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WATER HAMMER IN NUCLEAR POWER PLANTS

Office of Nuclear Reactor Regulation
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Division of Systems Safety
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
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1.0 INTRODUCTION

In the past few years, licensees of operating reactors have reported a large number of water hammer events during commercial operation. Most of these events resulted in damage to piping supports and restraints. A few cases involved small cracks or ruptures in feedwater systems. As a result, in 1977 the NRC staff initiated a review of reported water hammer events and of the potential for occurrence of water hammer in all fluid systems that could have an impact on plant safety. The objectives of the review were to identify the causes of water hammer events that could affect reactor safety and to recommend further staff actions needed to reduce the likelihood of such events.

The term water hammer, as used in this review, was generalized in meaning to include transients involving steam flow (steam hammer) and two phase flow (e.g., water entrainment in steam lines, steam bubble collapse) in addition to the classical water hammer transients such as those involving valve closing and pump startup in solid water systems. Water hammer in vessels (quench tank, torus) and pressure pulsations during steady operation (e.g., from positive displacement pumps, cavitating valves) were not included in the review. The review was also limited to fluid systems with lines larger than 1 inch in diameter.

The staff contacted nuclear steam system suppliers and architect-engineers to determine the extent to which water hammer has been considered in the design of the fluid systems for nuclear power plants. Piping code requirements and NRC licensing procedures were also reviewed in order to assess the requirements imposed on the plant designs heretofore. Other sources of information included Licensee Event Reports and responses by licensees to staff information requests. The staff also performed an independent review of the fluid systems to identify potential water hammer situations.

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The review was limited to one reactor design from each of the four reactor vendors. This limitation in scope is not considered significant because the recommendations are generic.

After the start of this review, the generic consideration of water hammer in nuclear power plants was incorporated into Task Action Plan A-1, "Water Hammer," which is described in NUREG-0371, "Approved Task Action Plans for Category A Generic Activities." Part of the task description from NUREG-0371 is reproduced in Appendix C for reference. This report on the results of the review is the first of several reports scheduled to be issued under Task Action Plan A-1.

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2.0 SUMMARY

The staff reviewed information on water hammer events obtained primarily from Licensee Event Reports and information requests to licensees. Approximately 30 fluid systems in each of four light water reactor designs were also reviewed to determine the potential for significant water hammer events. Of these, 10 systems were considered of importance either because of their safety function or the potential effect of water hammer events on the integrity of the reactor coolant pressure boundary.

In the review, the following classification of water hammer problems was adopted:

1. Pump startup with inadvertently voided discharge lines
2. Expected flow discharge into initially empty lines
3. Valve opening, closing and instability
4. Check valve closure and delayed opening
5. Water entrainment in steam lines
6. Column separation
7. Steam bubble collapse and mixing of subcooled water and steam from interconnected systems
8. Slug impact due to rapid condensation
9. Pump startup, stopping and seizure

Detailed discussions of these water hammer problems are presented in Appendix A.

The 10 systems of importance are presented in Table 2.1 along with the above water hammer concerns. On the basis of reactor operating experience, the most serious water hammer concerns are slug impact due to rapid condensation in certain PWR steam generators, pump startup with inadvertently

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TABLE 2.1 POTENTIAL WATER HAMMER EVENTS

System* / Event*	RCS	RHRS	ECCS	Main Steam System & Steam for Aux. FW RCIC & HPCI Turbines	Main & Aux. FW System	Component Cooling Water/ Service Water Systems	RCIC System	RWCU System
Pump Startup w/ Inadvertently Voided Discharge Lines		X	X		X	X	X	
Expected Flow Discharge into Initially Empty Lines	X[1]	X	X	X		X	X	
Valve Opening, Closing, and Instability	X	X	X	X	X	X	X	X
Check Valve Closure & Delayed Opening		X	X	X	X	X	X	
Water Entrainment in Steam Lines		X[2]	X[3]	X				
Transient Cavitation (Column Separation)	X	X	X		X	X	X	
Steam Bubble Collapse & Mixing of Subcooled Water & Steam from Interconnected Systems		X[2]	X			X		
Slug Impact Due to Rapid Condensation					X			
Pump Startup, Stopping, & Seizure	X	X	X		X	X	X	

*Combination of dynamic loads with OBE or SSE loads required for some events and systems

1 - Safety and Relief Valve Discharg
2 - BWRs only
3 - Turbine drives

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voided lines in ECC and RHR systems of BWRs and main feedwater line transients caused by flow control valves in BWRs and PWRs.

While the incidence of reported water hammer events at operating nuclear power plants has been relatively large, about 120 events in 400 reactor years of operation, only a few events during commercial operation have resulted in failure of pipe integrity. These failures were limited to small cracks or small broken lines or valves in the feedwater system. None of these events affected the health and safety of the public. The staff concludes that continued plant operation and licensing is warranted pending completion of Task Action A-1 (see Section 3.0 of Appendix C). However, as a result of this review, the staff concluded that the overall frequency of water hammer events was unnecessarily large and that steps in design and in plant operation should continue to be pursued with the goal of reducing the event frequency.

During the course of this review, recommendations were developed for preparation or modification of regulatory guides, specific actions for operating reactors, and initiation of technical studies and more detailed evaluations of specific water hammer concerns which are needed either to develop staff positions on water hammer or improve predictions of dynamic loads. With the exception of recommendations concerning regulatory guides and studies concerning safety valves with loop seals and feedwater check valves, all of the recommendations developed during the review have already been implemented or will be implemented under Task Action Plan A-1. These completed or scheduled staff actions and recommendations are summarized in Section 4.0 where they are related to particular tasks in Task Action Plan A-1 and to the detailed discussions of Appendix A.

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3.0 BACKGROUND

3.1 Fluid Transient and Mechanical Effects

Water hammer may be defined as the pressure change in the liquid in a closed conduit caused by a gradual or sudden change in the liquid velocity. A standard example in the texts on water hammer is the calculation of the pressure changes resulting from an instantaneous closure of a valve downstream of a reservoir. At the instant of valve closure, the layer of liquid next to the valve is stopped with a resultant instantaneous increase in pressure at the valve. The deceleration required to stop successive layers of the liquid leads to a compression pressure wave which propagates back towards the reservoir at the speed of sound, c, in the pipe. The compression wave is reflected at the reservoir as a rarefaction wave which returns at the same speed to the valve where it is reflected as another rarefaction wave. Tracking of the successive compression and rarefaction waves produced by these reflections at the valve and reservoir after the initial valve closure provides a history of pressure and velocity in the conduit and shows that the conditions existing just prior to valve closure are reached at the time $t = \frac{4L}{c}$ where L is distance between the valve and reservoir. In the absence of friction, this cycle of events would be repeated indefinitely with the period $\frac{4L}{c}$.

The magnitude of the pressure change, Wp, associated with either the compression or rarefaction waves is given by the relation:

$$Wp = pcv$$

where p is the liquid mass density, and v is the initial liquid velocity. The wave speed, which is determined primarily by the compressibility of the liquid, may be reduced by the elasticity of the pipe by about 20

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percent for some systems.[1] Uniformly entrained gas bubbles cause large reductions in the wave speed. For example, with 0.1 percent gas by volume, the wave speed in water is reduced by roughly 50 percent.[1]

For finite valve closure rates, the maximum pressure at the valve would still be given by Equation 1 provided the valve closure time is less than the transit time, 2L/c required for the first rarefaction wave to reach the valve. For typical piping lengths and wave speeds in nuclear plants, this transit time would be in the order of 100 milliseconds. Most power-operated valves for water systems have closure times ranging from several to 100 seconds. Consequently, the pressure increase generally would be much smaller than that given by Equation 1. However, sudden changes in feed water control valve position have occurred and resulted in large pressure changes.

Wood and Jones[2] have developed water hammer charts for estimating the peak pressure rise resulting from the closures of valves downstream of a reservoir. Depending on the valve closure characteristics, the peak pressure rise is about one-tenth that of Equation 1 for closure times from 10 to 100 times the transit time, 2L/c. However, it should be noted that the local pressure increase is not the only cause of water hammer damage. As discussed later, most of the reported damage can be attributed to forces produced during the transient at pipe bends and flow area changes. These forces cause pipe movement with resultant damage to such components as pipe hangers and restraints, and valve operators.

In nuclear plants, as in conventional plants, there are many mechanisms for initiating water hammer aside from valve closures. Pump starts and stops cause flow and pressure changes. The pressure pulses and dynamic loads from this source generally are small because of the relatively small rate of change in pump speed. Initiation of flow in a voided or partially voided line can result in significant dynamic loads when the liquid front reaches restrictions or directional changes in the piping.

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Although these mechanisms for initiating water hammer are different, the dynamics of the unsteady liquid motion and the associated dynamic loadings are similar and have been analyzed by a variety of methods in the past. Discussions of some of these classical water hammer problems in liquid systems and the methods of solution of the fluid transients are presented in Reference 1.

Other types of water hammer occurrences resulting from steam-water interactions cannot be predicted with sufficient accuracy for design calculations. For example, steam bubble collapse in water systems due to condensation and pressurization following pump startup may depend on factors such as bubble size and local condensation rate which can vary over wide ranges. Similar uncertainties arise in the calculation of the impact of water slugs in the feedrings of PWR steam generators. The predicted mechanical loads caused by the impact of steam-driven slugs of water in steam lines are also subject to large uncertainties since little is known about the quantities and distribution of the entrained water. In view of these uncertainties in the predicted loads, the only feasible design approach is to try to prevent the occurrence of water hammer due to these initiating mechanisms.

The present approach to the analysis of the piping system response to an expected water hammer occurrence is a two-stage process involving the calculation of transient water hammer pressures and forces which are then used as input to a structural analysis to obtain the piping and component stresses and support loads as a function of time. The two calculations are decoupled since the fluid transient calculation is based on the assumption that the piping network is rigidly fixed in space (i.e., the effect of piping deflections on the fluid transient response is not treated). However, the effect of local elastic or plastic behavior of the pipe walls may be included by modifying the wave speed of the pressure pulse.

The fluid transient conditions are calculated by solving the one-dimensional equations for pipe flow subject to boundary conditions representing the problem of interest. At components where the fluid imposes a force on the piping network, such as elbows and valves, the one-dimensional transient pressures and velocities are converted to time-dependent forces by application of Newton's second law to a control volume representing the component.

The computer codes WHAM (Ref. 3) and PTA (Ref. 4) are examples of the methods used to compute the fluid transient conditions for a liquid. Both codes have been used in calculations of three-dimensional piping networks and can be used to treat typical components such as pumps, elbows, junctions, and dead ends. The WHAM code uses the method of wave superposition to solve the one-dimensional water hammer equations. The effects of pipe friction are included by using discrete orifices between sections of pipe having negligible wall friction. The code approximates the effects of locally elastic pipe wall behavior by using a wave speed formulation in which the sonic speed is decreased by the effects of elasticity of the pipe wall. The PTA code uses the method of characteristics to solve the one-dimensional water hammer equations, treats continuous pipe wall friction by using steady-state flow friction factors, and approximates the effects of both elastic and inelastic pipe wall behavior by using a modified wave speed.

For compressible flow situations such as encountered in the calculation of the effects of steam hammer resulting from turbine stop valve closure, solution of the one-dimensional compressible flow equations by the method of characteristics and finite difference approximation to the flow equations can be used (see, e.g., Refs. 5, 6). For two phase flow situations such as encountered following the actuation of safety valves with loop seals, finite difference approximations to the flow equations have been used (e.g., Refs. 6, 7, 30).

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Two modes of failure are considered in evaluating the consequences of dynamic loads resulting from water hammer. The first type of failure is immediate due to the loads on the component caused by the water hammer. This type of failure ordinarily is ductile where the stresses exceed the yield or ultimate strengths of the material. The second type of failure results from the cumulative effect of the loads due to successive water hammer and other events. This type of failure occurs when the number of stress cycles exceeds the fatigue limit of the material. Because the failure results in crack initiation and growth, it can be detected in many instances by nondestructive testing before failure occurs. Typically, such crack growth in a pressure boundary will lead to a "leak before break" condition, thus aiding in detection.

The system components may be treated in terms of three classes: (1) those components or parts of components forming the pressure boundary of the system; (2) component parts internal to the pressure boundary and (3) component parts external to the pressure boundary.

For those components forming the pressure boundary of the system, the local pressure increases during the water hammer occurrence result in increases in the circumferential stress and a potential for ductile failure due to bulging or splitting. Loads are also transmitted to these components as the result of forces which are produced at various locations in the system where there is a change in pipe direction or flow area. These loads can cause pipe "jump" and result in axial forces and bending and torsional moments.

