AEOD SPECIAL EVALUATION

REVIEW OF OPERATI..G EVENTS OCCURRING
DURING HOT AND COLD SHUTDOWN AND REFUELING

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AEOD SPECIAL EVALUATION REPORT

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SUBJECT: REVIEW OF OPERATING EVENTS OCCURRING DURING HOT AND COLD SHUTDOWN AND REFUELING

EVENT DATES: Multiple

1 SUMMARY

This study examines operating experience during shutdown conditions (hot and cold shutdown and refueling). The objective of the study is to understand and categorize the types of events that have occurred at both pressurized water reactors (PWRs) and boiling water reactors (BWRs) while shutdown. Examples of the types of events are provided in this study. A *qualitative* judgement of the safety significance of some of the types of events is discussed. The primary source of event information was the Sequence Coding and Search System (SCSS) (see reference 1). The system was used to obtain events which occurred while shutdown during a target period from 1988 through July 1, 1990. In addition, known significant events occurring outside the target period and some events reported to the NRC through means other than Licensee Event Reports (LERs) have been included where appropriate. However, since the objective was to gain an overall understanding of the *types* of events which occur during shutdown, there was no intention in this study to gather *all* events occurring during shutdown over a specified period of time. A complete listing of the 348 domestic events obtained through the above process is provided in Appendix A. Approximately 30 percent of the events in Appendix A are discussed in this report.

Categories of events arising from this evaluation are: loss of shutdown cooling, loss of electrical power, containment integrity problems, loss of reactor coolant, flooding and spills, overpressurization of the reactor coolant system, and, for pressurized water reactors, boron problems. Other types of events [e.g., fires and void formation in PWRs] apparently occurring less frequently are also discussed in Section 2.4 below.

2 OPERATING EXPERIENCE

2.1 Shutdown Environment: Equipment Removed From Service and Multiple Activities On-Going Simultaneously

During shutdowns, a wide variety of both safety and non-safety equipment is taken out of service for testing and maintenance and multiple activities are on-going at the same time. The operations staff is relied upon to take appropriate action should the situation warrant.

As an example at *Millstone 3 on June 22, 1989*, it was discovered that technical specifications for the electrical distribution system were not met. At shift change, review of the electrical alignment indicated that a load center was being supplied by a cross-tie with a breaker trip setting of 600 amps. The normal trip setting was 1600 amp and it was determined that because of this cross-tie, the "A" electrical system could not meet its requirements. A contributing cause of this event was that adequate procedures did not exist to ensure Mode 5 (cold shutdown) and Mode 6 (refueling) electrical requirements were met. The event is an example of lack of procedural guidance to ensure that technical specification requirements are met during hot and cold shutdown and refueling (see reference 2).

A similar event occurred at *McGuire 2 on March 29, 1986*, when, due to personnel error, a charging pump was removed from service with the other charging pump not having an emergency electrical power source. Had a charging pump been required with a loss of offsite power, none would have been available until the second charging pump was returned to service (see reference 3).

A third example occurred at *Calvert Cliffs 2 on April 26, 1984*, in which the independence of the two Unit 2 diesel generators was lost. The plant was at mid-loop condition. The electrical alignment resulted in power to the auxiliaries (e.g., air start system) for diesel generator 21 being powered from diesel generator number 12. At this point, because of the lack of independence, diesel generator number 21 was declared inoperable. However, since only one diesel generator was required to be operable in refueling mode, no log entry was made of the diesel generator inoperability. The next shift crew failed to recognize the dependence of the number 21 diesel generator on the number 12 diesel, and, during this shift the number 12 diesel was removed from service and, thus, both diesel generators were out of service. When the situation was discovered, the number 12 diesel generator was returned to service. As corrective action, a list of diesel generator equipment required to be operable was prepared and training of the operations staff on the role of electrical distribution was emphasized (see reference 4).

An additional example of a similar event occurred at *Turkey Point 4 on February 9, 1989*, when the diesel generator associated with the operable residual heat removal (RHR) train was removed from service for about 3 and one-half hours for testing. The other train of RHR was out of service due to pump maintenance. Prior to removal of the diesel generator from service, the other diesel was tested, which removed it from service. The refueling cavity was flooded, no fuel movement was in progress at the time, and decay heat levels were very low. However, this type of event has the potential to cause a loss of RHR (see reference 5).

Also during shutdowns, separate maintenance or testing activities are generally on-going at the same time. Coordination of these activities is often a problem. As an example, at *Byron 2 on February 11, 1989*, with the refueling cavity flooded, a load sequencing test was in progress. Part of this test involved de-energizing an instrument bus. A containment pressure sensor supplying a signal to the reactor protection system was powered by this bus. At the same time work was underway on a second containment pressure sensor and this pressure channel was in test. When the instrument bus was deenergized for the sequencer test, the 2 out of 3 logic for high containment pressure was satisfied and a safety injection signal was initiated. This caused approximately 120 gallons of water to be injected into the vessel. The licensee reviewed controls to prevent unrelated work activities form causing such events. While the safety significance of this event was apparently low, it is an example of the type of events which can be caused by the numerous activities occurring at the same time (see reference 6).

2.2 Pressurized Water Reactor Events

In general, the relative vulnerability that shutdown conditions represent can be understood by considering the state of five variables within the plant. These are: (1) level of decay heat in the core (or time after shutdown); (2) reactor coolant system (RCS) integrity (i.e., open for maintenance or closed); (3) RCS inventory (i.e., reduced inventory, mid-loop, versus full RCS with steam generator primary side full); (4) sources of electrical power to safety related components (i.e., number of sources of offsite power and number of diesel generators); and (5) containment integrity (i.e. open or closed). This characterization can be displayed in a form shown in Table 2.1.

Table 2.1 Vulnerability of Reactor States

Variable	State of Variable Higher Vulnerability	 Lower Vulnerability
Decay Heat Level	High	Low
RCS Integrity	Open	Closed
RCS Inventory	Mid-Loop	Full
Electrical Sources	2	4
Containment Integrity	Open	Closed

Thus, events which have occurred with conditions similar to those on the right of Table 2.1 are generally less potentially severe than those on the left.

2.2.1 Loss of Shutdown Cooling

Prior AEOD studies (see references 7 and 8), chiefly AEOD Case study C503, "Decay Heat Removal Problems at U.S. Pressurized Water Reactors," Dr. H. Ornstein, December, 1985, identified losses of shutdown cooling in PWRs as a significant problem and made specific recommendations to correct the problems.

Losses of shutdown cooling have occurred frequently. Table 2.2 provides a list of some events involving loss of shutdown cooling. This table is not intended to list all losses of shutdown cooling events but does provide a representative set of such events. Review of the events in the references quoted below indicates that events at reduced inventory are more severe in terms of core exit temperature increase than events in other conditions.

Reference 9 (Vogtle 1 loss of offsite power and decay heat removal Incident Investigation Team report) provides a detailed evaluation of an event involving loss of decay heat removal. In that event, while decay heat level was relatively low, the remaining factors (i.e., only 2 of 4 sources of electrical power, both RCS and containment open, and the plant in reduced inventory condition) increased the plant's vulnerability. While the Vogtle 1 event did not result in RCS coolant reaching saturation, other events have resulted in boiling in the core (see for example references 10 and 11 for Diablo Canyon 2 and Waterford respectively). Thus, loses of shutdown cooling during PWR reduced inventory operation are potentially safety significant events. Events in Table 2.2 are listed in order of the maximum RCS temperature reached.

Table 2.2 Selected Loss of Shutdown Cooling Events

Plant	Date	Description
Waterford 3	07/14/86	Near mid-loop, a drain path was not closed and level dropped. The operating RHR pump began to cavitate and was shutdown. RHR was lost several times over a 3 and one-half hour period. RCS temperature rose 94 °F to 232 °F. (Ref. 11)
Diablo Canyon	04/10/87	At mid-loop, 7 days after shutdown with vessel head vented, a leak caused loss in RCS level and the RHR pump became air bound. RHR was lost for 1 hour, 28 minutes; RCS temperature rose 87 °F to 220 °F and pressure increased to 10 psig. (Ref. 10)
San Onofre 2	03/26/86	Level was near vessel flange as level was being lowered for maintenance. Running RHR pump became air bound and was secured. Second pump was started and also became air bound. Level indication in Tygon tube was not accurate. Level was raised and RHR was restored after 70 minutes; RCS temperature rose 96 °F to 210 °F with local boiling occurred in the core. (Ref. 9)
Crystal River 3	08/28/89	The RCS was closed and a steam generator was operable. The reactor was at about 165 psig and 190 °F. A step down transformer faulted due to insulation degradation caused by aging. This caused loss of RHR for 19 minutes; RCS temperature rose 20 °F to 210 °F. Various equipment status lights were lost, complicating the event. (Ref. 12)
Catawba 1	04/22/85	RCS level being lowered for maintenance with one train of RHR also out of service for maintenance. During the level reduction, the operating RHR pump became air bound. Level instrumentation was inaccurate. RHR lost for 81 minutes; RCS temperature rose to 177 °F. (Ref. 9)

Table 2.2 Selected Loss of Shutdown Cooling Events (continued)

Plant	Date	Description
Arkansas 1	12/19/88	Relay problem caused RHR suction valve in single suction line to clos[illegible] [illegible]HR lost for 12 minutes; RCS temperature rose [illegible] F to 147 °F. (Ref. 13)
Braidwood 2	02/23/89	Testing of ESF system caused RHR suction isolation valve to close; RHR lost for 43 minutes; RCS temperature rose 29 °F to 146 °F. (Ref. 14)
Vogtle 1	0[illegible]/20/90	At mid-loop, loss of offsite power to operating RHR pump and failure of diesel generator to continue to run. RHR restored after 41 minutes; RCS temperature rose 46 °F to 136 °F. (Ref. 9)
Crystal River 3	02/02/86	At mid-loop, RHR pump shaft broke after continuous operation for about 30 days. Placing second RHR pump in operation was delayed due to tripped breaker powering the suction valve. Temperature rose 33 °F to 131 °F. (Ref. 9)
Salem 1	05/20/89	Testing of ECCS SI accumulator resulted in discharge of nitrogen into RCS. RHR pumps became gas bound. RHR lost for almost 50 minutes; RCS temperature rose 30 °F to 122 °F. (Ref. 15)
Arkansas 1	12/06/89	A maintenance error resulted in loss of power to one 4160 V AC bus. This resulted in tripping of the operating RHR pump. RHR was lost for about 9 minutes; RCS temperature rose 17 °F to 120 °F. (Ref. 16)
McGuire 1	11/23/88	Near mid-loop conditions, while tensioning head and performing test, air was ingested into the RHR pump and RHR was lost for 39 minutes. RCS temperature rose 26 °F to 116 °F. (Ref 17)

Table 2.2 Selected Loss of Shutdown Cooling Events (continued)

Plant	Date	Description
Oconee 3	09/11/88	Near mid-loop, during performance of a test, confusion between personnel caused loss of electrical power and consequent loss of RHR for 15 minutes. RCS temperature increased 15 °F to 105 °F. (Ref. 18)
Palisades	12/16/87	Controller for shutdown cooling heat exchanger outlet valve caused valve to close. RHR lost for 19 minutes; RCS temperature rose 19 °F to 103 °F. (Ref. 19)
Arkansas 1	10/26/88	Near mid-loop with RCS open, loss of power to RHR flow controllers caused valves to close. RHR lost for 23 minutes; RCS temperature rose 18 °F to 87 °F. (Ref. 9)
Zion 2	12/14/85	Level below vessel flange with RHR and charging in operation, RHR pump vortexing occurred and pump was stopped. Level had decreased to the point where vortexing occurred. Control room level indication was in error. RHR was lost for 75 minutes and temperature increased 15 °F. (Ref. 9)

In addition to losses of shutdown cooling to the core, losses of fuel pool cooling have occurred. Because of the higher ratio of coolant mass to decay heat present in the fuel pool, these events have been less severe in terms of temperature increase than events involving losses of shutdown cooling. One such event occurred at *Oconee 3 on August 30, 1988*. The reactor had been defueled with one source of electrical power out of service. A false signal caused by maintenance personnel using a inadequate procedure resulted in loss of the source supplying electrical power to the spent fuel cooling pumps. Power was lost to the pumps for 18 minutes. Reference 20 describes the temperature increase as insignificant.

At *Haddam Neck on June 8, 1990*, while restoring the electrical alignment to normal following DC bus maintenance, power was lost to both spent fuel pit cooling pumps for about 25 minutes. No temperature increase was observed (see reference 21).

2.2.2 Loss of Electrical Power

Losses of electrical power to portions of the plant are common. These range from complete losses of offsite power, as described in reference 22, to losses of DC buses or instrument buses. These events are frequently contributors to other types of events. For example, about 30 percent of the PWR loss of shutdown cooling events listed in Table 2.2 were due to losses of electrical power. About one-half of the loss of electrical power events found in this study were accompanied by a loss of shutdown cooling. In general, the significance of loss of electrical power events are determined by the effect the loss of power has on the plant. For example, losses leading to loss of significant instrumentation or annunciators represent a significant class of events.

Table 2.3 Selected Loss of Electrical Power Events

Plant	Date	Description
Yankee Rowe	11/16/88	Due to a maintenance error, power was lost to two emergency 480 volt buses. One source of offsite power and one diesel generator were unavailable due to maintenance. The plant has three diesel generators. A tie breaker problem delayed restoring power to one of the busses. (Ref. 23)
Fort Calhoun	03/21/87	Due to a personnel error, all AC offsite electrical power was lost for 40 minutes and the one operable diesel generator did not start because the control switch was in the "off" position due to the erection of scaffolding around the diesel. The diesel was manually started and shutdown cooling was then restored after 5 minutes. (Ref. 22)
McGuire 1	09/16/87	Due to maintenance error during testing, the unit experienced a loss of offsite power for 6 minutes. The diesel generator started and loaded. The loss of power caused a loss of non-vital air which resulted in feedwater regulating valve closing and consequent reactor trip on Unit 2. (Ref. 24)
Harris	10/11/87	One incoming line to one of the safety buses was out of service for modification. The one remaining incoming line breaker tripped caused by accidental jarring of protection relays. The diesel generator on this bus started and loaded appropriately. (Ref. 25)
Wolf Creek	10/15/87	A series of engineered safety feature actuations occurred due to reduced battery voltage. The 480 volt AC bus which powers the battery chargers was removed from service for maintenance. At the onset of the work, a battery life time estimate was made. However, this time was exceeded and the 125 volt DC source was lost. (Ref 26)

Table 2.3 Selected Loss of Electrical Power Events (Continued)

Plant	Date	Description
Braidwood 2	01/31/88	Due to a maintenance error in an invertor cabinet, an instrument bus was deenergized. This caused loss of power to the source and intermediate range neutron detectors which in turn caused a reactor trip signal. (Ref. 27)
Indian Point 2	11/05/87	With all 3 diesel generators inoperable, a relay test in conjunction with failure to reset a relay after a pervious test caused a loss of power to the 480 V AC buses. Power was restored within 3 minutes. Additional battery cable work caused loss of instrument buses which complicated the event. (Ref. 28)
Millstone 2	02/04/88	In preparation for a test, the auxiliary contact of a breaker was energized causing loss of one train of vital 4160 V AC. The other train was out of service for maintenance. The diesel generator started but because of a sequencer failure did not load. Operator action was required to re-energize the bus and reestablish shutdown cooling. (Ref. 29)
Crystal River 3	10/16/87	The startup transformer was accidentally shorted in preparation for maintenance. One of two vital buses lost power. An engineered safeguards actuation complicated power restoration. Power to the control room annunciators and event recorder was lost. An Alert was declared. (Ref. 30)
Turkey Point 3	05/17/85	A offsite fire caused loss of power to the site. Unit 3 startup transformer was re-energized 31 minutes later. (Ref. 31)

Table 2.3 Selected Loss of Electrical Power Events (Continued)

Plant	Date	Description
Fort Calhoun	02/26/90	One of two diesel generators and one of two sources of offsite power were out of service. The containment equipment hatch was open. Power to 4160 safety buses was lost when a trip circuit breaker opened for unknown reasons. The operable diesel generator started. Because of a design feature would not load on the bus until the RHR pump motor connected to the bus was tripped. The pump motor breaker was opened and the bus was re-energized. A 2 °F RCS temperature increase occurred. (Ref. 32)

2.2.3 Containment Integrity Problems

There have been a number of containment integrity problems at PWRs. Most of these events occur due to multiple maintenance activities occurring at the same time. As an example, at *Farley 1 on September 30, 1989*, on two separate occasions core alterations were performed without containment integrity required by technical specifications. A pathway existed from the steam generator secondary side manway through the steam generator atmospheric relief valve. The event was caused because the operations staff failed to properly verify the position of the atmospheric relief valves using local indication. The procedure has been improved (Ref. 33). Additional similar events are described in references 34 through 36.

The apparent safety significance of these specific events is low because a fuel handling event would have to occur coincident with such an event to create a release. However, they represent a pattern which could make the results of a potential fuel handling event worse.

2.2.4 Loss of Reactor Coolant

There have been several events during shutdown involving losses of reactor coolant system inventory. The safety significance of this type of event is that it could potentially lead to voiding within the reactor vessel and possible eventual core uncovery. At *Braidwood 1 on December 1, 1989*, an event involved the loss of approximately 67,000 gallons of reactor coolant. The unit was in cold shutdown at 350 psig preparing for return to power when a residual heat removal pump suction relief valve lifted. The nature of the event was recognized by the operations staff which increased charging and isolated first one RHR system then the other in an effort to determine which train was leaking. The leak was isolated approximately 2 hours and 20 minutes after it started. Reference 37 indicates that procedures addressing losses of reactor coolant are not developed for shutdown conditions.

A similar event with much smaller loss of inventory occurred at *Farley 2 on November 27, 1987*, a residual heat removal system suction pressure relief valve opened when one of two series RHR containment sump suction valves was stroked. The sump suction valve was being opened to demonstrate operability. However, the procedure indicated that the RHR train should not have been operating. The pressure relief valve failed to reseat. The RHR train with the open relief valve was secured. The other RHR train was also in service. The licensee stated that the event was apparently caused by a pressure pulse when the sump suction valve was opened. The amount of water discharged was approximately 2200 gallons (see reference 38). Another loss of reactor coolant resulting from an improperly set RHR suction relief valve at Cook 2 is described in reference 39.

A loss of reactor coolant event at *Palisades on November 21, 1989*, (see reference 40) resulted from training and procedural problems. When the power operated relief valve (PORV) block valve was opened to test the PORV. When the block valve was opened, the PORV also opened. The valve cycled twice and the reactor depressurized from 2154 psi to 1565 psi.

