POWER REACTOR EVENTS & ISSUES

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Vol. 1

Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission

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IN THIS ISSUE . . .

Power Reactor Events & Issues is published by the Office for Analysis and Evaluation of Operational Data (AEOD) of the Nuclear Regulatory Commission. The publication reviews selected operating events that have occurred at nuclear power plants and presents the results of NRC-sponsored analysis of pertinent operating issues.

The feature article in this issue of ***Power Reactor Events & Issues*** is based on AEOD Engineering Evaluation Report AEOD/E91-01, "A Review of Water Hammer Events After 1985," written by E. J. Brown. This study was initiated following several instances of water hammer involving the service water system at Arkansas Nuclear One. The task was to evaluate the need to reissue previous NRC guidance about water hammer or to suggest additional measures to prevent or mitigate their occurrence.

This study concluded that the frequency of reported water hammer occurrences continues to drop and no new phenomena were identified as causes of water hammer. In addition, this study supports prior NRC conclusions regarding water hammer; however, some aspects that could impact safety and were identified in the study have not been previously emphasized.

A REVIEW OF WATER HAMMER EVENTS AFTER 1985

SUMMARY

Water hammer is a rapid pressure change caused by a change in the velocity of a fluid in a closed volume (Ref. 1). These pressure changes can create loads on piping and other components that exceed their design limits. Water hammer incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion.

Water hammer events involve not only piping systems but components connected to them, as well as components used to support them. Most of the damage caused by water hammer has been relatively minor (e.g., damaging only pipe hangers and restraints); however, there have been some incidents that have resulted in the loss of function of major components. Significant water hammer effects have also been observed in instrumentation systems.

After several water hammer incidents resulted in piping and valve damage, the water hammer issue was determined to be an Unresolved Safety Issue (USI), and classified as USI A-1, *Water Hammer* (Ref. 2). USI A-1 was considered to be resolved by the publication of NUREG-0927, Revision 1, *Evaluation of Water Hammer Occurrence in Nuclear Power Plants*, in March 1984 (Ref. 3). In resolving USI A-1, the NRC staff concluded that new requirements to reduce the number of water hammer events were not supported by cost-benefit guidelines. However, guidelines were provided concerning measures for preventing and mitigating water hammer.

The NRC reassessed resolution of the water hammer issue after a water hammer event at San Onofre Unit 1 in November 1985 damaged plant equipment and challenged the integrity of the plant's heat sink (Ref. 4). The reassessment, completed in 1986, reconfirmed the original conclusion that new or additional requirements to reduce the number of water hammer events were not supported by cost-benefit guidelines (Ref. 5, 6). The reassessment determined that the frequency of water hammer events had actually decreased significantly since the initial review performed for resolution of USI A-1. In addition, no new phenomena were identified as causes of water hammer. (USI A-1 identified 148 reported water hammer events from 1969 to 1980, while the reassessment after the San Onofre event identified 40 reported water hammer events from 1981 to 1985.)

While performing a Diagnostic Evaluation at Arkansas Nuclear One (ANO) during August and September 1989 (Ref. 7), NRC staff identified several water hammer incidents that involved the service water system (Ref. 8). These water hammer events at ANO were not reportable as Licensee Event Reports (LERs) under 10 CFR 50, §50.73. The Office for Analysis and Evaluation of Operational Data (AEOD) reevaluated water hammer events to assess the need to reissue previous NRC guidance about water hammer or to suggest additional measures to prevent or mitigate its occurrence. A summary of AEOD's reevaluation, based on AEOD Engineering Evaluation Report AEOD/E91-01, *A Review of Water Hammer Events After 1985* (Ref. 9), performed by E. J. Brown, is given below.

EVENT DESCRIPTIONS

AEOD's current assessment of water hammer identified 12 reported water hammer events between January 1986 and March 1990 (Table 1). These events were reviewed to determine whether they were associated with any new physical phenomena. This review also concentrated on identifying common mode failure aspects and lessons that may be useful to assist other plants in preventing situations that could result in water hammer. Summaries of the 12 reported water hammer events identified in E91-01 follow.