Components internal to the system pressure boundary include, for example, valve trim and pump impellers. These internal components cause partial or full reflection of the pressure waves and are subjected to asymmetric loading. Components external to the system pressure boundary include, for example, pipe hangers, snubbers, valve operators, and pump motors. These components are subjected to dynamic loads resulting from the pipe "jump" and, possibly, from impact with adjacent structures.

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There are numerous structural analysis codes which could be used to evaluate the response of piping systems to the dynamic loads obtained from fluid transient codes. Many of the codes are proprietary. However, one general purpose structural analysis code which is readily available is SAP IV (Ref. 8). An example of the application of this code to a PWR feedwater system which was approximated by finite element models of a collection of straight and elbow pipe elements is given in Reference 9.

3.2 Current Practice

In 1977, the staff obtained information from reactor suppliers and architect-engineering firms regarding their design procedures for accounting for water hammer. Most of the organizations have methods for calculating the dynamic loads associated with certain types of water hammer and then determining the mechanical effects in the piping systems. However, the approaches to the problem vary appreciably. This stems, in part, from lack of specificity in the code requirements. For piping design, ASME Boiler and Pressure Vessel Code, Section III, requires the designer to consider impact forces caused by either external or internal conditions. ANSI Codes for Pressure Piping, B 31.1 (1973) and B 31.7 (1969), also state that impact loads should be considered. None of these codes provides guidance as to load type, magnitude, pulse shape, or the type of analysis that should be performed.

In view of past experience, many of the organizations identified the feedwater and safety-related systems as areas in which water hammer is addressed. There is also a reliance on startup tests to demonstrate acceptable performance of the different fluid systems. Approaches used at the design stage to reduce the impact of potential water hammer include (a) increasing valve closure times, (b) uses of piping layouts to preclude water slugs in steam lines and vapor formation in water lines, (c) use of snubbers and pipe hangers, and (d) use of vents and drains.

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The Standard Review Plan (SRP), which is used by the staff for construction permit (CP) and operating license (OL) evaluations, includes a review of dynamic analyses and field test procedures to assure that the pipe support systems are adequately designed. However, water hammer is specifically identified in the SRP only for the feedwater and condensate system and ECC system. In addition, Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," requires testing of various fluid systems and Regulatory Guide 1.70, "Standard Safety Analysis Report Format," requests information on dynamic analyses and testing of all safety-related ASME Class 1, 2, and 3 piping systems.

The current Standard Review Plan (Section 3.9.2) specifies that piping vibrational and dynamics effects testing should be conducted during the initial testing program (preoperational testing and initial startup testing). The systems to be monitored should include: (a) ASME Code Class 1, 2, and 3 systems, (b) other high-energy piping systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level, and (c) seismic Category I portions of moderate-energy piping systems located outside of containment. The supports and restraints necessary for operation during the life of the plant are considered to be part of the piping systems. The purpose of these tests is to confirm that these piping systems, restraints, components and supports have been adequately designed to withstand flow-induced dynamic loadings, including those which may be due to water hammer, if the design calls for mitigating the consequences of water hammer forces.

Typical systems tested include:

. Emergency Core Cooling System
. Charging System

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- Letdown System
- Pressurizer Spray
- Shutdown Cooling System
- Steam Generator Blowdown System
- Boric Acid System
- Reactor Coolant Sampling
- Reactor Coolant Pump Controlled Bleed Off
- Atmospheric Dump and Turbine Dump/Bypass Lines

3.3 Water Hammer Experience in Light Water Reactors

The Technical Specifications issued when a nuclear power plant is licensed require submittal of Licensee Event Reports (LERs) in accordance with Regulatory Guide 1.16. This guide requires reporting of events, such as water hammer, that have an impact on the operation of safety-related equipment; however, if resultant damage is minimal, it need not be reported. Similarly, water hammer events in nonsafety-related systems need not be reported unless there is resultant damage in a safety system.* Events requiring a prompt notification (see Reg. Guide 1.16) result in site inspection and issuance of a staff inspection report. LERs of a less severe nature are reviewed by the NRC on a sampling basis. When events appear to be generic in nature, the Office of Inspection and Enforcement (IE) issues information circulars or bulletins to alert all potentially affected licensees.

NRC has conducted generic reviews and required corrective actions by licensees with respect to two types of water hammer in operating plants.

*Water hammer occurrences in some portions of secondary systems may affect the reactor coolant pressure boundary and safety-related systems. Hence, it is recommended in Section 4.2 of this report, that Regulatory Guide 1.16 be modified to require reporting of significant water (steam) hammer occurrences in the main steam lines up to and including the main turbine stop/control valve and the main feedwater lines back to the feed pumps.

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As the result of damaging water hammer events in certain PWR steam generators with top feedwater rings, letters requesting information pertinent to this type of water hammer were sent to licensees in 1975. In 1976, NRC engaged an engineering firm (CREARE, Inc.) to undertake a study of the cause and effects of past steam generator water hammer incidents. The results of this study were published in NUREG-0291 (Ref. 9). NRC actions with respect to PWR steam generator water hammer are summarized in Section A.8. As the result of damaging water hammer due to inadvertent closure of certain check valves in the main steam lines of PWR plants, information requests were sent in 1974 to licensees using these valves. The results of this review and specific corrective actions taken by the staff are summarized in Section A.4.

The information on water hammer events presented in this section was obtained primarily from a survey of Licensee Event Reports. As noted above, this source of information is not complete. Nevertheless, it is believed that the review of LERs has provided a reasonable assessment of the significant water hammer events that have occurred. The staff obtained information on some other water hammer events from information requests sent to licensees during the generic reviews noted above, from IE information requests in January 1974 concerning abnormal occurrences in piping systems and from informal contacts with some licensees in 1977. The information on water hammer events obtained in this manner is summarized in Tables 3.1 and 3.2. Where possible, the events are classified according to the initiating mechanism. Table 3.1 contains a summary of water hammer events due to rapid condensation downstream of a water slug in feedwater rings in some PWR steam generators. Table 3.2 summarizes other water hammer events in plant systems. Over half of the events listed in Tables 3.1 and 3.2 occurred during commercial operation.

Reactor plant experience relating to the steam generator feedwater water hammer phenomenon was reported by about a dozen PWR power plant licensees in response to the NRC requests in May 1975. The responses provided the

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TABLE 3.1

WATER HAMMER EVENTS IN FEEDWATER RINGS OF PWRS

Plant	NSSS Vendor	Commercial Operation	Date of Steam-Water Slug Water Hammer Event
Calvert Cliffs 1	CE	05/75	08/29/74, 12/30/74, 05/12/75*
Calvert Cliffs 2	CE	04/77	05/18/76*
D.C. Cook	W	07/75	01/02/76,* 03/10/77
Haddam Neck (Conn. Yankee)	W	01/68	*
Indian Point 2	W	09/73	11/13/73
San Onofre 1	W	01/68	04/29/72, 01/14/74
Surry 1	W	12/72	10/01/72
Turkey Point 3	W	07/72	01/14/73
Turkey Point 4	W	04/73	06/30/73, 01/05/74
Zion 1	W	06/73	09/26/76, 07/08/77*
Zion 2	W	12/73	08/29/74, 12/30/74, 05/76, 06/20/76, 07/10/77
Yankee Rowe	W	06/61	Before 1966
Millstone	CE	12/75	05/09/75*

*Reported as nondamaging.

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TABLE 3.2 OTHER WATER HAMMER EVENTS

Water Hammer Type	System	Plant	Reactor Type	Commercial Operation	Event Date	Damage
Pump Startup w/Inadvertently Voided Discharge Lines	RHR	Duane Arnold	BWR	02/01/75	04/06/77	Yes
	RHR	Hatch 1	BWR	12/31/75	12/15/74	Yes
	RHR	Quad Cities 1	BWR	02/18/73	04/04/72	Yes
	RHR	Brunswick 1	BWR	03/18/77	11/09/77	Yes
	RHR	Brunswick 2	BWR	11/03/75	09/05/75	Yes
	RHR	Brunswick 2	BWR	11/03/75	09/30/75	Yes
	RHR	Fitzpatrick	BWR	07/28/75	03/21/75	Yes
	RHR	Fitzpatrick	BWR	07/28/75	05/24/75	Yes
	RHR	Brunswick 2	BWR	11/03/75	09/30/75	No
	Core Spray	Dresden 2	BWR	06/09/72	03/29/71	Yes
	Core Spray	Dresden 2	BWR	06/09/72	07/11/76	Yes
	Core Spray	Dresden 3	BWR	11/16/71	11/27/74	Yes
	Core Spray	Duane Arnold	BWR	02/01/75	04/10/74	Yes
	Core Spray	Duane Arnold	BWR	02/01/75	02/11/77	Yes
	Core Spray	Oyster Creek	BWR	12/69	05/25/71	Yes
	SSI Discharge Line	San Onofre 1	PWR	11/01/68	10/20/73	Yes
	HPCI	Brunswick 2	BWR	11/03/75	03/24/78	---
	Emergency Equipment Cooling Water	Brown Ferry 1,2,3	BWR	08/01/74 03/01/75 03/01/77	------ ------ ------	--- --- ---
	RHR Service Water	Fitzpatrick	BWR	07/28/75	04/10/74	Yes
	Service Water	Salem 1	PWR	06/30/77	1977	---
	Main FW	Oconee 3	PWR	12/16/74	------	Yes
	LPSI	Palisades 1	PWR	12/31/71	05/14/74	Yes
Expected Transients in Initially Empty Lines	Main Steam Safety Valve	Robinson	PWR	03/07/71	04/70	Yes
	Press. Relief Valve Discharge line	Surry 1	PWR	12/22/72	01/73	Yes

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TABLE 3.2 OTHER WATER HAMMER EVENTS (Cont'd)

Water Hammer Type	System	Plant	Reactor Type	Commercial Operation	Event Date	Damage
Valve Opening, Closing and Instability	Main Steam	Arkansas 1	PWR	12/19/74	06/15/76	Yes
	Main Steam	Millstone 1	BWR	03/71	------	Yes
	Main Steam	Oconee 1	PWR	07/15/73	03/21/75	No
	Main Steam	Oconee 2	PWR	09/09/74	------	Yes
	Main Steam	Turkey Point 3	PWR	12/14/72	11/05/75	Yes
	Main Steam	Millstone 1	BWR	03/71	12/28/70	Yes
	Main Steam	San Onofre 1	PWR	01/01/68	01/14/74	Yes
	Main Steam	Oyster Creek 1	BWR	12/69	1971	Yes
	Main Steam	Dresden 2	BWR	06/09/72	09/28/71	---
	Main Steam	Pilgrim 1	BWR	12/72	07/24/72	Yes
	Main FW	Beaver Valley 1	PWR	10/01/76	12/27/76	Yes
	Main FW	Beaver Valley 1	PWR	10/01/76	11/05/76	Yes
	Main FW	Beaver Valley 1	PWR	10/01/76	01/05/77	Yes
	Main FW	Quad Cities 2	BWR	03/10/73	06/10/74	Yes
	Main FW	Quad Cities 2	BWR	03/10/73	08/12/75	Yes
	Main FW	Quad Cities 2	BWR	03/10/73	08/31/75	Yes
	Main FW	Turkey Point 3	PWR	12/14/72	------	No
	Main FW	Turkey Point 4	PWR	09/07/73	------	No
	Main FW	Pilgrim 1	BWR	12/72	01/06/76	Yes
	Main FW	Dresden 2	BWR	06/09/72	12/05/74	Yes
	Main FW	Dresden 2	BWR	06/09/72	02/18/76	Yes
	Main FW	Quad Cities 2	BWR	03/10/73	10/17/75	Yes
	Main FW	Dresden 3	BWR	11/16/71	06/23/74	Yes
	Main FW	Dresden 3	BWR	11/16/71	09/04/74	Yes
	Main FW	Oconee 1	PWR	07/15/73	------	No
	Main FW	Oconee 2	PWR	09/09/74	------	Yes
	Main FW	Oconee 3	PWR	12/16/74	------	Yes
	Main FW	Ginna	PWR	07/70	07/22/73	Yes
	Main FW	Millstone 1	BWR	03/71	12/26/74	Yes
	Main FW	Zion 1	PWR	12/31/73	06/20/76	No
	Main FW	Zion 2	PWR	09/17/74	03/18/75	Yes
	CVCS	Maine Yankee	PWR	12/28/72	06/02/77	No

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TABLE 3.2 OTHER WATER HAMMER EVENTS (Cont'd)

