionnaire.
- Reports and presentations in WGOE meetings.

In addition to asking for conventional information about events (such as what happened, what were the consequences, and so forth) the questionnaire also sought information on the regulatory aspects. Since the purpose of this study is to draw conclusions about the efficiency of corrective actions, several of the questions were aimed in that direction, including:

- Was the facility licensing basis changed?
- Were regulations changed?
- Were the corrective actions prescriptive in nature, or performance-based?
- Were corrective actions efficient, and did this close the issue?

The country-by country responses are discussed in the following chapter.

4. MEMBER STATE RESPONSES

4.1 General

A number of member states responded with discussions of events involving reduced inventory in the primary system, some of which had an actual loss of shutdown heat removal. Corrective actions were mentioned, in a varying level of detail. Also, the AIRS was consulted for additional detail on many of these events. The country-specific responses are presented in this section.

4.2 Belgium

4.2.1 Events

Belgium provided an IRS report 1628, regarding an event that took place at Doel-1 in October 1994. Prior to restart after a refuelling outage, a routine degassing operation was underway. The primary system is placed under a vacuum to enable degassing. The combination of the vacuum; too high a flow rate of shutdown cooling; too low level in the reactor system, and presence of air in the shutdown cooling pumps led to vortex formation and complete loss of shutdown cooling. A number of actions, including venting of the pumps, increase of water level, and throttling of flow allowed restoration of shutdown cooling. The primary temperature rose about 10 deg C during the transient. This event occurred at the end of refuelling, when considerable decay in the afterheat had taken place.

4.2.2 Corrective Actions

The corrective actions included procedures modifications, improvement in calibration and reliability of water level instruments, and training.

Some additional events were reported at WGOE meetings, although not to IRS. These were discussed in [2]. There were 5 spurious draining events of auxiliary nuclear circuits and an inadvertent level drop of the primary coolant during mid-loop operation, which required safety injection (all at the same NPP).

Root causes and lessons learned are discussed in [2], from the Belgian perspective. The generic lessons learned include:

- Written and auditable operation experience feedback process.
- Identification and reporting of events.
- Screening of events for in-depth analysis to be done by persons with good knowledge of HPP history and of the risk importance of events.
- Documentation of corrective actions for preventing recurring events.
- Process for detection of recurrent unavailabilities of safety-related equipment.
- Improved event analysis, to find root causes.

4.3 France

4.3.1 Events

France indicated that many incidents involving problems with mid-loop operation had been reported to the regulator. Generally they concerned uncontrolled decrease in primary circuit water level, but with no actual consequences. Only a few led to loss of, or degradation of, the RHR pumps. Since the 1996-97-time span, after corrective measures were taken, no other serious event has been observed. The IRS events described below are representative of the problems in France in the early 90s.

France has provided three IRS reports (659; 1197; 7208). In IRS 659 contains a description of loss of RHR at Blayais-4 in June 1985. The operators allowed the water level to drop to an abnormally low level. Air was ingested into the RHR system and two pumps lost their prime. After 25 minutes, water had been resupplied and the pumps were restarted. The direct cause of the event was an erroneous assessment of the primary water level required to drain the steam generator tubes. The procedure was deficient. Also, the level must be decreased below the alarm setpoint, so there is no additional alarm available for the level at which pumps lose prime. The water level instruments were insufficient. Corrective actions include modifications in procedures and an improved level measurement system.

At Belleville-2 (IRS 1197) there was a vortex formation in the RHR suction inlet which lead to fluctuations in pump current of the RHR pumps. This event happened in May 1991. The plant had been shutdown for 5 days. The water level was being lowered to the mid-loop condition. Due to anomalies in the measurement system, the level dropped too far. The situation lasted for less than one hour. The operator stopped the draining and initiated makeup. Five causes were listed:

- Failure to follow procedure.
- Discrepancies between the reference procedure and the plant-specific procedure.
- Unsuitable current scale in control room.
- Poor calibration of level sensor.
- Poor interpretation of the ultrasonic detector range.

Corrective actions included changes in procedures, modification of instruments, and more training.

A third event occurred at Blayais-3 in August 1996 (IRS 7208). The plant was in a refuelling outage. The plant was in cold shutdown, and the pressuriser vent hole was being opened. Since water was being injected into the system there was periodic drainage in order to maintain level. The draining was to last about 15 minutes every ten hours. Due to the multiple number of tasks assigned, the operator left the draining in progress. A low level light started flashing and this alerted the operator, who restored level. There was no effect on RHR.