TABLE 1 - WATER HAMMER EVENTS AFTER 1985			
Plant	Type	System	Event Description
Susquehanna Unit 2	BWR	RHR Shutdown Cooling Mode	Valve isolated while switching from Pump "D" to Pump "A." Partial pipe draindown - not refilled prior to restart.
Shearon Harris	PWR	Steam generator blowdown	Rapid motion of isolation valve resulted in snubber damage.
Trojan	PWR	Accumulator fill lines	Transferring water between accumulators with high differential pressure. No procedure.
South Texas Unit 1	PWR	Auxiliary Feedwater (AFW) Vent Lines	Pressure fluctuations developed by crossover flow control valve throttling.
Indian Point Unit 3	PWR	Feedwater	Cycling a flow control valve (FCV) caused pressure drop between FCV and isolation valve.
Oyster Creek	BWR	Isolation Condens-er (IC)	Steam lines to IC partially filled with water. Suspect-ed reverse flow through one-half of each IC.
Waterford Unit 3	PWR	Steam Generator Blowdown	Cycled inside-containment isolation valve to verify operability without closing outside isolation valve.
Oconee Unit 3	PWR	Main Steam	Suspected water accumulation in drain line for the turbine bypass line to the A condenser.
Palisades	PWR	Accumulator Injection	Safety injection tank vented to 50 psig and back leakage from primary system resulted.
ANO Unit 2	PWR	Steam supply to AFW	Condensate buildup at low point resulted in water slugging on Emergency Feedwater (EFW) pump startup.
Dresden Units 2 and 3	BWR	High Pressure Coolant Injection (HPCI)	Leaking HPCI injection valve and check valve caused FW leakage into HPCI system and void formation.
Dresden Unit 2	BWR	HPCI	FW leakage into HPCI system due to MOV failure to completely close after stroke timing test.

Susquehanna Unit 2

On October 12, 1986, the unit was in the shutdown cooling mode with the "D" residual heat removal (RHR) pump in service (Ref. 10). The "B" RHR pump was then started (see Figure 1). After switchover to the "B" pump was completed, pump "D" was shut down. However, at about the same time, outboard isolation valve F008 in the letdown line from the "B" recirculation pump to the suction of the "B" RHR pump closed, causing the "B" RHR pump to trip. Operations personnel reset the logic

and reopened valve F008. Reopening valve F008 resulted in water hammer and valve F008 again closed. The water hammer was caused by partial draindown of the system through a pathway to the condenser. This pathway is used to control the reactor water level while the RHR system is in service. The operators failed to fill and vent the RHR system piping after the "B" RHR pump tripped.

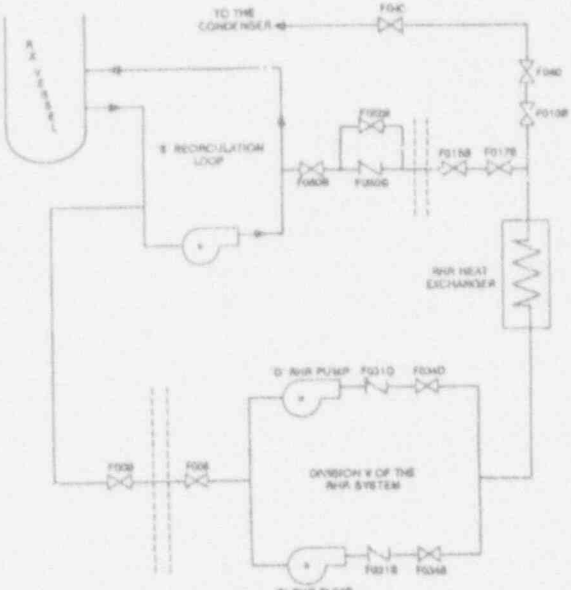

Figure 1

Susquehanna Unit 2 RHR Shutdown Cooling

Shearon Harris Unit 1

On April 22, 1987, plant personnel discovered a damaged snubber on a steam generator blowdown pipe (Ref. 11). In addition, two broken pipe supports and various pipe displacements were found. The steam generator blowdown piping is safety-related because of concerns of a high energy pipe break outside the reactor containment. The cause of water hammer was attributed to rapid motion of the blowdown isolation valve. The corrective action was to increase the stroke time of the valve.

Trojan

On May 12, 1987, the "A" accumulator 1-inch fill line ruptured at the nozzle-to-pipe weld while transferring water from the "A" accumulator to the "D" accumulator (Ref. 12). Approximately 2000 gallons of water were released to the containment. The "A" accumulator was at 583 psig and the "D" accumulator was depressurized. The weld failure was attributed to low-cycle, high-stress fatigue cracking. After repairing the weld, the fill line again ruptured while transferring water from the "A" accumulator to the "D" accumulator. This time, the "A" accumulator was at 650 psig. The "D" accumulator was still depressurized. The cause of the fill line rupture was excessive reverse flow through the packless diaphragm globe valve. This reverse flow caused cyclic vibrations.

