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Organisation de Coopération et de Développement Economiques
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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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RECURRING EVENTS

September 1999

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter; Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consist of all OECD Member countries, except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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Part 1 - Recurring Events

INTRODUCTION

A nuclear power plant is designed for a spectrum of incidents and accidents, ranging from a reactor trip without other complications to more serious events such as pipe ruptures. Certain portions of the plant are designed for even more significant events such as severe accidents. Several thousand reactor-years of experience have been recorded and many postulated events have in fact occurred. In some instances the same or similar event has occurred more than once within a single country or among several nations. Such cases are referred to as recurring events.

One way to reduce the likelihood, or severity (or both) of recurrence is to maintain and utilize a system for reporting of events, both at the national and the international levels. The international system is referred as the Incident Reporting System. Events to be reported to IRS include:

- The event itself is serious or important in terms of safety due to an actual or potential reduction in the plant's defense in depth;
- The event reveals important lessons learned that will help the international community to prevent its recurrence as a safety significant event under aggravated conditions or to avoid the occurrence of a serious or important event in terms of safety;
- The event is a repetition of a similar event previously reported to IRS, but highlights new important lessons learned for the international community.

National systems for reporting of events vary in scope; there is guidance on systems for feedback of experience from events in nuclear power plants. Further, the Nuclear Safety Convention, Article 19 – Operation – provides (section vii) that each Contracting Party shall take the appropriate steps to ensure that "programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies".

Recurring events may indicate that the operating experience system may need reinforcement in terms of actions taken, in that more aggressive and timely actions taken can reduce or eliminate the recurrences.

Four categories of recurring events were selected, somewhat randomly, to cover a range of plant performance, from general reactor behavior, to components, trains, and systems. Other examples also exist (such as BWR shroud cracking, thermal fatigue, control rod insertion problems) and could be used for future studies. The September 1999 meeting of PWG1[1] resulted in a decision to have an in-depth study of recurring events at the year 2000 meeting.

[1] This summary on Recurring Events is based on a longer report of the same title presented at the PWG1 meeting.

RECURRING EVENT DESCRIPTIONS

One notable recurring event is associated with the core meltdown at Three Mile Island, which occurred in March 1979. About 1 ½ years earlier, in September 1977, an event occurred at a similar plant, the Davis Besse plant in Ohio, USA. The common factors were loss of feedwater, opening of a relief valve on the pressurizer (and failing open), and operator intervention of the high-pressure injection system. The event occurred at about 10% power; had the reactor been at full power the consequences might have been much more severe. As it was, there was no fuel failure. This event was poorly reported and the potential safety significance was all but ignored. Other precursors to TMI exist, but also were not studied in detail and the lessons were not transmitted.

The national and international incident reporting systems have examples, much less dramatic, of recurring events. Some examples follow below.

LOSS OF RHR COOLING WHILE AT MIDLOOP IN A PWR

For various reasons during maintenance and refueling operation, a PWR may have the primary system water level reduced, to the point where the hot leg is only partly full. During such operation the decay heat removal path is through the RHR system, with heated water from the core flowing to the partly-full hot leg and thence down to the RHR pumps, heat exchangers, and return path to the cold leg. At such times a small error in accomplishing the desired water level can lead to drainage below the hot leg, and interruption of the heat transport to the RHR heat exchanger system. A partial list of such untoward events is shown in Table 1.

Table 1. Loss of RHR Cooling at Mid-loop.

DATE	PLANT NAME	**REPORT NUMBER**
April 1980	Davis Besse	**NRC Bulletin 80-12**
March 1981	**Beaver Valley**	**NRC IN 81-09**
April 1985	**Catawba**	**NRC IN 86-101**
June 1985	**Blayais-4**	**IRS 659**
October 1985	**Sequoyah-1**	**NRC IN 86-101**
December 1985	**Zion-2**	**NRC IN 86-101**
March 1986	**San Onofre 2**	**NRC IN 86-101 IRS 708**
April 1987	Diablo Canyon 2	**NRC IN 87-23 & NRC GL 87-12**
May 1989	Salem 1	**NRC IN 89-67**
July 1989	**Comanche Peak 1**	**IRS 1096**
May 1991	**Belleville 2**	**IRS 1197**
October 1991	**Vogtle 1**	**IRS 1340; 1991 ASP Report**
February 1992	**Prairie Island 2**	**1992 ASP Report**
October 1994	**Doel 1**	**IRS 1628**
August 1996	**Blayais 3**	**IRS 7208**

Some common deficiencies in the events in Table 1 included:

- Lack of knowledge between water level and RHR pump speed
- Operator procedures that did not adequately consider vortexing at the connection of the RHR line to the hot leg;
- Reactor water-level instruments that were erratic; inaccurate; did not have adequate range; were not checked out before use; or were not monitored as frequently as necessary;
- Operator training was deficient;
- Recovery procedures were not emphasized;
- The pressure to reduce outage duration resulted in multiple tasks with inherent conflicts.

OPERATIONAL FAILURE OF VALVES

Operational failure of valves due to the phenomena known as pressure locking or thermal binding has been reported at various places in the literature since 1964. Certain valves in safety systems may fail to operate when high-pressure fluids in the valve bonnet inhibit motion of the valve disc. Thermal binding may also inhibit disc motion. The IRS literature has at least 15 such failure reports over the last 20 years of the IRS. In spite of the reports, and discussions in various regular and special meetings of national and international bodies, the failures continue to be reported, even though the solution is well known.

SERVICE WATER AND ULTIMATE HEAT SINK PROBLEMS

A third example for consideration is in the service water and ultimate heat sink systems. Over the past 20 years there have been a number of total or partial failures in these systems. In particular there have been buildups of marine life that interfered with these safety systems.

Generally these failures were relatively benign, leading in some cases to a reactor trip. Of significance is that the causative factor is well known yet continues to occur from time to time.

The systems included the reactor building cooler coils; component cooling system; service water system; various heat exchangers; steam generator flow control valves; condenser; and circulating water screen. Marine life included Asian clams, barnacles, bivalve mollusks, shrimp, and other marine growth.

