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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	Docket No. 50-322 O.L.

ADDENDUM TO TESTIMONY OF MARC W. GOLDSMITH ON BEHALF OF SUFFOLK COUNTY REGARDING SUFFOLK COUNTY CONTENTION 4 - WATER HAMMER

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ADDENDUM TO TESTIMONY OF MARC W. GOLDSMITH
ON BEHALF OF SUFFOLK COUNTY REGARDING
SUFFOLK COUNTY CONTENTION 4 - WATER HAMMER

This addendum addresses two reports on water hammer provided by the NRC Staff on April 8, 1982: "Evaluation of Water Hammer Events in Light Water Reactor Plants," March 1982, by the Quadrex Corporation and "Compilation of Data Concerning Known and Suspected Water Hammer Events in Nuclear Power Plants," April 1982, by EG&G Idaho, Inc. ("EG&G").1/ The EG&G report compiles water hammer event data while the Quadrex report evaluates these compiled data and provides recommendations for minimizing water hammer. These reports, although in draft form, provide recent water hammer event data, along with new solutions and recommendations for preventing water hammer occurrences in nuclear power plants. These reports also illustrate the importance of incorporating past water hammer experience into plant procedures and the need for design modifications to prevent water hammer events.

NUREG-0582, "Water Hammer in Nuclear Power Plants," lists water hammer events that have occurred through 1978. The two reports mentioned above, however, include water hammer experiences through April 1981. Recent events, many of them serious, occurred at plants similar in design to Shoreham during pre-operational or

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1/ The Quadrex Report (minus Chapter 5 concerning PWR events) and the EG&E Report are attached as Exhibits 1 and 2, respectively.

startup experience at similar plants and incorporating this experience into Shoreham's program.

Included in the EG&G report was an evaluation of more PWR plants (44) than BWR plants (26). However, the BWR plants had approximately twice as many water hammer events (81) as the PWR plants (40). Approximately half of the events occurred within a two year period. This period included one year prior to commercial operation and one year after commencement of commercial operation.

All of the BWR plants were General Electric BWRs except one (13 were GE-Model 4, 7 were GE-3, and 6 were "others"). Out of the 81 BWR events, forty-eight occurred at GE-4 plants, twenty-nine occurred at GE-3 plants, and four occurred at the "other" plants (Shoreham is similar to a GE-4 design). Approximately three-fourths of the BWR events occurred in safety-related systems. One-third of the events occurred in the Residual Heat Removal System (RHR). In a BWR plant, the RHR has many functions, not all of which are directly safety-related, but which are connected to safety systems. One-fourth of the events occurred in the High Pressure Coolant Injection System (HPCI).

For both BWRs and PWRs, a third of the event reports indicated damage to piping-related systems. The damage involved welds, junctions, pumps, and other system components but valves were not damaged. The damage was significant, however, since it occurred in portions of the piping that could allow coolant leakage, if sufficiently damaged.

Water hammer events discussed in the reports were caused by either design and/or procedure-related failures. EG&G determined that BWR plants attributed the causes more to procedure-related problems than PWR plants. Therefore, preoperational and start-up experience at other plants with similar systems is important to BWRs in arriving at procedures that minimize water hammer.

The Quadrex Corporation report, an extension of previous evaluations performed by EG&G, evaluated the implications of the water hammers that occurred. None of the water hammer events placed a plant in a faulted or emergency condition. However, 18 of the water hammer events rendered a safety system inoperable. These included two events when flooding caused by nonsafety-related water hammer caused safety systems to become inoperable.

Of the 82 reported events, 72 were considered to be unplanned safety-related events. Sixty of the 72 safety-related water hammer events occurred in four systems: RHR (24 events), HPCI (20 events), core spray (9 events) and service cooling water (7 events). Three of these systems are of General Electric design and the fourth (service cooling water) is an architect/engineer design. Based on the frequency of BWR water hammers in the preoperation and startup phase, it appears that Shoreham could potentially have many water hammer occurrences during pre-operational or startup testing in safety-related systems.

Quadrex evaluated the water hammer events and found pumping water into a line containing voids to be the largest single cause

of BWR water hammers, and responsible for 43 events. This generic cause includes flow into voided lines, column separation and steam bubble collapse events.[2/] Because of the large number and severity of void related water hammer impacts, Quadrex states "the most serious concern is line voiding and should be a subject of regulatory action."

Quadrex also determined that current designs do not provide the operator with information concerning the existence of voids. For example, one concern is that certain safety systems may be more prone to water hammer under unplanned (i.e., accident condition) actuation than the reported data indicate. These systems are often vented prior to planned periodic testing or other usage to eliminate voids. An unanticipated start, such as would occur following a postulated accident, may occur with voids in lines and result in a water hammer.

Quadrex stated that "void caused water hammers could be greatly reduced or eliminated by the use of void detection and alarm, keep fill and modified venting systems." While Shoreham has keep fill and modified venting systems on the Emergency Core Cooling System (ECCS) and alarms on the keep fill system, it is unclear as to how these are to be used. The 2/1/82 version of the Shoreham Technical Specifications deletes channel checks on

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2/ Voids can occur through many means, including improper line filling during maintenance, gas evolvement, improper venting, leakage of water, in leakage of steam, and column separation following pump stoppage or valve closure.

keep fill alarms and a cursory review did not indicate any technical specification limits on operation. The keep fill system operability, or the system operability without keep fill operating, are unclear. The Quadrex report states that if voids are suspected then the system should be declared inoperable.

Many of the water hammer events were reported as having been caused by procedures. This was to be expected since operator actions are controlled by procedures. Therefore, it is essential that procedures be correct and complete. Certain good practices that aid in preventing water hammer, such as gradual line warmup, controlled valve opening, and draining and venting, are usually covered by procedures. However, as discovered by Quadrex during discussions with procedure writers and approvers, the potential for water hammer is generally not considered in procedure writing or review. It was also learned from these discussions that piping drawings such as isometrics that show relative piping and component elevations are not used in the writing of procedures or work instructions.

The Quadrex report contains specific recommendations for minimizing water hammer events. It is recommended that operating and maintenance procedures for systems in which safety-related water hammers can occur be reviewed for their effect on water hammer occurrence. Additionally, it is suggested that the relative elevations of system lines and components be considered in the writing of operating and maintenance procedures.

The foregoing information supports Suffolk County Contention 4 which suggests that LILCO incorporate similar BWR pre-operational or startup tests into Shoreham's program not only for design but also for operator training and procedure development.

Other recommendations for preventing water hammer are discussed in the Quadrex Report. Because past BWR (GE-4) water hammer experiences were numerous, preventive measures should be taken at Shoreham to minimize potential water hammer events. LILCO has taken some of these measures such as installing a keep fill system and a venting system. If Shoreham uses the keep fill alarm as an indicator of voids, then an important step will have been taken in reducing a key cause of water hammer. This requires a technical specification commitment. In addition, other plant startup and test data should be reviewed and that data should be used in procedure development and training operators to prevent water hammer. Therefore, many measures still remain to be taken at Shoreham to minimize water hammer.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	Docket No. 50-322 (OL)

CERTIFICATE OF SERVICE

I hereby certify that copies of the "ADDENDUM TO TESTIMONY OF MARC W. GOLDSMITH ON BEHALF OF SUFFOLK COUNTY REGARDING SUFFOLK COUNTY CONTENTION 4 - WATER HAMMER," dated April 21, 1982, have been served to the following this 21st day of April, 1982 by U.S. Mail, first class, except as otherwise noted.

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April 20, 1982

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EXHIBIT 1

EVALUATION OF WATER HAMMER EVENTS IN LIGHT WATER REACTOR PLANTS, MARCH 1982, BY THE QUADREX CORPORATION

QUADREX

EVALUATION OF WATER HAMMER EVENTS IN LIGHT WATER REACTOR PLANTS

4/9/82

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CONTROLLED No. 5

Rec'd 3/12/82

QUAD-1-82-018 AW STRK

EVALUATION OF WATER HAMMER EVENTS IN LIGHT WATER REACTOR PLANTS

DRAFT

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QUAD-1-82-018

EVALUATION OF WATER HAMMER EVENTS IN LIGHT WATER REACTOR PLANTS

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Draft Report

March 1982

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Prepared for EG&G Idaho, Inc.
Under Contract No. K7862

MAR 15 1982

DRAFT

ABSTRACT

This document presents the results of evaluations of water hammer events in LWR power plants. The evaluations were based upon reports of actual events, typical plant design drawings and operating procedures. Included in this report are design and operating recommendations for the prevention and mitigation of water hammer occurrence.

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1.0 INTRODUCTION

This report presents the results of an evaluation of actual and potential water hammer events occurring in LWR power plants. This work was performed by the Quadrex Corporation for EG&G, Idaho Incorporated and is an extension of previous evaluations performed by EG&G, Idaho.

The objectives of the work reported herein are to evaluate water hammer events that have occurred in commercial nuclear reactors and to develop methods for their prevention and mitigation. The evaluations are based upon the incident reports contained in reference 1, reviews of licensing event reports (LER), FSARs, typical plant design drawings, system descriptions and operating instructions and the operating and design experience of the authors. Event numbers used in this report are the same as those used in reference 1.

Steam generator water hammers (SGWH) are not included in the scope of this report.

A summary of the findings and recommendation of this study is presented in section 2.0. Generic and overview findings, evaluations and recommendations, that are based upon the individual system evaluations are contained in section 3.0. Individual system evaluations are contained in section 4.0 and 5.0 for BWR and PWR systems respectively. Section 6.0 presents recommended mechanisms and regulatory requirements for the prevention and mitigation of water hammer events.

2.0 SUMMARY

An evaluation of water hammers occurring in light water reactor plants was performed using the category I events listed in reference 1 as a basis. Recommendations for the mitigation and prevention of water hammers were made.

Water hammer damage for most instances was limited to the piping support systems.

The frequency and severity of water hammer in PWR plants was low. None of the 40 reported events disabled a safety system, had an adverse safety effect on the plant, or placed a plant in a faulted or emergency condition.

The frequency and severity of water hammer in BWR plants was higher than in PWR plants. Eighteen of the reported 82 events in BWR plants disabled a safety system. However, no event disabled more than one train or system with the possible exception of flooding events caused by water hammer in a nonsafety system. No event placed a plant in a faulted or emergency condition.

The predominant cause of water hammer events was the presence of voids or steam bubbles in pumped water lines. The presence of these voids was not readily detectable by operators. Other major causes of water hammer events were water entrainment in the HPCI turbine inlet and outlet lines, and in the isolation condenser inlet lines, and inadequate PWR feedwater control valves. The causes of several events were unknown. The damage from several events was the result of inadequate support design for loads resulting from anticipated valve closure induced steam hammers and safety/relief valve discharge. A detailed overview and generic evaluation are presented in section 3.0.

Recommendations for prevention and mitigation fall into two categories. The first category includes recommendations, often peculiar to a particular system or problem that should not be considered as regulatory items, but rather as suggestions to aid in prevention and mitigation. These recommendation are presented in the generic evaluation of section 3.0 and the individual system evaluations contained in sections 4.0 and 5.0. The second category of recommendations are those deemed significant enough to be a regulatory concern and are presented in section 6.0. These include:

- Mandatory void detection, keep fill and venting provisions for several systems
- Operator training
- Feedwater control valve design verification
- HPCI inlet line valve design features
- HPCI inlet line drain pot level detection
- HPCI and RCIC turbine exhaust line vacuum breakers, and
- Main steam and PWR - RCS support and component design basis.

The regulatory requirements should be implemented by a SRP or Branch Technical Position for plants in the design phase and by an IE bulletin or generic letter for operating plants.

3.0 GENERIC EVALUATIONS

This section contains generic evaluations of water hammer events and their causes and recommendations of measures for their mitigation and prevention. The evaluations contained in this section are based upon the individual systems evaluations contained in sections 4.0 and 5.0.

3.1 General Overview

3.1.1 PWR Systems

There were 40 category I events reported in reference 1. None of the water hammer events had any adverse safety effect on the plant. No water hammer event rendered a safety-related system inoperable or damaged the integrity of the reactor coolant boundary. For most of the events damage was limited to the piping support system.

The frequency and severity of safety-related water hammer events in the PWR systems are low with the exceptions of steam generator water hammer (SGWH) which is not within the scope of this study and feedwater contol valve (FCV) induced water hammer events.

Of the 40 reported events only 25 are considered to be unplanned safety-related water hammer events. See table 3-1. Four of the events (three in the Reactor Coolant System (RCS) and one in the main system) are considered relief valve discharge reaction forces (see sections 5.2 and 5.3) and five of the events, all in the main steam system, (see section 5.3) were valve closure initiated steam hammers which should be included in the design basis of the support system. Two RCS events (section 5.2) were not water hammers but the results of valves sticking open. Four events in the condensate and condenser system (see section 5.7) are not regarded as safety-related. Furthermore, three of these condenser events were not water hammer but jet impingement force incidents. Event 28 (section 5.8) in the circulating water system and event 11 (section 5.7) in the condensate system were considered to be neither safety-related nor water hammers.

Of the 25 safety-related water hammers, 13 occurred in the feedwater system (see section 4.land table 3-2). Eight of the feedwater system water hammers were related to the feedwater control valve. One other event was due to an improper procedure in opening a valve and another was due to a design error. The cause of three of the feedwater events were unknown. The damage reports indicate that the greatest forces were due to events occurring in the feedwater system. This is to be expected due to the large line size and the high fluid velocities and density in the feedwater system. The feedwater system, especially, the FCV requires regulatory measures.

Of the non-FCV events in the various systems (including FW); seven involved the presence of voids in a line, two involved improper valve usage, and one involved a design error. The causes of five of the events were unknown.

Of the five events with unknown causes, two in the CVCS system may not have been water hammer and were of low safety significance. Another event in the steam generator blowdown line is of low safety significance.

Regulatory action, discussed in section 6.3 is recommended to address feedwater control valve design (nine events) line voiding (seven events) operator training (two valve usage and perhaps the unknown events).

3.2 BWR Systems

The frequency and severity of safety-related water hammers in the BWR systems are moderate and are greater than for PWR systems.

There were 82 category I water hammer events reported in reference 1. None of the water hammer events placed a plant in a faulted or emergency condition. However, 18 of the water hammer events rendered a safety system inoperable. These included two events when flooding caused by nonsafety-related water hammers caused safety systems to become inoperable. A water hammer in conjunction with a stress induced condenser tube

rupture caused an isolation condenser to become inoperable. No events damaged the integrity of the reactor coolant boundary. For most of the events, damage was limited to the piping support system.

Of the 82 reported events 72 are considered to be unplanned safety-related water hammer events. One event in the main steam system (see section 4.6) is considered a relief valve discharge force. Two main steam events (see section 4.6) were valve closure initiated steam hammers, which should have been included in the design basis of the support system. One event in the RCIC system was a pump cavitation event (section 4.5). Six of the events, four in the condenser system, one in the auxiliary boiler (section 4.9) and one in the reactor water cleanup system (section 4.8) are not considered safety-related water hammer events.

Sixty of the 72 safety-related water hammer events occurred in four systems RHR (24), HPCI (20), and core spray (9) and service cooling water (7).

Two of the RHR events were actually initiated during HPCI turbine inlet line warmup. Other systems in which events occurred include isolation condenser (four), RCIC, (one) main steam, feedwater (three).

Pumping water into a line containing voids was the largest single cause of BWR water hammers and was responsible for 43 events. This generic cause includes flow into voided line, column separation and steam bubble collapse events. A generic discussion of line voiding is provided in section 3.3.

Seven events in the HPCI system (section 4.4) were caused by failure of drain pot systems. Six events, five in the HPCI system and one in the main steam system were caused by improper valve operation or warmup of the inlet line. Three events were caused by feedwater valve controller instabilities. Three events in the isolation condenser were caused by

reactor water surginy into the isolation condenser inlet line. Two events, one in main steam and one in the isolation condenser were caused by improper line s'ope.

The most serious concern is line voiding and should be a subject of regulatory action. A void indication and alarm system in combination with a requirement that a system be considered inoperable if there are voids in its pumped water line would essentially eliminate these water hammers. See section 3.3. Certain design modification and increased operator training would eliminate or greatly reduce the various turbine steam line water hammers. See section 3.5.

3.3 Line Voiding

Line voiding has been the single greatest cause of water hammer events identified in this report. Fifty-four percent (52 of 97) of the unplanned category I safety-related water hammer events were caused by pumping water into a line containing voids. The generic line voiding cause discussed in this section includes flow into voided lines, steam bubble collapse and column separation. The one common denominator in each case is that the event could have been avoided had the operator been aware of the void. Voids can occur through many means including improper line filling during maintenance, gas evolvement, improper venting, out leakage of water, in leakage of steam and column separation following pump stoppage or valve closure.

Voiding can generally occur in standby systems that are normally idle. Systems that continually run such as feedwater are started slowly and kept full by continuous operation. BWR systems are more prone to voiding than similar systems in PWRs. There are two main reasons for the differences between the BWR and PWR voiding frequency. The first is the elevation of the safety system's water source. The PWR pumps are supplied by the refueling water storage tank (RWST) which is maintained at an elevation above the pump discharge lines. The BWR safety systems most prone to line voiding, namely RHR and core spray, receive their supply from the suppression pool which is maintained at a level below the elevation of the pump discharge lines. Other systems which have

experienced less voiding are supplied by the condensate storage tank which in many plants is maintained at a level above of the pump discharge lines. The open service water systems for both BWR and PWR plants are supplied by sources below the level of the system lines. The second difference between BWR and PWR plants is the presence of steam water interfaces in BWR's permitting leakage of steam bubbles into the water lines.

The comparative studies of the RCIC, HPCI and AFW systems (section 3.4) indicate that line size is a factor in line voiding and its effects. Smaller lines appear to be less prone to observable water hammer than larger lines. This might be due to the fact that less leakage occurs through the valves of smaller lines. Another factor is that forces resulting from water hammers in small lines are smaller than those resulting from larger lines. Thus water hammers occurring in smaller lines may not be considered reportable or even detected if no damage occurred.

The addition of keep fill systems to BWR systems has reduced frequency of water hammers. (The water supply system for a PWR essentially acts as a keep fill system.) However, venting is also required to remove voids. In many plants venting is a difficult procedure due to the location of the vent valve. Venting may require wearing anticontamination clothing, entry into moderate radiation areas, considerable climbing and personal discomfort. Operations involving such difficulties are generally performed only to meet specific requirements or needs rather than routinely and frequently.

There is a concern that certain safety systems may be more prone to water hammer under unplanned (i.e., accident condition) actuation than the reported data indicates. These systems are often vented prior to planned periodic testing or other usage to eliminate voids. An unanticipated start, such as would occur following a postulated accident, may occur with voids in lines and result in a water hammer. Current designs do not provide the operator with information concerning the existence of voids.

Void caused water hammers can be greatly reduced or eliminated by the use of void detection and alarm, keep fill and modified venting systems.

3.4 Comparison of HPCI with Similiar Systems

This section compares HPCI (BWR) with the RCIC (BWR) and the AFW (PWR) systems in order to determine causes for the high frequency of water hammer events in the HPCI system. The RCIC and AFW systems are approximately one-tenth the size of the HPCI system, but are similiar in the following respects:

a. The system pumps are driven by steam turbines that are normally in a standby condition.

b. The systems are infrequently used.

c. The systems are surveillance tested monthly.

d. The systems pump ambient temperature water through normally unused lines to the feedwater lines at feedwater pressure.

Twenty water hammer events were noted in the HPCI system compared to only one in the AFW system and one in the RCIC system.

3.4.1 Steam Supply Lines

The supply lines for all three systems are normally kept warmed with steam up to the turbine stop valve and contain steam traps and drain pots. No steam supply events were noted in either the AFW or RCIC system. Eight steam supply incidents were noted in the HPCI system, three caused by valve operation and five caused by the failure of the steam trap level control and drain system.

There are two significant differences between the HPCI steam supply line and the RCIC and AFW steam supply lines. The first is the presence of a "seal in" feature on the HPCI inboard isolation valve. A seal in is a control feature, that causes the valve to open continuously to the full

open position upon actuation. This feature precludes using this valve for gradual line warmup or venting. The RCIC and AFW isolation valves generally do not have the seal in feature.

The second significant difference is size. The HPCI line is sized for approximately ten times the flow rate as the AFW and RCIC lines. The HPCI line thus is subject to considerably more steam condensation than the AFW and RCIC lines. It is possible that the drain pots of the AFW and RCIC systems have sufficient capacity to accommodate occasional malfunctions of the drain systems that may occur between periodic technical specification testing. Water hammer forces are also larger in a larger line. The events in the HPCI line have only caused minor damage. If these events had been scaled down by the 10:1 ratio of the RCIC and AFW systems, their effect may have gone unnoticed and thus unreported.

3.4.2 Steam Exhaust Lines

There were six events reported in the HPCI steam exhaust lines and only one in the RCIC and one in the AFW lines.

The HPCI and RCIC lines discharge into a water interface (the suppression pool), but the AFW line discharges into the atmosphere. Many of the HPCI events and the RCIC event occurred prior to the addition of vacuum breakers to the exhaust lines. The vacuum breakers prevent a vacuum from drawing suppresion pool water into the lines. The HPCI line is sized for ten times the flow rate of the AFW and RCIC turbines.

The reasons for the higher frequency of HPCI events may be size and the presence of the water-steam interface without vacuum breakers in some early installations.

3.4.3 Pump Discharge Line

Three water hammer events were noted in the HPCI pump discharge lines but none were noted in either the RCIC or AFW pump discharge lines.

The HPCI and RCIC lines are similar except for the HPCI lines being larger (10:1 flow area) and longer. Both pumps are normally aligned to the condensate storage tank for suction and in the systems reviewed discharge to the feedwater lines. The AFW lines are approximately the same size as the RCIC lines and are aligned to the refueling water storage tank and and discharge int) the feedwater lines. A more detailed discussion of line voiding is provided in section 3.3.

3.4.4 Conclusions of Comparison

Although there are several features that distinguish the HPCI system from the RCIC and AFW system, the difference that occurs in all three line types (turbine inlet, turbine exhaust and pump lines) is size. Larger lines may have a greater propensity for condensation (steam lines) and leakage caused voiding (water lines), which makes them more susceptible to water hammers. Water hammer forces and damage, increase with line size. Therefore, smaller water hammers occurring in the RCIC and AFW system may not be significant and thus not detected or reported.

3.5 Mitigation and Prevention of Water Hammer

This section provides a discussion of various generic methods to prevent and mitigate water hammer events. The inclusion of a method in this section should not imply that it is to be applied to all systems. The regulatory means for implementing these measures are discussed in section 6.0.

3.5.1 Line Void Detection, Filling and Venting

Fifty-four percent of the category I safety-related water hammer events reported in reference 1 occurred because water was pumped into a line that contained voids. These were primarily flow into void line events but also included steam bubble collapse and column separation. (See section 3.3 for further discussion of voiding.) All of these events could have been prevented if the operators had been aware of the existence of the void. A properly designed void detection and alarm system combined with a technical specification requirement that the system be considered

inoperable if voids were present would have eliminated these events. The following systems should have void detection and alarm:

o BWR:

- Core spray (pump discharge)	10 events
- RHR (all liquid lines)	22 events
- HPCI(HPCS) - pump discharge	3 events
- Cooling Water	8 events
	43

o PWR:

- ECCS (safety injection)	4 events
- Cooling Water	2 events
	6

In addition due to their requirements for rapid start or frequency of events the following systems should be provided with keep fill systems:

o BWR:

- Core Spray
- RHR
- HPCI
- RCIC

Keep fill systems are not required for PWR systems because the refueling water storage system acts as an intrinsic keep fill system. The use of a keep fill system for open loop service water systems is impractical due to the continual and large line losses. The service water lines are generally very large and very long. Furthermore, much of the line is remote from the main safety areas where the ECCS keep fill system is located, and there is considerable branching and many components served. The use of the ECCS keep fill system for a service water system is impractical. Therefore, the following recommendations are made for filling of open loop service water systems. For these systems it should be shown that either:

a. voids can be filled within prescribed time using a manually initiated fill system,

b. neither column separation nor voiding will occur during standby or following pump shutdown once the line has been filled and vented,

c. the system is designed with a startup mode that slowly fills and vents the discharge lines in such a manner as to prevent water hammer on pump startup. Low flow bypass valves or slow opening discharge valves are examples of features that can permit the system to meet this requirement. Analysis and testing would be required to show that slow fill and system minimum startup time requirements can be achieved or

d. analysis has determined that the system including its supports, is designed to maintain function following a postulated water hammer event.

Additionally, venting provisions should be provided at all points where voids could form either through maintenance, operating, draining, outleakage, gas evolvement, or in-leakage of steam or flashing fluid. The venting system shall be readily operable during all modes of plant operation. Suggested types of venting systems are remotely operated valves and valves located for ease of access. For some systems the use of vacuum breakers may be a desirable feature.

3.5.2 Operator Training

Most of the reported water hammer events involved plant operators and maintenance personnel to a varying degree. The reduction of water hammer events will require the participation of plant personnel. They frequently write the plant operating procedures, and ultimately approve them. The operators start the pumps, open the valves and place systems in operation, test, and maintain them. They would not knowingly initiate events that would cause a water hammer event.

Over 50% of the events occurred during plant startup and the twelve months following commercial operation. This indicates there is a learning period during which plant personnel and management become familiar with

the systems operation, change procedures, correct design errors, add equipment such as vents and drains, and make fewer errors. To be most effective, efforts to reduce water hammer events should start before plant operations and the learning by experience period starts.

An investigation of the general causes for the events, that involve plant design, training, operation and management, indicates the following:

- o There is often a lack of awareness among plant operators concerning the possibility of water hammer events occurring in a particular system or subsystem, their causes, and what the results of those events would be. Discussions with various plant operators reveals that they know from experience that water hammers occur, but in no case have they had specific training as to why or where water hammer events happen, what types of systems are susceptible, or what types of corrective actions are possible.
- o There is a lack of information available to the operators, concerning the existing conditions in the systems before the water hammer events occur. A review of the 82 BWR events and the 40 PWR events on the LER category I list of reference 1 reveals that in only 13 out of the 122 events was applicable instrumentation mentioned as part of the original design to give warning or as part of the repair effort to mitigate further events.
- o The implications of equipment malfunctions and maintenance related failures of components such as shutoff valves, steam traps, and check valves, are not fully considered as part of the causes of water hammer events by designers and plant operators.

Many water hammer events can be eliminated by design changes that provide the operator with more information (e.g., void detection and improved steam drain pot level indicators), preclude adverse conditions (e.g., vacuum breakers and keep fill systems) and minimize the potential for operator error (e.g., valve interlocks and operability requirements). However, there are many operations such as line warmup and venting that require operator knowledge of system conditions. Therefore, all plant operators

including personnel responsible for the writing of maintenance instructions and the supervision of maintenance activities should receive training in the causes and prevention of water hammer.

3.5.3 Turbine Exhaust Line Vacuum Breakers

The turbine exhaust lines of the HPCI and RCIC systems interface with the suppresion pool. Water hammers have been caused by suppression pool water being drawn into these lines due to vacuum formation (see sections 4.4 and 4.5).

Vacuum breakers should be incorporated on both the upstream and downstrean sides of the exhaust line stop/check valves for the following BWR systems:

- o HPCI
- o RCIC

3.5.4 Turbine Steam Line Drain Pots

The only system in which drain pot operation is considered a significant water hammer concern is the BWR HPCI system (see section 4.4). A comparison of the HPCI, RCIC and AFW systems (sections 3.4), indicates that the problem may be related to line size.

The adequacy of steam turbine inlet line drain pots sizing should be reviewed for all HPCI systems. If the size is determined to be inadequate, additional or larger drain pots shall be installed.

The operability of the steam line drain pot level switches should be verified monthly for the HPCI systems. Those systems in which level switch verification of operation and required maintenance can not be performed with the system in service shall be modified to permit such verification and maintenance.

3.5.5 Steam Supply Line Inlet Valves

Water hammer events have been caused by operation of the HPCI outboard steam line inlet valve. To prevent this the following is recommended.

An interlock should be provided that will preclude opening of the inboard isolation valve unless the outboard isolation is fully open. Neither valve should contain a seal-in feature on opening. The inboard valve should be designed for throttling and must be opened slowly to permit gradual line warmup and draining of all liquid. These requirements should apply to the following system:

o HPCI (BWR)

3.5.6 Anticipated Loads

Certain loads such as steam hammer due to rapid valve closure or forces caused by safety and relief valve actuation can not be prevented. As an example turbine stop valves close in approximately 0.1 to 0.2 seconds.

The forces generated by these loads should be considered in determining the design basis for the piping, its support system and other components such as valves. The inclusion of these loads in the design basis for piping is required by NUREG-0737, ASME B&V Code Section III and ANSI B31.1 (references 2, 3, and 4).

3.5.7 Operating and Maintenance Procedures

Many of the water hammer events were reported as having been caused by procedures. Additionally, other events may have been avoided had different procedures been available. This is to be expected since all operator actions are controlled by procedures. Therefore, it is essential that procedures be correct and complete.

Certain good practices that aid in preventing water hammer, such as gradual line warmup, controlled valve opening, draining and venting, are usually covered by procedures. However, discussions with procedure writers and approvers indicate that the potential for water hammer is generally not considered in procedure writing or review. It was also learned from these discussions that piping drawings such as isometrics that show relative piping and component elevations are not used in the writing of procedures or work instructions.

It is recommended that operating and maintenance procedures for systems in which safety-related water hammers can occur be reviewed for their effect on water hammer occurrence. Additionally, it is suggested that the relative elevations of system lines and components be considered in the writing or operating and maintenance procedures. Isometric piping drawings, sufficiently scaled to show relative elevations, would be useful in writing procedures and performing maintenance. It is suggested that the isometric drawings be available to operating and maintenance personnel as part of the system procedure package.

3.5.8 Line Sloping

A few events have been caused by the inability to properly vent or drain a line due to the location of high and low points. These conditions, however, are detected early, generally during plant startup. To prevent such incidents it is suggested that the design of lines be reviewed for proper slope, and the location of high and low points in both hot and cold conditions. A similar as-built review of the lines should be performed during startup and any necessary adjustments or modifications to the lines and their supports be made. Line isometric drawings should be updated to reflect as-built conditions.

TABLE 3-1

PWR CATEGORY I EVENTS

Total Category I events			40
o [1]Safety-related water hammer events			25
o Not safety-related water hammer events:			
	- Relief valve reaction forces	4	
	- Steam hammer, not safety-related	3	
	- Water hammers, not safety-related	1	
	- Neither water hammers nor safety-related	5	
	- Switch open relief valve	2	
		15	

[1]Includes three events that are of low safety significance and may not have been water hammer.

TABLE 3-2

PWR SAFETY-RELATED WATER HAMMERS

Total			25
Feedwater System		13	
o FCV related	9		
o Other and unknown	4		
Main Steam		1	
Auxiliary Feedwater Turbine		1	
[1]Steam Generator Blowdown		1	
[2]Chemical volume control		2	
Residual heat removal		1	
ECCS (Safety Injection)		4	
Cooling Water		2	

[1]Event of low safety significance

[2]Events probably not water hammer and low safety significance

TABLE 3-3

BWR Category I Events

Total BWR Category I Events			82
o	Safety-related unplanned water hammer events	72	
o	Nonsafety-related water hammer events:		
	- Relief valve discharge force	1	
	- Steam hammer bounded by design case	2	
	- Pump Cavitation	1	
	- Not safety-related	6	
		10	

4.0 BWR SYSTEM EVALUATIONS

This section contains evaluations of BWR plant system water hammer occurrences. Separate evaluations are provided for each system and are based upon events reported in reference 1. In addition to the information contained in reference 1, Licensing Event Reports (LERs), typical system P&IDs, physical drawings, system descriptions and operating instructions, and the design licensing and operating experience of the authors have been utilized in performing the evaluations and developing recommendations.

The evaluations of the safety significance of the systems were based on the following factors:

- o System redundancy,
- o System operability requirements,
- o Effects of a system failure on safe shutdown and the integrity of the reactor coolant and containment boundary, and
- o Ability to inspect the system.

Recommendations specific to the system being evaluated are presented in each evaluation. Generic recommendations that affect all systems such as those concerned with operator training and procedure writing are presented in section 3.5. The recommendations presented in these evaluations are not necessarily intended to be regulatory requirements but rather aids in preventing and mitigating water hammers. Recommendations deemed significant enough to be regulatory requirments are listed in section 6.3.

4.1 Core Spray System

4.1.1 System Description

The core spray system is an ECCS system designed to remove decay heat from the core following a postulated design basis LOCA. The core spray system, in conjuction with the automatic depressurization system, is capable of cooling the core independently of any other core cooling system.

The core spray system consists of one or two independent loops. Each full capacity loop includes, one or two pumps, piping and valves that convey water from the suppression pool to a spray sparger in the reactor vessel above tr e core and associated controls and instrumentation. A low flow bypass line is provided for pump protection.

A full-flow test line allows water to be circulated to the suppression pool for system testing during normal plant operation.

One testable check valve and one motor operated valve in each loop isolate the core spray system from the reator coolant boundary during normal plant operation. Most core spray systems have a keep full system and venting provisions to assure that the pump discharge line is always full of water. The keep full system generally consists of a continuously running low flow "jockey" pump that supplies water to the core spray pump discharge line. The venting system generally consists of manually operated valves that vent the discharge line high points.

4.1.2 Water Hammer Evaluation

4.1.2.1 Event Review

Table 4.1-1 lists the core spray water hammer events reported in reference 1. The cause listed for most (nine of ten) is flow into a voided line. The other event (steam-bubble collapse) would be initiated by similiar conditions as will be discussed later. However, it should be noted when using this data for cause evaluation that only five of the ten events were observed. The previous occurrence of a water hammer was surmised for the other five events on the basis of observed damage.

4.1.2.2 Causes of Water Hammer

The following mechanisms can initiate a water hammer event in a system without proper keep fill operation or venting:

a. Flow into voided core spray pump discharge line - The relative elevations and valving arrangement can cause voiding of lines due to normal system leakage over a period of time. Draining can occur because the high point of the pump discharge lines are usually 60 to 90 feet above the suppression pool. The pump suction valves must remain open to minimize equipment operation following a core spray actuation signal. Thus, water can drain back to the pool either through a leaking pump discharge check valve or leaking or inadvertently open valves in the bypass test line. The resulting voids may approach vacuum conditions, containing small amounts of gas and water vapor. In this case, there is practically no cushioning effect due to air compression. Thus, large water hammer pulses following pump start can be generated when water is stopped by a closed or partially closed valve.

A properly sized keep fill system, that is continously in operation, will replace the drained water and prevent vacuum conditions from occurring.

Voids containing either air or steam, however, can be introduced into the piping through many means. This is especially true during shutdown or maintenance periods. Voids will not be eliminated by the use of a keep full system alone, but must be removed by venting. Water hammer can occur when a slug of water is accelerated through a void and suddenly stopped even if the void consists of compressed gas or steam.

b. Leakage past core spray check valve and injection valve - Hot water from the reactor can leak into the core spray pump discharge line, then flash into steam, creating a steam void. When the pump starts, the steam-bubble collapses, causing a water hammer in the discharge line. Plants having a keep fill system probably will not experience this kind of incident, as the incoming water will condense the water as it flashes, thereby preventing the formation of a steam bubble. However, if the leak causes the sum of the line pressure

and elevation head to become greater then the jockey pump discharge pressure, the pump will not provide water to the line. Systems with high pressure alarms provide, the operator with a warning of this situation. Also valve leakage can cause a steam bubble to occur downstream of an isolation valve. This portion of the piping is generally not serviced by the keep fill system.

4.1.3 Safety Significance

The core spray system has a high safety significance because it is an ECCS system and is connected to the reactor coolant boundary. For most postulated accidents ECCS redundancy is provided by the HPCI (HPCS) system and the LPCI systems. There are, however, some postulated accidents, which in combination with a single active component failure, would require the use of the core spray system. Several water hammer incidents rendered the core spray system inoperative.

The connection to the reactor coolant boundary would only be of significance during an incident when the isolation valve outside of containment was open, because the closed isolation valve and the flued head restraint at the containment would prevent the transmission of the water hammer forces to the line inside containment. However, failure of the line outside of containment in combination with a leaking or failed check valve would violate the integrity of the reactor coolant pressure boundary. Water hammer events reported in the core spray system have not had sufficient energy to damage piping. Based on the above discussion, the safety significance of the core spray system connection to the reactor coolant boundary is small.

4.1.4 Recommendations for Prevention and Mitigation

4.1.4.1 Design Phase

a. All core spray systems should be provided with keep fill system, preferably a jockey pump, that is continuously operating. This is currently standard for most plants.

b. A vent system should be provided that vents all portions of the piping between the pump discharge and the RCP boundary. All venting should be at the line high point. Any portion of piping that is isolated from the system high point by a valve should have a separate vent point.

c. The vent system should either be automatic, remotely operated or designed and located in a manner to maximize the ease of line venting.

d. A monitoring/ alarm system should be provided to detect voids.

e. The system should be considered inoperable when voids are present in the piping.

f. A thorough design review should be made that identifies all portions of piping in which voids or steam bubbles can form under any operating condition. The operating conditions reviewed should include valve alignments that might occur during maintenance or through operator error.

4.1.4.2 Operational Phase

a. Valves should be leak checked at every fueling outage. When projected valve leakage is deemed to be large with respect to the keep fill system or void formation, repairs or replacements should be made.

b. Any time the system is to be maintained or aligned in a manner not covered by existing procedures, an evaluation of water hammer and venting requirements should be made.

4.2 Evaluation of the RHR System (BWR)

4.2.1 System Description

The combined RHR system is a group of related subsystems that share common components to perform separate functions at different times during normal plant operation, shutdown, and following postulated accidents. The primary system function is to remove heat from the fuel and the Nuclear Steam Supply System (NSSS) during plant shutdown and refueling operations, and following a postulated loss of coolant accident (LOCA). The system consists of two or more heat exchangers and three or more pumps, depending on plant size, and required piping, valves, and controls. The components are arranged in three separate subsystems, located in the plant's lower elevations, that circulate the coolant water between the fuel, NSSS, suppression pool and the heat exchangers. The most severe system temperature and pressure operating conditions are 150 psia and 350°F, which occur during the plant shutdown cooling and steam condensing modes.

4.2.1.1 Operating Modes

The system has seven principal operating modes, plus a test mode, which are described below.

4.2.1.1.1 Shutdown Cooling

This function of the RHR system removes decay and sensible heat from the nuclear boiler system after reactor shutdown. When reactor pressure is reduced to approximately 150 psia, an interlock allows the operator to realign the RHR pumps, to pump water from one of the reactor recirculation loops, through the RHR Heat Exchanger (Hx) for cooling, and return it to the reactor vessel through the recirculation lines,the feedwater systems or vessel penetrations.

4.2.1.1.2 Reactor Vessel Head Spray

This subsystem is an extension of the RHR shutdown cooling mode. During reactor cooldown, water is pumped from the reactor recirculation system through the RHR Heat Exchangers (Hx) and cooled. The water is then

sprayed inside the top of the reactor vessel head and condenses the steam that forms there during cooldown. This action expedites the cooling of the vessel head and helps to lower temperature induced stresses.

4.2.1.1.3 Containment Spray

This mode of RHR operation condenses steam and removes heat from the containment to prevent overpressuring the containment. After operator actuation, suppression pool water is pumped through the RHR Hx's by the RHR pumps to either or both of the independent containment spray piping headers, which are installed in an elevated section of the containment.

4.2.1.1.4 Low Pressure Coolant Injection (LPCI)

This subsystem is part of the Emergency Core Cooling System (ECCS) network, and in conjunction with the HPCS, LPCS, and ADS systems, will restore and maintain the reactor vessel water level required for core cooling following a loss of coolant accident. When the reactor vessel pressure reaches the low pressure setpoint value, the RHR pumps automatically pump water from the suppression pool directly into the vessel. One pump is a spare. The RHR system is aligned in the LPCI configuration during normal plant power operation.

4.2.1.1.5 Fuel Pool Cooling

This mode of the RHR system supplements the regular fuel pool cooling system when it is necessary to provide additional cooling capability, such as when a complete core is unloaded and stored in the fuel pool. Generally, removable piping spools are installed to connect the two systems. Water from the fuel pool is pumped through the RHR Hx by the RHR pumps, cooled, and then returned to the fuel pool.

4.2.1.1.6 Steam Condensing

This RHR system mode is operator actuated, and is used when the reactor coolant system is isolated from the main condenser. It may be used in conjunction with operation of the RCIC system, to remove decay heat from

the reactor. Steam is drawn from the main steam line, reduced in pressure and directed to the shell side of the RHR Hx where it is condensed by cooling water. The condensate flows to the suction side of the RCIC pump, which returns it to the reactor vessel or to the suppression pool. Noncondensibles are vented to the suppression pool.

4.2.1.1.7 Suppression Pool Cooling

This operator actuated mode of the RHR system ensures that the suppression pool temperature does not exceed a predetermined limit after heat from the reactor has been transferred into the pool. The heat transfer could be from a LOCA, an SRV discharge, or exhaust from the HPCI or RCIC turbines. Suppression pool water is pumped through the RHR Hx by the RHR pumps, where it is cooled and returned to the suppression pool.

4.2.1.1.8 Isolation Condenser

The isolation condenser system, which is a design feature included only in older BWR plants, has been removed from the evaluation of the RHR system. The isolation condenser has different design and operational requirements than the RHR system and is not connected to it. Therefore, the isolation condenser will be evaluated separately in section 4.3.

4.2.1.2 System Interfaces

The subsystems interface primarily with each other; however, there are system connections to the reactor vessel, the NSSS, the feedwater system, the fuel pool cooling system and to the RCIC system. The RHR steam condensing mode, using the system heat exchangers, interfaces with the RHR pumped water subsystems. See figure 4-1 for typical steam and water interfaces. The steam-water interface during all power operation modes except steam condensing occurs at valves -13 and -6 on figure 4-1. During the steam condensing mode the interface is at valve -7.

4.2.2 Water Hammer Evaluation

4.2.2.1 Event Review

Three types of recorded water hammer events have been noted in the RHR system, namely, "Flow Into Voided Lines" (FIVL), "Steam Bubble Collapse" (SBC), and "Steam Water Entrainment" (SWE).

a. The most common type of recorded event is FIVL (12 of 24), which occurs primarily at high point locations in the piping of pumped water systems. FIVL events result from poor venting and filling practices and procedures; eleven of the twelve events occurred because of venting or filling problems, or both. The cause of the twelfth event was unknown. Voiding of the lines occurs primarily due to leakage of water from the system.

   BWR event item 42 is an example. Following an RHR pump start, flow in the fuel pool cooling line, entered a pipe section not completely full of water, causing a FIVL event. Pipe supports were damaged and a piping section was overstressed. Procedural deficiencies such as indadequate operating instructions and test procedures, and an installation that did not ensure proper venting were the causes of the event. The operator was unaware of the void when the pump was started.

   The installation of keep fill systems in almost all BWR plants has reduced the incidence of FIVL events, but some still occur.

b. The second most commonly recorded RHR system water hammer type is SBC (7 of 24). SBC can occur at steam water interfaces such as the junction of the RHR steam condensing and shutdown cooling lines, or where a pressure drop could cause hot water to flash such as in the RHR pump suction lines.

BWR event item 33 is an example of the latter case. The plant was near shutdown conditions and RHR surveillance testing was in progress. The reactor side of the RHR Hx is normally kept in wet layup. Over a two-month period the "B" Hx had partially drained due to valve leakage (see figure 4-2). After the pump suction valves were opened (2A, B, C), a steam bubble was formed in the pump suction header when the pressure of the hot fluid entering the partially voided "B" Hx was reduced. The operator was unaware of the steam bubble, the opening of valve 1A pressurized the suction header and the steam bubble collapsed, causing a water hammer.

When handling water at or close to saturation, any appreciable pressure drop can cause flashing, as happened in event 33. Subsequent pressurization will cause a SBC unless valves are opened very slowly, or a small bypass around the valve is used. Venting of the steam bubble would have also eliminated the event or reduced its severity.

c. The third type of recorded water hammer is SWE. Two SWE events occurred in the RHR Hx steam condensing inlet line during warmup of the HPCI steam line, which shares portions of piping with the RHR steam line. (BWR event 50). A gradual steam line warmup, slow HPCI valve opening and inspection of steam line drains could have prevented the event.

For a compilation of RHR system events and causes, see table 4.2-1.

4.2.2.2 <u>Water Hammer Causes</u>

The specific causes of the 24 category 1 water hammer events in the RHR system have been separated into two classifications; those that occur in subsystems where water is pumped, and those that occur in steam condensing subsystems.

a. Pumped Water Subsystems

In the RHR, Head Spray, Containment Spray, LPCI, Fuel Pool Cooling and Shutdown Cooling subsystems, twelve of the sixteen events involved flow into a voided line (FIVL), two resulted from steam bubble collapse (SBC), and two were from unknown causes. Eleven of the twelve FIVL events resulted from poor venting and filling practices. One of the two SBC events was caused by the collapse of steam that flashed when hot water entered an RHR Hx that had become voided because of valve leakage.

One SBC event was caused by steam leakage into the water side of the RHR steam condensing/suppression pool cooling interface. For example, it is possible for a water hammer to occur on RHR pump start (see figure 4-1) when initiating suppression pool cooling.

Isolation valve (-5) and vent valve (-13) can be leaking steam and bubbles could be formed and entrained at the junction of the RHR steam condensing and suppression pool cooling line near valve (-7). The pressurization induced on RHR pump start could collapse the steam bubbles and result in a water hammer.

b. Steam Condensing Subsystems

In the RHR steam condensing subsystem, six of the eight events (14, 15, 16, 17, 20, 25) involved SBC and two (41, 50) were caused by steam water entrainment (SWE). The six SBC events occurred at the Brunswick plants, and were caused by steam leakage through valves into the RHR Hx steam inlet piping and the subsequent steam bubble collapse when water was admitted to the line. Five of the six SBC events occurred at the Brunswick 1 plant. The events occurred over a period of fourteen months, and were indicative of unsatisfactory maintenance and operating procedures. None of the events occurred during the steam condensing mode.

The two SWE events were caused by condensed steam that entered the RHR steam inlet piping, during the HPCI steam supply line warmup of the common piping.

For the overall RHR system, eleven of the twelve FIVL events and six of the eight SBC and SWE events resulted from poor procedures or operator error or both. Additional causes (more than one for some events) were lack of venting, incomplete inspection, the need for a keep fill system, and system leaks primarily through valves. In only two events was inadequate design cited as a cause. All of the above reported causes, except those caused by the need for a keep fill system, involved plant operators and maintenance people.

- The implications of equipment malfunctions and maintenance related failures of components such as shutoff valves, steam traps, and check valves, are not fully considered as part of the causes of water hammer events by designers and plant operators. For example, in eight of the 24 RHR system events, valve leakage in water and steam systems was a major cause of the event.

  Venting, filling, and draining of piping were not sufficiently considered in plant procedures prior to subsystem operation, particularly during testing and sytem startup operations. In 15 of the 24 RHR system water hammer events, inadequate venting and/or filling was stated as a contributing cause.

  System components have been used in an unintended manner. For example, using gate valves for throttling flow results in valve damage and subsequent leakage. A review of operator practices indicates that this is a common occurrence in pumped water systems, such as the RHR, when throttling valves are not supplied in the proper system locations.

  Procedures and procedural controls do not fully include consideration of the causes and effects of water hammer. In 16 of the 24 RHR system water hammer events, procedures and procedural controls were stated as a cause of the event.

4.2.3 Safety Significance

Evaluations of the safety significance attached to the water hammer events for each of the RHR subsystems are presented below. Each subsystem has been categorized as having either a high, medium, or low safety significance.

4.2.3.1 RHR - Head Spray: Medium

The subsystem is nonsafety related. It is operator actuated, and is used only during plant shutdown to condense steam inside the RPV head. Subsystem failure would result in a longer, slower RPV cooldown; however, in an extreme case it could threaten the integrity of the primary coolant pressure boundary. Inspection can be done only during plant shutdown. The single head spray subsystem event item 46 caused a crack in the piping, disabling the subsystem.

4.2.3.2 RHR - Containment Spray: Medium

The system is nonsafety related in some BWR plants, and safety-related in others. It is operator initiated and used as a backup to pressure suppression in a pressure suppression type of containment. In dry containments it is used to reduce post-accident pressure. If the system is safety-related, there are two redundant containment spray subsystems, either of which can accomplish the system objective. The systems can only be inspected during reactor shutdown, and are tested during surveillance testing. There were four containment spray system water hammer events (37, 48, 49, 75, all FIVL). One event, No. 75, disabled one of the two subsystems, leaving one operable.

4.2.3.3 RHR-LPCI: High

The system is safety-related and is automatically actuated as part of the ECCS. There are three separate LPCI subsystems, two of which can accomplish the system objective. Failure of one subsystem due to a water hammer would cause loss of system redundancy but still permit

system function. A large water hammer could threaten the integrity of the reactor coolant boundary or primary containment penetrations. The single water hammer event did not cause piping damage.

The system can be inspected during operation, and is tested during surveillance testing. The single system event, No. 53, caused damage to p'pe supports and moved some piping but was not severe enough to damage piping. The system remained operable.

4.2.3.4 RHR - Fuel Pool Cooling: Low

The system is nonsafety-related. It is operator initiated, and is used only during plant shutdown as a backup to the Fuel Pool Cooling System whenever extra cooling capacity is needed. If the system fails to operate, there are other cooling means which can be used. Inspection and testing can be done at any time.

There were three system water hammer events (42, 76 45). Following two events radiographic inspection of the piping was performed. In the third event a valve was damaged. In all cases the system remained operable.

4.2.3.5 RHR - Shutdown Cooling: Medium

The system is safety-related. It is operator initiated, and is used for low pressure reacter heat removal. The system has spare capacity available which could be used in any mode if one of the subsystems failed to operate or was disabled. In the shutdown cooling mode, the system is connected to the reactor coolant boundary and attached to the primary containment. A severe water hammer event could threaten the integrity of those boundaries. No events have been severe enough to damage either boundary. The system is inspected during operation and is tested during surveillance testing. There were seven system water hammer events, one of which, No. 33, caused damage to a pump suction valve, putting the valve out of service. An alternate suction line was available.

4.2.3.6 RHR - Steam Condensing: Medium

The system is nonsafety-related. It is operator initiated, and is designed to be used when the reactor is on hot standby or being shut down and is isolated from the condenser. The system has two 50% capacity loops. Failure of one-half of the system could cause a slowdown in cooling the NSSS, or require reactor blowdown to the suppression pool. The system is inspected during plant operation and tested during surveillance testing. There were eight water hammer events that involved this system. None of the events occurred while the system was in the steam condensing mode. One of the events, No. 41, caused a possible Hx inlet piping overstress condition. In all cases the system was returned to service.