The root cause of the fill line ruptures was inadequate procedures. Procedures were in place to transfer water between accumulators via the sample lines. No procedures existed for transferring water between accumulators via the fill lines. Water was transferred between the accumulators via the fill lines because the sample lines were tagged out for maintenance. A dynamic analysis, which modelled backflow through the fill line, showed that hydraulic loads far in excess of those necessary to fail the pipe would be imposed on the nozzle-to-pipe weld. In addition, a backflow test through a packless globe valve, similar to the valve in the accumulator fill line, resulted in a pipe failure at a flow of about 70 gpm. Operating procedures were revised to prohibit water transfer between the accumulators.

South Texas Unit 1

On November 5, 1987, while in hot shutdown and prior to initial criticality, a 1-inch diameter, double valve vent line in the pump discharge piping of the auxiliary feedwater (AFW) train "A" broke off (Ref. 13). A second failure occurred 3 days later in a similar manner in a double valve instrument

tap for the "D" train AFW pump discharge line (see Figure 2). The initial assessment cited the cause as water hammer resulting from improper venting of the AFW system. The AFW system however, continued to experience vibration when in operation. Subsequent testing revealed that over-throttling the flow control valves in trains "A" or "D" introduced pressure fluctuations at a dominant frequency of 24 hertz. This frequency matched the natural frequency of the piping system. Mechanical stops were installed to prohibit excessive throttling of the valves.

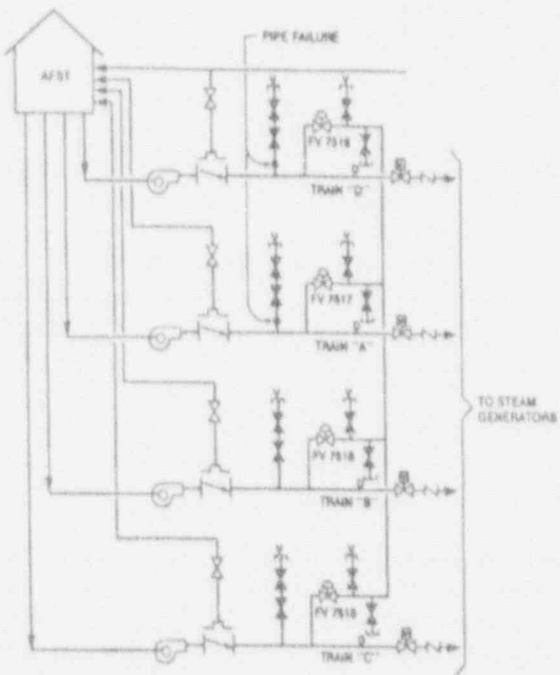

Figure 2

South Texas Unit 1 AFW System

Indian Point Unit 3

On March 31, 1988, main boiler feedwater pump (MBFP) No. 32 tripped with the reactor at full power (Ref. 14). The pump tripped in response to a "discharge valve not fully open" signal. This signal was generated by motion of the discharge valve (BFD-2-32) non-rotor-driven limit switch (Crane/Teledyne Model T 40-80). The limit switch motion was caused by a water hammer shock generated by the cycling of the No. 32 MBFP recirculation valve (FCV-1116) while the manual isolation valve (BFR-1-32) was closed. The water hammer occurred because of the pressure drop between the recirculation valve (FCV-1116) and the manual isolation valve (BFR-1-32). The cause of the water hammer was the recirculation valve cycling while personnel were troubleshooting a faulty limit switch on the valve.

Oyster Creek

On September 28, 1988, while operating at full power, operators determined that both the "A" and "B" isolation condensers (ICs) were operating in an unanalyzed condition (Ref. 15). Temperature data suggested that the steam lines to the ICs were at least partially filled with water. Analysis of the temperature data suggested existence of reverse flow through one-half of each IC and the possibility of subcooled condensate buildup in the steam lines to the IC. This raised concern over potential effects of increased piping loads, thermal stresses, and possible water hammer. The reactor was shut down. The event was the subject of an NRC Augmented Inspection Team review (Ref. 16) and extensive licensee followup.

The partial filling with water of the steam lines to the IC was outside the previously evaluated operating conditions. A subsequent licensee Technical Data Report concluded that water hammer had not occurred during this event and that other postulated transients would not result in water hammer (Ref. 17). Operating procedures were implemented to limit water accumulation in the steam lines.

Waterford Unit 3

During a routine system walkdown on July 14, 1989, with the plant at full power, plant personnel found a damaged pipe support on a steam generator blowdown pipe (Ref. 18). It was later determined that in this condition, the structural

integrity of the blowdown system (including the outside containment isolation valve and shield building penetration) could not be assured during a seismic event.