BWR INSTABILITY

There have been a number of reports on power oscillations and instability in boiling water reactors over the history of IRS. At least nine such events, from six member states, are in the database. NEA has sponsored a workshop on this subject in 1990. To date there have not been serious consequences. Several factors have contributed to these events, including skewed power shape; operation with mixed cores of fuel element designs, colder inlet feedwater, and, inadequate consideration of the proper operating envelope.

RISK IMPLICATIONS

Risk implications of recurring events has been made in some cases. Various studies on risk of mid-loop operation show that both risk at shutdown in general, and in particular risk due to reduced primary system inventory, can be an appreciable portion of the total plant risk. Risk quantification of events involving pressure locking and thermal binding of valves has not been done in general. If there were a common mode failure present at the system level in a vital mitigation system for a postulated LOCA, then the risk could approach the likelihood of the LOCA itself (i.e., risk-significant). One estimate of the core damage frequency attributable of loss of service water was reported in one event report as in the range of 10^{-3} to 10^{-5}. Blockage due to marine life would be a subset of the service water failure modes. The risk of BWR instability is generally thought to be low, on the empirical basis of operations to date.

POSSIBLE FACTORS CONTRIBUTING TO RECURRENCE.

Some possible explanations for event recurrence are:

- The affected utility (where the recurring event took place) failed to take timely action, or was not aware of the previous event, or believed that it was not vulnerable, or had plans for remedial action but had not yet implemented them, or was willing to assume the risk of recurrence for various reasons (such as intent to terminate operations in the near future);
- The cognizant regulatory authority was not aware of previous events, or was aware but had not taken prompt remedial regulatory actions, or had taken action but on more leisurely time scale (regulatory bodies frequently specify an acceptable time interval for incorporation of needed modifications);
- Central utility groups and allied research and development groups were working on a unified solution but had not yet reached agreement on the solution at the time of the recurrence;
- National and international groups responsible for collection and analysis of operating experience had not collected or analyzed the information from previous events, and thus had not disseminated proper warnings to the operating organizations.

More detailed study would be needed to associate a precise reason, in the above examples, as to what factors contributed to recurrence. Since recurrences took place in several member states, it would not be possible to ascribe specific reasons for recurrences; however, the four possibilities above might serve as a starting point.

It would be instructive to reach a more precise understanding as to the reasons for decisions to accept or reject changes in plant design and operation, and, where changes are accepted, decisions regarding schedules. It is known that some changes, for example those attendant to the strainer blockage issue attributable to a transient at the Barseback station, require considerable design, engineering, procurement of parts, and scheduling. It can take several years to implement a solution, once one is determined. Recurrence can take place during that time, which focuses additional attention on such aspects as the basis for continued operation and interim precautions involving operator attention and training.

Although direct causes for recurrence of an event may be known and described in event reports, it is much less likely that the programmatic factors, which lead to recurrence, are known. Some speculation is provided above, but more work would be needed to attain specificity. More likely, it is a combination of all of the cited factors.

It is generally agreed that the recurring problems have risk significance, in some cases. It is also true that in some countries the time scale for achievement of problem resolution is tempered by the perceived risk significance. There is no generally accepted international use of risk (for example, cdf) in this regard and this may contribute to the perception of recurrence.

Frequently the “fix” to a problem is left to additional knowledge, skill, and ability of the operational staff. However, it is seen that this added burden may be inadvisable, especially for infrequent events. A recent consultant report based on IRS reports expressed a concern over event recurrence perhaps attributable to loss of corporate knowledge.

In some cases the event represents a violation of the design basis; however, it is less than clear that reinforcement of the design basis takes precedence, in some countries, over the perceptions of risk significance. This can contribute to recurrence.

PART 2

Recurring Events

and the Possible Need to

Reinforce Operating Experience

Feedback Programs

September 1999

Denwood Ross

EXECUTIVE SUMMARY

Event recurrence, defined as repetition of an undesirable event or condition at several reactor facilities over a long period of time, is a concern for all of the involved parties. Such recurrence would represent a deficiency in the operating experience feedback programs.

To illustrate the problem, four case studies were selected:

1) Loss of residual heat removal while at mid-loop conditions;
2) Failure to operate of valves through the mechanisms of pressure locking or thermal binding;
3) Service water degradations by biofouling means;
4) BWR power oscillations.

Events were selected mostly from the IRS. However, a separate analysis was made for a group of INPO reports dealing with mid-loop transients in PWRs. In each of the four cases there were failures reported over a number of years, up to twenty years in some cases, where the direct and root causes were both quite similar and well known.

Although more prompt closure in terms of implementing solutions is the responsibility of the utility and the regulatory body, there is a contributory role that the international organization could play in assuring wider dissemination of information, and emphasis of the longer-running problems.

CHAPTER 1. INTRODUCTION

Operating experience feedback systems have existed at the national and international levels for many years. The need for such systems was underscored by the core damage events at TMI-2 and Chernobyl, but a strong supportive case can be made for such feedback systems even without these catastrophes. The national event reporting systems should ensure communication within the country, between and among the parties (i.e., utility, regulator, designer, fabricator, research institutes, and major component suppliers). The international parties (i.e., NEA, IAEA) can facilitate communication between member states. In all cases the primary responsibility for corrective actions lies totally within the national system.

The need for experience feedback was cited again in the Nuclear Safety Convention, Article 19. In particular, conclusions that are drawn from operating experience should be "acted upon".[2]

Within the NEA, there has been a Principal Working Group #1 since 1981. Reporting to the CSNI, PWG#1 is responsible for operating experience and human factors. The mandate of PWG#1 (Hada, 1994) includes an annual examination of the incidents reported during the previous year in order to select issues (either technical or human factors oriented) with major safety significance and to report them to the CSNI. The main goal of this element of the mandate is to identify issues of major safety significance and to discuss corrective actions taken in order to avoid occurrence of an accident.

In a similar vein, the IAEA undertakes an annual review of the incidents of the previous year, and documents this review in a report that is used by the national IRS coordinators.

Within the complex of national and international systems for experience reporting and feedback, the question arises as to frequency and importance of recurring events. Events or conditions of safety significance could be repetitive within a member state, or a number of states, for a variety of reasons, such as:

- Events are not reported well, or timely;
- It takes too long to formulate a corrective action program;
- The event is discounted as not particularly relevant (on grounds of perceived risk importance);
- Due to personnel turnover (loss of corporate knowledge) the utility or regulatory body may have insufficient current knowledge of old events.