Water Hammer Type	System	Plant	Reactor Type	Commercial Operation	Event Date	Damage
Check Valve Closure and Delayed Opening	Main Steam	Surry 1	PWR	12/22/72	10/05/73	Yes
	Main Steam	Surry 2	PWR	05/01/73	12/02/72	Yes
	Main Steam	Point Beach 2	PWR	10/01	04/08/73	Yes
	Cooling System for Reactor Auxiliaries	Calvert Cliffs 1	PWR	05/08/75	04/25/77	Yes
	Main Steam	Maine Yankee	PWR	12/28/72	12/02/72	No
	Recirculation Spray	Surry 1	PWR	12/22/72	11/29/74	Yes
	Containment Spray	Surry 1	PWR	12/22/72	12/27/74	Yes
	Recirculation and Containment Spray	Surry 2	PWR	05/01/73	12/27/74	Yes
	Core Spray	Millstone 1		03/71	04/17/78	Yes
Water Entrainment in Steam Lines	HPCI	Browns Ferry 1	BWR	08/01/74	10/14/72	Yes
	HPCI	Browns Ferry 1	BWR	08/01/74	04/14/74	Yes
	HPCI	Brunswick 2	BWR	11/03/75	08/30/76	Yes
	HPCI	Duane Arnold	BWR	12/01/75	06/11/74	Yes
	HPCI	Fitzpatrick	BWR	07/28/75	07/20/75	Yes
	HPCI	Fitzpatrick	BWR	07/28/75	07/28/75	Yes
	HPCI	Dresden 2	BWR	06/09/72	05/28/70	Yes
	RCIC	Browns Ferry 1	PWR	08/01/74	10/14/72	Yes
	Aux. FW Turbine	Palisades	PWR	12/31/71	------	Yes
	Aux. FW Turbine	Zion 1	PWR	12/31/73	06/16/74	Yes
	Isolation Condenser	Millstone 1	BWR	03/71	03/11/78	Yes

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TABLE 3.2 OTHER WATER HAMMER EVENTS (Cont'd)

Water Hammer Type	System	Plant	Reactor Type	Commercial Operation	Event Date	Damage
Steam Bubble Collapse and Mixing of Subcooled Water Steam from Inter-connected systems	Plant Steam Heating System	Big Rock Point 1	BWR	03/29/63	10/31/77	No
	HPCI	Monticello	BWR	06/30/71	1972	Yes
	Main FW	Oconee 1	BWR	07/15/73	------	No
	RHR	Brunswick 1	BWR	03/18/77	03/18/77	Yes
	RHR	Brunswick 2	BWR	11/03/75	12/29/76	Yes
	RHR	Dresden 2	BWR	06/09/72	09/28/71	Yes
	RHR	Monticello	BWR	06/30/71	------	Yes
	SIS Accumulator	Surry 1	PWR	12/22/72	11/20/74	Yes
	SIS Accumulator	Surry 2	PWR	05/01/73	1974	Yes
	RHR	Brunswick 2	BWR	11/03/75	04/18/77	Yes
	HPCI	Brunswick 1	BWR	03/18/77	05/20/77	Yes
	Main FW	Zion 1	PWR	12/31/73	12/31/73	No
	Main FW	Rancho Seco	PWR	04/17/75	1974	Yes
	Steam Generator Blowdown Line	Haddam Neck 1	PWR	01/01/6	03/17/78	Yes
Pump Startup	HPCI	Browns Ferry 3	BWR	03/01/77	01/18/77	Yes
	RHR	Pilgrim 1	BWR	12/72	10/17/75	Yes
Not Identified	Steam Generator Blowdown Line	Indian Point 2	BWR	08/73	------	Yes
	RHR steam line	Brunswick 1	BWR	03/18/77	12/20/77	Yes

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basis for our review of water hammer events believed to arise as a result of slug impact in the feedwater lines following uncovering of the feedwater rings. Although the cause would appear to be generic in nature, less than one-third of the operating PWRs have reported water hammer events associated with uncovering the feedwater ring inside the steam generator. These plants and dates of the events are shown in Table 3.1. An asterisk identifies events in which no observable damage was sustained in the plant. In the other cases, damage was noted to various components. Examples are: damaged feedwater piping hangers and feedwater isolation valve operators (Calvert Cliffs), crack in an 18-inch feedwater line inside containment at the wall penetration (Indian Point 2), failure of feedline check valve (Surry, Turkey Point), local permanent displacement of the feedwater piping (Surry, Calvert Cliffs), anchor bolts pulled out of concrete and deformed mounting plates for spring hangers (Surry), plastic deformation in the feedwater line at a 90 degree elbow upstream of the steam generator nozzle (Surry, Turkey Point), damaged main feedwater regulating valve (San Onofre), damaged knee supports and feedwater piping snubbers (San Onofre, Zion, D.C. Cook, damaged feedwater pipe hangers (Zion) damaged hydraulic fittings on the bypass control valve assembly and 1/2-inch long crack in auxiliary feedwater pipe (Cook). A discussion of staff action with respect to water hammer in PWR steam generators is presented in Appendix A (Section A.8).

Most of the water hammer events in Table 3.2 are attributed to the following mechanisms and systems: (1) stop valve closure or opening of isolation valves in the main steam system; (2) flow control valve opening, closure or instability in the main feedwater system; (3) entrainment of water slugs in the steam lines to the turbines in the HPCI and auxiliary feedwater systems; (4) condensation of steam bubbles in the RHR and safety injection system following flow startup or in other systems due to inadvertent mixing of steam and subcooled water; (5) inadvertently voided pump discharge lines in RHR, ECC and other systems; and (6) check valve closures in main steam systems and in water systems. In some cases, a

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single event in Table 3.2 represents the consequences of an undetermined number of water hammer occurrences which eventually resulted in the damage or corrective action noted in the event report. The consequences of the events listed in Table 3.2 ranged from no damage to damage to hangers and restraints to rupture of a small feedwater line and valves in the feedwater and steam system. These events are discussed in more detail in Appendix A.

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4.0 STAFF ACTIONS AND RECOMMENDATIONS

4.1 Introduction

Most of the recommendations developed during the course of this review have already been implemented or are scheduled to be implemented under Task Action Plan A-1, "Water Hammer". Because the staff actions to be discussed in the next section will be related to individual tasks under TAP A-1, the following summary of these tasks is provided for reference:

Task 1.0 Water Hammer Summary Reports
This task involves preparation of the present report and a final report summarizing the work accomplished under TAP A-1.

Task 2.0 Revision of CP and OL Review Procedures
This task involves preparation of changes to the Standard Format (Reg. Guide 1.70) and the Standard Review Plan needed to provide adequate coverage of water hammer. Requests for preparation or modification of Regulatory Guides and changes to the Standard Technical Specifications will also be made under this task.

Task 3.0 Water Hammer Positions for Operating Reactors
This task involves the development of specific staff actions to be taken with respect to water hammer problems in operating reactors. Subtask 3.1 is involved with actions needed to resolve any immediate problems. A long-term position to be taken after completion of Tasks 2.0 and 4.0 will be developed under Subtask 3.2.

Task 4.0 Water Hammer Safety Studies
This task involves specific studies to be made under Technical Assistance Contracts which will provide information needed to develop the CP and OL

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review procedures under Task 2.0 and the long-term staff position for operating reactors under Task 3.0.

4.2 Actions and Recommendations

Under Task 4.0 of Task Action Plan A-1, the following technical studies have been started:

1. a state-of-the-art review of experimental, analytical and design work reported in domestic and foreign literature which is pertinent to water hammer in nuclear power plants;

2. a review of water hammer events to date associated with power-operated valve opening, closing and instability and check valve closure to identify reasons for occurrence (e.g., inadequate design or procedure, operator error), corrections made to prevent further occurrence (e.g., design, operating procedures), success of corrective actions, and generic implications (see Sections A.3 and A.4);

3. an evaluation of safety-related systems representative of current PWR and BWR designs to (a) establish possible scenarios leading to damaging water hammer from voided lines, water entrainment and vapor bubble collapse, and (b) evaluate the effectiveness of particular subsystems and design features used to prevent water hammer due to voided lines (e.g., jockey pump systems in BWRs) and water entrainment (see Sections A.1, A.5 and A.7);

4. an evaluation of the potential for occurrence of water hammer for the following specific mechanisms and systems: (a) column separation in the reactor coolant, emergency core cooling, service water and main feedwater systems for representative BWRs and PWRs; (b) transient flow in voided containment spray lines between the isolation valves and sprays during spray initiation; and (c) slug impact due to rapid

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condensation in the feedwater spargers of BWRs (see Sections A.6, A.2 and A.8);

5. an evaluation of potential water hammer problems in new PWR steam generators with bottom feed into a preheater box (see Section A.8).

6. preparation of calculation methods, using existing computer programs, to provide piping system response to dynamic loads resulting from (a) water hammer initiated by control valve closure or check valve closure following a pipe break upstream of the valve, (b) pump startup with voided discharge lines, and (c) steam bubble collapse.

In addition to the above scheduled studies under TAP A-1, it is recommended that reviews be made of experimental data and analytical methods pertinent to the prediction of dynamic loads and stresses resulting from the actuation of pressurizer safety valves with loop seals and the closure of feedwater check valves following postulated pipe ruptures upstream of the valves. The objective of the reviews should be to determine the adequacy of current methods for calculating dynamic loads on the valves and associated piping and the need for any confirmatory experiments (see Sections A.3 and A.4).

Under Task 2.0 of Task Action Plan A-1, the information developed in the present review (Appendices A and B) will be used in conjunction with the information to be obtained in the above technical studies of Task 4.0 to develop changes to the Standard Format and Standard Review Plan needed to ensure adequate treatment of water hammer in the licensing review process. Requests for preparation or modification of Regulatory Guides and changes to Standard Technical Specifications will also be made under this task. At this time, the following recommendations with respect to Regulatory Guides are made:

1. Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," should be revised to include the design of piping with closed

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discharge systems, open discharge systems with long discharge piping and systems having slug flow such as from a water seal (see Section A.3).

2. Regulatory Guide 1.16 should be modified to require reporting of significant water (steam) hammer occurrences in steam lines up to and including the turbine stop valve and the feedwater lines back to the feedwater pumps (see Section 3.3).

3. A Regulatory Guide should be prepared which contains criteria for precluding or minimizing water hammer in safety-related systems.

The following licensing actions have already been taken or are scheduled to be taken under Task 3.0 of Task Action Plan A-1:

1. Task 3.1 was completed in 1977 with the issuance of letters to certain PWR licensees requiring submission of proposed plant design and/or procedural modifications to prevent damaging water hammers in the steam generators (see Section A.8). Reviews of the licensee responses are being made under the generic review program, "Steam Generator Feedwater Flow Instability" (see e.g., Ref. 20).

2. Operating PWR plants having the same feedwater flow control valves as those used in Beaver Valley I will be reviewed with respect to actions already taken or still needed to prevent damaging water hammer (see Section A.3).

3. After completion of the technical studies of Task 4.0, jockey pump, or equivalent, systems used to maintain water filled lines in the RHR and ECC systems of operating BWRs will be reviewed to determine changes needed to reduce the incidence of water hammers due to inadvertently voided discharge lines (See Section A.1).

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4. Additional actions with respect to operating reactors to be taken under Task 3.2 will be based on the results obtained in the water hammer studies of Task 4.0 and changes to the Standard Review Plan and Standard Technical Specifications made under Task 2.0 (See Appendix A).

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APPENDIX A - POTENTIAL WATER HAMMER PROBLEMS

The staff reviewed information on about 120 water hammer events obtained primarily from Licensee Event Reports and information requests. Approximately 30 fluid systems in each of four light water reactors were also reviewed to determine the potential water hammer problems. Of these, 10 systems were considered of importance either because of their safety function, or the potential effect on the reactor coolant pressure boundary (See Table 2.1).

In our review, the following classification of water hammer problems was adopted:

(1) Pump Startup with Inadvertently Voided Discharge Lines. In water systems designed for operation with full discharge lines, inadvertent voiding of the lines due to air entrapment or draining may result in excessive dynamic loads following pump startup and should be prevented.

(2) Expected Flow Discharge Into Initially Empty Lines. Discharge lines in water systems which are normally empty and discharge lines from various pressure relief valves should be designed to withstand the expected dynamic loads.

(3) Valve Opening, Closing and Instability. Rapid valve opening and closing in both water and steam systems and instability of control valves in water systems may cause excessive dynamic loads.

(4) Check Valve Closure and Sudden Delayed Opening. Normal check valve closure following pump stopping is not expected to result in large dynamic loads but should be considered in the system design. For certain check valves that perform a safety function, the valves and associated piping should be designed to withstand the large dynamic loads resulting from a postulated rupture upstream of the valve. The sudden opening of a stuck check valve after pump startup can also produce damaging pressure pulses.

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(5) Water Entrainment in Steam Lines. Water slugs driven by steam may cause excessive dynamic loads while being swept through bends in the lines and from impact on tees or closed or partially closed valves.