4.3.2 Corrective Actions

A Framatome Owner's Group was formed in mid-1992. This group had the task of initiating a specific technical program focused on transient scenarios associated with mid-loop conditions and loss of RHR. This group was concerned about analyses that confirmed that the riskiest phases are during operations at mid-loop, as well as during draining phases of the reactor cooling system. The major contributors to loss of RHR were the relative unreliability of the temporary level instruments, and operator errors. The main recommendations issued by this working group were:

1. Improve the reliability and accuracy of water level measurements.

2. Get early information to the operator regarding vortexing or cavitation.

3. Prevent a draining process in case of low level in the hot leg.

4. Install automatic water makeup system.

5. Install in-vessel temperature measurement specifically for use in mid-loop.

As a result, design and operational modifications in response to mid-loop problems include:

1. an automatic make-up function, actuated by a low level signal;

2. new and more reliable instrumentation;

3. shutting down of the cavity drain pump in situations of an impending vortex;

4. a vortex detection device;

5. changes in the operational technical specifications;

6. suppression of mid-loop before core unloading.

The regulatory authority, since 1994, required the operating utility to declare and precisely describe each scheduled operation at mid-loop conditions, and to explain the mitigation measures that would avoid loss of decay heat removal. The schedule for the above modifications was mutually agreed to by the operator and the regulator. The modifications were also approved by the Advisory Committee on Reactor Safety.

There was sufficient experience with mid-loop events in France to form the database for the above counter-measures, and thus the type and extent of the modifications were not influenced by experience in other countries although institutions such as IRSN are active in follow-up of international events.

4.4 Japan

4.4.1 Events

Japan has no records in IRS regarding loss of RHR while at mid-loop conditions. They did, however, respond to the questionnaire.

4.4.2 Corrective Actions

Based on events that took place in other countries, Japan has made some changes in design and operation relative to loss of DHR while in the shutdown mode. An additional narrow-range water level meter was added to increase monitoring capability during mid-loop conditions. In addition, operating procedures were changed to secure the water head during mid-loop, so that the NPSH was increased.

These changes were made by the utilities, based on the information available. Some utilities preferred ultrasonic level meters, and others preferred local instruments.

Some utilities, when establishing the periodic inspection program, contrive to decrease shutdown risk referring to the individual shutdown PSA, and sometimes change their systems and components inspection and maintenance schedule to increase reliability.

No incident/accident report is issued by the utilities to the agency regarding the mid-loop operation to date in Japan.

The facility licensing bases were not changed, and new regulations or orders were not issued. The motivation by the utilities was to increase plant safety margin, and decrease the core damage frequency. Japan believes that neither licensing nor orders are necessary.

4.5 South Korea

4.5.1 Events

South Korea stated that, although there is no official record in IRS regarding loss of RHR while at mid-loop, there were two events in this area that were analyzed by the operating utility. These occurred at Kori-2 in June 1984 and at Kori-3 in October 1987.

Initially these two events were not reported to the regulatory body, as the reporting requirements at that time did not cover these events. For the June 1984 event at Kori-2, there was a loss of RHR while at mid-loop during refuelling. Due to an instrument error, there was a false low-low level signal generated from the RWST, and, as a consequence the signal transferred the suction supply to the containment sump (which, of course, was dry at this time). The operating RHR pump failed due to loss of suction supply. The primary system increased in temperature by 95^{o} C. There was steaming from the primary system. There were a number of actions taken by the operating staff; RHR was off for about one hour.

In the second case, at Kori-3, there was a loss of suction to the RHR pumps during the RCS draining operation. The RHR was off for about 40 minutes, until level was restored by draining from the RWST. Primary temperatures rose to almost the boiling point.

4.5.2 Corrective Actions

There were some immediate countermeasures taken by the utility. Later the regulatory body issued a document entitled “The Safety Requirements during Mid-loop Operation for PWR”. These requirements included:

- Installation of added instrumentation to monitor various parameters associated with mid-loop operation;
- Development of Procedures;
- Implementation of Operator Training;
- Supplementing the Plant Technical Specifications.

The regulatory body based its requirements on information from IRS; from NRC issuances, and from analyses of experience noted above. The requirements were implemented by administrative order. New regulations were issued in March 1986. The schedule for compliance with the order was negotiated with the utility. These new requirements were characterized as prescriptive in nature. The regulatory body believes that this is now a closed issue. However, they intend to follow further developments in the international community.

4.6 Spain

4.6.1 Events

Spain reported two events of interest. One of these was reported to the IRS. On March 25, 1997, there was a flow reduction at Asco below the amount required by the technical specification. The reason was a reduced visibility in the spent fuel pool. At Almaraz, on 24 September 1990 (IRS 1187), there was total loss of RHR because of a misalignment of the one RHR train, and unavailability of the other train. This lasted for 46 minutes, and there was a heat-up of the primary coolant from 40 deg C to 70 deg C.

Some procedural changes were required:

- Avoid RHR maintenance during reduced inventory
- Specify minimum time before reducing inventory
- Specify RHR flow window to keep primary temperature below 60 deg C
- Intensify surveillance on cooling parameters during mid-loop

4.6.2 Corrective Actions

It was concluded that plant licensing basis, specifically the single failure requirement, was sufficient and no changes to the licensing basis were needed. A number of requirements were issued by CSN to the NPPs, in order to comply with the licensing basis, and these changes were regarded as prescriptive. For the most part, CSN issued these changes and prescriptive requirements on the basis of in-country experience, although the NRC documents on the 1987 Diablo Canyon experience (specifically, NRC Generic Letter 88-17, which was adopted by CSN) were influential.