Other reasons may exist also.

The IRS was examined to see whether there is concern over recurring events. It was desirable to extract reasons for event recurrence. Four cases were selected to illustrate the existence of recurring events.

[2] " …programmes to collect and analyze operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies …"

These should not be regarded as exhaustive, but only to support the generic conclusions regarding repetition.

Perhaps the ultimate example of event repetition is that of Three Mile Island. In September 1977, at the Davis-Besse NPP in the USA (also made by Babcock and Wilcox) there was a transient[3] that involved:

- Feedwater interruption caused by a spurious signal in the steam generator rupture control system (SFRCS);
- Isolation signal generated by the SFRCS, with isolation of main feedwater, with actuation of auxiliary feedwater;
- Rise in pressure of primary system with opening and sticking of pilot operated relief valve (PORV) in pressurizer;
- Rise in pressurizer level and manual scram due to high pressurizer level;
- Reactor pressure decreases due to flow out of the PORVs; PZR level drops;
- Initial starting of high-pressure makeup (by safety features actuation system, at 1600 psi); HPI secured manually when PZR level returned
- Twenty minutes after the trip the operators diagnosed the stuck-open PORV and isolated the leak via the block valve; system was saturated and boiling for much of this time;
- There was auxiliary feedwater to one steam generator throughout the 20 minutes.
- The first 20 minutes of this event greatly resembled TMI-2, except that this event was 9% power, and not 100% as for TMI-2. Also, there was some heat removal at Davis Besse via the steam generator, although the primary system was saturated.
- The primary system was stabilized at about one hour after incidence, with the system once again subcooled and pressurizer level at normal, and the PORV block valve closed.[4]

Some of these attributes were also present at TMI-2, including stuck-open PORV, loss of feedwater, and saturated primary system.[5]

Clearly an enlightened operating experience feedback program that explored the nuances of Davis Besse could have materially lessened the likelihood that the TMI event would have taken place. In fact, the Davis Besse event was not widely disseminated nor analyzed within the US or in other countries. This is a powerful incentive to report important events; to study repetitive events; and, to learn why they were not averted earlier.

An additional TMI precursor occurred in June 1975 at the Oconee-3 plant, a sister plant to TMI. The reactor was in a power reduction mode and had reached 15%. A feedwater transient occurred and the primary pressure increased to the PORV setpoint. Although the pressure dropped, the PORV did not reclose. The reactor tripped on low pressure and HPI started. The operator isolated the PORV, but reopened it upon high PZR level, and secured HPI. After the primary pressure reached 800 psi, the PORV

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[3] See NUREG 0560, NRC report on feedwater transients in B & W reactors.

[4] A more complete description of this event may be found in the Rogovin investigation report of TMI, Volume II, Part 1, Section C, Precursor events.

[5] There was an earlier event involving the opening of a pressurizer relief valve, and resultant level increase. This illustrated the nonconservative nature of coincidence logic for low pressurizer level-low pressurizer pressure as an initiator for ESF action such as ECCS or reactor trip.

was again isolated. During this interval the primary system was saturated with some boiling. Some of these feature:

- Stuck-open PORV
- Feedwater transient
- Securing of HPI
- Voiding of primary system
- High pressurizer level during a small break LOCA (due to void formation in primary system)

are common to both Davis Besse and TMI-2.

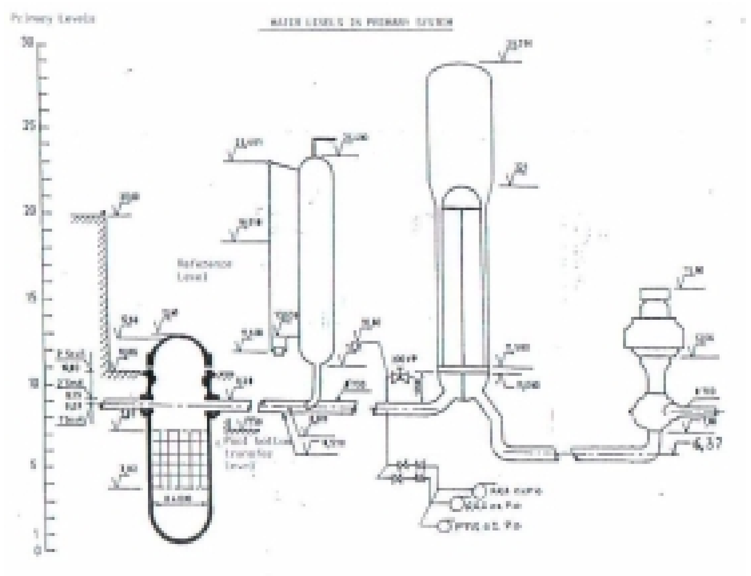
These two precursors would allow TMI to be labeled as a recurring event. Further, there was a feedwater transient at TMI-2 itself, during the unfueled commissioning phase in September 1977. Further complications resulted in interruption to the closed cooling water system, with loss of the reactor coolant pumps and HPI pumps. When the RCS pumps tripped and the PZR heaters tripped, the primary pressure decreased and voiding occurred in the primary system. Steam accumulated in the highest point of the primary system, and, even though pressure was dropping, the PZR level was rising. These anomalies were there to be seen, but were not acted on.

CHAPTER 2. CASE STUDIES

2.1 Loss of RHR While at Mid-Loop

For various reasons a PWR will, during maintenance, reduce the level in the reactor coolant system to what is known as mid-loop. The water level is drained such that the hot leg is less than full. Decay heat removal continues to be supplied by the RHR system. Care must be taken to assure that the RHR system flow rate is not so high as to reduce the NPSH for the pumps to a point where pump cavitation occurs.

In the figure to the right there is an approximate depiction of the relative elevation of key components in the RHR system as it relates to reduced inventory.



Over the past 20 years there have been a number of reported instances where RHR cooling has been lost while the PWR was at mid-loop conditions. Table 1 provides a list of these events.