(6) Transient Cavitation (column separation). The subsequent collapse of voids formed in water systems by low pressure transients resulting from pump stopping or seizure and change in valve setting or check valve closure may produce excessive dynamic loads.

(7) Steam Bubble Collapse and Mixing of Subcooled Water and Steam from Interconnected Systems. Damaging water hammer may result from the collapse of steam bubbles in water systems due to pressurization and condensation following pump startup or valve opening and from the mixing of steam and subcooled water from interconnected systems.

(8) Slug Impact Due to Rapid Condensation. The impact of water slugs, formed and driven by forces resulting from rapid condensation of steam on subcooled water in the feedwater rings and adjacent piping, has been identified as the cause of damaging water hammer in certain PWR steam generators with top feed.

(9) Pump Startup, Stopping and Seizure with Full Lines. Dynamic loads resulting from pump startup with full lines are expected to be relatively small, but should be considered in system design. Pump stopping in some systems may produce column separation. Postulated pump seizure can result in large dynamic loads and may cause column separation.

These potential water hammer problems are discussed in more detail in the following sections. A system review in terms of these problems is summarized in Appendix B.

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A.1 Pump Startup with Inadvertently Voided Discharge Lines Due to Entrapped Air or Draining

Severe water hammer may occur following pump startup or valve opening if the discharge lines are partially empty. A variety of mechanisms can be postulated to lead to partially empty lines. These include the water column separation mechanism discussed in Section A.6, and the formation of vapor bubbles in components due to flashing and heat and work input which is discussed in Section A.7. This section is concerned with voided lines caused by inadvertent air entrapment or draining. This type of voiding obviously applies to a large number of systems. However, the effects of the resulting water hammer are particularly pertinent to the residual heat removal (RHR), emergency core cooling (ECC), component cooling water, and service water systems because of their safety significance.

Air may be trapped in a system because of inadequate venting provisions or filling and venting procedures, evolution of gas from solution during low pressure transients, air admission at the system intake due to vortex formation, or operator error. The effect of air on water hammer might be beneficial for some transients if the air were dispersed through the liquid since there is a marked decrease in wave propagation speed with the addition of small amounts of gas (see e.g., Ref. 2). As noted in Section A.2, air may also be deliberately put into some systems to mitigate the consequences of expected transients. For example, in some circulating water systems air is introduced by vacuum breakers to prevent damaging low pressure following loss of pump power. However, the presence of large pockets of air in downstream portions of the system may result in excessive dynamic loads. The air results in higher liquid velocities during the initial portion of the transient with attendant increased loads on the piping and possible higher pressures during the latter portion of the transient as the air is compressed (see e.g., Refs. 1 and 11). If the water front reaches a closed valve, there will be sudden large impact loads. Even if a downstream valve is partially open, there can still be large impact loads when the liquid

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front reaches the valves since the valve pressure drop with air flow is appreciably less than that with water flow (Ref. 12).

In many systems (e.g., RHR and ECC systems for PWRs and some reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems for BWRs) the relative elevation of the water supply, injection points and isolation valves are such that the piping which is initially vented tends to remain full. For such systems the principal protection is obtained from administrative controls and proper vent locations and filling and venting procedures. However, even with an initially full system, there could be inadvertent draining. For example, one water hammer event in the ECC system for a PWR resulted from an air bubble which was inadvertently collected and trapped during a maintenance operation when the primary system was drained. Only a few water hammer events have been attributed to voided pump discharge lines for these systems.

For systems such as the RHR and core spray systems and some RCIC and HPCI systems in BWRs, the relative elevations and valving arrangement are such that voiding of lines due to normal system leakage will automatically occur unless a positive and continuous means for maintaining full lines is provided. Nearly all of the water hammer events resulting from voided pump discharge lines have been for these systems.

The draining problems for the core spray and RHR systems in BWRs result from the large difference in elevations of the suppression pool which serves as the water supply and the normally closed injection valves near the reactor vessel. Since the injection valves are about 60 feet above the suppression pool and the pump suction valves are open, water can drain back to the pool through a leaking pump discharge check valve and leaking or inadvertently open valves in the bypass test line. The resulting void may be at near-vacuum conditions, containing water vapor and small amounts of gas evolved from solution. In this case, there is essentially no cushioning due to air compression and large water hammer pulses following pump startup can be generated when the accelerating water front hits a closed or partially closed valve. Most of the reported water hammer events for these systems have occurred during surveillance testing of the pumps

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when the injection valves were closed. However, if the system operation had been initiated automatically, damaging water hammer would also have been expected as the result of the water front reaching a partially open injection valve and the check valves downstream of the injection valves.

As the result of water hammer occurrences in earlier plants, jockey pump systems were installed to maintain full lines by pumping suppression pool water to the RHR and core spray lines downstream of the pump discharge line check valves. The jockey pumps generally operate to maintain pressure downstream of the pump discharge check valve above the static head corresponding to the highest point in the RHR and core spray systems. In principle, the use of a properly designed jockey pump system, and adequate venting and filling and venting procedures, should prevent the occurrence of water hammer due to voided pump discharge lines. However, in view of the water hammer events that have occurred and the fact that some surveillance test procedures involve special filling and venting operations before pump startup to prevent water hammer, it is believed that additional improvements in system designs and procedures to prevent this type of water hammer are needed.

Damaging water hammer may also occur in service water systems as the result of pump startup with voided lines. The voids can be formed by improper filling and venting procedures, inadvertent draining, or water column separation when flow is terminated. In addition, the formation of voids at near vacuum conditions can occur as the result of draining of lines back to the ultimate heat sink when the pumps are stopped (e.g., loss of offsite power). The use of vacuum breakers in service water systems to prevent water hammer is discussed in Section A.2.

Twenty-two water hammer events in Table 3.2 are attributed to pump startup with voided discharge lines. Of these events, 16 occurred in the RHR, core spray or HPCI systems of BWRs. As noted previously, these systems are susceptible to draining and must be supplemented by a jockey pump, or equivalent, system to maintain full discharge lines. Three events occurred in service water systems

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of PWRs or BWRs which are also susceptible to draining. Only three of the 22 events occurred in systems which tend to remain full.

There have been a significant number of water hammer events attributed to inadvertently voided discharge lines in BWRs. Therefore, under Task 4.0 of Task Action Plan A-1, a more detailed study is being made (a) to identify design and procedural deficiencies leading to the reported water hammer events, and (b) to review representative jockey pump, or equivalent, systems for the RHR and core spray systems of BWRs. The objective of this study is to recommend design criteria criteria (including criteria for sensors and alarms) and changes to Technical Specifications for the jockey pump, or equivalent, systems that could be used to minimize the occurrence of water hammer. After completion of this study, jockey pump, or equivalent, systems used to maintain water filled lines in RHR, RCIC and ECC systems in operating BWRs will be reviewed.

A.2 Expected Flow Discharge into Initially Empty Lines

Most safety-related water systems are designed to have full discharge lines or are provided with special fill systems to maintain full lines in order to prevent damaging water hammer. However, there are some cases (e.g., containment spray line, service water system) where it may be impractical to maintain full water discharge lines. For these cases, the system should be designed to withstand the dynamic loads arising from the expected transient flow into an empty or partially empty line. Other lines which normally are empty or partially empty prior the transient are the discharge lines from various safety and relief valves in water and steam systems.

Water Lines

The containment spray lines are voided from the isolation valve to the spray headers. The pumps are actuated by a safety injection signal and the isolation valves subsequently open on a containment spray signal. Potential pipe failure can be caused by pipe jump due to the accelerating flow in the voided line and the impact as the flow front encounters elbows, tees, and valves in the piping.

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Since these systems are not subjected to a complete flow test, there is no operating experience concerning the adequacy of the designs with respect to dynamic loads. Similar voided line conditions have produced some damage in other piping systems.

Other water lines which may be empty include bypass test lines and minimum flow lines for some RHR and ECC systems, portions of the head spray line for BWRs and discharge lines from pressure relief valves on water systems. There is appreciable operating experience with these lines; however, no reports of damaging water hammer were found.

Damaging water hammer may also occur in open-ended systems such as the plant or RHR service water systems following restart of pumps after shutdown (e.g., after loss of offsite power). After shutdown, water may drain from portions of the pump discharge piping back through the pump to the ultimate heat sink and may also drain from other open ends of the system. The voids created in this manner may be at near-vacuum conditions. Vacuum breakers have been used (see e.g., Ref. 13) to reduce peak pressures by introducing air which provides a cushion between the water column driven by pump and the remaining water column in the system. Vacuum breakers have also been used to counteract the effect of damaging low pressure transients and to provide air cushions at points where voids caused by water column separation could occur following pump shutoff. Since the presence of the air pockets may lead to higher transient pressures than obtained with full lines, air release valves and flow throttling may also be used to control maximum pressures and dynamic loads (see e.g., Ref. 11).

Under Task 4.0 of Task Action Plan A-1, calculations will be made to determine the potential effects of water hammer in typical containment spray systems for PWRs and BWRs. If the results of this study indicate significant piping loads, it may be necessary to request licensees of operating reactors to demonstrate the adequacy of the containment spray systems.

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Discharge Lines from Safety and Relief Valves

Discharge lines from safety and relief valves on main steam lines and pressurizers are subjected to large dynamic loads following valve opening. These loads may produce peak stresses larger than those resulting from seismic loading (Ref. 14). The dynamic loads are high because of the short valve opening times (about 0.04 to 0.10 second), high inlet pressures (about 1100 to 2500 psia) and an initial atmospheric pressure in the discharge line. Dynamic loads on the discharge piping may be increased significantly if a water seal is used upstream of the valve, because of the reaction loads resulting from movement of the slug of water along the pipe (Refs. 14 and 30). Although pressurizer relief valves open more slowly, large dynamic loads may still be imposed on the valves and discharge piping.

No incidents involving damage to discharge piping from pressurizer safety valves were found. There is one event in Table 3.2 concerned with damage to the discharge line from an atmospheric dump valve. Events resulting in damage to portions of the discharge piping of BWR safety/relief valves located within the suppression pool have been reported, but are not included in Table 3.2. The dynamic loads on these discharge lines for both Mark I and Mark II type containments are being reviewed separately (see Ref. 15) under Task Number A-7, "Mark I Containment Long Term Program," Task Number A-8, "Mark II Containment Pool Dynamic Loads," and Task Number A-39, "Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containment."

No action is required for operating BWRs in view of the planned reviews under Task Numbers A-7, A-8 and A-39. No action is required for operating PWRs because of the limited number of damaging events reported.*

A.3 Valve Opening, Closing, and Instability

The rapid opening or closing of valves is a potential source of water hammer in a number of water and steam systems in nuclear plants. The valves to be considered include isolation and stop valves and various flow, pressure, and

*Recommendations pertaining to dynamic loads on the valves and discharge piping are given in Section A.3.

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temperature control valves. Flow control valve instability in water systems may also lead to significant water hammer loads.

This section is concerned only with power-operated or manually-operated valves and pressure relief valves. Water hammer due to closure of check valves is covered separately in Section A.4. For water systems, it is also assumed that the lines are full. Damaging hammer occurrences due to voided water system discharge lines are discussed in Sections A.1 and A.2. Water column separation is treated in Section A.6.

Water Systems

As noted in Section 3.1, one of the classical problems in the water hammer literature is the calculation of system response to the rapid closure of a valve. For water systems, this may result in large pressure increases and dynamic loadings of the piping system. For example, an assumed instantaneous complete closure of a valve in a water line with an initial velocity of 25 ft/sec and temperature of 300°F results in a pressure increase of roughly 1500 psi. However, damaging water hammer due to valve closure or opening in water systems usually is prevented, while still meeting operational requirements, simply by limiting the rate of change in valve setting.

Isolation valves are found in all water systems of safety significance. The closure time of power-operated isolation valves are much longer than the piping transit times ($t = 2L/C$) identified in Section 3.1. Typical closure times, which range from roughly 20 to 120 seconds, would result in small pressure increases and dynamic loadings. No water hammer events have been attributed to this type of valve closure. Therefore, it is recommended that no action be required of operating reactors.

There are control valves in water systems which may produce significant water hammer loads. These include the flow control valves in the main feedwater systems of PWRs and BWRs and pressure control valves in the letdown lines of the CVCS in PWRs. Other control valves such as those in the recirculation loop of a BWR and in plant systems such as the RHR, RWCU, auxiliary feedwater, component

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cooling water and plant and RHR service water systems are not expected to produce significant dynamic loads but should be considered in the system design.