CSN expressed the concern that too much time elapsed between the occurrence of the problem and the implementation of corrective actions. There were some misunderstandings about the changes in technical specifications that were necessary to correct the problem. At first, there was also less than satisfactory accounting for the need to comply with the single failure postulate. The plants were somewhat reluctant to adopt changes in the technical specifications, but changes were made to the technical specifications.

At present, CSN believes that the issue is apparently solved.

4.7 Switzerland

4.7.1 Events

Switzerland reported two events with a direct impact on decay heat removal, one in 1992 and one in 2000. The 1992 event was described in a monthly operating report, and the 2000 event was reported by a licensee event report. Both occurred in a Westinghouse-designed reactor. The event in 2000 was also reported to IRS (IRS 7472).

Another event occurred in September 2003 in a KWU design. The KWU design had an event while refilling the reactor cavity while the core was unloaded. Human error in valve alignment resulted in low level in the fuel pool and one train of shutdown cooling of the fuel pool was lost. Some administrative changes in system verification (double-checking) were made.

4.7.2 Corrective Actions

The corrective actions fell into two categories: generic and plant-specific. There were no direct regulatory requirements (or requests) to improve the safety and stability of mid-loop operation, with one exception being that for an ultrasonic mid-loop level device in Beznau. However, there were a number of indirect requirements, which had a significant impact on the safety of mid-loop operation, including:

- Plant-specific shutdown PSA (resulted in additional procedures and accident management measures)
- Plant-specific simulator (allows for specific training for the shift personnel management)
- Systematic evaluations of internal and external events (led to improvement in the diverse technical areas involved, such as operator training, procedures, level indication, etc.)
- Licensing of shift operators by examination, with regulatory oversight.

The plant-specific requirements are:

- For Beznau (a Westinghouse two-loop PWR):

  Level Measurements. There was a regulatory requirement for installation of an ultrasonic water level instrument for use in mid-loop operation. As a result, there are three diverse level instruments available at mid-loop:

  – Ultrasonic.
  – Differential pressure.
  – Local level gage, with indication in control room via video camera system.

  Information Display. There is also a new, digital, information system available to the operator regarding mid-loop conditions, including an optimal control path, and alarms when in off-normal situations.

  Procedures. Further, there are new and modified procedures to cope with difficult, off-normal situations.

  Piping systems were modified to permit recirculation of water from the containment sump, if mid-loop operation fails.

  Training. Enhanced simulator training is required prior to the reactor shutdown period, for the shifts to be involved in mid-loop operation.

- For Gösgen (a three-loop KWU reactor):

  Level instruments: The level of the hot-leg pipes, including the pressuriser, is measured by a differential pressure, with indication in the control room. There is also a local level device (which has to be installed during shutdown, prior to pressuriser level decrease), with indication, via video camera, in the control room. This level device also has a bistable that can close the primary discharge valve in the line from the RHR system to the volume control system. [The direct level measurement of the hot-leg pipes which was installed after the TMI-2 accident is foreseen to be removed due to the installation of these additional control devices).

Information Display: The regulatory body requested the installation of a modern digital display system to support the control of operations during normal and emergency conditions. A screen with specific parameters important to mid-loop operation allows a complete monitoring and control of RHR operation by shift personnel during mid-loop.

Simulator training. The plant operational crews received additional plant-specific simulator training as described for Beznau.

Procedures. Increased and improved procedures have been adopted for the potential situation of loss of all three RHR trains.

The requirements of the regulator (HSK) were influenced in part by its own events and also by events in other countries, as reported to IRS. It is believed by HSK that at present there are no open issues regarding mid-loop operation, and that no further regulatory requirements are foreseen.

4.8 United States

4.8.1 Events

The United States has a long history of events related to loss of decay heat removal while at mid-loop, actual loss or near-misses, starting from 1980 and continuing in various ways to 2005. These events are documented in several ways:

1. The event may be reported pursuant to the NRC reporting requirements, in conformance with NRC regulation 10 Code of Federal Regulations (CFR) 50.72 and 50.73 (the Licensee Event Report, or LER)

2. The NRC may choose to document the event in a report to the IRS

3. The NRC may issue a public report in the form of a bulletin, information notice, circular, or generic letter

4. The NRC may document a number of events in a generic operating experience report

5. The USA Institute of Nuclear Plant Operators (INPO) may document the event, or collection of events, in operating experience reports

The same plant event may be described in several of the five methods described above. Appendix E provides more detail on the various reports of mid-loop events in the USA.

Although there were many reported events over the past 25 years, the significant aspects were more or less repeated, and are listed below:

- Water level control was improperly maintained, frequently due to an inaccurate monitoring system.