4.2.3.7 RHR - Suppression Pool Cooling: High

The system is safety-related and is part of the ECCS. It is operator actuated, following a postulated accident for long term cooling of the suppression pool, using the RHR pumps and heat exchangers. The system is used to keep the pool water below the technical specification limit of 170°F, during plant operation, after SRV discharge or during steam exhaust from HPCI or RCIC system operation. There are two separate loops, either of which can achieve the system objective if the other were disabled by a water hammer event. The two loops will be used alternatively during the long term cooling process. The system is connected to the primary pressure boundary and to containment penetrations. An extreme water hammer event, could threaten the integrity of those boundaries. The system is inspected and tested during plant operation. No water hammer events were noted during this mode of RHR operation.

4.2.4 Recommendations for Prevention and Mitigation

4.2.4.1 Design Phase

a. All liquid filled lines should be provided with a keep fill system, preferably a jockey pump that is continuously operating. This is currently standard for most plants.

b. A vent system should be provided that vents all portions of the liquid filled piping between the pump discharge and the RCP boundary. All venting should be at the line high point. Any portion of piping that is isolated from the system high point by a valve should have a separate vent point.

c. The vent system should either be automatic, remotely operated or designed and located in a manner to maximize the ease of line venting.

d. A monitoring/alarm system should be provided tc detect voids.

e. The system should be considered inoperable when voids are present in the piping.

f. A thorough design review should be made that identifies all portions of piping in which voids or steam bubbles can form under any operating condition. The operating conditions reviewed should include valve alignments that might occur during maintenance or through operator error.

g. Where compatible with the system design, provide slow closing and opening flow regulating valves in manually started pumped water systems, instead of gate valves, for throttling service.

h. Establish a leak reduction maintenance program for all system valves in the discharge lines of the LPCI, containment spray and head spray subsystems, where experience indicates water hammer events are likely to occur.

i. Special filling and venting procedures should be used following maintenance outages that empty portions of the piping.

4.3 Evaluation of the Isolation Condenser System (BWR)

4.3.1 System Description

The isolation condenser system removes decay heat from the reactor core when the main condenser is not available. The isolation condenser, located outside containment, consists of two tube bundles immersed in a large water tank. Make-up water is available from the condensate storage tank or station firemain storage tanks, pumped by either condensate transfer or fire pumps.

The isolation condenser system is included only in the earlier BWR plants; those with dry containment, and a few of the first pressure suppression containment designs. Plants using isolation condensers are no longer being designed or constructed.

When the isolation condenser is in operation, steam flows from the reactor, through the tubes of the condenser. After condensing it returns by gravity to the reactor. The isolation condenser is located high in the reactor building to facilitate natural circulation. The valves on the steam inlet lines are normally open so that the tube bundles are at reactor pressure. The isolation condenser is placed in operation by opening the closed condensate return valves to the reactor system. This is done automatically by a high reactor pressure signal or it can be done manually. During operation, the water on the shell side of the condensers will boil and vent to the atmosphere while condensing the steam inside the tube bundles.

Radiation monitors and alarms are provided on the shell vents so that in the event of abnormal radiation levels, the tube side of the heat exchangers can be isolated from the reactor by closing isolation valves. Two isolation valves are provided in the lines connecting the isolation condenser and the reactor. One of the isolation valves is located inside the primary containment, and the other is located outside.

The system interfaces with the nuclear steam supply system through connections to the reactor recirculation piping and to the reactor vessel.

4.3.2 Water Hammer Evaluation

4.3.2.1 Event Review

Steam water entrainment (SWE) was the single type of water hammer event (4 of 4) that occurred in the system. There was some conjecture that Steam bubble collapse (SBC) may have also taken place in the tank water caused by the rupture of tubes in the condenser at the same time in one event (55).

- o In three events (55, 56, 58) water entered the steam inlet line and impacted the piping and condenser after transient reactor high water level caused water carryover into the steam line to the condenser.
- o In one event (62), during system start, the lack of venting and improper drainage caused condensed steam to initiate a water hammer.

For a list of isolation condenser events and causes see table 4.3-1.

4.3.2.2 Water Hammer Causes

Of the four water hammer events occurring in the Isolation Condenser Systems (all SWE), two occurred during plant power operation (55, 62) and two during plant shutdown (56, 58).

In three of the events (55, 56, 58) a reactor high water level transient caused water carryover to enter the steam inlet line.

Based on the reported events, the system is susceptible to water hammers caused by transient reactor high water level during isolation condenser operation. For example, in event 58, reactor normal water level had been maximized in accordance with TMI experience and requirements.

Following a scram, when the system was actuated, slugs of water entered the steam inlet piping and caused the event. As a result, instructions directing operators to maximize water level were revised.

Event 55 may have been a hydraulic transient caused by a stress corrosion induced condenser tube failure rather than a water hammer. However, a surge in reactor water level caused water to enter the isolation condenser inlet line. Damage was noted in both the condenser and its inlet line.

Three events 55, 56, and 58 all involved water entering the isolation condenser and occurred at one plant (Millstone 1) over a period of almost four years. No other plants have reported isolation condenser incidents. This indicates that there is a need for Millstone 1 to review their operating procedures with respect to isolation condenser operation and high reactor water levels. It should be noted that except for the damage caused by the tube rupture in event 55 which was attributed to stress corrosion, no damage was noted in an isolation condenser event. The only design related event 62 occurred during power testing. The design faults were corrected and the system has run for twelve years without accident.

The isolation condenser system is highly susceptibile to hydraulic transients. The system undergoes a series of mild hydraulic transients each time it is operated. Actuation of the system during high water level in the reactor vessel (above the isolation condenser steam supply connection) will result in a slug of water entering the steam filled piping causing momentum, impingment and water hammer forces impacting elbows and the condenser tube sheet. These forces can be more severe for an automatic actuation than for a manual one, because the rate of valve opening is not controlled during automatic initiation. If condenser tubes are weakened from stress corrosion, they can rupture, allowing a large steam bubble to form in the water side. The bubble collapses and forms again. This "chugging" can cause large vibrations and noise in the tank.

4.3.3 Safety Significance - Medium

The isolation condenser system is often safety-related. It serves as a replacement heat sink for the main condenser for decay heat removal after reactor scram. For wet containment plants this system can be replaced by reactor blowdown to the suppression pool and the RHR system cooling (all reported events were in wet containment plants). The system is connected to the reactor coolant pressure boundary and penetrates the containment. The RCP boundary and penetrations could be threatened by a severe water hammer. However, none of the water hammers damaged piping.

4.3.4 Recommendations for Prevention and Mitigation

4.3.4.1 Design Phase

No recommendations, since isolation condensers are no longer being considered for use.

4.3.4.2 Operational Phase

a. Check cold to hot movements of plant components, particularly piping and supports. Adjust supports as needed to reduce vibration and eliminate low spots in drain lines.

b. Procedures should be reviewed with respect to isolation condenser operation and high water levels.

4.4 Evaluation of High Pressure Coolant Injection (HPCI) System

4.4.1 System Description

The HPCI system consists of a steam turbine driven pump along with appropriate piping, valves, and controls and is part of the ECCS. It is designed to remove heat from the reactor following a postulated loss-of-coolant accident (LOCA) which does not rapidly depressurize the reactor. The HPCI system operates until the reactor pressure is below the pressure

at which either LPCI or the Core Spray system can maintain core cooling. If HPCI is unavailable, the Auto Depressurization system in conjunction with Core Spray or LPCI can provide the required core cooling.

4.4.1.1 Steam Turbine and Steam Lines

Steam, drawn from upstream of the main steam line isolation valves, drives the HPCI turbine. The two isolation valves in the steam line to the HPCI turbine are normally open to keep piping to the turbine at elevated temperatures and to permit rapid startup of the HPCI system.

To prevent the HPCI system supply line from filling with water, a condensate drain pot is provided upstream of the HPCI turbine stop valve. The drain pot normally routes condensate to the main condenser through an orificed line. The drain pot contains a level switch. A drain pot high level signal opens a bypass line to reduce the drain pot level and actuates an alarm.

Exhaust steam from the HPCI turbine is discharged to the suppression pool. The turbine exhaust line contains check valves to prevent back flow from the suppression pool. A drain pot at the low point in the exhaust line collects condensate which is discharged to a barometric condenser or the suppression pool.

4.4.1.2 Pump and Pump Discharge Lines

The HPCI system pumps water from either the condensate storage tank (normal alignment) or the suppression pool to a feedwater line in the steam tunnel. A minimum flow bypass to the suppression pool, is provided for pump protection. A system test line recirculates the pump discharge to the condensate storage tank during system testing.

The pump discharge line is provided with a vent system consisting of manually operated valves that vent the discharge line high points. Some of the HPCI systems are provided with a keep fill system that generally consists of a continuously running low flow "jockey" pump that supplies water to the pump discharge line to compensate for line leakage.

In BWR 5 and 6 plants, the steam turbine driven HPCI system has been replaced with an electric motor driven HPCS system. Thus, water hammer incidents associated with steam lines can not occur in these plants.

4.4.2 Evaluation of Water Hammer Events

4.4.2.1 Event Review

Table 4.4-1 presents a summary of HPCI system water hammer events reported in reference 1. The cause listed for most events (twelve of twenty) is steam-water entrainment. The other events were caused by steam-bubble collapse (four) and flow into-voided-line (three), and unknown (one). When using these data for cause evaluation, it should be noted that water hammer was actually observed in only ten out of nineteen cases. The previous occurrence of water hammer was surmised for the other events on the basis of observed damage.

4.4.2.2 Causes of Water Hammer

Possible mechanisms of observed water hammer occurrences are discussed below.

4.4.2.2.1 HPCI Turbine Steam Supply Line Water Entrainment

During normal reactor operation, both the inboard and outboard isolation valves are kept open to maintain steam in the line up to the closed stop valve at the turbine. The drain pot located upstream of the turbine stop valve routes condensed steam to the main condenser through the outlet steam trap. When a high drain pot level occurs, the steam trap bypass valve is automatically opened by a level switch. During HPCI turbine operation, the drain pot valve remains closed.

a. Failure of Steam Supply Line Drain System

The drain pot can fail to drain through the outlet steam trap because of plugging of the steam trap orifice. If the drain pot high level switch fails to open the steam trap bypass valve, water will accumulate in the drain pot and steam line. Under these

conditions, initiation of steam flow can cause a steam water entrainment water hammer. During normal HPCI standby conditions, the drain pot will be nearly empty. The level switch and bypass valve are rarely cycled. Such infrequent usage is conducive to the level switch or valve sticking. If the level switch is inoperative, a high water level can occur in the drain pot without any indication to the operator. Events 9, 10, 12, 68, and 69 are covered by this scenario.

b. Sequence of Isolation Valve Operation

There are no provisions for draining the steam line upstream of the outboard isolation valve. Therefore, if an isolation valve is closed, water will accumulate in the line upstream of the valve. Normally, the outboard valve is opened; then the inboard isolation valve is opened slowly for gradual admission of steam. The outboard isolation valve has a seal-in feature that causes the valve to open or close fully; thus the valve cannot be opened gradually. When the outboard valve is opened, with the inboard valve fully open, the steam flow rate builds up rapidly. Entrained liquid in the line flows rapidly through the line and is suddenly stopped at the first obstacle (the turbine stop valve) and large water hammer forces are generated capable of causing significant damage. Events 8, 30, 40, and 51 are covered by this scenario.

4.4.2.2.2 HPCI Turbine Exhaust Line

The turbine steam exhausts into the suppression pool after passing through two check valves, one located outside the drywell in a horizontal piping run inside the containment boundary and the other located inside the drywell in a vertical piping run. A drain pot at the low point in the exhaust line upstream of the check valves collects condensate which is discharged to a barometric condenser or the suppression pool through a drain pot valve. A level switch automatically opens the drain pot valve on high level.

a. Failure of Steam Exhaust Line Drain System

If the level switch fails to open the drain pot valve on high level, condensed steam will accumulate in the exhaust line. Under such a circumstance, the flow of steam will move this accumulated water, thereby causing a water hammer. This event is less severe than a similar one in the steam supply line, because the exhaust pipe ends in the suppression pool which has a free surface. Event 24 is covered by this scenario.

b. Rapid Steam Condensation

After turbine shutdown, rapid steam condensation in the exhaust line, can create a vacuum condition, drawing a water slug from the suppression pool into the exhaust line. The water slug, traveling at a high velocity, impacts the check valve disc resulting in a fast valve closure that can cause a water hammer. The resulting pressure differential can cause a rupture of the turbine exhaust rupture disc. The short operational periods during testing (less than two minutes) are particularly conducive to condensation, because the turbine housing and exhaust line inside walls remain cool and provide a subcooled condensing surface for the stagnant steam remaining in the pipe and turbine after shutdown. Events 7, 11, 61, and 81 are covered by this scenario.

c. Water Entrainment

The vacuum conditions, discussed in b above, can cause a slug of water to be trapped between the line check valves. On a subsequent turbine start, the water slug entrained between the two check valves can be propelled past the 90 degree elbow in the exhaust line to impact the suppression pool water interface causing a water hammer and reaction forces at the piping elbows. Event 2 appears to be covered by this scenario.

4.4.2.2.3 HPCI Turbine Gland Seal Condenser Steam Inlet Line

The labyrinth seal steam from either end of the turbine exhausts into a line which drains into the gland seal condenser feed line. The turbine gland seal leak-off drain pot drains both the feed line condensate and the turbine lower casing drains to the suppression pool through a thermostatic trap. If the drain has a high level, an air operated valve automatically opens and drains the pot to the gland seal condenser.

Failure of the turbine gland seal leak off drain pot to remove all the water in the line can result in accumulation of liquid in the gland seal condenser inlet line. Subsequent opening of the isolation valve can result in rapid condensation of steam in the line resulting in water hammer. Event 82 is covered by this scenario.

4.4.2.2.4 Pump Discharge Line

In some plants, the relative elevations and valving arrangement can cause voiding of lines due to normal system leakage over a period of time. The draining problem is primarily due to the difference between the elevation at the pump suction and the pump discharge line. The pump suction valves to the condensate storage tank must remain open to minimize the number of valves to be operated following an actuation signal. Thus, water can drain back from the discharge line to the source through a leaking check valve or leaking or inadvertently open valves in the bypass test line or leaking through the minimum flow line. The resulting voids may approach vacuum conditions, containing small amounts of dissolved gas and water vapor. In this case there is practically no cushioning effect due to air compression. Thus, large water hammer pulses following pump start can be generated when a slug of water is accelerated through a void and suddenly stopped. Draining to the source is not a problem in plants where the condensate storage tank level is higher than the high point in the discharge piping.

A properly sized keep fill system, that is continuously in operation, replaces the drained water and prevent vacuum conditions from occurring. Low pressure alarms alert the operators of excessive leakage.

Voids containing either air or steam, however, can be introduced into the piping in many ways. This is especially true during shutdown and maintenance periods. Voids will not be entirely eliminated by a keep fill system, but must be removed by periodic venting. Events 13, 19, 27, and 29 are covered by this scenario. The frequency of these incidents in HPCI, is considerably less than in the Core Spray system. This is to be expected because the elevation difference between the pump suction and the pump discharge line is less than that of the Core Spray system and the line sizes are smaller. Also for some plants, the condensate storage tank level is higher than the high point in the discharge piping.

4.4.3 Safety Significance

By virtue of being a part of the ECCS and a part of the reactor coolant pressure boundary (the steam side is connected upstream of the main steam line isolation valves), the HPCI system, has a high safety significance. If HPCI is inoperable, the Auto Depressurization system will provide adequate coverage through depressurization and subsequent use of Core Spay and Low Pressure Coolant Injection sytem. However, certain postulated accidents in combination with a single component failure require the use of the HPCI. It should be noted that some events (rupture disc activation, events 7, 11, and 61) rendered the system inoperative.

The connection to the reactor coolant pressure boundary would be of significance, if both isolation valves failed to respond to a containment isolation signal, subsequent to damage caused by water hammer or if the line was damaged upstream of the valves. However, no incident has occurred that caused piping damage. Based upon the above information the safety significance of being connected to reactor coolant pressure boundary is low.

4.4.4 Recommendations for Prevention and Mitigation

4.4.4.1 Design Phase

A design review should be performed that identifies all portions of piping in which voids or steam bubbles can form or collapse under any operating condition, including valve alignments that might occur during maintenance or through operating error.

4.4.4.1.1 Pump Discharge Line

a. All high pressure coolant injection systems should be provided with a keep fill system, preferably a continuously operating jockey pump. For most plants this feature is already provided.

b. A vent system should be provided that vents all portions of the piping between the pump discharge and isolation valve at the connection with the feedwater piping. All venting should be at the line high points. Any portion of piping that is isolated from the system high point by a valve should have a separate vent point.

c. The vent system should either be automatic, remotely operated, or designed and located for easy access and manual operation.

d. A monitoring/alarm system should be incorporated to detect system leakage and void formation.

4.4.4.1.2 Steam Supply Line

a. Normally the inboard steam line isolation valve is used for throttling. To cover all possible operating situations, the seal-in feature (if any exists) should be removed from the outboard valve opening circuit logic. This will permit slow opening and consequently gradual draining of entrained water. As an alternative, interlocks should be provided such that the inboard valve cannot be opened, unless the outboard valve is fully open. Another alternative would

be to provide for adequate and prompt draining of all portions of the steam supply line for all isolation valve positions. Removing the seal-in feature will be the least expensive solution, as it only involves disconnecting control wirings. Provision of interlocks, will be the best solution because it will minimize the probability of operator error.

b. Suitable provisions should be made to allow drain pot level switch maintenance during normal plant operation.

c. Adequacy of drain pot sizing should be reviewed. It may be advisable to increase the size of the drain pot or place two drain pots in parallel.

4.4.4.1.3 <u>Steam Exhaust Line</u>

a. Vacuum breakers should be incorporated both on the upstream and downstream sides of the exhaust line stop/check valves.

b. It is desirable to install a condensing sparger at end of the exhaust line in the suppression pool to reduce noise and vibration.

4.4.4.1.4 <u>HPCI Turbine Gland Seal Condenser Steam Inlet</u>

The gland seal leak-off drain pot should be sized adequately and should be designed for ease of maintenance during normal plant operation.

4.4.4.2 <u>Operational Phase</u>

a. Procedures should be reviewed for prcper warmup of the steam inlet line to minimize steam condensation.

b. Valves should be leak tested at every refueling outage. When projected valve leakage is deemed to be large with respect to the keep fill system capacity, repairs should be made.

c. The drain pot level switch and steam trap bypass valve should be exercised periodically.

d. Any time the system is to be maintained or aligned in a manner not covered by existing procedures, an evaluation of potential water hammer conditions and venting requirements should be performed.

4.5 Reactor Core Isolation Cooling System

4.5.1 System Description

The reactor core isolation cooling system (RCIC) provides makeup water to the reactor vessel, following a reactor vessel isolation from the main condenser accompanied by a loss of feedwater flow. The system is used to cool down and depressurize the plant to the point where the shutdown cooling mode of the residual heat removal (RHR) system can be utilized. If for any reason, the RCIC system is incapable of supplying sufficient flow for core cooling, the emergency core cooling systems (HPCI, ADS, CS, LPCI) are available to provide the required reactor coolant pressure boundary protection.

The RCIC system consists of a steam turbine driven pump unit, associated valves and piping capable of delivering makeup water to the reactor vessel. The steam is supplied to the turbine from a point upstream of the main steam line isolation valves. The pump is normally aligned to the condensate storage tank but can take suction from the residual heat removal system heat exchangers or the suppression pool. The pump discharges into a feedwater line outside containment on earlier model plants and into the reactor head spray line on later model plants. The pump discharge line is provided with venting provisions consisting of manually operated valves that vent the discharge line high points.

A full flow test line to the condensate storage tank and a minimum flow line to the suppression pool are also provided. The minimum flow valve automatically opens on a low flow signal and automatically closes on a high flow signal.

The two isolation valves in the steam line to the RCIC system turbine are normally open to keep piping to the turbine at main steam temperature and to permit rapid startup of the RCIC system. To prevent the steam supply line from filling with water, a condensate drain pot is provided upstream of the turbine stop valve. The drain pot normally routes condensate to the main condenser.

Exhaust steam from the turbine is discharged to the suppression pool through a line containing a check valve. A drain pot at the low point in the exhaust line collects condensate which is discharged to a barometric condenser. In most plants, the exhaust line contains a vacuum breaker to prevent the formation of a vacuum from steam condensation.

4.5.2 Evaluation of Water Hammer Events

4.5.2.1 Event Review

Table 4.5-1 presents a summary of reactor core isolation cooling system water hammer events reported in reference 1. Two events were observed, one was a steam water entrainment water hammer, the other event was a pump cavitation incident.

4.5.2.2 Water Hammer Causes

Possible mechanisms of observed water hammer occurrences, because of system design, equipment orientation, and operating procedure are investigated here.

4.5.2.2.1 Steam Side Event

After turbine shutdown, unless the line contains a vacuum breaker, rapid steam condensation in the exhaust line can create a vacuum condition. The vacuum can cause water from the suppression pool to be drawn into the exhaust line. The water slug, traveling at a high velocity, impacts the check valve disc resulting in fast valve closure, thereby causing water hammer due to sudden stoppage of the slug.

Short operational periods (less than 2 minutes) are particularly conducive to keeping the turbine housing and exhaust line inside walls cool thus providing a subcooled surface for condensation of steam remaining in the pipe and turbine after shutdown. Subsequent restart could cause the exhaust steam to expel the water slug and cause large forces when the slug impacts the suppression pool. Event 60 occurred due to this scenario.

4.5.2.2.2 Water Side Event

If the pump is started with the test return line valves fully open, the required discharge head is much lower than at the normal operating point. Under such conditions, the pump can cavitate. This could create excessive loading conditions and can cause severe damage to individual pump stages, particularly, if their axial movements have not been restrained. Event 79 is covered by this scenario. This event appears to be a pump cavitation event and not a water hammer event.

The various water hammer events that have occurred in the HPCI system (section 4.4.2), can conceivably happen in RCIC system. However, only one water hammer incident and one pump cavitation incident has been observed in the RCIC system. The smaller line size and length, and the recency of introduction of the RCIC system may be the reasons for the fewer incidents.

4.5.3 Safety Significance

By virtue of having to provide a core cooling function and being a part of the reactor coolant pressure boundary (the steam side is connected upstream of main steam line isolation valve), the reactor core isolation cooling system has a moderate impact on overall plant safety. If RCIC is inoperable, ECCS systems will provide at least two levels of redundancy.

The connection to the reactor coolant pressure would only be of significance, if both isolation valves failed to respond to the containment isolation signal subsequent to damage caused by water hammer. However

no water hammer incident involving the steam supply line has been observed. Based upon the above reasoning, the safety significance of being connected to reactor coolant pressure boundary is low.

4.5.4 Recommendations for Prevention and Mitigation

4.5.4 1 Design Phase

4.5.4.1.1 Steam Line

For relieving vacuum conditions in the turbine exhaust line, vacuum breakers should be incorporated on the downstream side of the exhaust line check valve.

4.5.4.1.2 Water Line

a. The reactor core isolation cooling system pump discharge side should be provided with a keep-fill system, preferably a continuously operating jockey pump.

b. To simulate normal operational discharge head and flow conditions during testing, a restricting orifice should be installed on the full flow test return line to the condensate storage tank.

c. Adequate provisions should be made for venting all portions of the piping between the pump discharge and connection with feedwater piping. All venting should be at the line high point. Any portion of piping that is isolated from the system high point by a valve shall have a separate vent point.

4.5.4.2 Operational Phase

a. Any time the system is to be maintained or aligned in a manner not covered by existing procedures, an evaluation of potential water hammer conditions and venting requirements should be performed.

b. The RCIC pump should not be started with the test return valve to condensate storage tank fully open. Because a minimum flow line has been provided, there is no danger of overheating if the pump is started against a closed discharge valve.

4.6 BWR Main Steam System

4.6.1 System Description

The main steam system supplies steam from the reactor vessel to the turbine-generator system. The system consists of main steam piping, safety-relief valves (S/RV), main steam line flow restrictors, turbine stop valves (TSV) and main steam isolation valves (MSIV). The steam bypass system bypasses flow to the condenser to control steam pressure during load rejections, reactor heatup, turbine start-up and reactor cooldown.

4.6.2 Evaluation of Water Hammer

4.6.2.1 Event Review

Four transient events that occurred in the main steam system are discussed below. Additonally, one event occurred in S/RV discharge line and one event in the steam bypass line. Of these six events, only three can be considered unanticipated water hammer events.

The water hammer events are summarized in table 4.6-1. The following comments are the result of a review of these events:

- Four events (events 52, 63, 65, and 71) occurred in the main steam lines. Events 52 and 71 were caused by turbine stop valve closure, resulting in piping support damage due to inadequate support design. In event 63, condensate in the main steam lines caused water hammer when the isolation valves were opened during the power escalation testing. The condensate in the line was caused by inadequate line drainage. A steam-water entrainment type water hammer probably occurred in event 65, when an isolation valve was suddenly opened

during start-up valve timing test. In event 65, a valve operator component was damaged due to inadequate valve design and poor line warmup procedure.

- Event 18 occurred in the S/RV discharge line. In event 18, it has been postulated that a sequence of S/RV openings resulted in damage to snubbers on a discharge line in the drywell. The postulated scenario is that, following a reactor scram, reactor pressure increased to the point that an S/RV opened. As a result of the S/RV opening, the manifold pressure increased causing the water initially in the adjacent discharge line sharing the same exhaust header to be pushed upward. The safety relief valve on the line in which the water was pushed upward was then actuated. The expulsion of this water slug from the discharge line caused high loads which resulted in damage to the snubbers. Analysis indicated that damage to the snubbers should not have occurred if they were functional.
- Event 67 occurred in the steam bypass line. Steam hammer was caused by cycling of the bypass valves due to an out-of-calibration control, resulting in damage to snubbers.

4.6.2.2 Water Hammer Causes

Of the six main steam events, three (events 63, 65, and 67) can be considered unanticipated water hammer events for which preventive measures are required.

Two events (events 52 and 71) were anticipated steam hammer events. Events 52 and 71 were caused by turbine stop valve closure.

Event 67 was attributed to cycling of the bypass valves due to an out-of-calibration control.

Event 63 was an unanticipated water entrainment in steam lines, which was caused by inadequate drainage.

Event 65 was considered a steam-water entrainment type water hammer due to sudden opening of the main steam isolation valve, which allowed hot steam to flow into insufficiently warmed downstream line.

Event 18 was a hydraulic transient in the S/RV discharge line with an entrained water slug in the discharge line. The pipe support system was determined through analysis to be able to withstand the loads, if proper inspection and maintenance procedures were followed.

o Two steam hammer events (events 52 and 71) were caused by sudden valve closure and inflicted damage to pipe support system components because of inadequate design. These events occurred just prior to or shortly after commercial operation, indicating that inadequate pipe support designs were found early and corrected. Rapid closure of the turbine stop valves and isolation valves is necessary in the main steam lines. Therefore, the design of the pipe support system components for the main steam lines should include consideration of steam hammer dynamic loads due to valve closure.

  In event 67, steam hammer was caused by valve cycling due to an out-of-calibration valve control. To prevent the control valve instabilities and fluctuations, inspection and calibration procedures should be followed to detect and correct control valve malfunctions.

  In event 65, water hammer due to poor line warmup procedure caused a valve operator component to fail. Inadequate valve design contributed to the valve failure. Event 63 was caused by water entrainment is steam lines due to inadequate drainage. Water hammer due to water entrainment should be eliminated from the main steam by proper draining design and operating procedure for warmup.

  Event 18 was not a true water hammer but a relief valve discharge hydraulic transient with a water slug in the S/RV discharge line and resulted in damage to the snubber due to inadequate inspection and maintenance procedures. It should be noted that BWR S/RV discharge lines normally contain a water slug at the exit.

4.6.3 Safety Significance

Safety significance of the main steam system is considered high. The main steam lines are safety-related up to and including the outboard isolation valves. The main steam system must be isolated to prevent radiological release during reactor accidents. Any pipe rupture upstream of the inboard isolation valve will result in a LOCA. The safety relief valves are a safety-related means of reducing the reactor pressure and removing the reactor heat. The reported water hammer events in the main steam system have resulted in damage to the pipe support system components due to either inadequate component design or improper inspection and maintenance procedures. One event caused damage to a valve operator component due to inadequate design. No events occurring in the main steam lines have been severe enough to cause piping damage, nor was any damage noted upstream of the MSIV.

4.6.4 Recommendations for Prevention and Mitigation

4.6.4.1 Design Phase

The following preventive measures should be implemented during the design phase:

a. The design of pipe support system components in the main steam lines should consider steam hammer dynamic loads due to valve closure.

b. Develop inspection and maintenance procedures for the pipe support system components in the main steam system, including S/RV discharge lines.

c. Develop inspection and calibration procedures to detect out-of-calibration valve controls and make necessary corrections as required.

d. Select valves with the components that are proven to be compatible with the duty cycle and service requirements.

e. Review the system design and operating procedures for the possibility of water entrainment in steam lines.

4.6.4.2 Operational Phase

Pipe support system components should be inspected periodically for evidence of wear and damage. Appropriate repairs or replacements should be made when required.

4.7 BWR Feedwater System

4.7.1 System Description

The major components of the feedwater system are feedwater pumps, feedwater regulating valves, and high pressure heaters. Condensate is pumped from the low pressure heaters by the feedwater pumps into the reactor vessel. The feedwater flow passes through the feedwater regulating valves which are automatically control the reactor water leve' [illegible] ut 50 percent of the plants use turbine speed control for feedwa[illegible] low control. During startup, the low flow bypass valves are utilized to control feedwater flow rate. At a flow rate typically about 20 percent of full flow rate, the control is transferred to the feedwater regulating valves and the low flow bypass valves are closed. Feedwater leaving the valves at a controlled rate, enters the final stages of the heating cycle (high pressure heaters) before entering the reactor vessel.

4.7.2 Evaluation of Water Hammer

4.7.2.1 Event History

Three unanticipated water hammer events occurred in the feedwater system. The water hammer events are summarized in table 4.7-1. In events 73 and 80, water hammer was triggered by feedwater regulating valve instability due to inadequate operator and controller design and/or poor inspection and maintenance procedures. Although damage was fairly extensive in both plants, the plant safety was not affected (see table 4.7-1). In event 35, water hammer was triggered by the feedwater

regulating valve closure due to malfunction of the control system. Damage was also fairly extensive in this event but the feedwater line was not damaged nor was plant safety effected. It is noted that the plants using the turbine speed control for feedwater flow control did not report water hammer.

4.7.2.1 Event Causes

The three events occurred after start of commercial operation. A logical explanation of the feedwater regulating valve instability and malfunction is that the valve operator and controller of these valves may have gradually deteriorated over the years. Therefore, periodic inspection, testing and maintenance procedures may be necessary to ensure that the valve operator and controller are performing properly. One additional consideration is inadequate design of the valve operator and controller. Plants have been plagued with excessive control system hunting and continuous valve cycling for many years. Many plants are now installing new types of regulating valves with improved performance. Some plants are also using a one element rather than a three element controller under steady-state conditions to reduce valve cycling. Three element controller causes valve cycling at low feedwater flows because the steam flow signal is not accurate enough at low flows; thus causing errors in three element control.

4.7.3 Safety Significance

The safety significance of the feedwater system is considered high. Loss of feedwater will lead to an emergency shutdown of the reactor. Pressure waves or vibrations have potential to damage check valves and the connected safety-related RCIC and HPCI lines and to overstress the reactor pressure vessel nozzles on the feedwater line. The reported water hammer events in the feedwater system have resulted in fairly extensive damage, although none of the events involved safety of the plants. All of the events were attributed to feedwater valve instability or malfunctions, resulting from valve operator and controller malfunctions. Therefore, the preventive measures should be taken to alleviate the valve operator and controller malfunctions.

4.7.4 Recommendations for Prevention and Mitigation

4.7.4.1 Design Phase

Three unanticipated events occurring in the feedwater system were attributed to the feedwater regulating valve instability and malfunction, which caused fairly extensive damage. Therefore, it is appropriate to take preventive measures during the design phase. The occurrence of spurious signals in a plant is not uncommon. The feedwater control system should be designed to prevent harmful response to such a signal. The following preventive measures should be implemented during the design phase:

a. Select feedwater regulating valves that are properly sized, have balanced trim and are resistant to internal damage.

b. Select a valve operator that can meet the feedwater regulating valve performance and response requirements. Installation of a hydraulic dampener on the valve stem should be considered for solution of the operator problem. Maximum valve operation speed under any signal should be limited to five percent/second.

c. Develop periodic inspection, testing and maintenance procedures to ensure good performance of the regulating valve operator and controller.

4.7.4.2 Preoperational Phase

The preventive measures that were incorporated into the feedwater system during the design phase should be verified during the preoperational phase. The feedwater regulating valves should undergo operability checks prior to being placed in service. Valve operator testing should be performed under all conditions to demonstrate that no controller action, including those from spurious signals, can cause water hammer or excessive vibrations.

4.7.4.3 Operational Phase

The feedwater regulating valve operator and controller should be periodically inspected and tested to ensure that they are in good working condition. Appropriate repair or replacement should be made when required.

4.8 Reactor Water Cleanup System

4.8.1 System Description

The reactor water cleanup system (RWCU) removes various impurities from the reactor water and provides a means for water removal from the primary system during startup, shutdown or refueling.

Primary water from the reactor recirculation pump suction line and the reactor vessel is pumped through regenerative and nonregenerative heat exchangers where it is cooled, and then through the filter-demineralizer units. The flow then continues through the shell side of the regenerative heat exchanger where it is heated before returning to the reactor through the feedwater line. During times of increasing water volume, excess water is removed from the reactor by blowdown through the cleanup system to the main condenser or alternatively to the radwaste system.

4.8.2 Water Hammer Evaluation

4.8.2.1 Event Review

Table 4.8-1 presents a summary of the only RWCU water hammer event reported in reference 1. It should be noted that water hammer was not actually observed in this case. The previous occurrence of water hammer was surmised on the basis of observed damage.

4.8.2.2 Cause of Water Hammer

a. A possible mechanism of water hammer occurrence based on reported events is discussed below:

- o During standby periods, reduced water temperatures can cause shrinkage and create voids in the system. Subsequent rapid opening of isolation valves can create a flow-into-voided-line water hammer. The resulting forces can damage the piping and adjacent components. Event 38 could have been caused by this scenario.

b. The snubber damage noted in event 38 could have been caused by line vibration induced by improperly installed valves or inadequate pipe supports. The crack in the affected pipe might have resulted from vibration or an existing material defect or both.

4.8.3 Safety Significance

The reactor water cleanup system has no safety function. However, the system is connected to reactor coolant pressure boundary and the observed event damaged this part of the system. Based on the above reasonings, the overall safety significance is to be rated as moderate.

4.8.4 Recommendations for Prevention and Mitigation

4.8.4.1 Design Phase

The isolation valve control should permit gradual opening.

4.8.4.2 Operational Phase

a. While initiating the reactor water cleanup system, the isolation values should be opened gradually, to avoid a sudden surge of water flow.

b. Valves should be leak tested periodically. When projected value leakage is deemed to be large, repairs should be made.

c. When the system is to be maintained or aligned in a manner not covered by existing procedures, an evaluation of potential water hammer conditions and venting requirements should be performed.

4.9 BWR Condenser System

4.9.1 System Description

The main condenser is the steam cycle heat sink. During normal operation, it receives and condenses main turbine exhaust steam, feedwater pump turbine exhaust steam, and turbine bypass steam. The main condenser is also a collection point for other steam cycle miscellaneous flows, drains, and vents. Hot well level controls provide automatic makeup or rejection of condensate to maintain a normal level in the condenser hot wells. The non-condensable gases contained in the turbine exhaust are collected in the condenser and removed by the condenser air removal system. The condensate pumps take suction from the condenser hot wells and pump water through the trains of heaters to the feedwater system.

4.9.2 Evaluation of Water Hammer Events

Four events occurred in the condenser system. These events are summarized in table 4.9-1. Two out of the four events (event 77 and 78) occurred in the circulating water line to the condenser. Event 77 was the first of two similar events, caused by inadvertent butterfly valve closure in the circulating water line during maintenance work. In event 77, a rubber expansion joint in the condenser water box ruptured, flooding the condensate pump room and damaging the RHR service water pumps and motors and other equipment. Event 78 was the second of the two similar events again caused by inadvertent valve closure during maintenance work. Events 54 and 72 occurred in the steam bypass line due to bypass valve opening. In both of these events, water hammer was caused by condensate accumulation in the header (or spargers). In event 54, the end cap of steam bypass header in the main condenser failed. In event 72, the condenser internal spargers and baffle were broken.

All of the four events appear to have been water hammer events, which occurred prior to or shortly after commercial operation. The cause of these events were detected early and corrected. Two of the events (events 77 and 78) were caused by inadvertent valve closure in the circulation water line during maintenance. The other two events (events 54 and 72) resulted from steam bypass valve opening because of steam-water entrainment in the header (or spargers) located inside in the condenser.

4.9.3 Safety Significance

The condenser system has no safety related function. However, the failure of the main condenser should not preclude operation of any essential system.

4.9.4 Recommendations for Prevention and Mitigation

The condenser system is not safety related. However, events 77 and 78 have safety significance, because a rubber expansion joint rupture in the condenser caused damage to engineered safety equipment due to flooding. Flooding of safety equipment is a safety issue. The Systematic Evaluation Program (SEP) and pipe rupture criteria (SRP 3.6.1 and BTP ASB-3-1, references 5, 6, and 7) require analysis for flooding caused by postulated pipe ruptures. Protection is provided by level switches that are installed to cause trips of the circulating water pumps should pit flooding occur. Events 54 and 72 in the condenser should not be considered a safety issue.

4.9.4.1 Design Phase

Although the condenser system is nonsafety-related, in events 77 and 78 the rupture of rubber expansion joint in line caused flooding and damaged engineered safety-system equipment. Adherence to pipe rupture criteria and SEP requires that safety related equipment be protected from flooding resulting from postulated pipe rupture. Therefore, the following preventive measure (item a.) is safety related and should be considered during the design phase. The other preventive measures (items b. and c.) would be desirable during the design phase.

a. Select proper locations and design enclosures for engineered safety-system equipment in the turbine building such that the safety equipment will not be damaged from flooding or other types of accidents. Provide level switches to detect flooding.

b. Develop inspection and maintenance procedures for the circulating water valve hydraulic system that prevent inadvertent valve closure, while circulating water is flowing. It is desirable that valve controls and the piping system be designed so that all portions of the piping and condenser can withstand valve closure.

c. Design steam bypass header (or spargers) to prevent condensate formation and accumulation.

4.9.4.2 Operational Phases

The condenser internals should be periodically inspected to verify that all components are in good working conditions (inspection procedure is required). Appropriate repair or replacement should be made when required.

4.10 BWR Cooling Water Systems

This section evaluates water hammer events occurring in several BWR cooling water systems that are important to safety. These systems include essential service water, RHR service water and component cooling water. Also included is an event in the cooling tower water system.

4.10.1 System Description

The service water and component cooling water systems provide essential cooling to safety-related equipment and may also provide cooling to nonsafety-related auxiliary components that are required for normal plant operation.

The service water system is an open loop system consisting of two or more pumps taking suction from the ultimate heat sink. The component cooling water system is a closed loop, solid water system with redundant heat exchangers that are cooled by the service water system. A surge

tank is connected to the pump suction headers. The pump discharges into a header with appropriate motor-operated valves in the line that allows the pumps be isolated from each other and the redundant trains from each other. Each train provides cooling to one or more of the redundant heat exchangers in the essential safety systems. The nonessential loads are isolated during accident conditions from the essential trains by quick acting isloation valves. Essential loads needed for shutdown or accident conditions, but not necessary during normal plant operation, have quick acting valves to bring the equipment on-line when needed.

4.10.2 Evaluation of Water Hammer Events

4.10.2.1 Event Review

Table 4.10-1 presents a summary of cooling water systems water hammer events reported in reference 1. No incidents were observed in a component cooling water system. This is to be expected, since each component cooling water system is a closed loop, solid water system and has a surge tank on the suction header. The cause listed for most events (five out of eight) is flow-into-voided-line. Two other events were caused by column separation and the cause for one event was not known. When using these data for cause evaluation, it should be noted, that water hammer was actually observed in event 47 only. The previous occurrence of water hammer was surmised for the other events on the basis of observed damage.

4.10.2.2 Cause of Water Hammer Events

Various possible mechanisms of observed water hammer occurrences, because of system design, equipment orientation and operating procedure are investigated here.

4.10.2.2.1 Flow-Into-Voided Line

a. During standby periods, the pressure on the discharge side of a pump can gradually decay, causing voids to form at the high points in the discharge line. On subsequent pump start, the water is

accelerated through a void and then is stopped upon impact with the upstream water column causing water hammer. Open loop service water systems, supplied by an open source, are particularly prone to this type of incident. Events 3, 4, 5 and 6 may have been caused by this scenario.

b. As the pump manual discharge valve is normally left open, the discharge side could drain through a leaking check valve during a prolonged standby period. On subsequent pump start, with the system partially drained, water hammer can result. Event 47 may have been caused by this scenario.

c. The cooling water line to the RHR service water pump motor cooler is tapped off of the pump discharge header. The water flows through the motor cooling water jacket to a floor drain. Manually-operated isolation valves, located before and after the cooling water jacket, are opened after pump start and closed after pump stop. If the operator forgets to close these valves after pump stop, the discharge side will start draining. On subsequent pump start, severe water hammer can result. Event 47 may have been caused by this scenario. Similar incidents are not expected to occur in plants where these valves close automatically following pump shut off.

4.10.2.2.2 Water Column Separation

Pressure transients propagated through a liquid system by sudden changes in valve position or pump failure can cause void formation if the pressure drops below the liquid vapor pressure. If the voids form over a considerable fraction of pipe the cross-section, the phenomenon is called column separation. Subsequent pump start or valve opening causes the water slug to accelerate through the void, then stop suddenly upon contact with the downstream water column. The resulting water hammer can cause severe damage. Events 70 and 74 may have been caused by this scenario.

4.10.3 Safety Significance

Water hammer induced failures in both redundant trains common headers or a water hammer failure in one train and a single active failure in the other could result in the loss of core and containment cooling following normal shutdown or postulated accidents. These failures would also cause loss of long-term cooling following a postulated LOCA. Based on the above reasoning, the safety significance of essential service water systems is high.

4.10.4 Recommendations for Prevention and Mitigation

4.10.4.1 Design Phase

a. A design review should be performed that identifies all portions of piping in which voids or column separation can occur under any operating condition, including pump trip and valve alignments that might occur during maintenance or through operating error.

b. A fill system should be incorporated to prevent void formation on the pump discharge side unless it can be shown that either voids can not form in the system or that the system can be safely started with voids present. Vacuum breakers should be installed in systems in which startup with voiding is deemed acceptable.

c. Manually-operated isolation valves for cooling the RHR service water pump motor, should be replaced by automatic valves that open on pump start and close following pump shut off.

d. A monitoring/alarm system should be incorporated to detect system leakage and void formation.

e. A vent system should be provided that vents all portions of the piping. All venting should be at the line high points. Any portion of that is isolated from the system high point by a valve should have a separate vent point.

f. The vent system should either be automatic, remotely operated or designed and located for easy access and manual operation.

4.10.4.2 Operational Phase

a. Valves should be leak tested periodically. When projected valve leakage is deemed to be large with respect to the keep fill system capacity, repairs should be made.

b. Standby pumps should preferably be started using either a low flow bypass line or against a closed discharge valve and then the discharge valve should be gradually opened.

c. Any time the system is to be maintained or aligned in a manner not covered by existing procedures, an evaluation of potential water hammer conditions and venting requirements should be performed.

4.11 BWR Plant Process Steam System

4.11.1 System Description

The plant process steam system supplies steam to various parts of the plant for heating purposes.

4.11.2 Cause of Water Hammer

One water hammer event (event 1) occurred in the plant process steam system. Event 1 was reported as a steam bubble collapse type water hammer in reference 1 and was caused by marginal design and procedural deficiency. The design allowed RCS water to backflow into the plant heating system external to the containment causing a water hammer in the steam supply line from the heating boiler. Operating procedures were either inadequate or not followed in lining up the valves.

4.11.3 Safety Significance

The safety significance of the plant process steam system is low. The plant process steam is not safety-related. The probability of damaging the safety-related systems due to a process steam pipe break is very low. The water hammer event in the plant process steam (event 1) was plant specific and caused no apparent physical damage.

4.11.4 Recommendations for Prevention and Mitigation

Event 1 occurred about 15 years after commercial operation. This implies that the water hammer was a rare event which was probably caused by operating error. This event is plant specific in that the plant process steam is connected to the RCS in this plant. The water hammer in the steam supply line from the heating boiler caused no apparent physical camage. Furthermore, the plant boiler system is not safety-related. For these reasons, consideration of preventive measures is not necessary for the other BWR plants.

TABLE 4.1-1

WATER HAMMER EVENTS IN BWR CORE SPRAY SYSTEM

NOTE 1: No water hammer was actually witnessed. The occurrence was surmised based on observed damage. The type of water hammer and cause can only be based on judgment.

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
31	Dresden 2	8/70	3/26/71	Testing	Possible flow-into-voided-line	Pump start/ valve opening	Failure of valve to open. Broken switch in the Limitorque operator.	Unknown	Existense of water hammer condition was inferred from the damage observed on valve operator switch. Note 1
32	Dresden 2	8/70	3/29/71	Testing	Flow-into-voided-line	Pump start/ valve opening	Water hammer on pump start. Pipe hangers damaged.	Operation of pump with empty discharge piping.	Update operating procedure so that discharge line is always filled and vented prior to pump start. Install keep fill system.
34	Dresden 2	6/70	7/11/76	95% Power	Possible flow-into-voided-line	Pump start/ valve opening	Outboard injection valve lost open/closed indication and could not be opened electrically. Valve limit switches damaged.	Unknown	Existence of water hammer condition was inferred from the damaged observed on valve limit switch. Note 1.
36	Dresden 3	10/71	11/27/71	Testing	Possible flow-into-voided-line	Pump start/ valve opening	Normally open valve inadvertently cycled closed and failed to open. Valve limit switches damaged.	Unknown	Existence of water hammer condition was inferred from the damaged observed on valve limit switch. Note 1
39	Duane Arnold	5/74	4/10/74	Cold Shutdown	Flow-into-voided-line	Pump start/ valve opening	Accidental actuation of core-spray. Anchors were pulled loose from one seismic restraints.	Inadequate administrative controls.	Core spray system should be rendered inoperative when the keep fill system is undergoing maintenance.

TABLE 4.1-1 (Continued)

WATER HAMMER EVENTS IN BWR CORE SPRAY SYSTEM

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
43	Duane Arnold	5/74	2/11/77	85% Power	Flow-into-voided-line	Pump start/ valve opening	Inadvertent initiation of core spray. Loud noises heard. Motor operated clutch housing of spray injection valve fractured.	Procedural deficiency.	Check adequacy of keep fill and vent systems against system leakage.
57	Millstone 1	12/70	4/11/78	50% Power	Unknown	Valve opening/ closing	Degradation of CS and LPCI piping support systems.	Unknown. Evidence of dynamic loading.	Existence of water hammer condition was inferred from the evidence of large dynamic loading. Note 1
59	Millstone 1	12/70	2/20/80	100% Power	Steam-bubble collapse	Pump start	Pipe support damage water hammer during spray pump operability surveillance.	Seat leakage past spray injection valve and check valve.	Review adequacy of keep fill system to prevent accumulation of steam bubbles past the injection valve. However, keep fill system cannot prevent formation or steam bubble between check valve and injection valve. Some sort of monitoring device needs to be incorporated.
64	Oyster Creek 1	12/69	1971	Testing	Flow-into-voided-line	Pump start/ valve opening	Water hammer on pump start. Pipe movement and possible over-stress condition at several points.	Inadequate design and operational procedures. Pump discharging on to empty pipe possibly due to leaky check valve.	Jockey pump system was installed as corrective measure.

NOTE 1: No water hammer was actually witnessed. The occurrence was surmised based on observed damage. The type of water hammer and cause can only be based on judgment.

TABLE 4.2-1

WATER HAMMER EVENTS IN BWR RESIDUAL HEAT REMOVAL SYSTEM (RHR)

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
46	Hatch-1	12/75	12/74	Shutdown Cooling	Flow into Voided Line	Pump Start Valve Opening	Observed leak in head spray line.	Operator error. Procedural deficiency. Design.	Head spray mode. Improper venting.
48	Fitzpatrick	7/75	3/75	Cold Shutdown	Flow into Voided Line	Pump Start Valve Opening	Damage found during inspection. Pipe restraints and snubber damaged.	RHR shutdown cooling operation with discharge piping not water filled.	Containment spray mode. Prior to keep fill system.
49	Fitzpatrick	7/75	5/75	Cold Shutdown	Probable Flow into Voided Line	Pump Start Valve Opening	Pipe movement reported. Pipe restraints and snubber damaged.	Unknown.	Containment spray mode. Prior to keep fill system. Repeat of 48. Note 1.
75	Quad Cities 1	2/73	4/72	Shutdown	Flow into Voided Line	Pump Start Valve Opening	Water hammer noted. Pipe restraints and hangers damaged.	Occurred during RHR system testing.	Containment spray mode. One system out of service.
37	Dresden 3	11/71	10/79	69% Power	Flow into Voided Line	Valve Opening	Damage found during inspection. Support bolts and spring hanger damage.	Probable water hammer prior to jockey pump installation.	Containment spray mode. Note 1.
53	Millstone 1	3/71	6/72	Unknown	Flow into Voided Line	Pump Start Valve Opening	Damage found during investigation. Severe pipe movement. Header damaged.	Inadequate operating procedures. Keep fill system not in service.	LPCI Mode. Note 1.
42	Duane Arnold	2/75	1/77	83% Power. System Test.	Flow into Voided Line	Pump Start Valve Opening	Damage found during inspection. Pipe restraints into hangers damaged. Piping overstressed.	Inadequate operating instructions, test procedures & installation. Improper venting before manual initiation.	Fuel pool cooling mode. Test procedures changed to require venting before test. Note 1.