The support was observed to be undamaged during a walkdown in June 1989. Between the two system walkdowns, personnel performed the operating test "Engineered Safety Feature Actuation Signal (ESFAS) Subgroup Relay Test." This test involved cycling the inside containment isolation valve to verify its operability in response to a containment isolation actuation signal or an emergency feedwater system actuation signal. The procedure did not require shutting the outboard containment isolation valve or minimizing the blowdown flow prior to opening the inboard isolation valve (see Figure 3). Thus, a water hammer transient most likely occurred when the inboard isolation valve was opened.

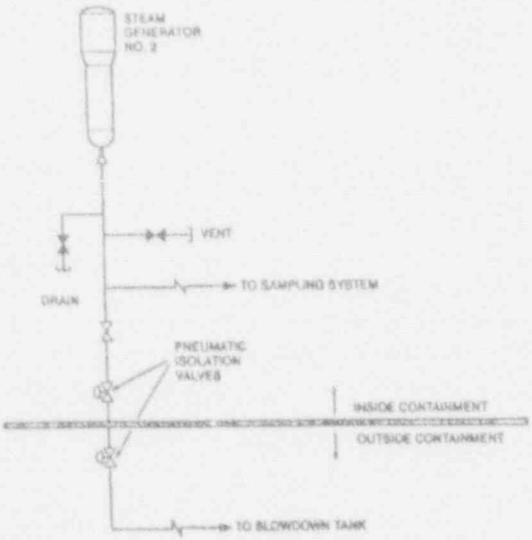

Figure 3

Waterford Unit 3
Steam Generator Blowdown System

Oconee Unit 3

After the main turbine tripped on March 6, 1989, the reactor tripped as expected (Ref. 19). Following the trip, a water hammer in the main steam turbine bypass line to the "A" condenser damaged three pipe supports and deflected the bypass line 12 to 18 inches. Subsequent investigations indicated that the most likely cause of the water hammer was water accumulation near an orifice in a drain line; however, no obstructions could be verified. A Station Problem Report was initiated to resolve problems with improper draining of the pipe (Ref. 20).

Palisades

On April 25, 1990, plant personnel found gross deformation of some of the seismic pipe supports in piping between the "A" and "D" safety injection tanks (SITs) and the primary reactor system (Ref. 21). Minor damage was also found on piping supports for the "B" and "C" SIT piping. An evaluation found that a water hammer occurred on October 1, 1987. This water hammer could have caused the damage; however, the exact cause was not clear. At the time of the water hammer, the reactor was in hot shutdown and the primary coolant system temperature was greater than 500-°F. The SITs normally have nitrogen cover gas at 200 psig, but the "A" tank had been vented to 50 psig. The licensee believes that this lower pressure in combination with possibly leaking check valves from the primary system resulted in flashing conditions and water hammer. Plant procedures were changed to prohibit bleed down of the SITs.

Arkansas Nuclear One Unit 2

A mechanical snubber on the main steam supply line to the emergency feedwater (EFW) system turbine-driven pump was found inoperable because of severe degradation of the snubber internals (Ref. 22). The damaged snubber was found while conducting inservice inspections in accordance with the plant's Technical Specifications during a refueling outage in February 1988. The failure mode indicated that the

snubber had been subjected to an overload condition during the just completed operating cycle. The most probable cause was determined to be condensate buildup at a previously unidentified low point in the EFW steam supply piping and subsequent "water slugging" upon starting the turbine-driven EFW pump. Poor system maintenance and inadvertent bypassing of steam traps for extended periods also contributed to the condensate buildup. This same snubber was found inoperable in the course of performing inservice inspections during earlier refueling outages. During one of these refueling outages, snubber failure was attributed to excessive "steam slugging" resulting from a past problem with overspeed trips of the turbine-driven EFW pump. During another previous refueling outage, snubber failure was attributed to vibration and overload. This failure mode was common to a large number of mechanical snubbers. Corrective action included modifying the EFW steam supply line to remove the low point and inspecting, cleaning, and rebuilding, as necessary, the steam traps.