- Due to the rapid pace of the refuelling and maintenance operations, frequently several risk-important activities were being conducted concurrently. This could include maintenance or inspection of one safety train (such as a diesel generator or RHR pump) and thus leave the plant without needed redundancy (for two-train systems). Also, it is possible that both the primary system and the reactor containment are open during mid-loop operations, and this is significant in the discussion of risk-importance.

- Mid-loop operations are not automated and thus subject to human errors.

- There are, in general, no automatic recovery systems such as a pumping system to restore inventory.

- Mid-loop operations may occur early in the shutdown cycle, when the time-to-boil (given an interruption in decay heat removal) can be only 15 minutes, or even less.

While not every event shared all of these aspects, the corrective actions should nonetheless address all of them. For this reason, it is not necessary to provide a detailed discussion of each event here; for further detail, refer to Appendix D.

4.8.2 Corrective Actions

The NRC did not issue any suggested corrective actions until 1988, in the form of its Generic Letter 88-17. Eight prior issuances during the period 1980-1988 had informed the regulated industry about event occurrence, but no specific action was requested. The format of Generic Letter 88-17 was to indicate the need to improve water level control and instrumentation, recovery procedures, control of containment integrity, and operator training. A generic letter cannot by itself be considered mandatory, or a change to the licensing basis. However, a utility can make a commitment to institute changes along the lines recommended in the generic letter, and then the NRC can inspect to assure that commitments are indeed being met.

For the next ten years after the issuance of the GL 88-17 mid-loop events continued to occur and the NRC considered a more binding corrective action in the form of a rule. A staff position paper[2], SECY-97-168 [10], was issued to solicit permission from the Commission of the NRC to issue a proposed rule that would cover shutdown activities. The basis for this proposed rule included the following:

- The proposed rule would provide a substantial increase in the overall protection to public health and safety, and

- The costs of the proposed rule are justified in view of the additional protection, and

- The practical effect of the rule implementation is not to raise the current level of safety being achieved through voluntary actions by all licensees, but

- The action is considered necessary to ensure a regulatory floor for all licensees and to preclude a withdrawal from current practices in light of continuing economic pressure to increase plant availability through shortened outages. This reference noted that a significant safety benefit relies upon measures for which a clear legal requirement does not exist.

[2] SECY 97-168 is a re-issue of SECY-94-176, and takes into account public comments received on SECY-94-176.

However, the Commission decided not to issue the proposed rule (as documented in the US Federal Register of Feb 4, 1999, volume 64, number 23) and rather rely on "voluntary" industry initiatives. These initiatives are documented in NUMARC-91-06, "Guidelines for Industry Actions to Assess Shutdown Management". The Commission decided to continue to monitor industry performance and possibly take further action if any adverse trends are identified. The NRC does have a policy of inspecting refuelling and other outage activities, and determining whether the licensee is adhering to commitments made in response to Generic Letter 88-17. This represents the current situation.

During the past 5 years (2000-2004), several events involving drain-down were terminated prior to reaching the elevation of the hot leg or loss of DHR capability. As might be expected, errors occur during transitions from one operating state to another, or during an activity that affects the flow paths or instruments used for inventory control. Some of these events could be described as near-miss events. A recent event at the Waterford PWR represented yet another example of a precursor to loss of shutdown cooling while in the mid-loop mode. The operating organization had neglected to provide a vent path and created a vacuum in the primary system during draindown that resulted in cavitation of a shutdown cooling pump. This is the second time for this sort of event at this plant. However, since a number of safety injection pumps were considered operable, the perceived risk was low, and this is a key factor in taking more strident action. As of mid-2005 the NRC concluded that the issue is not considered closed.

5. ANALYSIS AND CONCLUSIONS

5.1 Events

This report documents the practices of six countries that have reported instances of difficulties with loss of shutdown cooling while at mid-loop conditions. They are:

- Belgium
- France
- South Korea
- Spain
- Switzerland
- United States

The frequency of mid-loop events, averaged over the past 25 years, is on the order of two events per year, although the frequency over the past five years of the observation period (2000-2004) is less than that of the first five years (1980-1984).

5.2 Risk

There seems to be general agreement on the fact that shutdown risk can be important, in some cases even higher than that of the power operation, and that the risk associated with mid-loop conditions is a dominant contributor to shutdown risk for reactors experiencing mid-loop. This judgment was reached by a number of member states more than ten years ago.

For example The US NRC concluded, in a study on its Significant Determination Process, that the core damage frequency for a PWR whose level of protection was provided only strictly by legally enforceable requirements for mid-loop was on the order of 1E-2. In comparison, if the plant is operated in conformance with the voluntary aspects of GL-88-17 and the NUMARC 91-06 guidelines, then the core damage frequency is on the order of 1E-4 to 1E-6. The high value of 1E-2 would not be considered acceptable. Thus, it is seen that the existing level of safety is largely dependent on measures that are not traceable to specific underlying regulations, and that could, therefore, be withdrawn by licensees without prior staff approval.

5.3 Corrective Actions Alternatives

When events of a similar nature continue to recur for years, or even decades, it is appropriate to explore a range of possibilities, such as delineated in [12]:

- The operating organization failed to take timely actions, or was not aware of the previous events, or thought it was not applicable to the unit in question.