TABLE 4.2-1 (Continued)

WATER HAMMER EVENTS IN BWR RESIDUAL HEAT REMOVAL SYSTEM (RHR)

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
76	Quad Cities 1	2/73	4/72	Shutdown	Flow into Voided Line	Pump Start Valve Opening	Damage found during routine inspection. Valve motor housing failed. Damaged pipe restraints and hangers.	Deficient design and procedures. Improper venting upstream of check valves and of tie line to fuel pool cooling system.	Fuel pool cooling mode. Change fill and vent procedures. Use condensate transfer system to replace jockey pump. Note 1.
45	Duane Arnold	2/75	9/79	44% Power	Unknown	Valve Opening	Damage found during special inspection. Damaged pipe supports and restraints.	Apparent water hammer.	Fuel pool cooling mode. Note 1.
21	Brunswick 2	11/75	9/75	Shutdown Cooling	Flow into Voided Line	Pump Start Valve Operation	Water hammer heard. Pipe supports and snubber damage, pipe movement.	Inadequate operational test, maintenance, inspection and reporting procedures. Insufficient venting.	Shutdown cooling mode. Revise procedures for venting and keep fill system.
22	Brunswick 2	11/75	9/75	Operational Surveillance Test.	Flow into Voided Line	Pump Start	Water hammer heard. Damage same as 21.	Inadequate operational test, maintenance, inspection and reporting procedures. Insufficient venting.	Shutdown cooling mode. Add vent points. Revise procedures to minimize valve cycling.
23	Brunswick 2	11/75	9/75	Shutdown Cooling	Flow into Voided Line	Pump Start Valve Operation	Damage noticed during inspection. Damage same as 21.	Same as 21, 22.	See 21, 22. Note 1.
33	Dresden 2	6/70	9/71	At Power Surveillance Test	Steam Bubble Collapse	Valve Opening	Water hammer. Neutron flux spikes. Vibration alarm. Valve operator and insulation damage.	Inadequate operating procedures. Valve leakage drained RHR heat exchanger. High water temperature.	Shutdown cooling mode. Valve test to be run only at normal temp.

NOTE 2: Repeat events 14, 15, 16, 17, 20, 25, 26. Result from common cause, steam leakage through steam condensing system valves into RHR piping and Hx. Steam void detection needed.

TABLE 4.2-1 (Continued)

WATER HAMMER EVENTS IN BWR RESIDUAL HEAT REMOVAL SYSTEM (RHR)

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
66	Peach Bottom 2	7/74	11/75	Shutdown Depressurized	Unknown	Pump Start Valve Opening	Damage found during routine inspection. Broken rigid pipe support.	Unknown	Shutdown cooling mode. Note 1.
26	Brunswick 2	11/75	4/77	72% Power Torus Cooling	Steam Bubble Collapse	Pump Start Valve Opening	Snubber damage.	Administrative controls. Improper installation. Steam leak in vent valve.	RHR torus cooling mode. Venting required prior to manual pump start.
44	Duane Arnold	2/75	12/78	Cold Shutdown	Flow into Voided Line	Pump Start Valve Opening	Damage noted during special inspection. Snubber damage.	Defective procedure. System not maintained full during outage.	RHR mode. Need better review of procedures. Note 1.
20	Brunswick 1	3/77	4/81	75% Power	Possible Steam Bubble Collapse	Valve Opening	ound snubber damage n steam condensing ine to RHR Hx.	Possible water hammer. Lack of venting. Steam leak.	Steam condensing system leak. Increased venting to every 4 hours. Note 2.
14	Brunswick 1	3/77	3/77	96% Power	Steam Bubble Collapse	Pump Start Valve Opening	Snubber damage.	Inadequate operation and inspection procedures. Lack of venting. Steam leak.	Steam condensing system leak. Increase venting of steam condensing line. Note 2.
15	Brunswick 1	3/77	3/77	Torus Cooling	Steam Bubble Collapse	Pump Start	Snubber damage.	Inadequate operation and inspection procedures. Lack of venting. Steam leak.	Steam condensing system leak. Increased venting of RHR steam condensing line. Note 2.
17	Brunswick 1	3/77	12/77	91% Power	Steam Bubble Collapse	Unknown	Broken pipe restraint.	Inadequate detection and administrative procedures. Valves leaking steam into RHR steam inlet piping.	Steam condensing system leak. Pipe supports modified. Note 1. Note 2.

TABLE 4.2-1 (Continued)

WATER HAMMER EVENTS IN BWR RESIDUAL HEAT REMOVAL SYSTEM (RHR)

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
16	Brunswick 1	3/77	11/77	86% Power. Operability Test.	Possible Steam Bubble Collapse	Pump Start	Found broken snubber.	Apparent water hammer. Recurring event in steam condensing line.	Steam condensing system leak. Note 1. Note 2.
25	Brunswick 2	11/75	10/76	Unknown	Steam Bubble Collapse	Pump Start	Broken shock suppressor	Inadequate administrative procedures. Steam bubble in RHR Hx.	Steam condensing system leak. New procedures needed. Note 2.
41	Duane Arnold	2/75	9/74	90% Power	Steam Water Entrainment	Valve Opening	Insulation damage. Pipe movement. RHR Hx inlet pipe over-stressed.	Incomplete inspection evaluation. May be related to HPCI steam line event. Location same as 14,15, 25, 26.	Steam condensing inlet line damage probably caused by improper operation of HPCI system. See event 40.
50	Fitzpatrick	7/75	7/75	At Power	Steam Water Entrainment	Valve Opening	Damage found during routine inspection. Restraint damage.	Inadequate operational and inspection procedures. Event occurred during RHR Hx/HPCI steam line warmup.	Steam condensing line damage occurred during HPCI system warmup. See event 51. Note 1.

TABLE 4.3-1

WATER HAMMER EVENTS IN THE ISOLATION CONDENSER SYSTEM (BWR)

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
55	Millstone 1	3/71	2/76	100% Power	Steam Bubble Collapse or Steam Water Entrainment	Generator Trip, Turbine Trip	Ruptured condenser tube caused radiation leak. Reactor high water level caused slugs of water to enter Hx and cause internal damage. Tube rupture was attri attributed to corrosion rather than water hammer.	Inadequate failure mode alarm and detection system. Procedural deficiencies. Poor operator response. Feed-water valve lockup and MSIV opening caused water level surge over steam inlet.	System is prone to water hammers caused by high reactor water level. System design sensitive to steam bubble collapse and steam water entrainment.
56	Millstone 1	3/71	3/78	Plant Shut-down. Isolation Condenser in Service	Steam Water Entrainment	Steam Supply Line Valve Opening	Observed movement of steam supply lines.	Procedural Deficiency. A reactor vessel water level increase allowed carry over into steam supply line.	Comments same as Event 55. Snubbers were added to steam line.
58	Millstone 1	3/71	12/79	Plant Shut-down. Isolation Condenser in Service	Steam Water Entrainment	Steam Supply Line Valve Opening	Observed movement in piping. No damage.	Reactor water level had been maximized based on TMI experience which had allowed water to enter steam supply line.	Comments same as Event 55. Water level instruction revised.
62	Nine Mile Point 1	12/69	10/69	Power Testing	Steam Water Entrainment	Steam Line Valve Opening	Observed water hammer. Damage unknown.	Inadequate design. No provision for venting or draining piping. No pitch in piping. Too fast heatup when valves opened.	Added drain points. Changed valve and control design. No more water hammer events since commercial operation 12 years ago.

NOTE 1: No water hammer was actually witnessed. The occurrence of a water hammer was surmised based upon observed damage. Therefore the postulation of water hammer type and cause can only have been based upon engineering judgment.

TABLE 4.4-1

WATER HAMMER EVENTS IN BWR HIGH PRESSURE COOLANT INJECTION SYSTEM

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
2	Browns Ferry-1 GE-4	8/1/74	10/72	Pre-op. Testing	Steam Bubble Collapse	Valve Opening	Severe water hammer in HPCI turbine exhaust line. Exhausting steam noise	Inadequate design	Provide vacuum breakers and condensing sparger on turbine exhaust line. Steam exhaust line incident. See Event 81
7	Browns Ferry-1 GE-4	8/1/74	10/73	Power Testing	Possible Steam Bubble Collapse	Valve Opening	Automatic isolation of HPCI system. Turbine discharge inner rupture disc relieved under a vacuum condition.	Procedural deficiency and inadequate design. Vacuum condition created by rapid steam condensation in turbine exhaust line.	Some means of relieving vacuum close to turbine exhaust needs to be considered. Note 1. Steam exhaust line incident.
8	Browns Ferry-1 GE-4	8/1/74	4/4/74	Shutdown	Steam-Water Entrainment	Valve Opening	Water hammer Broken pipe Hangers. Inboard turbine journal bearing pedestal was fractured. Steam supply valve limit switch was broken.	Inadequate design and marginal operating procedures. Rapid opening of outboard isolation valve.	Remove seal-in feature from outboard isolation valve opening logic to allow throttling in the opening direction or provide interlocks such that inboard valve cannot be opened unless the outboard valve is fully open. Note that gate valves are not suitable for throttling. Alternatively, modify steam line drain system for prompt and adequate drainage. Steam supply line incident.
9	Browns Ferry-1 GE-4	8/1/74	1/27/80	Shutdown	Steam-Water Entrainment	Valve Opening	Crack in HPCI turbine coupling bearing support pedestal.	Possibly caused by an observed water hammer in the steam supply line while warming the HPCI system from an out-of service condition.	Note 1 Steam supply line incident.

TABLE 4.4-1 (Continued)

WATER HAMMER EVENTS IN BWR HIGH PRESSURE COOLANT INJECTION SYSTEM

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
10	Browns Ferry-1 GE-4	8/1/74	1/29/80	Shutdown	Steam-Water Entrainment	Valve Opening	Broken instrument sensing line hangers.	Possibly caused by an observed water hammer in the steam supply line while warming the HPCI system from an out-of-service condition.	Events 9 and 10 might have been caused by the same incident but the consequences were observed at different times. Note 1. Steam supply line incident.
11	Browns Ferry-2 GE-4	3/1/75	8/11/74	5% Power	Possible Steam-Water Entrainment	Valve Opening	Automatic isolation of HPCI system. Turbine exhaust rupture disc relieved.	Possible water in turbine exhaust line when steam was admitted. Exhaust drain line solenoid burned up due to wiring error on installation. Defective switching element inside level switch deactivated drain valve.	Water hammer caused by equipment failure due to maintenance error. Note 1. Steam exhaust line incident.
12	Browns Ferry-2 GE-4	3/1/75	2/16/80	Shutdown	Possible Steam-Water Entrainment	Valve Opening	Cracks in turbine coupling bearing support pedestal.	Design and operational deficiency possibly water hammer during HPCI system warm up from out-of-service condition.	Note 1 Steam supply line incident.
13	Browns Ferry-3 GE-4	3/1/77	1/26/77	100% Power	Unknown	Pump Start	Restraints on the pump discharge line and loose bolts and broken anchors.	Unknown	Note 1 Pump discharge side incident.

TABLE 4.4-1 (Continued)

WATER HAMMER EVENTS IN BWR HIGH PRESSURE COOLANT INJECTION SYSTEM

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
19	Brunswick-1 GE-4	3/18/77	3/28/81	90% Power	Possible Flow-Into-Voided-Line	Valve Opening	Damaged piping supports	Design and operational deficiency. Inadequacy of keep fill system relative to system leakage.	Note 1. Keep fill system by itself will not eliminate water hammer problem. Check keep fill system capacity against possible system leakage. Install monitoring system to detect system leakage and void formation. Pump discharge side incident.
24	Brunswick-2 GE-4	11/3/75	9/76	Shutdown	Steam-Water Entrainment	Valve Opening	Water Hammer Excessive turbine exhaust line movement resulting in shock suppressor and hanger damage at several locations.	Exhaust piping was not drained because of a malfunction of a drain level switch and failure of a solenoid valve.	Modify system design to allow maintenance of level switch during normal plant operation. Steam exhaust line incident.
27	Brunswick-2 GE-4	11/3/75	3/24/78	Shutdown	Possible Flow-Into-Voided-Line	Valve Opening	Snubber with broken shaft on HPCI discharge line.	Probably due to sticky check valve of keep fill system.	Note 1. Pump discharge side incident.
29	Brunswick-2 GE-4	11/3/75	2/28/81	90% Power	Possible Flow-Into-Voided-Line	Valve Opening	Damaged piping supports	Design and operational deficiency. Inadequacy of keep fill system relative to system leakage.	Note 1. Pump discharge side incident.

TABLE 4.4-1 (Continued)

WATER HAMMER EVENTS IN BWR HIGH PRESSURE COOLANT INJECTION SYSTEM

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
30	Dresden-2 GE-3	6/9/70	5/29/70	Power Testing	Steam-Water Entrainment	Valve Opening	Water hammer damage to piping.	Design and procedural deficiency.	Comment for Event 8 applies. Steam supply line incident.
40	Duane Arnold GE-4	2/1/75	6/11/74	30% Power	Steam-Water Entrainment	Valve Opening	Normally open outboard steam supply isolation valve was indicating closed; damage to pipe insulation, pipe hanger, seismic snubbers, pressure indicator and steam line drain pot indicator.	Design and procedural deficiency; operator error; movement and impact of water slug from steam condensation occurred in the steam supoly line when the outboard isolation valve was opened while the inboard isolation valve was full open.	Comment for Event 8 applies. Steam supply line incident.
61	Monticello GE-3	6/30/71	7/17/72	Surveillance Testing	Possible Steam Bubble Collapse	Turbine Exhaust Stop Check Valve Operation	Turbine trip. Failed check valve pin caused line blockage; steam issuing from relieved exhaust line rupture discs impinged on adjacent temperature switches rendering them inoperable.	Inadequate component and subsystem design.	Equipment failure leading to water hammer. Note 1. Steam exhaust line incident.
68	Peach Bottom-3 GE-4	12/23/74	2/14/75	Power	Steam-Water Entrainment	Valve Opening (Steam-Line)	Movement of steam supply line.	Inoperative component and administrative deficiency. Failure of steam tran to drain properly and drain pot level switch to trip on high level.	Equipment failure leading to water hammer. Steam supply line incident.

TABLE 4.4-1 (Continued)

WATER HAMMER EVENTS IN BWR HIGH PRESSURE COOLANT INJECTION SYSTEM

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
69	Peach Bottom-3 GE-4	12/23/74	12/2/75	57% Power	Steam-Water Entrainment	Valve Opening (Steam Line)	Movement of steam supply line.	Inoperative component and administrative deficiency. Failure of steam trap to drain properly and failure of the drain pot level switch to trip on high level.	Identical to Event 68. Steam supply line incident.
81	Vermont Yankee GE-4	11/30/72	1971	Pre-op. Testing	Steam Bubble Collapse	Exhaust Line Check Valve Closure	Water hammer	Design and procedural deficiency. Fast turbine exhaust line check valve closure due to vacuum condition.	Exhaust line check valve should preferably be located in horizontal position. Steam exhaust line incident. See Event 2
82	Vermont Yankee GE-4	11/30/72	6/76	88% Power	Possible Steam-Water Entrainment	Valve Opening/ Pump Start	Leakage	Operator error and design deficiency. Operator accidently drained reference leg of an RV level control instrument and gland seal head gasket failed on system start.	Note 1. Gland seal condenser steam line incident.
51	FitzPatrick GE-4	7/28/75	9/7/75	Startup	Steam-Water Entrainment	Valve Opening	Damaged several pipe restraints on steam line to RHR heat exchanger.	Insufficient drainage of condensed steam during HPCI line warmup.	Note 1 Provide adequate drainage of steam supply line. Steam supply line incident.

TABLE 4.5-1

WATER HAMMER EVENTS IN BWR REACTOR CORE ISOLATION COOLING SYSTEM

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
60	Monticello GE-3	6-30-71	1971		Steam Water Entrainment		Water Hammer	Inadequate design	Steam exhaust line incident. Provide vacuum breakers on turbine exhaust line.
79	Quad Cities-1 GE-3	2-18-73	10-29-76	94% Power	Pump Cavitation (Not a water hammer problem)	Valve Opening	Pump failed to develop required head and flow Two of five pump stages were severely damaged.	Faulty operational procedure and mechanical design. Pump startup with test return valve to condensate storage tank open.	Pump cavitation incident. Revise pump start procedure

Note 1: No water hammer was actually witnessed. The occurrence of a water hammer was surmised based upon observed damage. Therefore, the postulation of water hammer type and cause can only have been based upon engineering judgement.

TABLE 4.8-1

WATER HAMMER EVENTS IN BWR MAIN STEAM SYSTEM

EVENT NO.	PLANT/ DESIGN	COM. OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE & EVENT BASIS	COMMENTS
18	Brunswick-1 GE-4	3/18/77	12/12/79	0% Power	SRV discharge transient	SRV lifting	Discovered damaged snubbers. Variety of damage to ten different snubbers on S/RV discharge line.	Reactor scrammed and relief valves lifted. Analysis indicated that a water slug could cause the damage if selected snubbers were non-functional.	Note 1. Frequent recorded inspection of snubbers required to assure operability.
63	Nine Mile Point-1	12/69	1969-71	Power escalation testing	Steam water entrainment	Main steam lines valve opening	Water hammer in steam lines. Damage unknown	Inadequate design. Condensate formation and lack of adequate drain or properly pitched pipe to eliminate liquid accumulation. Two-phase flow caused turbine trip.	Identify situation and conditions that have potential for damaging water hammer. Check system design and operating procedure to eliminate steam-water entrainment.
52	Milestone-1 GE-3	3/71	12/09/70	Start-up (50% load turbine trip test)	Steam hammer	Turbine stop valve closure, turbine bypass valve instability	Damage to MS piping. Excessive movement of main steam lines and bypass lines. Fixed pipe support common to all four main steam lines, between outboard isolation and turbine stop valves, stressed beyond yield. Line movement damaged other support steel and instrument connections.	Inadequate piping support design (failure to consider dynamic forces generated by rapid closure of stop valves). During a planned turbine trip the rapid closure of the MS stop valves caused a transient. A contributing cause was malfunction of bypass valve actuator components.	Note 1. Piping support design should include loading due to dynamic force generated by rapid valve closure. Ensure that valve component selection is compatible with duty cycle and service requirement.
65	Oyster Creek-1	12/69	11/16/71	Start-up (Valve timing test)	Probably steam-water entrainment	Sudden MSIV opening	Incomplete MSIV closure. Valve operator cast iron speed cushion crushed with pieces preventing valve closure.	Inadequate valve design (component/ service condition incompatibility) and/or sudden valve opening with large pressure differential (800 psi) existing across valve disc.	Note 1. Ensure that valve component selection is compatible with duty cycle and service requirement. Check operating procedure for warm up during startup.

Note 1: No water hammer was actually witnessed. The occurrence of a water hammer was surmised based upon observed damage. Therefore, the postulation of water hammer type and cause can only have been based upon engineering judgement.

TABLE 4.6-1

WATER HAMMER EVENTS IN BWR MAIN STEAM SYSTEM

EVENT NO.	PLANT/ DESIGN	COM. OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE & EVENT BASIS	COMMENTS
67	Peach Bottom-3 GE-4	12/23/74	10/15/74	Power escalation testing	Steam hammer	Bypass valves cycling	Damage to piping system observed. Snubbers on piping between bypass valves and main condenser damaged.	Valve maintenance deficiency (improper EHC calibration). Out-of-calibration acceleration amplifiers in the electro hydraulic controls system caused valve cycling.	Note 1. Develop inspection and calibration procedures to detect out-of-calibration valve control and make necessary corrections as required.
71	Pilgrim-1 GE-3	12/72	7/24/72	Power escalation testing	Steam hammer	Startup test program involving repeated closure of turbine stop valve and control valve	Damage to piping system observed. Pipe hanger torn from support on one main steam line. Bent hangers on three other main steam lines downstream of MSIV near second elbow.	Inadequate piping support design (failure to consider cumulative concurrent loading). The additional dynamic loading induced by valve closure acted concurrently with existing loads to overstress a pipe support.	Note 1. Piping support design should include loading due to dynamic force generated by rapid valve closure. Work out inspection and maintenance procedures for the pipe support system components.

Note 1: No water hammer was actually witnessed. The occurrence of a water hammer was surmised based upon observed damage. Therefore, the postulation of water hammer type and cause can only have been based upon engineering judgement.

TABLE 4.7-1

WATER HAMMER EVENTS IN BWR FEEDWATER SYSTEM

EVENT NO.	PLANT/ DESIGN	COM. OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE & EVENT BASIS	COMMENTS
73	Pilgrim-1	12/72	1/6/76	Increasing power (above 20% rated power)	Unknown (vibration), possible column separation	FW regulating valve instability (valve cycling	FW system vibrations. Yoke of startup regulating valve fractured resulting in complete ejection of valve stem. Valve body cracked. Pipe hanger bent on FW line.	Inadequate valve operator design, component/service condition incompatible. Cycling of FW valve due to faulty pneumatic valve operator induced flow vibrations.	FW regulating valve instability caused water hammer. Select valve operator and controller that can meet the control valve performance and response requirements.
35	Dresden-3 GE-3	11/16/71	6/23/74	At power in run mode	Unknown	Regulating valve closure	FW regulating valve lock up; service air compressor, RWCU pump, and FW heater tripped; FW and reactor level decreased. FW low flow regulating valve opened and rotated with all air lines and electrical feeds broken. Damage to piping support system components in FW lines.	Specific cause not identified. FW valve vibration and inadvertent closure possibly related control system malfunction.	Note 1. See Event 73
80	Quad Cities-2 GE-3	2/18/73	8/31/75	Decreasing load for shutdown	Unknown (vibration)	Regulating valve instability	FW vibration alarm, turbine trip due to reactor high water level, reactor scram, FW system leak. FW low flow drain lines and high pressure heater bypass line broken.	Inadequate valve actuator design. FW system vibration possibly caused by flow/response condition of FW valve actuator piston.	See Event 73

TABLE 4.8-1

WATER HAMMER EVENTS IN BWR REACTOR WATER CLEANUP SYSTEM

NOTE 1: No water hammer was actually witnessed. The occurrence of water hammer was surmised based upon observed damage. Therefore the postulation of water hammer type and cause can only have been based on engineering judgment.

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
38	Dresden-3 GE-3	11/16/71	4/2/80	Refueling	Possible flow-into-voided line	Valve Opening/ Closing	Fully retracted mechanical snubber. Crack in affected pipe between first isolation valve and containment penetration.	Possible operation or material deficiency.	Note 1 After nine years of operation only one event was observed. Crack in the affected pipe did not cause any leak. Ultrasonic test detected the crack. Radiographic test did not confirm this indication. This suggests that the crack may be parallel to the surface.

Note 1: No water hammer was actually witnessed. The occurrence of a water hammer was surmised based upon observed damage. Therefore the postulation of water hammer type and cause can only have been based upon engineering judgement.

Note 2: The condenser system is non-safety related. However, nuclear safety considerations are involved when engineered safety system equipment is damaged due to flooding such as in Events 77 and 78.

TABLE 4.9-1

WATER HAMMER EVENTS IN BWR CONDENSER SYSTEM

EVENT NO.	PLANT/ DESIGN	COM. OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE & EVENT BASIS	COMMENTS
77	Quad Cities-1	2/18/73	6/9/72	Hot shut-down. (mainten-ance outage)	Probable column separation	Circulating water butterfly valve closure.	Sudden, inadvertent valve closure. While venting the recirculating valve hydraulic system, a rubber expansion joint in the condenser water box ruptured and recirculation water flooded the condensate pump room. Water immersion damaged RHR service water pumps and motors and other equipment.	Inadequate maintenance or repair practices. While performing condenser mod-ifications, a butterfly valve slammed closed while condenser circulating water pumps were in operation. This event occurred while venting the valve hydraulic oil system.	Note 1. Develop maintenance procedure for circulating water valve hydraulic system to prevent in-advertent valve closure, while circulating water is flowing. Select location and design en-closure of engineered safety equipment such that the safety equipment will not be damaged from flooding or other types of accident
78	Quad Cities-1	2/18/73	1973	Unavailable	Column separation	Circulating water butterfly valve closure.	Unavailable. Rupture of rubber expansion joint in line. Damaged engineered safety-system equipment due to flooding.	During maintenance work, malfunction caused a butter-fly valve to slam shut resulting in water hammer.	Note 1. See Event No. 77.
54	Millstone-1 GE-3	3/71	1972	Unavailable	Steam-water entrainment	Steam bypass valve opening.	Structural failure of steam bypass header in main condenser. End cap of steam bypass header failed.	Inadequate design of steam bypass header (hole sizes and numbers of holes in header insufficient). Con-densate formation and accumulation in steam bypass header.	Note 1. Check design for possib-ility of condensate formation and accumulation in steam bypass header.
72	Pilgrim-1 GE-3	12/72	1972	Unavailable	Probable steam-water entrainment	Steam bypass valve opening.	Broken spargers and baffle damage in condenser. Condenser internal spargers were broken.	Inadeuqate design of sparger. Probably condensate formation and accumulation in sparger.	Note 1. Check design for possib-ility of condensate formation and accumulation in sparger

NOTE 1: No water hammer was actually witnessed. The occurrence of water hammer was surmised based upon observed damage. Therefore the postulation of water hammer type and cause can only have been based on engineering judgment.

TABLE 4.10-1

WATER HAMMER EVENTS IN BWR COOLING WATER SYSTEMS

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
3	Browns Ferry-1 GE-4	8/1/74	5/6/73	-	Possible flow-into-voided-line. Non-essential water system.	Pump start	Failure of orifice gasket.	Design and procedural deficiencies. Voids form due to line leakage and dissolved gases collect at high points during standby periods. On pump start the water compresses these gases or forces them into solution such that the water inter faces come in contact causing damaging water hammer.	Note 1 Improve surveillance or add void alarm system.
4	Browns Ferry-1 GE-4	8/1/74	5/10/73	-	Possible flow-into-voided-line. RHR service water system.	Pump start	Failure of pipe coupling.	See Event 3	Note 1 See Event 3
5	Browns Ferry-1 GE-4	8/1/74	5/23/73	-	Possible flow-into-voided-line. RHR service water system.	Pump start	Failure of pipe coupling.	See Event 3	Note 1 See Event 3
6	Browns Ferry-1 GE-4	8/1/74	6/7/73	-	Possible flow-into-voided-line. RHR service water system.	Pump start	Failure of pipe coupling.	See Event 3	Note 1 See Event 3
28	Brunswick-2 GE-4	11/3/75	4/12/80	Zero Power	Unknown RHR service water system.	Valve Opening/ Closing	Partially buckled HX rib plate.	Procedural deficiency.	Note 1 Operating procedures should be revised to require venting.

TABLE 4.10-1 (Continued)

WATER HAMMER EVENTS IN BWR COOLING WATER SYSTEMS

EVENT NO.	PLANT/ DESIGN	COM.OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE AND EVENT BASIS	COMMENTS
47	FitzPatrick GE-4	7/28/75	4/10/74	Functional Testing	Flow-into-voided-line. RHR service water system.	Pump start/ valve opening.	Water hammer; piping movement. The bottom of the pump discharge basket strainer was blown off and grouting for strainer support was chipped. Buckled piping at seismic trunion locations. Bent seismic trunions, a pipe support with a failed turnbuckle, and a pipe support with failed anchors near the pump.	Procedural deficiency. The pump discharge valve and motor cooler isolation valves were not closed and remained open over night causing the system to drain.	The pump discharge valve and motor cooler isolation valves should normally be kept closed and be gradually opened on pump startup.
70	Peach Bottom 2 and 3 GE-4	7/5/74 12/73/74	5/76	Refueling Shutdown from 70% power.	Possible column separation. Cooling water system.	Pump start/ valve opening.	Water gushing from lift pump house door. Bolts holding suction bell housing to pump casing failed. Discharge piping cracked and base plate shifted.	Unknown; possible design/ procedural deficiency. Observed damage is indicative of water hammer on pump start due to column separation/voided line flow.	Not safety-related.
74	Pilgrim-1 GE-3	12/72	2/3/77	5% power	Possible column separation. Essential service water system.	Pump start/ valve opening.	Salt water service pump failed to start. The top column pipe had a 360 degree fracture just below its top flange allowing the pipe to drop into and jam the pump impeller.	Unknown; possible design/ procedural deficiency. Observed damage is indicative of column separation in discharge line.	Note 1 See Event 3

Note 1: No water hammer was actually witnessed. The occurrence of a water hammer was surmised based upon observed damage. Therefore the postulation of water hammer type and cause can only have been based upon engineering judgement.

TABLE 4.11-1

WATER HAMMER EVENTS IN BWR AUXILIARY BOILER SYSTEM

EVENT NO.	PLANT/ DESIGN	COM. OP. DATE	EVENT DATE	OPERATING MODE	WATER HAMMER TYPE	MECHANICAL FUNCTION	INITIAL INDICATION/ DAMAGE	CAUSE & EVENT BASIS	COMMENTS
1	Big Rock Pt.	3/29/63	10/31/77	Unavailable	Steam bubble collapse.	Plant heating boiler, valve opening.	Water hammer occurrence. No apparent physical damage. Event resulted in a minor, uncontrolled release of radioactive water to discharge canal.	Marginal design concepts and procedural deficiency. During manual valving operations RCS water backflowed into the plant heating system external to the containment causing a water hammer in the steam supply line from the heating boiler. Operating procedures were not followed in the valve lineup.	Plant specific event.

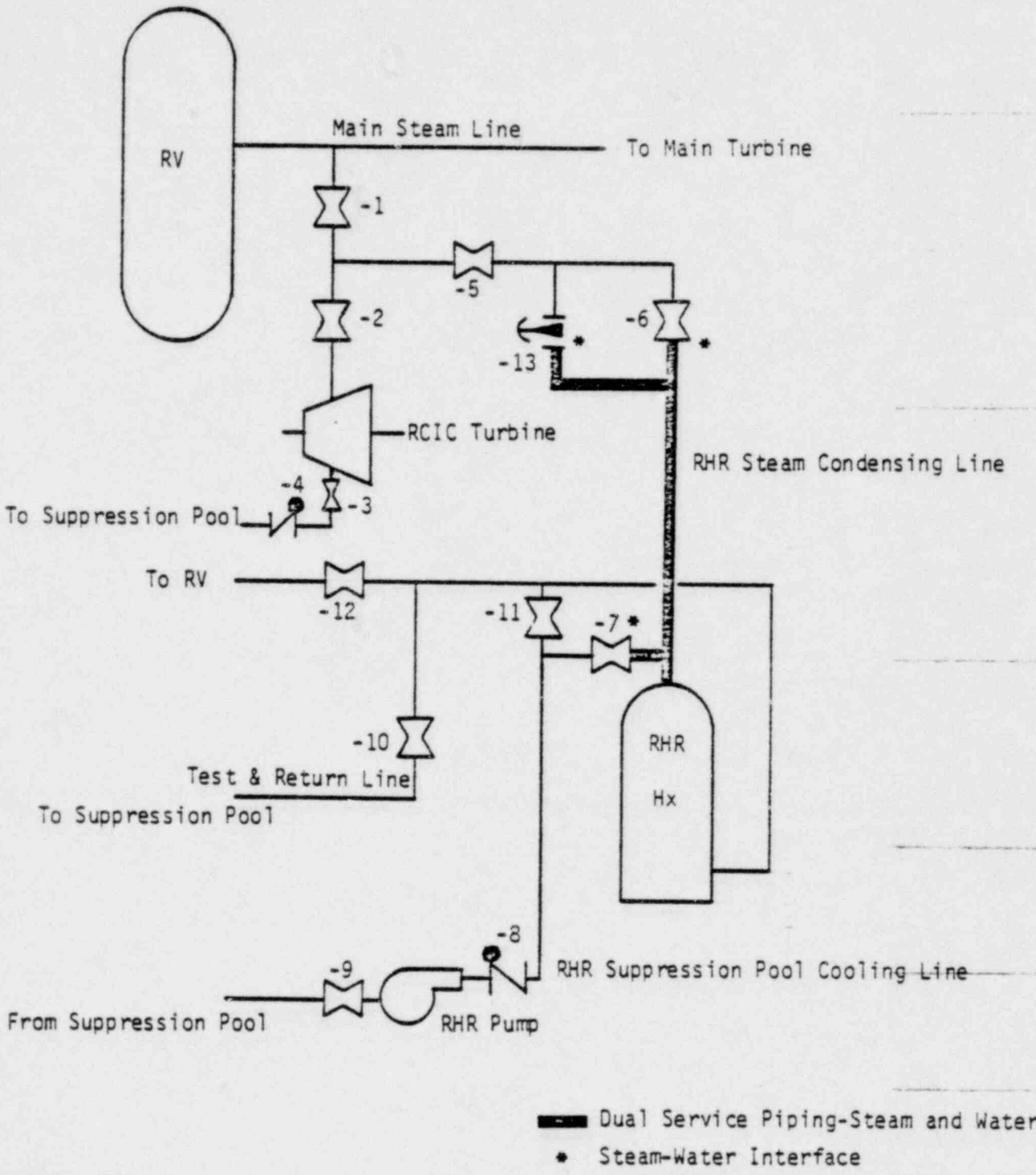

FIGURE 4-1

RHR STEAM CONDENSING SYSTEM SCHEMATIC

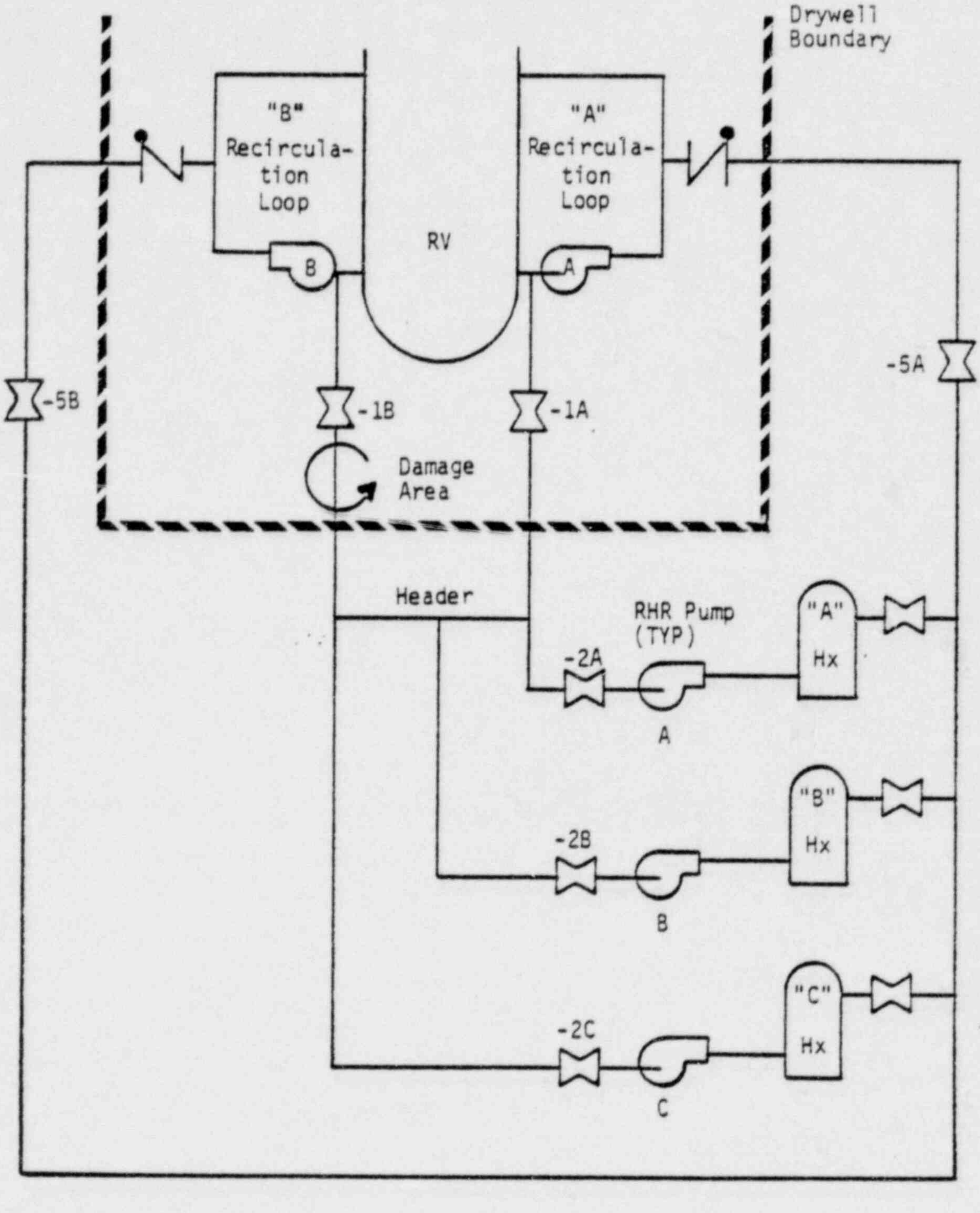

FIGURE 4-2

RHR SHUTDOWN COOLING SYSTEM SCHEMATIC

6.0 IMPLEMENTATION OF PREVENTIVE MEASURES

This section discusses the means of implementing the preventive and mitigative measures discussed in section 3.5.

6.1 Plants in Design or Construction

A generic Standard Review Plan (SRP) or Branch Technical Position (BTP) should be issued on water hammer. The SRP or BTP should be generic in nature and address the requirements contained in 6.3 below. The existing SRPs for the effected systems should refer to the generic SRP or BTP.

6.2 Operating Plants

A generic letter or IE Bulletin should be issued to operating plants listing the requirements contained in 6.3 below.

6.3 Requirements for the Prevention and Mitigation of Water Hammer in Light Water Reactor Plants

a. Operator Training

All plant operators, including personnel responsible for the writing of maintenance instructions and the supervision of maintenance activities, shall receive training on the causes and prevention of water hammer.

b. Operating and Maintenance Procedures

The applicant shall review all operating maintenance and testing procedures for the systems listed below for their appropriateness in preventing water hammer.

Applicable systems:

- o BWR
  - RHR
  - HPCI (HPCS)
  - Core Spray
  - Essential Cooling Water Systems

- Isolation Condenser
- Feedwater
- Main Steam

o PWR

- ECCS (Safety Injection)
- Feedwater
- Main Steam
- Essential Cooling Water Systems

c. Void Detection

Void Detection and alarm shall be provided for the systems listed below. Void detection shall be provided at all points in the normally liquid filled lines where voids or stream bubbles could form or collect. It shall be shown that all potential void points have been monitored. These systems shall be considered inoperable when voids are present. Open loop service water systems may be considered operable if analysis has been performed to demonstrate that there will be no adverse effects if the system is started with voids present.

Applicable systems:

o BWR

- RHR
- Core Spray
- HPCI (HPCS)
- Essential Cooling Water

o PWR

- ECCS
- Essential Cooling Water

d. Keep Fill Systems

Continuously operating keep fill systems shall be provided for the filling of voids in normally water filled lines in the systems

listed below. A jockey pump or a storage tank at a higher elevation then the lines of concern shall be considered adequate keep fill systems.

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Applicable System:

- o BWR
  - Core Spray
  - HPCI (HPCS)
  - RCIC
  - RHR

e. Filling of Safety-related, Open Loop Service Water Systems

One of the following shall be demonstrated for open loop service water systems.

1. Voids can be filled within the required start time through a manually initiated fill system. This provision is applicable to manually started systems only.
2. Neither column separation nor voiding can occur during standby or following pump shutdown.
3. The system is designed with a startup mode that slowly fills and vents the discharge lines in such a manner as to prevent water hammer on pump start up, or
4. The system is designed to maintain function following a postulated water hammer event.

f. Venting

Venting provisions shall be provided for the systems listed below. Venting shall be provided at all points in the normal lines where voids or steam bubbles could form or collect. It shall be demonstrated that all potential void points can be vented. The vent system shall either be automatic or shall be designed for ease of operator usage.

Applicable systems:

- o BWR
  - RHR
  - Core Spray

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- HPCI (HPCS)
- Essential Cooling Water
- RCIC

o PWR

- ECCS
- Essential Cooling Water

g. Turbine Exhaust Line Vacuum Breaker

Vacuum breakers shall be provided in the turbine exhaust lines that have a liquid interface. This provision is only applicable for safety related systems.

Applicable Systems:

o BWR

- HPCI
- RCIC

h. HPCI Steam Line Drain Pot

1. The adequacy of the sizing of the HPCI drain pot system shall be demonstrated.
2. The level indicators on the HPCI drain pot system shall be checked for operability on a periodic basis and repaired if necessary.

i. HPCI Turbine Inlet Line Isolation Valves

An interlock shall be provided that will preclude opening of the HPCI turbine inlet line inboard isolation valve unless the outboard isolation valve is fully open. Neither valve shall contain a seal-in feature on opening. The inboard valve shall be designed to permit gradual line warm up.

j. Feedwater Control Valve

The feedwater control valve supplier shall verify that the valve design parameters including actuator, CV, and trim are compatible with all final designed operating conditions of the condensate and

feedwater system. Furthermore, the valve and its control system shall be designed to minimize the potential for instability, vibrations, and water hammer.

k. Steam Hammer and Relief Valve Discharge

1. The design bases for the operability and support of main steam systems shall consider steam hammer resulting from the most rapid anticipated closure of all system valves including the turbine stop valves.
2. The design basis for the operability and support of the systems listed below shall consider forces resulting from safety and relief valve operation.
   - o BWR
     - Main Steam
   - o PWR
     - Main Steam
     - RCS Pressurizer

7.0 REFERENCES

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7. U.S. Nuclear Regulatory Commission, "Protection Against Postulated Piping Failures in Fluid Systms Outside Containment," Branch Technical Position ASB 3-1.

<u>EXHIBIT 2</u>

COMPILATION OF DATA CONCERNING KNOWN AND
SUSPECTED WATER HAMMER EVENTS IN NUCLEAR
POWER PLANTS, APRIL 1982, BY EG&G IDAHO, INC.

EGG-CAAD-5629
April 1982

COMPILATION OF DATA CONCERNING KNOWN AND SUSPECTED WATER HAMMER EVENTS IN NUCLEAR POWER PLANTS

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Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6451

4/9/82

EG&G Idaho, Inc.

FORM EG&G-398
(Rev. 11-79)

INTERIM REPORT

Accession No. ______________

Report No. EGG-CAAD-5629

Contract Program or Project Title: Technical Assistance for Resolution of Unresolved Safety Issue (USI) A-1, Water Hammer

Subject of this Document: Compilation of Data Concerning Known ~~Actual~~ and Suspected Water Hammer Events in Nuclear Power Plants

Type of Document: Final Report on Compilation of Water Events

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Date of Document: April ~~January~~ 1982

Responsible NRC Individual and NRC Office or Division: Aleck W. Serkiz, NRC-NRR

This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

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ABSTRACT

This report compiles data concerning known and suspected water hammer events reported by BWR and PWR power plants in the United States from January 1, 1969, to May 1, 1981. This information is summarized for each event and is tabulated for all events by plant, plant type, year of occurrence, type of water hammer, system affected, basis/cause for the event, and damage incurred. Information is also included from other events not specifically identified as water hammer related. These other events involved vibration and/or system components similar to those involved in the water hammer events. The other events are included to ensure completeness of the report, but are not used to point out particular facts or trends. This report does not evaluate findings abstracted from the data.

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SUMMARY

This report compiles data concerning known and suspected water hammer events reported by BWR and PWR power plants in the United States from January 1, 1969, to May 1, 1981. Incidents involving BWR relief valve discharge into the suppression pool were not scoped as part of this study, and are therefore excluded.

Information is included from other events not specifically identified as water hammer related. These other events involved vibration and/or system components similar to those involved in the water hammer events. The other events are included to ensure completeness of the report, but are not used to point out particular facts and/or trends.

2.

A total was reported of 81 BWR and 67 PWR water hammer events. The second grouping (of events not specifically identified as water hammer related) includes 43 BWR and 4 PWR events.

The background information,

The reason(s) for each occurrence and the corrective action(s) taken to prevent additional occurrences are summarized for each event in the appendixes. This information was obtained from various sources, and in some cases specific information was not available--necessitating interpretation to determine water hammer occurrences and other conditions.

Only the information for known or suspected water hammer events has been reviewed. This information has been tabulated by plant, plant type, year of event occurrence, type of water hammer, system affected, basis/cause for the event, and damage incurred. Numerous findings were abstracted from this tabulated data. To avoid highlighting any of these findings, none are summarized in this section. The reader who is interested only in these findings might go directly to the section titled Summarized Findings of the Study.

Though neither the exactitude of information for each event presented nor the completeness of event listings can be guaranteed, the ~~particular~~ facts and trends pointed out are considered valid.

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NOMENCLATURE

Acronyms

ANS	American Nuclear Society
AO	Abnormal occurrence
B&W	Babcock and Wilcox
BFAO	Browns Ferry abnormal occurrence
BWR	Boiling water reactor
CCW	Component cooling water
CE	Combustion Engineering
CS	Core spray
CST	Condensate storage tank
CVCS	Chemical and volume control system
ECCS	Emergency core cooling system
EECW	Emergency equipment cooling water
EHC	Electro hydraulic control
FW	Feedwater
GE	General Electric
HPCI	High-pressure coolant injection
Hx	Heat exchanger
IE	Inspection and enforcement
IEEE	Institute of Electrical and Electronic Engineers
IR	Inspection report
LER	Licensee event report
LPCI	Low-pressure coolant injection
MSIV	Main steam isolation valve
MWe	Megawatt electric
MWt	Megawatt thermal

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NDT	Nondestructive test
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
NUREG	Nuclear regulation
PWR	Pressurized water reactor
QA	Quality assurance
RCIC	Reactor core isolation cooling
RCS	Reactor coolant system
RHR	Residual heat removal
RO	Reportable occurrence
RT	Radiography
RV	Reactor vessel
RW	Reactor water
RWCU	Reactor water cleanup
SCW	Service cooling water
SGWH	Steam generator water hammer
SI	Safety injection
SRV	Safety relief valve
TMI	Three Mile Island
UE	Unusual event
USI	Unresolved safety issue
W	Westinghouse

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INTRODUCTION

Structural damage and inoperative systems attributed to water hammer have been reported during commercial operation of both BWR and PWR nuclear power plants--and have the potential to impact plant safety. To provide resource information for evaluating this impact on plant safety, information has been compiled about known and suspected water hammer events reported by BWR and PWR power plants in the United States from January 1, 1969, to May 1, 1981. Incidents involving BWR relief valve discharge into the suppression pool were not scoped as part of this study, and are therefore excluded.

Information is included from other events not specifically identified as water hammer related. These other events involved vibration and/or system components similar to those involved in the water hammer events. The other events are included to ensure completeness of the report, but are not used to point out particular facts and/or trends.

This work was performed at the Idaho National Engineering Laboratory for the Nuclear Regulatory Commission and is one of several investigations[1] used in resolving the generic safety issue regarding water hammer in nuclear plants.

The sources reviewed for information on these events were: records of nuclear plant experiences;[2] U.S. Library of Congress microfiche files[3] concerning nuclear plant dockets such as licensee event reports, abnormal occurrence reports, reportable occurrence reports, inspection and enforcement reports, and general correspondence between the licensee and the NRC; an NRC staff report on water hammer in nuclear plants.[4] Because in some cases specific information was not available, interpretation was necessary to determine water hammer occurrences and other conditions. Though neither the exactitude of information for each event presented nor the completeness of event listings can be guaranteed, the particular facts and trends pointed out are considered valid.

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Background information, reason(s) for occurrence(s), and corrective action(s) taken to prevent additional occurrence(s) are summarized for each event in the corresponding appendixes:

- o Appendix A--BWR Events
- o Appendix B--PWR Non-Steam-Generator Events
- o Appendix C--PWR Steam Generator Events.

The events in the appendixes ~~include the date of occurrence, and~~ are listed ~~both~~ by category, ~~and~~ alphabetically, by nuclear plant name. and by event date

The events in Appendixes A and B are classified as:

- o Category I--known and suspected water hammer events
- o Category II--events not specifically identified as water hammer related, out of interest because they involved vibration and/or system components similar to those involved in water hammer events.

Tables 1 and 3-7 list BWR Category I events and Tables 8 and 10-14 list PWR non-steam-generator events.

As previously stated, the Category II events are included to ensure completeness of the report, but are not used to point out particular facts or trends. The information from the Category II event summaries in Appendixes A and B is tabulated by system and type of event in Tables 2 (BWR) and 9 (PWR). No further treatment of the Category II events is provided.

This report is in two major sections: Review of Water Hammer Events; Summarized Findings of the Study. The review, consisting of three subsections (BWR Events, PWR Non-Steam-Generator Events, and PWR Steam Generator Events) presents ~~data abstracted from the respective~~ appendixes.

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Each of these subsections is reviewed independently of the other subsections. The findings section summarizes the particular facts and trends obtained as a result of the review.

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REVIEW OF WATER HAMMER EVENTS

For each Category I event summary in the appendixes, the following items are given:

- Plant name, event date or date of damage discovery, and the plant system operating mode preceding or at the time of event occurrence or damage discovery

- Source of information

- The mechanical function preceding or possibly precipitating the event

- The initial indication of the problem

- The system, subsystem, or component involved, the suspected type of water hammer, and the damage or malfunction incurred as a result of the event

- The cause or basis for the event

- The corrective measures undertaken for prevention.

Concerning water hammer, four basic types have been identified (Reference 4) as initiating mechanisms:

- Water column separation in liquid flows

- Flow-into-voided- (or partially-voided-) lines

- Steam-bubble collapse or mixing of subcooled water and steam from interconnected systems

- Water entrainment in steam lines.

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Situations not identifiable, but considered a result of acoustic wave propagation or reflection from valve opening, closure and instability, flow instability, or rapid check valve closure were categorized as either unknown or steam hammer.

To facilitate the compilation of data presented in the event summaries, the Category I events have been subdivided as follows:

- Class 1--Events that occurred prior to a plant's commercial operation date (87% occurred within 12 months prior and 98% occurred within the 15 months prior to the commercial operation date)

- Class 2--Events that occurred within one year after a plant's commercial operation date

- Class 3--Events that occurred after the first year of a plant's commercial operation.

General information about the various plants and information abstracted from the event summaries in Appendixes A and B have been tabulated in various ways in an attempt to find unusual situations or trends. Table 1 (BWR Category I events) and Table 8 (PWR Category I events) list events by plant system and type of water hammer. Table 3 (BWR Category I events) and Table 10 (PWR Category I events) are comprehensive listings of:

- Commercial operation dates

- Years of commercial operation

- The number of Category I events per year per plant by event class

- The totals of events per plant

- The incident rate for Class 3 events per plant.

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Table 4 (BWR Category I events) and Table 11 (PWR Category I events) list events per plant system by event class for each reporting plant. Table 5 (BWR Category I events) and Table 12 (PWR Category I events) summarize events by class for the three major plant system classifications (Main, Auxiliary, and Safety). Table 6 (BWR Category I events) and Table 13 (PWR Category I events) list basis/cause and damage incurred for each event and plant system.

Table 7 (BWR Category I events) and Table 14 (PWR Category I events) list the normalized incident rates per year for each class of events. The plant years in these two tables are based on the amount of time in a year that a plant fell under the classification of concern. As an example, assume that a plant has an initial commercial operation date of June 1, 1975. It would be a Class 1 plant in 1974 for the period from June 1, 1974, to January 1, 1975, and in 1975 for the period from January 1, 1975, to June 1, 1975. It would be a Class 2 plant in 1975 for the period from June 1, 1975, to January 1, 1976, and in 1976 for the period from January 1, 1976, to June 1, 1976. It would be a Class 3 plant after June 1, 1976. The Class 1 period is based on one year, because 87% of these events occurred within the 12 months prior to commercial operation and 98% occurred within the 15 months prior to commercial operation. Figures 1-6 are graphic displays of the data in Tables 7 and 14. Figures 1-3 follow Tables 1-8 and Figures 4-~~8~~6 follow Tables 9-15. These tables and figures will be referenced as they are discussed in the next two sections.