Table 1.Loss of RHR While at Mid-loop Conditions

DATE	PLANT NAME	REPORT NUMBER
April 1980	Davis Besse	**NRC Bulletin 80-12**
March 1981	**Beaver Valley**	**NRC IN 81-09**
April 1985	**Catawba**	**NRC IN 86-101**
June 1985	**Blayais-4**	**IRS 659**
October 1985	**Sequoyah-1**	**NRC IN 86-101**
December 1985	**Zion-2**	**NRC IN 86-101**
March 1986	**San Onofre 2**	**NRC IN 86-101** **IRS 708**
April 1987	Diablo Canyon 2	**NRC IN 87-23 &** **NRC GL 87-12**
May 1989	Salem 1	**NRC IN 89-67**
July 1989	**Comanche Peak 1**	**IRS 1096**
May 1991	**Belleville 2**	**IRS 1197**
October 1991	**Vogtle 1**	**IRS 1340; 1991** **ASP Report**
February 1992	**Prairie Island 2**	**1992 ASP Report**
October 1994	**Doel 1**	**IRS 1628**
August 1996	**Blayais 3**	**IRS 7208**

In the NUREG-1410 reference, there is an analysis (see Appendix D of that report) of some mid-loop events that took place in the 1985-86 time span. The plants involved were San Onofre 2, Zion 2, Sequoyah 1, and Catawba. Some common deficiencies noted were:

- Lack of knowledge about the correlation between water level and pump speed at the onset of vortexing;
- Operator procedures that did not adequately consider vortexing;
- Reactor vessel water-level instruments that were erratic or inaccurate, did not have adequate range, were not checked adequately before use, or were not monitored as frequently as necessary during use.

Despite the occurrence of these events, there is a continuing history of occurrence of problems while at mid-loop, as briefly discussed below. The eight events below were obtained from a search of the IRS database. There is no reason to assume that the national authority reported all mid-loop events to IRS, so the actual number of events is likely to be higher. This operations history may indicate that either experience feedback is unduly slow, or else not being adequately disseminated or heeded. Some text from several of the IRS reports is provided below:

1. *Blayais-4; June 1985; IRS 659 "At the beginning of the annual outage for refueling and when draining the steam generator tubes, the operators allowed the water level in the primary system to drop to an abnormally low level. Air was sucked into the spent fuel pool cooling and residual heat removal systems leading to loss of prime to two pumps of these systems. The low flow signal appeared followed by amperage drop. Another pump was then started before a complete diagnosis and failed in the same conditions. The RHR function was completely lost. …. Several causes: procedural mistake concerning the required level; took too long to diagnose the problem; inadequate level instruments. RHR lost for 20 minutes, with little heap.*

2. *San Onofre-2; March 1986; IRS 708. "While in cold shutdown the shutdown cooling system experienced a total loss of flow for 49 minutes. Using the established level indications, which was later found to be in error, the RCS was drained to a level when vortexing occurred at the suction connection, causing the pumps to become airbound. There was local boiling in the core during the event. Changes were made in the plant design, procedures, level instruments, and operator training.*

3. *Comanche Peak; July 1989; IRS 1096. This event took place prior to initial fuel loading. While at mid-loop it is necessary to throttle RHR to 2000 gpm, instead of the default value of 4,400 gpm. The reactor was at mid-loop when an inverter failed to the flow control valve and the valve failed open, increasing the RHR flow rate to the default value. Procedures were changed to demand manual throttling of RHR while at mid-loop.*

4. *Belleville-2; May 1991; IRS 1197. "While draining the RCS, five days after shutdown, an uncontrolled drop in the RCS level caused the two trains of RHR to malfunction. There were fluctuations in the amperage indicating vortexing. The causes were attributed to failure to follow procedures; discrepancies between the actual plant and the reference plant design; unsuitable control room instruments (amperage); poor level calibration; and operator errors. Corrective measures corresponding to these causes were taken.*

5. *Vogtle-1; October 1991; IRS 1340. "The event occurred during a draindown of the reactor cavity, following core reload. There were inadequacies in the procedure for mid-loop operations. Vortexing occurred and RHR was lost for 16 minutes. A new sight glass for reactor*

level had been installed, but not tested or aligned. Some valves were mispositioned in this level system.

6. *Prairie Island-2; February 1992; IRS 1340. "The reactor was being taken to mid-loop. There were numerous level systems (direct with tygon tube; two electronic). After several hours of draining, vortexing occurred with low flow signals and fluctuating amperage. Loss of RHR occurred for 21 minutes. Causes included inadequate procedures, nonconservative operator actions, and unanticipated performance of electronic level instruments.*

7. *Doel-1; October 1994; IRS 1628. "Prior to restart the RCS is degassed under vacuum to remove oxygen. During the degassing the RHR was lost. It was attributed to a combination of excessive RHR flow rate combined with low level that contributed to vortexing. The RCS temperature rose about 20 deg F. There were deficiencies in procedures and operator actions. However, the "main root cause" was attributed to the use of model tests that were done by Westinghouse. These tests were done to establish a range of permissible flow rates that would avoid vortexing. However, the tests were done at atmospheric pressure, and thus were not applicable to the partial vacuum conditions that prevail during degassing.*

8. *Blayais-3; August 1996. IRS 7208. "The RCS opening was in progress and the RCS level was above the flange. Draining was necessary to compensate for seal injection. Several failures occurred in the draining operation monitoring and level were not maintained properly. While a repetitive draining was underway, the operator became distracted and forgot about the draining. The RCS level alarmed, indicating that the level was 29 cm below the flange. In this case RHR was not lost. There was concern over the human factors aspect and the management of alarms in the control room. In this case, the alarm was in a control area far removed from the central control room. Many other alarms were indicated in other systems, indicating a need for some sort of alarm filter. Changes are to be made in equipment and procedures, and in training and in task coordination and planning.*

The US Institute for Nuclear Power Operations (INPO) prepared analyses and comments on events and conditions in plants. These are referred to as Significant Event Reports (SER); Operations and Maintenance Reminders (O & MR); Significant Event Notification (SEN); and, Significant Operating Experience Report (SOER). In a report documenting a significant transient at the Vogtle reactor in March 1990 the NRC published a summary of ten years' history of INPO reports with respect to loss of decay heat removal (see NUREG-1410, Appendix E). A brief summary of the INPO work is provided herein as Appendix A. The INPO reports collectively document 22 instances of loss of decay heat removal, usually while at reduced reactor system inventory. Corrective actions are along the lines discussed in this section. Information from INPO was not readily available for the time period after 1990.[6]

Clearly this record documents the concern for recurrence. It is significant that even after the multiple number of INPO reports, and after issuance of guidance from the NRC, events continued to occur in the US. Whether the problem can be considered solved among the members of NEA is problematic, and may need further study.