- The regulatory authority was not aware of the events, or had not imposed on the licensee the duty to take timely corrective actions
- Work on the appropriate corrective actions was in progress, but not fully implemented
- The event was considered to be of lesser importance and risk than other plant modifications, and thus was not being pursued as rapidly as needed
- Overall, the operating experience feedback programme was not fully effective
- The root cause of the event had not been correctly identified, and thus the corrective actions were not responsive
- The contributing factors or causes ere not appropriately taken into account in identifying the corrective actions
- What was thought to be a solution was, in fact, not one, or else the problem was generic, and what was a fix for one aspect did not cover all aspects.

These eight possibilities were reviewed in light of the more than 50 events related to loss of RHR while at mid-loop. It is believed, based on the large number of public notices that both the operating organization and the regulators knew about these events. Thus, the first two possibilities may be ruled out. Taken as a whole, it is likely that the main reasons for the long duration of these recurring events lies in the belief that operator actions, coupled with minimal equipment and procedures, suffice keep the plant within the boundaries of safe operation, and thus no mandatory regulatory actions are needed.

Where member states took firm and binding regulatory action, with a broad range of compensatory features, there were no further events

5.4 Member State Responses

Most countries have elected to issue binding regulatory requirements for corrective actions. This includes France, South Korea, Spain, and Switzerland. Notably the US has decided to rely more on industry actions, and this merits further discussion.

In SECY-00-0116 [11], the NRC staff discussed industry initiatives in the regulatory process. Four of the presumptions of the use of industry initiatives in the regulatory process are:

1. The use of industry initiatives in the regulatory process can provide effective and efficient use of resources and resolution of issues;
2. Industry initiatives will be controlled and monitored by the staff so as to provide reasonable assurance that the health and safety of the public will be maintained;
3. The industry initiative will be conducted in compliance with the Commission's regulations[3], as applicable, and
4. The industry initiative will not be inimical to the common defence and security.

[3] The use of industry initiatives may differ somewhat from the viewpoint of the Commission paper issued three years previously [10] to the point that a significant level of safety is dependent on measures that are not traceable to specific underlying regulations.

On the basis of the experience gained from "voluntary" action in the USA, in particular in response to the guidelines expressed in NUMARC 91-06, in contrast to the more forceful regulatory actions of France, South Korea, Spain, and Switzerland, it appears that closure is more readily achieved by mandatory action by the regulatory authority.

The lack of closure that sometimes accompanies "voluntary" actions is illustrated by a comparative examination of the following three generic issuances from the US:

- Degradation of RHR System. Information Notice 81-09; March 1981.
- "Loss of shutdown cooling capability is a potentially significant contributor to the total risk associated with nuclear power. It is expected that licensees will review the circumstances of this event and take appropriate actions to assure shutdown cooing system operability."
- Recent Events Involving Reactor Coolant System Inventory Control during Shutdown. Information Notice 97-83. December 1997.
- "The USNRC is issuing this information notice to alert addressees to two events involving inadequate control of reactor coolant system inventory. It is expected that recipients will review this information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems."
- Outage Planning and Scheduling-Impacts on Risk. Information Notice 2005-16. June 20, 2005.
- "The USNRC is issuing this information to inform addressees about recent experiences in which outage planning and scheduling and adverse human performance for PWRs and BWRs have had a significant impact on shutdown risk. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems."

These three notices, spaced 24 years apart, use somewhat similar language regarding expectations with respect to "appropriate actions" in response to very similar events.

There was another USA event in April 2005 which resulted in cavitation of the shutdown cooling pump. This event, and other recent events that took place in the post-2000 time period, was the subject of IN 2005-16 mentioned earlier. In two of the events discussed in IN 2005-16 there was loss of shutdown heat removal while at mid-loop conditions or in the process of going to mid-loop. However, these events were caused by interruption of heat removal by failures in the secondary cooling water (e.g., component cooling water). These types of failures, although important, were not the focus of the current study.

Clearly the voluntary nature of the corrective actions is less than effective in terms of closing this issue. Actions of a mandatory nature by the regulatory body, such as rulemaking or plant-specific orders are examples of regulatory action.

APPENDIX A: REFERENCES

1. Recurring Events. NEA/CSNI/R(99)19. September 1999.

2. Recurring Events, Volume 2. NEA/CSNI/R(03)XX. December 2003.[4]

3. Proceedings of the Workshop "How to Prevent Recurring Events More Effectively". NEA/CSNI/R(2002)25. 6-8 March 2002; Boettstein, Switzerland

4. Development of a Low-Power and Shutdown Risk Assessment Program. USA Advisory Committee on Reactor Safeguards Letter. June 11, 1999.

5. Risk-Based Analysis of Operational Events for Swiss Nuclear Power Plants. Beutler, G. et al. March 28, 2001.

6. Safety of Reactor Low-Power and Shutdown States. Eurosafe Tribune #002. October 2002.

7. A Compendium of Practices on Safety Improvements in Low-power and Shutdown Operating Modes. NEA/CSNI/R (97)17. March 1998.