Dresden Units 2 and 3

A series of three events involving water hammer in the high pressure coolant injection (HPCI) system occurred at Dresden Units 2 and 3 (Ref. 23). Two of these events were reviewed by an NRC Augmented Inspection Team in October and November 1989 (Ref. 24). Preliminary indications of a precursor to the initial Unit 2 damage in the HPCI system were provided by increasing HPCI cubicle temperatures in May 1989. The pipe temperature at the HPCI pump was 140°F, while the piping between MOVs 2-2301-8 and 2-2301-9 was 160°F (see Figure 4). Further pipe temperature measurements taken in July 1989 found the HPCI pump discharge piping at 175°F, while the pipe between MOVs 2-2301-8 and 2-2301-9 was 220°F. Closing MOV 2-2301-9 produced pipe temperatures of 106°F after 12 hours. Pipe temperatures returned to 220°F after MOV 2-2301-9 was reopened. This showed that check valve 2-2301-7 and MOV 2-2301-8 were both leaking feedwater back to the HPCI system piping.

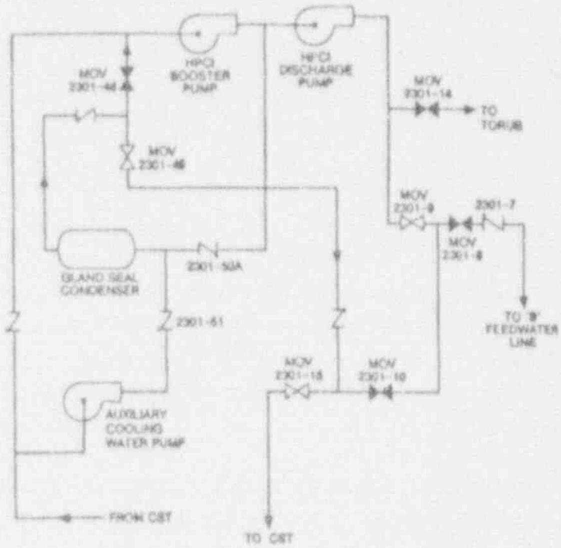

Figure 4

Dresden Units 2 and 3
Basic HPCI System, Normal Valve Line-up

A pipe temperature survey conducted in October 1989 revealed that the HPCI pump discharge pipe temperature was 246°F, while the temperature of the piping between MOVs 2-2301-8 and 2-2301-9 was 275°F. An evaluation determined that steam voids could form in certain sections of the pipe under these conditions. Inspection of the Unit 2 HPCI discharge piping supports found deficiencies in 47 percent (or 16) of the supports. The Unit 2 HPCI system valve lineup was changed so that the injection valve function was moved fro MOV 2-2301-8 to MOV 2-2301-9 (see Figure 5). Also, with MOV 2-2301-8 open, MOV 2-2301-10 becomes an isolation valve that is now subject to feedwater pressure.

As a result of the elevated temperatures discovered on the Unit 2 HPCI pump discharge piping, personnel investigated HPCI pump discharge piping at Unit 3. Temperature measurements obtained with an infrared thermometer in October 1989 found HPCI pump discharge pipe temperatures ranging from 256°F between MOVs 3-2301-8 and 3-2301-9, to 112°F at the discharge of the HPCI pump. This was evidence of possible steam void formations in the Unit 3 HPCI pump

discharge piping. Further monitoring of the piping temperature and pressure in early November 1989 revealed that the temperature in the piping was increasing. The temperature increase was attributed to leakage past MOV 3-2301-10. This valve was operated both electrically and manually in order to seat the valve. MOVs 3-2301-15 and -49 were also closed. After MOVs 3-2301-15 and -49 were closed, the pump discharge pressure increased from 650 psig to 1070 psig. This pressure increase confirmed that the leakage through MOV 3-2301-10 was successfully stopped.

Inspection of Unit 3 HPCI discharge pipe supports found deficiencies in 52 percent (or 21) of the supports. The Unit 3 HPCI system valve lineup was changed so that MOVs 3-2301-15 and -49 were normally closed and MOV 3-2301-48 was normally open (see Figure 6).

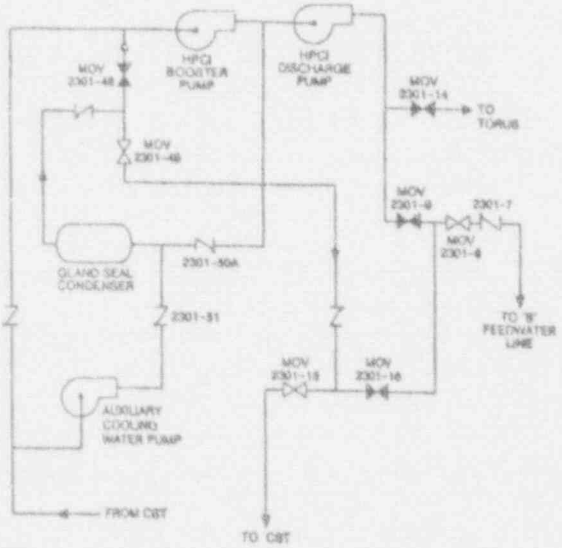

Figure 5

Dresden Unit 2 HPCI Alternate Valve Line-up
(Post-October 1989 Event)