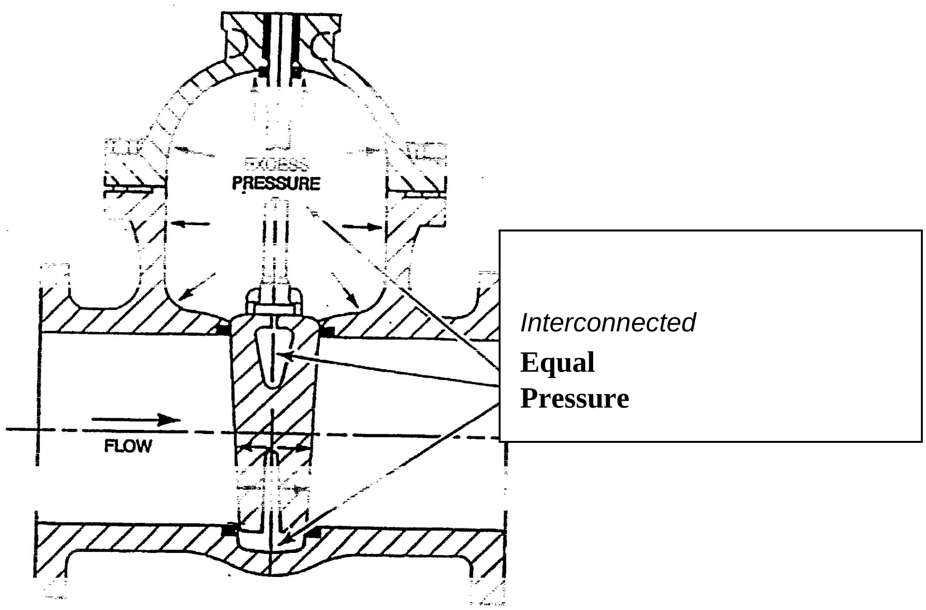
[6] The INPO information also includes reports from Cattenom (May, 1988) and Ringhals (1984).

2.2 Operational Failure of Valves due to Pressure Locking and Thermal Binding

Operating experience has shown for many years that double-disc and flexible-wedge gate valves in many safety applications have not been operable due to pressure locking or thermal binding. These valves, as a result of their design, can become pressure locked by being placed in operating configurations which subject them to high-pressure fluids in the bonnet. Such forces oppose motion of the valve disc from the seat and these forces were not considered when sizing the valve's motor operator. Thermal binding occurs because the flexible-wedge gate body contracts a greater amount during cooldown than the valve disc and pinches the disc in the valve seat.

The potential for these kinds of failure has been well known for many years, and well publicized. In spite of numerous communications from the regulator, the utilities, the vendors, and industry organs, failures persisted for at least 30 years.

A partial history of failures and communications follows below in Table 2.



Pressure Locking Flexible-Wedge Gate Valve

Table 2. Chronology of Valve Failures from Pressure Locking and Thermal Binding

DATE	*REFERENCE*	*DESCRIPTION*
June 1964	US Navy Bureau of Ships Instruction 9490.72	A general warning on the phenomena of pressure locking and thermal binding
December 1969	IRS 1391	The RHR containment sump isolation valve failed due to pressure locking.
March 1977	NRC Information Notice 81-31	Safety injection valves fail to operate against differential pressure
August 1982	IRS 0225	Two RHR valves to open on command due to pressure locking.

DATE	*REFERENCE*	*DESCRIPTION*
September 1983	IRS 1391	RHR heat exchanger outlet valves fail to open due to pressure locking.
November 1983	IRS 1391	RHR isolation valve can not be opened due to thermal binding
February 1984	IRS 1391	Three reactor depressurization valves fail to open due to thermal binding.
January 1988	IRS 1391	Two RHR hot leg valves fail to open due to pressure locking.
May 1988	IRS 1391	HPCI admission valve fails to open due to pressure locking.
July 1988	IRS 1391	HPCI steam supply isolation valve would not open due to thermal binding.
November 1988	IRS 1391	Containment spray valve fails to open due to pressure locking.
May 1989	IRS 1391	RHR suction valve fails to open due to pressure locking.
1989	NRC Generic Letter 89-10; IRS-0925	The GL requests a program of tests, inspection, and surveillance to assure that valves function under normal and abnormal conditions.
January 1990	IRS 1391	Letdown cooler isolation valve would not open due to thermal binding
June 1990	IRS 1391	PORV block valve would not open due to thermal binding
March 1991	IRS 1391	RHR suppression pool valves would not open due to thermal binding
July 1991	IRS 1391	Low-pressure ECCS valve fails to open due to pressure locking.
January 1992	IRS 1391	RHR suction valve from suppression pool fails to open due to pressure locking.

DATE	*REFERENCE*	*DESCRIPTION*
April 1992	NRC IN 92-26	Pressure locking of MOV flexible wedge gate valves
May-July 1995	IRS 1599	LPCI and core spray valves failed due to pressure locking
November 1995	NRC IN 96-08	HPCI valve failed due to pressure locking
October 1996	IRS 7220	HPIS gate valve failed to open due to pressure locking

Various causes were listed for these valve failures:

- The plant did not consider the problem credible because of prior history, and that industry communications were too general. Also, the engineering department did not build a case to convince management of the problem, and could not identify valves at risk. Even after engineering did recommend action, management failed to act on the basis of no prior events at a given plant.
- Training for plant staff on causes and cures for pressure locking and thermal binding was not provided. Various generic communications were placed on "required reading" for maintenance (but not operations) personnel. Even after training programs were instituted for maintenance and operations, there was none provided for engineering, in some cases.
- Surveillance tests for operability were not representative of the valve loading during the conditions inherent to the transient in question. For example, LPSI injection valve tests were performed by cycling the valves within 48 hours after cold shutdown.
- Vendor recommendations for valve modification were in some cases neglected in favor of maintenance and operational procedures.
- For some time there were no uniform guidelines on how to identify valves that might be susceptible to pressure locking or thermal binding.
- The valve design is deficient.
- The operating experience feedback program was deficient in that event repetition occurred

2.3 Service Water System Failures

The service water system in the nuclear power plant serves a broad variety of roles. In addition to transporting decay heat to the ultimate heat sink, the service water (a safety-related system) also may provide cooling for the emergency diesels, the containment fan coolers, control room chillers, component cooling (such as seal water) and other safety-related functions.