8. Region IV Refuelling Outage Risk—An Operational Perspective. NRC Letter, Region IV to Licensees August 29, 2000

9. Proposed Staff Plan for Low Power and Shutdown Risk Analysis to Support Risk-Informed Regulatory Decision Making. NRC Commission Paper SECY-00-0007.January 12, 2000.

10. Issuance for Public Comment of Proposed Rulemaking for Shutdown and Fuel Storage Pool Operation. NRC Commission Paper SECY-97-168. July 40, 1997

11. Industry Initiatives in the Regulatory Process. NRC Commission Paper SECY-00-0116. May 30, 2000

12. Recurring Events. CSNI Technical Opinion Paper No. 3. NRA #04388, ISBN: 92-64-02155-8. December 31, 2003.

[4] See CSNI Technical Opinion Papers-No. 3. December 31, 2003. Recurring Events. NEA #04388, ISBN: 92-64-02155-8 for a summary discussion of [1] and [2].

APPENDIX B: SCOPE OF CORRECTIVE ACTIONS REPORT

Title	Update of Recurring Event Report
Objectives and expected products	Explore methods for analyzing recurrent events. Make the definitions of recurrent events clear. Review the experienced recurring events. Provide input for updating the report on Recurring Events.
Scope /Justification	Definitions of recurring events. Methodologies for analyzing a set of events. Single events could be of low safety significance but together they show something else. Root cause analysis for recurring events. Corrective programmes (this will be investigated in 2004). Elements of good practices, ownership of problem Communication
Safety significance, use and users of the results	Recurring events can be of great safety significance. They remove resources from other safety issues. Something in the feed back loop is not closed. This is of great interest to Regulatory Bodies as well as utilities. WANO will cosponsor this activity.
Schedule and milestones	Call for papers and invitations March 2001. Workshop took place in March 2002. The report has been made available in 2003 and A Technical Opinion Paper is in NEA publication process. The WGOE has, in accordance with the CSNI guidance to prioritise its activities, decided to continue by investigating corrective actions against recurrence (including if they were mandatory/voluntary, link to licensing basis, if they were prescriptive/performance based, how communicated to others, did they solve the problems,) in order to close the feedback loop. The group plans to issue a report about corrective actions with regard to loss of RHR in outage conditions including the mid-loop by the end of 2004.
Lead organization(s)	WANO and NEA cosponsor
Participants	Persons involved in operating experience, utility people, regulatory bodies
Financing (if relevant)	
Requested action	CSNI approved in Dec 2003

APPENDIX C: QUESTIONNAIRE FOR LOSS OF DECAY HEAT REMOVAL WHILE AT REDUCED LEVEL CONDITIONS

1. How many events have taken place in your country? Do you know some events that could have lead to similar loss of decay heat removal conditions and that occurred despite the corrective actions that have been taken (near misses)?

2. What are the references that document these events/near misses?

3. Based on these reports and occurrences, what modifications in design and operation have taken place?

4. How did you (or the Licensee) decide what modifications needed to be done?

5. Did the facility licensing basis get changed?

6. Were new regulations issued, or were regulatory orders issued?

7. How was the schedule for the modifications developed, and who agreed to the schedule?

8. Do you regard the modifications as being general in nature~~,~~ or did the regulatory authority issue prescriptive requirements?

9. To what extent were the modifications influenced by activities in another country or group of countries?

10. Overall, what is your opinion of the success that you have had in closing this issue, in terms of a timely solution that is technically supportable?

11. In consideration of the actions taken to date and the experience gained in other countries, do you think the issue will remain solved in your country?

APPENDIX D: ACRONYMS AND ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AIRS	Advanced Incident Reporting System (sometimes referred to simply as IRS)
CNRA	Committee on Nuclear Regulatory Activities
CSNI	Committee on Safety of Nuclear Installations
INPO	Institute for Nuclear Power Operations
LOCA	Loss of Coolant Accident
RHR	Residual Heat Removal system; sometimes known as DHR, decay heat removal system
WGOE	Working Group on Operating Experience