Dresden Unit 2

In March 1990, after completing routine HPCI valve operability surveillance tests and while performing quarterly valve timing tests on the Unit

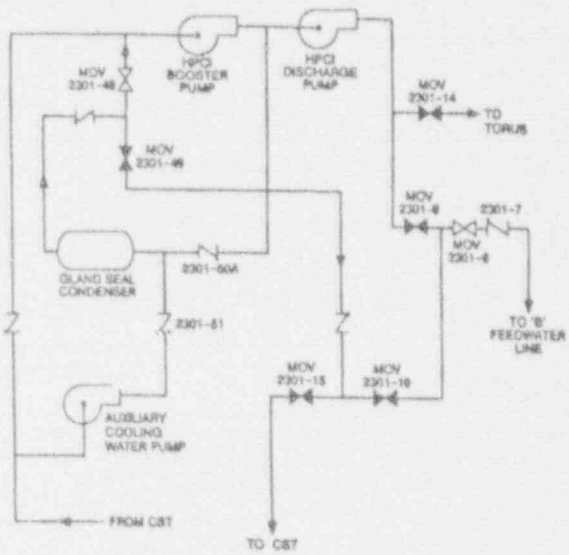

Figure 6

Dresden Units 2 and 3
HPCI Revised Alternate Valve Line-up
(Unit 2 Post-March 1990 Event and
Unit 3 Post-October 1989 Event)

2 HPCI pump discharge valves, the Unit 2 Shift Supervisor heard banging noises coming from the HPCI pump discharge piping (Ref. 23). The valve timing test was terminated and the HPCI system valve lineup was returned to the configuration shown in Figure 5. The pipe banging and motion was monitored until it eventually ceased about 1-1/2 hours later. Valve manipulation and temperature measurements along the HPCI pump discharge pipe led the licensee to conclude that feedwater leakage back through HPCI test return valve MOV 2-2301-10 was the root cause of this event.

Based on further investigation, the licensee postulated that MOV 2-2301-10 did not fully close after performing the HPCI system tests. However,

may or may not fully close when given a close signal. This valve does not have a seal-in feature to complete the stroke after closure initiation. In addition, the limit switches are set to provide a torque switch bypass function in the open direction until the valve is 25 percent open. This limit switch also controls the indicated "valve closed" light in the control room. Thus, an operator removing a closure signal shortly after the control room panel lights indicate that the valve has closed could leave the valve nearly 25 percent open. A procedure was introduced to continue the closure signal 30 seconds after the panel lights indicate MOV 2-2301-10 has closed. The Unit 2 HPCI system valve lineup was again revised, and left to match the HPCI system valve lineup for Unit 3 (see Figure 6).

DISCUSSION

From Table 1, it is evident that water hammer events have occurred in both Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs). The BWR events involved the shutdown cooling mode of RHR, the isolation condenser, and HPCI. These BWR systems have all been identified with water hammer events in previous studies. The PWR events involved main steam, AFW steam supply, steam generator blowdown, feedwater, accumulator (fill lines and injection to the reactor coolant system), and the AFW system vent lines. Most of these PWR systems have been associated with water hammer in previous studies.

The causes of these recent water hammer events are consistent with those reviewed in previous studies. However, previous studies did not emphasize the interactions between systems. For example, the event at South Texas Unit 1 was initially believed to be water hammer caused by improper venting of the AFW system. Subsequent evaluation determined that the flow control valves were introducing pressure fluctuations when they were in a highly-throttled position. The pressure fluctuations matched one of the natural frequencies of the piping system, causing pipe rupture.

The closure of shutdown cooling isolation valves has been identified with fluid systems interactions. These interactions appear to perturb monitored parameters resulting in undesired isolation signals. Shutdown cooling will be lost and water hammer may result if realignment procedures are not adequate. This phenomenon does not seem to be well understood since several BWR plants have experienced this problem and have attempted different solutions with varying degrees of success.

The recent water hammer events seem related to lack of implementation of the guidance issued in the resolution to USI A-I (Ref. 3). For instance, some causes cited were failure to fill and vent properly, rapid valve stroking, lack of guidance about system configuration, low point water accumulation, depressurization of a system which could cause local flashing, and bypassing steam traps. Thus, water hammer can result when plant staff are not vigilant concerning system conditions and changes in system arrangement.