Various sorts of failures and degradations have occurred over the past 20 years in the service water systems. To illustrate the recurring nature of some service water failures, the subset of events that involved the presence of marine organisms was examined, as noted below.

Table 3. Buildup of Marine Life in Various Systems

DATE	REFERENCE	EVENT
September 1980	**IRS 0162.G2**	**Asian Clam Buildup on Reactor Building Cooling Coils**
June 1981	**IRS 0162.G!**	**Barnacles restrict flow in component cooling system**
February 1983	**IRS 0348**	**Marine growth suspected in failure to operate of four 30-in valves in SW system**
October 1985	**IRS 0162**	**Service water system was impaired by marine growth in heat exchanger**
March 1988	**IRS 0914**	**Biofouling led to flow blockage in steam generator; flow control valves clogged with Asian clam shells.**
1996	**IRS 7150**	**Bivalve mollusks invaded diesel intercooler, main condenser, and turbine oil cooler**
April 1997	**IRS 7136**	**Shrimp invasion clogs circulating water screen and leads to reactor trip.**

2.4 BWR Instability

Under certain conditions a BWR may inadvertently operate in an unstable mode. There is a long history of such operation as shown in Table 4.

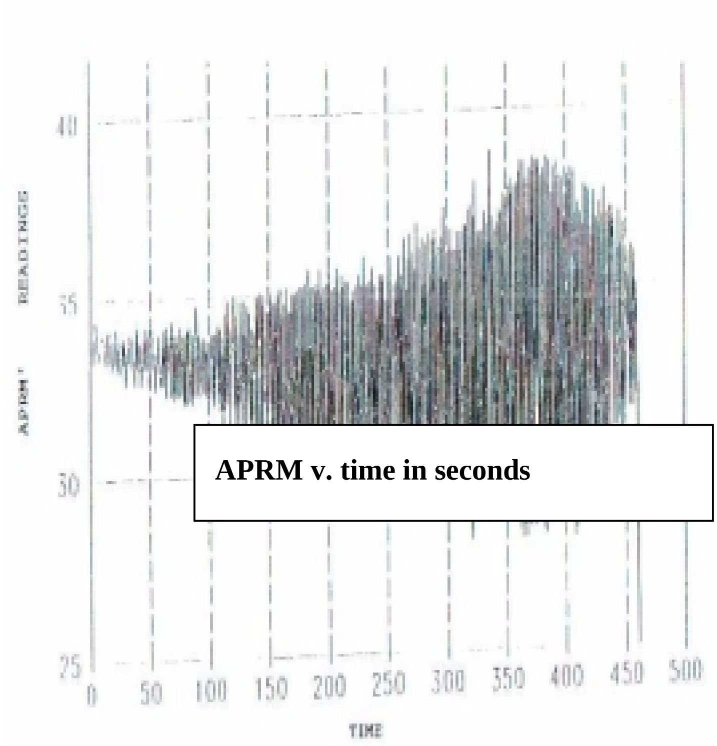
Table 4. Chronology Of Bwr Instability Events

Date	Reference	Discussion
June 1982	IRS 220; Caorso	53% power; 38% flow 2.5 sec period of oscillation; did not meet GDC # 12
October 1983	IRS 363; Caorso	Oscillations induced by pressure fluctuations during special tests
December 1984	IRS 677; Garona	14% instabilities; operation beyond control rod pattern line
June 1988	NRC Bulletin 88-07; Lasalle 2 BWR	Oscillations after recirc pump trip; inadequate insts. , procedures, training.
January 1991	IRS 1226; Cofrentes	Restart after scram; at 31% flow & 41% power; 10% oscillations
July 1991	IRS 1247; ISAR-1	During power reduction the reactor entered the unstable zone
August 1992	IRS 1388; WNP-2	Instability during startup in an unexpected region of power/flow/rod patterns
February 1994	IRS 1545; Muehleberg	Core stability tests as part of power upgrade; induced by pump trips
January 1995	IRS 1469; Laguna Verde	Oscillations during startup were in unexpected zone, much like WNP-2.

There were at least nine reports of BWR instability over the 1982-1995 time span, which indicates that recurrence is taking place, at least for this 13-year interval. Six different countries reported the instances of instability. In retrospect the direct causes were established and included:

- Operation outside the recommended power/flow/rod pattern zone;
- Operation with bottom-skewed flux shape;
- Startup operations with transient reductions in feedwater;
- Operation with cold feedwater;
- Operation with mixed cores;
- Inadequate instrumentation, procedures, and training.

In most cases there was information or guidance from the reactor vendor, or owner's groups, or the regulatory body, or from previous instances of instability. Nonetheless the instabilities recurred over this 13-year history. One illustration of Average Power Range Monitor (APRM) readings v. time in seconds for power oscillations is shown in the figure to the right.



APRM v. time in seconds

As expected, regulatory guidance varies among the reporting countries, as does the interface with the various vendors and operators. For example, the USNRC has very general guidance in its General Design Criterion #12 which provides that operators should assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. In 1986 the NRC issued Generic Letter 86-02 which requested BWR licensees to examine core reloads and impose such operating limits as necessary to ensure compliance with GDC #12. This letter cited General Electric SIL (Service Information Letter) #380 as competent guidance on operation with respect to instability.

In 1988, following the LaSalle incident (NRC Bulletin 88-07 and supplement 1) issued additional guidance, and cited an interim letter from General Electric to BWR licensees. Plants that did not have automatic scram protection for regional oscillations were requested to manually scram the reactor in the event of loss of the recirculation pumps. More justification of the operating boundaries (power, flow rate, rod position) was needed in some cases.