Table 15 (PWR SG events) lists plants reporting events, and gives their dates of commercial operation, NSSS vendor, and the number of events reported by year. This table is discussed in the final section of this review.

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BWR Events

Twenty-six BWR plants were involved in this study. Dresden I and Humbolt Bay are currently not operating, but are included because they are part of the base from which information was extracted. The following information was taken from the tables and figures previously mentioned.

This study involves 26 BWR plants at 18 locations, and covers only the 81 Category I events. All plants are General Electric BWRs except LaCrosse, which is an Allis Chalmers plant. Here is the plant breakdown:

- Thirteen (50%) are GE-4
- Seven (27%) are GE-3
- Six (23%) are "other."

Category I events were reported by

- Eleven (85%) of the GE-4 plants
- All of the GE-3 plants
- Three (50%) of the "other" plants.

Category I events occurred in

- Forty-eight (59%) of the GE-4 plants
- Twenty-nine (36%) of the GE-3 plants
- Four (5%) of the "other" plants.

This should be the number of events occurring in each type of plant

Note the percentage of events per plant type versus the percentage of plants involved.

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The plants reported Category I events as follows (see Table 3):

- Twenty-one (81%) reported at least one
- Five (19%) reported 40 (49%)
- Eight (31%) reported 54 (67%)
- Eleven (42%) reported 66 (81%)
- Allis Chalmers reported none.

Twenty-nine (36%) of the events are Class 1 (see Table 3). Fifteen (58%) of the plants reported Class 1 events. Seven (24%) of the Class 1 events occurred at Browns Ferry 1.

Thirteen (16%) of the events are Class 2 (see Table 3). Eight (31%) of the plants reported Class 2 events. Three (23%) of the Class 2 events occurred at Brunswick 1.

Thirty-nine (48%) of the events are Class 3 (see Table 3). Seventeen (65%) of the plants reported Class 3 events. Seven (18%) of the Class 3 events occurred at Millstone 1, and 22 Class 3 events (56%) occurred at Brunswick 1 and 2, Dresden 3, Duane Arnold, and Millstone 1.

The average incident rate for those plants reporting Class 3 events is 0.32 events per year per reactor (see Table 3). Of those plants reporting Class 3 events, the following have two to three times the average incident rate:

- Brunswick 1 (0.96)
- Brunswick 2 (0.91)
- Duane Arnold (0.76)
- Millstone 1 (0.76).

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The average incident rate for all plants for Class 3 events is 0.22 events per year per reactor. The incident rates for GE-3 (0.31) and GE-4 (0.30) plants are slightly above the average rate for all plants.

Sixty-six (81%) of the Category I events occurred in the Reactor Safety Systems--RHR, ECCS, and SCW (see Tables 1, 4, and 5). Forty-six (70%) of the Safety System events occurred in the GE-4 plants, which comprise 13 (50%) of the total number of plants.

Twenty-seven (33%) of the Category I events occurred in the RHR. Ten (37%) of the RHR events occurred at Brunswick 1 and 2. Six similar events in the RHR occurred within 14 months at Brunswick 1 and 2, accounting for 38% of the 16 Brunswick 1 and 2 events.

Twenty (25%) of the Category I events occurred in the HPCI (ECCS). Eight (40%) of the HPCI events occurred at Browns Ferry 1, 2, and 3, accounting for 67% of the 12 Browns Ferry 1, 2, and 3 events.

Ten (12%) of the Category I events occurred in the SCW. Four (40%) of the SCW events occurred at Browns Ferry 1, accounting for 44% of the 9 Browns Ferry 1 events. Three similar SCW events occurred within one month at Browns Ferry 1.

Eleven (14%) of the Category I events occurred in the Main Reactor Systems--Main Steam, Turbine, Condenser, and Feedwater (see Tables 1, 4, and 5). Eight (73%) of the Main Reactor System events occurred in the GE-3 plants. These plants account for 7 (27%) of the plants involved in this study.

Three (4%) of the Category I events occurred in the Auxiliary Reactor Systems--RCIC and RWCU (see Tables 1, 4, and 5). All three events occurred in the GE-3 plants.

One (1%) of the Category I events occurred in an "other" system.

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The incidence of events occurring in the Reactor Safety Systems is high for all three classes of events (see Table 5), but is slightly higher for Class 1 events (26 of 29 for 90%) than for Class 2 events (10 of 13 for 77%) and Class 3 events (30 of 39 for 77%).

A high percentage of the basis/cause for the events has been attributed to design or procedure deficiencies (see Table 6). The design-related incidence is significant for all three classes, but is higher for Class 1 events (16 of 29 for 55%) than for Class 2 events (4 of 13 for 31%) and Class 3 events (15 of 39 for 39%). The procedure-related incidence is also significant for all three classes, but again is higher for Class 1 events (18 of 29 for 62%) than for Class 2 events (5 of 13 for 39%) and Class 3 events (20 of 39 for 51%). It should be noted that some events are both design and procedure related.

The GE-4 plants attributed the basis/cause more to procedure-related problems (28%) than to design-related problems (15%).

The GE-3 plants attributed the basis/cause more to design-related problems (16%) than to procedure-related problems (13%).

Of the 10 events where the basis/cause of the event was attributed to component malfunction or failure -

- Seven (70%) were in the HPCI
- Three (30%) were a result of drainpot/switch problems
- One (10%) was a result of a check valve failure.

A high percentage of the event reports stated that the piping support systems were damaged (see Table 6). The incidence declines from Class 1 events (18 of 29 for 62%) to Class 2 events (7 of 13 for 54%) to Class 3 events (16 of 39 for 41%). The incidence of piping-related damage was significant for Class 1 events (11 of 29 for 38%) and Class 3 events (14 of 39 for 36%), but lower for Class 2 events (2 of 13 for 15%)--see Table 6.

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Twenty-nine (60%) of the GE-4 plant events resulted in piping support damage. Twelve (41%) of the GE-3 plant events resulted in piping support damage.

Thirteen (27%) of the GE-4 plant events resulted in piping-related damage. Thirteen (45%) of the GE-3 plant events resulted in piping-related damage.

The incident rates (events per year per reactor) for Class 3 events are more stable and lower than the Class 2 rates, which are lower than the Class 1 rates (see Table 7 and Figures 1-3). The average incident rate for all plants for Class 3 events is 0.22 events per year per reactor (see Table 3b).

A breakdown of the Category I events according to water hammer type (see Table 1) shows that:

- Steam-bubble collapse caused 13 events
- Flow-into-voided-line caused 29 events
- Steam-water entrainment caused 20 events
- Water column separation possibly caused five events
- Fourteen events were not specifically identifiable.

The involvement of the various types of water hammer relative to specific systems and subsystems for Category I events is discussed next.

Main Reactor Systems

Four main reactor subsystems (Main Steam, Turbine, Condenser and Feedwater) had 11 identified events.

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Main Steam. Six events were identified. Two were steam-water entrainment and four were steam hammer. Most events involved rapid isolation-valve actuation with high-pressure steam flow; one involved water accumulation in steam lines.

Turbine. No event was identified under Category I.

Condenser. Two events were identified. A column separation resulted from an inadvertent rapid valve closure. A steam-water entrainment resulted from condensate formation without adequate drainage.

Feedwater. Three events were identified. One involved column separation, and two were listed as unknown. Feedwater valve malfunction led to these events.

Auxiliary Reactor Systems

Two auxiliary subsystems (RCIC and RWCU) had three identified events. One unknown event in the RWCU system caused a cracked pipe. The two events in the RCIC involved steam-water entrainment caused by condensate formation and flow-into-voided-line caused by pump voiding problems.

Reactor Safety Systems

Four reactor safety subsystems (RHR, HPCI, CS, and SCW) had 66 identified events.

Residual Heat Removal (RHR). The RHR is used for both safety and auxiliary purposes. Because the RHR components are interconnected and a malfunction or failure during the auxiliary mode could adversely affect a safety function, all RHR events were classified under safety systems. Twenty-seven events were identified. Nineteen of these events involved an RHR auxiliary function, of which eight events were directly safety related. Thirteen events were identified as flow-into-voided-line, seven as steam-bubble collapse, four as steam-water entrainment, and three as unknown. Sixteen of the events in the RHR subsystem functions occurred via

voiding caused by leakage, venting, and filling problems. The steam-supply-and-exhaust-to-the-heat-exchanger/condenser portion of the RHR had eleven of the events. In addition to the voiding problems, one event occurred when water was introduced into the isolation condenser steam supply line because the reactor vessel water level had been maximized. The water level change was based on TMI experience. One other event involved an isolation condenser tube failure.

High-Pressure Coolant Injection (HPCI). Twenty events were identified. Twelve were identified as steam-water entrainment, six of which occurred during HPCI warmup. Five events involved water in steam lines because of drainage problems and one involved a gland seal failure. Four events were identified as steam-bubble collapse, and involved vacuum conditions caused by check-valve closure or rapid condensation due to lines being too cool. Three events were identified as flow-into-voided-line and resulted because of problems in the keep-full system. One was listed as unknown.

Core Spray (CS). Nine events were identified. Seven were identified as flow-into-voided-line, of which four involved filling, venting, or keep-full system problems, and three involved valve switches damaged by previous occurrences. One event, identified as steam-bubble collapse, also involved leakage and the keep-full system. One event was listed as unknown.

Service Cooling Water (SCW). Ten events were identified. Five were identified as flow-into-voided-line (of which four occurred at one plant), and involved filling and venting problems and one inadvertent line drainage. Three were identified as column separation, of which one involved an inadvertent rapid valve closure, and two involved unknowns. Two of the SCW events were listed as unknown types of water hammer.

Other Systems

One event occurred in the nonnuclear plant boiler steam system due to steam-bubble collapse. The situation involved possible breaching of the containment and interaction with nuclear systems, and thus was deemed to be of interest.

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TABLE 1. TYPES OF WATER HAMMER EVENTS OCCURRING IN BWR PLANT SYSTEMS (CATEGORY I)

Systems of Occurrence	Number of Events	Subsystem	Number and Type of Water Hammer	Event Number (Appendix A)
Main reactor (11 events)	6	Main steam	2 Steam-water entrainment	18,53
			4 Steam hammer	51,63,65,69
	--	Turbine		
	2	Condenser	1 Column separation	76 c
			-1 Steam-water entrainment	53,70
	3	Feedwater	1 Column separation	71
			2 Unknown	35,78
Auxiliary reactor (3 events)	2	RCIC	1 Flow-into-voided-line	77
			1 Steam-water entrainment	59
	1	RW cleanup	1 Unknown	38
Reactor safety (66 events)	3	RHR fuel pool cooling	2 Flow-into-voided-line	41,74
			1 Unknown	44
	5	RHR shut-down cooling	3 Flow-into-voided-line	21,22,23
			1 Steam-bubble collapse	33
			1 Unknown	64
	11	RHR steam supply and exhaust to Hx/ condenser	1 Flow-into-voided-line	20
			5 Steam-bubble collapse	14,15,17,25, 54
			4 Steam-water entrainment	49,55,57,61,
			1 Unknown	16
	1	RHR head spray	1 Flow-into-voided-line	45
	2	RHR un-identified	1 Flow-into-voided-line	43
			1 Steam-bubble collapse	26
	3	RHR con-tainment spray	3 Flow-into-voided-line	47,48,73

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TABLE 1. (continued)

Systems of Occurrence	Number of Events	Subsystem	Number and Type of Water Hammer	Event Number (Appendix A)
	2	LPCI (RHR) (ECCS)	2 Flow-into-voided-line	37,52
	20	HPCI (ECCS)	3 Flow-into-voided-line	19,27,29
			4 Steam-bubble collapse	2,7,60,79,8,
			12 Steam-water entrainment	9,10,11,12, 24,30,40,50, 66,67,80
			1 Unknown	13
	9	Core spray (ECCS)	7 Flow-into-voided-line	31,32,34,36, 39,42,62
			1 Steam-bubble collapse	58
			1 Unknown	56
	10	Service cooling water	3 Column separation	68,72,75
			5 Flow-into-voided-line	3,4,5,6,46
			2 Unknown	28,81
Other systems (1 event)	1	Plant process steam	1 Steam-bubble collapse	1
Total events	81			

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TABLE 2. NON-WATER-HAMMER EVENTS OCCURRING IN BWR PLANT SYSTEMS (CATEGORY II)

Systems of Occurrence	Number of Events	Subsystem	Number and Type of Event	Event Number (Appendix A)
Main reactor (16 Events)	1	Main steam	1 Potential for steam hammer	91
	8	Turbine	3 Potential for steam-bubble collapse	98,117,119
			2 Potential for steam hammer	92,93
			3 Potential for variety of water hammer	95,101,104
	--	Condenser		
	7	Feedwater	1 Component failure	94
			1 Component failure causing vibration	102
			4 Vibration causing component failure	115,120,121, 122
			1 Vibration restricting operation	124
Auxiliary Reactor auxiliary (7 events)	7	RCIC	5 Component failure that could lead to steam-bubble collapse	85,86,87, 116,123
			1 Potential for flow-into-voided-line or steam-bubble collapse	99
			1 Component failure that could lead to water hammer	100
	--	RW cleanup		
Reactor safety (20 events)	--	Fuel pool cooling		
	3	RHR shut-down cooling	2 Vibration causing component failure	107,108
			1 Component failure	113

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TABLE 2. (continued)

Systems of Occurrence	Number of Events	Subsystem	Number and Type of Event	Event Number (Appendix A)
	1	RHR steam supply and exhaust to Hx/Condenser	1 Component failure	103
	--	RHR head spray		
	--	RHR unidentified		
	--	Containment spray		
	--	LPCI		
	7	HPCI	1 Component failure that could lead to steam-bubble collapse	84
			3 Component failure	82,97,114
			2 Component failure that could lead to water hammer	83,88
			1 Component failure causing vibration	96
	1	Core spray	1 Component failure that could lead to flow-into-voided-line	118
	8	Service cooling water	5 Vibration causing component failure	89,90,106, 110,112
			3 Vibration causing component damage	105,109,111
Other (0 events)	--	Plant process steam		
Total events	43			

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TABLE 3a. BWR CATEGORY I EVENT BREAKDOWN BY PLANT

Plant	Commercial Operation Date	Years of Commercial Operation by NSSS Vendor Class ~~Form~~ From 1/1/69 to 5/1/81: GE-4	GE-3	Other	Number of Events by Year ( ) - Class 1 [ ] - Class 2 Class - 3: 69	70	71	72	73	74	75	76	77	78	79	80	81	Event Summaries: Category I #	%	Class 1 #	%	Class 2 #	%	Class 3 #	%	Class 3 Events Per Year[a]
Big Rock Pt.	3/29/63	--	--	12.33	--	--	--	--	--	--	--	--	1	--	--	--	--	1	1.2	--	--	--	--	1	2.6	.08
Browns Ferry 1	8/1/74	6.75	--	--	--	--	--	(1)	(5)	(1)	--	--	--	--	--	2	--	9	11.1	7	24.1	--	--	2	5.1	.35
Browns Ferry 2	3/1/75	6.17	--	--	--	--	--	--	--	(1)	--	--	--	--	--	1	--	2	2.5	1	3.4	--	--	1	2.6	.19
Browns Ferry 3	3/1/77	4.17	--	--	--	--	--	--	--	--	--	--	(1)	--	--	--	--	1	1.2	1	3.4	--	--	--	--	--
Brunswick 1	3/18/77	4.12	--	--	--	--	--	--	--	--	--	--	(1)[3]	--	1	--	2	7	8.6	1	3.4	3	23.1	3	7.7	.96
Brunswick 2	11/3/75	5.41	--	--	--	--	--	--	--	--	(3)	[2]	1	1	--	1	1	9	11.1	3	10.3	2	15.4	4	10.3	.91
Cooper	7/1/74	6.83	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Dresden 1	7/4/60	--	--	9.50	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Dresden 2	6/9/70	--	10.89	--	--	(1)	[2] 1	--	--	--	--	1	--	--	--	--	--	5	6.2	1	3.4	2	15.4	2	5.1	.20
Dresden 3	11/16/71	--	9.46	--	--	--	--	--	--	2	--	--	--	--	1	1	--	4	4.9	--	--	--	--	4	10.3	.47
Duane Arnold	2/1/75	6.25	--	--	--	--	--	--	--	(2)	--	--	2	1	1	--	--	6	7.4	2	6.9	--	--	4	10.3	.76
Fitzpatrick	7/28/75	5.76	--	--	--	--	--	--	--	(1)	(3)[1]	--	--	--	--	--	--	5	6.2	4	13.8	1	7.7	--	--	--
Hatch 1	12/31/75	5.33	--	--	--	--	--	--	--	(1)	--	--	--	--	--	--	--	1	1.2	1	3.4	--	--	--	--	--
Hatch 2	9/5/79	1.65	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Humbolt Bay	8/63	--	--	7.50	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
La Crosse	11/1/69	--	--	11.50	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Millstone 1	3/71	--	10.17	--	--	(1)	--	2	--	--	--	1	--	2	1	1	--	8	9.9	1	3.4	--	--	7	17.9	.76
Monticello	6/30/71	--	9.84	--	--	--	[1]	1	--	--	--	--	--	--	--	--	--	2	2.5	--	--	1	7.7	1	2.6	.11
Nine Mile Pt. 1	12/69	--	--	11.42	(1)	--	--	--	--	--	--	--	--	--	--	--	--	1	1.2	1	3.4	--	--	--	--	--
Oyster Creek	12/69	--	--	11.42	--	--	2	--	--	--	--	--	--	--	--	--	--	2	2.5	--	--	--	--	2	5.1	.19
Peach Bottom 2	7/5/74	6.82	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	1	1.2	--	--	--	--	1	2.6	.17
Peach Bottom 3	12/23/74	6.35	--	--	--	--	--	--	--	(1)	[2]	1	--	--	--	--	--	4	4.9	1	3.4	2	15.4	1	2.6	.19
Pilgrim 1	12/72	--	8.42	--	--	--	--	(1)[1]	--	--	--	1	1	--	--	--	--	4	4.9	1	3.4	1	7.7	2	5.1	.27
Quad Cities 1	2/18/73	--	8.20	--	--	--	--	(3)	[1]	--	--	1	--	--	--	--	--	5	6.2	3	10.3	1	7.7	1	2.6	.14
Quad Cities 2	3/10/73	--	8.14	--	--	--	--	--	--	--	1	--	--	--	--	--	--	1	1.2	--	--	--	--	1	2.6	.14
Vermont Yankee	11/30/72	8.34	--	--	--	--	(1)	--	--	--	--	1	--	--	1	--	--	3	3.7	1	3.4	--	--	2	5.1	.27

a. This column lists the number of Class 3 events divided by the amount of time a plant is in the Class 3 period.

Table 3a

See table 10b

Table 3b

TABLE 3b. BWR CATEGORY I EVENT BREAKDOWN BY GROUPS OF PLANTS

Plant	Number of Plants	Years of Commercial Operation by NSSS Vendor Class From 1/1/69 to 5/1/81: GE-4	GE-3	Other	Number of Events by Year: 69	70	71	72	73	74	75	76	77	78	79	80	81	Event Summaries: Category I #	%	Class 1 #	%	Class 2 #	%	Class 3 #	%	Class 3 Events Per Year[a]
All plants	26	73.95	65.12	63.67	1	2	7	9	6	9	11	8	10	4	5	6	3	81	100.0	--	--	--	--	39	100	.22
All plants reporting Class 1	15	--	--	--	1	2	1	5	5	7	6	--	2	--	--	--	--	--	--	29	100	--	--	--	--	--
All plants reporting Class 2	8	--	--	--	--	--	3	1	1	--	3	2	3	--	--	--	--	--	--	--	--	13	100	--	--	--
All plants reporting Class 3	17	50.21	65.12	23.75	--	--	3	3	--	2	2	6	5	4	5	6	3	--	--	--	--	--	--	39	100	.32
All GE-4 plants	13	73.95	--	--	--	--	1	1	5	7	10	4	8	2	3	4	3	48	59.3	--	--	--	--	18	46.2	.30
GE-4 plants reporting Class 1	10	--	--	--	--	--	1	1	5	7	6	--	2	--	--	--	--	--	--	22	75.9	--	--	--	--	--
GE-4 plants reporting Class 2	4	--	--	--	--	--	--	--	--	--	3	2	3	--	--	--	--	--	--	--	--	8	61.5	--	--	--
GE-4 plants reporting Class 3	8	50.21	--	--	--	--	--	--	--	--	1	2	3	2	3	4	3	--	--	--	--	--	--	18	46.2	.43
All GE-3 plants	7	--	65.12	--	--	2	4	8	1	2	1	4	1	2	2	2	--	29	35.8	--	--	--	--	18	46.2	.31
GE-3 plants reporting Class 1	4	--	--	--	--	2	--	4	--	--	--	--	--	--	--	--	--	--	--	6	20.7	--	--	--	--	--
GE-3 plants reporting Class 2	4	--	--	--	--	--	3	1	1	--	--	--	--	--	--	--	--	--	--	--	--	5	38.5	--	--	--
GE-3 plants reporting Class 3	7	--	65.12	--	--	--	1	3	--	2	1	4	1	2	2	2	--	--	--	--	--	--	--	18	46.2	.31

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Table 3b continued

TABLE 3b. (continued)

Plant	Number of Plants	Years of Commercial Operation by NSSS Vendor Class Form 1/1/69 to 5/1/81: GE-4	GE-3	Other	Number of Events by Year: 69	70	71	72	73	74	75	76	77	78	79	80	81	Event Summaries: Category I #	%	Class 1 #	%	Class 2 #	%	Class 3 #	%	Class 3 Events Per Year[a]
All other plants	6	--	--	63.67	1	--	2	--	--	--	--	--	1	--	--	--	--	4	4.9	--	--	--	--	3	7.7	.05
Other plants reporting Class 1	1 of 3	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	3.5	--	--	--	--	--
Other plants reporting Class 2	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Other plants reporting Class 3	2	--	--	23.75	--	--	2	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	3	7.7	.13

a. This column lists the number of Class 3 events divided by the amount of time a plant is in the Class 3 period.

TABLE 4a. BWR CATEGORY I EVENTS PER SYSTEM[a] BY EVENT CLASS

Plant	NSSS Vendor Class	Events Per System by Event Class: Class 1									Class 2									Class 3									
		1	2	3	4	5	6	7	8	9	1	2	3	4	5	6	7	8	9	1	2	3	4	5	6	7	8	9	10
Big Rock Pt.	Other	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1
Browns Ferry 1	GE-4	--	--	--	--	--	--	3	--	4	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	2	--	--	--
Browns Ferry 2	GE-4	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--
Browns Ferry 3	GE-4	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Brunswick 1	GE-4	--	--	--	--	--	1	--	--	--	--	--	--	--	--	3	--	--	--	1	--	--	--	--	1	1	--	--	--
Brunswick 2	GE-4	--	--	--	--	--	3	--	--	--	--	--	--	--	--	1	1	--	--	--	--	--	--	--	1	2	--	1	--
Dresden 2	GE-3	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	2	--	--	--	--	--	--	1	--	1	--	--
Dresden 3	GE-3	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	1	1	--	1	--	--
Duane Arnold	GE-4	--	--	--	--	--	--	1	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	3	--	1	--	--
Fitzpatrick	GE-4	--	--	--	--	--	3	--	--	1	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--
Hatch 1	GE-4	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Millstone 1	GE-3	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	4	--	2	--	--
Monticello	GE-3	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--
Nine Mile Pt. 1	Other	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Oyster Creek	Other	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	1	--	--
Peach Bottom 2	GE-4	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--
Peach Bottom 3	GE-4	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	2	--	--	--	--	--	--	--	--	--	--	1	--
Pilgrim 1	GE-3	1	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	1	--
Quad Cities 1	GE-3	--	--	--	--	--	2	--	--	1	--	1	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--
Quad Cities 2	GE-3	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--
Vermont Yankee	GE-4	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	1	--

a. LEGEND for systems:

Main	Auxiliary	Safety		Other
1 = Main steam	4 = RCIC	6 = RHR	8 = Core spray	10 = Plant process steam
2 = Condenser	5 = RW cleanup	7 = HPCI	9 = SCW	
3 = Feedwater				

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Table 4a

Table 4b

TABLE 4b. SUMMARY OF BWR CATEGORY I EVENTS BY VENDOR CLASS AND EVENT CLASS

NSSS Vendor Class	Events Per System by Event Class																											
	Class 1									Class 2									Class 3									
	1	2	3	4	5	6	7	8	9	1	2	3	4	5	6	7	8	9	1	2	3	4	5	6	7	8	9	10
GE-4	1	--	--	--	--	8	7	1	5	--	--	--	--	--	4	4	--	--	1	--	--	--	--	6	7	1	3	--
GE-3	2	--	--	--	--	2	1	--	1	--	2	--	1	--	--	--	2	--	1	--	3	1	1	6	1	4	1	--
Other	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	1	--	1
Total	3	--	--	--	--	11	8	1	6	--	2	--	1	--	4	4	2	--	3	--	3	1	1	12	8	6	4	1

Delete ( ) around # and number under # column
Keep ( ) around % and add around number under % column
easier to read with one line under # and % # (%)

TABLE 4c. SUMMARY OF BWR CATEGORY I EVENTS BY VENDOR CLASS AND SYSTEM

NSSS Vendor Class	Events Per System																			
	1		2		3		4		5		6		7		8		9		10	
	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)
GE-4	(2)	33	--		--		--		--		(18)	66	(18)	90	(2)	22	(8)	80	--	
GE-3	(3)	50	(2)	100	(3)	100	(2)	100	(1)	100	(8)	30	(2)	10	(6)	67	(2)	20	--	
Other	(1)	17	--		--		--		--		(1)	4	--		(1)	11	--		(1)	100
Total	(6)	100	(2)	100	(3)	100	(2)	100	(1)	100	(27)	100	(20)	100	(9)	100	(10)	100	(1)	100

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TABLE 4d. SUMMARY OF BWR CATEGORY I EVENTS BY VENDOR CLASS AND MAJOR SYSTEMS

NSSS Vendor Class	Events Per Major Systems							
	Main Systems (#)	Main Systems (%)	Auxiliary Systems (#)	Auxiliary Systems (%)	Safety Systems (#)	Safety Systems (%)	Other (#)	Other (%)
GE-4	(2)	18	--	--	(46)	70	--	--
GE-3	(8)	73	(3)	100	(18)	27	--	--
Other	(1)	9	--	--	(2)	3	(1)	100
Total	(11)	100	(3)	100	(66)	100	(1)	100

delete ( ) around # and number under that column
keep ( ) around % and add to number under that column

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TABLE 5. BWR CATEGORY I EVENTS BY MAJOR SYSTEMS AND EVENT CLASS

	Events by Event Class					
Major Systems[a]	All Classes (#) (%)	Class 1 (#) (%)	Class 2 (#) (%)	Class 3 (#) (%)	Class 3 (Pre-78) (#) (%)	Class 3 (78 to Present) (#) (%)
Main reactor	(11) 14	(3) 10	(2) 15	(6) 15	(5) 24	(1) 6
Auxiliary reactor	(3) 4	(0) 0	(1) 8	(2) 5	(1) 5	(1) 6
Reactor safety	(66) 81	(26) 90	(10) 77	(30) 77	(14) 67	(16) 89
Other	(1) 1	-- --	-- --	(1) 3	(1) 5	(0) 0

a. LEGEND for systems: Caps.

Main reactor systems: Main steam, Condenser, Feedwater
Auxiliary reactor systems: RCIC, RW cleanup
Reactor safety systems: RHR, HPCI, Core spray, SCW
Other: Plant Process Steam.

delete ( ) around # and numbers under that column
keep ( ) around % and add to numbers under that column

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TABLE 6a. BWR CATEGORY I EVENTS BY PLANT, BASIS/CAUSE, AND DAMAGE

				Basis/Cause for Event			Items Damaged		
Event	Plant	Vendor Class	Plant System	Class 1	Class 2	Class 3	Class 1	Class 2	Class 3
1	Big Rock Pt	Other	Plant process steam	--	--	Design/ procedure	--	--	Release of rad. water to disch. canal
2	Browns Ferry 1	GE-4	HPCI	Design	--	--	Not indicated (steam noise)	--	--
3	Browns Ferry 1	GE-4	SCW	Design/ procedure	--	--	Orifice gasket	--	--
4	Browns Ferry 1	GE-4	SCW	Design/ procedure	--	--	Pipe coupling	--	--
5	Browns Ferry 1	GE-4	SCW	Design/ procedure	--	--	Pipe coupling	--	
6	Browns Ferry 1	GE-4	SCW	Design/ procedure	--	--	Pipe coupling	--	
7	Browns Ferry 1	GE-4	HPCI	Procedure/ design	--	--	Rupture disc	--	--
8	Browns Ferry 1	GE-4	HPCI	Procedure/ design	--	--	Supports/valve Switch/pedestal	--	--
9	Browns Ferry 1	GE-4	HPCI	--	--	Procedure	--	--	Bearing support pedestal
10	Browns Ferry 1	GE-4	HPCI	--	--	Procedure	--	--	Supports
11	Browns Ferry 2	GE-4	HPCI	Installation	--	--	Rupture disc	--	--
12	Browns Ferry 2	GE-4	HPCI	--	--	Procedure	--	--	Bearing support pedestal
13	Browns Ferry 3	GE-4	HPCI	Unknown	--	--	Support	--	--
14	Brunswick 1	GE-4	RHR steam supply/exhaust	Procedure	--	--	Snubbers	--	--
15	Brunswick 1	GE-4	RHR steam supply/exhaust	--	Procedure	--	--	Snubbers	--
16	Brunswick 1	GE-4	RHR steam supply/exhaust	--	Procedure	--	--	Snubber	--

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TABLE 6a. (continued)

Event	Plant	Vendor Class	Plant System	Basis/Cause for Event: Class 1	Basis/Cause for Event: Class 2	Basis/Cause for Event: Class 3	Items Damaged: Class 1	Items Damaged: Class 2	Items Damaged: Class 3
17	Brunswick 1	GE-4	RHR steam supply/exhaust	--	Procedure	--	--	Support	--
18	Brunswick 1	GE-4	Main steam	--	--	Not indicated	--	--	Snubbers
19	Brunswick 1	GE-4	HPCI	--	--	Component malfunction	--	--	Weld failure/ guide seal
20	Brunswick 1	GE-4	RHR steam supply/exhaust	--	--	Design/ procedure	--	--	Snubber
21	Brunswick 2	GE-4	RHR shutdown cooling	Procedure	--	--	Supports/insulation	--	--
22	Brunswick 2	GE-4	RHR shutdown cooling	Procedure	--	--	Supports/insulation	--	--
23	Brunswick 2	GE-4	RHR shutdown cooling	Procedure	--	--	Supports/insulation	--	--
24	Brunswick 2	GE-4	HPCI	--	Component failure/ inspection	--	--	Support/ snubber	--
25	Brunswick 2	GE-4	RHR steam supply/exhaust	--	Procedure	--	--	Snubber	--
26	Brunswick 2	GE-4	RHR unidentified	--	--	Installation/ administrative controls	--	--	Snubbers
27	Brunswick 2	GE-4	HPCI	--	--	Inoperative check valve	--	--	Support
28	Brunswick 2	GE-4	SCW	--	--	Design/ procedure	--	--	Hx rib plate
29	Brunswick 2	GE-4	HPCI	--	--	Component malfunction	--	--	Weld failure/ guide seal
30	Dresden 2	GE-3	HPCI	Design/ procedure	--	--	Supports/snubbers Piping/insulation	--	--
31	Dresden 2	GE-3	Core spray	--	Design	--	--	Valve switch	--
32	Dresden 2	GE-3	Core spray	--	Procedure	--	--	Supports	

TABLE 6a. (continued)

Event	Plant	Vendor Class	Plant System	Basis/Cause for Event: Class 1	Basis/Cause for Event: Class 2	Basis/Cause for Event: Class 3	Items Damaged: Class 1	Items Damaged: Class 2	Items Damaged: Class 3
33	Dresden 2	GE-3	RHR shutdown cooling	--	--	Procedure	--	--	Insulation
34	Dresden 2	GE-3	Core spray	--	--	Design/ procedure	--	--	Valve switch
35	Dresden 3	GE-3	Feedwater	--	--	Unknown	--	--	Supports/piping
36	Dresden 3	GE-3	Core spray	--	--	Design/ procedure	--	--	Valve switch
37	Dresden 3	GE-3	RHR LPCI	--	--	Design/ procedure	--	--	Supports
38	Dresden 3	GE-3	RW cleanup	--	--	Design/ procedure	--	--	Pipe cracked
39	Duane Arnold	GE-4	Core spray	Administrative controls	--	--	Support	--	--
40	Duane Arnold	GE-4	HPCI	Proced/design/ operator	--	--	Supports/snubber Instruments/insul.	--	--
41	Duane Arnold	GE-4	RHR fuel pool cooling	--	--	Procedure	--	--	Support
42	Duane Arnold	GE-4	Core spray	--	--	Procedure	--	--	Valve operator
43	Duane Arnold	GE-4	RHR unidentified	--	--	Procedure	--	--	Snubbers
44	Duane Arnold	GE-4	RHR fuel pool cooling	--	--	Not indicated	--	--	Supports
45	Hatch 1	GE-4	RHR head spray	Operator/ procedure/ design	--	--	Pipe cracked	--	--
46	Fitzpatrick	GE-4	SCW	Administrative controls/ procedure	--	--	Strainer/supports	--	--
47	Fitzpatrick	GE-4	RHR containment spray	Design	--	--	Supports/snubber		

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TABLE 6a. (continued)

				Basis/Cause for Event			Items Damaged		
Event	Plant	Vendor Class	Plant System	Class 1	Class 2	Class 3	Class 1	Class 2	Class 3
48	Fitzpatrick	GE-4	RHR containment spray	Not indicated	--	--	Supports/snubbers	--	--
49	Fitzpatrick	GE-4	RHR steam supply/exhaust	Procedure	--	--	Support	--	--
50	Fitzpatrick	GE-4	HPCI	--	Design	--	--	Support	--
51	Millstone 1	GE-3	Main steam	Design	--	--	Supports/piping	--	--
52	Millstone 1	GE-3	RHR LPCI	--	--	Procedure	--	--	Support shift
53	Millstone 1	GE-3	Condenser	--	--	Design	--	--	Piping
54	Millstone 1	GE-3	RHR steam supply/exhaust	--	--	Condenser tube failure	--	--	Condenser
55	Millstone 1	GE-3	RHR steam supply/exhaust	--	--	Procedure	--	--	No damage (pipe movement)
56	Millstone 1	GE-3	Core spray	--	--	Design	--	--	Supports
57	Millstone 1	GE-3	RHR steam supply/exhaust	--	--	Procedure	--	--	No damage
58	Millstone 1	GE-3	Core spray	--	--	Component malfunction	--	--	Support
59	Monticello	GE-3	RCIC	--	Design	--	--	Not indicated	--
60	Monticello	GE-3	HPCI	--	--	Component failure	--	--	Rupture disc
61	Nine Mile PT. 1	Other	RHR steam supply/exhaust	Design	--	--	Not indicated	--	--
62	Oyster Creek	Other	Core spray	--	--	Design/ procedure	--	--	Possible overstress
63	Oyster Creek	Other	Main steam	--	--	Design	--	--	Valve
64	Peach Bottom 2	GE-4	RHR shutdown cooling	--	--	Not indicated	--	--	Support

TABLE 6a. (continued)

Event	Plant	Vendor Class	Plant System	Basis/Cause for Event: Class 1	Class 2	Class 3	Items Damaged: Class 1	Class 2	Class 3
65	Peach Bottom 3	GE-4	Main steam	Maintenance	--	--	Snubbers	--	--
66	Peach Bottom 3	GE-4	HPCI	--	Component failure	--	--	No damage	--
67	Peach Bottom 3	GE-4	HPCI	--	Component failure	--	--	Not indicated (Line movement)	--
68	Peach Bottom 2/3	GE-4	SCW	--	--	Not indicated	--	--	Pipe cracked/ failed bolts
69	Pilgrim	GE-3	Main steam	Design	--	--	Supports	--	--
70	Pilgrim	GE-3	Condenser	--	Design	--	--	Condenser	--
71	Pilgrim	GE-3	Feedwater	--	--	Design	--	--	Valve/support
72	Pilgrim	GE-3	SCW	--	--	Unknown	--	--	Pipe fracture
73	Quad Cities 1	GE-3	RHR containment spray	Not indicated	--	--	Supports	--	--
74	Quad Cities 1	GE-3	RHR fuel pool cooling	Design/ procedure	--	--	Supports	--	--
75	Quad Cities 1	GE-3	SCW	Procedure	--	--	Expansion joint	--	--
76	Quad Cities 1	GE-3	Condenser	--	Component malfunction	--	--	Expansion joint	--
77	Quad Cities 1	GE-3	RCIC	--	--	Procedure/ design	--	--	Pump
78	Quad Cities 2	GE-3	Feedwater	--	--	Design	--	--	Piping
79	Vermont Yankee	GE-4	HPCI	Design/ procedure	--	--	Not indicated	--	--
80	Vermont Yankee	GE-4	HPCI	--	--	Operator/ design	--	--	Seal gasket
81	Vermont Yankee	GE-4	SCW	--	--	Procedure	--	--	Support

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TABLE 6b. BWR CATEGORY I EVENT TOTALS BY BASIS/CAUSE AND EVENT CLASS

	Events by Event Class					
	Class 1		Class 2		Class 3	
Basis/Cause	(#)	(%)	(#)	(%)	(#)	(%)
Design	(5)	17	(4)	31	(5)	13
Design/operater	--	--	--	--	(1)	3
Design/procedure/operator	(2)	7	--	--	--	--
Design/procedure	(9)	31	--	--	(9)	23
Procedure	(6)	21	(5)	39	(11)	28
Procedure/admin control	(1)	3	--	--	--	--
Operater, admin control, installation/maintenance	(3)	10	--	--	(1)	3
Component failure/ malfunction	--	--	(4)	31	(6)	15
Unknown	(3)	10	--	--	(6)	15
TOTAL	(29)	100	(13)	100	(39)	100
Design related[a]	(16)	55	(4)	31	(15)	39
Procedure related[a]	(18)	62	(5)	39	(20)	51

a. Some events are both design and procedure related.

delete ( ) around # and numbers under that column
keep ( ) around % and add to numbers under that column

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TABLE 6c. BWR CATEGORY I EVENT TOTALS BY DAMAGE AND EVENT CLASS

Items Damaged	Class 1 (#)	Class 1 (%)	Class 2 (#)	Class 2 (%)	Class 3 (#)	Class 3 (%)
Snubbers, supports snubber/supports	(10)	35	(7)	54	(14)	36
Supports/snubbers/piping	(3)	10	--	--	(1)	3
Supports/snubbers/misc	(4)	14	--	--	--	--
Supports/valve	(1)	3	--	--	(1)	3
Valve, valve switch	--	--	(1)	8	(4)	10
Piping, pumps, and related	(8)	28	(2)	15	(13)	33
Coolant release to canal	--	--	--	--	(1)	3
Pedestal, insulation	--	--	--	--	(3)	8
Unknown, no damage, not indicated	(3)	10	(3)	23	(2)	5
TOTAL	(29)	100	(13)	100	(39)	100
Snubber, support related[a]	(18)	62	(7)	54	(16)	41
Piping related[a]	(11)	38	(2)	15	(14)	36

a. Some events are snubber, support, and piping related.

delete ( ) around # and numbers under that column
keep ( ) around % and add to numbers under that column

TABLE 6d. BWR CATEGORY I EVENT TOTALS BY BASIS/CAUSE AND VENDOR CLASS

Basis/Cause	Events by NSSS Vendor Class					
	GE-4		GE-3		Other	
	(#)	(%)	(#)	(%)	(#)	(%)
Design	(3)	6	(9)	31	(2)	50
Design/operator	(1)	2	--	--	--	--
Design/procedure/ operator	(2)	4	--	--	--	--
Design/procedure	(9)	19	(7)	24	(2)	50
Procedure	(16)	33	(6)	21		
Procedure/admin. control	(1)	2	--	--	--	--
Operator, admin. control, installation maintenance	(4)	8	--	--	--	--
Component failure/ malfunction	(6)	13	(4)	14	--	--
Unknown, no damage, not indicated	(6)	13	(3)	10	--	--
TOTAL	(48)	100	(29)	100	(4)	100
Design related[a]	(15)	31	(16)	55	(4)	100
Procedure related[a]	(28)	58	(13)	45	(2)	50

a. Some events are both design and procedure related.

delete ( ) around # and numbers under that column
keep ( ) around % and add to numbers under that column

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TABLE 6e. BWR CATEGORY I EVENT TOTALS BY DAMAGE AND VENDOR CLASS

Items Damaged	Events by NSSS Vendor Class: GE-4 (#)	GE-4 (%)	GE-3 (#)	GE-3 (%)	Other (#)	Other (%)
Snubbers, supports snubber/supports	(23)	48	(8)	28	--	--
Supports/snubbers/piping	(1)	2	(3)	10	--	--
Supports/snubbers/misc	(4)	8	--	--	--	--
Supports/valve	(1)	2	(1)	3	--	--
Valve, valve switch	(1)	2	(3)	10	(1)	25
Piping, pumps, and related	(12)	25	(10)	34	(1)	25
Coolant release to canal	--	--	--	--	(1)	25
Pedestal, insulation	(2)	4	(1)	3	--	--
Unknown, no damage, not indicated	(4)	8	(3)	10	(1)	25
TOTAL	(48)	100	(29)	100	(4)	100
Snubber, support related[a]	(29)	60	(12)	41	--	--
Piping related[a]	(13)	27	(13)	45	(1)	25

a. Some events are snubber, support, and piping related.

Delete ( ) around # and numbers under that column
Keep ( ) around % and add to numbers under that column

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Table 7.

Tables 7 & 14 should be laid out the same

TABLE 7. BWR CATEGORY I NORMALIZED INCIDENT RATES

	Class 1			Class 2			Class 3		
Event Year	Plant Years[a]	Number of Events	Incident Rate	Plant Years[a]	Number of Events	Incident Rate	Plant Years[a]	Number of Events	Incident Rate
69	3.23	1	.31	.33	--	--	3.00	--	--
70	1.90	2	1.05	3.23	--	--	3.33	--	--
71	1.71	1	.58	1.90	3	1.58	7.56	3	.40
72	3.51	5	1.42	1.71	1	.58	9.46	3	.32
73	1.76	5	2.84	3.51	1	.28	11.17	--	--
74	4.91	7	1.43	1.76	--	--	14.68	2	.14
75	2.67	6	2.25	4.91	3	.61	16.43	2	.12
76	1.62	--	--	2.67	2	.75	21.34	6	.28
77	.38	2	5.26	1.62	3	1.85	23.00	5	.22
78	.32	--	--	.38	--	--	24.62	4	.16
79	.68	--	--	.32	--	--	25.00	5	.20
80	NA	NA	NA	.68	--	--	25.32	6	.24

a. Plant years are the total amount of time that all plants are within the time period for a given event class (1, 2, or 3) for a given event year.

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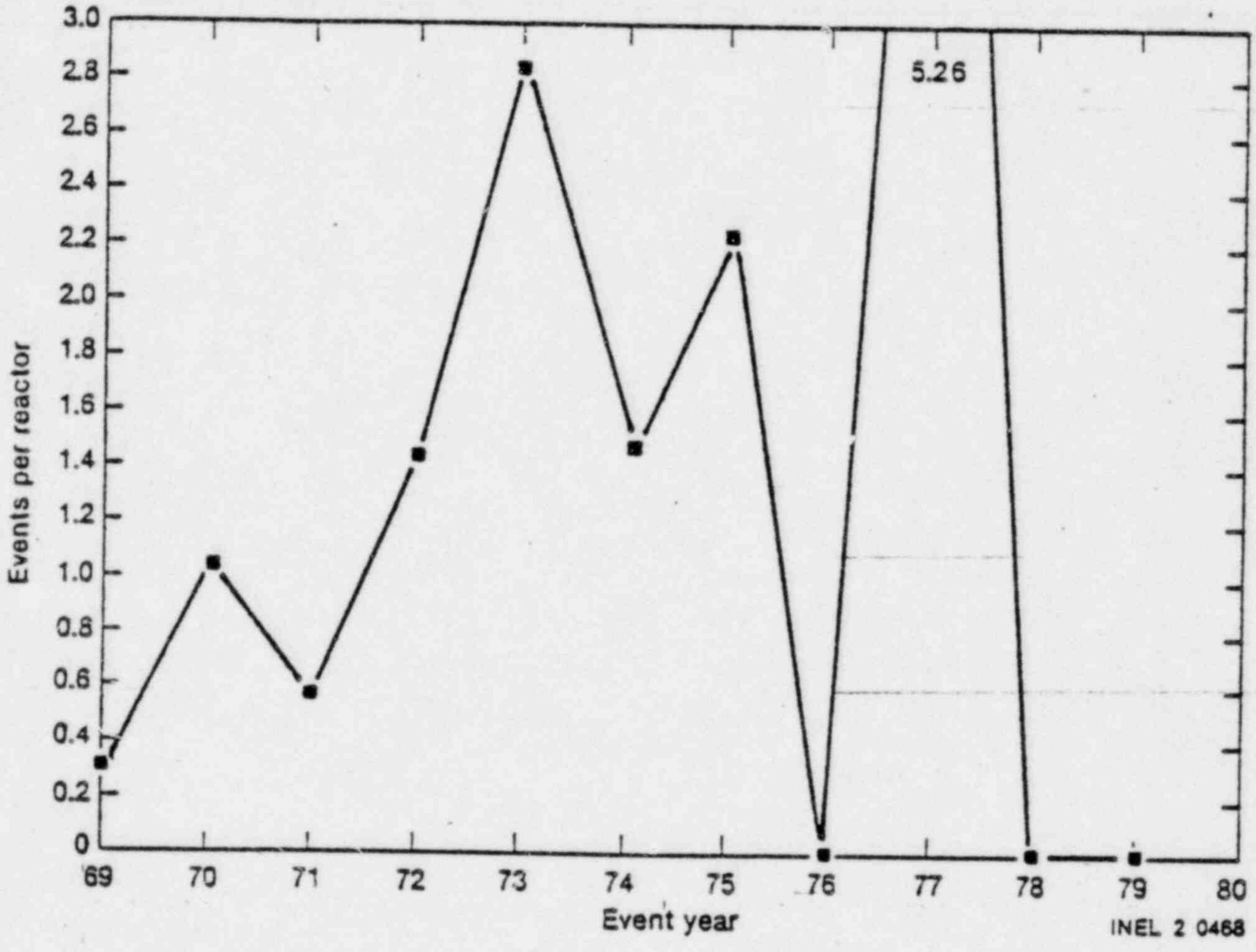
Figure 1



Need Table

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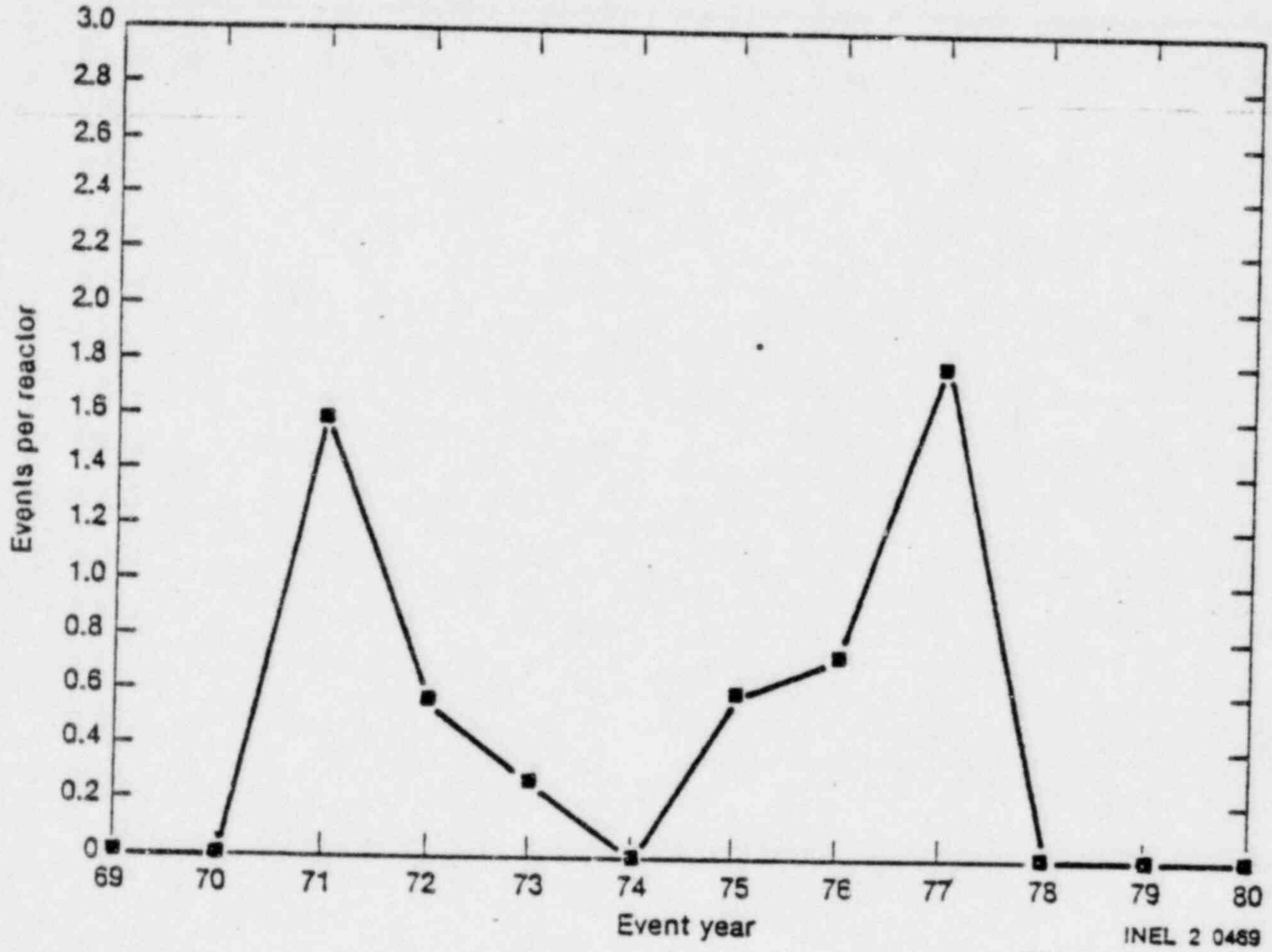
Figure 2



Need title

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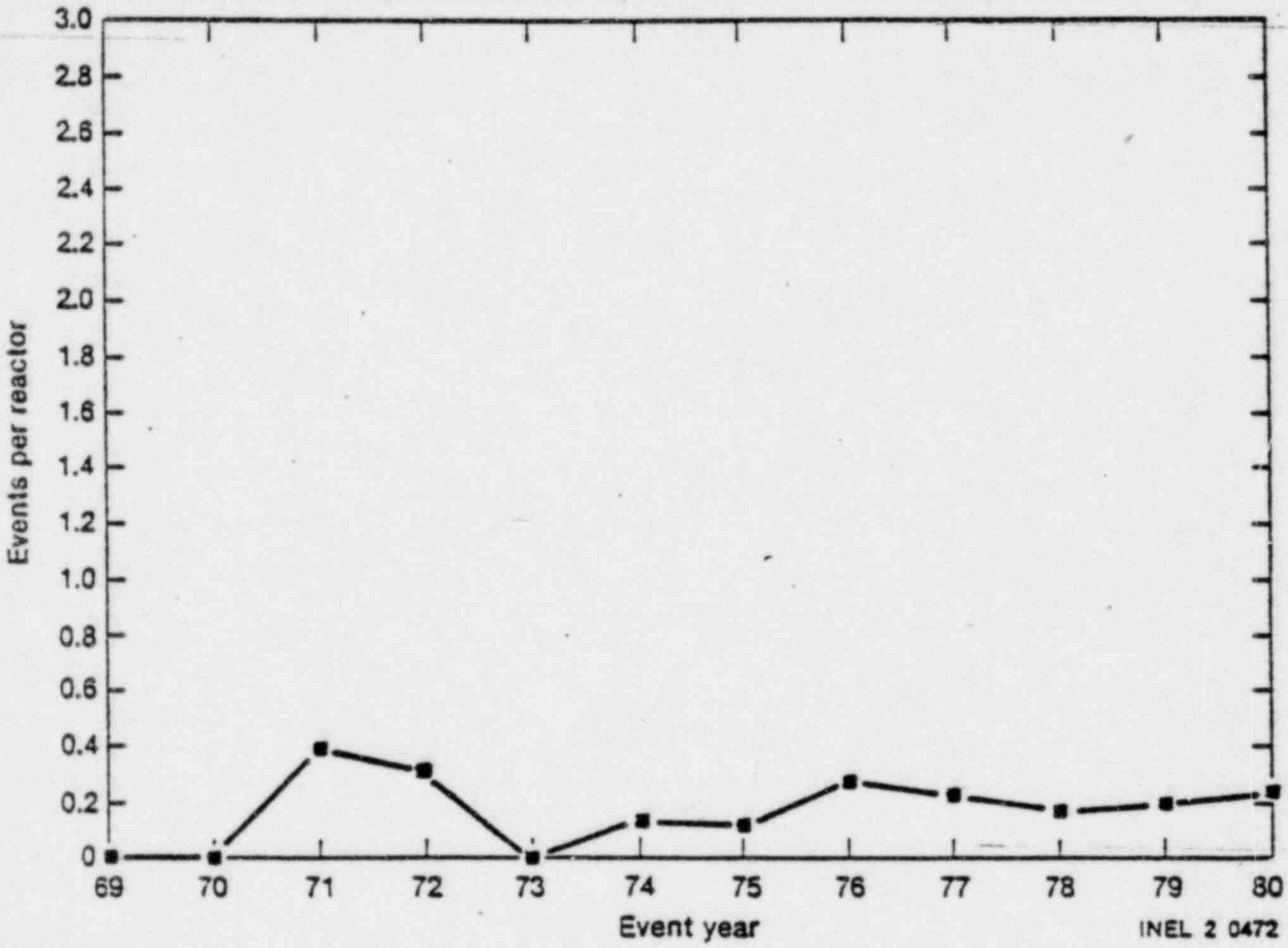
Figure 3



Need Title

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PWR Non-Steam-Generator Events

Forty-four PWR plants were involved in this study. Indian Point 1, TMI-1, and TMI-2 are currently not operating, though hearings are underway regarding TMI-1 restart ~~TMI-1 is scheduled to resume operation shortly~~. These plants are included in the study, because they are part of the base from which information was extracted. The following information was taken from the tables and figures previously mentioned.