APPENDIX E: MID-LOOP EVENTS IN THE USA

E-1. Issuances by the NRC

Date	Issuance	Substance
1980	IN 80-20	Discusses DHR loss at Davis Besse; due to electrical faults, RHR pumps were aligned to containment sump, which was empty. Thus there was no RHR function for about 2 ½ hours.
1980	Bulletin 80-12	Noted several losses of RHR, including the Davis Besse event above. Identified extensive maintenance activities, and inadequate procedures, as major contributors.
1980	Generic Letter	Asks that technical specifications by amended with respect to decay heat removal. Need to assure redundancy in all modes. Model tech specs included with letter.
1981	IN 81-09	Discusses event at one plant (Beaver Valley) where RHR flow was severely degraded for about an hour while the water level was near the hot leg midpoint. Problems with level measurement. Air entrained to RHR pumps.
1981	Circular 81-11	Summarized several instances of loss of decay heat removal. Indicated need for BWRs to provide additional controls. Not directly related to the loss of suction scenario for PWRs.
1986	IN 86-101	Noted loss of decay heat removal due to loss of fluid levels in reactor coolant system. Advised licensees of continuing problems during PWR outages with procedures and instrumentation where water level is lowered. Also noted industry reports of 12 PWR events between 1977 and 1981. In addition, there were six events in 1984. Noted deficiencies related to operator knowledge; operating procedures; and, water level instrumentation. IN summarized four events for illustration.
1987	IN 87-23	Provided detailed summary of event at Diablo Canyon 2 which occurred in April 1987. No decay heat removal for 1 ½ hours, during which vessel water increased from about 31 deg C to 100 deg C, with steaming from open primary system. Causes included erroneous level measurement, inadequate knowledge of cavitation, poor work coordination, and inadequate training and procedures. Referenced NRC Case Study AEOD/C503 which depicted 23 events since 1981 that involved loss of pump suction while at reduced water level.
1987	Generic Letter 87-12	Letter summarized concern over events of past few years and requested that licensee provide information on how plant will be operated during reduced water level scenarios such that the licensing basis is maintained. This includes procedures, instrumentation, training, containment status, equipment needs, and additional resources to the operators. More information was provided in NUREG-1269 (which was a detailed study by NRC of the Diablo Canyon event mentioned above).

1988	Generic Letter 88-17	This letter notes continuing difficulties with response to GL 87-12, as well as additional events that occurred at Sequoyah and Waterford since the issuance of that GL. Noted deficiencies in procedures, hardware, and training. Stated that expeditious action was needed.
1989	IN 89-67	Notified licensees about injection of nitrogen from accumulators which could affect decay heat removal. Cited event at Salem in May 1989 where, due to operator error, nitrogen was injected for about one minute, and sent the pressuriser full. The operator mistakenly drained liquid from system and this resulted in total loss of RHR flow. Primary system heated from 92 deg F to 122 deg F.
1990	IN 90-06	This IN covers operability of RHR while at mid-loop conditions. At Comanche Peak-1, one train of RHR was operating at a flow rate of 2,000 gpm. An air-operated flow control valve failed to the fully open position, and the flow rate increased to 4,400 gpm. This resulted in a vortex potential at the pump suction line to the hot leg.
1992	IN 92-16, and supplements 1 and 2	The topic is loss of flow from RHR during refuelling cavity draindown. One event was at Vogtle-1 in October 1991. While at mid-loop conditions, a new sight glass was being used for level indication. However, it had not been aligned or tested for use. As a result the level was reduced too far, and a vortex formed and the RHR pump cavitated. A 20 deg F heatup occurred. In Supplement 1, an event at Prairie Island-1 was discussed. There, while at mid-loop, there was conflict with some level measurement devices. A vortex formed, as indicated by pump current and flow meters. RHR was lost for 21 minutes. In supplement 2, conflicts with level measurement at three more PWRs were summarized.
1993	IN 93-72	The NRC informed the industry about observations from shutdown risk and outage management pilot team inspections. The NRC conducted pilot team inspections at five plants on initiatives to improve shutdown safety. This IN noted that the core damage frequency for shutdown operations may be a substantial fraction of the total cdf. Two areas of concern were listed: i) risk assessment for pre-outage planning, emergent work and schedule changes, and ii) implementation of defense-in-depth for equipment availability.
1996	IN 96-37	This notice provided information on an inadvertent drain down at Surry 1 due to the expansion of a gas bubble trapped in the reactor vessel head.
1996	IN 96-65	This information notice related to the effect that undetected accumulation of nitrogen gas might have on water level indicators during shutdown activities. It referenced the Surry 1 event just above. A related event occurred at Haddam Neck. Nitrogen gas inadvertently flowed from the volume control tank through several leaky valves into the reactor vessel. This caused the vessel level to decrease. This gas influx has the potential to interfere with RHR to the point of common mode failure of all the pumps.
1997	IN 97-83	Two plants, Sequoyah 1 and Zion 2, experienced a draindown with inadequate level indication. In both cases RHR continued to function, but it was necessary to add fluid to the reactor coolant system to compensate for the inadequate level information.
1999	IN 99-14	This information notice discussed unanticipated reactor water draindown events at Arkansas Unit 2. During a draindown procedure the level dropped about 25 cm lower than desired. The rate of level decrease was about 80 cm per minute. The event was terminated before the level reached the point of a vortex or pump cavitation. Some causative factors included an inexperienced crew; need for more pre-briefing; need for additional operator in visual contact with the level instrumentation, and need for more training.

2000	IN 2000-13	This IN discusses a review of refuelling outage risks. Of the PWR outages that used midloop configuration, 9 of 15 did so with concurrent unavailability of either an emergency DG or with performance of significant switchyard and DG maintenance. Contingencies and other strict controls were adopted.