Losses of reactor coolant have also occurred in refuel mode during refueling cavity filling. At *Indian Point 2 on March 25, 1989*, (see reference 41) within 4 hours, the cavity level was observed to drop 5 inches. The leak was estimated at 60 gpm. The leak was subsequently determined to be due to nuclear instrument box cover and due to two locations on the vessel to cavity seal.

At *Surry 1 on May 27, 1988*, the refueling cavity was flooded and all fuel had been removed from the vessel. The instrument air supply was isolated to support maintenance. This resulted in the reactor cavity ring losing pressure and subsequent leakage by the ring. Approximately 25,800 gallons of water drained from the cavity (see reference 42).

At *San Onofre 2 on June 22, 1988*, the fuel pool level dropped and about 6000 gallons of fuel pool water drained into the reactor cavity. The flow path was through the spent fuel pool purification pump suction lines between the fuel pool and the cavity. The event resulted from inadequate procedure which had an incorrect order for closing isolation valves (see reference 43).

At *Sequoyah 1 on May 23, 1988*, with the unit near mid-loop while attempting to start the second train of shutdown cooling, the wrong valve was opened and RCS inventory was pumped back to the refueling water storage tank. Approximately 6000 gallons of RCS inventory was lost and level dropped to the point that RHR pump cavitation occurred and RHR was lost. Gravity feed from the refueling water storage tank (RWST) was used to replace the lost inventory and RHR was restored with no adverse temperature increase (see reference 44). Similar events occurred at Waterford (reference 45), at Braidwood 2 (reference 46) and at Catawba 2 (reference 47) where misalignments resulted in RCS water flowing back to the refueling water storage tank.

During vessel level reduction, at *Byron 1 on September 19, 1988*, the cavity water level was being reduced to replace a vessel stud hole protective insert. The upper internals assembly was in place and one RHR pump was running to lower level. The reduction of cavity level was being visually observed. The other RHR pump began to ingest air and was stopped. The cause of the loss of RHR was that the upper internals created a flow restriction and water level in the vessel was actually less than that observed from the containment operating floor. Level was increased and RHR restored (see reference 48).

2.2.5 Flooding and Spills

There have been a number of events involving flooding or spills. The events themselves appear to have had a lower safety significance. However, just as with fires discussed in Section 2.4.1, the safety significance of flooding events may be significant depending upon the type and amount of equipment which is made inoperable by flooding. Reference 49 discusses flooding events specifically in significantly more detail than can be done in this report. A brief discussion of relevant shutdown events follows.

At *McGuire 2 on September 5, 1989*, one train of containment spray was overpressurized, causing a containment spray heat exchanger bottom flange gasket to fail. This caused a leak of approximate 10,000 gallons of reactor coolant into the auxiliary building. The event resulted from an inappropriate valve lineup which connected the containment spray system to the pressurized RHR system. Additionally a lack of attention to detail by operations staff was involved (see reference 50).

At *South Texas 1 on April 9, 1990*, The reactor cavity was being filled. One of two required spool pieces for the cavity was not installed. A drain path from the lower internals storage area to the containment resulted. Approximately 17,000 gallons of water from the refueling water storage tank spilled into containment (see reference 51).

At *Salem 1 on December 22, 1987*, one of service water bays was flooded due to failure of valves to completely close. This resulted in flooding of the turbine building and the Cardox Room (see reference 52).

Additionally, there have been several spills out of steam generator manways when RCS level was raised too high. Two examples are given in references 53 and 54.

2.2.6 Pressurization of the Reactor Coolant System

Pressurizations of the reactor coolant system (RCS) or of auxiliary systems attached to the RCS have occurred during shutdown. Events found in this evaluation have not been of high safety significance. With only one exception, pressurizations of the RCS have been limited by the plant's low temperature overpressure protection system.

These events generally are of three types, e.g., operations with the RCS completely full ("water solid") with pressure control problems, occurrences of inadvertent safety injections, or pressurization of systems attached to the RCS.

Regarding operations with the vessel "water solid", at *Catawba 1 on March 20, 1990*, the RHR suction relief valve opened during a reactor pressure increase from atmospheric to 100 psig. However, vessel pressure indicted near zero gage. Subsequent investigation determined that the root valves for the control room pressure indication were closed. The actual vessel pressure had reached approximately 175 psig (see reference 55).

Other pressurization events occurring during water solid conditions often happen due to isolation of letdown paths or inadvertent opening of charging paths. References 56 through 58 provide examples of this type of event.

One event, at *Millstone 3 on January 19, 1988*, involved failure of the low temperature overpressure protection system to operate. The event was mitigated by operator action. In this event, isolation of the RHR letdown path caused a pressure excursion. The peak RCS pressure reached was 526 psia. The low temperature overpressure protection system did not work because a power supply which was required for system operation had its fuses removed without knowing what effect this removal would have on the system (see reference 59).

Regarding safety injections which produce pressurizations, these events (see references 60 and 61) have been limited by the plants overpressure protection system.

Regarding pressurization of auxiliary systems that are connected to the RCS, these events are potentially significant events because of the possibility of pressurization of systems designed for pressures less than RCS pressure, i.e., intersystem loss of coolant. One such event was found in this evaluation. At *Salem 2 on October 27, 1989*, one train the RHR suction piping was pressurized to 600 psig. This pressure is 150 psig above its design pressure. The event occurred because the RHR cold leg injection valve did not seat properly and allowed RCS coolant to pressurize the line. The licensee stated in reference 61 that the failure of the valve to seat was most likely due to the conditions when RHR flow had been terminated. This condition was that the RHR system was water-solid at 350 psig with no differential pressure across the injection valve to assist in seating it (see reference 62).

2.2.7 Boron Problems

This evaluation found two types of PWR problems with boron. First dilutions have occurred while plants were shutdown. Second, boration capability required by technical specification has been lost. The domestic events found in this study have low safety significance.

At *Surry 2 from April 14 to 23, 1989*, reactor coolant system inventory was increasing and boron concentration was decreasing for unknown reasons. Over this period attempts were made to discover the cause of the dilution. On April 23 it was discovered that the source of the primary grade water was a reactor coolant pump standpipe makeup valve leak. The dilution was less than 80 ppm over the 9 days and shutdown margin requirements were maintained (see reference 63). Other dilution events are described in references 64 and 65.

A second boron related problem is loss of boration capability. An example event occurred at *Turkey Point 3 and 4 three times between May 28 and June 3, 1987*. On May 28, 1987, an attempt was made to add borated water to the volume control tank using the Unit 4 boric acid transfer pumps (BATPs). No flow was passed. Unit 3 BATPs were used to add borated water to Unit 4. Work orders for the Unit 4 BATPs were prepared. On May 29, 1987, the Unit 3 BATPs were used in an attempt to add borated water to Unit 3. No addition occurred. Similar problems occurred on June 1 and 3, 1987. The cause was subsequently determined to be nitrogen, which is used as a cover gas over the BATP seal water tank. Nitrogen had entered one of the BATPs and, because of interconnection of the systems had gas bound *all* of the pumps (see reference 66).

Gas binding of the charging pumps occurred at *Arkansas 2 on May 4, 1988* due to emptying the volume control tank because of a faulty level indication. Makeup capability was restored

4 hours later (see reference 67). In addition, references 68, 69, and 70 provide information about technical losses of boration capability at McGuire 2, Wolf Creek and Turkey Point 3 respectively.

At a *foreign reactor in 1990*, an unexpected RCS boron dilution took place (see reference 71). The vessel level was at about mid-loop, boron concentration was at 2083 ppm, and the RHR system was providing cooling. The secondary side of the number 2 steam generator (s/g) was being filled. However, a s/g tube had been cut, a portion removed, and had not been plugged. Thus, as the secondary side of the steam generator was filled, water leaked from the secondary side of the s/g to the primary system. This leakage resulted in a reduction in boron concentration to about 2001 ppm. The maximum leak rate was estimated to be 45 gpm which was limited by the size of the s/g tube. This event was bounded by a plant analysis. It was estimated that, had the dilution continued, the core might have reached criticality in about 4 hours.

In addition to the actual event, the failure of a s/g primary nozzle dam was also postulated after a sufficient time to fill the s/g cold leg side below the tube sheet. This would release about 1050 gallons of non-borated water into the cold leg and then the core. Several variations of this scenario were evaluated. The worst scenario involves starting a reactor coolant pump in a loop where there is little mixing and thus a slug of non-borated water enters the bottom of the reactor core. This postulated event could produce a criticality event with high flux. Because of this postulated event, in which several separate contributing situations occur simultaneously, the national safety authorities required that methods be put in place to ensure that steam generator tubes which are remove are in fact plugged.

2.2.8 PWR Summary

Based on the operating experience found during this evaluation and discussed above, review of the seven types of events indicates that the most severe type of event has been losses of shutdown cooling while at reduced inventory. However, other types of events (e.g., boron dilutions) have the *potential* to be severe.

2.3 Boiling Water Reactor Events

2.3.1 Loss of Shutdown Cooling

Losses of shutdown cooling have occurred frequently at boiling water reactors (BWRs).

At *Susquehanna 1 on February 3, 1990*, in cold shutdown a surveillance test was performed requiring that the residual heat removal (RHR) system be stopped (see reference 72). RHR could not be restored due to a short circuit in a reactor protection system distribution bus which in turn caused the isolation signals to the RHR suction valves. These signals could not be reset until power was restored to the reactor protection system (RPS) bus. The reactor water cleanup system was also isolated. An alert was declared when temperature increased from 188 °F to 200 °F in 30 minutes causing the reactor to enter the hot shutdown operating condition. According to plant procedures, the safety-relief valves were used to control reactor coolant temperature. One loop of the RHR system was placed in suppression pool cooling mode. The control rod drive system was the source of makeup water. During the event the suppression pool temperature rose from 62 °F to 69 °F. The temperature and pressure had risen to 253 °F and 20 psig respectively after 2 hours and 22 minutes when the isolation signal was reset. The short was caused by the configuration of the circuit breaker in the RPS distribution panel.

At *Fitzpatrick on October 31, 1988*, a loss of electrical power occurred when the only source of offsite power was lost due to a line fault possibly caused by high winds. The diesel generators started and loaded. Shutdown cooling was lost for approximately 95 minutes when the RHR system isolation occurred. Shutdown cooling could have been returned to service sooner but was not needed. Temperature rose approximately 10 °F (see reference 73).

At *River Bend on June 13, 1989* various engineered safety features occurred due to personnel shorting two leads during transformer maintenance. Shutdown cooling was lost for approximately 24 minutes. However, reference 74 states that no adverse consequences occurred during the loss of shutdown cooling.

AT *Fermi 2 on March 18, 1988*, during maintenance an RHR isolation occurred due to loss of power. Operator action restored shutdown cooling in 23 minutes. Two alternate methods of shutdown cooling also existed (see reference 75).

There have been several loss of shutdown cooling events at BWRs resulting from hydraulic pressure transients while starting RHR pumps. The high pressure isolation is intended to prevent overpressurization of the RHR system. The starting of RHR pumps may create pressures at or near this isolation set point. At *Fitzpatrick on March 20, 1990*, the reactor pressure was 6 psig (see reference 76). The isolation set point was about 60 psig. When the injection valve was opened following start of a second RHR pump, an isolation occurred causing loss of RHR for 1 hour and 22 minutes. However, RHR could have been returned to service sooner but was not required.

Similar examples occurred at *Fitzpatrick on January 20, 1990*, at *Susquehanna 1 on March 21, 1984*, at *Pilgrim on December 9, 1989*, and at *Susquehanna 1 on January 7, 1989* (see references 77, 78, 79, and 80 respectively).

As with PWRs, losses of spent fuel pool cooling have also occurred at BWRs. At *Duane Arnold on July 9, 1990*, vital offsite power was lost for approximately 35 minutes (see reference 81). Fuel pool cooling was lost for about 20 minutes. However, no increase in spent fuel pool temperature was observed.

2.3.2 Loss of Electrical Power

As with PWRs discussed above, losses of electrical power to portions of the plant during periods of shutdown are common. Complete losses of vital offsite power have occurred and well as losses to smaller portions of plant electrical equipment. These events are frequently contributors to other types of events. For example, about 66 percent of the BWR loss of shutdown cooling events were due to losses of electrical power. As with PWRs, about one-half of the loss of electrical power events found in this study were accompanied by a loss of shutdown cooling. As with PWRs, in general, the significance of loss of electrical power events are determined by the effect the loss of power has on the plant. Table 2.4' lists several losses of electrical power events to provide perspective on this type of event.

Table 2.4 Selected Loss of Electrical Power Events

Plant	Date	Description
Pilgrim	11/12/87	In cold shutdown (having been so for 17 months), with the alternate source of offsite power out of service for plant modifications, snow caused loss of the both transmission lines. The diesel generators started and loaded. One of the diesel generators was shutdown after 9 hours due to operational problems. Offsite power was available after 11 hours. However, it was not restored for 21 hours. (Ref. 9)
Nine Mile Unit 2	12/26/88	All offsite power was lost due to explosion of current transformer in switchyard. The other source of offsite power was out of service. The site diesel generators started and carried sequenced loads. The alternate source of power was subsequently restored. (Ref. 82)
Millstone 1	04/29/89	With the fuel offloaded to the spend fuel pool, a loss of normal power occurred with the gas turbine emergency generator out of service. The emergency diesel generator started and powered sequenced loads. The event occurred during modifications due to a maintenance error. (Ref. 83)
Washington Nuclear Unit 2	05/14/89	With the reactor in refueling with the vessel head off and the cavity flooded, a partial loss of offsite power occurred when two offsite feeders were lost due to operator error. The diesel generator started and loaded as designed. The loss of power to the reactor protection system caused an isolation of RHR which was restored about 1 hour later. (Ref. 84)

Table 2.4 Selected Loss of Electrical Power Events (continued)

Plant	Date	Description
River Bend	03/25/89	In refueling with the vessel head removed, division II lost power for about 4 and one-half hours due to a cable failure. The associated diesel generator was out of service for maintenance as was the 125 volt battery. The loss of power resulted in a loss of control annunciators and status lights for about 40 minutes until the battery could be returned to service. (Ref. 85)
Limerick	03/30/90	In cold shutdown, a reactor protection system uninterruptible power supply (RPS/UPS) failed causing loss of power to a RPS/UPS panel. Among other things, this resulted in loss of RHR for 17 minutes with resultant increase in reactor coolant temperature of less than 1 °F. (Ref. 86)

2.3.7 Containment Integrity Problems

Containment integrity (secondary containment in BWRs) problems have occurred. The safety significance of the events which were found in this evaluation are apparently low. Three examples are:

At *Washington Nuclear Project 2 from May 5 to May 29, and on May 31, 1989*, secondary containment integrity was not maintained as required. A main steam isolation valve had been previously disassembled. A turbine throttle valve was removed providing a path from the secondary containment to the atmosphere from the main steam isolation valve through the steam line to the throttle valve. This event was caused by failure to recognize that a path existed. When recognized, the main steam line isolation valve (MSIV) was reassembled (see reference 87).

At *Clinton on March 4, 1989*, during the preparation for a local leak rate test, the valve alignment was revised. This process resulted in a direct path from the steam tunnel in secondary containment to the turbine building (which is outside secondary containment. Fuel movement was in progress at the time. The event occurred because a licensed operator failed to factor existing valve alignments into the additional alignments used for the test (see reference 88).

At *Oyster Creek on November 8, 1988*, both doors of a temporary airlock built to support isolation condenser maintenance may have been open for up to 6 and one-half hours. This was a degradation of secondary containment. This event was caused by administrative problems with the work package for the isolation condenser maintenance (see reference 89).

2.3.3 Loss of Reactor Coolant

There have been several events during shutdown involving losses of reactor coolant system inventory. The safety significance of these events is that they could potentially lead to core uncovery. At *Vermont Yankee on March 9, 1989*, in refueling with the head off and water level approximately 290 inches above the top of the fuel (13 inches below the vessel flange), maintenance activities associated with RHR valve operators were in progress (see reference 90). The "B" loop of RHR was in service. In part, these activities involved the following sequence: The circuit breakers for the "A" loop RHR pumps were opened and racked down and their control circuits were deenergized. This deenergization caused the "A" loop RHR pump minimum flow valve to open. Approximately 20 hours later, the suction valves for the "A" loop of RHR were manually opened. This established a gravity drain path from the vessel to the suppression chamber from the common supply line to the RHR system through "A" RHR loop suction valves and then through the "A" loop minimum flow valve to the suppression pool.

When the valves were opened, the level dropped about 18 inches. The operations staff attributed the drop in level to refilling drained portions of the "A" RHR loop. They did not know the minimum flow valve was open. However, when the operations staff was informed about 15 minutes later that the level had dropped an additional 18 inches, the operations staff began an investigation of possible leakage paths. About 47 minutes after the event started, the leakage path was identified, and personnel were sent to close the manual isolation valves for the two "A" loop RHR pumps. This closure was accomplished about 54 minutes after the start of the event with a level decrease of 72 inches. The total amount which leaked from the reactor to the suppression pool was about 10,300 gallons.

At *Washington Nuclear Plant 2 on May 1, 1988*, in cold shutdown with one loop of RHR operating, shutdown cooling was being shifted to the other loop of RHR (see reference 91). In order to perform this shift, the RHR suction valve from the vessel was closed. However, this suction valve requires about 120 seconds to fully close. After the operations staff moved the switch to close the suction valve, the suppression pool suction valve was opened. For about 40 seconds both valves were open providing a gravity drain path from the reactor vessel to the suppression pool. About 10,000 gallons of water were drained to the suppression pool. The lowest vessel level was 142 inches above the top of active fuel. In order to prevent similar events, the procedure was changed. A design change to prevent the concurrent cycling of the two valves was to be implemented.

Several events have occurred due to inappropriate or improper filling of the RHR system. An event occurred at *Fermi 2 on March 17, 1987*, during the later stages of a plant shutdown in which the RHR system was being prepared for service (see reference 92). Due to the way in which the RHR system was warmed and filled, the system was not totally filled and, when the system was started, vessel level dropped 34 inches from 199 to 165 inches above top of active fuel. The problem was due to procedural inadequacies and personnel error in that inability to use a path suggested in plant procedures was not compensated for properly.

A second RHR filling event occurred at *Pilgrim on December 3, 1988* (see reference 93). Due to licensed personnel error, one train of RHR which had previously been out of service and partially drained for testing was partially empty when the system was being filled. This was due to the method in which the operations (concurrent filling and venting) were preformed. The vessel level dropped in excess of 20 inches.

At *Susquehanna 2 on April 27, 1985*, vessel level dropped about 35 inches when one of the RHR trains was placed in service. Also subsequently, water hammers occurred due to collapse of steam in the voided RHR piping (see reference 94).