Additional guidance[7] flowed from the NRC in 1994 (see IRS 1528) for long-term solutions to instabilities in BWRs. The BWR owner's group had been studying the situation for several years, and had issued its report in 1991, which contained the proposed long-term solution. This generic letter had a number of requested technical actions which could include procedures, training, hardware additions, and supporting analyses which together would comprise the long-term solution.

The NEA-CSNI sponsored an International Workshop on BWR Stability at Brookhaven, New York in October 1990.[8]

Thus, the interval from knowledge of the potential of an undesirable event to the promulgation of the intended final solution can easily take a decade.

[7] Generic Letter 94-02).

[8] See CNSI Report #178.

CHAPTER 3. RISK IMPLICATIONS

Risk of Mid-Loop Operation.

The risk implications of loss of RHR at mid-loop have been considered (Shutdown, 1993). This reference summarizes the risk perspectives that had been reached in other assessments, as follows:

- NSAC-84 documented a risk assessment on the Zion PWR. The mean cdf at shutdown was estimated at 1.8×10^{-5}, of which 61%, or about 1×10^{-5} was attributed to failures in reduced-inventory operations;

- The NRC issued NUREG/CR-5015 which found that the mean cdf at shutdown was estimated at 5.2×10^{-5}, most of which was attributed to loss of RHR. The two dominant core damage sequences involved loss of RHR suction as a result of over-draining the RCS. Operator errors dominated the risk in this instance.

- The Seabrook PRA for shutdown found a total shutdown cdf of 3.5×10^{-5}, which was dominated (82%) by loss of RHR initiators.

- The findings of the report noted that, from the risk perspective, the most significant concerns included failures during mid-loop operation; operator errors (especially failure to restore shutdown cooling at mid-loop, and procedural errors); loss of RHR shutdown cooling induced by operator errors or by cavitation due to over-draining.

In a separate report (Barriere, 1994) there was an observation on the contribution of human error. Thirty-two events at PWRs (of which 17 involved loss of shutdown cooling) were studied. A total of 39 errors were identified of which 9 were errors of omission and the remaining 30 were errors of commission. These errors of commission involved commission of undesired tasks, analyses or steps, or involved errors of commission induced by other human factors elements, such as a faulty reactor coolant level indicator.

Risk of Pressure Locking or Thermal Binding of Valves

Risk numbers were not immediately available but some upper limit could be inferred from the potential, for example, for common-mode failure of an ECCS system, such as high-pressure injection. In this case, the cdf would approach the initiator value, such as 10^{-3}, if the system were unavailable for a year. More precise techniques such as used in the accident sequence precursor programs, would shed more light on conditional core damage probability.

Risk of Service Water Faults

In IRS 0920 there was an estimate of the core damage frequency attributable to loss of service water as being in the range of 1×10^{-3} to 1×10^{-5}.

Risk of BWR Power Oscillations

BWR oscillations are not generally characterized as a precursor to core damage. Divergent oscillations, or even limit cycle operations, may cause undue stress on the fuel, with local failures. In theory oscillations could lead to a boiling crisis with fuel failure due to undercooling. In some cases oscillations could lead to violation of core safety limits, which would indicate the need for automatic protection.

However, since the general view is that instability as currently experienced is not a dominant contributor to risk, there may have been a tempering of efforts to avoid recurrence.

CHAPTER 4. CONCLUSIONS AND RECOMMENDATIONS

Some common attributes for these four case studies can be seen:

- The problems had existed for many years;
- The problem was well documented and distributed to the concerned parties, and were well-known by international groups and the subject of discussion at various meetings;
- In most cases the solution to the problem was identified in the early events, but not implemented;
- Human error plays an important role
- The event sequences are important from a risk perspective, in some cases, and more or less risk-significant on others;
- The operating experience feedback programs were not effective in achieving closure;

Observations:

Some factors that might contribute to recurrence include:

- The affected utility (i.e., where the recurring event took place) failed to take timely action, or was not aware of the previous event, or believed that it was not vulnerable to the event, or had plans for remedial action but had not yet implemented them, or was willing to assume the risk of recurrence for various reasons (such as an intent to decommission the plant in the near future);
- The cognizant regulatory authority was not aware of previous events, or was aware but had not yet taken prompt remedial regulatory actions, or had taken action but had not imposed a short time scale for incorporation of a remedy, or had not imposed the proper remedy;
- Central utility groups and allied r & d groups were working on a unified solution but had not yet reached agreement on the solution at the time of recurrence;
- National and international groups responsible for collection and analysis of operating experience had not collected or analyzed the information from previous events, and thus had not disseminated proper warnings to the operating organizations and regulatory bodies;
- Perceptions of risk significance might be tempering the time scale for remediation; in some cases the desire to bring plants back into the design basis for the plant might be in conflict with the desire to regulate or operate in proportion to the perceived risk.[9]

[9] For example, if BWR power oscillations were not thought to be of prime risk significance, then this would tend to mitigate the efforts to impose a prompt conformance to the criterion of avoidance of operation in the unstable zone.

- More prompt closure - that is, remediation of the root causes - is the proper role of the utility operator and national regulatory body;

Some recommendations follow:

- The international bodies (such as PWG1 of the NEA-CSNI, and the IAEA Technical Committee of IRS Coordinators) could more diligently track the events which seem to be recurring10 and highlight these events at annual meetings and in various publications;
- The causes for delay in implementation of solutions could be studied by the international bodies. In particular there should be a study to determine if loss of corporate knowledge is a contributor. Changes in training programs for utility staff and regulatory staff might be effective in assuring that lessons learned have a more permanent nature.
- Further, it seems appropriate to consider the extent to which risk-based regulation is playing a role, in that some recurring events may at the same time indicate both operation outside the design basis, but of minimal risk to core damage. This could be a joint project for PWG-1 and PWG-5.

[10] If this were done, it would be important to capture, in a recurring event, the reasons that the earlier events had not been considered.

REFERENCES

Hada, M. Report of the Task Group Reviewing the Activities of the PWG#1NRA/SEN/SIN/WG1 (94) 5. October 1994.

Haried, J.A. Evaluation of Events Involving Service Water Systems in Nuclear Power PlantsNUREG/CR-2797. November 1982.

Lam, P. & Leeds, E. Operating Experience Feedback Report – Service Water System Failures and Degradations NUREG-1275, Vol. 3. November 1989.