This study involves 44 PWR plants at 30 locations and covers only the 40 Category I events. Here is the plant breakdown:

- Twenty-seven (61%) are Westinghouse (W)
- Eight (18%) are Combustion Engineering (CE)
- Nine (21%) are Babcock and Wilcox (B&W).

Category I events were reported by

- Thirteen (48%) of the W plants
- Three (38%) of the CE plants
- Five (56%) of the B&W plants.

Category I events occurred in

- Twenty-seven (68%) of the W plants
- Five (13%) of the CE plants
- Eight (20%) of the B&W plants.

This should be the number of events per plant type.

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The percentage of events occurring in the three different vendor plants is in proportion to the number of plants a vendor has supplied.

The plants reported Category I events as follows (see Table 10):

- Twenty-one (48%) reported at least one
- Thirteen (62%) reported only one.

Eleven (28%) of the events occurred at two (5%) of the plants and 27 (68%) occurred at eight (18%) of the plants.

Eleven (28%) of the events are Class 1 (see Table 10). Seven (16%) of the plants reported Class 1 events. Five (45%) of the Class 1 events occurred at Beaver Valley 1.

Six (15%) of the events are Class 2 (see Table 10). Six (14%) of the plants reported Class 2 events. No plant had more than one Class 1 event.

Twenty-three (58%) of the events are Class 3 (see Table 10). Sixteen (36%) of the plants reported Class 3 events. Five (22%) of the Class 3 events occurred at San Onofre 1. Ten (43%) of the Class 3 events occurred at Ginna, San Onofre 1, and Zion 1.

The average incident rate for those plants reporting Class 3 events is 0.19 events per year per reactor (see Table 10). The only plants above this average are:

- Beaver Valley 1 (0.33)
- Ginna (0.29)
- San Onofre (0.41)
- Zion 1 (0.32).

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The average incident rate for all plants for Class 3 events is 0.09 events per year per reactor. The incident rates for all the W and B&W plants are close to this average, with the CE plants being somewhat lower.

Eight (20%) of the Category I events occurred in the Reactor Safety Systems--RHR, ECCS, and SCW (see Tables 8, 11, and 12). One (3%) of the Category I events occurred in the RHR. Four (10%) of the Category I events occurred in the ECCS. Three (8%) of the Category I events occurred in the SCW.

Thirty (75%) of the Category I events occurred in the Main Reactor Systems--RCS, Condenser, Main Steam, and Main Feedwater (see Tables 8 and 11). Five (13%) of the Category I events occurred in the RCS. Four (10%) of the Category I events occurred in the Condenser system. Six (15%) of the Category I events occurred in the Main Steam system. Fifteen (38%) of the Category I events occurred in the Main Feedwater System, of which 13 (93%) occurred in six Westinghouse plants.

Two (5%) of the Category I events occurred in Auxiliary Reactor Systems--RCIC and RWCU (see Tables 8 and 11).

The incidence of events occurring in the Main Reactor Systems is high for all three classes of events, but decreases from Class 1 (9 of 11 for 82%) and Class 2 (6 of 6 for 100%) to Class 3 (15 of 23 for 65%) as the Reactor Safety System incidence rises from Class 1 (1 of 11 for 9%) and Class 2 (0) to Class 3 (7 of 23 for 30%)--see Table 12.

A high percentage of the basis/cause for the events has been attributed to design or procedure deficiencies (see Table 13). The design-related incidence decreases from Class 1 (10 of 11 for 91%) to Class 2 (5 of 6 for 83%) to Class 3 (14 of 23 for 61%) as the procedure-related incidence increases from Class 1 (0) to Class 2 (1 of 6 for 17%) to Class 3 (11 of 23 for 48%). It should be noted that some events are both design and procedure related. All three different types of vendor plants attributed the basis/cause more to design-related problems than to procedure-related problems.

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A significant percentage of the event reports listed damage to the piping support systems for Clas (5 of 11 for 45%) and Class 2 (2 of 6 for 33%), and to piping-related systems for Class 1 (5 of 11 for 45%) and Class 2 (3 of 6 for 50%). The incidence of events causing damage to piping support systems was high 3 of 23 for 57%) than piping-related systems (5 of 23 for 22%) for Class 3 events (see Table 13). Piping support damage resulted from:

- Fifteen (56%) of the W events
- One (20%) of the CE events
- Four (50%) of the B&W events.

Piping-related damage resulted from:

- Nine (33%) of the W events
- Two (40%) of the CE events
- Two (25%) of the B&W events.

The incident rates (events per year per reactor) for Class 3 events are more stable and lower than the Class 2 and Class 1 rates (see Table 14 and Figures 4-6). The average incident rate for all plants for Class 3 events is 0.09 events per year per reactor (see Table 10).

A breakdown of the Category I events according to water hammer type (see Table 8) shows that:

- Steam-bubble collapse caused two events
- Flow-into-voided-line caused 13 events
- Steam-water entrainment caused one event

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- Water column separation caused no events

- Steam hammer caused 21 events

- Three events were not identified.

The steam hammer category includes a phenomenon called wave reflection. Such events usually involved the partial or total interruption of flow in a fluid system because of pump seizure, valve closure, or valve flutter. In liquid systems, this event is usually classified as column separation if the pressure drop is sufficient to produce vaporization.

The involvement of the various types of water hammer in specific systems and subsystems for Category I events is discussed next.

Main Reactor Systems

Four main reactor subsystems (RCS, Condenser, Main steam, and Main Feedwater) had thirty identified events.

Reactor Coolant System (RCS). Five events were identified. All involved flow-into-voided-line and occurred in the pressurizer discharge line when the pressurizer relief valve opened and a slug of water was propelled into an essentially voided line. In four of the incidents, the relief valve was installed with an upstream water seal to prevent valve seat erosion. In the remaining incident, the valve stuck open due to boric acid buildup.

Condenser. Four events were identified. One event, identified as possible wave reflection (steam hammer), involved the rapid closure of a deaerator level regulating valve, coupled with a long pipe run. Three events, identified as flow-into-voided-line, involved the condenser tubes and inlet piping.

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Main Steam. Six events were identified only as steam hammer or unknown, of which three were caused by inadvertent valve actuation.

Main Feedwater. Fifteen events were identified. One event, identified as steam-bubble collapse, involved a valve lineup that allowed a back flow from the steam generator to the feedwater inlet line. One event, identified as steam-water entrainment, involved the turbine exhaust line drain, which had too much back pressure and allowed the collection of rain water. Eleven events were identified as steam hammer, of which eight were directly related to valve failure, design, and instability, two were related to valve adjustment and opening, and one was unknown. Two events were not identified.

Auxiliary Reactor Systems

The Chemical and Volume Control system had two identified events. In one event, identified as flow-into-voided-line, vent line vibration caused pipe shear, which resulted in reactor coolant release. One event, identified as steam hammer, involved valve instability after the positioner fell off.

Reactor Safety Systems

Three reactor safety subsystems (ECCS, RHR, and SCW) had eight identified events.

Emergency Core Cooling System (ECCS). Four events were identified. Three events, identified as flow-into-voided-line, involved the safety injection lines not being water filled. One event, identified as steam-bubble collapse, involved pressure reduction in a safety injection line during testing.

Residual Heat Removal (RHR). One event, identified as steam hammer, involved a pump start with an incorrect valve line up.

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Service Cooling Water (SCW). Three events were identified. One event, identified as flow-into-voided-line, involved the cooling water flow to the RHR Hx not being throttled. Two events were identified as steam hammer, of which one involved a control valve line from the diesel generator air cooler Hx, and the other involved the failure of a tsunami gate in the circulating water system.

Tables 8-14 and figures 4, 5, 6 go here

PWR Steam Generator Events — This comes after the figures and Tables for PWR non-SG events

The following information was taken from Table 15 and Appendix C. Thirteen plants reported 27 of the PWR steam generator events:

- W plants reported 23 (85%)
- CE plants reported 4 (15%)
- B&W plants reported none.

Zion 1 (8) and Zion 2 (5) reported 13 (48%) of the events.

Six (22%) of the events are Class 1. Six (22%) of the events are Class 2. Fifteen (56%) of the events are Class 3, of which 11 (73%) occurred at Zion 1 and 2.

Many of the events tabulated represent one or more events recorded during a single short time span under the same plant conditions. Many events were not observed at the time of occurrence, but the damage observed indicated that these events were caused by SGWH. Past SGWH events have varied greatly in magnitude and consequences. Effects reported have ranged from minor noises and feedwater piping vibration to major feedwater system support damage and feedwater piping ruptures.

The most recent events occurred at Palisades and San Onofre 2 (see Appendix C SGWH Event Items 8 and 9, respectively) during testing. The event at Palisades occurred during a test that, if successful, might have allowed an administrative increase of auxiliary feedwater flow rate during

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plant startup or during shutdown recoveries. The event at San Onofre was caused by a very high auxiliary feedwater flow rate (approximately 1200 gpm) to a steam generator after the water level had been below the feedring for about two hours. Even though the feedring was equipped with "J" tubes to reduce the drainage flow rate from the feedring and adjacent feedwater piping, the two-hour no-flow period was excessive and complete voiding may have occurred, resulting in SGWH and feedring collapse. Rapid steam condensation caused the pressure to drop inside the feedring, and the pressure in the steam generator caused it to collapse.

No event has been recorded for a Zion steam generator equipped with feedrings having "J"-shaped top discharge tubes. The modification of all feedrings to the "J" tube configuration at the Zion plants is expected to be completed by February 1982 (see SGWH Event Item 27).

Only one SGWH occurrence (San Onofre 2, Event Item 9 of Appendix C) has been recorded for plants that are in conformance with the requirements set forth in the NRC's Branch Technical Position (BTP) ASB 10.2, "Design Guidelines for Water Hammers in Steam Generators With Top Discharge Feedring Designs." This BTP is attached to Standard Review Plan (SRP) 10.4.7, "Condensate and Feedwater System," of NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." In this instance, the feedring had been uncovered for an extended period of time, resulting in water being replaced by steam in the feedring even though a top discharge geometry was employed.

No SGWH has been reported at plants in conformance with BTP ASB 10.2, and in which auxiliary feedwater flow was started shortly after the interruption of main feedwater flow.

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TABLE 8. TYPES OF WATER HAMMER EVENTS (NON-STEAM-GENERATOR) OCCURRING IN PWR PLANT SYSTEMS (CATEGORY I)

Systems of Occurrence	Number of Events	Subsystem	Number and Type of Water Hammer	Event Number (Appendix B)
Main reactor (30 events)	5	RCS (pressurizer)	5 Flow-into-voided-line	1,10,15,22, 33
	4	Condenser	3 Flow-into-voided-line 1 Steam hammer	2,4,5 14
	6	Main steam	5 Steam hammer 1 Unknown	3,17,18,21, 35 16
	15	Main feedwater	1 Steam-bubble collapse 1 Steam-water entrainment 11 Steam hammer 2 Unknown	24 37 6,7,8,11, 12,13,30,31, 38,39,40 32,36
Auxiliary reactor (2 events)	--	Steam generator blowdown		
	--	Auxiliary feedwater turbine		
	2	CVCS	1 Flow-into-voided-line 1 Steam hammer	27 19
Reactor safety (8 events)	4	ECCS	3 Flow-into-voided-line 1 Steam-bubble collapse	23,26,29 34
	1	RHR	1 Steam hammer	25
	--	Containment spray		
	3	Service cooling water	1 Flow-into-voided-line 2 Steam hammer	9 20,28
Total events	40			

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TABLE 9. NON-WATER-HAMMER EVENTS (NON-STEAM-GENERATOR) OCCURRING IN PWR PLANT SYSTEMS (CATEGORY II)

Systems of Occurrence	Number of Events	Subsystem	Number and Type of Event	Event Number (Appendix B)
Main reactor (0 events)	--	RCS		
	--	Condenser		
	--	Main steam		
	--	Main feed-water		
Auxiliary reactor (2 events)	1	Steam generator blowdown	1 Potential for steam-water entrainment	42
	--	Auxiliary feedwater turbine		
	1	CVCS	1 Possible flow-into-partially-voided-line	43
Reactor safety (2 events)	--	ECCS		
	--	RHR		
	1	Containment spray	1 Component malfunction	44
	1	Service cooling water	1 Component malfunction	41
Total events	4			

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Table 10 a

TABLE 10a. PWR CATEGORY I EVENT BREAKDOWN BY PLANT

Plant	Commercial Operation Date	Years of Commercial Operation by NSSS Vendor From 1/1/69 to 5/1/81: W	CE	B&W	Number of Events by Year ( ) - Class 1 [ ] - Class 2 Class - 3: 69	70	71	72	73	74	75	76	77	78	79	80	81	Event Summaries: Category I #	%	Class 1 #	%	Class 2 #	%	Class 3 #	%	Class 3 Events Per Year*
Arkansas 1	12/19/74	--	--	6.37	--	--	--	--	--	(1)	[1]	1	--	--	--	--	--	3	7.5	1	9.1	1	16.7	1	4.3	.19
Arkansas 2	3/26/80	--	1.10	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Beaver Valley 1	4/30/77	4.00	--	--	--	--	--	--	--	--	--	(4)	[1]	--	--	1	--	6	15.0	5	45.5	--	--	1	4.3	.33
Calvert Cliffs 1	5/8/75	--	5.02	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Calvert Cliffs 2	4/1/77	--	4.08	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Cook 1	8/27/75	5.68	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Cook 2	7/1/78	2.83	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Crystal River 3	3/13/77	--	--	4.13	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Davis-Besse 1	7/31/78	--	--	2.75	--	--	--	--	--	--	--	--	(1)	--	--	--	--	1	2.5	1	9.1	--	--	--	--	--
Farley 1	12/1/77	3.42	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Fort Calhoun 1	6/20/74	--	6.86	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Ginna	3/70	11.17	--	--	--	--	1	--	1	--	1	--	--	--	--	--	--	3	7.5	--	--	--	--	3	13.0	.29
Haddam Neck	1/1/68	12.33	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	NA	--	NA	--	--	--	--
Indian Point 1	62-63	12.00	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	1	2.5	NA	--	NA	--	1	4.3	.08
Indian Point 2	8/73	7.75	--	--	--	--	--	--	--	[1]	--	--	--	--	--	1	--	2	5.0	--	--	1	16.7	1	4.3	.15
Indian Point 3	8/30/76	4.67	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Kewaunee	6/74	6.92	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Maine Yankee	12/28/72	--	8.34	--	--	--	--	(1)	[1]	--	--	--	1	--	--	--	--	3	7.5	1	9.1	1	16.7	1	4.3	.14
Millstone 2	12/26/75	--	5.35	--	--	--	--	--	--	--	(1)	--	--	--	--	--	--	1	2.5	1	9.1	--	--	--	--	--
North Anna 1	6/6/78	2.90	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
North Anna 2	12/14/80	.38	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Oconee 1	7/15/73	--	--	7.79	--	--	--	--	--	--	--	--	--	--	1	--	--	1	2.5	--	--	--	--	1	4.3	.15
Oconee 2	9/9/74	--	--	6.64	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Oconee 3	12/16/74	--	--	6.38	--	--	--	--	--	--	[1]	--	--	--	--	--	--	1	2.5	--	--	1	16.7	--	--	--
Palisades	12/31/71	--	9.34	--	--	--	--	--	--	1	--	--	--	--	--	--	--	1	2.5	--	--	--	--	1	4.3	.12
Point Beach 1	12/21/70	10.36	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Point Beach 2	10/1/72	8.58	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Prairie Island 1	12/16/73	7.38	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Prairie Island 2	12/21/74	6.36	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--

TABLE 10a. (continued)

Plant	Commercial Operation Date	Years of Commercial Operation by NSSS Vendor From 1/1/69 to 5/1/81: W	CE	B&W	Number of Events by Year ( ) - Class 1 [ ] - Class 2 Class - 3: 69	70	71	72	73	74	75	76	77	78	79	80	81	Event Summaries: Category 1 #	Category 1 %	Class 1 #	Class 1 %	Class 2 #	Class 2 %	Class 3 #	Class 3 %	Class 3 Events Per Year[a]
Rancho Seco	4/17/75	--	--	6.04	--	--	--	--	--	(1)	--	--	--	1	--	--	--	2	5.0	1	9.1	--	--	1	4.3	.20
Robinson 2	3/7/71	10.15	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	1	2.5	--	--	--	--	1	4.3	.11
Salem 1	6/30/77	3.83	--	--	--	--	--	--	--	--	--	--	(1)	--	--	--	--	1	2.5	1	9.1	--	--	--	--	--
San Onofre 1	1/1/68	12.33	--	--	1	--	--	--	1	1	--	--	--	--	1	1	--	5	12.5	NA	--	NA	--	5	21.7	.41
St. Lucie 1	12/21/76	--	4.36	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Surry 1	12/22/72	8.36	--	--	--	--	--	--	[1]	--	--	--	--	--	--	--	--	1	2.5	--	--	1	16.7	--	--	--
Surry 2	5/1/73	8.00	--	--	--	--	--	--	--	q^e	--	--	--	--	--	--	--	1	2.5	--	--	--	--	1	4.3	.14
TMI 1	9/2/74	--	--	4.57	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
TMI 2	12/30/78	--	--	0.24	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Trojan 1	5/20/76	4.95	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Turkey Point 3	12/14/72	8.38	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	1	2.5	--	--	--	--	1	4.3	.14
Turkey Point 4	9/7/73	7.65	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	1	2.5	--	--	--	--	1	4.3	.15
Yankee Rowe	7/61	12.33	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	NA	--	NA	--	--	--	--
Zion 1	12/31/73	7.34	--	--	--	--	--	--	--	[1]	--	2	--	--	--	--	--	3	7.5	--	--	1	16.7	2	8.7	.32
Zion 2	9/17/74	6.62	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	1	2.5	--	--	--	--	1	4.3	.18

a. This column lists the number of Class 3 events divided by the amount of time a plant is in the Class 3 period.

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TABLE 10b. PWR CATEGORY I EVENT BREAKDOWN BY GROUPS OF PLANTS

Plant	Number of Plants	Years of Commercial Operation by NSSS Vendor From 1/1/69 to 5/1/81: W	CE	B&W	Number of Events by Year: 69	70	71	72	73	74	75	76	77	78	79	80	81	Event Summaries: Category I #	%	Class 1 #	%	Class 2 #	%	Class 3 #	%	Class 3 Events Per Year[a]
All plants	44	196.67	44.45	44.91	1	1	1	1	4	7	5	8	4	2	3	3	--	40	100	--	--	--	--	23	100	.09
All plants Reporting Class 1	7	--	--	--	--	--	--	1	--	2	1	4	3	--	--	--	--	--	--	11	100	--	--	--	--	--
All plants Reporting Class 2	6	--	--	--	--	--	--	--	2	2	2	--	--	--	--	--	--	--	--	--	--	6	100	--	--	--
All plants Reporting Class 3	16	95.39	17.68	20.20	1	1	1	--	2	3	2	4	1	2	3	3	--	--	--	--	--	--	--	23	100	.19
All W plants	27	196.67	--	--	1	1	1	1	3	4	2	7	2	1	2	3	--	27	67.5	--	--	--	--	18	78.3	.10
W plants Reporting Class 1	2	--	--	--	--	--	--	--	--	--	--	4	2	--	--	--	--	--	--	6	54.5	--	--	--	--	--
W plants Reporting Class 2	3	--	--	--	--	--	--	--	1	2	--	--	--	--	--	--	--	--	--	--	--	3	50.0	--	--	--
W plants Reporting Class 3	11	95.39	--	--	1	1	1	--	2	2	2	3	--	1	2	3	--	--	--	--	--	--	--	18	78.3	.21
All CE plants	8	--	44.45	--	--	--	--	1	1	1	1	--	1	--	--	--	--	5	12.5	--	--	--	--	2	8.7	.05
CE plants Reporting Class 1	2	--	--	--	--	--	--	1	--	--	1	--	--	--	--	--	--	--	--	2	18.2	--	--	--	--	--
CE plants Reporting Class 2	1	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	1	16.7	--	--	--
CE plants Reporting Class 3	2	--	17.68	--	--	--	--	--	--	1	--	--	1	--	--	--	--	--	--	--	--	--	--	2	8.7	.13
All B&W plants	9	--	--	44.91	--	--	--	--	--	2	2	1	1	1	1	--	--	8	20.0	--	--	--	--	3	13.0	.08
B&W plants Reporting Class 1	3	--	--	--	--	--	--	--	--	2	--	--	1	--	--	--	--	--	--	3	27.3	--	--	--	--	--
B&W plants Reporting Class 2	2	--	--	--	--	--	--	--	--	--	2	--	--	--	--	--	--	--	--	--	--	2	33.3	--	--	--
B&W plants Reporting Class 3	3	--	--	20.20	--	--	--	--	--	--	--	1	--	1	1	--	--	--	--	--	--	--	--	3	13.0	.17

a. This column lists the number of Class 3 events divided by the amount of time a plant is in the Class 3 period.

TABLE 11a. PWR CATEGORY I EVENTS PER SYSTEM[a] BY EVENT CLASS

		Events Per System by Event Class																							
		Class 1								Class 2								Class 3							
Plant	NSSS Vendor	1	2	3	4	5	6	7	8	1	2	3	4	5	6	7	8	1	2	3	4	5	6	7	8
Arkansas 1	B&W	1	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	1	--	--	--	--	--
Beaver Valley 1	W	--	2	--	3	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1
Davis-Besse 1	B&W	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Ginna	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	3	--	--	--	--
Indian Point 1	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--
Indian Point 2	W	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--
Maine Yankee	CE	--	--	1	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	1	--	--	--
Milestone 2	CE	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Oconee 1	B&W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--
Oconee 3	B&W	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Palisades	CE	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--
Rancho Seco	B&W	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--
Robinson 2	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--
Salem 1	W	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
San Onofre 1	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	3	--	1	--	1
Surry 1	W	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Surry 2	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--
Turkey Point 3	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--
Turkey Point 4	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--
Zion 1	W	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--	--	--	--	2	--	--	--	--
Zion 2	W	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	1	--	--	--	--

a. LEGEND for systems:

Main	Auxiliary	Safety
1 = RCS	5 = CVCS	6 = ECCS
2 = Condenser		7 = RHR
3 = Main steam		8 = SCW
4 = Main feedwater		

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Table 11b

TABLE 11b. SUMMARY OF PWR CATEGORY I EVENTS BY VENDOR AND EVENT CLASS

NSSS Vendor	Events Per System by Event Class																							
	Class 1								Class 2								Class 3							
	1	2	3	4	5	6	7	8	1	2	3	4	5	6	7	8	1	2	3	4	5	6	7	8
W	--	2	--	3	1	--	--	--	2	--	--	1	--	--	--	--	--	1	2	10	--	3	--	2
CE	--	--	1	--	--	--	--	1	--	--	1	--	--	--	--	--	--	--	--	--	1	1	--	--
B&W	2	--	--	1	--	--	--	--	1	1	--	--	--	--	--	--	--	--	2	--	--	--	1	--
Total	2	2	1	4	1	--	--	1	3	1	1	1	--	--	--	--	--	1	4	10	1	4	1	2

TABLE 11c. SUMMARY OF PWR CATEGORY I EVENTS BY VENDOR AND SYSTEM

NSSS Vendor	Events Per System															
	1		2		3		4		5		6		7		8	
	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)	(#)	(%)
W	(2)	40	(3)	75	(2)	33	(14)	93	(1)	50	(3)	75	--		(2)	67
CE	--		--		(2)	33	--		(1)	50	(1)	25	--		(1)	33
B&W	(3)	60	(1)	25	(2)	33	(1)	7	--		--		(1)	100	--	
Total	(5)	100	(4)	100	(6)	100	(15)	100	(2)	100	(4)	100	(1)	100	(3)	100

delete ( ) around # and numbers under that column

keep ( ) ~~around % and add to numbers~~ under that column

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TABLE 11d. SUMMARY OF PWR CATEGORY I EVENTS BY VENDOR AND MAJOR SYSTEMS

NSSS Vendor	Main Systems (#)	Main Systems (%)	Auxiliary Systems (#)	Auxiliary Systems (%)	Safety Systems (#)	Safety Systems (%)
	Events Per Major Systems					
W	(21)	70	(1)	50	(5)	63
CE	(2)	7	(1)	50	(2)	25
B&W	(7)	23	--	--	(1)	13
Total	(30)	100	(2)	100	(8)	100

delete ( ) around # and numbers under that column

Keep ( ) around % and add to numbers under that column

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TABLE 12. PWR CATEGORY I EVENTS BY MAJOR SYSTEMS AND EVENT CLASS

	Events by Event Class											
Major Systems[a]	All Classes (#)	(%)	Class 1 (#)	(%)	Class 2 (#)	(%)	Class 3 (#)	(%)	Class 3 (Pre-78) (#)	(%)	Class 3 (78 to Present) (#)	(%)
Main reactor	(30)	75	(9)	82	(6)	100	(15)	65	(10)	67	(5)	63
Auxiliary reactor	(2)	5	(1)	9	(0)	0	(1)	4	(1)	7	(0)	0
Reactor safety	(8)	20	(1)	9	(0)	0	(7)	30	(4)	27	(3)	38

a. LEGEND for systems:

Main reactor systems: RCS Condenser, Main steam, Main feedwater
Auxiliary reactor systems: CVCS
Reactor safety systems: ECCS, RHR, SCW.

delete ( ) around # and numbers in that column
keep ( ) around % and add to numbers in that column

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TABLE 13a. PWR CATEGORY I EVENTS BY PLANT, BASIS/CAUSE AND DAMAGE

Event	Plant	NSSS Vendor	Plant System	Basis/Cause for Event: Class 1	Basis/Cause for Event: Class 2	Basis/Cause for Event: Class 3	Items Damaged: Class 1	Items Damaged: Class 2	Items Damaged: Class 3
1	Arkansas 1	B&W	RCS	Design	--	--	Supports	--	--
2	Arkansas 1	B&W	Condenser	--	Design	--	--	Condenser	--
3	Arkansas 1	B&W	Main steam	--	--	Procedure	--	--	Not indicated
4	Beaver Valley 1	W	Condenser	Design	--	--	Tubing	--	--
5	Beaver Valley 1	W	Condenser	Design	--	--	Tubing	--	--
6	Beaver Valley 1	W	Main feedwater	Design	--	--	Snubbers/Valves/ instr/insul/fittings	--	--
7	Beaver Valley 1	W	Main feedwater	Design	--	--	Supports/instr. lines	--	--
8	Beaver Valley 1	W	Main feedwater	Design	--	--	Supports/drain line/ valves	--	--
9	Beaver Valley 1	W	SCW	--	--	Procedure	--	--	Embedment plate
10	Davis-Besse 1	B&W	RCS	Design	--	--	Not indicated	--	--
11	Ginna	W	Main feedwater	--	--	Design/ procedure	--	--	Valve
12	Ginna	W	Main feedwater	--	--	QA/Manuf.	--	--	Supports/insul.
13	Ginna	W	Main feedwater	--	--	Design	--	--	Tubing/insul.
14	Indian Pt. 1	W	Condenser	--	--	Design/QA	--	--	Piping/valve
15	Indian Pt. 2	W	RCS	--	Design	--	--	Rupture disc/ grout/radio- activity rise in contain- ment snubber/ support/drain line	--
16	Indian Pt. 2	W	Main steam	--	--	Design/ procedure	--	--	Snubber/support/ drain line
17	Maine Yankee	CE	Main steam	Operator	--	--	Not indicated	--	--
18	Maine Yankee	CE	Main steam	--	Design	--	[illegible]	Valves	--
19	Maine Yankee	CE	CVCS	--	--	QA	[illegible]	--	Valve/drain line

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TABLE 13a. (continued)

Event	Plant	NSSS Vendor	Plant System	Basis/Cause for Event: Class 1	Basis/Cause for Event: Class 2	Basis/Cause for Event: Class 3	Items Damaged: Class 1	Items Damaged: Class 2	Items Damaged: Class 3
20	Millstone 2	CE	SCW	Design	--	--	HX nozzle cracked	--	--
21	Oconee 1	B&W	Main steam	--	--	Design/install	--	--	Snubbers
22	Oconee 3	B&W	RCS	--	Maintenance	--	--	Rupture disc/insul/release of coolant to sump	--
23	Palisades	CE	ECCS	--	--	Procedure	--	--	Support
24	Rancho Seco	B&W	Main feedwater	Design	--	--	Support	--	--
25	Rancho Seco	B&W	RHR	--	--	Procedure	--	--	Supports
26	Robinson	W	ECCS	--	--	Design/procedure	--	--	Supports
27	Salem	W	CVCS	Design	--	--	Pipe break	--	--
28	San Onofre 1	W	SCW	--	--	Design/QA	--	--	Tsunami gate
29	San Onofre 1	W	ECCS	--	--	Design/procedure	--	--	Support/valve bolts
30	San Onofre 1	W	Main feedwater	--	--	Design/QA	--	--	Snubber/supports
31	San Onofre 1	W	Main feedwater	--	--	Maintenance	--	--	Snubber
32	San Onofre 1	W	Main feedwater	--	--	Design	--	--	Supports
33	Surry 1	W	RCS	--	Design	--	--	Snubber	--
34	Surry 2	W	ECCS	--	--	Procedure	--	--	Support
35	Turkey Pt. 3	W	Main steam	--	--	Design/Proced	--	--	Support
36	Turkey Pt. 4	W	Main feedwater	--	--	Design	--	--	Snubber
37	Zion 1	W	Main feedwater	--	Design/procedure	--	--	Support	--
38	Zion 1	W	Main feedwater	--	--	Design	--	--	No damage
39	Zion 1	W	Main feedwater	--	--	Procedure	--	--	No damage
40	Zion 2	W	Main feedwater	--	--	Design	--	--	No damage

TABLE 13b. PWR CATEGORY I EVENT TOTALS BY BASIS/CAUSE AND EVENT CLASS

	Events by Event Class					
	Class 1		Class 2		Class 3	
Basis/Cause	(#)	(%)	(#)	(%)	(#)	(%)
Design	(10)	91	(4)	57	(5)	23
Design/QA	--	--	--	--	(3)	14
Design/install	--	--	--	--	(1)	5
Design/procedure	--	--	(1)	14	(5)	23
Procedure	--	--	--	--	(6)	23
QA/Manuf, QA, maint	--	--	(1)	14	(3)	14
Operator	(1)	9	--	--	--	--
TOTAL	(11)	100	(6)	100	(23)	100
Design related[a]	(10)	91	(5)	83	(14)	61
Procedure related[a]	--	--	(1)	17	(11)	48

a. Some events are both design and procedure related.

Delete ( ) around # and numbers in that column

keep ( ) around % and add to numbers in that column

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TABLE 13c. PWR CATEGORY I EVENT TOTALS BY DAMAGE AND EVENT CLASS

Items Damaged	Events by Event Class: Class 1 (#)	Class 1 (%)	Class 2 (#)	Class 2 (%)	Class 3 (#)	Class 3 (%)
Snubbers, supports snubber/supports	(2)	18	(2)	43	(10)	41
Snubbers/supports/ piping related	(1)	9	--	--	(1)	5
Snubbers/supports/misc	(1)	9	--	--	(1)	5
Supports/valves	(1)	9	--	--	(1)	5
Valve	--	--	(1,	14	(1)	5
Piping and related	(4)	36	(1)	14	(4)	18
Coolant release/rupture disc	--	--	(2)	29	--	--
Insulation, grout, plate	--	--	--	--	(1)	5
Unknown, no damage, not indicated	(2)	18	--	--	(4)	18
TOTAL	(11)	100	(6)	100	(23)	100
Snubber, support related[a]	(5)	45	(2)	33	(13)	57
Piping related[a]	(5)	45	(3)	50	(5)	22

a. Some events are snubber, support, and piping related.

Delete ( ) around # and numbers in that column
Keep ( ) around % and all around numbers in that column

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TABLE 13d. PWR CATEGORY I EVENTS BY BASIS/CAUSE AND VENDOR

	Events by NSSS Vendor					
	W		CE		B&W	
Basis/Cause	(#)	(%)	(#)	(%)	(#)	(%)
Design	(13)	48	(2)	40	(4)	50
Design/QA	(3)	11	--	--	--	--
Design/install	--	--	--	--	(1)	13
Design/procedure	(6)	22	--	--	--	--
Procedure	(3)	11	(1)	20	(2)	25
QA/manuf, QA, maint	(2)	7	(1)	20	(1)	13
Operator	--	--	(1)	20	--	--
TOTAL	(27)	100	(5)	100	(8)	100
Design related[a]	(22)	81	(2)	40	(5)	63
Procedure related[a]	(9)	33	(1)	20	(2)	25

a. Some events are both design and procedure related.

delete ( ) around # and numbers in that column
keep ( ) around % and all the numbers in that column

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TABLE 13e. PWR CATEGORY I EVENTS BY DAMAGE AND VENDOR

	Events by NSSS Vendor					
	W		CE		B&W	
Items Damaged	(#)	(%)	(#)	(%)	(#)	(%)
Snubbers, supports snubber/supports	(9)	33	(1)	20	(4)	50
Snubbers/supports/ piping related	(2)	7	--	--	--	--
Snubbers/supports/misc	(2)	7	--	--	--	--
Supports/valves	(2)	7	--	--	--	--
Valve	(1)	4	(1)	20	--	--
Piping and related	(6)	22	(2)	40	(1)	13
Coolant release/rupture disc	(1)	4	--	--	(1)	13
Insulation, grout, plate	(1)	4	--	--	--	--
Unknown, no damage, not indicated	(3)	11	(1)	20	(2)	25
TOTAL	(27)	100	(5)	100	(8)	100
Snubber, support related[a]	(15)	56	(1)	20	(4)	50
Piping related[a]	(9)	33	(2)	40	(2)	25

a. Some events are snubber, support, and piping related.

Delete ( ) around # and numbers in that column
keep ( ) around % and add to numbers in that column

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TABLE 14. BWR CATEGORY I NORMALIZED INCIDENT RATES

	Class 1			Class 2			Class 3		
Event Year	Plant Years[a]	Number of Events	Incident Rate	Plant Years[a]	Number of Events	Incident Rate	Plant Years[a]	Number of Events	Incident Rate
69	.86	--	--	N/A	N/A	N/A	4.0	1	.25
70	2.04	--	--	.86	--	--	4.0	1	.25
71	1.52	--	--	1.96	--	--	4.86	1	.21
72	5.58	1	.18	1.52	--	--	6.82	--	--
73	6.25	--	--	5.58	2	.36	8.33	2	.24
74	7.58	2	.26	6.25	2	.32	13.90	3	.22
75	3.28	1	.30	6.66	2	.30	19.98	2	.10
76	4.30	4	.93	3.28	--	--	26.71	4	.15
77	3.69	3	.81	4.84	--	--	29.00	1	.03
78	2.52	--	--	3.70	--	--	33.81	2	.06
79	.81	--	--	2.52	--	--	36.73	3	.08
80	1.19	--	--	.81	--	--	39.00	3	.08

a. Plant years are the total amount of time that all plants are within the time period for a given event class (1, 2, or 3) for a given event year.

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Table 14.

Figure 4

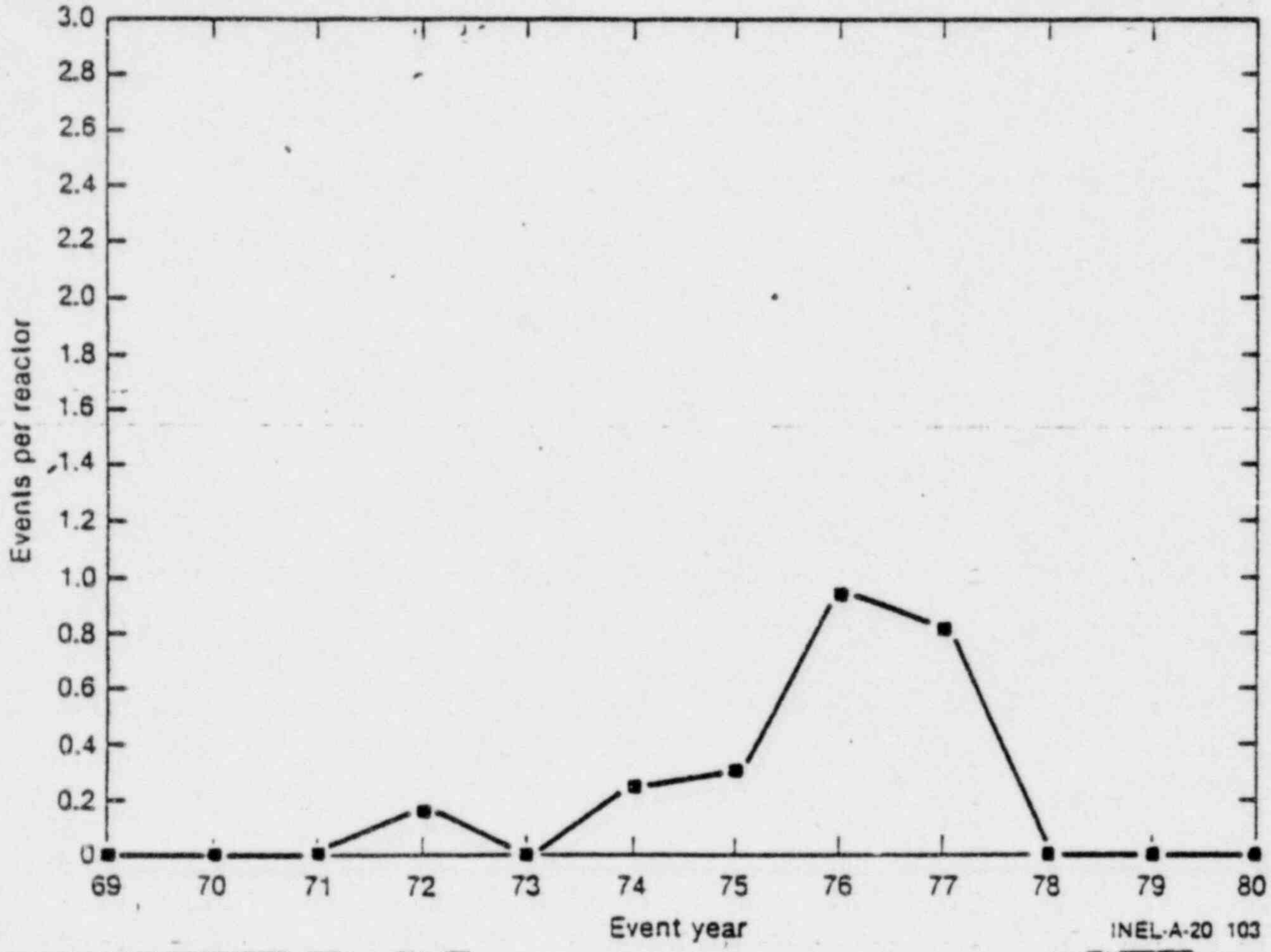

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Figure 5

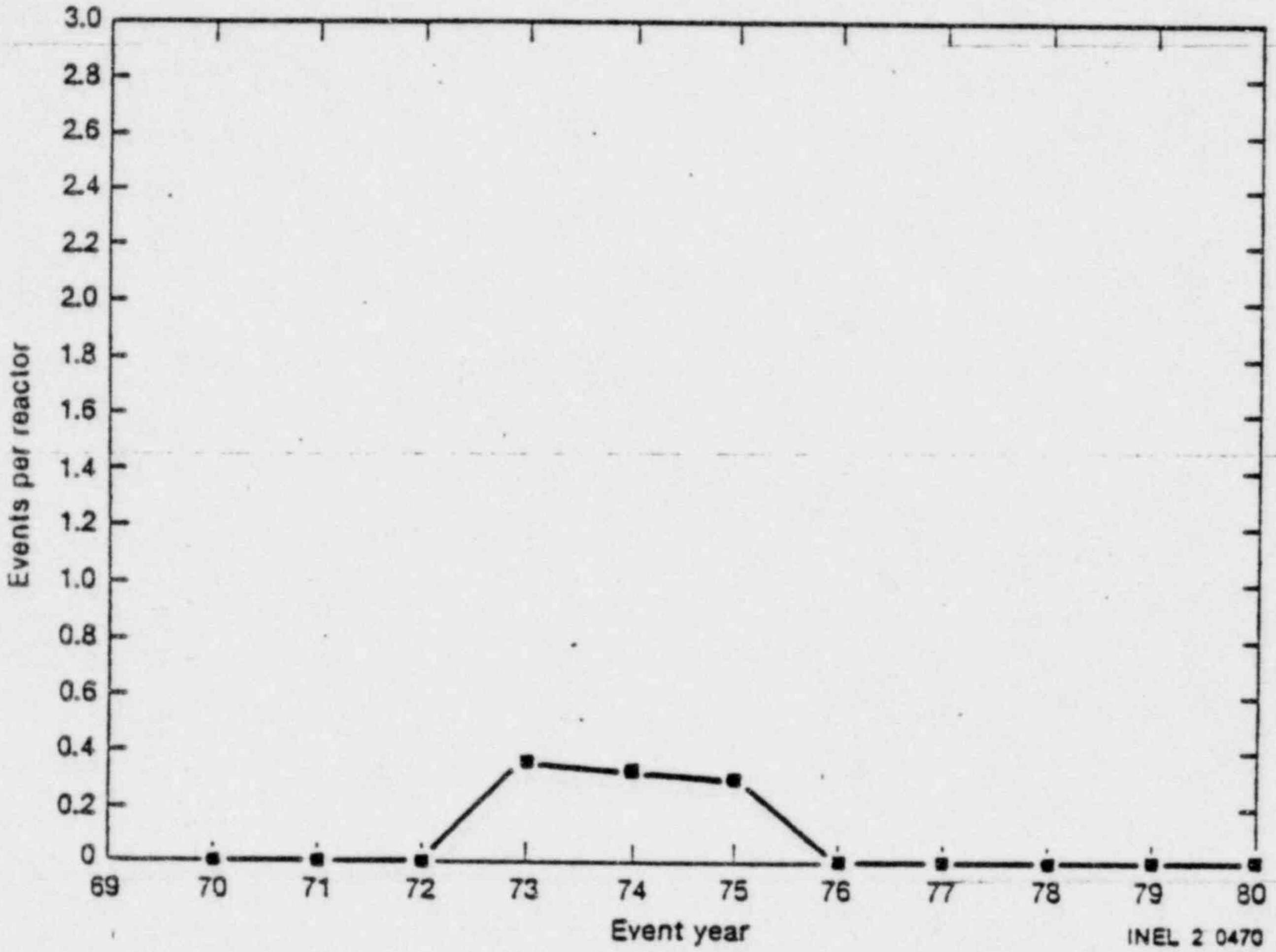

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Figure 6

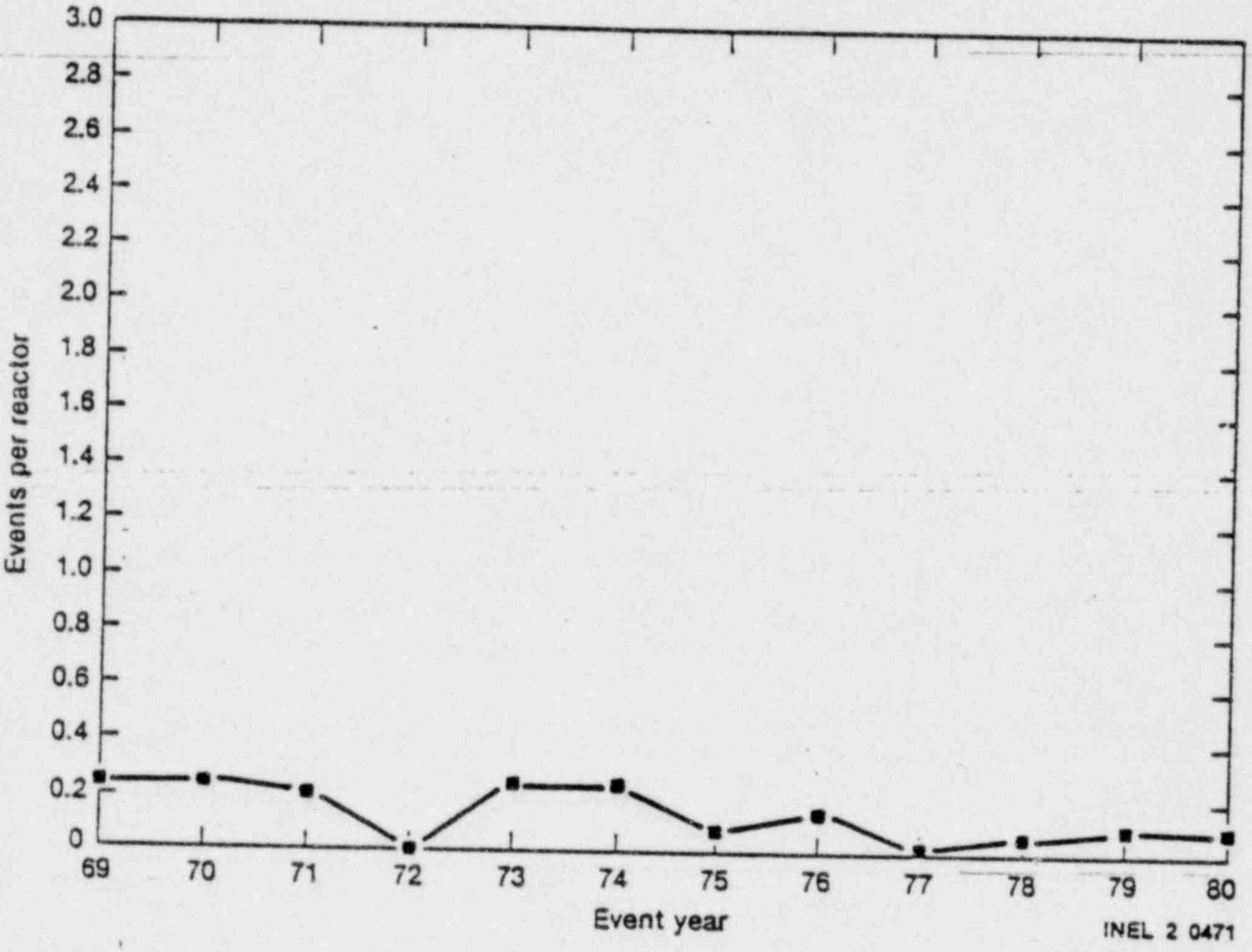

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TABLE 15. PWR STEAM GENERATOR EVENT BREAKDOWN BY PLANT

Plant	Commercial Operation Date	NSSS Vendor	71 or Before	72	73	74	75	76	77	78	79	80	81
				Number of Events by Year: ( )-Class 1			[ ]-Class 2			-Class 3			
Calvert Cliffs 1	5/8/75	CE	--	--	--	--	1	--	--	--	--	--	--
Calvert Cliffs 2	4/1/77	CE	--	--	--	--	--	(1)	--	--	--	--	--
Cook 1	8/27/75	W	--	--	--	--	[1]	--	1	--	--	--	--
Indian Point 2	8/73	W	--	--	[1]	[1]	--	--	--	--	--	--	--
Millstone 2	12/26/75	CE	--	--	--	--	(1)	--	--	--	--	--	--
Palisades	12/31/71	CE	--	--	--	--	--	--	--	--	--	--	1
San Onofre 2	N/A	W	--	--	--	--	--	--	--	--	--	--	(1)
Surry 1	12/22/72	W	--	(1)	--	--	--	--	--	--	--	--	--
Turkey Point 3	12/14/72	W	--	--	[1]	--	--	--	--	--	--	--	--
Turkey Point 4	9/7/73	W	--	--	(1)	[1]	--	--	--	--	--	--	--
Yankee Rowe	7/61	W	1	--	--	--	--	--	--	--	--	--	--
Zion 1	12/31/73	W	--	--	--	--	--	1	2	2	3	--	--
Zion 2	9/17/74	W	--	--	--	(1) [1]	--	2	--	--	--	1	--

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SUMMARIZED FINDINGS OF THE STUDY

In this section we summarize the findings from our compilation of known and suspected water hammer occurrences in BWR and PWR plants. This summary groups:

- The numbers of events and plants
- The systems involved
- The basis/cause for the events
- The damage incurred
- The types of water hammer observed.