The flow control valves in main feedwater systems of both BWRs and PWRs have the potential for producing significant water hammer loads as the result of relatively high fluid velocities and short closure and opening times. Twenty-two events in Table 3.2 are attributed to main feedwater flow control valve opening, closing, or instability. In three events the water hammer resulted from a sudden flow rate decrease following valve failure in which the plug separated from the valve stem and blocked the flow. These valve failures may be attributed in part to piping vibrations during normal operation. The remaining incidents involved sudden valve opening or closing and valve instability. Components damaged as the result of these water hammers included piping supports and restraints, valve bodies and operators, and the piping.

Resolution of the feedwater control valve instability problems and determination of the causes of some of the events involving sudden opening and closing of these feedwater control valves may require consideration of a large number of contributing factors. These include unbalanced hydraulic forces on the valve plug, damping due to frictional forces exerted by the valve stem packing, valve flow gain (ratio of flow to plug lift), and the dynamic characteristics of the valve and valve operator. Treatment of the hydraulic forces involves consideration of possible resonances between the valve and the compressible water columns in the piping. In some instances, the hydraulic forces apparently caused the valve to override the force applied by the operator. In addition, the head-flow characteristics of the condensate booster pumps, condensate pumps and feed pumps may be contributing factors.

At one installation (Beaver Valley 1), the proposed solution involved a reduction in trim size (originally oversized for the application) and replacement of the original plug-type trim with a cylindrical trim designed to reduce cavitation and unbalanced forces. Approximately eight other PWR plants originally had the same type of control valve. Some of these plants have also modified the valve trim. All PWR plants having this model flow control valve will be reviewed to determine if any corrective actions are needed.

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Under Task 4.0 of TAP A-1, a more detailed review will also be made of operating experience at both PWR and BWR facilities to determine the causes of this type of water hammer in main feedwater lines and the design and procedural changes made to prevent the recurrence of water hammer. The review will include consideration of piping vibration during normal operation. This may have contributed to the valve and operator failures which initiated some events. Actions with respect to operating reactors will be based on the conclusions reached from this review.

In the letdown line of the Chemical and Volume Control System (CVCS), orifices or control valves are used downstream of the regenerative heat exchanger to control the flow from the reactor coolant system which is at a pressure of about 2250 psia. To prevent flashing, the pressure downstream of these offices or flow control valves is maintained at pressures of roughly 400 psia by a pressure control valve. In view of the high primary system pressure, there is a potential for appreciable dynamic loading when the pressure control valve is actuated. Therefore, careful design of the valve and associated actuator and control system is required to prevent large pressure changes. Only one water hammer event associated with the pressure control valve in the CVCS was found (see Table 3.2). Hence, it is concluded that no action should be taken with respect to operating reactors.

Steam Systems

Steam hammer, the transient arising from a sudden change in the steam flow rate, can cause relatively large dynamic loads in steam lines and, in some cases, has caused damage to piping supports and restraints. There are a number of expected transients involving sudden opening or closing of valves which can cause significant dynamic loadings and which should be considered in the design of steam piping systems.

For the main steam lines, which have typical operating pressures of about 900 psia and velocities of about 150 feet per second, closure of the turbine stop/control valve would be expected to produce the most severe steam hammer loading resulting from closure of a power-operated valve. The pressure

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transients following rapid valve closure in steam systems are more gradual and less severe than those in water systems. For typical closure times of 0.1 to 0.2 second, pressure fluctuations following valve closure may reach 10 to 20 percent of the operating pressure. Rapid closure of the turbine stop valve may also result in significant transient loads on the steam supply lines to auxiliary turbines (e.g., RCIC, HPCI, and auxiliary feedwater turbines). The closure times for power-operated isolation valves in steam lines are typically several seconds and should result in smaller transient loadings. Closure of check valves in steam lines is discussed in Section A.4.

Anticipated transients involving rapid opening of valves in steam lines include: (a) opening of the steam bypass line to the main condenser, (b) main steam dump valve and main steam safety valve discharge to atmosphere for PWRs, (c) PWR pressurizer power-operated relief valve and safety valve discharge to the quench tank, and (d) main steam safety/relief valve discharge for BWRs. Opening times are about 0.2 second for steam bypass valves, about 1 second for power-operated relief valves, and from about .04 to .10 second for safety valves. As noted in Section A.2, the transient loads for safety valve lines are high because of the short opening times, high inlet pressures (about 1100 to 2500 psia), and initial atmospheric pressure in the discharge line. If a water seal is used upstream of the safety valve, the dynamic loading on the relief valve could be increased significantly as the results of the acceleration of the water slug down the line with valve discharge.*

Ten events in Table 3.2 are attributed to water (steam) hammer caused by valve opening or closure (other than check valves) in main steam lines. Of these, four occurred during the startup testing program which includes tests to find any deficiencies in the piping support and restraint design. Two events were attributed to operator error either during reactor startup or valve testing. The remaining events involved anticipated transient loads caused by closure of

*This section is concerned only with dynamic loads on the supply steam lines. The dynamic loads on the discharge piping from safety valve and relief valves is discussed in Section A.2.

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the turbine stop valve or opening of the bypass valve. In most cases, the events resulted in some damage to piping supports and restraints. No event resulted in rupture of a steam line. Corrective action to prevent recurrence of damage during these transients involved the use of additional supports and restraints. In view of the corrective actions already taken, no action for operating reactors is recommended at this time.

Two events in Table 3.2 involved damaging steam hammer resulting from opening of pressure relief valves. Both events occurred during the initial testing program and involved rupture of the steam supply piping. No action is recommended for operating reactors because of the limited number of damaging events reported.

It is recommended that a review be made of experimental data and analytical methods pertinent to the evaluation of dynamic loads on pressurizer safety valves and discharge piping. The objective of the review should be to determine the adequacy of current methods for calculating dynamic loads for safety valves with loop seals and the need for any confirmatory experiments. It is also noted that Regulatory Guide 1.67, entitled "Installation of Overpressure Protection Devices," is limited to methods for implementing GDC 1 with respect to the design of piping for safety valve and relief valve stations having open discharge systems with limited discharge pipes and inlet piping that neither contains a water seal nor is subject to slug flow of water following opening of the valves. It is recommended that Regulatory Guide 1.67 be revised to include the design of piping with closed discharge systems, open discharge systems with long discharge piping and systems having slug flow such as from a water seal.

A.4 Check Valve Closure and Delayed Opening

Large check valves are used in the pump discharge lines of a number of water systems (e.g., ECC, RHR, RCIC, main feedwater, auxiliary feedwater systems). The valves prevent reverse flow following loss of pump power in a single loop or loss of power to one or more pumps in parallel loops and may provide isolation for overpressure pressure protection of portions of the system. In some cases

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the valves also have a containment isolation function. Large check valves are also used in the main steam systems for some PWRs and in the turbine exhaust lines of the HPCI and RCIC systems of BWRs.

If the pump power is lost in a given loop, there is a flow coastdown followed by flow reversal which is stopped by the check valve. Ideally, the valve would close at the zero flow condition and result in no water hammer on closing. However, for rapid flow reversal the valve is late in closing and significant pressure transients may be generated that propagate throughout the system. The effect of the check valve closure is the possible failure of the valve itself and/or piping damage. Examples of calculations for the case of check valve closure in a nuclear system are given in References 7, 16 and 17. Corrective measures to minimize the consequences of this type of water hammer may include the use of hydraulic dampers to modify closing times, and changes in piping layout and snubbers. The magnitude of the hydraulic loads imposed on the piping and components by normal check valve closure due to loss of pump power should not be large. However, the effects of check valve closure should be considered in the system design. Damage to check valves has also occurred as a result of disc oscillation ("flutter") or intermittent opening and closing ("percolation") in steam lines (e.g., main steam lines in PWRs and steam exhaust lines of HPCI and RCIC in BWRs).

Rapid check valve closure will occur as the result of a postulated pipe rupture upstream of the valve when it is open during normal operation. Closures of this type are not assumed for check valves in the ECC system when operating following a postulated LOCA. However, there are check valves in systems used in normal plant operation which are used to mitigate the consequences of a postulated pipe rupture upstream of the valve. Examples are (a) the main feedwater check valves for BWRs and PWRs, (b) check valves used in the main steam lines of some PWRs to stop blowdown of all steam generators following a postulated steam line break, and (c) check valves in the steam supply lines to the auxiliary feedwater turbines in PWRs. For these check valves, analyses should be performed to confirm that

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the valve and associated piping can withstand the effects of the water or steam hammer resulting from valve closure and perform the required safety function.

Of the nine water hammer events in Table 3.2 listed under Check Valve Closure and Delayed Opening, four were associated with inadvertent closure of valves in the main steam lines of PWRs. The valves serve as isolation valves in the event of a postulated main steam line break. The closures, which resulted either from inadvertent actuation signals or from fluid forces, resulted in valve damage. Four events resulting from check valve closure in liquid systems were attributed to valve slamming following pump shutoff or flow reversal during surveillance tests. One event resulted from delayed opening of check valves in the RHR system of a BWR.

As the result of the adverse operating experience with the check-type main steam isolation valves, Information Request No. 74-2 was sent by the staff to all PWR stations using these valves. The information obtained from these requests and from meetings with valve manufacturers and utility representatives was used as input to a generic study conducted by the staff. The staff's concern was the ability of the valves to withstand dynamic forces associated with rapid closure in the event of a steam line rupture. From this review it was concluded that some upgrading of both the materials and the design of the larger swing-check and angle lift-check MSIVs might be required. The affected PWR plants were requested to supply analyses or tests to confirm that the integrity of the MSIVs would be maintained under postulated steam line break conditions. Safety evaluations of the analyses and modifications to the valve design and materials required to meet the safety objective were completed in 1976 (see e.g., Ref. 18).

Damaging water hammer may also occur if there is a sudden opening of a stuck check valve. One water hammer event in the RHR system of a BWR (see Table 3.2) was attributed to the delayed opening of check valves in the pump discharge lines. The check valves, which were located in vertical lines, apparently were stuck in the closed position during pump startup and subsequent opening of a gate valve in the discharge line. The buildup of upstream pressure caused a sudden opening of the valve and a resultant pressure pulse which damaged piping supports and restraints.

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With respect to water hammer resulting from check valve closure or delayed opening, no immediate action is recommended for operating reactors in view of the corrective actions already taken for the steam lines and the limited number of events for liquid systems. However, it is recommended that the staff initiate a more detailed review of the effects of water hammer resulting from line rupture upstream of check valves in feedwater lines. The review should include evaluation of the adequacy of current analytical methods used to predict dynamic loads and the need for any confirmatory experiments.

A.5 Water Entrainment in Steam Lines

About half of the damaging water hammer events in steam systems were attributed to a water entrainment mechanism in which slugs of water are driven along piping by the steam. The principal damage probably results from impact of the water slugs on system components such as isolation valves and turbine stop and control valves. However, excessive reaction loads may also occur as the water slugs are accelerated through bends in the piping system.

The systems which should be considered with respect to this type of water hammer include the main steam system, those RHR systems for BWRs which have the steam condensing mode of operation, and various auxiliary systems having pumps with steam turbine drives such as the auxiliary feedwater system, and the RCIC and HPCI systems.

The two obvious sources of water in the steam lines of these systems are (1) normal carryover from PWR steam generators or BWR pressure vessels and excessive carryover resulting from high water levels in these components, and (2) condensation resulting from normal heat losses in lines carrying steam or during the introduction of steam to cold lines. The traditional design approaches used to remove water produced by these mechanisms include (1) sloping of horizontal lines downward in the direction of steam flow to promote drainage, and (2) provision of drainage features at all low points where water can accumulate. The drainage features should have provisions for local draining at points where partial damming of draining lines is caused by globe valves or other obstructions. In addition, the designer should avoid creating local nondraining pockets such

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as caused, for example, by installation of gate valves with stems oriented below the horizontal. Where possible, self draining of low points should be provided. The uses of steam traps or other automatic or manual methods for draining are dependent for successful operation on adequate maintenance and test procedures. To prevent water hammer during startup of cold lines, small bypass valves or limitations on the minimum actuation times of injection valves can be used to provide slow warmup.

In addition to carryover and condensation there can be other less obvious sources of water in steam lines. In one LER, the water source in the exhaust line from the auxiliary feedwater turbine was rainwater which was not removed because of an inadequate drainage system. For steam lines exhausting below the water surface in pools or tanks, steam condensation following steam shutoff can produce low line pressures and result in water being drawn up into the lines. A design solution for this problem for the turbine exhaust lines of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems has involved the incorporation of vacuum breakers and check valves arranged to allow air flow into the line but preclude steam flow to the torus.