E -2. NRC Reports to the IRS

IRS Number	Event Date	Plant	Summary
0030	April 1980	Davis Besse	This is the event described in IN 80-20, in Table 1. This IRS describes other events at this plant of a similar nature. Stated root causes include extensive maintenance activities and inadequate procedures and administrative controls. Long-term corrective actions were taken pursuant to NRC Bulletin 80-12 (see Table 1).
238	March 1981	Beaver Valley 1	The reactor was shut down and at reduced inventory. There was a low flow alarm on RHR and low pump current. Normal flow was recovered in about one hour. The direct cause was a faulty water level sensing instrument. The only corrective action mentioned was to monitor level by a redundant instrument.
492	August 1984	Arkansas One	This event commenced about 36 hours after shutdown. The plant was in cold shutdown in anticipation of a reactor coolant pump seal replacement. The system was at 140 deg F. The RCS level was about 5 cm above the hot leg.
708	March 1986	San Onofre 1	There was a loss of shutdown cooling due to improperly functioning water level indicators. A vortex formed at the suction inlet (hot leg) and flow was lost for 49 minutes. An immediate cause was that a non-conservative error in the level indicator existed. As a result, the operators were not able to diagnose that the system was at mid-loop conditions. Corrective actions included changes in design, procedures, and configuration control. Training was improved.
918	1988 (Generic)	Multiple plants	In several plants there was a loss of decay heat removal capability due to rapid refuelling cavity pump-down. The plants included Byron 2, Diablo Canyon 2 and Wolf Creek. Even though there was water above the vessel flange, there is the potential for a vortex and air binding of the RHR pumps. The difficulty is that there was a restriction the flow path through the internals.
1088	March 1990	Vogtle 1	There was a risk-significant event at Vogtle involving a loss of all AC to the safety buses while the system was at mid-loop, the primary system was open, and the containment hatch was open. With no AC, decay heat removal was lost for about 36 minutes. The primary water heated up from 90 deg F to 136 deg F. Some direct causes were: a switchyard event interrupted off-site power; one (of two) a diesel was out for maintenance; and, the other diesel failed to start and run.

1096	July 1989	Comanche Peak 1	Although there was not an actual loss of flow in the RHR system, there was a potential for vortexing following a failure in the air supply which causes a flow control valve to fail wide open, and an increase in flow from 2000 gpm to 4400 gpm. At the time the water level was slightly above the hot leg. Corrective actions included manual positioning of the injection valves to limit the maximum flow rate.
1242	May 1991	Waterford	There was a loss of shutdown cooling during a disassembly of a safety injection system check valve. When the valve was opened, air entered the suction piping and cavitated the pump. Shutdown cooling was lost for 19 minutes. [The NRC issued In 01-73 as a result.] The containment hatch was open, but was closed within one-half hour. The coolant system heatup rate was about ½ deg/minute. Corrective actions included review, evaluation, and possible alterations to work controls and scheduling policies.
1340	October 1991	Vogtle 1	During reactor system drain-down, there was a loss of flow from the RHR system. The plant was in refuelling mode, with the head removed. One RHR pump was removing decay heat, and the other was draining the refuelling cavity. The level indicator was not aligned correctly. Oscillations occurred in the shutdown heat removal pump current, indicating a vortex formation. Flow rate was decreased, and water was added to the refuelling cavity. One action was that NRC issued IN 92-16. Following a similar event at Prairie Island, NRC issued Supplement 1 to this information notice.
7027	March 1991	Oconee 3	This IRS report is a summary of a review of human factors aspects of an event involving loss of shutdown cooling for 17 minutes. The water level had been drained to the top of the vessel in preparation for reinstallation of the head. One low pressure pump was removing decay heat. A valve was mistakenly opened and this resulted in a loss of water level occurred. Corrective actions included revision of procedures, training, and communication.
7189	August 1996	Haddam Neck	Undetected accumulation of gas was present in the reactor coolant system. This event was discussed in IN96-65 [Table 1].
7321	May 1999	Generic; Multiple Plants	There was an unanticipated reactor water draindown at Arkansas 2. [This was discussed in IN 99-14; see Table 1.]
7412	September 2000	Generic; Multiple Plants	This IRS reports provides a review of refuelling outage risks 9 (see IN 2000-13; see Table 1.)

E-3. Reports by INPO

Date	Plant
April 1980	Davis Besse
March 1981	Beaver Valley 1
June 1981	Trojan
April 1983	McGuire q
May 1983	North Anna 1
May 1983	Surry 1
?	Calvert Cliffs 1
Mat 1984	Trojan
August 1984	Arkansas 2
December 1983 / January 1984	McGuire 2
September 1984	Zion 2
June 1983	Beaver Valley 1
October 1982	Summer
May 1984	Cook 2
October 1985	Sequoyah
March 1986	San Onofre 2
February 1985	Crystal River
July 1986	Waterford
April 1987	Diablo Canyon 2
September 1988	Byron 1
June 1988	Cook 2
September 1988	Oconee 3