Results of the steam generator events are presented in the last part of this section.

The data base for this study is quite limited. However, valid conclusions may be obtained if the limits of the data base are observed.

Water Hammer Events

There are approximately 70% more PWR plants (44) than BWR plants (26) involved in this study, but the BWR plants had approximately twice as many events (81) as the PWR plants (40). When the 27 steam generator events are added, the PWR total is still less than the BWR total.

A small number of plants account for a large number of the events. This is true for both BWR and PWR plants.

The numbers of events for each type of plant (GE-3, GE-4, W, CE, and B&W) are in approximately the same proportion as the number of plants for each type.

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Approximately half of the events occurred prior to and within one year after the commercial operation dates of the plants--an average period of two years for each plant.

After the first year of commercial operation, the average incident rate (events per year per reactor) is low for both BWRs (0.22) and PWRs (0.09). The yearly normalized rate, though changing slightly from year to year, remains low.

Systems Affected

Approximately three fourths of the BWR events occurred in safety-related systems. One third of the events occurred in the RHR. In a BWR plant, the RHR has many functions, not all of which are directly safety related, but which are connected to safety systems. One fourth of the events occurred in the HPCI.

Three fourths of the PWR events occurred in main reactor systems. Approximately one third of the events occurred in main feedwater systems. Thirteen of the 15 main feedwater events occurred in Westinghouse plants.

Basis/Cause for Events

Basis/cause denotes an attempt to identify the basic reason for an occurrence. Some events are both design and procedure related.

PWR plants have attributed basis/cause more to design-related problems than have BWR plants. For both types of plants, the design-related basis/cause impact is considerably less for Class 3 events than for Class 1 events.

BWR plants have attributed the basis/cause more to procedure-related problems than have PWR plants. For BWR plants, the impact of procedure-related problems has been considerable for all three classes of events. For PWR plants, procedure-related problems ranged from zero of 10 events for Class 1 to 11 of 23 events for Class 3.

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For PWR plants, the events were attributed to a design-related basis/cause more often in W plants (22 of 27 for 81%) than in CE plants (2 of 5 for 40%) or in B&W plants (5 of 8 for 63%). For BWR plants, events were attributed to a design-related basis/cause more often in GE-3 plants (16 of 29 for 55%) than in GE-4 plants (15 of 48 for 31%).

For PWR plants, events were attributed to a procedure-related basis/cause more often in W plants (9 of 27 for 33%) than in CE plants (1 of 5 for 20%) or in B&W plants (2 of 8 for 25%). For BWR plants, events were attributed to a procedure-related basis/cause more often in GE-4 plants (28 of 48 for 58%) than in GE-3 plants (13 of 29 for 45%).

Damage Incurred

For both BWRs and PWRs, half of the event reports indicated that damage occurred in the piping support systems.

For both BWRs and PWRs, a third of the event reports indicated damage to piping related systems. These involved welds, junctions, pumps, and other system components not including valves. These systems include those portions of the piping that could allow coolant leakage if sufficiently damaged.

Types of Water Hammer

The four types of water hammer identified (in Reference 4) as initiating mechanisms are:

- Water column separation in liquid flows
- Flow-into-voided-line
- Steam-bubble collapse or mixing of subcooled water and steam from interconnected systems
- Water entrainment in steam lines.

Those not identifiable, or considered to be a result of wave propagation/reflection are categorized as unknown or as steam hammer.

Excluding PWR steam generator events, 13 of the 15 steam-bubble collapse events occurred in BWR plants. Seven occurred in the RHR, the majority involving leakage, filling, and venting problems. Four occurred in the HPCI, involving valve closures in the exhaust line.

Thirteen flow-into-voided-line events occurred in PWR plants. Twenty-nine occurred in BWR plants. All events involved systems not continuously in use and involved line voids caused mostly by leakage and by lack of good filling and venting measures. All but one of the BWR events occurred in reactor safety systems.

Twenty of the 21 steam-water entrainment events occurred in the BWR plants. Twelve of these occurred in the HPCI and involved water accummulation without drainage because of defective equipment, or occurred when the HPCI was being warmed up from an out-of-service condition. The majority of the remaining events involved water accumulation without proper drainage.

Five water column separation events were identified, occurring only in the BWR plants. Three occurred in the SCW, involving the cooling tower, the salt water pump, and a valve closure. Two events, one in the feedwater and one in the condenser system, were due to inadvertent valve actuation.

Twenty-four events in PWR plants were listed as steam hammer/unknown events. Fourteen were so listed in BWR plants.

PWR Steam Generator Events

The majority (23 of 27 for 85%) of the events occurred in Westinghouse plants, with 13 (48%) occurring at Zion 1 and 2.

Approximately half of the events (12 of 27 for 44%) occurred prior to and within one year of commercial operation.

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No events occurred in plants which were in conformance with BTP ASB 10.2 when auxiliary feedwater was started shortly after the interruption of main feedwater flow.

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reference are missing

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22-141 50 SHEETS
22-142 100 SHEETS
22-144 200 SHEETS

APPENDIX A

BWR EVENTS

PRELIMINARY

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Category I: Reported BWR Water Hammer Events

BWR EVENT ITEM 1

(a) PLANT, MODE, EVENT DATE	Big Rock Pt., valve lineup prior to startup, 10/31/77.
(b) REFERENCE	50155-810 RO-77-44 (11/9/77).
(c) SYSTEM, MECHANICAL FUNCTION	Plant Heating Boiler (steam supply line), valve opening.
(d) INITIAL INDICATION	Water hammer occurrence.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. No apparent physical damage. Event resulted in a minor, uncontrolled release of radioactive water to discharge canal.
(f) CAUSE AND BASIS FOR EVENT	Marginal design concepts employed and procedural deficiency. During manual valving operations, RCS water backflowed into the plant heating system external to the containment, causing a water hammer in the steam supply line from the heating boiler. Operating procedures were not followed in the valve lineup.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Administrative controls instituted to preclude routine system use. Plant boiler steam supply line valves to Hx were locked closed.

PRELIMINARY

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BWR EVENT ITEM 2

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 1, preoperational testing (HPCI startup test), 10/72.
(b)	REFERENCE	50259 RO-74-2 (2/8/74).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line), valve opening.
(d)	INITIAL INDICATION	Severe water hammer in HPCI turbine exhaust line.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Exhausting steam noise a problem.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Addition of check valves as vacuum breakers decreased line vibration but exhausting steam noise was a problem.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	First, one additional snubber installed to restrain exhaust line. Second, vacuum breakers (check valves) installed in exhaust line. Third, a condensing sparger was installed at end of exhaust line in suppression pool.

NOTE: First modification gave no abatement. Second change decreased line motion. Third fix arrested noise and line movement and considered adequate.

PRELIMINARY

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BWR EVENT ITEM 3

(a) PLANT, MODE, EVENT DATE	Browns Ferry 1, currently unavailable, 5/6/73.
(b) REFERENCE	50259 (7/9/73).[a]
(c) SYSTEM, MECHANICAL FUNCTION	Cooling water (emergency equipment, orifice gasket), pump start.
(d) INITIAL INDICATION	Gasket failure.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Failure of orifice gasket.
(f) CAUSE AND BASIS FOR EVENT	Design and procedural deficiencies. Event initiated on pump start following standby. Voids form and air collects at high points in the lines during standby periods. On pump start the water compresses the air voids or forces the air into solution such that the water interfaces come into contact, causing damaging water hammer. See 4, 5, 6.
(g) CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Water charging connections were provided RHR SW and EECW systems at appropriate locations to maintain full water lines without void formation at high points. See 4, 5, 6.

a. Four separate events are reported in this reference. See Events 4, 5, and 6. Cause and corrective measures were the same for all four events.

PRELIMINARY

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BWR EVENT ITEM 5

(a) PLANT, MODE, EVENT DATE	Browns Ferry 1, currently unavailable, 5/23/73.
(b) REFERENCE	50259 (7/9/73).[a]
(c) SYSTEM, MECHANICAL FUNCTION	Service water (RHR/supply line), pump start.
(d) INITIAL INDICATION	Pipe coupling failure.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Failure of pipe coupling on service water supply to 1A RHR Hx.
(f) CAUSE AND BASIS FOR EVENT	Design and procedural deficiencies. See 3.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	See 3.

a. Four separate events are reported in this reference. See Events 3, 4, and 6. Cause and corrective measures were the same for all four events.

PRELIMINARY

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BWR EVENT ITEM 6

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 1, currently unavailable, 6/5 or 6/7/73.
(b)	REFERENCE	50259 (7/9/73).[a]
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, supply line), pump start.
(d)	INITIAL INDICATION	Pipe coupling failure.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Failure of pipe coupling on service water line to 1C RHR Hx. Loosening of a relief valve. Separation of building seal from pipe. Hanger displacement on service water line.
(f)	CAUSE AND BASIS FOR EVENT	Design and procedural deficiencies. See 3.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	See 3.

a. Four separate events are reported in this reference. See Events 3, 4, and 5. Cause and corrective measures were the same for all four events.

PRELIMINARY

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BWR EVENT ITEM 7

(a) PLANT, MODE, EVENT DATE	Browns Ferry 1, power escalation testing (HPCI startup test), 10/5/73.
(b) REFERENCE	50259-184 BFAO-7325W (10/15/73).
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (turbine discharge line, rupture disc), valve opening.
(d) INITIAL INDICATION	Automatic isolation of HPCI system.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Failed rupture disc.
(f) CAUSE AND BASIS FOR EVENT	Procedural deficiency and inadequate operating procedures. Inadequate design caused potential vacuum condition. During a HPCI startup test the system was isolated automatically by failure of the inner rupture disc, due to a vacuum condition created by rapid steam condensation rather than overpressure. The turbine casing/discharge piping was not up to operating temperatures and the steam within these confines condensed.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Repairs made and design investigation for avoidance of long term recurrence.

PRELIMINARY

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BWR EVENT ITEM 8

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 1, shutdown following scram, 4/4/74.
(b)	REFERENCE	50259 BFAO-7422W (4/12/74).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (steam supply line, valve), valve opening.
(d)	INITIAL INDICATION	Water hammer.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Broken and bent vertical pipe hanger rods. Rigid wall restraints were bent/buckled. Hydraulic restraint clevis sheared off and wall plate buckled. Hanger lugs attached to pipe were deformed. A wall anchor assembly for the turbine discharge pipe was struck by the steam line and buckled. The inboard turbine journal bearing pedestal was fractured. Steam supply valve limit switch was broken.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design and marginal operating procedures. The HPCI steam supply line was isolated for maintenance (while the reactor was in operation) by closing the outboard primary containment isolation valve. The inboard valve remained open, thus exposing a portion of the line to reactor steam. The next day a scram shutdown operation. On the following day, while still shut down, the HPCI steam supply line was charged by opening the outboard isolation valve; excessive lateral and vertical movement resulted from the impact of a slug of water created by condensed steam. Contributing factors were the valve arrangement and positioning. When the outboard isolation valve was opened to warm the HPCI steam supply line, the valve opening contacts sealed-in as per design and the valve went full open. See 30, 40.

PRELIMINARY

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(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Repairs made and system tested functionally. Operating procedures implemented to minimize condensate accumulation for steam supply line valve alignment and isolation. A permanent modification to the piping design is to be made.

PRELIMINARY

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BWR EVENT ITEM 9

(a) PLANT, MODE, EVENT DATE	Browns Ferry 1, refueling outage, 1/27/80.
(b) REFERENCE	50259 LER 80-008 (2/25/80).[a]
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (steam supply line, turbine), valve opening.
(d) INITIAL INDICATION	Discovery of pedestal damage by painter.
(e) WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Crack in HPCI turbine coupling bearing support pedestal.
(f) CAUSE AND BASIS FOR EVENT	Possibly caused by an observed water hammer in the steam supply line while warming the HPCI system from an out-of-service condition.
(g) CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Pedestal replaced. Units 2 and 3 being inspected. Formal study by turbine maintenance section in progress.

a. This event and Event 10 could possibly have a common cause (the same water hammer occurrence). The reference for Event 10 does not, however, relate back to the reference for this event.

PRELIMINARY

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BWR EVENT ITEM 10

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 1, refueling outage, 1/29/80.
(b)	REFERENCE	50259 LER 80-010 (2/27/80).[a]
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (instrument lines), valve opening.
(d)	INITIAL INDICATION	Discovery of broken hangers by personnel in area.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Broken instrument sensing line hangers in HP pump room.
(f)	CAUSE AND BASIS FOR EVENT	Possibly caused by an observed water hammer while warming the HPCI system from an out-of-service condition.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Hangers repaired. Hangers in Units 2 and 3 inspected and found undamaged.

a. This event and Event 9 could possibly have a common cause (the same water hammer occurrence). The reference for this event does not, however, relate to the reference for Event 9.

PRELIMINARY

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BWR EVENT ITEM 11

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 2, 5% power (HPCI startup test), 8/11/74.
(b)	REFERENCE	50260-229 AO-745W (8/21/74).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line, rupture disc), valve opening.
(d)	INITIAL INDICATION	HPCI system isolation.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. HPCI turbine exhaust rupture disc failed resulting in HPCI isolation.
(f)	CAUSE AND BASIS FOR EVENT	Improper installation, insufficient inspection and check-out (component failure). Possible water in turbine exhaust line when steam was admitted. Exhaust drain line solenoid burned-up due to wiring error on installation. Defective switching element inside level switch deactivated drain valve.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Wiring errors corrected and defective and failed components replaced or repaired. All components and system functionally tested.

PRELIMINARY

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BWR EVENT ITEM 12

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 2, Hot Shutdown, 2/16/80.
(b)	REFERENCE	50260 LER 80-010 (3/13/80).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine coupling pedestal), valve opening.
(d)	INITIAL INDICATION	Discovery of pedestal damage during special inspection using NDT. Visual did not show.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Cracks in HPCI turbine coupling bearing support pedestal.
(f)	CAUSE AND BASIS FOR EVENT	Possibly water hammer during HPCI system warmup from out-of-service condition. Design deficiencies allow water hammer while warming HPCI system from out-of-service condition.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Pedestal replaced, Unit 3 inspected with no cracks found. Turbine maintenance section investigation in progress.

PRELIMINARY

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BWR EVENT ITEM 13

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 3, 100% power (HPCI initiation during reactor scram), 1/26/77.
(b)	REFERENCE	50296-455 RO-77-2 (2/17/77).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (pump discharge line), pump start.
(d)	INITIAL INDICATION	Discovery of pipe restraint damage.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. Restraints on the HPCI pump discharge line had loose bolts and broken anchors.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable. A contributing cause was improper installation of the line restraints.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Restraints reinstalled as designed and Browns Ferry 1 and 2 plants inspected for similar installation deficiencies.

PRELIMINARY

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BWR EVENT ITEM 14

(a)	PLANT, MODE, EVENT DATE	Brunswick 1, 96% power, 3/15/77.
(b)	REFERENCE	50325-506 LER-77-24 (4/7/77).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (Hx, steam condensing line), pump start/valve opening.
(d)	INITIAL INDICATION	Snubber damage found during operator rounds.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Broken snubber on steam condensing inlet line to 1A RHR Hx. Snubber damaged again. See 15. Same snubber damaged at Brunswick 2. See 25.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate operational and inspection procedures.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	RHR systems vented once per shift as temporary measure.

PRELIMINARY

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BWR EVENT ITEM 15

(a)	PLANT, MODE, EVENT DATE	Brunswick 1, torus cooling mode, 3/31/77.
(b)	REFERENCE	50325-506 LER 77-24 (4/7/77).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (Hx, steam condensing line), pump start.
(d)	INITIAL INDICATION	Damage to snubbers.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Broken snubber on steam condensing inlet line to 1A RHR Hx. A repeat of previous failure. See 14. Same snubber damaged at Brunswick 2. See 25.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate operational and inspection procedures. Occurred when RHR pump was started for torus cooling.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	RHR systems vented once per shift as temporary measure.

PRELIMINARY

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BWR EVENT ITEM 16

(a)	PLANT, MODE, EVENT DATE	Brunswick 1, 86% power, 11/9/77.
(b)	REFERENCE	50325 LER 77-98 (12/5/77).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (steam condensing line), pump start.
(d)	INITIAL INDICATION	Inoperative snubber found by operator.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. Broken snubber.
(f)	CAUSE AND BASIS FOR EVENT	Apparent water hammer. RHR pump operability test being run daily because HPCI inoperative.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Operators cautioned about insuring the system is filled and rented prior to pump start. Part on order to repair leaking valves.

PRELIMINARY

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BWR EVENT ITEM 17

(a) PLANT, MODE, EVENT DATE	Brunswick 1, 91% power, 12/20/77.
(b) REFERENCE	50325-672 LER 77-118 (1/20/78).
(c) SYSTEM, MECHANICAL FUNCTION	RHR (Hx, steam condensing line), unknown.
(d) INITIAL INDICATION	A broken pipe restraint was found during a routine inspection.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. A snubber on the RHR steam condensing line to the RHR Hx was found broken.
(f) CAUSE AND BASIS FOR EVENT	Inadequate detection and administrative procedures. RHR valves were leaking steam into the Hx causing the event. A contributing cause was the slippage of the pipe off of a trapeze hanger prior to the event.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Piping supports modified to minimize movement. Leaking valves to be repaired.

PRELIMINARY

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BWR EVENT ITEM 18

(a)	PLANT, MODE, EVENT DATE	Brunswick 1, 0% power, 12/12/79.
(b)	REFERENCE	50325 LER 79-107 (1/14/80).
(c)	SYSTEM, MECHANICAL FUNCTION	Mainsteam (SRV, snubbers), SRV lifting.
(d)	INITIAL INDICATION	Discovery of damaged snubber during periodic test.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Variety of snubber damage to ten different snubbers.
(f)	CAUSE AND BASIS FOR EVENT	Reactor scrammed, relief valves lifted, and analysis indicated that a water slug could cause the damage if selected snubbers were nonfunctional.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Repairs, inspections and tests, commitments, analysis of actual event and piping integrity analysis being done.

PRELIMINARY

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BWR EVENT ITEM 19

(a)	PLANT, MODE, EVENT DATE	Brunswick 1, 90% power, 3/28/81.
(b)	REFERENCE	50325 Letter (3/30/81), 50325 LER 81-034 (4/24/81).[a] See 29.
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (injection line), valve opening.
(d)	INITIAL INDICATION	Damage found during inspection of piping supports.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Failed weld, bent pin, bent steel, broken guide seal.
(f)	CAUSE AND BASIS FOR EVENT	Attributed to one or more water hammer events although no known or observed events have taken place. Cause is believed to be in the keep-full system and system leakage.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Daily venting begun. Observation of lines during core spray and HPCI operability tests required. Evaluate and modify valves to insure no leakage. Evaluate and modify the keep-full system as required.

a. As a result of this event, further inspection of this plant and Brunswick 2 was begun. The results of the inspections for both plants were attached as part of this LER. Cause and corrective measures were the same for both plants.

PRELIMINARY

DRAFT

BWR EVENT ITEM 20

(a) PLANT, MODE, EVENT DATE	Brunswick 1, 75% power, 4/14/81.
(b) REFERENCE	50325 LER 81-046 (4/28/81).
(c) SYSTEM, MECHANICAL FUNCTION	RHR (steam supply to Hx), valve opening/closing.
(d) INITIAL INDICATION	Snubber damage found after changing RHR mode.
(e) WATER HAMMER TYPE, DAMAGE	Unknown. Broken snubber shaft on RHR Hx steam supply (line for steam condensing mode).
(f) CAUSE AND BASIS FOR EVENT	Possibly water hammer. RHR Loop 1A placed in torus circulation mode for sampling.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Snubber repaired. Steam condensing lines in both RHR loops inspected. Repeat of operating method resulted in no damage. Venting increased from monthly to daily until evaluation justifies less. Venting of steam condensing line every 4 hours. Investigate causes.

PRELIMINARY

DRAFT

BWR EVENT ITEM 21

(a) PLANT, MODE, EVENT DATE	Brunswick 2, shutdown cooling, 9/5/75.
(b) REFERENCE	50324-451 IR 75-53 (11/10/75).[a]
(c) SYSTEM, MECHANICAL FUNCTION	RHR (shutdown cooling lines), pump start/valve operation.
(d) INITIAL INDICATION	Damaged piping supports were noticed during a routine NRC inspection on 10/7/75. Water hammer heard on 9/5/75.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. The following are cumulative effects from three probable events (Items 21, 22, and 23): Metal insulation cover loose on steam condensing line to RHR Hx. RHR Hx relief valve blowdown line movement. Snubber between RHR pump discharge and Hx inlet pipe disconnected (no pin installed). Shutdown cooling suction and demineralizer water fill line movement. Dislocation and slippage of pipe supports on steam condensing line to RHR Hx.
(f) CAUSE AND BASIS FOR EVENT	Inadequate operational test, maintenance, inspection and reporting procedure. Possibly related to insufficient venting points and inadequate test, operational or maintenance procedures of discharge pipe and shutdown cooling suction. See 22 and 23.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Shutdown cooling procedures were revised to minimize air entrainment and to maintain a keep-full condition in the shutdown cooling suction. See 22 and 23.

a. Three separate events are reported in this reference. See Events 22 and 23.

PRELIMINARY

DRAFT

BWR EVENT ITEM 22

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, RHR operational surveillance test, 9/30/75.
(b)	REFERENCE	50324-451 IR 75-53 (11/10/75).[a]
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (shutdown cooling suction line), pump start.
(d)	INITIAL INDICATION	Damaged piping supports were noticed during a routine NRC inspection on 10/7/75. Water hammer heard on 9/30/75.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. See 21.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate operational test, maintenance, inspection and reporting procedure. See 21.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Additional venting points added. RHR valve and pump operability procedures revised to minimize valve cycling and partially draining the system. See 21.

a. Three separate events are reported in this reference. See Events 21 and 23.

PRELIMINARY

DRAFT

BWR EVENT ITEM 23

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, shutdown cooling, 9/30/75.
(b)	REFERENCE	50324-451 IR 75-53 (11/10/75).[a]
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (shutdown cooling lines), pump start/valve operation.
(d)	INITIAL INDICATION	Damaged piping supports were noticed during a routine NRC inspection on 10/7/75. Reactor level decrease noted on 9/30/75. Water hammer was not heard, but is believed to have occurred.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. See 21.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate operational test, maintenance, inspection and reporting procedure. See 21.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	See 21.

a. Three separate events are reported in this reference. See Events 21 and 23.

PRELIMINARY

DRAFT

BWR EVENT ITEM 24

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, shutdown, 9/76.
(b)	REFERENCE	50324-1018 (12/15/76) LER 76-140 (9/16/76).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line/drain pot, switch), valve opening.
(d)	INITIAL INDICATION	Water hammer occurrence.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Shock suppressor and hanger damage at several locations.
(f)	CAUSE AND BASIS FOR EVENT	Insufficient inspection and check-out (component failure). Excessive turbine exhaust line movement. The HPCI turbine exhaust piping was not drained (when HPCI system was initiated) because of a malfunction of a drain level switch and failure of a solenoid valve. Poor repeatability and annunciation failure of level switch.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	A vent, drain and a second set of isolation valves were installed at the steam drain line pots to allow switch calibration during normal plant operations. A preventive maintenance program was initiated for monthly testing of HPCI and RCIC steam line drain pot switches.

PRELIMINARY

DRAFT

BWR EVENT ITEM 25

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, currently unavailable, 10/76.
(b)	REFERENCE	50324 LER 76-164 (1/28/77).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (Hx, line), pump start.
(d)	INITIAL INDICATION	Delayed repair of broken shock suppressor.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Broken shock suppressor shaft.
(f)	CAUSE AND BASIS FOR EVENT	Steam bubble in RHR Hx during previous RHR pump start. Administrative oversight allowed plant startup and operation with broken suppressor because of a misplaced trouble ticket.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Currently unavailable.

PRELIMINARY

DRAFT

BWR EVENT ITEM 26

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, 72% power (torus cooling mode), 4/13/77.
(b)	REFERENCE	50324-1120 RO 77-29 (5/11/77).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (line, valves), pump start/valve opening.
(d)	INITIAL INDICATION	Damage to snubbers.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Attachment for RHR snubber sheared in half.
(f)	CAUSE AND BASIS FOR EVENT	Improper installation and lack of administrative controls. Occurred when loop was placed in torus cooling mode. Vent valves had been leaking steam, causing steam voids in RHR lines.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Administrative controls (caution tags) were instituted to require venting prior to manual start of RHR pump until steam leakage problem is resolved. Vent valve repaired to stop steam leakage into RHR loop.

PRELIMINARY

DRAFT

BWR EVENT ITEM 27

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, shutdown, 3/24/78.
(b)	REFERENCE	50324 LER 78-27 (4/19/78).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (discharge line, check valve), valve opening.
(d)	INITIAL INDICATION	Discovery of damaged snubber.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Snubber with broken shaft on HPCI discharge line.
(f)	CAUSE AND BASIS FOR EVENT	Inoperative component. Probable sticking of keep-full station check valve, preventing piping from filling properly. A scram and HPCI injection occurred earlier in the day. Previous check valve sticking problems encountered.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Scheduled inspection for next system outage. Investigate component/service condition suitability.

PRELIMINARY

DRAFT

BWR EVENT ITEM 28

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, 0% power, 4/12/80.
(b)	REFERENCE	50324 LER 80-030 (5/8/80).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (RHR 2B Hx), valve opening/closing.
(d)	INITIAL INDICATION	Discovery of damage during inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. Partially buckled Hx rib plate.
(f)	CAUSE AND BASIS FOR EVENT	Possibly water hammer, but none recalled in this system.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Replaced rib plate, 2A Hx inspected with no damage found, 1A and 1B Hx are to be inspected. Operating procedures revised to require venting. Evaluate in 6 months to see if venting still required.

PRELIMINARY

DRAFT

BWR EVENT ITEM 29

(a) PLANT, MODE, EVENT DATE	Brunswick 2, 90% power, 3/28/81.
(b) REFERENCE	50325 LER 81-034 (4/24/81).[a] See 19.
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (injection line), valve opening.
(d) INITIAL INDICATION	Damage found during inspection of piping supports.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Failed weld, bent pin, bent steel, broken guide seal.
(f) CAUSE AND BASIS FOR EVENT	Attributed to one or more water hammer events although no known or observed events have taken place. Cause is believed to be in the keep-full system and system leakage.
(g) CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Daily venting begun. Observation of lines during core spray and HPCI operability tests required. Evaluate and modify valves to ensure no leakage. Evaluate and modify the keep-full system as required.

a. As a result of the Brunswick 1 event reported in this LER, further inspection of this plant and Brunswick 1 was begun. The results of the inspections for both plants were attached as part of this LER. Cause and corrective measures were the same for both plants.

PRELIMINARY

DRAFT

BWR EVENT ITEM 30

(a) PLANT, MODE, EVENT DATE	Dresden 2, power escalation testing, 5/29/70.
(b) REFERENCE	Nuclear Safety, 12(1), p. 40, Jan - Feb, 1971; 50237-147 (3/3/71), -53 (7/6/70), -47 (6/2/70).
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (turbine, inlet steam line), valve opening.
(d) INITIAL INDICATION	Water hammer and damage to piping system.
(e) WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Dented pipe section, damaged insulation, and broken or bent pipe hangers and seismic snubbers.
(f) CAUSE AND BASIS FOR EVENT	Procedural deficiency (lack of instructions). Operating and maintenance procedures allowed water accumulation in steam line during HPCI isolation period. Valve line-up at start of maintenance and return to operational mode allowed reactor water to flow into the line and become entrapped. See 8, 40.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Revised procedures for valve closure when removing HPCI from service and adopted procedures to eliminate water accumulation in the steam header and line. Interlock system installed between valves inside and outside of drywell.

PRELIMINARY

DRAFT

BWR EVENT ITEM 31

(a)	PLANT, MODE, EVENT DATE	Dresden 2, pump operability check, 3/26/71.
(b)	REFERENCE	50237-164 (4/3/71).
(c)	SYSTEM, MECHANICAL FUNCTION	Core spray (valve, switch), pump start.
(d)	INITIAL INDICATION	Valve failed to open electrically.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Broken switch in the discharge valve limitorque operator.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design for service environment. Broken switch in the limitorque operator. This condition may be the result of one or more previous events.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Broken switch replaced and unit tested for operability.

PRELIMINARY

DRAFT

105

BWR EVENT ITEM 32

(a)	PLANT, MODE, EVENT DATE	Dresden 2, core spray system test, 3/29/71.
(b)	REFERENCE	50237-156 (4/24/71), -160 (4/29/71) -168 (5/14/71).
(c)	SYSTEM, MECHANICAL FUNCTION	Core spray (line), pump start/valve opening.
(d)	INITIAL INDICATION	Water hammer on pump start.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Seismic restraints impaired (one damaged, three pulled out of concrete and two loosened). Pipe hangers damaged (one hanger loaded to plastic condition and bowed, one permanently deformed, one had a cast iron coupling fractured and clevis elongated). One pipe sprung.
(f)	CAUSE AND BASIS FOR EVENT	Procedures not followed for manual keep-full and venting prior to pump start. Evidently core spray pumps were operated an unknown number of times with empty discharge piping.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	All cast pipe hanger parts will be replaced with forged parts to enhance load carrying capability and inhibit dynamic loading fracture type and components. A "jockey" pump system will be installed.

PRELIMINARY

DRAFT

BWR EVENT ITEM 33

(a) PLANT, MODE, EVENT DATE	Dresden 2, at power (surveillance of shutdown cooling isolation valves), 9/28/71.
(b) REFERENCE	50237-204 (11/1/71).
(c) SYSTEM, MECHANICAL FUNCTION	RHR (Hx valve, header), valve opening.
(d) INITIAL INDICATION	Hydraulic shock/water hammer. Recirculation pump vibration alarm initiated and a number of RV flow level and flux spikes were recorded.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Dislocated insulation downstream of shutdown cooling isolation valve and pipe movement. Valve closure/opening and valve motor/operator malfunction.
(f) CAUSE AND BASIS FOR EVENT	Inadequate operating procedures. After RHR shutdown cooling system operation, normal practice was to leave the reactor coolant side of the Hx full of water. Because two months elapsed since the system was last activated, valve leakage may have caused partial drainage of the Hx. On pump start, flow passed through the header and flashed to steam, since the header was partially voided also. A steam bubble was thus formed in the header and collapsed as system pressure was restored by opening an inlet isolation valve, causing a surge of water in the header toward a closed valve. A contributing factor was the higher than normal shutdown coolant water temperatures used.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Future valve testing to be conducted only at normal shutdown coolant temperatures.

PRELIMINARY

DRAFT

BWR EVENT ITEM 34

(a) PLANT, MODE, EVENT DATE	Dresden 2, 95% power (core spray valve operability testing), 7/11/76.
(b) REFERENCE	50237-1088 RO 76-49 (8/9/76).
(c) SYSTEM, MECHANICAL FUNCTION	Core spray (valve, switch), pump start/valve opening.
(d) INITIAL INDICATION	Core spray outboard injection valve lost open/closed indication. Could not be electrically reopened.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Valve limit switch damaged causing valve malfunction.
(f) CAUSE AND BASIS FOR EVENT	Design and procedural deficiencies. It is assumed that water hammer events that occurred prior to the jockey pump addition damaged the switch. The switch then failed after subsequent valving cycles.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Switch replaced and components tested.

PRELIMINARY

DRAFT

BWR EVENT ITEM 35

(a) PLANT, MODE, EVENT DATE	Dresden 3, at power in run mode (power increase), 6/23/74.
(b) REFERENCE	50249-450 (7/2/74).
(c) SYSTEM, MECHANICAL FUNCTION	Main feedwater (regulating valve), valve closure.
(d) INITIAL INDICATION	FW reg valve lock up, service air compressor, RWCU pump, and FW heater tripped, FW and reactor level decrease.
(e) WATER HAMMER TYPE, DAMAGE	Unknown. FW low flow reg valve open and rotated with all airlines and electrical feeds broken. Upstream pipe support broken and concrete base cracked. The "A" FW reg valve upstream pipe support pad pulled out of cement base. The "B" FW reg valve air supply line broken off. Pipe support movement on all three reactor feed pump lines. Bent warming line on "A" feed pump and loose insulation on "A" minimum flow line to condenser. Tack welds broken on pipe hangers on heater outlet. Heater pipe support displaced. Extraction steam piping support pedestal movement and pad cement base cracked. Three pipe support pedestal cement pads cracked under the hotwell of feed pump suction header.
(f) CAUSE AND BASIS FOR EVENT	Specific cause not identified. FW valve vibration and inadvertent closure possibly related to control system malfunction.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	All affected piping and components repaired and welds inspected and NDT conducted. FW control circuit instrumented to determine the existence of spurious signals.

PRELIMINARY

DRAFT

BWR EVENT ITEM 36

(a) PLANT, MODE, EVENT DATE	Dresden 3, startup (valve operability check), 11/27/74.
(b) REFERENCE	50249-536 (12/17/74).
(c) SYSTEM, MECHANICAL FUNCTION	Core spray (valve, switch), pump start/valve opening.
(d) INITIAL INDICATION	Valve being tested would not reopen electrically.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Valve limit switch damaged causing valve malfunction.
(f) CAUSE AND BASIS FOR EVENT	Design and procedural deficiencies. Reactor scram on this day caused by water hammer which damaged the limit switch for the core spray valve.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Main steam line switches relocated to an area not subject to core spray transient. All adjacent electrical switches inspected for serviceability.

PRELIMINARY

DRAFT

BWR EVENT ITEM 37

(a)	PLANT, MODE, EVENT DATE	Dresden 3, 69% power, 10/5/79.
(b)	REFERENCE	50249 LER 79-028 (11/1/79).
(c)	SYSTEM, MECHANICAL FUNCTION	LPCI (drywell spray header), valve opening.
(d)	INITIAL INDICATION	Damage found during inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Bent load carrying support bolts, two bottomed out spring hangers.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable (probably water hammer prior to ECCS jockey pump installation).
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Bolts replaced, springs reset. No further action deemed necessary.

PRELIMINARY

DRAFT

BWR EVENT ITEM 38

(a)	PLANT, MODE, EVENT DATE	Dresden 3, refueling, 4/2/80.
(b)	REFERENCE	50249 LER 80-015 (4/14/80).
(c)	SYSTEM, MECHANICAL FUNCTION	Reactor water cleanup, valve opening/closing.
(d)	INITIAL INDICATION	Fully retracted mechanical snubber found during refueling.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. Crack in affected pipe between first isolation valve and containment penetration.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Pipe replaced. Other cleanup piping weld penetrant tested with no cracks found. System isolation valve circuit to be modified to enable throttling.

PRELIMINARY

DRAFT

BWR EVENT ITEM 39

(a)	PLANT, MODE, EVENT DATE	Duane Arnold, cold shutdown (RHR shutdown cooling), 4/10/74.
(b)	REFERENCE	50331-175 (4/19/74); 50331 UE-2 (5/10/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Core spray (line, switch), pump start/valve opening.
(d)	INITIAL INDICATION	Accidental actuation of core spray.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Anchors were pulled loose from one seismic restraint.
(f)	CAUSE AND BASIS FOR EVENT	An operator accidentally activated a temporary test switch lying on the floor in front of the console. The discharge piping was only partially full because the keep-full pump was inoperative. For the existing plant conditions, the keep-full system was not required.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Relocated test switch to a location where accidental actuation was not possible. Maintenance procedures revised to preclude conditions of this nature.

PRELIMINARY

DRAFT

BWR EVENT ITEM 40

(a) PLANT, MODE, EVENT DATE	Duane Arnold, 30% power, steady state, 6/11/74.
(b) REFERENCE	50331-215 AO-74-11 (6/21/74).
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (steam line, isolation valve), valve opening.
(d) INITIAL INDICATION	A normally open HPCI outboard steam supply isolation valve was indicating closed without HPCI automatic isolation signal.
(e) WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. HPCI inoperable. Damage to pipe insulation, a pipe hanger, seismic snubbers, a pressure indicator, and an HPCI steam line drain pot indicator.
(f) CAUSE AND BASIS FOR EVENT	Procedural deficiency (no procedures for situation). Operator error (methods not followed). Design inadequacy. Control panel inspection indicated closure of the HPCI outboard steam supply isolation valve. Since an auto-isolation signal was not present the operator opened the valve. The HPCI fire protection system then actuated (later found to be caused by leaking steam that dislodged a switch). Movement and impact of a water slug from steam condensation occurred in the HPCI steam supply line when the outboard isolation valve (a normally open seal-in type) was opened while the inboard isolation valve was full open. Operating procedures did not cover opening of the outboard valve while the inboard valve was full open. In addition, proper steam line warmup methods were not followed. See 8, 30.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Outboard isolation valve seal-in feature eliminated. Procedures revised to emphasize warmup of both RCIC and HPCI steam supply piping.

PRELIMINARY

DRAFT

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BWR EVENT ITEM 41

(a)	PLANT, MODE, EVENT DATE	Duane Arnold, 83% power, 1/31/77.
(b)	REFERENCE	50331-957 RO 77-8 (2/11/77), -1216 RO 77-8 update (1/20/78), -1220 RO 77-29 (4/20/77), -1217 RO 77-29 update (1/20/78).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (fuel pool cooling line), pump start/valve opening.
(d)	INITIAL INDICATION	Pipe support damage found during normal plant inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. The mounting bracket for the hydraulic shock suppressor on the RHR to fuel pool cooling line was found loose. Later, a calculation indicated that a section of the RHR to fuel pool cooling line was overstressed and pipe hangers and restraints were found damaged.
(f)	CAUSE AND BASIS FOR EVENT	Procedural deficiency (inadequate operating instructions, test procedures, and instal[illegible]on). Induced on RHR pump start when line not completely filled with water. A procedural deficiency in the operating instructions did not contain precautions to insure proper venting when the system was manually initiated.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Monthly LPCI test procedures revised to require venting prior to and after each test. All concrete anchor attachments inspected and corrected. Anchor to be tested at a later date for loading capability. Test conducted to verify no pipe movement on RHR pump start.

PRELIMINARY

DRAFT

115

BWR EVENT ITEM 42

(a) PLANT, MODE, EVENT DATE	Duane Arnold, 85% power (core spray logic testing), 2/11/77.
(b) REFERENCE	50331-958 LER 77-9 (2/25/77).
(c) SYSTEM, MECHANICAL FUNCTION	Core spray (line, valve), pump start/valve opening.
(d) INITIAL INDICATION	Accidental actuation of core spray. Loud noises heard.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. The motor operated clutch housing on a core spray injection valve fractured.
(f) CAUSE AND BASIS FOR EVENT	Procedural deficiency. The system core spray was probably initiated when temporary test jumpers were removed and possibly shorted across electrical contacts. Partially drained discharge piping may have resulted due to leakage at a bypass test valve.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Clutch housing replaced with one made from material to increase ductility for less susceptibility to fracture type stress failures. An evaluation of the adequacy of the RHR and core spray fill pump was initiated. Adjacent components inspected for serviceability.

PRELIMINARY

DRAFT

BWR EVENT ITEM 43

(a)	PLANT, MODE, EVENT DATE	Duane Arnold, cold shutdown - no fuel, 12/21/78.
(b)	REFERENCE	50331 LER 78-035 (1/19/79).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (piping), pump start/valve opening.
(d)	INITIAL INDICATION	Damage noted during special inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Snubber swivel bent, snubber wall attachment plate pulled 1/4" from wall.
(f)	CAUSE AND BASIS FOR EVENT	Unplanned RHR system startup due to defective procedure. System had not been maintained full due to extended outage.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Swivel replaced, realigned and wall attachment secured.

PRELIMINARY

DRAFT

BWR EVENT ITEM 44

(a)	PLANT, MODE, EVENT DATE	Duane Arnold, 44% power, 9/27/79.
(b)	REFERENCE	50331 LER 79-025 (10/19/79).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (to fuel pool line), valve opening.
(d)	INITIAL INDICATION	Damaged pipe supports found during special inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. Two seismic restraints and one pipe support found not intact.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable (apparently caused by a water hammer).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	1 of 3 supports repaired, RT of line in progress. Investigation is in progress. Status of other 2 supports unknown.

PRELIMINARY

DRAFT

BWR EVENT ITEM 45

(a)	PLANT, MODE, EVENT DATE	Hatch 1, shutdown cooling, 12/15/74.
(b)	REFERENCE	50321-218 AO-74-27 (1/16/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (head spray line), pump start/valve opening.
(d)	INITIAL INDICATION	Observed leak in head spray line.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Crack in head spray line.
(f)	CAUSE AND BASIS FOR EVENT	Operator error/procedural deficiency/ design. Potential for operator to actuate wrong valve for system venting.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Section of line removed and a new section of pipe and a new anchor installed.

PRELIMINARY

DRAFT

BWR EVENT ITEM 46

(a)	PLANT, MODE, EVENT DATE	Fitzpatrick, functional testing, 4/10/74.
(b)	REFERENCE	50333 (5/3/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, lines), pump start.
(d)	INITIAL INDICATION	Water hammer occurrence.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Piping movement. The bottom of the pump discharge basket strainer was blown off and grouting for strainer support was chipped. Buckled piping at seismic trunnion locations. Bent seismic trunnions, a pipe support with a failed turnbuckle, and a pipe support with failed anchors near the pump.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate administrative controls and temporary operating procedures. Due to a misunderstanding the pump discharge and motor cooler isolation valves were not closed and remained open overnight causing the system to drain. The event occurred the next day on pump startup with the pump discharge valve fully open and a partially drained system. Normally, the pump discharge valve is closed and gradually opened after a pump start.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Currently unavailable.

PRELIMINARY

DRAFT

BWR EVENT ITEM 47

(a) PLANT, MODE, EVENT DATE	Fitzpatrick, cold shutdown, 3/21/75.
(b) REFERENCE	50333 AO 75-31 (3/26/75).
(c) SYSTEM, MECHANICAL FUNCTION	RHR (containment spray), pump start/ valve opening.
(d) INITIAL INDICATION	Damage found during inspection.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Pipe restraints and a snubber damaged on "A" side.
(f) CAUSE AND BASIS FOR EVENT	Operation of RHR in shutdown cooling mode with discharge piping not water filled.
(g) CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Keep-full system planned.

PRELIMINARY

DRAFT

BWR EVENT ITEM 48

(a)	PLANT, MODE, EVENT DATE	Fitzpatrick, cold shutdown, 5/24/75.
(b)	REFERENCE	50333-334 AO 75-48 (5/28/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (containment spray line), pump start/valve opening.
(d)	INITIAL INDICATION	Pipe movement reported.
(e)	WATER HAMMER TYPE, DAMAGE	Probable flow-into-voided-line. Damaged pipe restraints and snubber on "B" side.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable. Unusual service condition.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	System repair and NDT conducted.

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BWR EVENT ITEM 49

(a)	PLANT, MODE, EVENT DATE	Fitzpatrick, at power, 7/20/75.
(b)	REFERENCE	50333-346 AO 75-63 (7/25/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (Hx/HPCI steam supply line), valve opening.
(d)	INITIAL INDICATION	Damaged pipe restraint found during routine inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Pipe restraint on RHR steam supply line to the Hx was pulled loose.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate operating procedures. Event occurred while warming up steam supply to HPCI turbine.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Inspected and replaced damaged pipe restraint.

PRELIMINARY

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BWR EVENT ITEM 50

(a)	PLANT, MODE, EVENT DATE	Fitzpatrick, reactor startup (HPCI line warmup), 9/7/75.
(b)	REFERENCE	50333-385 AO 75-73 (9/15/75).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (HPCI steam supply line to RHR Hx), valve opening.
(d)	INITIAL INDICATION	Discovery of pipe restraint damage.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Several pipe restraints sustained damage on HPCI steam line to RHR Hx.
(f)	CAUSE AND BASIS FOR EVENT	Design deficiency (insufficient drainage). Event occurred on warmup of HPCI steam supply line during reactor startup.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Provisions to drain HPCI steam supply line to Hx.

PRELIMINARY

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BWR EVENT ITEM 51

(a) PLANT, MODE, EVENT DATE	Millstone 1, startup (50% load turbine trip test), 12/9/70.
(b) REFERENCE	50245-96 (3/23/71), -62 (3/3/71), -48 (12/19/70).
(c) SYSTEM, MECHANICAL FUNCTION	Main steam (turbine bypass lines, valves), valve instability.
(d) INITIAL INDICATION	Damage to piping system.
(e) WATER HAMMER TYPE, DAMAGE	Steam hammer. Excessive movement of all four main steam lines. Fixed pipe support common to all four main steam lines (located between outboard isolation and turbine stop valves) stressed beyond yield. Line movement damaged other steel supports and line connections. Excessive line movement in bypass lines between bypass valves and condenser.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design (failure to consider concurrent loading conditions). During a planned turbine trip, the rapid closure of the main steam stop valves caused a transient. A contributing cause was the malfunction of three separate components associated with bypass valve actuation. The design of a fixed pipe support anchor common to all four main steam lines did not consider dynamic forces generated by rapid closure of the stop valves.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Bypass valve actuation components repaired or modified. New supports were designed and installed on main steam lines and bypass lines. All systems tested at 50%, 75% and 100% load rejection turbine trips.

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BWR EVENT ITEM 52

(a)	PLANT, MODE, EVENT DATE	Millstone, early operational phase, 6/12/72.
(b)	REFERENCE	50245-138 (7/13/72).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (LPCI crossover line), pump start/valve opening.
(d)	INITIAL INDICATION	Investigation of pipe hangers revealed some supports had shifted from original position.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Severe pipe and pipe support movement. LPCI cross tie header shifted from original position and damaged.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate operating procedures. Probably occurred during early operational phase while LPCI keep-full line was not in service.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Necessary repairs made and investigations of event undertaken.

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BWR EVENT ITEM 53

(a)	PLANT, MODE, EVENT DATE	Millstone 1, currently unavailable, second half of CY 1972.
(b)	REFERENCE	50245-194 (2/12/73).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (steam bypass header at condenser), valve opening.
(d)	INITIAL INDICATION	Structural failure of steam bypass header end caps at the main condenser.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. End cap failures of steam bypass header numerous times.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Condensate formation and accumulation in steam bypass header.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Existing drain holes in header enlarged and additional drain holes added to permit condensate drainage and prevent buildup.

PRELIMINARY

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BWR EVENT ITEM 54

(a) PLANT, MODE, EVENT DATE	Millstone 1, 100% power (surveillance and maintenance), 2/12/76.
(b) REFERENCE	50245 (2/17/76); Nuclear Industry, 23(2), P. 17, Feb. 76); 50245 RO 76-4/T (2/26/76), -648 RO 76-4/10 (3/5/76), Insp. 76-5 (3/26/76), -678 (3/31/76), RO 76-4/10 Update (9/28/76).
(c) SYSTEM, MECHANICAL FUNCTION	RHR (isolation condenser, supply line), valve opening.
(d) INITIAL INDICATION	Main transformer deluge alarm signal. Generator trip followed by turbine and reactor trip. RHR isolation condenser noise.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse or steam-water entrainment. One RHR isolation condenser steam tube failed and allowed release via the vent stack. The majority of tube inlet and thermal ferrules collapsed. The mounting studs on both heat shields were bent or broken. One shield was pulled away and one had a cracked weld. The parallel supply line showed evidence of substantial movement as indicated by restraint penetration of insulation blanket. The vertical supply line showed a small downward movement.
(f) CAUSE AND BASIS FOR EVENT	Inadequate failure mode alarm and detection systems, procedural deficiencies, and operator response. Not water hammer induced, but primarily a result of RHR isolation condenser steam tube failure. Initiated by a chain of events starting with generator trip caused by arcing in main transformer by inadvertent actuation of deluge system. A number of unusual situations then evolved from mechanical malfunctions and operator interpretations that have water hammer implications when viewed in context with a later occurrence. See 55. The tube failure was a result of structural degradation from

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stress corrosion cracking. Damage to internals and movement of supply lines bore indications of being thermally stress induced.

(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION

Repaired and modified isolation condenser. Added temperature alarm to shell side water. Increased sensitivity of vent and radiation alarm and frequency of shell side water activity samples. Provisions instituted for noncontaminated shell side makeup and periodic maintenance program.

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BWR EVENT ITEM 55

(a)	PLANT, MODE, EVENT DATE	Millstone 1, shutdown, 3/11/78.
(b)	REFERENCE	50245 RO 78-7/3L (4/10/78).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (isolation condenser, steam supply line), N/A.
(d)	INITIAL INDICATION	Observed movement of isolation condenser steam supply lines.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment or steam-bubble collapse. Movement of isolation condenser steam supply lines.
(f)	CAUSE AND BASIS FOR EVENT	Procedural deficiency. A reactor vessel water level increase occurred and allowed water carry-over into the isolation condenser steam supply lines presumably while the RHR condenser was functioning.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Recommendation to install additional sway arrestors in condenser steam supply lines.

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BWR EVENT ITEM 56

(a)	PLANT, MODE, EVENT DATE	Millstone 1, 50% power, 4/17/78.
(b)	REFERENCE	50245 RO 78-8/1T (4/20/78).
(c)	SYSTEM, MECHANICAL FUNCTION	ECCS (core spray, LPCI), valve opening/closing.
(d)	INITIAL INDICATION	Deficiencies found during newly incorporated ISI.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. Degradation of CS and LPCI piping support systems.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable. Water hammer not specifically called out. Evidence of dynamic loading beyond load carrying capacity. Review determined inadequate original designs.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Review and subsequent design modifications and repairs instituted.