The events at Trojan and Dresden Units 2 and 3 illustrate this situation. The events involved system alignment changes in which the significance may not have been fully appreciated. At Trojan, there was an attempt to transfer water to accumulator "D" from accumulator "A" via the fill lines for each accumulator because the sample lines, which had procedures for such application, were tagged out for maintenance. There were no procedures for transferring water between accumulators by use of the fill lines. There were two attempts at water transfer, and two pipe ruptures. These attempted water transfers used an approach that had not been reviewed for systems with a pressure differential of several hundred pounds per square inch. Thus, the need to fill the accumulator led to using an unauthorized approach without appropriate checks to satisfy safety considerations.

The series of water hammer events in the HPCI system at Dresden Units 2 and 3 were all related to operating the plant with leaking isolation valves. Plant operation with the leaking valves was initially attempted by using monitoring techniques intended to identify undesirable system temperature conditions. Subsequent efforts to operate with leaking valves involved both

monitoring and realignment of MOVs. Water hammer events occurred in each instance. Simple monitoring was inadequate because temperatures could still increase to the point where local steam voids could form and subsequent water hammer could occur.

HPCI system valve realignment at Dresden Unit 2 involved closing the normally open isolation valve outside containment (making this valve an injection valve), opening the normally closed isolation valve inside containment, and closing the condensate storage tank return valve (making this valve a pressure isolation valve). Realigning the valves in this arrangement has several features of interest: 1) assurance of MOV operability against full differential pressure (about 1200 psi), 2) a relationship between the new alignment and valve position changes required for MOV tests, and 3) MOV control features such as signal seal-in or torque switch bypass aspects.

Also at Dresden, the normally closed HPCI injection valve (which leaked) was evaluated for operation against full differential pressure (1135 psi) as a result of IE Bulletin 85-03, *Motor Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings*, issued November 15, 1985 (Ref. 25). However, the MOVs used in the realigned configuration (normally open isolation valve and condensate return valve) were not evaluated for operation against high differential pressures until NRC Bulletin 85-03, Supplement 1, was issued on April 27, 1988 (Ref. 26). Supplement 1 addresses the inadvertent operation (closure or opening due to mispositioning) of motor-operated valves. Thus, a system realignment prior to April 1988 could have resulted in placing MOVs in a configuration for which operability requirements may not have been adequately addressed.

Plant Technical Specifications at Dresden still required MOV stroke tests and HPCI system flow tests as part of the inservice test (IST) program. In order to conduct these tests, it was necessary to use the leaking valves temporarily as isolation valves against the feedwater system pressure. This aspect, in conjunction with the condensate return valve that did not have a seal-in feature for the closure signal eventually lead to formation of local voids and water hammer in the HPCI system. (The valve was 25 percent open when the control panel light indicated closed. The control room light indicated closed because the MOV control features used the same limit switch to set the open torque switch by-pass and the closing light indication; this approach may still be used in several plants.) Therefore, a relatively simple realignment of MOVs to provide HPCI injection evolved into a complex situation requiring detailed knowledge of subcomponent control features and settings and system (component) operational requirements (including IST tests) in order to protect the HPCI system from potentially damaging water hammer events as well as providing assurance for injection capability.

FINDINGS

Evaluation of the data identified from the search for water hammer events for January 1986 through March 1990, resulted in the following findings:

1. There were 12 water hammer events identified from January 1986 through March 1990. This indicates the event frequency is dropping compared with previous studies (148 events from 1969 to 1980 and 40 events from 1981 to 1985).

2. The causes of the 12 recent water hammer events reviewed in AEOD/E91-01 (Ref. 9) were similar to causes identified in previous studies. Thus, there were no new phenomena identified.

3. The 12 water hammer events appear related to either a failure to implement the guidance issued in the resolution to USI A-1); a less than vigilant attitude concerning system conditions, operations, or changes in system alignments that could result in water hammer; or an insufficient understanding of system conditions, including component operational characteristics, that could cause water hammer.

4. The causes of the 12 water hammer events include failure to fill and vent properly, too rapid valve stroking, lack of guidance about system configuration, water accumulation at low points, depressurization of a system which could lead to local flashing, and bypassing steam traps.

5. Some water hammer events show that hydrodynamic interactions between systems may occur.

6. System realignments involving MOVs may involve very complex issues that could affect component operability or possibly result in water hammer events.