Eleven water hammer events in Table 3.2 were attributed to steam-driven slugs of water. Of these, two occurred in the steam lines of auxiliary feedwater turbines of PWRs and were attributed to inadequate drainage. Two events occurred in the exhaust lines of HPCI and RCIC systems and apparently resulted from suppression pool water being drawn into the line. The other seven events occurred in the HPCI system and were the result of either inadequate draining while the system was in operational status or procedural or system inadequacies during system warmup. Of the five events occurring during system warmup, three resulted in significant system damage. These were attributed to water slugs collected upstream of a normally open outboard isolation valve in the steam supply line which was closed either inadvertently or for maintenance operations. Water hammer occurred following subsequent opening of the valve which has a seal-in feature to give full opening after receipt of the initiating signal. The addition of valve interlocks, removal of the seal-in feature or changes in operating procedures were used to prevent recurrence of this type of water hammer.

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It is concluded that the mechanisms leading to water hammer due to water entrainment are sufficiently well understood at this time and that this type of water hammer can be prevented by careful attention to system design and the use of thorough maintenance, testing, and operational procedures. The water hammer incidents to date have been the result of inadequacies of design or procedures which could be corrected to prevent the recurrence of these events.

There have been a number of incidents involving water hammer in the HPCI and auxiliary feedwater turbine steam systems. However, the loss of the turbine drive for these auxiliary system pumps does not lead directly to unacceptable consequences because of the availability of other redundant systems such as the automatic depressurization system (BWR) and electrically-driven auxiliary feedwater pumps (PWR).

No immediate action is recommended for operating reactors since the occurrence of this phenomenon is generally related to administrative controls which are subject to inspection by IE in connection with the review of such events. However, under Task 4.0 of TAP A-1, representative steam supply and exhaust systems for HPCI, RCIC, and auxiliary feedwater turbines will be reviewed to establish criteria to be used in preventing this type of water hammer.

A.6 Transient Cavitation (Water Column Separation)

Pressure transients propagated through a liquid system by sudden changes in a valve position, check valve closure, or a pump failure or seizure may cause local pressures to drop to the liquid vapor pressure and result in cavitation. The local pressure reductions during transients may also cause the release of appreciable quantities of dissolved gas. If the vapor cavities occupy a significant fraction of the pipe cross section area, this phenomenon of vapor cavity formation in a full flowing system is referred to as column separation. The concern with respect to water hammer is the possibility of excessively high impact pressures and system loads resulting from subsequent collapse of the vapor cavities. The cavity collapse may occur during the transient or following subsequent pump startup or valve opening.

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Transient cavitation involves nonequilibrium thermodynamic processes in both the formation and collapse of the cavities and is affected by entrained or dissolved gasses. Several models for treating transient cavitation in pipes have been developed (e.g., Refs. 19-23). In a recent article in which transient cavitation was treated by a column separation model, it was shown that analyses which neglect column separation effects give erroneous pressure histories both for locations which cavitate as well as for locations which do not cavitate but are influenced by cavitation occurring elsewhere (Ref. 22). The subsequent collapse of the vapor cavities can also produce larger dynamic loads than those predicted by analyses which neglect column separation.

Transient cavitation may have occurred, but was not identified as a contributing factor to any of the reported water hammer occurrences in liquid systems. It is more likely to occur in systems operating at pressures close to the saturation pressure and has been considered for the circulating water system for the main condensers. Under Task 4.0 of TAP A-1, a study will be made of the potential for occurrence of transient cavitation (column separation) in selected safety-related systems of PWRs and BWRs following expected maximum power operated valve closure rates, check valve closures and pump stoppage or seizure. The systems to be evaluated will include the primary coolant, ECC, RHR service water and component cooling water systems.

A.7 Steam Bubble Collapse and Mixing of Subcooled Water and Steam from Interconnected Systems

If large steam bubbles or pockets are produced locally in systems which normally carry subcooled water, the subsequent bubble collapse due to pressurization and condensation following pump startup or valve opening may produce damaging water hammer. Damaging water hammer due to rapid condensation of steam bubbles may also occur during deliberate or inadvertent mixing of subcooled water and steam from interconnected systems.

In this section the discussion of steam bubble collapse is concerned only with (a) steam bubbles formed in water lines by flashing following local or overall system depressurization, (b) steam bubbles formed in water lines by energy input

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at heat exchangers, steam generators and pumps, and (c) mixing of steam and water from interconnected systems. The collapse of vapor pockets formed by transient cavitation is covered in Section A.6. Water hammer due to the slug impact resulting from rapid condensation of steam pockets formed by local drainage of the feedwater spargers and associated piping in PWR system generators is treated in Section A.8.

Flashing Following Local or System Depressurization

Four of the water hammer events listed in Table 3.2 under Steam Bubble Collapse were attributed to depressurization. Of these, two occurred in the safety injection system accumulator discharge lines of PWRs during leakage tests of the check valve. The steam bubble was formed between the check valve and closed gate valve to the accumulator by flashing as the result of excessive test drainage. Water hammer occurred when the gate valve was reopened. This type of water hammer was eliminated by modifying the test procedure to maintain pressure between the valves above saturation pressure and, hence, prevent flashing. Water hammer may also occur as the result of local depressurization during valve opening and closing sequences in surveillance tests. This mechanism was postulated to be the cause of a water hammer occurrence during surveillance testing of the timing of RHRS isolation valves for a BWR while the reactor was at power. To prevent the recurrence of this type of water hammer, it was concluded that future tests would be conducted only after the reactor was cooled down to the normal RHR system operating temperature.

Water hammer has also occurred during initiation of the shutdown cooling mode of the RHR system of a BWR as the result of a steam bubble in the discharge line between the outboard RHR injection valve and the reactor recirculation loop. During the reactor depressurization, this line apparently was hot enough to result in flashing of water. A check valve downstream of the inboard injection valve prevented reflooding of this section of the pipe. Following initiation of RHR shutdown cooling, the rapid injection of cold water into the trapped steam pocket resulted in water hammer. One solution has involved slow flooding and pressurization of this section of line prior to RHR startup in the shutdown cooling mode. It is noted that bubble formation between check and closed

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injection valves might also occur in other systems (e.g., RHR and ECC systems for PWRs) and for inadvertent rather than planned depressurization (e.g., inadvertent safety/relief valve opening or LOCA). Hence, a further review of this potential water hammer mechanism and means for preventing its occurrence is recommended.

One could conjecture that water hammer could occur in the reactor coolant system following a LOCA as a result of rapid condensation of steam following injection of cold ECC water. The concern is that water hammer might lead to damaging ECCS piping and component loads. Significant water hammer of this type has not occurred in semi-scale and LOFT tests. Minor pressure fluctuations (N 10 psi) were observed (Ref. 24). This suggests that the postulated mechanism should not be of concern. It is also not expected to be of concern for spray-type ECC systems such as used in a BWR where condensation is on water drops or jets. Piping loads associated with this postulated type of water hammer, which could only occur at power system pressures when there is sufficient steam formed by flashing to interact with the injected water, are expected to be small relative to blowdown loads. However, there are uncertainties both with the mechanism and resulting loads. Hence, the staff will continue to review future LOFT, semi-scale and ECC injection tests such as described in References 25 and 26, for information pertinent to water hammer.

Energy Input at Heat Exchangers, Steam Generators and Pumps

Steam bubbles could be formed in the component cooling water and service water systems at off-design conditions as the result of heat input at the heat exchangers. The normal secondary side temperatures of the heat exchangers are well below saturation temperature. However, low secondary side flow rates due to operator error or pump or valve failure could result in steam bubble formation. Some plants have high temperature alarms on the secondary side which are set welll below the saturation temperature. It is noted that the occurrence of water hammer would require an operator error (i.e., flow initiation following a condition giving bubble formation). Hence, administrative controls are important in preventing this type of water hammer.

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The formation of steam bubbles in the primary coolant system of PWRs at the steam generators would require both low flow rates and coolant pressures well below the normal operating pressure. This might occur during a cooldown and depressurization under natural circulation conditions (e.g., loss of offsite power or damage to RCP seals and deliberate pump shutoff*). There is a low probability of occurrence of this type of transient. In addition, bubble formation could be precluded by control of primary system pressure. As in the case of water hammer involving the heat exchangers, administrative control is the key to preventing water hammer under these circumstances.

Steam bubble formation at pumps could occur as the result of overheating of the water due to low flow rates, gross cavitation during pump runout and inadequate head in the pump suction line. Since pump damage will result, such occurrences are prevented by system design and operating procedures, regardless of water hammer considerations. Minimum flow bypass lines with automatic valve alignment are provided to prevent damage due to low water flow rates through the pump. Orifices are provided in the discharge lines to prevent damage due to pump runout. There have been no water hammer occurrences attributed to the collapse of steam bubbles formed by energy input at heat exchangers, steam generators or pumps.

Mixing of Subcooled Water and Steam from Interconnected Systems

The RHR system of a BWR that has a steam condensing mode of operation is susceptible to steam bubble formation due to steam leakage past valves. When not in the steam condensing mode, the RHR system heat exchangers are full of water and valves in the steam lines leading to the heat exchangers are closed. If the steam valves leak, a steam bubble may form in the heat exchangers and result in water hammer following RHR pump startup. Of the 13 events in Table 3.2 which are listed under Steam Bubble Collapse, three

*Formation of a steam bubble in the primary coolant system of a PWR has occurred under these circumstances.

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are attributed to this mechanism. Near-zero leakage steam isolation valves, supplemented by sensors and alarms to indicate leakage, could be used to prevent this type of water hammer.

The steam condensing mode of operation of the RHR system for a BWR is used to condense reactor steam while the reactor is isolated for the main condenser and reactor cooling water is being supplied by the RCIC system. Steam at reactor pressure is throttled to the desired pressure for the RHR heat exchanger where it is condensed and subcooled. The heat exchanger and some associated piping must be valved off, drained, and slowly heated with steam before operating in the steam condensing mode. Since the system involves an interconnected steam region and cold water lines, the evaluation of potential water hammer events should be concerned with the effects of inadvertent water valve positioning. This can occur because of valve failure or operator error during operation in the steam condensing mode or during the transition between operation in the steam condensing mode and other modes where the heat exchangers are kept full by the jockey pumps. No water hammer event attributed to this mechanism has been reported.

Condensation of the exhaust steam from the HPCI and RCIC turbines in the suppression pool is characterized by intermittent formation and collapse of large pockets of water vapor. The resulting dynamic conditions at the pool can produce exhaust line vibrations and oscillations of the exhaust line check valves which have a containment isolation function.* Two of the Steam Bubble Condensation events listed in Table 3.2 were concerned with exhaust line flow oscillations which lead to the failure of check valves in the HPCI system. Resolution of the problem involved the use of spargers and redesigned check valves.

*Steam bubble collapse in the suppression pool can also lead to excessive loads on the safety/relief valve discharge lines in the torus. This problem is discussed in Section A.2.

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Recommendations

Of the 14 recorded water hammer events in Table 3.2 associated with steam bubble collapse or steam condensation (other than in PWR steam generators), four events were attributed to steam bubbles formed by depressurization, three to steam leakage in the RHR system heat exchangers of BWRs, three to steam bubbles in PWR feedwater lines, two to flow oscillations in the HPCI steam exhaust lines caused by bubble collapse in the suppression pool, one to the inadvertent injection of water into a BWR heating supply line and one to condensation in the steam generator blowdown line. In view of the variety of mechanisms and systems associated with these events, a more detailed study is being made of water hammer resulting from steam bubble collapse in safety-related systems of PWRs and BWRs. This study, which is part of Task 4.0 of Task Action Plan A-1, has the objectives of identifying any additional potential sources of this type of water hammer and defining pertinent design criteria to be used to prevent occurrence of this type of water hammer. No action is required for operating reactors until completion of this review.