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BWR EVENT ITEM 57

(a) PLANT, MODE, EVENT DATE	Millstone 1, during shutdown following reactor scram, 12/19/79.
(b) REFERENCE	50245 LER 79-036 (1/18/80).
(c) SYSTEM, MECHANICAL FUNCTION	RHR (isolation condenser, steam supply line), valve opening.
(d) INITIAL INDICATION	Movement in piping observed.
(e) WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. No damage.
(f) CAUSE AND BASIS FOR EVENT	During shutdown from scram, water hammer type pipe movement observed. Investigation determined that water had been introduced into isolation condenser steam supply line because reactor vessel water level had been maximized, based on TMI experience.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION .	Instructions to maximize reactor vessel level revised.

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BWR EVENT ITEM 58

(a)	PLANT, MODE, EVENT DATE	Millstone 1, 100% power steady state, 2/20/80.
(b)	REFERENCE	50245 LER 80-005 (3/17/80), LER 80-005 update (4/10/80).
(c)	SYSTEM, MECHANICAL FUNCTION	ECCS (core spray), pump start.
(d)	INITIAL INDICATION	Damage found during core spray subsystem A pipe support inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Pipe support damage. Valve motor to operator bolts loose.
(f)	CAUSE AND BASIS FOR EVENT	Water hammer during routine core spray pump operability surveillance which was caused by steam bubble in piping due to seat leakage past core spray injection valve and core spray check valve.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Keep-full system modified. Plans being made for check valve replacement and modification to injection valve to monitor seat integrity. Valve bolts replaced with Ny-loc cap screws.

PRELIMINARY

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BWR EVENT ITEM 59

(a)	PLANT, MODE, EVENT DATE	Monticello, currently unavailable, 1971.
(b)	REFERENCE	50263-115 (2/28/72).
(c)	SYSTEM, MECHANICAL FUNCTION	RCIC (turbine exhaust line), unknown.
(d)	INITIAL INDICATION	Water hammer.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. No damage indicated.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (no allowance for prevention of vacuum condition). Condensation of residual steam in turbine exhaust line after RCIC turbine shutdown. Torus water drawn into evacuated line.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Vacuum breakers installed.

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BWR EVENT ITEM 60

(a) PLANT, MODE, EVENT DATE	Monticello, HPCI flow rate surveillance, 7/17/72.
(b) REFERENCE	50263-139 (7/27/72).
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (turb ne exhaust line, stop check valve), check valve operation.
(d) INITIAL INDICATION	HPCI turbine trip.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Rupture discs on HPCI turbine exhaust line to suppression tank relieved and steam impingement rendered temperature switches inoperable. Turbine exhaust high pressure alarm and turbine trip failed to activate.
(f) CAUSE AND BASIS FOR EVENT	Inadequate component and subsystem design. The HPCI turbine exhaust stop check valve pin failed causing the valve disc to disengage and block the line. The rupture discs in the turbine exhaust line to the suppression c amber relieved. The turbine exhaust high pressure alarm failed to initiate because of a delayed response time believed associated with sensing line design. Steam issuing from the relieved rupture discs impinged on adjacent temperature switches rendering them inoperable.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	A modified disc pin was installed according to manufacturers recommendation. Electrical box with temperature switch wiring modified.

PRELIMINARY

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BWR EVENT ITEM 61

(a) PLANT, MODE, EVENT DATE	Nine Mile Pt. 1, power escalation testing, 10/12/69.
(b) REFERENCE	50220-46 (7/10/70), -39 (4/20/70).
(c) SYSTEM, MECHANICAL FUNCTION	RHR (emergency condenser steam lines), valve opening.
(d) INITIAL INDICATION	Observed water hammer.
(e) WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. Currently unavailable.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design (no provisions for adequate venting or water drainage). Piping not drained, pitched or vented properly. Too fast a heatup of lines when isolation valves were opened.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Removed check valves from emergency condenser bleed line to main steam line. Included automatic closure of bleed line isolation valves in response to RV isolation signal. Additional drain points added to horizontal run of steam supply lines to emergency condenser.

PRELIMINARY

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BWR EVENT ITEM 62

(a) PLANT, MODE, EVENT DATE	Oystar Creek 1, preoperational testing, startup to mid 1971.
(b) REFERENCE	50219-155 (12/15/71), -140 (6/25/71), -146 (9/22/71), -172 (3/6/72).
(c) SYSTEM, MECHANICAL FUNCTION	Core spray (line, valve), pump start.
(d) INITIAL INDICATION	Water hammer on pump start.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Pipe movement and possible overstress condition at several points.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design and operational procedures. Empty discharge piping on pump start as a result of leaky check valves.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Jocky pump system installed to maintain continuously filled loops. Stress, fatigue and water hammer engineering studies and analysis and NOT were performed to verify adequacy of core spray system subject to prior occurrences.

PRELIMINARY

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BWR EVENT ITEM 63

(a) PLANT, MODE, EVENT DATE	Oyster Creek 1, startup testing (valve timing test), 11/16/71.
(b) REFERENCE	50219-153 (12/14/71), -170 (2/28/72).
(c) SYSTEM, MECHANICAL FUNCTION	Main steam (lines, valve), valve closure.
(d) INITIAL INDICATION	Incomplete MSIV closure.
(e) WATER HAMMER TYPE, DAMAGE	Steam hammer. Valve operator cast iron spud cushion crushed with pieces preventing valve closure. Similar damage on same valve in another loop.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design (component/service condition incompatibility). Sudden valve opening with large pressure differential (800 psia) existing across valve disc.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Valves repaired and tested. Corrective action taken will be reevaluated after analysis of damaged spud cushions.

PRELIMINARY

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BWR EVENT ITEM 64

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 2, shutdown and depressurized, 11/17/75.
(b)	REFERENCE	50277-565 RO 75-76 (11/26/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (shutdown cooling suction line), pump start/valve opening.
(d)	INITIAL INDICATION	A rigid pipe support was found broken on a routine inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. Broken rigid pipe support on RHR shutdown cooling suction line downstream of outer isolation valve.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable (mechanical failure).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Pipe support replaced. Engineering investigation being conducted.

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BWR EVENT ITEM 65

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 3, power escalation testing shutdown, 10/15/74.
(b)	REFERENCE	50278-215 AO 74-16 (10/25/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (lines), valve closure.
(d)	INITIAL INDICATION	Damage to piping support system found during routine inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Steam hammer. Snubbers on piping between bypass valves and main condenser damaged. Snubbers on one bypass valve detached (the nuts and bolt holding the snubber eye vibrated loose). The snubber pipe clamps on two additional bypass valves rotated. Another valve snubber was broken.
(f)	CAUSE AND BASIS FOR EVENT	Maintenance deficiency (improper EHC calibration). Out-of-calibration acceleration amplifiers in the electro hydraulic controls system caused valve cycling during a prior turbine overspeed test.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Repairs made and NOT performed on system. Components and piping pressure tested. EHC system acceleration amplifiers recalibrated to preclude valve fluctuations. Operational tests were satisfactory.

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BWR EVENT ITEM 66

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 3, approximately 100% power, 2/14/75.
(b)	REFERENCE	50278-283 AO 75-12 (2/24/75).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (steam supply line/drain pot), valve opening.
(d)	INITIAL INDICATION	Movement of HPCI steam supply line.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. No induced damage.
(f)	CAUSE AND BASIS FOR EVENT	Inoperative component (alarm signal) and administrative deficiency. Failure of steam trap to drain properly and drain pot level switch to trip on high level. Drain pot level switch linkage defective. Steam trap orifice plugged with dirt.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Components refurbished and tested for operability.

PRELIMINARY

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BWR EVENT ITEM 67

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 3, 57% power, 12/2/75.
(b)	REFERENCE	50278-476 AO 75-73 (12/12/75).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (steam supply line, steam trap, switch), valve opening.
(d)	INITIAL INDICATION	Movement of HPCI steam supply line.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-water entrainment. No damage indicated.
(f)	CAUSE AND BASIS FOR EVENT	Inoperative component, alarm signal failure, and administrative deficiency. Failure of steam trap to drain properly due to friction in trap internals and failure of the drain pot level switch to trip on high level allowed water accumulation in the steam supply line.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Defective components repaired or replaced.

PRELIMINARY

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BWR EVENT ITEM 68

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 2 and 3, refueling shutdown 70% power, 5/76.
(b)	REFERENCE	Atomic Energy Clearing House, 21 (42), pp 32-38, 10/20/75.
(c)	SYSTEM, MECHANICAL FUNCTION	Cooling water (cooling tower lift pump), pump start/valve opening.
(d)	INITIAL INDICATION	Water gushing from lift pump house door.
(e)	WATER HAMMER TYPE, DAMAGE	Probably column separation. Bolts holding suction bell housing to pump casing failed. Discharge piping cracked and base plate shifted.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Necessary repairs and checkout made.

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BWR EVENT ITEM 69

(a) PLANT, MODE, EVENT DATE — Pilgrim 1, power escalation testing, 7/24/72.

(b) REFERENCE — 50293-89 (9/5/72).

(c) SYSTEM, MECHANICAL FUNCTION — Main steam (lines, valves), valve closure.

(d) INITIAL INDICATION — Damaged piping supports found during inspection.

(e) WATER HAMMER TYPE, DAMAGE — Steam hammer. Pipe hanger torn from support on one main steam line, and bent hangers on three other main steam lines downstream from MSIV's near second elbow.

(f) CAUSE AND BASIS FOR EVENT — Inadequate design (failure to consider cumulative/concurrent loading). Startup testing involved repeated turbine stop and control valve closures with normal turbine trips. The additional dynamic loading induced by the valve closure acted concurrently with existing loads to overstress a pipe support carrying more than its desired dead weight. The attachment method did not utilize the support load carrying capability.

(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION — Damaged components were repaired and NDT conducted. Pipe hanger attachments reinforced for better load distribution. Rigid pipe supports set to designed static load for proper dead weight distribution. A vertical load restraining structure was designed and installed. Verification by turbine stop valve closure testing.

PRELIMINARY

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BWR EVENT ITEM 70

(a)	PLANT, MODE, EVENT DATE	Pilgrim 1, currently unavailable, 1972.
(b)	REFERENCE	50293-119 (2/8/73).
(c)	SYSTEM, MECHANICAL FUNCTION	Condenser (spargers), valve opening.
(d)	INITIAL INDICATION	Broken spargers and baffle damage in condenser.
(e)	WATER HAMMER TYPE, DAMAGE	Probably steam-water entrainment (or) steam-bubble collapse. Condenser internal spargers broken.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (probably condensate formation and accumulation in sparger).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Baffles repaired.

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BWR EVENT ITEM 71

(a)	PLANT, MODE, EVENT DATE	Pilgrim 1, power increase, 1/6/76.
(b)	REFERENCE	50293-602 IE 76-01 (1/29/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (regulating valve), valve instability.
(d)	INITIAL INDICATION	FW system vibrations.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (vibration)--possibly column separation. Yoke of startup valve fractured, resulting in complete ejection of valve stem. Valve body cracked. Pipe hanger bent on FW line.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (component/service condition incompatibility). Cycling of FW valve due to faulty pneumatic valve operator induced flow vibrations.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Replaced failed valve and NDT performed on welds in affected FW piping. Replacing pneumatic valve operators on FW regulating valves with type less susceptible to vibratory response.

PRELIMINARY

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BWR EVENT ITEM 72

(a)	PLANT, MODE, EVENT DATE	Pilgrim 1, 5% power, 2/3/77.
(b)	REFERENCE	50293-829 RO 77-3/3L (3/3/77).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (pump, piping), pump start/valve opening.
(d)	INITIAL INDICATION	Attempt to start salt water service pump unsuccessful.
(e)	WATER HAMMER TYPE, DAMAGE	Probably column separation. The top column pipe had a 360 degree fracture just below its top flange, allowing the pipe to drop into and jam the pump impeller.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable. The location of damage is indicative of column separation in a discharge line.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Necessary repairs, checkout and verification tests made and completed.

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BWR EVENT ITEM 73

(a)	PLANT, MODE, EVENT DATE	Quad Cities 1, shutdown, 4/3/72.
(b)	REFERENCE	50254-74 (4/5/72).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (piping, containment spray) pump start/valve opening.
(d)	INITIAL INDICATION	Water hammer occurred on 4/3/72.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Damaged pipe hangers and restraints. One containment spray system rendered inoperable.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable (water hammer during RHR system testing).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Repairs to system and testing conducted.

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BWR EVENT ITEM 74

(a)	PLANT, MODE, EVENT DATE	Quad Cities 1, shutdown, 4/4/72.
(b)	REFERENCE	50254-80 (4/11/72).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (fuel pool cooling line), pump start/valve opening.
(d)	INITIAL INDICATION	Damaged seismic restraints and pipe hangers found during routine inspection on 4/4/72. Noise heard 4/2/72 on pump start.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Damaged hangers, pipe restraints, and seismic restraints. Mechanical failure of containment spray valve motor housing prevented electrical valve operation.
(f)	CAUSE AND BASIS FOR EVENT	Design and procedural deficiency (improper venting of discharge piping). When venting at high points only, several possible points for air entrapment or accumulation are possible upstream from all discharge check valves and the tie line to the fuel pool cooling system.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Filling and venting procedures were altered to use the condensate transfer system which provides for a higher pressure and larger flow water supply for filling than the jockey pump system previously used.

PRELIMINARY

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BWR EVENT ITEM 75

(a) PLANT, MODE, EVENT DATE	Quad Cities 1, hot shutdown (maintenance outage), 6/9/72.
(b) REFERENCE	50254 (6/17/72); Nuclear Safety, Vol. 13 No. 6, pp 490-493, Nov. - Dec., 1972; 50254-103 (7/10/72), -108 (7/24/72), -110 (7/31/72).
(c) SYSTEM, MECHANICAL FUNCTION	Cooling water (main condenser), valve closure.
(d) INITIAL INDICATION	Sudden, inadvertent valve closure.
(e) WATER HAMMER TYPE, DAMAGE	Probably column separation. A ruptured rubber expansion joint in the condenser water box. Recirculation water flooded the condensate pump room. Water immersion damaged RHR service water pumps and motors and other equipment.
(f) CAUSE AND BASIS FOR EVENT	Inadequate maintenance or repair practices. While performing condenser modifications, a butterfly valve slammed closed while condenser circulating water pumps were in operation. This event occurred while venting the valve hydraulic oil system.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Repaired or replaced damaged components and systems. New maintenance procedures for venting and circulating water valve hydraulic system were instituted. A number of corrective measures were initiated to isolate and prevent the inadvertent flooding of safety systems/components located in the condensate pump room.

PRELIMINARY

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BWR EVENT ITEM 76

(a) PLANT, MODE, EVENT DATE	Quad Cities 1, currently unavailable, 1973.
(b) REFERENCE	50254 RO 73-4 (4/13/73).
(c) SYSTEM, MECHANICAL FUNCTION	Condenser circulating water system, valve closing.
(d) INITIAL INDICATION	Currently unavailable.
(e) WATER HAMMER TYPE, DAMAGE	Column separation. Rupture of rubber expansion joint in line. Damaged engineered safety-system equipment due to flooding.
(f) CAUSE AND BASIS FOR EVENT	During maintenance work, malfunction caused a butterfly valve to slam shut resulting in water hammer.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Modification of room construction and relocation of equipment.

PRELIMINARY

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BWR EVENT ITEM 77

(a) PLANT, MODE, EVENT DATE	Quad Cities 1, 94% power (RCIC pump operability surveillance), 10/29/76.
(b) REFERENCE	50254-937 RO 76-33 (11/10/76).
(c) SYSTEM, MECHANICAL FUNCTION	RCIC (pump), valve opening.
(d) INITIAL INDICATION	RCIC tech spec pump flow and pressure requirements not achieved.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-void within pump. Two of five pump stages were severely damaged.
(f) CAUSE AND BASIS FOR EVENT	Faulty operational procedures and mechanical design. Pump startup procedure called for test return valve to contaminated CST to be open prior to rolling the RCIC turbine. Conceivably the pump could pump itself dry and a water slug would be forced into the void thus created within the pump. The pump design did not restrict the axial movement of each individual stage for excessive loading conditions.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Operational procedures revised to initiate pumps start with discharge valve to condensate storage tank closed. Pump design modified to restrain individual pump stages.

PRELIMINARY

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BWR EVENT ITEM 78

(a) PLANT, MODE, EVENT DATE	Quad Cities 2, shutdown, 8/31/75.
(b) REFERENCE	50265-610 AO 75-36 (9/10/75).
(c) SYSTEM, MECHANICAL FUNCTION	Main feedwater (regulating valve), valve instability.
(d) INITIAL INDICATION	FW vibration alarm, turbine trip, reactor scram, FW system leak.
(e) WATER HAMMER TYPE, DAMAGE	Unknown (vibration). FW low flow drain lines and high pressure heater bypass line broken.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design (component duty cycle mismatch). FW system vibration possibly caused by flow/resonance condition of FW valve actuator piston.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Lines replaced and welds repaired. FW regulation valve operator actuators replaced with a different operational type using a diaphragm instead of actuated piston-cylinder type currently used.

PRELIMINARY

DRAFT

BWR EVENT ITEM 79

(a) PLANT, MODE, EVENT DATE	Vermont Yankee, HPCI turbine preoperation test, 1971.
(b) REFERENCE	50271-66 (11/30/71).
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line check valve), check valve closure.
(d) INITIAL INDICATION	Water hammer.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. No damage indicated.
(f) CAUSE AND BASIS FOR EVENT	Design and procedural deficiencies caused creation of vacuum condition. Vacuum in HPCI turbine exhaust line sucked in water from the suppression pool and caused rapid check valve closure. The condition was abetted by valve orientation (installation in a vertical pipe) and lack of vacuum breakers.
(g) CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	HPCI and RCIC turbine exhaust line swing check valves relocated to horizontal pipe runs. Vacuum breakers (check valves) installed in HPCI and RCIC turbine exhaust lines inside the suppression pool chamber.

PRELIMINARY

DRAFT

BWR EVENT ITEM 80

(a)	PLANT, MODE, EVENT DATE	Vermont Yankee, 88% power (ECCS operation), 6/76.
(b)	REFERENCE	50271 (7/15/76).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (pump, condenser), valve opening/ pump start.
(d)	INITIAL INDICATION	Leakage.
(e)	WATER HAMMER TYPE, DAMAGE	Probably steam-water entrainment. HPCI pump gland seal and condenser head and gasket failure.
(f)	CAUSE AND BASIS FOR EVENT	Operator error and design deficiency. An operator accidentally drained the reference leg of a RV level control instrument. FW regulation valves closed, RV water level decreased to scram set point and the HPCI gland seal and head gasket failed on ECCS start.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Damaged component parts replaced or repaired.

PRELIMINARY

DRAFT

BWR EVENT ITEM 81

(a) PLANT, MODE, EVENT DATE	Vermont Yankee, 99% power, 12/4/79.
(b) REFERENCE	50271 LER 79-032 (1/2/80).
(c) SYSTEM, MECHANICAL FUNCTION	Condenser storage tank, valve opening/closing.
(d) INITIAL INDICATION	Damaged hanger found during support inspection.
(e) WATER HAMMER TYPE, DAMAGE	Unknown. Damage to pipe hanger.
(f) CAUSE AND BASIS FOR EVENT	Not conclusive, but attributed to water hammer developed during full flow surveillance testing of associated safety system.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Hanger replaced with one that allows axial movement plus "other" measures to minimize possibility of damage.

PRELIMINARY

DRAFT

Category II: BWR Non-Water-Hammer Events

BWR EVENT ITEM 82

(a)	PLANT, MODE, EVENT DATE	Browns Ferry 2, power escalation testing, 8/13/74.
(b)	REFERENCE	50260-231 AO-746W (8/23/74).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (injection valves), valve opening.
(d)	INITIAL INDICATION	HPCI injection valves failed to open.
(e)	EVENT TYPE, DAMAGE	Unknown. HPCI injection valve failed in the closed position. Collar broken off stem nut and several bolts securing the valve operator and cap were broken. A second valve was found in a similar condition.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable. Torque and limit switches were not damaged and appear to be set properly. Condition similar to RHR and FW valve vibration.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Necessary repairs, checkout and tests made.

PRELIMINARY

DRAFT

BWR EVENT ITEM 83

(a)	PLANT, MODE, EVENT DATE	Brunswick 1, cold shutdown (leak rate test), 10/23/76.
(b)	REFERENCE	50324-969 RO 76-152 (11/8/76), RO 76-152 Supp. (10/25/76).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line check valve), check valve operation.
(d)	INITIAL INDICATION	Component functional testing.
(e)	EVENT TYPE, DAMAGE	Unknown. Same as Brunswick 2, but disc still in place. See 88.
(f)	CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. Same as Brunswick 2. See 87.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	See 88.

PRELIMINARY

DRAFT

BWR EVENT ITEM 84

(a) PLANT, MODE, EVENT DATE	Brunswick 1, cold shutdown (leak rate test), 5/20/77.
(b) REFERENCE	50325-517 LER 77-35 (6/3/77).
(c) SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line check valve), check valve instability.
(d) INITIAL INDICATION	Discovery of check valve failure.
(e) EVENT TYPE, DAMAGE	Possible steam-bubble collapse. HPCI steamline exhaust check valve failed. Disc hinge fractured.
(f) CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. Flow oscillations in the HPCI exhaust line at low speed, no load conditions resulted in check valve oscillations. Extended periods of low speed HPCI operation were incurred in trouble shooting for startup testing.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Disc hinge redesigned and material changed to increase tensile and impact properties.

PRELIMINARY

DRAFT

BWR EVENT ITEM 85

(a)	PLANT, MODE, EVENT DATE	Brunswick 2 (post critical testing 8% power), 4/13/75.
(b)	REFERENCE	50324-345 AO 75-30 (9/11/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RCIC (turbine exhaust line check valve), check valve operation.
(d)	INITIAL INDICATION	RCIC turbine trip due to high exhaust pressure.
(e)	EVENT TYPE, DAMAGE	Possible steam-bubble collapse. Check valve flapper disconnected and jammed in the line restricting exhaust flow to torus.
(f)	CAUSE AND BASIS FOR EVENT	Component mechanical failure. RCIC turbine tripped on high exhaust pressure.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Necessary repairs and checkout made.

PRELIMINARY

DRAFT

BWR EVENT ITEM 86

(a) PLANT, MODE, EVENT DATE	Brunswick 2, 63% power, 10/75.
(b) REFERENCE	50324-466 LER 75-118, -467 LER 75-119, -469 LER 75-120 (11/25/75).
(c) SYSTEM, MECHANICAL FUNCTION	RCIC (turbine exhaust line check valve), check valve operation.
(d) INITIAL INDICATION	RCIC turbine trip due to high exhaust pressure.
(e) EVENT TYPE, DAMAGE	Possible steam-bubble collapse. Check valve flapper was jammed partially closed restricting RCIC turbine exhaust flow to torus.
(f) CAUSE AND BASIS FOR EVENT	Component mechanical failure. The RCIC turbine tripped repeatedly due to high exhaust pressure.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Necessary repairs and checkout made.

PRELIMINARY

DRAFT

BWR EVENT ITEM 87

(a) PLANT, MODE, EVENT DATE	Brunswick 2, hot standby, 2/76.
(b) REFERENCE	50324-631 LER 76-32, -632 LER 76-33, -633 LER 76-34, -634 LER 76-35, -635 LER 76-36, -636 LER 76-37, (2/27/76).
(c) SYSTEM, MECHANICAL FUNCTION	RCIC (turbine exhaust line check valve), check valve operation.
(d) INITIAL INDICATION	RCIC turbine tripped repeatedly on high exhaust pressure.
(e) EVENT TYPE, DAMAGE	Possible steam-bubble collapse. Check valve flapper stud was broken and flapper was blocking exhaust flow.
(f) CAUSE AND BASIS FOR EVENT	Component mechanical failure. The RCIC turbine tripped repeatedly on high exhaust pressure.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Necessary repairs and checkout made. Engineering effort undertaken to improve valve design.

PRELIMINARY

DRAFT

BWR EVENT ITEM 88

(a)	PLANT, MODE, EVENT DATE	Brunswick 2, cold shutdown (leak rate test) 10/23/76.
(b)	REFERENCE	50324-969 RO-76-152 (11/8/76), RO-76-152 Supp (10/25/76).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line check valve), check valve operation.
(d)	INITIAL INDICATION	Component functional testing.
(e)	EVENT TYPE, DAMAGE	Unknown. HPCI turbine steam line exhaust check valve disc missing and lodged at stop check valve. Broken retaining stud on disc. Similar event, see 83.
(f)	CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. The retaining stud on the disc had broken off allowing disc to be blown downstream. The retaining nut became loose allowing the stud to carry the full weight of the disc.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Valve internals repaired and modified to improve and increase disc load carrying capability.

PRELIMINARY

DRAFT

BWR EVENT ITEM 89

(a)	PLANT, MODE, EVENT DATE	Cooper, 84% power, 1/3/76.
(b)	REFERENCE	50298-514 RO 76-4 (2/2/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, line), N/A.
(d)	INITIAL INDICATION	Damaged pipe restraints.
(e)	EVENT TYPE, DAMAGE	Unknown (vibration). RHR A Hx SW pipe hangers broken (fatigue failure in the bolt holes for support bolt). Broken seismic support and rigid pipe hanger. Similar event, see 90.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (design/service condition deficiency). Vibration. Loading on pipe hangers was close to design value. Insufficient margin for service condition.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Pipe hangers were strengthened by using larger sized straps.

PRELIMINARY

DRAFT

BWR EVENT ITEM 90

(a)	PLANT, MODE, EVENT DATE	Cooper, currently unavailable, 1/12/76.
(b)	REFERENCE	50298-512 RO 76-6 (2/2/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, line), N/A.
(d)	INITIAL INDICATION	Damaged pipe restraints.
(e)	EVENT TYPE, DAMAGE	Unknown (vibration). One RHR B Hx SW pipe hanger failed at bolt hole. Prior failure of same type of hanger on other loop. See 89.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (design/service condition deficiency). Vibration. See 89.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	See 89.

PRELIMINARY

DRAFT

165

BWR EVENT ITEM 91

(a)	PLANT, MODE, EVENT DATE	Dresden 3, power escalation testing (pressure regulator tests), 4/7/71.
(b)	REFERENCE	50249 (5/4/73).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (turbine bypass lines, valves), valve instability.
(d)	INITIAL INDICATION	Bypass valve instability.
(e)	EVENT TYPE, DAMAGE	Potential steam hammer. Valve instability and cycling caused pulsating, oscillatory steam flow (rated flow delivery impaired). No damage incurred but same potential exists as Peach Bottom 3 (Item 65).
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (improper EHC design). Bypass valve instability and oscillations caused by erroneous EHC pressure sensor signals. The pressure sensing lines were too long and vertical loops and bends (in the sensing lines) presumably caused voids to occur which produced false pressure amplification signals at the sensor.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Sensor hydraulic line shortened and abrupt elevation changes eliminated.

PRELIMINARY

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BWR EVENT ITEM 92

(a)	PLANT, MODE, EVENT DATE	Dresden 3, 75% power, 9/12/76.
(b)	REFERENCE	50249-902 RO 76-15 (10/12/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (turbine moisture separator, drain line), valve opening/closing.
(d)	INITIAL INDICATION	Steam leak at piping elbow found during routine inspection.
(e)	EVENT TYPE, DAMAGE	Potential steam hammer. Pipe crack at welded support. Similar event, see 93.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (insufficient line restraints). A surge in process steam resulted in failure of a support bolt on a rigid pipe hanger that increased loading on a downstream support welded to the process line.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Additional line restraints were added.

PRELIMINARY

DRAFT

BWR EVENT ITEM 93

(a) PLANT, MODE, EVENT DATE	Dresden 3, shutdown, 6/7/77.
(b) REFERENCE	50249-1123 RO 77-20 (7/1/77).
(c) SYSTEM, MECHANICAL FUNCTION	Main steam (turbine moisture separator, drain line), valve opening/closing.
(d) INITIAL INDICATION	Leakage at welded joint.
(e) EVENT TYPE, DAMAGE	Potential for steam hammer. Pin hole leak in weld crack located at junction of a rigid vertical standpipe support. Failure of same weld on another loop line on 9/76. See 92.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design (insufficient line restraints). Repeated surges of process steam in line caused vibration (fatigue).
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Additional line restraints were added.

PRELIMINARY

DRAFT

BWR EVENT ITEM 94

(a)	PLANT, MODE, EVENT DATE	Millstone 1, 97% power, 12/26/74.
(b)	REFERENCE	50245-484 (2/26/75); 50245 UE 75-1 (1/23/75).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (regulating valve), valve closure.
(d)	INITIAL INDICATION	Decrease FW flow and sudden decrease in reactor water level.
(e)	EVENT TYPE, DAMAGE	Unknown. FW regulation valve stem sheared.
(f)	CAUSE AND BASIS FOR EVENT	Component failure. Broken FW regulating valve stem allowed full closure of valve plug causing a RV water level decrease and initiating a scram.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	New parts installed and system returned to service.

PRELIMINARY

DRAFT

BWR EVENT ITEM 95

(a)	PLANT, MODE, EVENT DATE	Monticello 1, startup testing, 1971.
(b)	REFERENCE	W. Hill: IEEE Transactions on Nuclear Science NS-19(1), pp. 846-850, Feb. 1972; 50263-115 (2/28/72).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (turbine moisture separator, drain tank lines and control valves), system vibration/valve instability.
(d)	INITIAL INDICATION	Excessive turbine trips and reactor scrams related to moisture separator/ drain tank liquid levels.
(e)	EVENT TYPE, DAMAGE	Steam flashing leading to column separation, steam-bubble collapse, or steam condensation/slug impact. Valve instability. Failure of drain control valves to remain fully open due to oscillations and inadvertent closures. Piping vibration due to flashing.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (deficient EHC and mechanical systems design). The extended periods of flashing in vent and drain lines raised the pressure in the drain tank higher than that in the separator thus forcing water back into the separator while water was still being collected. The high water level in the separator eventually reached the turbine trip point. The drain tank level sensors responded to the water level decrease in the drain tank and closed the drain valve. The vent line was not large enough to relieve the pressure differential across the tanks. Inadequate instrumentation and controls were a contributing factor.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The moisture separator level control system was modified to increase responsiveness to sudden plant power level transients. Turbine trips were prevented by installing a differential pressure switch to activate dump valves on high level indication.

BWR EVENT ITEM 96

(a)	PLANT, MODE, EVENT DATE	Monticello, refueling shutdown (containment leak check), 4/30/74.
(b)	REFERENCE	50263 AO 74-05 (5/20/74).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line isolation valve), valve instability.
(d)	INITIAL INDICATION	Valve leakage.
(e)	EVENT TYPE, DAMAGE	Unknown. Excessive valve leakage. Valve seat loose.
(f)	CAUSE AND BASIS FOR EVENT	Design deficiency (component/service condition). Vibration and oscillation during HPCI operation possibly induced by exhaust flow fluctuations.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Valve seat and disc lapped. Disc reseated and tightened properly.

PRELIMINARY

DRAFT

BWR EVENT ITEM 97

(a)	PLANT, MODE, EVENT DATE	Monticello, shutdown (HPCI leak rate test), 1/16/75.
(b)	REFERENCE	50263-452 AO 75-1 (2/7/75).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line check valve), check valve operation.
(d)	INITIAL INDICATION	Leakage in excess of acceptance criteria.
(e)	EVENT TYPE, DAMAGE	Unknown. Excessive leakage caused by improper disc seating due to bent hinge pin.
(f)	CAUSE AND BASIS FOR EVENT	Mechanical failure. Similar occurrences prior.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Disc pin straightened.

PRELIMINARY

DRAFT

BWR EVENT ITEM 98

(a)	PLANT, MODE, EVENT DATE	Monticello 1, 100% power, 5/15/76.
(b)	REFERENCE	50263-667 RO 76-04 (6/11/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (turbine moisture separator, drain tanks, lines, valves), N/A.
(d)	INITIAL INDICATION	Crack in drain line tee.
(e)	EVENT TYPE, DAMAGE	Potential steam-bubble collapse due to flashing in drain tank. A forged butt weld tee cracked at the vent line branch connection.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (fatigue environment). Cyclic vibration (fatigue).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Vent line removed and crack repaired.

PRELIMINARY

DRAFT

BWR EVENT ITEM 99

(a)	PLANT, MODE, EVENT DATE	Monticello, 100% power (RCIC test), 8/2/76.
(b)	REFERENCE	50263, 1976 Annual Operating Report, 1977; 50263-715 RO 76-11 (9/1/76).
(c)	SYSTEM, MECHANICAL FUNCTION	RCIC (turbine exhaust line), valve opening.
(d)	INITIAL INDICATION	RCIC turbine tripped on overspeed.
(e)	EVENT TYPE, DAMAGE	None observed. Potential for flow-into-voided-line or steam-bubble collapse. RCIC turbine tripped on over-speed on two occasions.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design and maintenance/repair procedures. Valve leakage from FW system created a steam void in discharge piping. The condenser cooling water supply valve was open for repair and allowed warm water to circulate through the pump.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Cooling water supply valve repaired and closed. An additional discharge valve was closed to prevent steam void formation.

PRELIMINARY

DRAFT

BWR EVENT ITEM 100

(a)	PLANT, MODE, EVENT DATE	Monticello, refueling shutdown, 10/4/77.
(b)	REFERENCE	50263 RO 77-24-5 (11/18/77).
(c)	SYSTEM, MECHANICAL FUNCTION	RCIC (turbine exhaust line check valve), check valve operation.
(d)	INITIAL INDICATION	Excessive check valve leakage.
(e)	EVENT TYPE, DAMAGE	Unknown. Bent disc washer resulted in improper disc seating causing excessive leakage.
(f)	CAUSE AND BASIS FOR EVENT	Component mechanical failure. A bent disc washer caused improper seating of disc.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Necessary repairs and checkout made. A valve modification was made to prevent disc from striking valve body.

PRELIMINARY

DRAFT

BWR EVENT ITEM 101

(a)	PLANT, MODE, EVENT DATE	Nine Mile Pt. 1, power escalation testing, 1969-1971.
(b)	REFERENCE	50220-69 (8/26/71), -83 (2/25/72), -99 (7/5/72), -55 (12/9/70), -39 (4/20/70).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (turbine moisture separator, reheater, drain tank), system vibration/valve instability.
(d)	INITIAL INDICATION	Excessive turbine trips and reactor scrams.
(e)	EVENT TYPE, DAMAGE	Steam flashing leading to column separation, steam-bubble collapse, condensation/slug impact. Excessive turbine trips due to high water level in moisture separator. Reheater drainage problems.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (deficient EHC and mechanical system design). Inadequate drain piping/vents and moisture separator baffles. Lack of adequate instrumentation and controls. Two phase flow in drain lines.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Incorporation of new instruments and controls and water injection system to assure single phase flow in drain lines. New drain piping, tanks, vents and enlarged drain piping installed. Moisture separator baffles replaced by Herzog Cones for increased structural integrity and improved flow.

PRELIMINARY

DRAFT

BWR EVENT ITEM 102

(a)	PLANT, MODE, EVENT DATE	Nine Mile Pt. 1, 300 MWe, 12/74.
(b)	REFERENCE	50220 (2/27/75).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (control valve, pump), valve instability.
(d)	INITIAL INDICATION	FW system vibrations.
(e)	EVENT TYPE, DAMAGE	Unidentified (vibration). FW control valve stem broken. Pump cavitation with impeller erosion.
(f)	CAUSE AND BASIS FOR EVENT	Component failure. Broken FW control valve stem caused vibrations in FW system. Increased plugging of outlet laterals of condensate demineralizers produced a high pressure differential across the demineralizers resulting in low FW pump suction pressure and induced cavitation erosion on impeller.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	New parts installed and system returned to service.

PRELIMINARY

DRAFT

BWR EVENT ITEM 103

(a) PLANT, MODE, EVENT DATE	Oyster Creek 1, operational, 12/11/71.
(b) REFERENCE	50219-160 (1/12/72), -170 (2/28/72).
(c) SYSTEM, MECHANICAL FUNCTION	RHR isolation condenser (vent line), currently unavailable.
(d) INITIAL INDICATION	Line failure.
(e) EVENT TYPE, DAMAGE	Unknown. The isolation condenser vent line failed at the main steam line connection downstream of the main steam and condenser vent line isolation valves.
(f) CAUSE AND BASIS FOR EVENT	Installation and maintenance deficiency. Excessive motion of the vent line. The first pipe hanger upstream of the failure was found disconnected permitting the line to move freely.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Line repaired and restraint connected properly. Additional pipe support installed.

PRELIMINARY

DRAFT

BWR EVENT ITEM 104

(a)	PLANT, MODE, EVENT DATE	Oyster Creek 1, startup testing, 1972.
(b)	REFERENCE	50219-172 (3/2/72), -213 (8/31/72)
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (turbine moisture separator, reheater, drain tank), system vibration/ valve instability.
(d)	INITIAL INDICATION	Excessive turbine trips and reactor scrams.
(e)	EVENT TYPE, DAMAGE	Steam flashing leading to column separation, steam-bubble collapse, or steam condensation/slug impact, valve instability. Excessive turbine trips due to high water level in moisture separator or drain tanks.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (deficient EHC and mechanical system design). Pressure fluctuations in flash (drain) tank and undersized vent and drain lines. Failure to identify the need for check valves in flash tank riser.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	See 101.

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BWR EVENT ITEM 105

(a) PLANT, MODE, EVENT DATE	Peach Bottom 2, 100% power, 11/22/74.
(b) REFERENCE	50277-339 RO 74-53 (12/2/74).
(c) SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, discharge valves), valve instability.
(d) INITIAL INDICATION	Inspection revealed damaged component.
(e) EVENT TYPE, DAMAGE	Unknown (vibration). All cap screws that attach the motor operators to the valve yokes were loose.
(f) CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. Excessive vibration. Locking devices were not used on screws when initially installed and possibly not torqued tightly enough.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Cap screws replaced, torqued to proper values and sealed with antivibration compound. Six other valves will be inspected for similar defects.

PRELIMINARY

DRAFT

BWR EVENT ITEM 106

(a) PLANT, MODE, EVENT DATE	Peach Bottom 2, 100% power, 4/1/75.
(b) REFERENCE	50277-395 RO 75-24 (4/11/75).
(c) SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, discharge valves), valve instability.
(d) INITIAL INDICATION	Valve closure indication received but flow continuation still being maintained.
(e) EVENT TYPE, DAMAGE	Unknown (vibration). Valve disc lock-nut welds fractured letting the nut back off and causing the valve stem to vibrate and incur fatigue failure. The valve seat backed out and was loose in the valve body. The threads in the valve body were completely destroyed.
(f) CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. High valve vibration due to throttling mode of valve.
(g) CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Damaged parts repaired or replaced. Modification of valve internals to improve serviceability under vibratory environment was done as an interim solution. Final modification replaced control valves with three position orifice plates for flow control.

PRELIMINARY

DRAFT

BWR EVENT ITEM 107

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 2, shutdown cooling, 6/21/75.
(b)	REFERENCE	50277-451 AO 75-49 (7/1/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (shutdown cooling lines, valves), valve instability.
(d)	INITIAL INDICATION	Valve limit light signal indicated abnormal condition.
(e)	EVENT TYPE, DAMAGE	Unknown (vibration). Valve motor failed electrically from excessive loading due to jammed valve stem. Valve stem key fell out permitting stem rotation which upset valve operator calibration.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design/service condition. Excessive valve vibration when throttling during shutdown cooling mode.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Repairs and NOT undertaken. Valve locked in full open position.

PRELIMINARY

DRAFT

BWR EVENT ITEM 108

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 2, shutdown cooling, 11/9 and 11/13/75.
(b)	REFERENCE	50277-556 (11/21/75), -560 (11/19/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (shutdown cooling lines, valves), valve instability.
(d)	INITIAL INDICATION	Line and valve water leaks.
(e)	EVENT TYPE, DAMAGE	Unknown (vibration). Water leakage in vicinity of motor operated valve from a crack in the vent line. Cracked socket weld where the process line connects to the header. Later leakage and a crack in the body of the RHR relief valve was observed.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design/service condition. Normal throttling of the motor operated valve during the shutdown cooling mode induced vibrations. Probably insufficient valve support.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Additional supports to be added in vicinity of valve.

PRELIMINARY

DRAFT

BWR EVENT ITEM 109

(a) PLANT, MODE, EVENT DATE	Peach Bottom 3, at power, 11/22/74.
(b) REFERENCE	50278-232 RO 74-28 (12/2/74).
(c) SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, discharge valves), valve instability.
(d) INITIAL INDICATION	Inspection revealed damaged component.
(e) EVENT TYPE, DAMAGE	Unknown (vibration). Half of the cap screws that attach motor operator to valve yokes were broken.
(f) CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. Excessive vibration. Locking devices were not used on screws when initially installed and possibly not torqued tightly enough.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	See 105.

PRELIMINARY

DRAFT

BWR EVENT ITEM 110

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 3, shutdown, 12/8/74.
(b)	REFERENCE	50278-248 (12/9/74), -249 AO 74-35 (12/18/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, injection valve), valve instability.
(d)	INITIAL INDICATION	Component failure.
(e)	EVENT TYPE, DAMAGE	Unknown (vibration). Motor support vibrated loose allowing motor to separate from operator. The cap screws had vibrated loose.
(f)	CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. Vibration.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Damaged components replaced and necessary repairs made. Cap screws installed with lock washers to inhibit vibrational effects.

PRELIMINARY

DRAFT

BWR EVENT ITEM 111

(a)	PLANT, MODE, EVENT DATE	Peach Bottom 3, shutdown, 12/10/74.
(b)	REFERENCE	50278-250 AO 74-36 (12/24/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, discharge valve), valve instability.
(d)	INITIAL INDICATION	Component damage.
(e)	EVENT TYPE, DAMAGE	Unknown (vibration). Valve cap screws broken and valve stem bent due to vibration.
(f)	CAUSE AND BASIS FOR EVENT	Valve instability. Vibration.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Damaged parts repaired or replaced. Valve operator installed with high tensile cap screws.

PRELIMINARY

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BWR EVENT ITEM 112

(a) PLANT, MODE, EVENT DATE	Peach Bottom 3, shutdown, 1/10/75.
(b) REFERENCE	50278-265 AO 75-5 (1/20/75).
(c) SYSTEM, MECHANICAL FUNCTION	Service water (RHR Hx, discharge valve), valve instability.
(d) INITIAL INDICATION	Component failure.
(e) EVENT TYPE, DAMAGE	Unknown (vibration). Valve support flange failed (an integral part of the motor end bell casing used to bolt the motor to valve operator gear case). Valve packing hammered out of position by valve stem vibration. Fatigue failure of motor end bell case.
(f) CAUSE AND BASIS FOR EVENT	Component design/service condition deficiency. High vibration due to valve throttling.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Damaged parts replaced.

PRELIMINARY

DRAFT

BWR EVENT ITEM 113

(a)	PLANT, MODE, EVENT DATE	Pilgrim 1, shutdown cooling, 9/75.
(b)	REFERENCE	50293 AO 75-25 (10/13/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR (shutdown cooling lines, valve), currently unavailable.
(d)	INITIAL INDICATION	Inoperable outboard isolation valve.
(e)	EVENT TYPE, DAMAGE	Unknown. Outboard isolation valve became inoperable. The valve operator motor separated from the gear box.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design/service condition. A system transient occurred while placing the RHR in the shutdown cooling mode. The valve motor mounting arrangement may have been a contributing cause.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Necessary repairs made.

PRELIMINARY

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BWR EVENT ITEM 114

(a)	PLANT, MODE, EVENT DATE	Pilgrim 1, shutdown (HPCI pump test), 4/8/78.
(b)	REFERENCE	50293-1006 LER 78-13/3L (4/28/78).
(c)	SYSTEM, MECHANICAL FUNCTION	HPCI (turbine exhaust line rupture disc), currently unavailable.
(d)	INITIAL INDICATION	Annunciation of steam leak detection system for HPCI torus area high temperature.
(e)	EVENT TYPE, DAMAGE	Unknown. HPCI turbine exhaust line rupture disc ruptured.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (insufficient margin to compensate for vacuum or pulsating flow conditions). Pulsating pressure changes in the exhaust line accelerated the work hardening process by flexing the diaphragm thus contributing to the failures.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Backing plates designed to minimize rupture disc flexure mode.

PRELIMINARY

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BWR EVENT ITEM 117

(a)	PLANT, MODE, EVENT DATE	Quad Cities 1, 24% power, 9/21/77.
(b)	REFERENCE	NRC Grey Book (9/77); Operating Units Status Report NUREG Series; 50254, RO-77-39 (10/4/77).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (turbine moisture separator, drain tank, level controller), system vibration/valve instability.
(d)	INITIAL INDICATION	Turbine trip.
(e)	EVENT TYPE, DAMAGE	Potential steam-bubble collapse due to flashing in drain tank and drain lines. Moisture separator drain tank level controller vibrated loose severing instrument air lines. Broken drain line pipe hangers.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design (insufficient line restraints). Loss of level controller caused a high water level in the drain tank and moisture separator initiating turbine trip with severe vibration.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Failed component replaced and relocated to minimize vibratory environment effect.

PRELIMINARY

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BWR EVENT ITEM 118

(a)	PLANT, MODE, EVENT DATE	Quad Cities 1, shutdown (core spray test), 9/77.
(b)	REFERENCE	50254 RO 77-75/3L (10/28/77).
(c)	SYSTEM, MECHANICAL FUNCTION	Core spray (valve), pump start/valve opening.
(d)	INITIAL INDICATION	Core spray minimum flow valve failed to close when pump was shut down. Outboard pump discharge valve inoperative.
(e)	EVENT TYPE, DAMAGE	Potential for flow-into-voided-line. During repairs to the minimum flow valve, the outboard pump discharge valve was found inoperable. An inspection revealed a broken limit switch with oversized lugs for each discharge valve, a cracked valve yoke and motor housing and fractured housing support bolts on the outboard valve. The presence of rust in the yoke crack was indicative that the crack was present for some time.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate inspection techniques. The nature of the damage is such that a contributing cause could be related to a previous water hammer event.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Necessary repairs, checkout and testing conducted.

PRELIMINARY

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SGWH EVENT ITEM 7

(a) PLANT, MODE, EVENT DATE	Millstone 2, preoperational testing, 1975.
(b) REFERENCE	D. C. Switzer, Northeast Nuclear Energy Co., letter to R. Reid, NRC, Subject--"Millstone Nuclear Power Station, Unit No. 2, Feedwater System Piping," 8/22/79.
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d) INITIAL INDICATION	Water hammer heard by plant personnel.
(e) DAMAGE	None.
(f) CAUSE AND BASIS FOR EVENT	Currently unavailable. A steam generator water hammer occurred in an unidentified feedwater line.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	In 1977, "J" shaped discharge tubes were installed atop all steam generator feedrings and the bottom discharge holes were plugged. The feedwater piping geometry in all loops was altered to reduce the length of horizontal piping adjacent to each steam generator to less than five feet. Testing was performed during hot standby conditions to verify the adequacy of these modifications to preclude further steam generator water hammer. Administrative auxiliary feedwater flow limitations were implemented during periods of main feedwater deactivation.

PRELIMINARY

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SGWH EVENT ITEM 8

(a) PLANT, MODE, EVENT DATE	Palisades, hot testing, spring 1981.
(b) REFERENCE	S. D. MacKay, NRC, Draft Report "Safety Evaluation in Support of a License Amendment to Permit Performance of a Water Hammer Test at the Palisades Plant," 4/16/80.
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d) INITIAL INDICATION	Currently unavailable.
(e) DAMAGE	Currently unavailable.
(f) CAUSE AND BASIS FOR EVENT	A steam generator water hammer occurred during water hammer testing. The purpose of the testing was to show that auxiliary feedwater flow rates of greater than 150 gpm per steam generator would not result in water hammer. Had the testing been successful, administrative guidelines may have been changed to permit higher auxiliary feedwater flow rates.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Currently unavailable.

PRELIMINARY

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SGWH EVENT ITEM 9

(a) PLANT, MODE, EVENT DATE	San Onofre 2, preoperational hot testing, 3/30/81.
(b) REFERENCE	50361 PNO-V 81-40 (7/16/81).
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d) INITIAL INDICATION	Damage found during inspection.
(e) DAMAGE	The event caused collapse and twisting of the "J" tube equipped feedring. Feedring mounts were broken or damaged at nearly all locations.
(f) CAUSE AND BASIS FOR EVENT	Damage to the feedring of an unspecified steam generator was found during a July 14, 1981 inspection and was attributed to steam generator water hammer. The event occurred undetected during a March 30, 1981 water hammer test. The auxiliary flow rate to the damaged steam generator was about 1200 gpm at the time of the event and the feedring had been uncovered for two hours.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The event and subsequent damage is currently under investigation by the licensee.

PRELIMINARY

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SGWH EVENT ITEM 10

(a)	PLANT, MODE, EVENT DATE	Surry 1, hot shutdown, 10/1/72.
(b)	REFERENCE	W. R. Butler, NRC, memo to A. Schwencer, NRC, Subject--"Water Hammer in Feedwater Systems," 5/26/78.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d)	INITIAL INDICATION	Loud noise heard plus safety injection signal.
(e)	DAMAGE	The following mechanical effects were noted on the No. A feedwater line: (a) an observed displacement of 7 to 10 inches clockwise around the containment building wall and 4 to 5 inches radially inward toward the containment centerline; (b) all seven feedwater linespring hanger support assemblies were displaced to varying degrees; (c) all seven feedwater line hydraulic shock suppressors failed; (d) damage to the feedwater line check valve consisting of bolting flange distortion, breaking off of a 180° segment of a metal sheathed asbestos gasket, and a leaking crack in the valve body.
(f)	CAUSE AND BASIS FOR EVENT	While the unit was being maintained in hot shutdown conditions, a steam generator water hammer in the No. A feedwater line was indicated by a loud noise followed by a safety injection signal.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	All damaged components were repaired or replaced and a computer analysis was performed to verify the integrity of the No. A feedwater line. In units 1 and 2, "J" shaped discharge tubes were installed atop all feedrings and the bottom discharge holes were plugged and the geometry of all feedwater lines adjacent to the steam generators was altered to reduce the horizontal runs of piping.

PRELIMINARY

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SGWH EVENT ITEM 11

(a)	PLANT, MODE, EVENT DATE	Turkey Point 3, currently unavailable, 1973.
(b)	REFERENCE	Robert E. Uhrig Florida Power and Light, letter to George Lear, NRC, Subject--"Water Hammer in PWR Feedwater Systems," 7/3/75.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d)	INITIAL INDICATION	Damage found during an in-containment inspection of the feedwater system.
(e)	DAMAGE	The damage consisted of body-to-bonnet bolt elongation and leakage on the No. 3 feedwater piping check valve.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable. A steam generator water hammer occurred in the No. B feedwater piping.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The damaged components were repaired or replaced.