7. The water hammer event at Susquehanna 2 involved closure of the letdown isolation valve during attempts to use the shutdown cooling mode of RHR. The RHR pump tripped and shutdown cooling was lost when the isolation valve closed. Several events similar to the Susquehanna 2 event with respect to the loss of shutdown cooling were identified (Ref. 9).

CONCLUSIONS

The assessment process for USI A-1 was based on a disciplined approach to review each event and identify affected systems, determine event frequency, and establish the phenomena that caused the water hammer.

Based on the similar evaluation performed in AEOD/E91-01 (Ref. 9), it was found that the 12 reported water hammer events represent a reduction in frequency of occurrence of water hammer at operating plants and that no new physical phenomena were identified as causes of water hammer. Therefore, the recent operating experience is consistent with the resolution conclusions for both USI A-1 and the subsequent reassessment after the San Onofre event in 1985. Thus, there appears to be no need to revise NUREG-0927.

Although these recent water hammer events are comparable with those reviewed in previous studies, there are some aspects that were not previously emphasized which impact safety. The specific areas include:

1. failure to implement the guidance issued in the resolution to USI A-1,

2. hydrodynamic interaction between systems,

3. system realignments involving MOVs that may involve complex issues concerning component operability as well as water hammer, and

4. closure of letdown isolation valves in the shutdown cooling mode of RHR for BWR plants.

The first three areas were clearly evident in the 12 water hammer events reviewed. The fourth area, as demonstrated by the water hammer event at Susquehanna 2, shows that the closure of the letdown isolation valves can cause water hammer events. However, closure of the isolation valve will also result in loss of shutdown cooling.

NRC issued Information Notice (IN) 91-50 to nuclear power plant licensees as a result of this study. This IN discussed the events reviewed in this report, indicated that NUREG-1027, Revision 1, (resolution of USI A-1), addressed many of the causes cited by these water hammer events, and alerted the industry about hydrodynamic interactions between systems as well as system realignment concerns.

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11. ABSTRACT (200 words or less)

Power Reactor Events & Issues is published by the Office for Analysis and Evaluation of Operational Data (AEOD) of the U.S. Nuclear Regulatory Commission (NRC). This publication presents assessment of selected operating events at U.S. nuclear power plants. The objective of this periodical is to feedback the results of NRC-sponsored analysis of pertinent power reactor events and issues to the nuclear industry, specifically, to the Operations, Maintenance, and the Training Managers at all the operating reactor sites. The feature article in this issue is based on AEOD Engineering Evaluation Report AEOD/E91-01, "A Review of Water Hammer Events After 1985," written by E. J. Brown. This study was initiated following several instances of water hammer involving the service water system at Arkansas Nuclear One. The task was to evaluate the need to reissue previous NRC guidance about water hammer or to suggest additional measures to prevent or mitigate their occurrence. This study concluded that the frequency of reported water hammer occurrences continues to drop and no new phenomena were identified as causes of water hammer. This study supports prior conclusions regarding water hammer; however, some aspects that could impact safety and were identified in the study have not been previously emphasized.

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TO: RECIPIENTS

SUBJECT: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA PUBLISHES POWER REACTOR EVENTS & ISSUES (NUREG/BR-0171)

Enclosed is the first issue of *Power Reactor Events & Issues* (PREI), NUREG/BR-0171, Vol. 1. This periodical replaces an earlier AEOD bi-monthly publication, *Power Reactor Events*, NUREG/BR-0051, that was discontinued in 1989. Our plans are to issue PREI semi-annually, with the capability of issuing special reports when events warrant a more timely issuance.

In Spring of 1992, we sent the draft issue of PREI to the NRC program offices, regions and the industry-supported organizations for their comments on the potential usefulness of this publication. The highlights of this peer review were that such a publication would be beneficial to the plant staff provided it was not redundant to other NRC feedback documents, did not carry a regulatory tone, and if the information presented was factual and non-judgmental.

Each future issue of PREI will normally contain at least one article based on an AEOD study of U.S. reactor operating experience. We plan to use this periodical to provide timely feedback of reactor operational experience, both domestic and foreign, without duplication of information contained in other NRC generic communications. Using a broad dissemination, our goal is to feedback operating experience to the plant site management (i.e., operations, maintenance, and training organizations) as well as to the industry-supported organizations and the NRC staff.

In order to make the future issues of this periodical of interest to you, please use the response form attached to this letter to provide your comments regarding the usefulness of the information presented in PREI. We would also welcome suggestions for topics you would like to see covered in the future issues.

T M Novak

Thomas M. Novak, Director
Division of Safety Programs
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Enclosure: As stated

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