An event occurred at *Limerick on April 7, 1989*. In refueling with the vessel cavity flooded, a craftsman performing maintenance removed the packing from the wrong valve. The valve was in the operating RHR train. This resulted in reactor coolant spraying out of the valve from which the packing had been removed. The control room was notified and the RHR system was stopped and the valve was isolated. The other RHR train was placed in service. Reference 95 indicates that the leak was about 1 gpm.

2.3.4 Flooding and Spills

At *April 19, 1989 at River Bend*, a freeze seal failed on a 6-inch service water line. The freeze seal was being used to allow inspection and repair work on manual isolation valves on a auxiliary building cooler. Portions of the auxiliary building were flooded by approximately 15,000 gallons of water. The flooding caused loss of the operating residual heat removal system, spent fuel cooling and normal lighting in the auxiliary building, control building and the reactor building. RHR was restarted in 17 minutes with no increase in reactor coolant observed. The fuel pool temperature increased to 123 °F (see reference 96).

At *Clinton on March 20, 1989*, during cold shutdown, about 40,000 gallons of water were gravity drained for the containment refuel pool to the drywell. This spill resulted in about 4 inches of water in the drywell. The event occurred when the service air was isolated from containment resulting in loss of the air inflated seals for the gate separating the dryer storage pool from the reactor cavity pool. The reactor cavity pool was empty but the dryer pool drained into the reactor cavity pool. The drywell head was not fully tightened and water entered the drywell. The event was caused because the isolation of service air from the containment was not adequately evaluated prior to the isolation. Equipment was submerged. In addition, evidence of equipment being wetted above the standing water level was found (see reference 97).

At *LaSalle 1 on November 2, 1989*, during the process of filling the reactor vessel, a flange cover on one of the main steam lines leaked. The flange cover was over a safety-relief valve opening. The valve had been removed for maintenance. This leakage indicated that the main steam line plug was leaking. The steam line safety relief valve flange was tightened. However, about 3000 gallons of water entered the drywell and wetted equipment (see reference 98).

2.3.5 Pressurizations of the Reactor Coolant System

Events involving pressurization of BWRs have occurred. At *Clinton on March 19, 1989*, in cold shutdown with the reactor coolant pressure about 1000 psig with the vessel and attached piping water "solid" for reactor coolant pressure boundary pressure testing, concurrently with this test, scram time testing was being conducted (see reference 99). When the scram time testing was completed, the mode switch had to be moved from the refuel to the shutdown position. It was recognized that this action might potentially increase reactor pressure. Because of this potential increase, the reactor pressure was reduced to 900 psig and the operations staff discussed appropriate actions should an increase in reactor pressure occur.

When the mode switch was moved, the scram inlet and exhaust valves of all control rods opened. Because of the valve opening and because the control rod drive pump was running, the reactor pressure increased at about *seven psig per second*. Prior to operator action, four safety-relief valves opened to relieve pressure. The maximum pressure noted at a local test gauge was 1130 psig. To prevent recurrence, operations staff will be briefed on this event and procedures will be changed.

At *Peach Bottom 3 on October 23, 1989*, during hydrostatic testing with the reactor pressure about 1000 psig excess flow check valve testing was in progress. Additionally, control rod drive scram time testing was on-going. In an effort to increase reactor temperature, the amount of letdown through the reactor water cleanup system was decreased. The plant process computer was used to monitor reactor pressure. However, a problem had occurred and the computer was not updating its pressure indication. In fact, it was later determined that reactor pressure had begun to increase at about 6 psi per minute. The narrow range pressure recorder and the high pressure alarm were inoperable at the time contributing to the event. In addition numerous on-going activities distracted the control room staff and also contributed to the event. When pressure reached 1055 psig, a reactor scram signal was generated (all rods already were fully inserted). This resulted in stopping the control rod drive (CRD) flow to the vessel and the high pressure condition cleared very soon after the scram signal. In order to avoid recurrence of this type of event, a coordinator position will be established to minimized the impact of separate activities on the operations staff (see reference 100).

At a *foreign reactor in 1989*, near the end of a refueling outage the reactor vessel pressure reached 1150 psig with the reactor coolant temperature between 79 and 88°F. Information about the event indicates that this pressure substantially exceeds the allowable pressure at a temperature of 88 °F (see reference 101).

At the time of the event, vessel level was about 6 inches below the vessel flange. Also, the main steam line isolation valves were shut as were the safety-relief valves, the vessel head vent and the reactor water cleanup system. RHR was in service. To support maintenance activities, the control rod drive pump was started. This action injected about 25 gpm into the vessel. Reactor pressure increased to the RHR isolation set point and RHR isolated automatically. Information about the event indicates that the operations staff was not aware of the isolation of the RHR system. When RHR isolated, the vessel was effectively "bottled up" except for the CRD injection. The vessel pressure increased to 1150 psig at which point the CRD pump tripped due to low suction pressure.

2.4 Other Types of Events

In addition to the above event groupings, there are several other types of events which have apparently occurred less frequently. These have not been included in the above event categories but are potentially safety significant. These include personnel industrial accidents, fires, core alteration/fuel handling problems (including fuel damage); improper fuel loading situations involving potentially inadequate neutron monitoring; radiation releases; and, in PWRs, void formation in the vessel.

Personnel industrial accidents involving personnel injury are not discussed. However, some examples of this type of event which are safety significant are discussed above in Section 2.3.2, Loss of Electrical Power.

2.4.1 Fires

Fires have occurred during periods of shutdown. The safety significance of any fire depends on its location. However, fires are generally potentially significant especially when coupled with the potential lack of accessibility which a fire can cause. The unpredictability of fires and their effects forces plant operators to rely on a program for protection against and response to fires (Appendix R to 10 CFR Part 50).

Several fires during shutdown have been due to the presence of hydrogen gas. At *Robinson, on January 7, 1989*, two separate fires were observed associated with the station and instrument air systems. These two air systems were cross-connected at the time because the air compressor for the station air system was out os service. The fires were the result of hydrogen gas entering the air systems. An incorrect line up was performed on January 6, 1989, which, allowed hydrogen to enter both the instrument and station air systems. Hydrogen reached flammable concentrations in the air systems' piping (see references 102 and 103).

The main generator hydrogen cooling system was being tested for leakage following extensive maintenance. The objective of the operation was to pressurize the generator hydrogen system using air. Instead, a connection from the plant air systems to the plant bulk hydrogen gas supply resulted in connecting a rubber hose to the wrong location. This connection established a flow path between the higher pressure hydrogen supply and the plant air systems. The cause of the event was a breakdown in administrative controls in that the maneuver was performed without a procedure, without checking the appropriate system drawings, and also included a failure to verify an isolation valve closure. Longer term corrective actions included ensuring a procedure will be used which describes the connection point for the rubber hose, ensuring that potential hazards associated with hydrogen systems are understood, and the clearance procedure will be modified to prevent recurrence of valve problems.

At *Wolf Creek, on October 14, 1987*, a welder was preparing to make a second tackweld on a pressurizer instrument sensing line and when he struck an arc. At that time, the welder heard a loud rushing sound lasting 5 to 10 seconds. The welder also saw a sheet metal cover taped over the pressurizer safety valve flange blown off. Also, additional personnel inside containment informed the control room that they heard a loud noise (see reference 104). It was concluded that a hydrogen gas ignition had taken place in the pressurizer. Preliminary analyses indicated that the pressurizer was not adversely affected by the ignition and the maximum pressure was 30 psig. The hydrogen gas came from pressurizer drain down activities.

At *Byron 2, on January 15, 1989*, the contents of an emergency core cooling safety injection accumulator ignited. A person was using an electro-mechanical sampler to collect a sample of the atmosphere when a rush of warm gas and a small cloud came out of the accumulator for approximately 5 seconds. Noise and vibration also occurred.

Subsequent analysis of a previously obtained sample indicated that the concentration of hydrogen gas was sufficient to be flammable but not explosive. Calculations indicted that the maximum pressure which might have occurred during the ignition was below the accumulator design pressure. Inspection indicated no evidence of damage (see reference 105).

The licensee concluded that the hydrogen came from the waste gas system due to the similarity between the gas compositions in the two volumes. One of the gas decay tanks provides cover gas through a regulator for various tanks including the reactor coolant drain tank (RCDT). The presence of the cover gas prevents a vacuum from being formed when the tanks are drained. When the accumulator drain valve to the RCDT was open and the accumulator was open to the containment, there was adequate pressure differential for gases to flow from the RCDT to the accumulator. Procedures were revised to required isolation of the RCDT following the draining operation. Controls will also be implemented to required a combustibility analysis prior to collecting gas using an electro-mechanical sampler.

At *Fermi 2, on October 24, 1989*, A cable fire occurred in the drywell (see reference 106). An extension cord or temporary lighting cord caught fire. Efforts to extinguish the fire were delayed by accessibility problems. The fire lasted nineteen minutes. There was no damage to plant equipment.

At *Beaver Valley 1 on December 13, 1989*, with the unit in cold shutdown, a fire occurred in the "B" diesel generator engine control cabinet. This cabinet is located adjacent to the diesel generator. The fire occurred while the diesel was supplying power to the "B" train emergency buses. The diesel was manually tripped locally and power was restored to the "B" electrical buses from offsite power. The fire was extinguished in about 3 minutes. Shutdown cooling and other required safety systems were not affected (see reference 107).

2.4.2 Core Alteration/Fuel Handling Problems

There have numerous fuel handling problems during shutdown. Several NRC information notices have been issued describing some of these events (see references 108 through 112). Types of fuel handling/core alteration events which have occurred are fuel assemblies being raised too high and no longer meeting the minimum height requirement for water above the fuel assemblies, dropped fuel assemblies, damage of fuel assemblies due to striking structures while moving the assemblies, fuel pins dropped or damaged during reconstitution or inspection, fuel assemblies lifted out of the core by the upper guide structure during their removal, and failure of a lifting rig attachment at St. Lucie 1 (see reference 111).

Potential consequences of fuel handling events involving the movement of one fuel assembly are analyzed as part of the normal "Chapter 15" safety analysis (see for example reference 113). Potentially more serious PWR fuel handling events involving multiple assemblies lifted by the upper guide structures are possible. One event occurring at *Indian Point 3 on October 4, 1990*, resulted in two fuel assemblies being lifted out to the core by the upper internals when it was removed during a refueling outage (see reference 114). This type of event, may create the possibility of an unanalyzed fuel handling accident since these analyses only consider a single dropped fuel assembly.

2.4.3 Core Alterations with Inadequate Neutron Monitoring

Several events involving core alterations with inadequate neutron monitoring have occurred. In the first example, at *Nine Mile 1, on January 15, 1990*, it was discovered that the source range monitor (SRM) in the quadrant into which fuel was being loaded was bypassed which is a violation of technical specifications (see reference 115). Thus, the SRM would not have produced a scram signal on high flux. However, the flux levels were monitored and criticality was extremely unlikely.

At *Clinton on January 22, 1989*, in refueling while performing core alterations, it was discovered that there was no operable source range monitor in the quadrant next to he one in which alterations were being performed. This detector was inoperable because its high voltage cable was disconnected. The operations staff was not aware that the detector was not in service. The detector was declared operable even though it had no power. (see reference 116)

2.4.4 Radiation Releases

There have been radiation releases while plants were shutdown (see for example reference 15). However, these releases are no different than those during power operation and, thus, will not be considered further in this evaluation.

2.4.5 PWR Vessel Void Formation

One event involving the formation of a void in a PWR vessel head was found in this study. At *Turkey Point 4, between October 21, 1987, and November 3, 1987*, voids occurred in the head seven different times (see reference 118). On October 21, 1987, the heated junction thermocouple vessel level system indicated that a void existed in the upper vessel head area. The unit was at atmospheric pressure in preparation to draw a bubble in the pressurizer. The void was compressed and eliminated by increasing reactor coolant system (RCS) pressure to 30 psig. Initially, the origin of the void was believed to be hydrogen gas coming out of solution as the RCS was depressurized. Subsequent analysis of the RCS coolant hydrogen concentration indicated that this belief was not correct.

At Turkey Point 4, the process of filling and venting the RCS continued and subsequent to RCS depressurization on October 25, 1987, a second void was indicated in the vessel head. At this time the vessel head and pressurizer were vented to the pressurizer relief tank (PRT). The gas in the PRT was analyzed and indicated a nitrogen gas concentration of about 96 percent. Subsequently again on October 25, another void was indicated. The vessel head and pressurizer were again vented to the PRT. Subsequently on October 25, the licensee concluded that the source of the nitrogen gas was an emergency core cooling safety injection accumulator. The event was further complicated by communication problems related to the head vent valve position and possible blockage of the head vent at the orifice. The source of the nitrogen gas which formed the voids was ECCS SI accumulators which caused nitrogen saturated water to enter the RCS.

During cold shutdown conditions at PWRs, due to the increased solubility of gases in water, the formation of voids in the vessel would have to be caused by introduction of gaseous materials into the RCS. The existence of reactor head vents allows the removal of these voids. Based on this one event, there appears little safety significance associated with formation of voids during hot or cold shutdown.

2.4.6 Loss of Annunciators

At *Brunswick 2 on December 22, 1989*, an Alert was declared while in refueling condition because audible annunciators were lost from 3:25 p.m. until 4:44 p.m. This occurred during a plant modification. Visual alarms continued to function (see reference 119).

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24. Licensee Event Report 87-021, Duke Power Co., Docket No. 50-369, October 29, 1987.

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93. Licensee Event Report 88-005, Boston Edison Co., Docket No. 50-293, December 30, 1988.

94. Licensee Event Report 85-016, Pennsylvania Power and Light Co., Docket No. 50-388, May 28, 1985.

95. Licensee Event Report 89-024, Philadelphia Electric Co., Docket No. 50-352, May 3, 1989.

96. NRC Inspection Report 50-458/89-20, AIT for River Bend, May 12, 1989.

97. NRC Inspection Report 50-461/89014(DRP) for Clinton, June 8, 1989.

98. Region III Morning Report November 7, 1989.

99. Licensee Event Report 89-016, Illinois Power Co., Docket No. 50-461, April 18, 1989.

100. Licensee Event Report 89-006, Philadelphia Electric Co., Docket No. 50-278, November 22, 1989.

101. Proprietary reference.

102. Licensee Event Report 89-001, Carolina Power and Light Co., Docket No. 50-261, February 3, 1989.

103. NRC Inspection Report 50-261/88-36 for H.B. Robinson, February 6, 1989.

104. NRC Inspection Report 50-482/87-27 for Wolf Creek, November 30, 87.

105. NRC Inspection Report 50-455/89001(DRP) for Byron Station, February 28, 1989.

106. Immediate Notification Report 16921 from FERMI Unit 2, October 24, 1989.

107. Region I Morning Report of December 13, 1989.

108. NRC Information Notice 80-01: Fuel Handling Events, January 4, 1980.

109. NRC Information Notice 31-23: Fuel Assembly Damaged due to Improper Positioning of Handling Equipment, August 4, 1981

110. NRC Information Notice 85-12: Recent Fuel Handling Events, February 11, 1985.

111. NRC Information Notice 86-06: Failure of Lifting Rig Attachment While Lifting the Upper Guide Structure at St. Lucie Unit 1, February 3, 1986

112. NRC Information Notice 86-58: Dropped Fuel Assembly, July 11, 1986.

113. Vogtle Units 1 and 2 Final Safety Analysis Report, Chapter 15.

114. Immediate Notification Report 19526 from Indian Point 3, October 4, 1990.

115. Licensee Event Report 90-001, Niagara Mohawk Power Corp., Docket No. 50-220, February 21, 1990.

116. Licensee Event Report 89-007, Illinois Power Co., Docket No. 50-461, February 22, 1989.

117. Licensee Event Report 88-024, Consumers Power Co, Docket No. 50-255, January 23, 1989.

118. NRC Inspection Report 50-250/87-46 for Turkey Point Units 2 and 3, December 17, 1987.

119. Immediate Notification Report 17411 Brunswick Unit 2, December 22, 1989.

APPENDIX A

LIST OF EVENTS FOUND IN EVALUATION

SHUTDOWN AND REFUEL EVENTS DATABASE

	PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
280	ANO	1 LC	?	20 MIN LOSS SDC DUE LOSS POWER TO DHR OUT COOL VLVS-18F INC	Y	Y	313/88-14(S),	09/24/88	.F.
244	ANO	1 LC	OPE	MD LP PMPS,S/G OPN:VLVS CLOSE 23MIN/18F INC VLV CLOSED	Y	Y	N1410-C29,	10/26/88	.F.
252	ANO	1 LC	OPE	LOSS SDC 12 MIN DELT=12F FR135 TO 146F DUE TO CNTRACT MOV RL	Y	Y	313/88-24(R),	12/19/88	.F.
134	ANO	1 LE ES LC	LPE	2 EVNTS:LOSS 480 V-DG START BUSES-X-CONN FOR MAINT	N	Y	313/89-40,	12/05/89	.F.
253	ANO	1 SP	ADM	29 GPM LK DEV PD-UNISOLABLE	N	Y	313/88-23(S),	12/16/89	.F.
94	ANO	2 LE LC	OPE	SEC OFF TRIP BKR TIES 4160 TO 480 SDC LOST 8 MIN	N	Y	R4 04/14/88,	04/12/88	.F.
266	ANO	2 LC ES	OPE	12VDC PWR DE-ENG ALSO OTHER MAINT=-GOT ALL ESFS LC FOR 5 MIN	N	Y	368/88-07(S),	04/23/88	.F.
287	ANO	2 BN	EQI	VCT WENT DRY-GAS BINDING LOSS OF ALL NORM & EMRG BN CAP	N	Y	368/88-08(S),	05/04/88	.F.
188	ANO	2 LE LC	ADM	DURING TEST LOSS OFF SITE AND SDC FOR 1 MIN	N	Y	368/89-22(S),	11/14/89	.F.
343	BEAVER VALLEY	1 OT	OPE	1 TRN LTOP INOPW/RCPS RUNNG-RT VLV FOR PRESS SENSR SHUT			R1/111589,	11/13/89	.F.
185	BEAVER VALLEY	1 ES	LPE	SI SIGNAL DUE TO WRONG TRAIN OF SSPS BEING ENERGIZED	N	N	334/89-??(S),	12/13/89	.F.
333	BEAVER VALLEY	1 OT	EQU	FIRE TRN B DG ENG CNTL CABINET NEXT TO DG B RUNNING	Y	Y	R1/12/13/89,	12/13/89	.T.
46	BEAVER VALLEY	2 ES LE	OPE	TECHS DOING WORK CAUSED BKR TO OPEN LOST PWR D/G + START	N	Y	412/88-04,R1-2/2/88,	02/01/88	.F.
271	BRAIDWOOD	1 IN	OPE	WRONG CHNL CONNECTION CAUSES SI 1250GAL	N	Y	456/88-02(S),	01/25/88	.F.
208	BRAIDWOOD	1 ES IN	ADM	MS LO PRESS RX TRIP SI MSIV CLOSE DUE TO PERS	N	?	456/89-02(S),	04/16/89	.F.
203	BRAIDWOOD	1 OS	ADM	HPSI RQD TO B PLCD INSERV. VLV NOT CAPABLE	Y	Y	456/89-11(S),	09/20/89	.F.
136	BRAIDWOOD	1 LL SP	EQI	6BK GAL OUT RHR RELF VLV WHILE DRAWING BUB	Y	Y	EN 17235, R3/89-030(DRP),	12/01/89	.F.
48	BRAIDWOOD	2 LE OT	OPE	D/G START DUE TO TECH PROBLEM	N	Y	457/88-04,R3-2/2/88,	01/29/88	.F.
61	BRAIDWOOD	2 LE ES	OPE	PAINTERS BUMP CABNT AND CAUSE LE TO INST BUS RX SCRAM SIG	N	Y	457/88-03,	01/31/88	.F.
193	BRAIDWOOD	2 LC	PRO	2B RHR IN OP 2A OOS, ESF RESP TM TEST SENT SIGNAL CAUSE ISO	N	Y	457/89-01(S),	02/23/89	.F.
310	BRAIDWOOD	2 LL SP	PRO	MIS ALIGNMT OF RHR TO RWST WHILE PRESS RESULTS IN LOSS OF IN	Y	Y	457/90-02, [9.8KGAL LST]	03/18/90	.F.
1	BROWNS FERRY	1 LE ES	ADM	LOSS OF XFORMER CAUSES LOSS OF RAW COOLING AND SRT OF EQ COL	N	Y	259/89-12-1, N1410C-16,	05/05/89	.F.
209	BROWNS FERRY	2 LE	EQI	BUS FLT CUASES LOSS OFF SITE DGS START OK	N	?	N1410-C16, NRCIN89-64,	03/09/89	.F.
21	BROWNS FERRY	2 OS	EQI	FAILURE TO MEET T.S. DUE TO EQPMT PROB WITH RHRSW PMP	N	N	260/89-11,	04/04/89	.F.
246	BROWNS FERRY	2 OT	VAR	VARIOUS BF RHR OOS BUT STILL USABLE	Y	Y	260/89-18 ETC (S)	06/23/89	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

	PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
299	BRUNSWICK	1 LC ES	OPE	DUR TEST GOT RWCU & RHR ISOLATION 47 MIN 2DEG INC FLOD UP	Y	Y	324/88-26,	11/16/88	.F.
37	BRUNSWICK	1 OT	ADM	SBGT SYS INOP DUE MISCOMM BETWEEN OPS AND DESGN PROB	N	N	325/88-[illegible]-1,	12/14/88	.T.
240	BRUNSWICK	1 OS	AMD	CORE SPRY AND RHR OOS AT SAME TIME VIO TS	N	Y	325/89-08(S),	03/17/89	.F.
298	BRUNSWICK	2 OT	LPE	CR W/DRWN WITH SHORTING LNKS INSTALLED	N	Y	324/88-06(S),	03/08/88	.F.
239	BRUNSWICK	2 LC	LPE	ACTION TO PREVENT ISO DURING TEST FAILED	N	Y	324/89-15(S),	09/11/89	.F.
334	BRUNSWICK	2 OT	OPE	LOSS OF AUDIBLE ALARMS IN CR ALRT DCLRD	Y	Y	EN/174[illegible], EN	12/22/89	.F.
328	BRUNSWICK	* LC	UNK	SDC & RWCU ISO 38 MIN INC 6F TO 126F	Y	Y	EN18595,	05/30/90	.F.
285	BYRON	1 LC LL	DES	RED INV-HD OFF 4IN CAV LVL LVL REDUCED AND RHR LOST	Y	Y	454/88-07, N1410-C28,	09/19/88	.F.
24	BYRON	2 OT	ADM	HYDROGEN BURN DUE TO PASSAGE FROM WASTE GAS SYS TO ACCU	Y	Y	R3/IR 89-01/DRP,	01/16/89	.T.
25	BYRON	2 ES OS	PRO	2ND TRN OF SI START UNEXPECTEDLY DUE TO SURV	Y	Y	455/89-01-1,R3/IR 455-49-01(DR	02/11/89	.F.
17	CALLAWAY	1 LE ES	EQI	ESF DUE TO LOSS OF 120V RAD MON POWER	N	N	483/87-30,	10/23/87	.F.
140	CALLAWAY	1 ES IN	LPE	2 SI DURING TEST PRESS, FLOW, VOL INC NOT GIVEN	N	Y	483/89-05,	05/18/89	.F.
183	CALVERT CLIFFS	1 ES	LPE	2 STPS SKIPPED IN RETURING ACT LOGIC CAB TO OP	N	?	317/89-??(S)	03/20/89	.F.
309	CALVERT CLIFFS	1 IN ES	OPE	DURING TEST OF SI INST GOT SI:600GAL 14MIN		Y	317/90-03,	03/08/90	.F.
41	CALVERT CLIFFS	2 OS	LPE	D/G AUX POWERED BY OTHER D/G IF LOSS OF OFF SITE	Y	Y	318/84-05,	04/26/84	.T.
195	CALVERT CLIFFS	2 CN	ADM	2 EVNTS: CNTM INTEG NOT SAT DURING FUEL MOVE	N	Y	318/89-06(S),	04/17/89	.F.
196	CALVERT CLIFFS	2 CN	ADM	FUEL MOVE W/OUT CNTMT INTEG BOTH CNTMT VENT VLVS NOT CLOSED	N	Y	318/89-22(S),	12/27/89	.F.
142	CALVERT CLIFFS	* OT	ADM	2 UNIT:HPSI NOT LOCKED SHUT DURING WAT.SOLID	N	Y	317/89-19,	11/28/89	.F.
331	CATAWBA	1 LC	?	LVL BEING LWRD. PMP VORTX. 81MIN/TO 177F	Y	Y	N1410-C24,IN86101,	04/22/85	.F.
38	CATAWBA	1 OS OT	OPE	WRONG S/D INST GIVEN FOR SERV WATER ON SHRD SYSTEM	N	Y	413/87-36,	08/30/87	.T.
62	CATAWBA	1 LE ES	EQI	TEST CAUSES LD SHED, D/G START FAILURE TO REEN OTHER D/G OUT	N	Y	413/87-42,	1/17/87	.F.
284	CATAWBA	1 OT	ADM	DIF OF .5% BETWN SIGHT GLS AND CR LVL IND RHR RED BELOW 3KGP	N	N	413/88-27(S),	12/30/88	.F.
11	CATAWBA	1 LE PR	LC OPE	LOSS OF A TRAIN VITAL AC DUE TO WRONG TIME DELAY IN RELAY	Y	Y	413/89-01-02,	01/07/89	.F.
115	CATAWBA	1 PR OS	ADM	PRESS TO 460 PSI WHILE PRESS PRI SYSTEM DUE TO ROOT VLVS CLD	Y	Y	413/90-18,SP REPT(4/26/90)	03/20/90	.F.
311	CATAWBA	1 IN	ADM	5KGAL INJ EXCEED 200F/HR COOLDN RT RPT 3/26 AGAIN	Y	Y	413/90-22,	03/25/90	.F.
326	CATAWBA	1 SP LL	LPE	PATH FROM RHR TO RWST PRESS AT 335# DRP TO 110#. 5KGAL	Y	Y	EN18679,	06/11/90	.F.
269	CATAWBA	2 IN ES	OPE	MIDLP,SG ED CUR TSTNG INJ 400G DUE TO TEST ERROR	Y	Y	414/88-03,	02/09/88	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P U	REFERENCES	DATE	
202 CATAWBA	2 OS SP	LPE	FLOOD CAUSES RAD MON OUT WITH CNTMT PURG VLV OPEN	N Y	414/89-06(S),	04/03/89	.F.
231 CLINTON	1 LC	OPE	DUR SURV JUMPER WAS GRNDED FUSE BL CAUSE RHR ISO	N Y	461/89-05(S),	01/10/89	.F.
232 CLINTON	1 OT OS	ADM	CORE ALTS IN QUADNT W/OUT OP SRM IN ADJ QUAD-SRM WAS OOS	Y Y	461/89-07(S),	01/22/89	.T.
233 CLINTON	1 LL SP	LC PRO	DUR SURV RHR VLV SHUT MIN FLO OPENED AND REDRT WATER TO POOL	Y Y	461/89-09(S),	02/03/89	.F.
234 CLINTON	1 OT	LPE	LVL IN UPPER POOL LT 23 FT ABOVE W/ONLY 1 RHR LOOP OP	Y Y	461/89-11(S),	02/26/89	...
235 CLINTON	1 CN	LPE	DIR PATH STM TUN IN 2 CNTMT TO TUR BLDG WITH CORE ALTS	N Y	461/89-14(S),	03/04/89	.F.
63 CLINTON	1 PR OT	PRO	OVER PRESS DURING HYDOR SRVs OPEN	Y Y	461/89-16,	03/19/89	.F.
86 CLINTON	1 LL SP	LPE	40K GAL WATER FROM CNTMT REFUEL POOL TO DRYWELL WET. EQMT	Y Y	R3/IR 89-14(DRP),IN 89-63,	03/20/89	.F.
87 CLINTON	1 ES	OPE	SCRM ON HI IRM DUE TO JIGGLNG WIRES DURING SRM TROUBLE SHOOT		461/89-18,R3/IR 8914(DRP),	04/14/89	.F.
88 CLINTON	1 ES	OPE	SCRM DUE TO TOUCHING LEADS DURING SURV	N N	461/89-20,R3 IR 89-14(DRP),	04/15/89	.F.
89 CLINTON	1 OT	LPE	SCRM DIS VOL INSTRU VLVS NOT LOCKED IN PROPER POSITION	N Y	R3 IR 89-14(DRP),	04/20/89	.F.
210 CLINTON	1 LV	ADM	VLV LEAK AND CLOSED VLV CAUS RHR TO BE EMPTY LVL DPS 36.5"	Y Y	461/89-30,	07/14/89	.F.
308 COMANCHE PEAK	1 IN ES	PR EQI	250F, 380#,SI FLOW TO RCS 8KGAL-DUR RECVRY PZR HTUP RT EXCD	Y	445/90-04,R4/90-11	03/12/90	.F.
29 COOK	1 OT	PER	D/G FAILURE DUE TO MISADJMT AND ELE PRO WITH OVERSPD TRIP	N N	R3/IR 89-20(DRS),	04/10/89	.F.
270 COOK	2 LC LL	?	NO FUEL PMP DN PAST UPL INT LD LVL (EE 285 BYRON)	Y Y	N1410-C28,	06/16/88	.F.
58 COOK	2 SP	EQI	RHR HX OUTLT S/V LIFTED 300P LO RELEASING WATER PRT RUPT DIS	N Y	R3 IR/89-09(DRP), R5-2/16/89,	02/16/89	.F.
170 COOK	2 OT	OPE	INDIVID KILLED WORKING ON 4160 S/G-OTHERS INJURED-NO LOSS SD	N Y	EN18886,	07/13/90	.T.
65 COOPER	1 ES IN	OPE	D/G,CORE SPRAY STARTS DURING TESTING 2K GAL INJECT LVL UP 10	N Y	298/88-17,	05/26/88	.F.
66 COOPER	1 ES IN	LPE	D/G LOADING TEST CAUSES INJ OF APR 15K GAL CORE SP & RHR	Y Y	298/88-18,	06/09/88	.F.
64 COOPER	1 ES IN	OPE	CORE SPRY 500 GAL TO VESSEL DUE TO MISPLACED JMPR DURING TST	N Y	289/89-16,	05/10/89	.F.
321 COOPER	4 LC ES	EQI	SDC LOST 15 MIN NO INC IN TMP - LOSS OF RPS BUS DUE MG BKR T	Y Y	298/90-05,	04/30/90	.F.
302 CRYSTAL RIVER	3 LC	OTH	MD LP: RHR PMP SHFT BRK. INC FR 98 TO 131F.	Y Y	N1410-C25,	02/02/86	.F.
112 CRYSTAL RIVER	3 LE OS	ES LPE	LOSS PWR LOSS ANNCRS IN PREP FOR TEST	Y Y	302/87-21, IR87-36, N1410C8,	10/14/87	.F.
111 CRYSTAL RIVER	3 LE ES	OPE	LOSS OF PWR FOR 1 HR, DEATH DUE TO SHORTING IN SWTCH YD	Y Y	302/87-25, IR/87-36, R210,N141	10/16/87	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