NUREG-1449. Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States. September 1993.

Barriere, M. et al. An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Issues.

NUREG/CR-6094. June 1994

NUREG-1410 Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990. June 1990.

APPENDIX A

INPO REPORTS ON LOSS OF DECAY HEAT REMOVAL

FOR THE DECADE 1980-1990
AS DOCUMENTED IN NUREG-1410.

1. INPO SER 74-81. There was a loss of DHR at the Davis Besse station in April 1980. While at a reduced water level, there was a loss of a non safety 13.8 kv feeder bus. As a consequence there was some actuation of safeguards, which interfered with DHR.
2. INPO SER 78-81. There was a loss of DHR at Beaver Valley 1 while at reduced water level. Some errors in level measurement were mentioned, as well as the caution about shifting to a standby RHR pump, if suction is lost on the running pump.
3. INPO SER 87-81. A loss of DHR at Trojan while at reduced water level was discussed. The level indication was incorrect, as influenced by a partial vacuum in the RCS.
4. INPO SER 51-83. This SER discussed a loss of DHR at McGuire 1 while the level was being reduced. The level indication system was out of service.
5. INPO SER 60-83. There was loss of DHR at North Anna 2 (May 1983) and Surry 1 (May 1983) while draindowns of the reactor vessels were in progress. Note was made of the need to obtain reliable water level information and to have reliable communication between the control room and the workers (presumably at the reactor vessel).
6. INPO O& MR 171. This report discussed the loss of DHR at Calvert Cliffs 1 during refueling with a reduced reactor coolant system inventory. Apparently some testing with automatic actuation took place (perhaps similar to item 1, above).
7. INPO SER 79-84. This report was a composite of losses of DHR at Trojan, Arkansas Unit 2, and McGuire 2. In each case the reactor inventory was being reduced. Emphasis was made on the following topics:
   a. Potential for loss of DHR is greater just after shutdown;
   b. Accurate water level measurement is needed; redundancy should be provided; low-level alarms should be provided; and, trend recorders are indicated;
   c. Use of tygon tubing connected to drain lines has caused numerous incorrect readings;
   d. The RCS inventory does not have to drop below the suction pipe level for loss of DHR, as vortexing can take place;
   e. Recovery methods for DHR should be included in training and procedures.

8. INPO SOER 85-4. This SOER discussed loss of DHR at Zion 1 (September 1984); Beaver Valley 1 (June 1983); Summer (October 1982) and Cook 2 (May 1984). More emphasis was placed on training, procedures for surveillance, maintenance, and operation of the RHR system; recovery procedures for loss of DHR; elective maintenance of the DHR system; disabling of certain autoclosure valves in the RHR system; and, use of independent level measurement systems.

9. INPO O&MR 295. This report discussed the loss of DHR while at reduced water level at Sequoyah in October 1985. Reminders were issued to cover procedures for switching DHR pumps; investigation of causes for tripping a running pump (concern over common mode failure); and, methods for restoring water to the vessel.

10. INPO SER 17-86. This SER discussed the loss of DHR at San Onofre 2 in March 1986. Note was made of the need to have instrumentation to monitor reactor coolant level; need to resolve discrepancies in level indication; and, need to have in the plant procedures at least one HPI to refill the reactor vessel.

11. INPO SER 23-86. This SER discussed the loss of DHR at Crystal River in February 1985. It observed that five recent events had air entrainment caused by vortexing in the DHR suction line, which caused loss of DHR. It observed the need to more closely observe the operation of the system, and the water level. Procedures should reflect this concern. Removal of the fuel is preferred if there is to be long-term operation with reduced inventory.

12. INPO SER 31-86. This SER discussed the loss of DHR at Ringhals 4 in August 1984. This was for information only, as preventive measures were discussed in SOER 85-4.

13. INPO SER 35-86. This SER discussed the loss of DHR in July 1986 at Waterford 3 while the reactor level was being reduced. Industry was requested to consider monitoring of level during draindown, with comparisons between independent level systems. Operators should be aware of plant status at all times. Recovery procedures should be available, and level indications should be capable of operation during a partial vacuum (tygon tubing systems are not).

14. INPO SEN 8. This SEN had preliminary information on the loss of DHR at Diablo Canyon in April 1987, while the inventory was being reduced.

15. INPO SER 15-87. This SER provided more details on the Diablo Canyon event. Several measures were proposed to industry, including a caution about the relatively short time to heat the reactor vessel fluid following a loss of DHR; need to evacuate certain areas around the RCS, should a loss of DHR occur; utility of a gravity feed from the refueling water storage tank, for providing vessel makeup; and, need for procedures to vent gas and vapor from the RHR pumps, if they become bound during the transient.

16. INPO SOER 88-3. This SOER discussed the loss of DHR at Waterford, Diablo Canyon, San Onofre, and Zion 2. A number of recommendations were made, along the lines previously discussed, in the areas of level control, procedures for operation of the DHR system, training of operational staff, need for alarms, and, alternate means of monitoring the reactor coolant temperatures.

17. INPO SER 36-88. This SER discussed the loss of DHR at Byron 1 in September 1988; Cook 2 in June 1988; and Cattenom in May 1988. In each case the reactor vessel water level was lowered. A new observation was made on the maximum acceptable drain rate and the corresponding drain-down-time limit.

18. INPO SER 5-89. This SER covered the loss of DHR at Oconee 3 while a surveillance test was being performed with reduced inventory in the vessel. The observation was that supervision should clearly understand the impact that the test or surveillance could have, and if appropriate the test should be rescheduled.

The table on the next page summarizes the dates and plant names for the INPO reports above. Over the period of observation there were 22 instances of loss of DHR. In several instances the event recurred at the same plant. Many of the factors were common, including difficulties with level measurement, procedures and training, alarms systems, recovery procedures, and maintenance factors related to desire to reduce refueling outage time.

Plants Covered by INPO Reports in Decade 1980-1990

Date	*Plant*
April 1980	**Davis Besse**
March 1981	**Beaver Valley 1**
June 1981	**Trojan**
April 1983	**McGuire 1**
May 1983	**North Anna 2**
May 1983	**Surry 1**
?	**Calvert Cliffs 1**
May 1984	**Trojan**
August 1984	**Arkansas 2**
December 1983 *And 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**