PRELIMINARY

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SGWH EVENT ITEM 12

(a)	PLANT, MODE, EVENT DATE	Turkey Point 4, currently unavailable, prior to 1974.
(b)	REFERENCE	See 11.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d)	INITIAL INDICATION	Damage found during an in-containment inspection of the feedwater system.
(e)	DAMAGE	The damage consisted of body-to-bonnet bolt elongation and leakage on the No. 3 feedwater piping check valve.
(f)	CAUSE AND BASIS FOR EVENT	Currently unavailable. A steam generator water hammer occurred in the No. 8 feedwater piping.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The damaged components were repaired or replaced.

PRELIMINARY

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SGWH EVENT ITEM 13

(a) PLANT, MODE, EVENT DATE	Turkey Point 4, currently unavailable, during first half of 1974.
(b) REFERENCE	Nuclear Power Experience, Nuclear Power Experience, Inc., PWR Vol. VI.E.69. Robert E. Uhrig, Florida Power & Light, letter to George Lear, NRC, Subject--"Water Hammer in PWR Feedwater Systems," 7/3/75.
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d) INITIAL INDICATION	Damage found during an in-containment inspection of the feedwater system.
(e) DAMAGE	The damage found consisted of: (a) some expansion bolts for two hyraulic pipe restraints had been pulled about one inch out of a concrete wall; (b) two spring hanger mounting plates had been deformed; (c) a slight plastic deformation in a 90° elbow located in the piping leading to the steam generator feedwater nozzle.
(f) CAUSE AND BASIS FOR EVENT	Currently unavailable. A steam generator water hammer occurred in the No. A feedwater piping.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	All damaged components were repaired or replaced. The feedwater piping geometry of all loops in Unit No. 3 and 4 was altered to reduce the horizontal run of piping adjacent to each steam generator to less than five feet. Administrative controls were implemented to limit auxiliary feedwater to the minimum required for decay heat removal during periods of main feedwater deactivation.

PRELIMINARY

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SGWH EVENT ITEM 14

(a) PLANT, MODE, EVENT DATA — Yankee Rowe, currently unavailable, between unit startup (1/61) and 1965.

(b) REFERENCE — J. L. French, Yankee Atomic Electric Company (YAEC), letter to P. A. Purple, NRC, Subject--"Response to May 13, 1975 NRC letter on Steam Generator Water Hammer," 7/18/75.

Informal conversations, Richard Aron (YAEC), and D. D. Christensen, EG&G Idaho, September-October 1979.

(c) SYSTEM, MECHANICAL FUNCTION — Steam generator, currently unavailable.

(d) INITIAL INDICATION — Two feedwater and steam generator inspections, one each in 1964 and 1965, revealed damage in the steam generator attributable to steam generator water hammer. The events were unobserved by plant personnel.

(e) DAMAGE — The damage found during the 1964 inspection consisted of weld failures causing displacement of the feedring inlet cover plate in three (unspecified) of the four steam generators. The damage found in 1965 consisted of broken feedring supports and weld failures on the feedring mixing tees of an unspecified number of steam generators. A maximum feedring displacement of two inches was found in one of the steam generators.

(f) CAUSE AND BASIS FOR EVENT — Currently unavailable. Periodic and undocumented water hammer events of a minor and undamaging nature occurred from January 1961 through 1963.

(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION — Subsequent to the 1964 and 1965 inspections, necessary repairs or replacement of damaged components were performed prior to resumption of reactor operation. In 1964, the feedring cover plates were resecured with U-bolts and, in 1965, new feedring supports were installed and shoe pads were added between the feedrings and the steam generator shell inner contact points. During the 1966 refueling outage, the following modifications

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SGWH EVENT ITEM 14 (continued)

(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION (continued)

were made to each of the main feedwater lines: (a) the geometry of each line was altered to eliminate the horizontal run of piping adjacent to each steam generator; (b) a steam line was installed to supply steam to the high pressure feedwater heater from the plant auxiliary boilers to provide feedwater preheating during reactor startup periods; (c) low flow bypass lines bypassing the main feedwater regulating valves were installed in each feedwater line in conjunction with low flow bypass valves for more precise feedwater flow control during startup and low power operating conditions.

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SGWH EVENT ITEM 15

(a)	PLANT, MODE, EVENT DATE	Zion 1, cold shutdown, 9/26/76.
(b)	REFERENCE	T. Printz, Commonwealth Edison, informal information transmittals to D. D. Christensen, EG&G Idaho, 2/79-3/79.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, valve opening.
(d)	INITIAL INDICATION	Piping support damage found.
(e)	DAMAGE	The damage found on September 25, 1976 consisted of eight broken hangers on an unspecified number of feedwater lines. No additional damage was attributed to the September 26, 1976 event.
(f)	CAUSE AND BASIS FOR EVENT	The No. 1D feedwater loop was unisolated and brought into service to bring the unit to cold shutdown following discovery of feedwater system damage on September 25, 1976. A steam generator water hammer occurred in the No. 1D feedwater line causing a safety injection signal.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	All damaged hangers were either repaired or replaced prior to continued operation.

PRELIMINARY

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SGWH EVENT ITEM 23

(a) PLANT, MODE, EVENT DATE	Zion 2, hot shutdown, 8/29/74.
(b) REFERENCE	G. J. Pliml, Commonwealth Edison, letter to R. A. Purple, NRC, Subject--"Response to May 13, 1975 NRC letter on Steam Generator Water Hammer," 7/17/75.
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, pump stop.
(d) INITIAL INDICATION	Safety injection signal.
(e) DAMAGE	The event resulted in two broken snubbers on the No. 2C feedwater line.
(f) CAUSE AND BASIS FOR EVENT	The No. 2B reactor coolant pump tripped with the unit at 30% power, resulting in a reactor trip from a low No. 2B steam generator water level signal. A steam generator water hammer in the No. 2C feedwater piping mechanically induced a No. 2D steam generator low pressure signal, resulting in a safety injection signal. A steam generator water hammer in the Zion units typically results in a safety injection signal due to mechanical vibration of feedwater line mounted steam pressure transmitters falsely sensing high steam line differential pressure between loops A/C and loops B/D.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The snubbers were repaired prior to unit startup.

PRELIMINARY

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SGWH EVENT ITEM 24

(a)	PLANT, MODE, EVENT DATE	Zion 2, hot shutdown, 12/30/74.
(b)	REFERENCE	See Item 23.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, valve closure.
(d)	INITIAL INDICATION	Safety injection signal.
(e)	DAMAGE	None.
(f)	CAUSE AND BASIS FOR EVENT	The closure of the No. 2C feedwater regulating valve due to a severed electrical connection caused a reactor trip from 82% power. A steam generator water hammer in the No. 2B or 2D feedwater piping resulted in a safety injection signal (see 23).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	None.

PRELIMINARY

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SGWH EVENT ITEM 25

(a)	PLANT, MODE, EVENT DATE	Zion 2, hot shutdown, 5/25/76.
(b)	REFERENCE	J. S. Bitel, Commonwealth Edison, letter to J. G. Heppler, NRC, Subject--"Description of the May 25, 1976 Zion Unit No. 1 Safety Injection Event," 8/12/76.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, pump/valve operation.
(d)	INITIAL INDICATION	Water hammer heard by control room personnel.
(e)	DAMAGE	None.
(f)	CAUSE AND BASIS FOR EVENT	The No. 2B feedwater pump tripped, causing a reactor trip from 85% power. While auxiliary feedwater was being added to the steam generators, a steam generator water hammer occurred in the No. 2C feedwater line causing a safety injection signal (see 23).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Administrative auxiliary feedwater flow limitations (105 gpm per steam generator) were implemented during periods of main feedwater deactivation.

PRELIMINARY

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SGWH EVENT ITEM 26

(a) PLANT, MODE, EVENT DATE	Zion 2, hot shutdown, 6/20/76.
(b) REFERENCE	J. S. Bitel, Commonwealth Edison, letter to J. G. Heppler, NRC, Subject--"Description of the June 20, 1976 Zion Unit No. 2 Safety Injection Event," 8/13/76.
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, pump/valve operation.
(d) INITIAL INDICATION	Safety injection signal.
(e) DAMAGE	None.
(f) CAUSE AND BASIS FOR EVENT	While auxiliary feedwater was being added to the steam generator after a planned turbine trip, a steam generator water hammer occurred in the No. 2C feedwater line, causing a safety injection signal (see 23).
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Administrative auxiliary feedwater flow limitations (50 gpm per steam generator) were implemented during periods of main feedwater deactivation in Unit No. 2. An analysis of the feedwater systems was performed using a hypothesized water hammer forcing function to verify that the structural integrity of the feedwater piping would be maintained under future hammer conditions. Accelerometers were installed on the Unit No. 2 feedwater lines to monitor and gather data of physical effects of future steam generator water hammer events.

PRELIMINARY

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SGWH EVENT ITEM 27

(a)	PLANT, MODE, EVENT DATE	Zion 2, hot shutdown, 9/3/80.
(b)	REFERENCE	50304 RO 80-26.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d)	INITIAL INDICATION	Safety injection signal.
(e)	DAMAGE	None.
(f)	CAUSE AND BASIS FOR EVENT	Following a reactor scram due to a low level in the No. 2C steam generator, a steam generator water hammer occurred in the No. 2B feedwater line, resulting in a safety injection signal (see 23).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Installation of "J" shaped top discharge tubes and the plugging of all bottom discharge feedrings will be performed for the remaining unaltered steam generators (Nos. 2A and 2B) of Unit No. 2 during the September 1981 outage. The modifications to the No. B steam generator of Unit No. 1 will be performed during the February 1982 outage. The identical modifications were made to the No. 1A and 1D steam generators during the February 1981 outage. The February 1982 work will mark completion of the Zion plant steam generator modifications.

PRELIMINARY

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BWR EVENT ITEM 123

(a)	PLANT, MODE, EVENT DATE	Quad Cities 2, refueling (testing), 9/76.
(b)	REFERENCE	Currently unavailable.
(c)	SYSTEM, MECHANICAL FUNCTION	RCIC (turbine exhaust line, check valve), check valve operation.
(d)	INITIAL INDICATION	Excessive leakage of the RCIC turbine exhaust check valve.
(e)	EVENT TYPE, DAMAGE	Possible steam-bubble collapse. Excessive leakage past RCIC turbine exhaust check valve. Valve flapper assembly broken.
(f)	CAUSE AND BASIS FOR EVENT	Mechanical failure. Corresponding valve failed due to a broken disc on Unit 1. See 116.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Necessary repairs and checkout made.

PRELIMINARY

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BWR EVENT ITEM 124

(a)	PLANT, MODE, EVENT DATE	Vermont Yankee, unknown, 7/20/73.
(b)	REFERENCE	50271 (2/28/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (regulating valve), valve closure.
(d)	INITIAL INDICATION	FW system vibrations and minor leakage.
(e)	EVENT TYPE, DAMAGE	Unknown (vibration). Severe FW system vibrations (FW regulating valve) with restricted plant power output level. Three separate occurrences.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate maintenance and repair practices or procedures. Oversize FW valve trim induced flow instability.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	New parts installed and system returned to service.

PRELIMINARY

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APPENDIX B

PWR NON-STEAM-GENERATOR EVENTS

PRELIMINARY

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Category I: Reported PWR Water Hammer Events

PWR EVENT ITEM 1

(a)	PLANT, MODE, EVENT DATE	Arkansas 1, power escalation testing (pipe shock test), 9/12/74.
(b)	REFERENCE	50313-217 UE 74-3 (10/11/74).
(c)	SYSTEM, MECHANICAL FUNCTION	RCS (pressurizer relief valve and discharge piping), valve opening.
(d)	INITIAL INDICATION	Excessive pipe movement readings.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Several hanger rods were bent.
(f)	CAUSE AND BASIS FOR EVENT	Probably inadequate design. Cause not given. Probably same as 33, if pressurizer relief valves are of similar design.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Two additional hydraulic snubbers were added to each of the discharge lines.

PRELIMINARY

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PWR EVENT ITEM 2

(a)	PLANT, MODE, EVENT DATE	Arkansas 1, unknown, 3/75.
(b)	REFERENCE	Arkansas Power and Light Seminar, B&W, W. Cavenaugh (4/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Condenser (inlet piping tubes), unidentified (probably valve opening).
(d)	INITIAL INDICATION	Condenser damage noted during routine inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Failed turbine bypass spargers, impingement plates on turbine-to-condenser expansion joint, and tie rods and expansion joints.
(f)	CAUSE AND BASIS FOR EVENT	Unidentified (probably inadequate valve design or inadequate piping design). Could be water piston effect from dump valves. See Beaver Valley Events 4 and 5.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Replacement carbon steel spargers with longer ligaments between holes and tie rods of Cr-Mo steel were used. Stronger welds, more tabs on impingement plates and additional pipe restraints were added.

PRELIMINARY

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PWR EVENT ITEM 3

(a)	PLANT, MODE, EVENT DATE	Arkansas 1, heatup, 6/76.
(b)	REFERENC :	50313 (11/9/77).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (isolation valve), valve opening.
(d)	INITIAL INDICATION	Water hammer noise.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (steam hammer due to wave reflection). No reported damage.
(f)	CAUSE AND BASIS FOR EVENT	Poor operating procedure. Main steam isolation valves were inadvertently opened and admitted a slug of hot steam into the only partially warmed main steam line, causing a steam hammer.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Plant procedure was revised to provide timely opening of main steam isolation valves, i.e., to warm the main steam line earlier in the heatup process.

PRELIMINARY

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PWR EVENT ITEM 4

(a)	PLANT, MODE, EVENT DATE	Beaver Valley 1, 10% power, 7/76.
(b)	REFERENCE	50334 (3/15/77), 50334 (3/2/77), 50295 (8/5/77).
(c)	SYSTEM, MECHANICAL FUNCTION	Condenser (inlet piping tubes), valve opening.
(d)	INITIAL INDICATION	Tube leak. First of two events. See 5.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Broken flow deflector and condenser tubes.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Water piston effect from heater drain tank high level dump valve on the flow deflector plate in the condenser.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	This attempt changed angle of deflector plate. Second attempt (see 5) added braces to plate to prevent contact with tubes in case of failure. Replacement of baffle with sparger is under consideration.

PRELIMINARY

DRAFT

PWR EVENT ITEM 5

(a)	PLANT, MODE, EVENT DATE	Beaver Valley 1, 30% power, 10/76.
(b)	REFERENCE	50334 (3/15/77), 50334 (3/2/77), 50295 (8/5/77).
(c)	SYSTEM, MECHANICAL FUNCTION	Condenser (inlet piping tubes), valve opening.
(d)	INITIAL INDICATION	Tube leak. Second of two events. See 4.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Broken flow deflector and condenser tubes.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Water piston effect from heater drain tank high level dump valve on the flow deflector plate in the condenser.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	First attempt (see 4) changed angle of deflector plate. This attempt added braces to plate to prevent contact with tubes in case of failure. Replacement of baffle with sparger is under consideration.

PRELIMINARY

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PWR EVENT ITEM 6

(a)	PLANT, MODE, EVENT DATE	Beaver Valley 1, 50% power, 11/5/76.
(b)	REFERENCE	50334 LER 76-083 (11/19/76), -445 (1/21/77), -464 (2/17/77).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (flow control valve), valve instability.
(d)	INITIAL INDICATION	Water hammer noise and variations in steam generator water level. First of three similar events. See 7, 8.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Damage occurred to instrument lines, valves, insulation, fittings, and shock suppressors.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Sudden feedwater flow oscillations, probably due to feedwater control valve oscillations, caused vibrations in both feedwater and auxiliary feedwater piping. Valve was in automatic mode.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Motor operated isolation valve on auxiliary feedwater was replaced by a manual valve. Main and auxiliary feedwater lines were instrumented for use in attempts to recreate incident.

PRELIMINARY

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PWR EVENT ITEM 7

(a) PLANT, MODE, EVENT DATE	Beaver Valley 1, 73% power, 12/27/76.
(b) REFERENCE	50334 LER 76-089 (1/10/77), -445 (1/21/77), -464 (2/17/77).
(c) SYSTEM, MECHANICAL FUNCTION	Main feedwater (flow control valve), valve instability.
(d) INITIAL INDICATION	Feedwater flow oscillations and feedwater system vibrations. Second of three similar events. See 6, 8.
(e) WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Damage to instrument lines and support.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design. Feedwater flow regulating valve was dynamically unstable due to improper trim and allowed valve opening inappropriate to control signal. Valve was in automatic mode.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Replaced instrument line connectors. Mounted I/P converter to building. Installed control signal limiter to prevent valve control from operating valve in unstable range.

PRELIMINARY

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PWR EVENT ITEM 8

(a) PLANT, MODE, EVENT DATE	Beaver Valley 1, 74% power, 1/5/77.
(b) REFERENCE	50334 LER 77-002 (1/19/77), -445 (1/21/77), -464 (2/17/77).
(c) SYSTEM, MECHANICAL FUNCTION	Main feedwater (flow control valve), valve instability.
(d) INITIAL INDICATION	Feedwater flow oscillations and feedwater system vibrations. Third of three similar events. See 6, 7.
(e) WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Damage to instrument lines, drain lines, valves, supports, and related items.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design. Feedwater flow regulating valve was dynamically unstable due to improper trim and allowed valve opening inappropriate to control signal. Valve was in automatic mode.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	New trims were installed in feedwater regulating valves and flow control valves. Valves and lines were extensively instrumented.

PRELIMINARY

DRAFT

PWR EVENT ITEM 9

(a)	PLANT, MODE, EVENT DATE	Beaver Valley 1, 0% power, 7/1/80.
(b)	REFERENCE	50334 LER 80-046 (7/30/80).
(c)	SYSTEM, MECHANICAL FUNCTION	CCW (to RHR Hx), pump start.
(d)	INITIAL INDICATION	Damage found during routine inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. The water hammer resulted in bowing of the embedment plate for Component Cooling Water (CCW) to the RHR heat exchangers and spalling of the surrounding concrete in a few locations.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate procedures. The water hammer was postulated to have occurred when CCW flow to the RHR heat exchangers was not throttled and flow surged into a partially voided line.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	An evaluation showed the embedment plate to be structurally sound and procedures were put into effect to throttle CCW flow to the RHR heat exchangers.

PRELIMINARY

DRAFT

PWR EVENT ITEM 10

(a)	PLANT, MODE, EVENT DATE	Davis-Besse 1, hot functional testing, 8/5/77.
(b)	REFERENCE	50346-472 (8/19/77).
(c)	SYSTEM, MECHANICAL FUNCTION	RCS (pressurizer relief valve and discharge piping), valve opening.
(d)	INITIAL INDICATION	Severe movement of discharge piping.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. No damage reported.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Same as 33.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Supports were added to line. Heaters were added to line upstream of valve to cause flashing of loop seal water upon valve opening. Discharge piping temperature is to be monitored daily.

PRELIMINARY

DRAFT

PWR EVENT ITEM 11

(a)	PLANT, MODE, EVENT DATE	Ginna, at power, 6/71.
(b)	REFERENCE	50244-73 (7/9/71).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (pump suction line), valve opening.
(d)	INITIAL INDICATION	Water hammer noise.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Suction valve position indicator damaged. Some damage occurred which was not directly related to the water hammer.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design and poor procedures. Temperature control valve failed and closed while normal condensate bypass valve was in closed position. Emergency feedwater valve opened and relatively cold emergency water mixed with heater drain pump discharge, causing water hammer.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Currently unavailable.

PRELIMINARY

DRAFT

PWR EVENT ITEM 12

(a) PLANT, MODE, EVENT DATE	Ginna, 1455 MWt, 7/22/73.
(b) REFERENCE	50244-185 (8/21/73).
(c) SYSTEM, MECHANICAL FUNCTION	Main feedwater (flow control valve), valve instability.
(d) INITIAL INDICATION	Water hammer noise.
(e) WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Cracked support adjacent to valve, skewed rod hanger supports, damaged feedwater pipe insulation.
(f) CAUSE AND BASIS FOR EVENT	Poor quality control during previous valve repair. Feedwater control valve plug separated and induced rapid feedwater flow fluctuations. Plug had not been torqued properly on previous repair, causing holding pin to fail.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Steam vibrations were monitored on feedwater valves. In April '75, after additional analysis, plug throttling design was changed to port throttling design.

PRELIMINARY

DRAFT

PWR EVENT ITEM 13

(a)	PLANT, MODE, EVENT DATE	Ginna, return to power, 6/75.
(b)	REFERENCE	50244 (7/17/75).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (f˜ow control valve), valve instability.
(d)	INITIAL INDICATION	Feedwater piping vibration.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Pressure gauge tubing was broken. Vent valve vibrated partly open. Tubing pulled out of transmitter fitting. Feedwater piping insulation was shaken loose.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Feedwater flow valve instability in 30% to 40% load range.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	External bubble dampers were added to each feedwater valve. Valve vibration was monitored. Installation of permanent hydraulic dampers was planned for the 1976 maintenance outage.

PRELIMINARY

DRAFT

PWR EVENT ITEM 14

(a)	PLANT, MODE, EVENT DATE	Indian Point 1, currently unavailable, 1970.
(b)	REFERENCE	50003-44 (6/29/70), -64 (1/26/71).
(c)	SYSTEM, MECHANICAL FUNCTION	Condenser (deaerator level regulating valve), unidentified (probably valve opening).
(d)	INITIAL INDICATION	Not identified (probably water hammer noise).
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (probably water hammer due to wave reflection). Cracks in piping and valve damage.
(f)	CAUSE AND BASIS FOR EVENT	Either inadequate design or poor quality control. Valves malfunctioned, causing severe water hammer in main condensate system. Probably too rapid closing due to misplaced valve positioners.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Longer run of straight piping was provided before and after the valves. Valve positioners were redesigned and relocated.

PRELIMINARY

DRAFT

PWR EVENT ITEM 15

(a)	PLANT, MODE, EVENT DATE	Indian Point 2, increasing power level, 5/13/74.
(b)	REFERENCE	50247-261 (5/14/74).
(c)	SYSTEM, MECHANICAL FUNCTION	RCS (pressurizer relief valve and discharge piping), valve opening.
(d)	INITIAL INDICATION	Sudden pressure reduction in pressurizer relief tank.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Tank rupture disc blew open. Concrete grouting on tank pedestals was cracked. Sheet metal covering for the containment insulation was damaged. Radioactivity level inside containment increased slightly.
(f)	CAUSE AND BASIS FOR EVENT	Probably inadequate design. Cause not given. Probably same as 33, if pressurizer relief valves are of similar design.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Currently unavailable.

PRELIMINARY

DRAFT

PWR EVENT ITEM 16

(a)	PLANT, MODE, EVENT DATE	Indian Point 2, hot shutdown, 6/7/80.
(b)	REFERENCE	50247 LER 80-005 (7/7/80).
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator blowdown piping, currently unavailable.
(d)	INITIAL INDICATION	Control room indication of secondary side leakage to the containment atmosphere.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. The water hammer resulted in a failed snubber pipe clamp, a broken spring hanger support rod, and a crack in the shell drain.
(f)	CAUSE AND BASIS FOR EVENT	Probable water hammer dynamic loading and inadequate design.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	All damaged components were replaced and the drain line was eliminated. Procedures (unspecified) were revised to preclude water hammer conditions in this system.

PRELIMINARY

DRAFT

PWR EVENT ITEM 17

(a)	PLANT, MODE, EVENT DATE	Maine Yankee, currently unavailable, 12/2/72.
(b)	REFERENCE	50309-86 AO 72-2 (12/14/72).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (dump valves), valve closing.
(d)	INITIAL INDICATION	Drop in steam generator pressure.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (probably steam hammer due to wave reflection). No damage reported.
(f)	CAUSE AND BASIS FOR EVENT	Personnel error. False signal caused steam dump valves to open, dropping steam generator pressure. Flow check valves then closed, causing sudden increase in steam generator pressure, thus initiating scram.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Steam dump valve signals removed from transient recorder patch panel. Personnel instructed to take utmost caution when connecting parameter signal to test recorder.

PRELIMINARY

DRAFT

PWR EVENT ITEM 18

(a)	PLANT, MODE, EVENT DATE	Maine Yankee, 80% power, 10/11/73.
(b)	REFERENCE	50309-183 (2/27/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (excess flow check valves), valve flutter and closure.
(d)	INITIAL INDICATION	Low steam generator level.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (steam hammer due to wave reflection). No damage reported.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Valves failed closed. Safety valves unseated during subsequent transient (repeated problems with flutter from these valves have been noted).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Extensive redesign of valves was required.

PRELIMINARY

DRAFT

PWR EVENT ITEM 19

(a)	PLANT, MODE, EVENT DATE	Maine Yankee, low power physics test, 6/2/77.
(b)	REFERENCE	50309 LER 77-007 (6/7/77).
(c)	SYSTEM, MECHANICAL FUNCTION	CVCS (letdown pressure control valve), valve instability.
(d)	INITIAL INDICATION	Fluctuation in the charging and volume control system letdown flow rate.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Valve positioner fell off and severed an associated air line. Drain line and pressure tap on letdown line were broken.
(f)	CAUSE AND BASIS FOR EVENT	Possibly inspection or quality control. Valve positioner bolts worked loose, allowing positioner to fall off. Loss of air signal to valve controller caused flow oscillations and resulting water hammer.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Positioner is routinely checked to avoid recurrence.

PRELIMINARY

DRAFT

PWR EVENT ITEM 20

(a)	PLANT, MODE, EVENT DATE	Millstone 2, construction phase, 7/22/75.
(b)	REFERENCE	50336-240 (7/31/75).
(c)	SYSTEM, MECHANICAL FUNCTION	Service water (diesel generator air cooler Hx line), valve opening/closure.
(d)	INITIAL INDICATION	Water hammer on startup and shutdown of diesel generator.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (wave reflection or flow-into-voided-line). Inlet nozzle to Hx water box overstressed and fractured.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Failure to recognize water hammer potential. Outlet control valve arrangement on service water line to Hx initiated hydraulic transient.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Piping restraints installed to prevent transmission of thrust forces. Expansions joints replaced by conventional piping. The cast iron inlet water box replaced with epoxy-coated steel box. Outlet control valve scheme modified to reduce hydraulic transient forces.

PRELIMINARY

DRAFT

PWR EVENT ITEM 21

(a) PLANT, MODE, EVENT DATE	Oconee Unit 1, 0% power, 6/28/79.
(b) REFERENCE	50269 LER 79-020 (8/10/79).
(c) SYSTEM, MECHANICAL FUNCTION	Main steam, currently unavailable.
(d) INITIAL INDICATION	Damage found during routine inspection.
(e) WATER HAMMER TYPE, DAMAGE	Unknown. A steam hammer resulted in damage and inoperability of two hydraulic suppressors on the main steam relief valve line.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design. One suppressor failed due to inadequate design of the anchoring base plate and the other failed due to low reservoir fluid level caused by rotation of the suppressor.
(g) CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	The base plate was redesigned and a new one was installed. All suppressors were modified to ensure no future rotation.

PRELIMINARY

DRAFT

PWR EVENT ITEM 22

(a)	PLANT, MODE, EVENT DATE	Oconee 3, power reduction (routine maintenance), 6/75.
(b)	REFERENCE	50287-310 (8/8/75).
(c)	SYSTEM, MECHANICAL FUNCTION	RCS (pressurizer relief valve and discharge piping), valve opening.
(d)	INITIAL INDICATION	Unexplained drop in RCS pressure.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Tank rupture disc blew open. Mirror insulation separated from bottom nozzle of pressurizer. 1500 gallons of coolant were released to containment sump.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate valve maintenance. Pressurizer relief valve was stuck in open position due to boric acid crystal buildup.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	All pressurizer relief valves were inspected for boric acid crystal buildup. Valve cycling prior to startup was incorporated into station operating procedures.

PRELIMINARY

DRAFT

PWR EVENT ITEM 23

(a)	PLANT, MODE, EVENT DATE	Palisades, currently unavailable, 5/14/74.
(b)	REFERENCE	50255-324 AO 74-08 (7/19/74), -334 (8/13/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Safety systems (ECCS, low pressure SI pump suction line), valve opening.
(d)	INITIAL INDICATION	Pipe restraint pulled loose from mounting.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-partially-voided-line. Anchor bolts on pump suction line pipe restraint pulled out of mounting.
(f)	CAUSE AND BASIS FOR EVENT	Poor operating procedure. Air was apparently introduced into the system during the testing and filling of the sodium hydroxide system.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Applicable operating procedures were reviewed and revised to eliminate possibility of introducing air into line.

PRELIMINARY

DRAFT

PWR EVENT ITEM 24

(a)	PLANT, MODE, EVENT DATE	Rancho Seco, hot functional testing, 1974.
(b)	REFERENCE	50312 (6/19/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (flow control valve), valve throttling.
(d)	INITIAL INDICATION	Water hammer noise.
(e)	WATER HAMMER TYPE, DAMAGE	Steam-bubble formation and collapse. Seismic support was damaged.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Valve lineup allowed backflow from steam generator into feedwater inlet line, into feedwater cleanup line, and (flashing) into condenser when valve throttling reduced inlet line pressure.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	New feedwater cleanup lines were installed upstream of check valves and flow control valves. Check valves prevent blackflow from steam generator during feedwater cleanup with minifeed to steam generator. Procedures were changed to require use of new lines under these conditions.

PRELIMINARY

DRAFT

PWR EVENT ITEM 25

(a)	PLANT, MODE, EVENT DATE	Rancho Seco, refueling shutdown, 12/15/78.
(b)	REFERENCE	50312 LER 78-017 (1/12/79).
(c)	SYSTEM, MECHANICAL FUNCTION	RHR, pump start.
(d)	INITIAL INDICATION	Damage found during inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-a-line-with-a-closed-valve. Damage to four support hangers, one snubber on the No. B Decay Heat System and one hanger on the No. A system.
(f)	CAUSE AND BASIS FOR EVENT	Incorrect procedures. Event occurred due to incorrect valve lineup before starting the No. B RHR pump.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	All damaged components were repaired or replaced.

PRELIMINARY

DRAFT

725

PWR EVENT ITEM 26

(a)	PLANT, MODE, EVENT DATE	Robinson 2, currently unavailable, 12/19/78.
(b)	REFERENCE	50261 LER 79-012 (5/22/79).
(c)	SYSTEM, MECHANICAL FUNCTION	ECCS (safety injection), pump start.
(d)	INITIAL INDICATION	Damage found during inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Six supports of the cold leg safety injection line were found in a faulted condition (inoperable during certain design plant conditions).
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. The water hammer was postulated to have occurred during testing when flow was admitted into the voided injection line.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	New supports, with a higher design capacity, were installed to replace those damaged and two additional supports were installed on the injection line.

PRELIMINARY

DRAFT

PWR EVENT ITEM 27

(a)	PLANT, MODE, EVENT DATE	Salem 1, hot shutdown, 1/3/77.
(b)	REFERENCE	50272 LER 77-002/3L (1/12/77).
(c)	SYSTEM, MECHANICAL FUNCTION	CVCS (letdown header vent line), currently unavailable.
(d)	INITIAL INDICATION	Leak detection system alarmed.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (probably flow-into-partially-voided-line). Pipe break with reactor coolant release resulted.
(f)	CAUSE AND BASIS FOR EVENT	Unknown (probably inadequate design). Excessive vibration of the vent line caused the pipe to shear downstream of the joint to the main letdown line.
(g)	CORRECTIVE MEASURE UNDERTAKEN FOR PREVENTION	Vent line was removed, and remaining socket was plugged and welded. Non-destructive testing was performed on new weld and on similar vent line welds.

PRELIMINARY

DRAFT

PWR EVENT ITEM 28

(a)	PLANT, MODE, EVENT DATE	San Onofre 1, plant operating, 10/9/69.
(b)	REFERENCE	50206-40 (10/24/69).
(c)	SYSTEM, MECHANICAL FUNCTION	Circulating water (tsunami gates), N/A.
(d)	INITIAL INDICATION	Pieces of concrete observed coming up on bar rakes.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (water hammer due to wave reflection). Gate was cracked, deformed, and torn loose from guide slots. Hydraulic actuator was damaged.
(f)	CAUSE AND BASIS FOR EVENT	Bonding material between reinforced concrete gate slabs and stubs holding the slabs to the actuator failed. Gate dropped into water intake flow path.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	A new gate was fabricated with an improved mechanical design for attaching the gate slab to the actuator yoke.

PRELIMINARY

DRAFT

PWR EVENT ITEM 29

(a) PLANT, MODE, EVENT DATE	San Onofre 1, normal operation, 10/21/73.
(b) REFERENCE	50206-246 (10/31/73), -271 (2/5/74).
(c) SYSTEM, MECHANICAL FUNCTION	ECCS (safety injection valve), valve opening.
(d) INITIAL INDICATION	SI valve motor operator casing found damaged.
(e) WATER HAMMER TYPE, DAMAGE	Flow-into-partially-voided-line. Valve operator casing bolts and pipe hanger support failed.
(f) CAUSE AND BASIS FOR EVENT	Inadequate design. Safety injection line had an air bubble. Line design did not permit adequate on-line venting.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Vent piping changes were made to facilitate on-line venting of applicable sections of safety injection piping. Procedures were written to specify frequent venting.

PRELIMINARY

DRAFT

PWR EVENT ITEM 30

(a) PLANT, MODE, EVENT DATE	San Onofre 1, currently unavailable, 1/74.
(b) REFERENCE	50206 (2/8/74).
(c) SYSTEM, MECHANICAL FUNCTION	Main steam (line, feedwater line/supports), unidentified (probably valve closure).
(d) INITIAL INDICATION	Damaged knee supports and snubber.
(e) WATER HAMMER TYPE, DAMAGE	Unknown (probably water hammer due to wave reflection). Knee supports on main steam line, and a knee support and snubber on main feedwater line were damaged.
(f) CAUSE AND BASIS FOR EVENT	Poor design and quality control procedures contributed to damage. Cause of water hammer is unknown. Original design assumptions included lateral loading, which was neglected in final design. Anchor plates were incorrectly installed. Bolts were wrong size.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Design and quality control problems were remedied.

PRELIMINARY

DRAFT

PWR EVENT ITEM 31

(a) PLANT, MODE, EVENT DATE	San Onofre 1, maintenance outage, 5/14/79.
(b) REFERENCE	50206 LER 79-009 (6/19/79).
(c) SYSTEM, MECHANICAL FUNCTION	Main feedwater (regulating valve), closure.
(d) INITIAL INDICATION	Sudden stop and restart of flow. Failed snubber on No. A main feedwater line.
(e) WATER HAMMER TYPE, DAMAGE	Damaged snubber found during functional testing on 6/5/79. Water hammer detected on 5/14/79 following a unit trip from full power.
(f) CAUSE AND BASIS FOR EVENT	Misadjustment of feedwater valve control circuit. Event occurred due to valve closure caused by misadjustment of the feedwater regulating valve control circuitry.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The control circuitry was adjusted and the damaged snubber was repaired.

PRELIMINARY

DRAFT

PWR EVENT ITEM 32

(a) PLANT, MODE, EVENT DATE	San Onofre 1, refueling outage, 5/15/80.
(b) REFERENCE	50206 LER 80-021 (5/29/80).
(c) SYSTEM, MECHANICAL FUNCTION	Main feedwater, currently unavailable.
(d) INITIAL INDICATION	Damage found during routine inspection.
(e) WATER HAMMER TYPE, DAMAGE	Unknown. Broken hanger clamp and deformed pipe guide.
(f) CAUSE AND BASIS FOR EVENT	Unknown mechanical loading (possibly a water hammer) and probably inadequate design.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The damaged supports were repaired or replaced.

PRELIMINARY

DRAFT

PWR EVENT ITEM 33

(a)	PLANT, MODE, EVENT DATE	Surry 1, currently unavailable, 1/73.
(b)	REFERENCE	50280 (9/17/73).
(c)	SYSTEM, MECHANICAL FUNCTION	RCS (pressurizer relief valve and discharge piping), valve opening.
(d)	INITIAL INDICATION	Broken seismic snubber on discharge piping.
(e)	WATER HAMMER TYPE, DAMAGE	Flow-into-voided-line. Seismic snubber was broken due to displaced discharge line piping.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Discharge piping upstream of relief valves holds a water seal against valves, preventing erosion by steam or gas during normal operation. When valves open, water slug is impelled through piping into relief tank with high impact force unless vaporized.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Additional bracing was added to the discharge piping. Size of snubber was increased to resist additional hydraulic forces. Analysis indicated valve design was adequate.

PRELIMINARY

DRAFT

PWR EVENT ITEM 34

(a) PLANT, MODE, EVENT DATE	Surry 2, startup (accumulator discharge line leak test), 1974.
(b) REFERENCE	50281 (11/29/74).
(c) SYSTEM, MECHANICAL FUNCTION	ECCS (SI accumulator discharge line), valve opening.
(d) INITIAL INDICATION	Damaged pipe restraint support.
(e) WATER HAMMER TYPE, DAMAGE	Steam-bubble collapse. Pipe restraint support was damaged.
(f) CAUSE AND BASIS FOR EVENT	Poor operating procedure. Line pressure was reduced below saturation pressure during leak testing. Water hammer occurred when accumulator discharge valve was opened.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Operating procedure was modified to insure that line pressure was maintained at a level above the saturation pressure during leak testing.

PRELIMINARY

DRAFT

PWR EVENT ITEM 35

(a)	PLANT, MODE, EVENT DATE	Turkey Point 3, refueling shutdown, 11/5/75.
(b)	REFERENCE	50250-434 UE 75-3 (12/5/75).
(c)	SYSTEM, MECHANICAL FUNCTION	Main steam (spring hanger), possibly valve closing.
(d)	INITIAL INDICATION	Distortion of spring support.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (probably steam hammer due to wave reflection). Distortion of spring support.
(f)	CAUSE AND BASIS FOR EVENT	Unidentified (probably inadequate piping design and/or poor procedure). Piping was not designed to withstand transient loads. A spurious closure of the main steam isolation valve at full power may have generated enough transient load (probably water hammer) to distort the hanger.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Hanger was redesigned to carry anticipated transient load from spurious valve closure. Similar hangers were evaluated to see if the design was adequate. Possible water hammer was not addressed.

PRELIMINARY

DRAFT

PWR EVENT ITEM 36

(a)	PLANT, MODE, EVENT DATE	Turkey Point 4, 0% power, 6/11/79.
(b)	REFERENCE	50251 LER 79-009 (7/11/79).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater, currently unavailable.
(d)	INITIAL INDICATION	Damage found during routine inspection.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown. A snubber on the No. C steam generator feedwater line was found in a "locked up" position.
(f)	CAUSE AND BASIS FOR EVENT	Unknown mechanical loading and probably inadequate design.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The snubber, rated at 10,000 lb, was replaced with one rated at 15,000 lb.

PRELIMINARY

DRAFT

PWR EVENT ITEM 37

(a)	PLANT, MODE, EVENT DATE	Zion 1, 750 MWe, 6/6/74.
(b)	REFERENCE	50295-324 (6/14/74).
(c)	SYSTEM, MECHANICAL FUNCTION	Auxiliary feedwater (turbine exhaust line, pump), pump start.
(d)	INITIAL INDICATION	Water hammer noise.
(e)	WATER HAMMER TYPE, DAMAGE	Water entrainment in steam lines. Pipe hanger was damaged.
(f)	CAUSE AND BASIS FOR EVENT	Either inadequate design or poor procedure. Water hammer occurred after reset of overspeed trip valve and restart of the auxiliary feedwater pump. Rain water entered the open exhaust, flooding the turbine. Back pressure in the drain system prevented the draining of this water.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Drain lines were temporarily routed to floor. Traps were inspected and cleaned.

PRELIMINARY

DRAFT

PWR EVENT ITEM 38

(a)	PLANT, MODE, EVENT DATE	Zion 1, 875 MWe (hot shutdown), 5/76.
(b)	REFERENCE	50295 (10/27/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (flow control valve), not identified (probably valve instability).
(d)	INITIAL INDICATION	Water hammer noise.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (probably water hammer due to wave reflection). No damage. Safety injection occurred.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Rapid feedwater flow increase to steam generator due to water hammer, probably from too rapid trim on a flow control valve. See 7, 8, 40.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Feedwater piping was physically monitored. Auxiliary feedwater flow was limited to 50 gpm per steam generator.

PRELIMINARY

DRAFT

PWR EVENT ITEM 39

(a)	PLANT, MODE, EVENT DATE	Zion 1, currently unavailable, 9/26/76.
(b)	REFERENCE	50295-827 (12/13/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater (isolation valves), valve opening.
(d)	INITIAL INDICATION	Audible water hammer noise heard in control room. SI injection due to pressure spike in steam generator feedwater line.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (caused by mixing of cold and hot water). No damage due to water hammer.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate procedure. Feedwater line had sagged due to unrelated problems and was isolated. Later, when attempting to unisolate the line, the water hammer occurred due to the mixing of relatively cool water with hot feedwater.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	None for water hammer.

PRELIMINARY

DRAFT

PWR EVENT ITEM 40

(a)	PLANT, MODE, EVENT DATE	Zion 2, 875 MWe (hot shutdown), 6/76.
(b)	REFERENCE	50304 (10/12/76), 50304 (10/13/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Main feedwater, (flow control valve), not identified (probably valve instability).
(d)	INITIAL INDICATION	Water hammer noise.
(e)	WATER HAMMER TYPE, DAMAGE	Unknown (probably water hammer due to wave reflection). No damage. Safety injection occurred.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate design. Rapid feedwater flow increase to steam generator due to water hammer, probably from too rapid trim on a flow control valve. See 7, 8, 38.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Feedwater piping was physically monitored. Auxiliary feedwater flow was limited to 50 gpm per steam generator.

PRELIMINARY

DRAFT

Category II: PWR Non-Water-Hammer Events

PWR EVENT ITEM 41

(a)	PLANT, MODE, EVENT DATE	Calvert Cliffs 1, 100% power, 4/25/77.
(b)	REFERENCE	50317 LER 77-040 (5/25/77). See 46.
(c)	SYSTEM, MECHANICAL FUNCTION	SCW (for reactor auxiliaries), check valve closure or delayed opening.
(d)	INITIAL INDICATION	Currently unavailable.
(e)	EVENT TYPE, DAMAGE	Component malfunction. Check valve disc wedged closed on seat.
(f)	CAUSE AND BASIS FOR EVENT	Internal disc stop incorrectly adjusted due to inadequate instructions.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Stop readjusted, valve tested by cycling 12 times.

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PWR EVENT ITEM 42

(a)	PLANT, MODE, EVENT DATE	Haddam Neck, currently unavailable, 3/17/78.
(b)	REFERENCE	50213 LER 78-003 (4/17/78).
(c)	SYSTEM, MECHANICAL FUNCTION	RCS (steam generator blowdown line), currently unavailable.
(d)	INITIAL INDICATION	Leak at blowdown line junction.
(e)	EVENT TYPE, DAMAGE	Potential for steam-water entrainment water hammer. Damage was a leak in piping.
(f)	CAUSE AND BASIS FOR EVENT	Incident involved mixing of subcooled water and steam from interconnected system.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Leak was weld repaired. Use of thermal sleeve, expansion joint or similar device is under investigation.

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PWR EVENT ITEM 43

(a)	PLANT, MODE, EVENT DATE	Indian Point 1, 94% power, 10/74.
(b)	REFERENCE	50003 (11/1/74).
(c)	SYSTEM, MECHANICAL FUNCTION	CVCS (outlet filter drain valves), unidentified (probably valve opening).
(d)	INITIAL INDICATION	Radiation activity.
(e)	EVENT TYPE, DAMAGE	Possibly flow-into-partially-voided-line. Leaking valve diaphragms resulted.
(f)	CAUSE AND BASIS FOR EVENT	Inadequate inspection and replacement procedure. Normal wear, plus shock loading during system startup repressurization caused diaphragm leakage on drain valves for purification outlet filters.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	Valve diaphragms were replaced. No further action was taken, since no previous valve malfunctions or failures had occurred. Water hammer was not addressed.

PRELIMINARY

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PWR EVENT ITEM 44

(a)	PLANT, MODE, EVENT DATE	Surrey 1 and 2, currently unavailable, 11/74.
(b)	REFERENCE	50281-293 (2/14/75).
(c)	SYSTEM, MECHANICAL FUNCTION	Containment spray (isolation check valves), valve closure.
(d)	INITIAL INDICATION	Leakage in containment isolation check valve.
(e)	EVENT TYPE, DAMAGE	Component malfunction. Cracks found at the outside keyways of the rockshafts on recirculation spray check valves.
(f)	CAUSE AND BASIS FOR EVENT	Design. Too high impact loading on disc due to improper counter weight design.
(g)	CORRECTIVE MEASURES UNDER-TAKEN FOR PREVENTION	Counterweights were modified and repositioned on the valve weight lever to provide the correct seating torque.

PRELIMINARY

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APPENDIX C

PWR STEAM GENERATOR EVENTS

PRELIMINARY

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SGWH EVENT ITEM 1

(a) PLANT, MODE, EVENT DATE	Calvert Cliffs 1, preoperational hot testing, 5/13/75.
(b) REFERENCE	R. W. Reid, NRC, letter to A. E. Lundvall, Baltimore Gas and Electric Co., Subject, "Transmittal of Amendments to Facility Operating Licenses," 3/10/80.
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d) INITIAL INDICATION	Water hammer heard.
(e) DAMAGE	None.
(f) CAUSE AND BASIS FOR EVENT	Following a unit trip on loss of main feedwater, three steam generator water hammers were heard while main feedwater was being reestablished to the steam generators.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	"J" shaped top discharge tubes were installed on the feedrings of both units and the bottom discharge holes were plugged. Procedures were put into effect to require the use of the auxiliary feedwater system (rather than the main feedwater system) during recovery from reactor trips.

PRELIMINARY

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SGWH EVENT ITEM 2

(a)	PLANT, MODE, EVENT DATE	Calvert Cliffs 2, preoperational hot testing, 5/19/76.
(b)	REFERENCE	See 1.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d)	INITIAL INDICATION	Currently unavailable.
(e)	DAMAGE	None.
(f)	CAUSE AND BASIS FOR EVENT	A steam generator water hammer occurred during testing of the effectiveness of internal standpipes in the steam generator. The event occurred at 5% of main feedwater rated flow as the steam generator water level reached the feedring in an unspecified steam generator.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	See 1.

PRELIMINARY

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SGWH EVENT ITEM 3

(a)	PLANT, MODE, EVENT DATE	Donald C. Cook 1, currently unavailable, between 11/14/75 and 1/2/76.
(b)	REFERENCE	Nuclear Power Experience, Nuclear Power Experience, Inc., PWR Vol. VI.E.119. 50316 RO 76-03 (1/28/76).
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, currently unavailable.
(d)	INITIAL INDICATION	Snubber damage found.
(e)	DAMAGE	Three damaged hydraulic snubbers were found on the section of the No. 4 feedwater line located inside the containment crane wall.
(f)	CAUSE AND BASIS FOR EVENT	A steam generator water hammer was postulated to have occurred in the No. 4 feedwater line sometime between November 14, 1975 (the date of an extensive incontainment inspection by the licensee) and January 2, 1976 (the date the damage was found).
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The damaged snubbers were either repaired or replaced. Administrative controls were put into effect to limit auxiliary feedwater flow to 150 gpm per steam generator whenever the feedwater sparges become uncovered. An analysis of the feedwater system was performed, using a hypothesized water hammer forcing function to verify that the structural integrity of the feedwater piping and support structures would be maintained under future water hammer conditions.

PRELIMINARY

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SGWH EVENT ITEM 4

(a)	PLANT, MODE, EVENT DATE	Donald C. Cook 1, reactor trip recovery, 3/10/77.
(b)	REFERENCE	Nuclear Power Experience, Nuclear Power Experience, Inc., PWR Vol. VI.E.119. 50315 RO 77-14, RO 77-15.
(c)	SYSTEM, MECHANICAL FUNCTION	Steam generator, pump/valve operation.
(d)	INITIAL INDICATION	Water hammer heard by plant personnel.
(e)	DAMAGE	The event resulted in three failed hydraulic snubbers on the No. 1 feedwater line. Two failed snubbers were outside containment and the third was inside containment. A crack was found in the valve body in the auxiliary feedwater line to the No. 3 steam generator and was attributed to this same event.
(f)	CAUSE AND BASIS FOR EVENT	A steam generator water hammer event occurred in the No. 1 feedwater line during recovery from a reactor trip. The event was believed to be directly caused by increasing auxiliary feedwater flow to over 200 gpm per steam generator, which was in violation of the administrative prescribed maximum of 150 gpm.
(g)	CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The snubbers and auxiliary feedwater valve were repaired or replaced and a feedwater system inspection was performed. The reactor operators were reminded to adhere to the auxiliary feedwater flow maximum flow procedural guidelines. The feedwater piping geometry of all loops was altered to eliminate the horizontal run of piping adjacent to each steam generator. The installation of "J" shaped top discharge tubes was performed on all feedwater feedrings and the bottom discharge holes were plugged.

PRELIMINARY

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SGWH EVENT ITEM 5

(a) PLANT, MODE, EVENT DATE	Indian Point 2, startup, 11/13/73.
(b) REFERENCE	J. A. Block et al., An Evaluation of PWR Steam Generator Water Hammer, Creare, Inc., NUREG-0291, December 1976. Nuclear Power Experience, Nuclear Power Experience, Inc., PWR Vol., VI.E.34.
(c) SYSTEM, MECHANICAL FUNCTION	Steam generator, pump/valve operation.
(d) INITIAL INDICATION	Water hammer heard by plant personnel.
(e) DAMAGE	The water hammers resulted in a 180° circumferential crack in the No. 22 feedwater piping within a few feet of the steam generator. Localized bulging and stress cracking occurred in the feedwater piping around the containment penetration.
(f) CAUSE AND BASIS FOR EVENT	A steam generator water hammer event occurred in the No. 22 steam generator feedwater piping following a plant trip caused by a high steam generator water level signal from the No. 23 steam generator. This event occurred shortly after the automatic initiation of auxiliary feedwater and a second event occurred 50 minutes later in the same piping.
(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION	The damaged piping and associated hardware were repaired or replaced and the No. 22 feedwater piping geometry was altered to eliminate the horizontal piping run adjacent to the steam generator. Modifications to all feedwater lines incuded: (a) the addition of restraints to prevent future excessive pipe rebound under water hammer conditions; (b) the installation of hydraulic dampers to the main feedwater valves to prevent rapid closure upon receipt of closure signals. After the preceding modifications, a test program was begun to investigate the effect of

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SGWH EVENT ITEM 5 (continued)

(g) CORRECTIVE MEASURES UNDERTAKEN FOR PREVENTION (continued)	auxiliary feedwater flow rates on the occurrence of steam generator water hammer. The tests showed the system to be susceptible to SGWH with auxiliary feedwater flows of 200 gpm or greater.

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