	PLANT		CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
258	CRYSTAL RIVER	3	ES IN	OPE	INADV INJ SI 1K GAL VIO PROCD	N	Y	302/88-21(S),	10/14/88	.F.
259	CRYSTAL RIVER	3	LC	PRO	PRESS INC HTUP SDC ISO VLV CLOSED PROCD WRONG PRESS	N	Y	302/88-22(S),	10/23/88	.F.
179	CRYSTAL RIVER	3	LE	ADM	DEG VOLT DG START VOLTS RECOVERED SDC NOT LOST	N	?	302/89-13(S),	04/09/89	.F.
19	CRYSTAL RIVER	3	OT	PER	UNEXPECTED HEAT UP OF EFW PENETRATION ABOVE LIMIT DUE LACK P	N	N	302/89-18,	05/29/89	.F.
159	CRYSTAL RIVER	3	LE LC	UNK	ELE STRM LOSS OFF SITE NAT CIRC FOR 2 HRS.	Y	Y	302/89-25,	06/29/89	.F.
180	CRYSTAL RIVER	3	LC LE	EQI	LOSS SDC HT UP TO MODE 4 ALSO VLV FAILED TO OPN FR CNTL RM	Y	Y	302/89-31(S),	08/28/89	.F.
147	DAVIS BESSE	1	OT	LPE	ENTRY INTO MD 6 W/OUT AUD SOURCE RNG MONITOR	N	Y	346/90-03,	02/12/90	.F.
320	DAVIS BESSE	1	IN	UNK	INJ 1K GAL TO VSL LO PRES INJ	Y	Y	346/90-10,	05/18/90	.F.
105	DAVIS-BESSE	1	LC OT	EQI	AFTER TRIP ATTEMPT TO STRT DHR AND SW FAIL DUE TO AIR AND PR	N	Y	IR 346/87-25(AIT),346/87-11, +	09/07/87	.F.
255	DIABLO CANYON	1	OT	EQI	CH VLV IN RHR LEAKING-COOL DN TO FIX THIS RHR OOS	N	?	275/88-17(S),	06/25/88	.F.
293	DIABLO CANYON	2	LC	ADM	DN PATH CAV PMPS SDC LSOT 1 HR28MIN TO 220F	Y	Y	N1410C26,	04/10/87	.F.
211	DRESDEN	2	ES IN	EQI	DURING TEST CR SPRY AND LPCI STARTED LVL INC 40"	Y	Y	237/89-05(S),	02/05/89	.F.
300	DUANE ARNOLD	1	IN ES	EQI	IN CORE SPRY TO FLOODED UP REFL CAV-NOT OVERFILLED	Y	Y	331/88-13,	10/26/88	.F.
133	DUANE ARNOLD	1	LE LC	OPE	DEFUELED-FLOODED UP:DRNG ELEC CALIB-1 D/G OOS-FU POL CL LOST	Y	Y	EN 18859	07/09/90	.F.
281	FARLEY	1	CN	ADM	PERS, EQMT HATCHS OPEN MORE THAN 7 DAYS-FIRE BARRIER PROB	N	?	348/88-11(S),	04/13/88	.F.
4	FARLEY	1	CN	ADM	MAN AT REL VLV 3 S/G AND MANWYS TO 2NDARY SIDE OPEN PATH OPN	N	Y	R2-10/03/89, 348/85-05,IR89-22	10/02/89	.F.
72	FARLEY	2	ES	OPE	TESTING- JUMPER TOUCHED WRONG PLACE SHORTED	N	Y	364/87-06,	11/15/87	.F.
71	FARLEY	2	SP	LPE	RHR RELF VLV OPEN 2500GAL TO PRT PRT RUPT DSK RUPTD		Y	364/87-08, EV FU 87-176,	11/27/87	.F.
199	FARLEY	2	CN	LPE	PACKING FR S/G CHEM INJ VLV WITH S/G MNWAY & HND HLDS OPEN	N	Y	364/89-04(S),	04/20/89	.F.
199	FARLEY	2	ES	LPE	AT RED INV GOT SI SIGNAL	N	?	364/89-05(S),	04/29/89	.F.
204	FARLEY	2	CN	LPE	TS ON CNTM OP AFTER MULTIPLE CNTMT ENTRIES NOT DONE	N	Y	364/89-11(S),	09/25/89	.F.
18	FERMI	2	LL OS ES	LPE	LOSS OF LEVEL DUE TO RHR SYS NOT FULL WHEN STARTED	N	Y	341/87-07,	03/17/87	.F.
68	FERMI	2	LC	EQI	LOSS OF SDC 35 MIN DUE TO FAIL OF MAS TRIP UNIT -ISO SIG BOT	N	Y	341/87-09,	03/25/87	.F.
70	FERMI	2	LC	LPE	SDC ALMOST LOST - 1 MIN ONLY DURING MAINT ON RPS M/G ST	N	Y	341/88-06	02/29/88	.F.
69	FERMI	2	LC	LPE	PM ON MOD POW UNIT CAUSE LOSS CNTL PWR TO RHR VLV-NO RHR 23M	N	Y	341/88-13-1,	03/18/88	.F.
109	FERMI	2	IN OT	LPE	SLC PUMPED INTO VESSEL GOT 20 PPM IN VESSEL	Y	Y	IR 341/88-12, R3 04/11/88,	04/08/88	.T.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE
67 FERMI	2 LC	EQI	LOSS OF SDC RENDERING BOTH TRAINS INOP 37 MIN	N	Y	341/88-16,	04/20/88 .F.
212 FERMI	2 LC LE	EQI	LOSS OFF SITE CAUSE SDC ISO 20 MIN	Y	Y	341/89-03(S),	01/10/89 .F.
241 FERMI	2 LC	LPE	FUEL POOL CLNG LOST-DEFUELED-TRIP OF 2 BUSES	N	Y	341/89-23(S),	09/24/89 .F.
242 FERMI	2 LC ES	UNK	PWR LOST TO RPS BUS B-WHEN PWR RSTRD VLV CLOSED	N	Y	341/89-29(S),	10/23/89 .F.
157 FERMI	2 OT	UNK	FIRE IN DRYWELL DUE TO SMALL CABLE OR EXT CORD	Y	Y	EN 16921,	10/24/89 .T.
146 FERMI	2 OT	PRO	INOP OF EMG SERV WAT..R & EMB EQMT COOLING SYS-FL TO REST REL	N	?	341/89-31,	11/20/J9 .F.
15 FITZPATRICK	1 LE LC OS	EQI	LOSS OF ALL OFF SITE PWR WITH ESFS AND LOSS SDC	Y	Y	333/88-11, N1410-C15,	10/31/88 .F.
301 FITZPATRICK	1 LC	PRO	LOSS SDC 2 MIN NO HT UP OBSVD	Y	Y	333/88-12(S).	11/09/88 .F.
131 FITZPATRICK	1 LC ES	PRO	SDC ISO WHILE STARTING SYS	N	Y	333/90-02,	01/20/90 .F.
116 FITZPATRICK	1 LC	DES	LOSS SDC DUE TO HYDRAU XIEND WHEN STARTING RHR PMP	N	Y	333/90-111,	03/20/90 .F.
316 FITZPATRICK	1 LC	ADM	LOSS OF SDC ON ISO 1 HR - JUMPER FELL OFF DURING MAINT	Y	Y	333/90-13,	04/09/90 .F.
130 FITZPATRICK	1 ES LC	EQI	SDC ISO DUE TO CHATTERING PRESS SWITCH SYS PRES OSC	Y	Y	333/90-16,	04/24/90 .F.
128 FITZPATRICK	1 LC ES	PRO	SDC ISO WHILE STARTING SYS	N	Y	EN 18770,	06/26/90 .F.
8 FORT CALHOUN	1 LE LC	OPE	LOSS ALL OFF SITE PWR FOR 40 MIN LOSS SDC FOR APRPX 5 MIN	Y	Y	285/87-08, N1410-C" C26,	03/21/87 .F.
42 FORT CALHOUN	1 LC	PRO	I&C CAUSED HI PRESS SIGNAL ISOLATED SDC-SDC LOST 5 MIN	Y	Y	285/89-01,	01/08/89 .F.
138 FORT CALHOUN	1 LE LC	UNK	ONE SOURCE OFFST PWR OUT, 1 DG OUT. LOST SDC FOR LT 1 MIN	Y	Y	285/90-06, R4/90-21,	02/26/90 .F.
337 FORT CALHOUN	1 SP	OTH	2K GAL SPLD DUE MONOMTR EFF BETW.. .AV AND SFP	Y	Y	R4/031390,	03/12/90 .F.
96 GRAND GULF	1 OT	EQI	RHR HX OTLT VLV FAILED TO FULLY OPEN	N	N	50.72 15371	04/18/89 .F.
75 GRAND GULF	1 LC	EQI	2 EVENTS DUE TO 1)FUSE REPLMT & 2)LIFTED WIRE DRIG SURV M5/4	Y	Y	416/89-04,	04/20/89 .F.
92 GRAND GULF	1 LE OS	EQI	LOSS 1 DIV (II) OF PWR 4.5 HRS DIV 2 D/G OO" 125 VDC LST	Y	Y	R4/05/03/89,	05/02/89 .F.
327 HADDAM NECK	1 LC LE	UNK	LOSS PWR TO FUEL POOL COOLING 25 MIN NO INC OBS	Y	Y	EN18652,	06/08/90 .F.
12 HARRIS	1 LE OT	MAN	LOSS OF A TRAIN PWR DUE TO BRKR TRIP DUE TO VIBRATION	N	Y	400/87-59-02,	10/11/87 .F.
74 HARRIS	1 LC	EUI	TESTING CAUSES RHR ISO VLV TO CLOSE 2X	Y	Y	400/87-60	10/15/87 .F.
341 HARRIS	1 OT SP	OPE	RCS WATER INTO INST AIR-AT MD LP-25 GAL	N	?	R2/110289,	10/30/89 .F.
189 HARRIS	1 LC	UNK	SPURIOS CLOSE OF SDC SUCT VLV -5F INC	N	Y	400/89-22(S),	12/10/89 .F.
213 HOPE CREEK	1 LC	PRO	ISO SDC DURING SURV FAILED TO INC STP IN PRO	Y	Y	354/89-05(S),	03/01/89 .F.
245 HOPE CREEK	1 LC ES	MAN	1 EPA OPENED DUE TO LO VOLT SIG LOSS SDC-REPEATEDLY OCCURD	N	Y	354/89-22,	10/16/89 .F.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
73 INDIAN POINT	2 OS LE	PRO	TEST CAUSES LOSS OF 480 VOLT ALSO SIMUL LOSS OF INST	Y	Y	247/87-13-02, R1 IR 87-32,	11/05/87	.F.
108 INDIAN POINT	2 OT SP	OPE	SERVICE WATER VLV INAD OPENED CAUSING OUTSIDE TRENCH TO FLD	N	?	R1 11/20/87,	11/18/87	.F.
100 INDIAN POINT	2 LL OT SP	EQI	REACTOR CAVITY SEAL LEAK THRU NI COVER	N	Y	R1 IR 247/89-9,	03/25/89	.F.
53 INDIAN POINT	2 LL OT	PRO	DRAIN DOWN INAD WHILE USE TYGON TUBES TO MEAS LVL	Y	Y	IR 247/89-09,	03/28/89	.F.
165 INDIAN POINT	2 OT	PRO	PRES SPRY LINE OPNED WITH TO HI DELTA T	N	Y	247/89-09(S),	06/26/89	.T.
169 INDIAN POINT	3 ES	MAN	SI SIG NO INJ. UNEVEN FILL OF STM FL MEAS LINES	N	Y	286/89-01(S),	02/04/89	.F.
347 " "	3 OT	UNK						
156 LASALLE	1 SP	UNK	DURING FLOODUP-FLNG CVR ON STM LN (REMOVED VLV) WAS LEAKING	N	Y	R3/11/7/89DAILY	11/02/89	.F.
221 LASALLE	1 ES	ADM	FAIL TO MAINTN VLV DE-ENERG ISO OF SDC DUR TEST-SDC NOT IN S	N	?	373/89-28(S),	12/04/89	.F.
214 LASALLE	2 LC	OPE	SDC ISO DUR SURV. VALVE WAS INADVERT RE-ENERG AND CLOSED	Y	Y	374/89-03(S),	01/28/89	.F.
219 LASALLE	2 LC	EQI	HOLDING RELAY WRONG DURING TEST CAUSE SDC ISO	N	Y	374/89-12(S),	08/29/89	.F.
132 LASALLE	2 ES LC	DES	SDC ISO DURNG SYS START FLOW OSC	N	Y	374/90-03,	03/17/90	.F.
2 LIMERICK	1 LL LS SP	MAN	PACKING FROM WRONG VLV REMOVED. RCS LEAK/DW SPRAY DOWN.L SDC	N	Y	352/89-24,	04/07/89	.F.
51 LIMERICK	1 LL SP	PRO	DROP OF LVL DUE TO VLV OPEN DURING TST WITH CNTMT NOT CLOSED	N	Y	352/89-27-1	04/23/89	.F.
22 LIMERICK	1 OT	EQI	RWCU ISO DUE TO DIF FLOW SIGNAL SYS HAD NOT BEEN FILLED	N	N	352/89-33,	05/13/89	.F.
313 LIMERICK	2 LE LC	EQI	LOSS OF UPS CAUSED LOSS PWR LOSS SDC & RWCU SMOKE AT INVERTR	Y	Y	353/90-07,	03/30/90	.F.
251 MAINE YANKEE	1 ES SI	LPE	DUR PLNT COOLDN SI WHEN OPS FAILED TO BLOCK SI AT SI PRESS	N	Y	309/88-11(S),	12/22/88	.F.
10 MCGUIRE	1 LE OT LC	PER	UNIT 1 BLACK OUT UNIT T 2 TRIP DUE TO TESTING RELAYS	Y	Y	369/87-21, N1410-C8 C27,	09/16/87	.F.
283 MCGUIRE	1 LC	[illegible]S	CNTMT SP TST-RHR CAV-PMP SD 39 MIN INC 90 TO 116F ==RED INV	Y	Y	369/88-49,	11/23/88	.F.
267 MCGUIRE	1 LC LE	ADM	BKR TST:TRN B LOSS PWR RHR PMP LOST AND RESTARTED 4F INC	N	Y	369/88-38(S),	11/29/88	.F.
35 MCGUIRE	2 OS	LPE	BOTH CHARGINE PUMPS TECH. INOP	N	Y	370/86-06,	03/29/86	.F.
160 MCGUIRE	2 OT	OPE	LTOP PROT CKTRY PROB. DISCOVERED APROX 1 YR AFTER WIRING ER	N	?	370/89-05,	07/07/88	.T.
60 MCGUIRE	2 BN	ADM	BOTH CH PMP TECH INOP NO BN PATH	N	Y	370/89-07	08/10/89	.F.
9 MCGUIRE	2 LL OT	PRO	CNTMT SPRAY SYS PRESS TO 325 VS 220 LIMIT SPILL 15000GAL	N	Y	R3-9/6/89,370/89-10,	09/05/89	.F.
97 MILLSTONE	1 OS		REMOVAL OF BUSES FOR INSUL CHKING POTENTIAL ES PMP OR SBGT	N	Y	R1 IR 245/ 89-9,	04/01/89	.F.
20 MILLSTONE	1 LE OS	LPE	LOSS OF NORMAL POWER WHILE MODIFING XFORMER LOGIC - D/G STRT	N	Y	245/89-12, N1410-C16,	04/29/89	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
99 MILLSTONE	1 OS	LPE	SBGT INOP DRNG FUEL MOVE DUE TO ISO OF AIR SUPPLY NO KNOW WH	?	Y	R1 IR 245/89-08,	05/02/89	.F.
78 MILLSTONE	2 LE OS	LPE	LOSS OFPWR TO HALF SYS WITH OTHER HALF OOS.	N	Y	336/88-05,	02/04/88	.F.
79 MILLSTONE	2 LE OS	EQ	LOSS PWR ONE TRAIN DUE TO EQI FAIL	N	Y	336/88-02,	01/19/89	.F.
80 MILLSTONE	2 LE ES	OPE	LOSS OF PWR ON FAC 2 DUE TO WIREING ERROR	N	Y	336/88-03,	01/30/89	.F.
77 MILLSTONE	3 LE LC	ADM	LOSS OF 4160V BUSS DUE TO MECH SHOCK RELAXD REQUIREMTS	N	Y	423/87-38,IR 423/87-24,	11/10/87	.F.
102 MILLSTONE	3 SP	ADM	SWITCH FR RHR OUTPT HT LEG TO VERFY CK VLV RESULTS IN WATER	Y	Y	R3/11/13/87,IR 423/	11/11/87	.F.
34 MILLSTONE	3 OT	ADM	REPLACEMT OF UPPER INTERNALS W/O SRO OR CNTL RM COMO	N	N	423/87-47,	11/30/87	.F.
76 MILLSTONE	3 PR	OPE	LTOPS CHALLENGED WHEN FUSE RENDRD IT INOP - LTOP STUDY	Y	Y	423/88-05,	01/19/88	.F.
190 MILLSTONE	3 ES SI	LPE	SI W/INJECTION TO CORE WHILE AT MID-LOOP	Y	Y	423/89-05(S),	02/17/89	.F.
6 MILLSTONE	3 OS	PRO	SHUTDN ELEC ALIGN D.D NOT MEET T.S - CROSS TIE UNDER RATED	Y	Y	423/89-15,	06/22/89	.F.
191 MILLSTONE	3 OT	LPE	BOTH CHNLS OF SRM HI FLX ALARM BLOCKED TS VIO	N	Y	423/89-23(S),	12/03/89	.T.
143 MONTICELLO	1 LC LL	LPE	PULLED FUSE CAUSE SDC ISO AND OPN OF BYPS LINE CAUS LOSS LVL	Y	Y	263/89-10,	06/20/89	.F.
155 MONTICELLO	1 ES	LPE	CNTME ISO WITH RHR NOT IN SERV-WOULD HAVE ISO	N	N	263/89-03,	10/23/89	.F.
154 MONTICELLO	1 OT	PRO	DURING TESTING LO LO SET REACHED 2 SRVS OPENED	N	Y	263/89-33,	10/25/89	.T.
315 NINE MILE	1 OT	LPE	LOADING QUAD W/ SRM BYPASSED	Y	Y	220/90-01,	01/15/90	.T.
292 NINE MILE	2 LC	OPE	SDC LOST 7 MIN-ALT SDC AVAILABLE IN PROC.	Y	Y	410/88-59,	10/25/88	.F.
14 NINE MILE	2 LE LC	ES EQI	LOSS OF ALL OFF SITE WITH ESFS AND LOSS OR RWCU COOLING	Y	Y	410/88-62, N1410-C15,	12/26/88	.F.
215 NINE MILE	2 LC	ADM	SCD ISO PROC DEF	N	Y	410/89-01(S),	01/22/89	.F.
216 NINE MILE	2 ES IN	LPE	ELE MAINT SURV LPCI AUTO START AND INJECT	Y	Y	410/89-04(S),	02/28/89	.F.
217 NINE MILE	2 LC LE	ES UNK	LOSS ELE GOT SBGT,LOSS SDC, RHR RESTOREDSRV WATER TO ORG CN	Y	Y	410/89-10(S),	03/21/89	.F.
319 NINE MILE	2 LC	PRO	LOSS RPS PWR: SCD ISO RX RECIRC IN PLACE 20 MIN 10F INC	Y	Y	410/90-11,	05/18/90	.F.
291 NINE NILE	2 LC	DES	SDC ISO DURING BACKFIL SENS LINE-ALT SCD AVAIL	Y	Y	41088-07,	02/01/88	.F.
338 NO EVENT HERE	0						03/12/90	.F.
264 NORTH ANNA	1 PR	OTH	PORV OPENED WHEN 2ND RCP WAS STARTED	N	Y	338/88-08(S),	02/02/88	.F.
186 NORTH ANNA	1 LE LC	LPE	TESTING CAUSES LOSS OF BUS AND SDC FOR VERY SHORT PERIOD	N	Y	338/89-06(S),	02/25/89	.F.
187 NORTH ANNA	1 LE LC	OPE	SHORTED LEAD LOST BUS SDC LOST DURING MAINT	N	Y	338/89-10(S),	04/16/89	.F.
103 NORTH ANNA	2 ES IN	PR LPE	OPENING WRONG VLV RESULTS IN HI HD INJ AND PORVs OPENING	N	Y	339/87-13,50.72 10477,	10/26/87	.F.
82 NORTH ANNA	2 LC	OPE	PAINTER CLOSE AIR VLV-CCW COOLING TO RHR HX LOST VLV CLOSED	N	Y	339/89-07,	04/03/89	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE
137 NOT AN EVENT	1						04/19/89 .F.
277 OCONEE	3 LC LE	PRO	MFD BS LST OTHR OOS-SP FU CLNG LOST CR OFF LDD LT 20MN	Y	Y	287/88-04,	08/30/88 .F.
278 OCONEE	3 LC LE	ADM	LS ALL SF BUSES AND SDC DRND TO " MID LOOP 15F/15MN NOTC,INS	Y	Y	287/88-05, N1410-C9,	09/11/88 .F.
344 OCONEE	3 SP	UNK	2K GAL TO CNTMT VLV NOT SHUT RX SPRY SYS	N	?	EVNT BRF 111589	11/15/90 .F.
295 OYSTER CREEK	1 CN	ADM	BOTH DRS OF THP ARLK OPEN GT 6.5H WITH MNWY ON ISO CND OPEN	N	Y	219/88-30(S), (SEC CNTMT)	11/08/88 .F.
16 PALISADES	1 LC LL SP	ADM	WRONG VALVE TEST RESULTS IN LOSS SDC AND SPILL 1000GAL	N	Y	255/87-35, N1410-C27,	10/15/87 .F.
110 PALISADES	1 LC	EQI	FAILED CNTLR CAUSED LOSS OF RHR - TEMP INC =19F	Y	Y	IR 255/87-32,R3 12/22/87,	12/16/87 .F.
248 PALISADES	1 ES	PRO	AUX FD ACTUATION WITH AFW RUNNING AT TIME	N	Y	255/88-07(S),	04/29/88 .F.
275 PALISADES	1 OT	UNK	DURING UPR GD STRU REMOVAL FUEL ASSY LEFT HANG FROM IT	N	Y	255/88-15(s),	09/03/88 .T.
249 PALISADES	1 OT	OPE	CNTNTS OF WST GS DCY TNK RELEASED TS VIO VLV LN UP WRONG	N	Y	255/88-24(S),	12/24/88 .T.
166 PALISADES	1 PR	UNK	EXPD AIR POCKETS TRAPED IN RHR SYS EXPANDED CAUSEING PORVS O	N	?	255/89-02(S),	02/15/89 .F.
167 PALISADES	1 OT SP	MAN	LEAK THRU COLD LG DRN VLVS TO PSDT VLV PROBS	N	Y	255/89-03(S),	02/18/89 .F.
135 PALISADES	1 ES LL OT	LPE	DEPRESS DURING PORV TEST DUE TO VLV CHARACTERISTICS	Y	Y	R3/255/89-33(DRP),	11/21/89 .F.
273 PALO VERDE	1 SP OT	EQI	LEAK IN SDC SYS BOLTS NOT UP TO SPEC	N	?	528/88-22(S),	03/19/88 .F.
274 PALO VERDE	2 ES IN PR	LPE	SIT TANK ONLY INJECTED PZR TO 100%, PRESS TO 250	N	Y	529/88-05(R),	02/21/8[illegible] .F.
57 PALO VERDE	3 OT	AMD	BUBBLES IN TYGON TUBE USED FOR LVL & RHR FLOW MES TOO HI		Y	R5/IR 86-16, R5-4/3/89,	03/30/89 .T.
288 PEACH BOTTOM	2 LC	EQI	BKR TRIP CNTMT ISO-LOST RHR & RWCU LO HT LD	Y	Y	277/88-04,	03/02/88 .F.
306 PEACH BOTTOM	2 LC	ADM	SPUR HI PRES SGNL CAUSE SDC ISO-PLANT SD 16 MTHS	N	Y	277/88-17(S),	03/03/88 .F.
307 PEACH BOTTOM	2 LC	PRO	FLS HI PRESS SDC ISO-RESTORED NO PROB	N	Y	277/88-19(S),	07/29/88 .F.
289 PEACH BOTTOM	2 LC ES	ADM	SDC, RWCU ISO LO DECAY HT LD-SD FOR 18 MONTHS	Y	Y	277/88-21,	09/13/88 .F.
236 PEACH BOTTOM	3 OS PR OT	ADM	HYDRO TST:PRESS TO 1055 SCRAM COMP HANG UP-ALRM,RECRDR OOS	Y	Y	278/89-04(S),	10/23/89 .F.
220 PEACH BOTTOM	3 LC	UNK	FALSE LO LVL CAUSE SDC ISO	N	Y	278/89-08([illegible]	10/27/89 .F.
294 PERRY	1 LC	LPE	DE-ENG RPS GOT NS'4 ISO W/RHR ISO RESET NO PROB	Y	Y	440/88-21,	06/06/88 .F.
93 PERRY	1 ES	OPE	SPKNG ON LPRM CAUSES SCRM TWICE DIV 1 PWR OOS	N	N	50.72 151134,	03/28/89 .F.
119 PILGRIM	1 LC ES	OPE	SDC ISO WHILE CAVITY FULL AND DC HT NR ZERO DUE TO ERROR	N	Y	293/89-039	10/15/87 .F.
238 PILGRIM	1 LE	OTH	17 MO S/D: SNOW,ETC LOST ALL OFF SITE 11 HRS DG S/D PROB	Y	Y	N1410-C15,	11/12/87 .F.
120 PILGRIM	1 LC ES	PRO	SDC ISO DURING TEST POOR JUMPER INSTALLATION	N	Y	293/87-15, 293/89-39	12/07/87 .F.