A.8 Slug Impact Due to Rapid Condensation

A mechanism believed to be responsible for a number of water hammer occurrences in top-feed steam generators involves the sudden impact of slugs of water formed and driven by forces resulting from the rapid condensation of steam on subcooled water. Figure 1 shows the layout of the feedwater line and feedring for a steam generator with top feed. During power operation, the water level is well above the feedwater ring. However, during a transient or accident when the main feedwater flow is shut off, the water level falls below the feedwater ring. Water in the ring and horizontal portion of the feedwater pipe then drain out through the holes in the bottom of the ring and through the clearance between the thermal sleeve and the feedwater line nozzle. The auxiliary feedwater pumps used in PWRs to recover water level have a capacity which is approximately 3.5%

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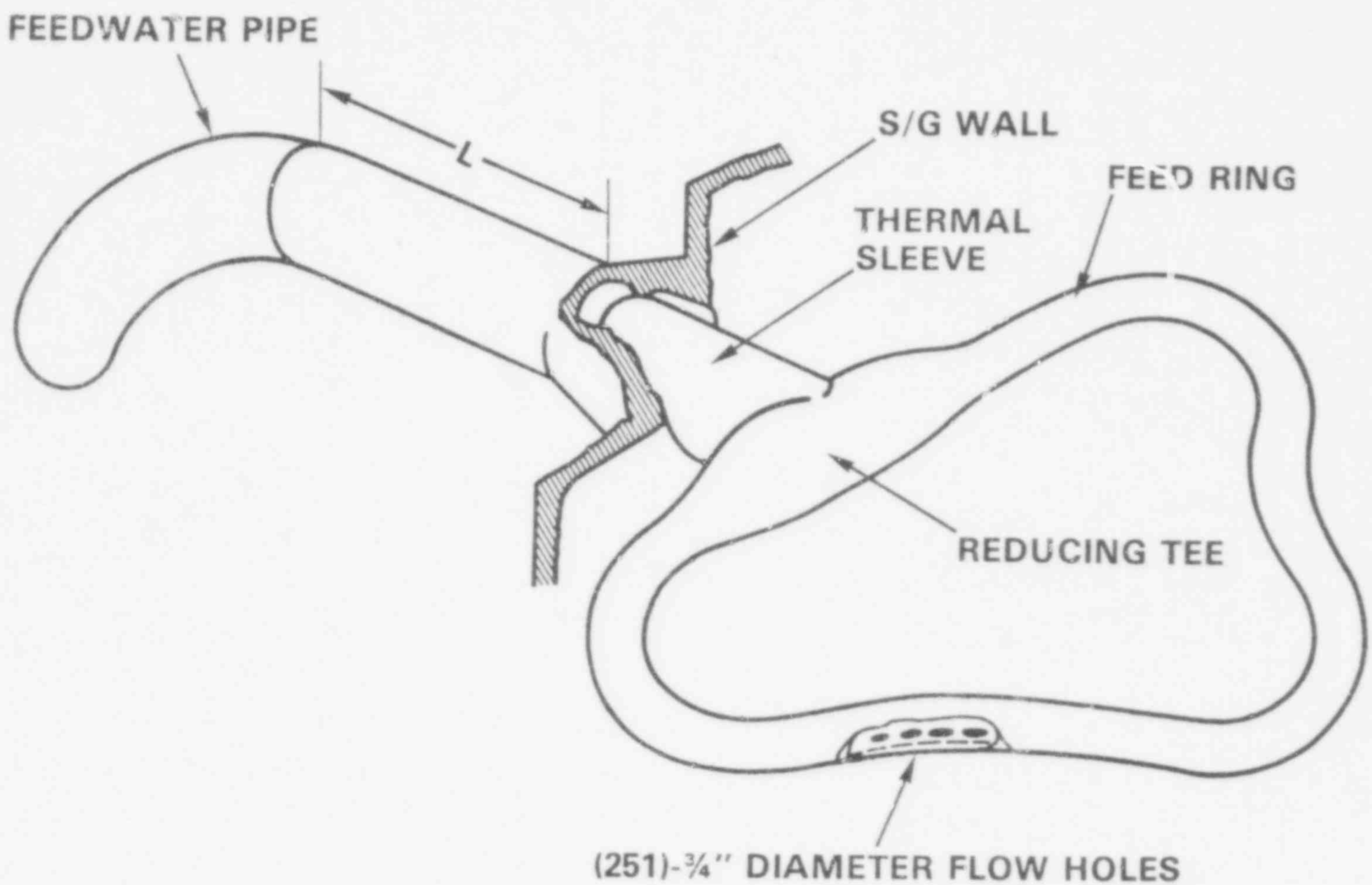

Figure 1

FEEDRING ASSEMBLY
FEEDRING TYPE STEAM GENERATOR

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of full power feedwater flow. Hence, the auxiliary feedwater flow alone cannot keep the horizontal run of pipe full of water. During water level recovery, the steam in this region will condense on the subcooled auxiliary feedwater in the vicinity of the elbow upstream of the feedring and for some distance down the line so that at least the surface of the water will reach the saturation temperature. At some point the counterflow of steam may disturb the surface of the water and expose more subcooled water surface area which then requires a greater steam flow. The counter flow of steam interfering with water flow can result in a water seal at the tee to the feedring, a rapid condensation of the isolated steam, and the acceleration of a slug of water toward the feedwater piping. This process of slug formation as well as slug formation in the feedring is discussed in Ref. 10.

As discussed in Section 3.3, numerous water hammer events of this type have occurred in operating plants. The damage was usually limited to pipe support or restraint failure. In one instance, however, the feedwater line cracked at the containment penetration due to a water hammer that originated in the piping adjacent to the steam generator and propagated throughout the feedwater system. In view of the potential for significant damage, the only feasible design approach is to prevent the occurrence of this type of water hammer. Methods for precluding this slug impact type of water hammer have included the use of top discharge feedwater rings, short elbows, loop seals, venting, and main and auxiliary feedwater control. Some of these approaches are discussed in Ref. 10.

For all CP and OL application reviews, the NRC now evaluates systems with steam generators that have the feedwater inlet at a high elevation (most of Westinghouse and Combustion Engineering plants) for damaging water hammer effects. The current Standard Review Plan specifies the following requirements for applicants having steam generators utilizing top feed:

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"To eliminate or reduce possible water hammer in the feedwater system:

1. Prevent or delay water draining from the feedring following a drop in steam generator water level by means such as J-Tubes.

2. Minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest possible (less than 7 feet) horizontal run of inlet piping to the steam generator feedring.

3. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater and possible draining of the feedring. Provide the procedures for these tests for approval before conducting the tests."

The top discharge on the feedring prevents the rapid drainage of the feedring, and a short horizontal run of piping on the inlet to the steam generator feedring might reduce the magnitude of a water hammer and also lower the probability of the formation of trapped steam which can condense rapidly and cause water hammer.

Starting with Trojan, all licensed Westinghouse and Combustion Engineering plants equipped with steam generators with a top feedring have met the above criteria and have performed a test to confirm the adequacy of plant operating procedures to avoid damaging water hammer.

The NRC has considered close elbows in conjunction with J-tubes as a means of avoiding those conditions that could result in water hammer. However, the maximum acceptable length of horizontal piping could not be ascertained. The CREARE study did not determine the probability of slug formation as a function of pipe length. Furthermore, a simple conservative calculation based on the

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Joukowski equation shows that with a steam space only one foot long, and a water slug one foot long accelerated by the steam generator pressure, the resultant slug impact pressure could be over 10,000 psi. As noted previously, however, the use of short lengths may reduce the probability of slug formation.

The CREARE report states on page 217 that top discharge devices greatly reduce the drainage rate while rapid reestablishment of feedwater flow reduces the drainage period. Only if these recommendations are followed together can these means act to limit the volume of water drained and the consequent size of the steam void. This combined effect has not been included in the PWR vendor position statements. With a top discharge feedring, the drain rate is about 20 gpm compared to a minimum auxiliary feedwater flow of about 100 gpm. The auxiliary feedwater could be initiated within about 30 seconds - feeding into an almost full feedwater line and feedring (feedring capacity 100 to 200 gallons). It therefore is reasonable to expect that the feedring can be filled prior to contact between cold auxiliary feedwater and any residual steam in the feedring.

We have concluded that the most effective way of preventing water hammer caused by this phenomenon is to keep the feedwater line and feedring filled with water at all times.

In September 1977, the NRC informed all PWR licensees that water hammer events due to the rapid condensation of steam in the feedwater lines of steam generators represented a safety concern and that further actions by licensees having Westinghouse and Combustion Engineering designed nuclear steam supply systems are warranted to assure that an acceptably low risk to public safety due to such events is maintained. These licensees were requested to submit proposed plant design and/or procedural modifications, if any, which would be necessary to assure that the feedwater lines and spargers remain filled with water during normal as well as transient operating conditions. Reviews of the licensee responses are being made under the generic review program "Steam Generator Feedwater Flow Instability" (Ref. 18).

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Recent developments have indicated that steam generators with preheaters (RESAR-41 and CESSAR-80) may have problems with steam condensation in the preheater box area. If cold auxiliary feedwater is pumped through the main feedwater line to the preheater, the entrapped steam might condense rapidly and result in damaging water hammer. Preliminary results of 1/8-scale tests by Westinghouse indicate that water hammer occurs in the preheater box of the steam generators. Westinghouse recommends that the auxiliary feedwater be pumped into the top of the steam generator at low power levels to eliminate the possibility of steam condensation in the box area. Westinghouse has provided test data to demonstrate the effectiveness of plant modifications to avoid water hammer. The test data are presently being reviewed. An area of concern with the 1/8-scale tests is the ability to extrapolate the results of these tests to full-scale plant designs. Under Task 4.0 of Task Action Plan A-1, this potential water hammer which could damage the preheater box, tubes, and the feedwater piping is being evaluated to identify the most appropriate plan of action.

Boiling water reactors (BWRs) also have a feedwater ring inside the reactor pressure vessel but the holes are located on the side instead of on the bottom of the ring. Many of the BWR feedwater systems include two feedwater lines that enter the containment dry well and then split into four lines that connect to four feedwater nozzles on the reactor vessel. Connected to the inside of the nozzles are four separate sections of the sparger in a segmented ring around the inside of the reactor vessel. The HPCI system is connected to one feedwater line just outside of containment. There is almost no horizontal feedwater piping connected to the reactor vessel nozzles since a downward 90° elbow is connected close to the nozzle.

The sequence of events following a turbine trip in a BWR is similar to that which occurs in a PWR. Following a turbine trip from full power, the water level in the reactor vessel drops approximately 6 feet, exposing the feedwater ring. The HPCI and RCIC systems are actuated by a low water level signal. The reactor vessel is automatically refilled and the feedwater sparger is recovered after every turbine trip. Although the conditions in a BWR feedwater ring following a turbine trip are similar to those in a PWR steam generator feedwater ring following a turbine trip, there have been no reports of damage to feedwater

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rings or piping due to water hammer from slug impact in a BWR. The absence of a significant horizontal piping run would be expected to prevent slug formation in the piping, while the segmented ring may reduce the probability of slug formation in the ring or at the junction of the ring and the nozzle.

In view of the absence of reported water hammer in the BWR feedwater systems due to slug impact, system modifications should not be required at this time. However, the NRC should continue to monitor the performance of BWR feedwater systems and should review the presently available data to determine whether undetected water hammers may have occurred and contributed to the damage that has been reported in BWR feedwater rings. Under Task 4.0 of TAP A-1, a study will be made to determine if the potential hydraulic conditions in the BWR spargers are similar to those in PWR steam generators. If the results of these studies indicate a significant potential for this type of water hammer, appropriate action will be taken at that time.

A.9 Pump Startup, Stopping, and Seizure with Full Lines

Normal pump startup in systems with full lines usually should not lead to significant dynamic loads. The maximum head developed by the pump is included in the operating pressure, and the transient is relatively slow as the result of pump inertia and the long opening time of downstream valves. However, these dynamic loads may be significant for some systems and should be considered in the design and operating procedures. In one water hammer event in Table 3.2, damage to improperly installed restraints in an HPCI discharge line was attributed to surging of the line during HPCI initiation. Pump stopping is of concern primarily with respect to the possibility of column separation as discussed in Section 4.6. Postulated seizure of pumps in safety-related systems can lead to large dynamic loads and column separation. In view of the limited number of events, no action is warranted for operating reactors.

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APPENDIX B - SYSTEM REVIEW

This appendix contains brief system descriptions and listings of potential water hammer concerns for systems of representative PWR and BWR plants. The various water hammer concerns are identified with specific subsections of Appendix A where more detailed discussions of the water hammer mechanisms are presented.

B.1 Emergency Core Cooling System

The function of the emergency core cooling system (ECCS) is to cool the reactor and in the case of PWR reactors to provide additional shutdown capability (boric acid injection) following certain accidents.

In PWR plants the ECCS is composed of five subsystems:

(1) Core flooding (accumulators) system

(2) High pressure injection system

(3) Low pressure injection system

For the ECCS, the following potential water hammer concerns are identified:

(1) Inadvertent voiding of pump discharge lines for PWRs and BWRs (Section A.1).