SHUTDOWN AND REFUEL EVENTS DATABASE

	PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
84	PILGRIM	1 LC ES	PRO	RWCU ISO DURING MAINT	N	N	293/88-04,	02/02/88	.F.
47	PILGRIM	1 OT	UNK	SPURIOUS ATWS SIGNAL TRIPS RECIRC PMPS COLD S/D	N	N	R1-2/4/88,	02/03/88	.F.
23	PILGRIM	1 LL ES LC	LPE	SDS SUCT LINE NOT FILLED PRIOR TO SDC DUE TO MISSED STEP		Y	293/88-25,	12/03/88	.F.
237	PILGRIM	1 LE ES	EQI	LOSS PRFD ELE, DG STRT, 1,2 CNTMT ISO, RWCU ISO CABLE DAMAGE	J	Y	293/89-10(S), N1410-C15,	02/21/89	.F.
118	PILGRIM	1 LC ES	PRO	RHR ISO DURING STARTING OF SYS IN S/D PROCESS PROB DU FLO OS	Y	Y	293/89-39,	12/09/89	.F.
312	PILGRIM	1 LC	UNK	LOSS SDC 30 MIN DUE TO FAIL TO XFER PWR PROPERLY	N	Y	293/90-05(R),	03/20/90	.F.
129	PILGRIM	1 LC ES	PRO	SDC ISO WHILE STARTING SYS DUE TO PRESS SURGE WHEN STARTING	N	Y	EN 18823,	07/03/90	.F.
335	POINT BEACH	1 OS CN	ADM	AUTO CLS SIG FOR CNTMT PURG & EXHAUST TAGGED OUT FUEL MV ON	N	Y	EN/12145,	04/28/88	.F.
141	POINT BEACH	1 OT	DES	LTOP INOP USING N-2	N	?	266/89-05,	05/03/89	.T.
324	POINT BEACH	1 OT	UNK	F/A STUCK DAMAGED DURING MOVMT-TORN PIN BENT	Y	Y	EN18452,	04/21/90	.T.
325	POINT BEACH	1 PR	UNK	FILLING SI ACCUM GOT PORV OPEN FOR LOTP	N	Y	EN18452,	05/12/90	.F.
39	POINT BEACH	* LE OS	EQI	ESF/TRIP WHEN BATTERY REMOVED FROM SERVICE	N	Y	266/87-04	05/15/87	.F.
81	PRAIRIE ISLAND	2 LC OT	OPE	WORK RESULTS IN INCREASED RHR FLOW, RHR PMPS ARE TRIPPED	Y	Y	R3/01/ /88, IR 282&306/88-01,	01/15/88	.F.
178	PRAIRIE ISLAND	2 CN	LPE	STM PIPING OPEN-VLVS IN AUX BLDG OPEN PATHS CREATED	N	Y	282/89-04(S),	04/20/89	.F.
83	RANCHO SECO	1 LC	EQI	DROP LINE ISO PRESS SENSOR OUT OF CAL CLOSED VLV	N	Y	312/89-02,	02/28/89	.F.
182	RANCHO SECO	1 LC OT OS	LPE	SDC OOS LONGER THAN 1 HR TS VIO	N	Y	312/89-09(S),	07/25/89	.F.
339	RANCHO SECO	1 LC	ADM	LOSS SDC 10 MIN NO INC TEMP	N	N	312/89-11,	10/21/89	.F.
152	RIVER BEND	1 LC ES	PRO	SDC ISO DURING TEST 2 MIN ISO FU PO COOL, RWCU AVAILABLE	N	Y	458/87-22,	10/16/87	.F.
27	RIVER BEND	1 OS OT	UNK	FAILURE OF DIV 2 D/G WITH DIV 1 D/G DISASSEMBLED-NO ON ST AC	N	Y	458/87-25,	10/21/87	.F.
229	RIVER BEND	1 ES LC LE	LPE	LOSS OF PWR TO DIV 2 ISO LOGIC OP OPEND BKR 12 MIN SDC LOS	N	Y	458/89-12(S),	03/25/89	.F.
153	RIVER BEND	1 LC	OPE	IMPROPER JUMPER PLACEMT FELL OFF FUSE BLEW TEMP INC 6DEG	Y	Y	458/89-15,	03/29/89	.F.
91	RIVER BEND	1 SP LC LE	ADM	FAILURE OF FRZ SEAL ON 6" SW LINE W/ SW IN SERV SP 15K GAL	Y	Y	R4IR89-20(AIT), IR 89-24,89-20	04/19/89	.F.
230	RIVER BEND	1 LC ES	UNK	TEST CONN SHORT CAUSE PWR XIENT AND ESP TRIP LOGIC GOES	N	Y	458/89-21(S),	04/27/89	.F.
44	RIVER BEND	1 LE LC ES	ADM	SHORTED LEADS CAUSE ESF INC ISO INC LOSS OF RHR FOR 24 MIN	Y	Y	458/89-29,R4-6/14/89,R489-2d,N	06/13/89	.F.
314	RIVER BEND	1 LC	EQI	LOSS OF SDC NO INC TEMP	N	Y	458/90-11,	03/26/90	.F.
26	ROBINSON	2 OT	PER	H-2 IN PLANT AIR SYS-FLAMMABLE MIX VLV LINEUP PROB	N	Y	261/89-01,	01/07/89	.T.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
345 ROBINSON	2 PR OT	LPE	SOLID 350#,160F 2 RCPS OP PRESS CNTL VLV ISO AND LTOP ACTIV	Y	Y	LTR RNPD/89-3661-1107B(,	10/16/89	.F.
28 SALEM	1 LL OT	PRO	900 GAL SPILL DUE TO LEVEL MEAS AND PROC PROB	Y	Y	R1/IR 87-28,	10/09/87	.F.
332 SALEM	1 SP OT	OPE	2 VLV REALY OPN-FLNGS LOSE. SRVC WAT BAY FLOODED	Y	Y	SP REPT 87-10,	12/22/87	.F.
33 SALEM	1 LC	LPE	LOSS OF RHR TEM INC. APRX 30DEG DUE TO N-2 FROM ACCU	N	Y	272/89-19,	05/20/89	.F.
104 SALEM	2 LE	PRO	POTENTIAL LOSS OF 125 V DC BATTS DUE TO WRONG CORRECT FTR UD	N	?	50.72 10923, IR 311/87-36,	12/09/87	.F.
181 SALEM	2 LC	UNK	DROP LN ISO-LOSS SDC DUE HI RCS PRESS LOST 10 MIN	N	Y	312/89-02(S),	10/24/89	.F.
337 SALEM	2 PR	EQI	RHR SUCT PIPE PRESS TO 150 OVR 450-RHR INJ VLV FAIL TO SEAT	Y	Y	311/89-[illegible],	10/27/89	.T.
144 SALEM	2 ES OT	PRO	MSIV ISO CLOS OF ONLY VLV FOR STM TO CONDNSR-ATMS DMPS USED	Y	Y	311/90-07,	01/17/90	.F.
192 SEABROOK	1 LC	PRO	WHILE REST PWR LOST ONE/2 TRNS OF RHR SCT VLV CLOSED B OUT M	N	Y	443/89-12(S),	09/05/89	.F.
346 SEABROOK	1 LC OS	PRO	LOSS SDC DUE LOSS CNT PWR TO ISO VLV 50MIN NO INC TEMP	N	Y	443/89-12,	10/11/89	.F.
330 SEQUOYAH	1 LC	ADM	XOCER PMPS RUN BOTH VORTX LOST SDC 43MIN	Y	Y	N1410-C24, IN86101, OMR295,	10/09/85	.F.
45 SEQUOYAH	1 LC LL OT	EQU	USE OF SITE GLASS FOR LVL WHILE DRAINED CAUSES LOSS SDC/SPIL	Y	Y	327/87-12-1,R2/IR 87-24, N1410	01/28/87	.F.
49 SEQUOYAH	1 SP	LPE	OPEN PATH FROM RWST TO RCS-SPILL 3000GAL	Y	Y	327/87-13-1,R2/IR 87-24,	02/01/87	.F.
50 SEQUOYAH	1 OT SP	ADM	SPILL WHILE FILLING AND PRESSURIZING DUE TO OPEN VLV	N	Y	R2/IR 87-24,	04/29/87	.F.
36 SEQUOYAH	1 OT	OPE	WRONG AND INAD PROCEDURES RELATIVE TO T.S.	N	Y	327/87-74,	12/22/87	.F.
262 SEQUOYAH	1 LC LL	OPE	RED INVT:B RHR LOST-DIVT FL TO RWST LVL DROP PMP LOST 6KGAL	Y	Y	327/88-21,	05/23/88	.F.
106 SEQUOYAH	2 LC	EQI	RHR LOST FOR 3 MIN DURING SURV NO DEC HT SO NO TEMP INC	N	Y	50.72 10810,	11/28/87	.F.
184 SEQUOYAH	2 ES	ADM	RX TRIP DUE TO SI SIGNAL DUE TO ADM	N	?	328/89-02(S),	03/25/89	.F.
59 SEQUOYAH	2 OT	PRO	FUSES RENDER ONE SAFETY CKT INOP FOR 1.5MO D/G WK TOOK PLACE	N	Y	328/88-38,	09/29/89	.F.
218 SHOREHAM	1 LC ES	OPE	DURING TESTING GOT SDC ISO	N	Y	322/89-06(S),	07/07/89	.F.
194 SONGS	1 BN	LPE	CAVITY BEING WASHED WITH UNBOR WATER IN OP BUS AND INOP SRM	Y	Y	206/89-02(S),	01/23/89	.F.
297 SONGS	2 LC	ADM	LVL BE LWRED PMP CAV. 2ND PMP CAV. LVL INC. 70MIN/TO 210F BL	Y	Y	N1410-C25,	03/26/86	.F.
282 SONGS	2 SP	DES	9K GAL DRND FR SPNT FL PL TO RX CAVITY-VIO DESIGN	N	Y	361/88-17(S),	06/22/88	.F.
197 SONGS	2 CN	LPE	3/4 IN HOT LG VNT DRN VLVS OPEN VENT TO OUTSIDE DUR CORE ALT	N	Y	361/89-22(S),	06/17/89	.F.
198 SONGS	2 OS	N/A	BOTH TRNS EMG CHILLERS INOP VOLUNTRY INVOLVS UNIT 3 ALSO	N	?	362/89-09(S),	09/07/89	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
340 SONGS	2 SP LL	OPE	100 GAL DRN FRM RCS TO RWST AT MID LOOP - WRON VLV OPND -21N	Y	Y	R5/11/02/89,	11/02/89	.F.
247 SONGS	3 LC	ADM	MD LP LVL INDICATION PROBS LOSS SDC	Y	Y	N1410-C28,	07/07/88	.F.
107 SOUTH TEXAS	1 SP OT	?	PIPE RUPTURE IN AFW SYS CAUSED BY WATER HAMMER	N	Y	IR 498/87-71, 50.72 10766,	11/22/87	.F.
85 SOUTH TEXAS	1 OT	LPE	SEC WATER GT 50F ABOVE RCS AND STARTED RCP	N	Y	498/88-01,	01/02/88	.T.
272 SOUTH TEXAS	1 OT	DES	HI PT IN PIPE FR RWST TO CH SYS MEANS AIR BINDING A CT HTS	N	N	498/88-??(S),	05/17/88	.F.
322 SOUTH TEXAS	1 SP	OPE	SPL PIECE NOT INSTALLED-DRAIN PATH 17K GAL TO CNTMT	Y	Y	EN18186,	04/09/90	.F.
163 SOUTH TEXAS	1 OT	ADM	WHILE COLD VESSEL PRESSZD W/OUT LOTP	N	?	R4/5/3/90DAILY,	05/02/90	.T.
201 ST LUCIE	2 OT SP	LPE	FL POOL LVL HI ENGH TO FLOOD INTAKE VENTS-VENT SYS INOP	N	Y	389/89-01(S),	02/22/89	.F.
268 SUMMER	1 IN	OPE	TEST CAUSED SI 6 MIN	N	Y	395/88-13(S),	12/11/88	.F.
256 SURRY	1 OT	OPE	PORV CYCLED WHILE SOLID DUE TO CONSERVATIVE SET PT	N	Y	280/88-21,	07/28/88	.F.
276 SURRY	1 SP	ADM	REFL CV SL RING LK 25,800 GAL DU CLOSING AIR VLVCR OFFLDD	Y	Y	280/88-30(R),	09/01/88	.F.
173 SURRY	1 LE LC	EQI	FAILED CKT BKR CAUSE LOSS ELS,RHR	N	Y	280/89-03(S),	02/04/89	.F.
174 SURRY	1 LC	LPE	VLVS CLOSED IN RHR, CCW SYSS. NO RHR FOR PERIOD OF TIME	N	Y	280/89-09(S),	03/18/89	.F.
175 SURRY	1 LE ES	OPE	BKR CHK IN S/YCAUSE LOSS 230KV AND EMEG BUSES NO LOSS SDC	N	N	280/89-13(S),	04/13/89	.F.
176 SURRY	1 BN	MAN	INLEAK OF WATER CAUSE BN REDUCT. RCP STNDPIPE MAKUP WATER	N	Y	280/89-16(S),	04/23/89	.F.
5 SURRY	2 CN	PRO	INSIDE AND OUTSIDE SI TRIP VLVS OPEN - PATH OUT	N	Y	281/81-78,	11/27/81	.F.
257 SURRY	2 OT	PRO	PORV CYCLD WHILE SOLID WHEN RCP STARTED 375PSIG	N	Y	281/88-14(S),	05/28/88	.T.
177 SURRY	2 BN	LPE	SLOW DIL DUE TO IMPRO BLNDR SETTING AND DICS NOT TO CLOSE CN	Y	Y	281/89-15(S),	10/25/89	.F.
168 SURRY	2 OT SP	UNK	LOOP SEAL PROB CAUSE PRZ SAFE VLV TO OPEN AT 2335 DN TO 2255	N	Y	281/89-17(S),	11/06/89	.F.
113 SURRY	* LE LC	EQI	FAILED LT ARR CAUSES SW YD EVENT LOSS OF EMEG BUS	N	Y	280/89-10, N1410-C9,	04/06/89	.F.
117 SUSQUEHANNA	1 LC ES	PRO	RHR SD COOL ISO DURING SWITCHING PMPS PROB DUE FLOW OCSLNS	Y	Y	387/84-20,	03/21/84	.F.
127 SUSQUEHANNA	1 ES LC	PRO	SDC ISO WHILE SWAP PMPS	N	Y	387/87-028,	09/13/87	.F.
125 SUSQUEHANNA	1 ES LC	PRO	2 ATMPTS TO PLACE RHR IN SERV FAIL 3RD OK FLOW OSC	Y	Y	387/88-11.	06/02/88	.F.
124 SUSQUEHANNA	1 LC ES	PRO	SDC ISO WHILE SWITCH PMPS 2 EVNTS 2ND WATER HAMR	Y	Y	387/89-03,	01/07/89	.F.
222 SUSQUEHANNA	1 LC ES	OPE	FUSE BLEW-SDC ISOLATION FOR 1 HR	Y	Y	387/89-15(S),	05/21/89	.F.
162 SUSQUEHANNA	1 LC LE	EQI	TEMP RISE TO 253F SDC LOST DUE TO LOSS OF RPS BUS AND ISO	Y	Y	EN 17688, 387/90-05,	02/03/90	.F.
123 SUSQUEHANNA	2 ES LL	PRO	OPENING RHR CAUSED LVL DRP/WTR HAMR 2 EVNTS (NO SDC ISO)	Y	Y	388/85-16,	04/27/85	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

	PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
122	SUSQUEHANNA	2 LC ES	PRO	'/C ISO SWAP PMPS 30 F RISE - WATER HAMMER	Y	Y	388/86-15-1,	10/12/86	.F.
126	SUSQUEHANNA	2 ES LC	PRO	RHR ISO DURING PREP TO PUT RHR INS SRV	Y	Y	388/88-03,	03/05/88	.F.
303	SUSQUEHANNA	2 LC ES	EQI	LOSS SDC RWCU DURING TRIP OF PWR SUPPLY TO A RPS	Y	Y	388/88-05,	03/19/88	.F.
279	TMI	1 IN OT	PRO	[illegible] INJT TST GOT PRESS ..C AND TEMP DECREASE	N	Y	289/88-02,	06/19/88	.F.
250	TMI	1 OT	LPE	EXCEEDED HEAT UP RT DURING STARTUP	N	Y	289/88-05(S),	09/25/88	.T.
265	TROJAN	1 PR	PRO	SOLID-PRES INC 305 TO 430P LETDN ISO AS PART OF TEST-PORV OP	N	Y	244/88-20(S),	06/30/88	.F.
56	TROJAN	1 OT		LOSS OF N-2 SUP TO [illegible] LOOP RESULTS IN DECLR RHR TN INOP	N	Y	50.72 # 15621,	05/15/89	.T.
139	TURKEY POINT	3 LE	?	BKFDING LOST ELE DG START XFER TO UNIT 4	N	Y	N1410-C7,	04/29/85	.F.
121	TURKEY POINT	3 LE	OTH	FIRE NR SUB STA LOSS OF 500KV LN OFF ST LOST DGS RAN 2HRS	Y	Y	N1410-C7,	05/17/85	.F.
161	TURKEY POINT	3 OT	DES	LTOP PROT PROB-PORV OPEN TIME TOO LONG	N	?	250/88-21,	09/13/88	.T.
254	TURKEY POINT	3 LC	OPE	CHN CLB OF BAST LVL SHORTCKT RHR SUCT ISO VLV CLOSE "5 MIN	Y	Y	250/88-29(S.,	11/17/88	.F.
171	TURKEY POINT	3 OT OS	ADM	B DG OOS, 3A INK CL WATER PMP OOS IF LOOP, NO INK WATER	Y	Y	250/89-01(S),	01/10/89	.F.
172	TURKEY POINT	3 BN OS	ADM	2 HT TRACE CKTS BATS OOS SAME TIME	N	Y	250/89-08(S),	03/20/89	.F.
114	TURKEY POINT	4 OS	LPE	AIR TO AIR ST SYS OF D/G ISO D/G WOULD NOT HAVE STARTED	Y	Y	251/89-01,	02/09/89	.F.
90	TURKEY POINT	4 OT VD	EQI	N-2 VOID FORM IN UPPER HEAD SEVERAL TIME 10/21-11/03	Y	Y	R2 IR 87-46, R2/10/26/87,	10/21/89	.F.
40	TURKEY POINT	* OS OT BN	EQI	LOSS OF BORIC ACID FLOWPATHS TO BOTH UNITS	N	Y	250/87-17,	05/28/87	.F.
243	VERMONT YANKEE	1 LE	OTH	OFF-SITE GRD DISTB CAUSED BKR TO OPN LOSS OFF - BK AVAIL DGS	N	Y	N1410-C14,	08/17/87	.F.
52	VERMONT YANKEE	1 OT OS	LPE	T.S. PROBLEM DUE TO SYS OUT OF SERV	Y	Y	271/87-11,	08/20/87	.F.
13	VERMONT YANKEE	1 LL	PER	RX VSL LVL DECREASE DUE TO PERS ERROR - DROP OF 72 IN	Y	Y	271/89-13,	03/10/89	.F.
98	VOGTLE	' ES OT	OPE	BIT INFLOW OF 600 GAL @ 125F INTO PZR @ 415F PZR LVL FR 51 T	Y	Y	R2 C2/05/88,	02/03/88	.F.
205	VOGTLE	' OS	ADM	A TN CHILLER OUT FOR MAINT B TRN INADV TAKEN OUT FOR CALB	N	Y	424/89-03(S),	01/19/89	.F.
206	VOGTLE	1 OT	ADM	DEPRES OF RHR W/OUT PROC-TEST RET LINE VLVS OPEN 14 HRS	N	?	425/89-03(S),	10/02/89	.T.
145	VOGTLE	1 LC LE	OPE	LOSS OF ELE CAUSES LOSS SDC AT MID LOOP-1IT	Y	Y	424/90-06, N1410,	03/20/90	.F.
207	VOGTLE	2 IN ES	LPE	WRONG SWITCH-SI 2900 GAL INJ	N	Y	425/89-06(S),	03/18/89	.F.
95	WATERFORD	3 LE	?	LOSS OF OFF SITE FOR 33 MIN DGS WORKED	N	Y	N1410-C7	12/21/85	.F.
296	WATERFORD	3 LC LL	ADM	RCS DNED 2 PATHS. FAIL TO STP 2ND PATH 2HR 15 MIN TO 232F	Y	Y	382/86-15, N1410-C25,	07/14/86	.F.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE	
286 WATERFORD	3 LC	ADM	LVL BE LWRD: LOST SDC CAVITATION LVL RAISED	Y	Y	N1410-C27,	05/12/88	.F.
342 WATERFORD	3 OS	OTH	1 DG OOS THE OTHER FAILED IMPURITY LMTS	N	?	EN16982,	10/31/89	.F.
290 WNP	2 LL SP	LPE	RHR & SUP LINED UP TO VESSEL PATH TO SUP POOL 10K GAL TO SP	Y	Y	397/88-11,	05/01/88	.F.
304 WNP	2 LE LC	ES ADM	FLDED UP-LOSS OF PRS PWER CAUSE SDC ISO	Y	Y	397/88-15,	05/15/88	.F.
305 WNP	2 LC	LPE	FLDED UP-SDC ISO 20 MIN	Y	Y	397/88-21,	05/30/88	.F.
148 WNP	2 LC LE	UNK	2 EVENTS:ELE XIENT CAUSE LOSS OF ELE TO 120VAC VLVS CLOSED	Y	Y	397/89-10,	05/01/89	.F.
149 WNP	2 LC ES	EQI	2ND EVENT: LOSS ELE CAUSE ESFS VLVS CLOSE	Y	Y	397/89-10, R5 5//89	05/03/89	.F.
43 WNP	2 LE LC	OS LPE	WRONG FUSE PULLED RESULTS IN ESFS AND LOSS OF SDC	Y	Y	397/89-16, N1410-C16,	05/14/89	.F.
223 WNP	2 LC	OPE	8 MIN LOSS SDC CONTRACTOR LIFTED WIRE	Y	Y	397/89-19(S),	05/24/89	.F.
224 WNP	2 LC	UNK	3 HR LOSS OF SDC DUE	Y	Y	397/89-20(S)	05/27/89	.F.
225 WNP	2 LC LE	UNK	EPA BKR TRIPD LOSS OF PWER TO RPS CAUSE ISO +	N	Y	397/89-21(S),	05/27/89	.F.
226 WNP	2 CN	ADM	2 EV: 2-CNTMT:MSIC OPEN IN STM TUN. IN RX BLDG TURB THVLV OP		Y	397/89-22(S),	05/30/89	.F.
227 WNP	2 LC ES	ADM	EPA BKR TRIP-ESFS SDC ISO-RECIRC PMP STARTED W/OUT REV	N	Y	397/89-23(S),	05/31/89	.F.
228 WNP	2 LC	PRO	SCD ISO DURING EXCS FL VLV TESTING HI PRES SIG	N	Y	397/89-25(S),	06/17/89	.F.
317 WNP	2 LC ES	UNK	3 SIMLR EVNTS:LOSS RPS BUS RHR ISO RWCU OOS-MAINT	Y	Y	397/90-09,	05/30/90	.F.
318 WNP	2 ES LC	UNK	RPS PWR LST:SDC &RWCU ISO 13F INC 40 MIN	Y	Y	397/90-13,	06/10/90	.F.
32 WOLF CREEK	1 OT	ADM	PRESS. H-2 BURN DUE TO H-2 ACCUMU IN PZR AND WELDING	N	Y	R4/IR87-27(P6),	10/14/87	.T.
30 WOLF CREEK	1 LE LC	ES LPE	DEATH, LOSS OF SDC, LOSS OF ELEC		Y	482/87-48-1, IR87-27 (P9)	10/14/87	.F.
31 WOLF CREEK	1 LE ES	LC LPE	LOSS OF DC CAUSES ESFS-LAKE WATER (ESS SER WAT) PMP'D TO S/G	N	Y	482/87-49, R4/IR87-27, N1410C9	10/15/87	.F.
101 WOLF CREEK	1 OT	?	D/G GOV SETTINGS WRONG	N	N	IR 482/87-33,50.72 10921,	12/09/87	.F.
54 WOLF CREEK	1 OT	LPE	HD OFF CHKING ACCU INADVERT OPEN ACCU VLV AND N-2 TO RCS	N	Y	R4/IR 88-37, R4 11/21/88,	11/21/88	.F.
55 WOLF CREEK	1 BN OS	OT LPE	NO BORATION PATH WHILE CORE ALTERATIONS ON-GOING	N	Y	482/88-25(S),R4/IR 88-37,	11/23/88	.F.
164 WOLF CREEK	1 PR	EQI	CH PMP SPD CNTL PROB-PRESS DAMAGE TO DN STM RELF VLV	N	Y	R4/5/3/90DAILY,	05/02/90	.F.
7 YANKEE ROWE	1 LE	?	LOSS OF PWR DUE TO OPN BKRS IN WRONG SEQ	N	Y	N1410-C8,	06/01/87	.F.
150 YANKEE ROWE	1 OT SP	OPE	WRONG TEST PLUG-PORV OPENS AT OP TEMP,PRESS	N	Y	029/88-07,	04/30/88	.T.
151 YANKEE ROWE	1 LE	PRO	E BUSES LOST POWER DURING TEST OF GENERATOR	N	?	029/88-10,	11/16/88	.F.
3 ZION	1 CN	ADM	AFW CK VLV IN CNTMT AND AT RL VLV OUT REMOVED - PATH OUT	N	Y	R3-9/22/89	09/21/89	.F.
158 ZION	1 OT	UNK	FIRE IN CH PMP BEARING SHORT FIRE -DEFUELED	Y	Y	R3/102489DAILY	10/23/89	.F.
323 ZION	1 OT	EQI	VNT VSSL AND PZR HDS TO VNT RCS-RAD REL UNMON	Y	Y	EN18215,	04/11/90	.T.

SHUTDOWN AND REFUEL EVENTS DATABASE

PLANT	CATEGORIES	CAUSE	DESCRIPTION	P	U	REFERENCES	DATE
329 ZION	2 LC	ADM	LEAKS LOST RCS LVL VORTXING SDC LSOT 75 MIN/15F INC	Y	Y	N1410-C24, IN 86101, SOER88-3	12/14/85 .F.
261 ZION	2 EN	LPE	TESTING RHR CK VLV: 3500G INJTD PRESS S/B 1800 PRESS WAS 900	N	Y	304/87-06,	07/29/87 .F.
260 ZION	2 IN PR	LPE	TESTING: INADV SI BKRS CLOSED OUT OF SINC FORV OPND	N	Y	304/88-12(R),	12/11/88 .F.
263 ZZ NO EVENT HERE	0						01/01/01 .F.

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Attachment 1

Low Power and Shutdown Events

The attached table provides 65 candidate shutdown and low power events for re w and consideration for further analysis. These candidates were obtained from the AEOD shutdown events compendium, from previous zero and low power ASP cases, and from previous AEOD studies. Those obtained by foreign event review are found in a separate table (Attachment 2).

The groupings of events and their order in this table are:

Loss of Cooling
Loss of Reactor Coolant,
Flooding,
Pressurization,
Containment Integrity,
Loss of Electrical Power,
Reactivity Events,
Mode Entry Events, and
Other events which do not fit the above categories.

Within these categories, the events are ordered by reactor type (e.g., PWR followed by BWR) and by date.

SHUTDOWN EVENTS

Loss of Cooling

PLANT	DATE	DESCRIPTION
Millstone 2	12/09/81	In mode 5, SDC pump tripped due to testing and RCS temperature rose 118 °F to 208 °F in about *2 minutes*. Heatup and cooldown limits were exceeded. Mode 4 was entered without satisfying Mode 5 LCOs. Appendix G analyses were preformed indicating design basis not downgraded.
Crystal River 3	02/02/86	At mid-loop, RHR pump shaft broke after continuous operation for about 30 days. Placing second RHR pump in operation was delayed because a tripped breaker was powering the suction valve. Temperature rose 33 °F to 131 °F. (Ref. 9)
Waterford	07/14/86	Near mid-loop, a drain path was not closed and level dropped. Operating RHR pump began to cavitate and was shutdown. RHR lost several times over a 3 and one-half hour period. RCS temperature rose 94 °F to 232 °F.
Diablo Canyon 2	04/10/87	At mid-loop, 7 days after shutdown with head vented, a leak caused loss in RCS level and the RHR pump became air bound. RHR was lost for 1 hour, 28 minutes; RCS temperature rose 87 °F to 220 °F and pressure increased to 10 psig.
Arkansas 1	12/19/88	Relay problem caused RHR suction valve in single suction line to close. RHR lost for 12 minutes; RCS temperature rose 12 °F to 147 °F.
Brunswick 1 & 2	04/17/81	A total Unit 1 loss of RHR occurred when the Unit 1 RHR heat exchanger baffle plate failed allowing service water to bypass the RHR HX tubes. Alternate cooling path through spent fuel pool HXs was used. The baffle failure was caused by excessive differential pressure across the baffle when a second RHR pump was started due to reduced cooling due to marine organisms blocking HX tubes.

Losses of Cooling (continued)

PLANT	DATE	DESCRIPTION
Fitzpatrick	10/31/88	Loss of electrical power caused loss of RHR. Temperature rose 10 °F
Susquehanna 1	02/03/90	After test, RHR could not be restored due to short circuit. RWCU also isolated. Temperature rose to 253 °F and pressure to 20 psig. RHR restored after about 2 and one-half hours.
Brunswick 2	01/16/82	At power, turbine trip and subsequent reactor scram occurred due to loss of vacuum. MSIV closure occurred due to steam flow sensor problem. RCIC was started and operators attempted to start suppression pool cooling but a low suction header pressure signal would not allow start of RHR service water. Signal was caused by sediment in the sensing line and a failed sensing switch. Power supply for one loop's pressure sensor was also turned off. The MSIVs were opened to allow condenser cooling.
Salem 1	03/16/82	In mode 5, vital bus was de-energized causing loss of power to 1 CCW pump and 2 service water (SW) pumps. The other CCW and service water pumps were out of service for maintenance. Thus, a complete loss of CCW and service water resulted. The immediate effect was to cause all charging pumps, boron injection paths, RHR trains, and D/Gs to be inoperable. Vital bus was restored within 1 hr restoring SW. CCW restored 2 hrs later.
Palisades	01/08/84	De-fueled, with both the main transformer and one of two D/Gs out of service, a problem developed on an incoming electrical feeder. Since one D/G was out of service, the plant should not have disconnected from the problem feeder. However, the disconnect was made and the plant was powered by one D/G. When this was done, all SW was lost because one SW pump was out of service and the other 2 SW pumps were powered by the inoperable D/G. Lack of SW caused loss of running D/G and loss of all AC.

Loss of Reactor Coolant		
PLANT	DATE	DESCRIPTION
Haddam Neck (also under flooding)	08/21/84	The refueling cavity seal failed and about 200,000 gallons of reactor coolant spilled into the containment in about 20 minutes. Containment sump overflowed and level in containment reached about 18 inches. It was postulated that for a worst case scenario, that the fuel pool could have been drained. The fuel transfer canal was not in use and the fuel pool was isolated from the cavity at time of the event. NRC Bulletin 84-03 was issued.
Surry 1	05/27/88	About 25,800 gallons of reactor coolant were drained from the reactor cavity due to leaking reactor cavity ring due to isolation of the instrument air for maintenance.
Palisades	11/21/89	Due to pressurizer PORV design characteristics, when the PORV block valve opened, the PORV also opened and RCS pressure dropped from 2154 to 1565 psi.
Braidwood 1	12/01/89	At 350 psig, approximately 67,000 gallons of reactor coolant was lost when a RHR pump suction relief valve lifted. About 2 and one-half hours was required to isolate the leak.

Loss of Reactor Coolant (continued)

PLANT	DATE	DESCRIPTION
Grand Gulf	04/03/83	While aligning RHR system for LPCI standby after a surveillance test. a failed "open" indication led operators to believe the SDC suction valve was closed. It was still partially open. Suppression pool suction valve was opened establishing a drain path and vessel level decreased about 50 inches and an isolation occurred stopping the drain down.
LaSalle 1	09/14/83	A testable isolation check valve stuck open and a manual valve down stream of the check valve was inadvertently left open. The RHR injection valve was opened completing a path from the recirculation line to the suppression pool. The vessel level dropped about 50 inches and an isolation occurred stopping the drain down. Although this isolation function was in service, *it was not required to be operable in modes 4 and 5*. Although most of the coolant entered the suppression pool, some entered the drywell via an open drywell spray line.
LaSalle 2	03/08/84	Due to testing and an inadvertently open breaker supplying power to an isolation valve, RHR pump was operating at shutoff head conditions. The minimum flow valve opened and coolant was pumped by the RHR pump to the suppression through the minimum flow line. Level decreased about 60 inches. The event was stopped by operator action and would have been automatically stopped by a low level isolation had it been required.

Loss of Reactor Coolant (continued)

PLANT	DATE	DESCRIPTION
Washington Nuclear Project 2	08/23/84	A shutdown cooling suction isolation valve closed due to a high flow signal. However, the closure was not alarmed in the control room. The closed suction isolation valve allowed water in the RHR line to drain to the radwaste system. When the isolation valve was reopened, reactor coolant filled the emptied line and level dropped about 25 inches to the low level isolation set point.
Hatch 2	05/10/85	The vessel level was above the main steam lines due to refueling. An inadequate procedure allowed the ADS 2 minute timer to elapse and 7 ADS valves opened for about 17 minutes and water drained through the main steam line to the safety-relief valve discharge line to the suppression pool. Vessel level dropped about 42 in.
Peach Bottom 2	09/24/85	To perform a test on one loop of RHR, the full flow test return valves were opened for the test. The operations staff did not know that the RHR suction valve in another loop had been inadvertently opened. When the full flow test return valves were opened, a path was created from the vessel to the suppression pool. The level decreased to the isolation set point which terminated the loss of level.
Washington Nuclear Project 2	05/01/88	During transfer of SDC from one loop to the other, valve closure required about 120 seconds and this resulted in two valves being open at the same time. This caused a leak of about 10,000 gallons from the vessel to the suppression pool.

Loss of Reactor Coolant (continued)

PLANT	DATE	DESCRIPTION
Vermont Yankee	03/09/89	With the refueling cavity flooded, valve alignment in the RHR system established a gravity drain path from the vessel to the suppression pool. About 54 minutes after the start of the event, the leakage path was identified and the leak was stopped. About 10,300 gallons of coolant were drained to the suppression pool.

Flooding		
PLANT	DATE	DESCRIPTION
Haddam Neck (also under loss RCS coolant)	08/21/84	The refueling cavity seal failed and about 200,000 gallons of reactor coolant spilled into the containment in about 20 minutes. Containment sump overflowed and level in containment reached about 18 inches. It was postulated that for a worst case scenario, that the fuel pool could have been drained. The fuel transfer canal was not in use and the fuel pool was isolated from the cavity at the time of the event. NRC Bulletin 84-03 was issued.
Salem 1	12/22/87	In mode 5 for refuel outage, with two blank flanges installed on service water piping -- one where a valve was removed and one at another location, a leak occurred. Flanges had been loosened after original installation to provide a necessary leak path. Due to valve failure and flanges being loose, a large amount of service water (about 144,000 gal.) entered one service water bay. The water level was above the service water pump motors in the bay. Some electrical equipment and the Cardox Room outside the bay were wetted.
McGuire 2	09/05/89	About 10,000 gallons of water entered auxiliary building when one train of containment spray was overpressurized and the bottom flange gasket of a heat exchanger failed.
South Texas	04/09/90	While the reactor cavity was being filled, one of two required spool pieces was not installed. This created a drain path from the lower equipment storage area to the containment and 17,000 gallons of water entered containment.

Flooding (continued)		
PLANT	DATE	DESCRIPTION
Clinton	03/20/89	About 40,000 gallons of water were gravity drained from the containment refuel pool to the drywell. Equipment was submerged and other equipment was wetted.
River Bend	04/19/89	When a freeze seal failed on a 6-inch service water line, portions of the auxiliary building were flooded by about 15,000 gallons of water. The flooding caused loss of the operating RHR train, spent fuel pool cooling and normal lighting in several areas.

SHUTDOWN EVENTS

Pressurizations		
PLANT	DATE	DESCRIPTION
Millstone 3	01/19/88	Event involved failure of the low temperature overpressure protection system to operate. The event was mitigated by operator action. In this event, isolation of the RHR letdown path caused a pressure excursion. The peak RCS pressure reached was 526 psia. The low temperature overpressure protection system did not work because a power supply which was required for system operation had its fuses removed without knowing what effect this removal would have on the system.
Surry 1	04/15/88	Both pressurizer PORVs failed to open manually. Torque values for diaphragm hold down bolts were not specified and actuator operation was intermittent. Required to be operable for LTOP.
Salem 2	10/27/89	One train the RHR suction piping was pressurized to 600 psig (150 psig above design). The event occurred because the RHR cold leg injection valve did not seat properly and allowed RCS coolant to pressurize the line.
Peach Bottom 3	10/2[illegible]/89	During excess flow check valve testing at 1000 psig, control rod drive scram time testing was also on-going. Letdown was decreased to increase temperature. Plant process computer used to monitor reactor pressure was not updating its pressure indication. Narrow range pressure recorder and the high pressure alarm were inoperable at the time. Reactor pressure increased 6 psi per minute. At 1055 psig, a scram signal was generated, resulting in stopping CRD flow to vessel and high pressure condition cleared.

Containment Integrity Problems

PLANT	DATE	DESCRIPTION
Salem 2	05/25/83	In refueling, containment integrity could not be established within the required 8 hours with the electrical power available. Condition lasted about 9 hours.
San Onofre 1	02/13/85	In mode 3, a security officer requested to unlock both normal and emergency escape hatches, manipulated the emergency hatch so that both doors were open. Condition existed about 21 hours.
Farley 1	04/15/85	During refueling, both doors of personnel air lock were inadvertently left open for about 4 hours.
McGuire 1	05/25/85	In refueling, during core alterations, containment ventilation cooling water system vent valve was found locked open. This configuration created a path from the upper containment to the auxiliary building through the open valve inside containment.
Farley 1	9/30/89	On two separate occasions core alterations were performed without containment integrity required by technical specifications. A pathway existed from the steam generator secondary side manway in containment through the steam generator atmospheric relief valve.
Fermi 2	09/04/85	At about 3% power, it was discovered that primary containment torus air sampling system test penetration valve was open and the penetration cap was removed. The condition may have existed for up to 75 days.

Containment Integrity Problems (continued)

PLANT	DATE	DESCRIPTION
Peach Bottom 2	08/17/86	The reactor mode was inadvertently increase from cold to hot shutdown due to instrumentation problems during testing. Containment integrity was not established during the period lasting about 4 and one-half hours.
Oyster Creek	11/8/88	Both doors of temporary airlock built to support isolation condenser maintenance open for up to 6½ hours. Event caused by administrative problems with the work package for isolation condenser maintenance.
Clinton	03/04/89	In preparation for local leak rate test, valve alignment was revised. Process resulted in a direct path from the steam tunnel in secondary containment to turbine building. Fuel movement was in progress at the time.
Washington Nuclear Project 2	05/05/89	Secondary containment integrity not maintained as required. Main steam isolation valve had been previously disassembled. A turbine throttle valve was removed providing path from the secondary containment to the atmosphere. Event caused by failure to recognize that a path existed. When recognized, the main steam line isolation valve (MSIV) was reassembled.

Loss of Electrical Power

PLANT	DATE	DESCRIPTION
Indian Point 3	11/16/84	While in cold shutdown for more than 1 month, loss of offsite power occurred due to sheet metal blowing across phases of buswork. 1 D/G out of service, other 2 D/Gs started but an output breaker to one vital bus did not close because the breaker for normal offsite power had not opened. Other attempts to energized that bus failed due to control power fuses being blown.
Fort Calhoun	03/21/87	Due to personnel error, all AC offsite electrical power was lost for 40 minutes and the one operable diesel generator did not start because control switch was in "off" position due to erection of scaffolding around diesel. The diesel was manually started and shutdown cooling was then restored after 5 minutes.
McGuire 1	09/08/87	At mid-loop, valve between diesel fuel storage tank and day tank for one diesel closed in error. Other D/G was inoperable. Diesel would have tripped had operators not found problem. Day tank was available.
McGuire 1	09/16/87	Due to maintenance error during testing, the unit experienced loss of offsite power for 6 minutes. Diesel generator started and loaded. The loss of power caused a loss of non-vital air which resulted in feedwater regulating valve closing and consequent reactor trip on *Unit 2*.
Harris	10/11/87	One incoming line to one of the safety buses was out of service for modification. The one remaining incoming line breaker tripped due to accidental jarring of protection relays. The diesel generator on this bus started and loaded appropriately.

Loss of Electrical Power (continued)		
PLANT	DATE	DESCRIPTION
Wolf Creek	10/15/87	A series of engineered safety feature actuations occurred due to reduced battery voltage. The 480 volt AC bus which powers the battery chargers was removed from service for maintenance. At the onset of the work, a battery life time estimate was made. However, this time was exceeded and the 125 volt DC source was lost.
Millstone 2	02/04/88	In preparation for a test, the auxiliary contact of a breaker was energized causing loss of one train of vital 4160 V ac. The other train was out of service for maintenance. The diesel generator started but because of a sequencer failure did not load. Operator action was required to re-energize the bus and reestablish shutdown cooling. (Ref. 29)
Vogtle 1	03/20/90	At mid-loop, loss of offsite power to operating RHR pump and failure of diesel generator to continue to run. RHR restored after 41 minutes; RCS temperature rose 46 °F to 136 °F.
Pilgrim	11/12/87	In cold shutdown, with the alternate source of offsite power out of service, snow caused loss of the both transmission lines. The diesel generators started and loaded. One of the diesel generators was shutdown after 9 hours due to operational problems. Offsite power was available after 11 hours. However, it was not restored for 21 hours.
Nine Mile 2	12/26/88	All offsite power lost due to explosion of current transformer in switchyard. Other source of offsite power out of service. Diesel generators started and loaded. Alternate source of power was subsequently restored.

SHUTDOWN EVENTS

Reactivity Events

PLANT	DATE	DESCRIPTION
San Onofre 2	03/14/82	Improper valve alignment during nitrogen backflushing in CVCS resulted in LPSI pumps used for shutdown cooling becoming gas bound. RHR was reestablished in about 90 minutes. However, during the restoration, LPSI suction was aligned to the RWST. Due to boron stratification in the RWST, measured boron concentration in the RWST was higher than that of RWST water pumped into the RCS. Thus, the addition of RWST water to the RCS resulted in dilution to less than 2000 ppm.
Callaway 1	07/19/84	Inadequate surveillance procedures made both trains of the SRM doubling circuit inoperable for about 5 hours in mode 5. The SRM trip and high flux alarms were still operational.
Oconee 2	04/11/85	Due to malfunction in power range recorder indicating less than actual flux, the high flux trip set point was exceeded and the reactor tripped.
San Onofre 3	04/13/86	An unplanned criticality occurred following a spurious reactor trip the previous day. Trainee was withdrawing rods. Estimated critical rod position was in error due to incorrect xenon reactivity. Reactor became critical sooner than expected and it was not recognized by the operator or his supervisor. Due to rod misalignments, the core protection calculator tripped the reactor. At this time the critical condition was recognized.
Vogtle 1	06/06/87	During startup due to error in the estimated critical position, reactor was taken critical before expected. Trainee was performing startup and SRO was inattentive. Procedure failed to adequately treat moderator temperature's effect on boron worth. Reactor trip occurred due to high source range flux.

Reactivity Events (continued)

PLANT	DATE	DESCRIPTION
Surry 2	10/25/89	Due to problems with the boric acid blender in the CVCS, the mode of operation for the blender was changed to manual. RCS filling was in progress and the flows were increased to expedite filling of the RCS. This resulted in increased boron concentration. Adjustment caused the boron concentration to decrease below the required 2000 ppm.
Millstone 1	11/12/76	During local shutdown margin tests, the wrong rod was withdrawn followed by withdrawal of the rod determined to have a high worth for the test. This resulted an unplanned criticality and reactor scram. In an attempt to complete the test, the similar error in rod selection was made a second time. During this second test, the rod was reinserted before a scram occurred.
Browns Ferry 2	02/22/84	Reactor scrammed due to high flux on two IRMs due to continuous withdrawal of a "high worth" control rod. Procedures were inadequate to ensure "high worth" rods were identified.
Hatch 2	11/07/85	While equalizing pressure around MSIVs, higher than expected demand resulted in vessel level reduction and increased feedwater demand. Increased feedwater flow caused power level increase and reactor scram.

Reactivity Events (continued)		
PLANT	DATE	DESCRIPTION
Peach Bottom 3	03/18/86	With the RWM inoperable, and a second operator verifying correct rod selection for withdrawal, the wrong rod was selected and verified by the second operator. Due to other duties the operator failed to subsequently note the error. RSCS initiated a rod block since the rod pattern was incorrect. The RSCS was bypassed without checking the rod causing the block. The RWM was returned to service and it indicated rod was in fact fully inserted versus fully withdrawn. A ontrolled shutdown including .. reactor scram was then ·erformed.
River Bend	06/14/86	During steam line drain valve alig ment, pressure dropped causing a vessel level sv ell and consequent feedwater reduction. Power decreased and IRMs were down ranged. Subsequently, water level decreased and the feedwater control system increased flow (startup regulating valve full open) resulting in power increase above set point for IRM range. Reactor scrammed.
Oyster Creek	12/24/86	At about 1%, high flux reactor scram occurred due to cold feedwater transient. Transient occurred during power reduction due to personnel error.

Mode Entry Events

PLANT	DATE	DESCRIPTION
Trojan	08/18/82	Both trains of SI were blocked for about 43 hours while in modes 3 and 4. While in mode 5, both trains were blocked due to testing. The plant status panel indicated this condition, however, it was not logged and proper procedures were not followed for removal of the system from service.
North Anna 1	12/06/82	An inadvertent automatic SI occurred which blocked additional SI actuations until reset by closing the reactor trip breakers. The procedure used to recover from the SI actuation, implied it was acceptable to leave the SI blocked for up to 30 hours which is not correct. The SI was unblocked about 22 hours later.
Oconee 3	04/10/87	Heating up, breakers for HPI suction valves from BWST were left tagged out. This is a technical specification violation because two flow paths to HPI are required. Cause was failure to review tag out logbook and poor communication between operations staff and support.

Other Types of Problems

PLANT	DATE	DESCRIPTION
Turkey Point 3	07/22/85	After a reactor trip, "B" S/G bypass feedwater valve would not open. AFW started on "B" low level. Later "C" S/G bypass feedwater valve would not close. "C" level rose and "B" feedwater pump tripped. AFW started again. No. "2" AFW flow control valve would not close.
Browns Ferry 1	06/27/84	At 1%, at about 400 psig, main steam relief valve opened. Torus temperature reached technical specification limit. Reactor was scrammed and valve reseated when reactor pressure was reduced.
Salem 1	07/16/84	0% power, one charging pump seized during surveillance and was inspected. Metal filings and resin particles were found in the pump. Similar material was found in the suction lines to all charging pumps. Potential existed to cause loss of all centrifugal charging pumps.
Quad Cities 1	08/08/84	While starting RHR, it was discovered that both LPCI injection valves would not open. One of these valves would open about 25%. Cause was valve stem damage due to design problem.

Attachment 2

Low Power and Shutdown Events from Foreign Experience

The attached table provides shutdown and refuel events obtained by review of foreign events. There are 20 events in the table. Event dates ranged from August 1975 to the present. The numbers in parentheses following the event are internal tracking numbers.

SHUTDOWN EVENTS

Loss of Reactor Coolant

TYPE	DESCRIPTION
PWR	Plant air system maintenance deflated seal which isolates spent fuel pool. Seal was subsequently ruptured during attempts to reinflate it. Leakage was controlled. Year: 1981. (166)
PWR	Due to thermal shock to drain line, a glass inspection window broke. Steam escaped at about 10 ton/hr from an unisolable leak. 4 and 1/2 hours were required to make containment entry. About 50 tons of coolant were lost. Year: 1982. (165)
PWR	Steam generator sensing line failed. A manual scram was initiated and the steam generator was isolated. Prior to event, RHR was being recirculated to RWST and was being aligned to provide RHR. Several procedure steps were skipped and about 3800 gallons of RCS coolant was diverted to RWST. Reactor pressure decreased from about 355 psig to 85 psig. SI was initiated to recover pressure. Year: 1983 (19,20)
PWR	Unisolable leak (about 1.8 gpm) due to fatigue failure occurred upstream of the isolation valve in the RHR system. Steam generators were not available for cooling. Repair was completed in 1.75 hours during which RCS temperature increased 70 °F. Year: 1987. (89)

SHUTDOWN EVENTS

Pressurizations

TYPE	DESCRIPTION
PWR	While cold, test of ECCS and failure to close and lock injection valves caused ECCS flow to enter RCS. Max RCS pressure reached 1250 to 1350 psi causing a cold overpressure condition. Year: 1981. (424)
PWR	During startup, check valve in RHR did not close. 3 operating shifts apparently ignored the open check valve even though it was indicated by the plant computer. An attempt was made to close the check valve by opening a valve in the RHR test line. This released coolant through the RHR relief valve. If the test valve had not closed, a LOCA could have occurred. Year: 1987. (465, 466)
BWR	Vessel pressure reached 1150 psig with the reactor coolant temperature between 79 and 88°F. Pressure exceeds allowable pressure at a temperature of 88 °F. Vessel level about 6 inches below flange. Main steam line isolation valves and safety-relief valves were shut. RHR was in service. Control rod drive pump was started. This injected about 25 gpm into the vessel. Reactor pressure increased to the RHR isolation set point and RHR isolated automatically. Operations staff was not aware of the RHR isolation. Vessel was effectively "bottled up" except for the CRD injection. Pressure increased to 1150 psig at which point the CRD pump tripped due to low suction pressure. Year: 1989. (HIWRT)

SHUTDOWN EVENTS

Loss of Electrical Power

TYPE	DESCRIPTION
PWR	With all fuel in the fuel pool, a new transformer was installed. The new transformer failed causing loss of all offsite power. Two of 3 D/Gs started as expected. Due to breaker problem with swing D/G, it would not close on its bus. Cooling water pumps for D/G supplying power did not start. D/G could have failed. Manual action was taken to reconnect plant to offsite sources. Year: 1986. (298)
PWR	One train of raw cooling water and one train of the component cooling water were out of service. Fuel was in the spent fuel pool. One source of offsite power and one D/G were out of service. Train A offsite power was lost and the cooling circuits connected to train B were isolated. Control logic for the available D/G was supplied through a electrical panel which was lost. Fuel pool cooling was lost for 49 minutes. No temperature increase occurred. Year: 1988. (85)
PWR	With one transformer and one D/G out of service for maintenance and with all fuel in spent fuel pool, all spent fuel cooling was lost for 17 minutes when another offsite source was lost and the diesel generator did not start because a startup valve was left closed after repair work. No temperature increase occurred. Year: 1988. (84, 86)
PWR	Cold shutdown, during testing, loss of safety busses occurred. An instrumentation bus was lost. 6 of 8 D/Gs did not start and RHR was lost for 20 min. RCS temperature increased 100 ° F. Access control prevented entry to the D/G rooms to manually start the D/Gs. Year: 1988. (307)

SHUTDOWN EVENTS

Reactivity Events	
TYPE	DESCRIPTION
PWR	Boron makeup system malfunction caused demineralized water to enter CVCS instead of requested makeup blend. Dilution from 2200 to 1400 ppm occurred. Problem due to electrical circuit. Year: 1975. (531)
PWR	During S/G tube maintenance secondary to primary leak caused dilution. PPM decreased about 50 ppm before event was stopped. Year: 1982. (534)
PWR	During testing after initial fuel loading, power supplies were turned off and neutron monitoring was not available during subsequent testing. Alarms indicating this fact were not noticed by operators and the audio alarm was turned off. Potential boron dilution might not have been detected. Year: 1987. (94)
PWR	RCS boron dilution took place. Vessel level was at about mid-loop, boron concentration was at 2083 ppm, and the RHR system was providing cooling. Secondary side of the number 2 steam generator (s/g) was being filled. S/G tube had been cut, a portion removed, and had not been plugged. Thus, water leaked from the secondary side of the S/G to the primary system. Leakage reduced boron concentration to 2001 ppm. Maximum leak rate estimated to be 45 gpm. Event was bounded by plant analysis. Had dilution continued, criticality might have been reached in about 4 hours. Failure of S/G primary nozzle dam also postulated after a sufficient time to fill the s/g cold leg side below the tube sheet. Worst scenario could produce criticality with high flux. Year: 1990. (HIWRT)

SHUTDOWN EVENTS

Mode Entry Events	
TYPE	DESCRIPTION
PWR	Plant heated up from cold to hot shutdown without high or low pressure SI. Due to maintenance problems. HPSI breaker was open, LPSI injection isolation valves were closed, and initiation controls were unavailable due to incorrect switch position. Year: 1986. (96)
BWR	During shutdown margin tests, testing involving rod withdrawal was performed prior to testing of the RPS with the fast acting hydraulic scram system unavailable (electric system was available). Year: 1987. (341)

SHUTDOWN EVENTS

Other Types of Problems

TYPE	DESCRIPTION
PWR	During startup, steam generator tube rupture occurred. Rapid decrease of pressure and pressurizer level occurred. Use of turbine driven AFW pump caused radioactivity release. Year: 1979. (15)
PWR	The autostart feature of the AFW was inoperable due to maintenance error. Loss of feedwater occurred and the AFW system had to be started manually. Year: 1986. (95)
PWR	Four of 5 pressurizer (PZR) level instruments were left closed. During subsequent startup, while forming the bubble, the 4 level instruments gave false indications and the 5th one, although correct, was thought to be in error. PZR level was allowed to drop and 2/3 of heaters were uncovered and damaged. Year: 1989. (72)