(2) Anticipated dynamic loads on pressure relief valve discharge lines and other initially empty water lines (e.g., bypass test lines) for PWRs and BWRs (Section A.2)

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(3) Closing of isolation valves and actuation of pressure relief valves for PWRs and BWRs (Section A.3)

(4) Closing of HPCI turbine stop valve for BWRs (Section A.3)

(5) Closure or delayed opening of check valves in pump discharge lines for PWRs and BWRs (Section A.4)

(6) Intermittent closure of check valves in exhaust line of HPCI turbine for BWRs (Sections A.4 and A.7)

(7) Water entrainment in steam supply and exhaust lines of HPCI turbine for BWRs (Section A.5)

(8) Column separation effects from postulated pump seizure for PWRs and BWRs (Section A.6)

(9) Collapse of steam bubbles formed as result of local or system depressurization (Section A.7)

(10) Dynamic loads from pump startup and seizure for PWRs and BWRs (Section A.9)

B.2 Residual Heat Removal System

For pressurized water reactors, the RHR system serves to remove decay heat during the latter stages of plant cooldown and to maintain the plant in a cold shutdown condition. It also serves dual ECCS functions of injecting water from the refueling water storage tank to the reactor vessel and of recirculating coolant to the reactor vessel from the containment sump following a LOCA. The system may be used to provide initial RCS circulation prior to startup and to fill and drain the refueling cavity during refueling operations.

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For boiling water reactors, the RHR system functions in the shutdown cooling mode and provides the ECCS function of low pressure water injection from the suppression pool to the reactor following a LOCA. The system is used in the pool cooling mode to remove heat from the suppression pool and in the containment spray mode to limit pressure and temperature in the torus and drywell following a LOCA. Finally, for some plants the RHR system is used in the steam condensing mode to condense reactor steam while the reactor is isolated from the main condenser and makeup water is supplied by the reactor core isolation cooling (RCIC) system.

For the RHR system, the following potential water hammer concerns are identified:

(1) Inadvertent voiding of pump discharge lines for PWRs and BWRs (Section A.1)

(2) Anticipated dynamic loads on pressure relief valve discharge lines and other initially empty water lines (e.g., test lines) for PWRs and BWRs (Section A.2)

(3) Closing of isolation valves, control valve actuation, and opening of pressure relief valves for PWRs and BWRs (Section A.3)

(4) Closure or delayed opening of check valves in pump discharge lines for PWRs and BWRs (Section A.4)

(5) Water entrainment in discharge lines of pressure relief valves and steam supply lines for RHR heat exchangers of BWRs (Section A.5)

(6) Column separation effects from postulated pump seizure for PWRs and BWRs (Section A.6)

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(7) Steam bubble formation in RHR heat exchangers of BWRs due to steam leakage (Section A.7)

(8) Inadvertent water valve opening when BWR RHR system operating in steam condensing mode (Section A.7)

(9) Steam bubble formation due to depressurization for PWRs and BWRs (Section A.7)

(10) Dynamic loads from pump startup and postulated seizure for PWRs and BWRs (Section A.9)

B.3 Reactor Coolant System

The reactor coolant system (RCS) in a PWR consists of between two to four similar loops connected in parallel to the reactor pressure vessel. Each loop has one or two pumps and a steam generator. A pressurizer with safety/relief valves is connected to one loop. In some plants, the RCS has loop isolation valves. In addition, there are connections with isolation valves to other systems such as ECC, RHR, and chemical and volume control systems.

For a BWR, there are two recirculation loops which provide the driving flow for jet pumps within the reactor vessel. Each loop has a variable speed pump or constant speed pump with a flow control valve and isolation and bypass valves. In addition, there are connections, with isolation valves from the recirculation loops or reactor vessel to the ECC, RHR and RWCU systems.

For the reactor coolant system, the following potential water hammer concerns are identified:

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(1) Discharge lines of pressurizer relief and safety valves for PWRs (Section A.2)

(2) Actuation of flow control valves in BWR recirculation loops (Section A.3)

(3) Actuation of pressurizer safety and relief valves for PWRs (Section A.3)

(4) Column separation effects following rapid flow control valve closure in BWRs and postulated pump seizure in PWRs and BWRs (Section A.6)

(5) Postulated pump seizure for PWRs and BWRs (Section A.9)

B.4 Main Steam Supply System

In a PWR, the main steam lines transfer the high pressure steam to the main turbine as well as to auxiliary equipment such as the auxiliary feedwater pump turbine or reheat steam. Each steam line is furnished with atmospheric relief valves and several code safety valves. Main steam line isolation is accomplished by a hydraulically operated globe valve and a swing check valve or a bidirectional isolation valve in each line just outside containment. In the event of loss of load, a fast acting turbine stop valve will operate to prevent additional steam flow to the main turbine and the steam flow is then bypassed to the steam dump system. The swing check valves prevent backflow of steam from the other steam generators for a steam line break inside containment. Isolation and control valves are also located in lines leading to the auxiliary systems.

In the BWR, the main steam lines conduct steam from the reactor to the main steam turbine, as well as to the reactor feed pump turbines and auxiliary systems such as the HPCI and RCIC turbines. After leaving the reactor vessel, the steam passes through main steam isolation valves to the turbine.

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The safety/relief valves are mounted on a horizontal portion of the main steam lines inside the drywell.

For the main steam system, the following potential water hammer concerns are identified:

(1) Anticipated dynamic loads in discharge lines of safety and relief valves for PWRs and BWRs and atmospheric dump valves for PWRs (Section A.2)

(2) Opening of safety, relief, and atmospheric dump valves (Section A.3)

(3) Closing of turbine stop/control valves and isolation valves and inadvertent opening of isolation valves (Section A.3)

(4) Inadvertent closure or closure following postulated line rupture of check valves for PWRs (Section A.4)

(5) Water entrainment in steam lines for PWRs and BWRs (Section A.5)

B.5 Main and Auxiliary Feedwater Systems

The main feedwater system, in series with the condensate system, provides feedwater at the required temperature, pressure, and flow rate to the reactor for boiling water reactor (BWR) plants and to the steam generators for pressurized water reactor (PWR) plants. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the low pressure feedwater heaters to the feedwater pumps, and then is pumped through the high pressure feedwater heaters to the nuclear steam supply system. On BWR plants the feedwater is pumped through a feedring into the reactor while on PWR plants the feedwater is pumped through a feedring into a steam generator. Usually at least one check valve and one motor operated valve isolate the steam generator or reactor from that part of the feedwater

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system outside the containment. A control valve in the feedwater line is used to regulate the flow.

The auxiliary feedwater system in some installations is normally operated during startup, hot standby and shutdown to supply feedwater system for pressurized water reactor (PWR) plants. A typical system has redundant auxiliary feedwater trains with pumps separated from the steam generator by at least one motor operated valve and one check valve. Usually the auxiliary feedwater pumps feed into the steam generator through the main feedwater piping system.

For the main and auxiliary feedwater systems, the following potential water hammer concerns are identified:

(1) Rapid opening, closing, and instability of feedwater flow control valves for PWRs and BWRs (Section A.3)

(2) Check valve closure in pump discharge lines for PWRs and BWRs and in steam supply lines of auxiliary feedwater turbines (Section A.4)

(3) Water entrainment in steam supply and exhaust lines for turbine drives of main and auxiliary feedwater pumps for PWRs and BWRs (Section A.5)

(4) Transient cavitation effects following pump stopping or control valve closure for PWRs and BWRs (Section A.6)

(5) Slug impact due to rapid condensation for PWR steam generators and spargers at the reactor vessel of BWRs (Section A.8)

B.6 Containment Spray System

The containment spray system for a PWR is designed to provide cooling to limit the containment pressure in the event of a high energy line break

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inside of containment. This system does not function during normal plant operation. It consists of pumps, spray headers, and isolation valves. Redundant spray systems operate independently. The pumps initially draw water from the borated water storage tank (BWST). Subsequently, the suction is switched to the containment sump after the BWST is substantially depleted. For BWRs, containment spray water is supplied by the RHR system. The spray lines are normally voided from the inboard isolation valves to the spray headers. The pumps are actuated by a safety injection signal and the isolation valves subsequently open on a containment spray signal.

For the containment spray system, the following potential water hammer concerns are identified:

(1) Pump startup with inadvertently voided pump discharge lines upstream of the containment isolation valves (Section A.1)

(2) Anticipated transient flow through empty portions of the discharge lines from the containment isolation valves to the spray headers (Section A.2)

(3) Isolation valve closure in the pump discharge lines (Section A.3)

(4) Check valve closure in the pump discharge lines (Section A.4)

B.7 Component Cooling Water and Service Water Systems

These systems provide essential cooling to safety-related equipment and may also provide cooling to nonsafety-related auxiliary components that are used for normal plant operation.

These water systems vary from plant to plant and the components served by the system may differ. Basically, the essential service water system is

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composed of two or more pumps per plant taking suction from the ultimate heat sink. The component cooling water system is a closed loop, solid water system with redundant heat exchangers cooled by the service water system. The pump discharge goes into a header with appropriate motor-operated valves in the line so that the pumps can be isolated from each other or 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 during an accident and/or the nonseismic service water trains are isolated during accident conditions from the essential seismic Category I water trains by quick acting isolation valves. Essential loads needed for shutdown or accident conditions, but not necessary during normal plant operation, have quick acting (solenoid or air operated) valves to bring the equipment on-line when needed.

Water hammer induced failures in both redundant trains or a water hammer failure in one train and an active single failure in the other could result in the loss of core and containment cooling following normal shutdown or postulated accidents. These failures would be significant for long-term cooling following a postulated LOCA.

For the component cooling water/service water systems the following potential water hammer concerns are identified:

(1) Pump startup with inadvertently voided discharge lines (Section A.1)

(2) Pump startup or pump restart in lines with vacuum breakers and air release valves (Section A.2)

(3) Check valve closure (Section A.4)

(4) Transient cavitation (water column separation) effects following pump stopping (Section A.6)

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(5) Steam bubble formation in heat exchangers (Section A.7)

(6) Pump startup and stopping (Section A.9)

B.8 Reactor Core Isolation Cooling System

The RCIC system for a BWR is designed to provide core cooling during reactor shutdown and loss of the reactor feedwater system by pumping makeup water into the reactor pressure vessel. The system has one centrifugal pump with a steam turbine drive unit and associated piping and instrumentation. Water is supplied from either the condensate storage tank or the RHR heat exchangers (with RHR in steam condensing mode). The suppression pool is a backup supply. The water is discharged to the reactor vessel (feedwater line or head spray) or to a full-flow test return line to the condensate storage tank. Reactor steam taken from a point upstream of the main steam isolation valves is fed to the RCIC turbine and exhausted to the suppression pool.

For the RCIC system, the following potential water hammer concerns are identified:

(1) Pump startup with inadvertently voided discharge lines (Section A.1)

(2) Anticipated transients in empty water lines (Section A.2)

(3) Turbine stop valve and isolation valve closure (Section A.3)

(4) Closure of check valves in turbine exhaust line and pump discharge line (Section A.4)

(5) Water entrainment in turbine supply and exhaust lines (Section A.5)

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B.9 Other Systems

There are systems of potential concern which do not have a safety function; however, they are connected to the primary system. A pressure pulse generated in these other systems could result in a piping failure at the connection. The reactor water cleanup system in BWRs and those portions of the chemical and volume control system from PWRs which are not safety-related are two such systems that potentially could initiate a LOCA.

These systems or portions of systems contain pumps and isolation and check valves; however, no significant safety water hammer incidents have been attributed to them. No action is recommended for operating reactors because of the absence of any adverse experience. These systems should be considered in the generic studies covering isolation control and check valve closures to evaluate the significance of potential water hammers.

For the CVCS and RWCU systems, the following potential water hammer concerns are identified:

(1) Isolation and control valve closure (Section A.3)

(2) Check valve closure (Section A.4)

B.10 Systems With Essentially No Water Hammer Concerns

There are other systems in nuclear power plants not connected to the primary or secondary systems that were judged not to have significant water hammer concerns because of long records of favorable experience, separation of redundant systems, or other mitigating design features. A brief discussion of these systems is presented below.

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Emergency Diesel Engine Cooling Water System

This is a liquid filled system with an expansion tank that circulates water between the diesel and heat exchanger. A temperature regulator valve controls flow. Favorable operating experience, as indicated by the results of preoperational and routine inservice testing, has not shown water hammer to be a problem with existing system designs. In addition, preoperational testing will identify any new system designs where it may be a problem thus permitting redesign of the system before operation.

Emergency Diesel Engine Starting System

Air accumulators used for the starting system can collect water because of the compression process. The quantity of water is normally reduced by the installation of moisture separators and air driers between the compressors and receivers, or by periodic blowing down of the receivers. The effectiveness of these measures is verified by preoperational and routine inservice testing.

Fire Protection System

Water hammer that is caused by pump startup with a partially voided or voided line is a possibility in the fire protection system. The void would occur between the water supply and the fire pump discharge check valve. The design of the system avoids this problem. The fire protection system normally is a water solid system from the water source through the pumps to the discharge of the system. To keep the system solid, priming supplies or other devices are "provided for fire pumps that may at any time take suction under a lift." (Ref. 27) For a fire protection system to be approved for insurance purposes as well as for NRC approval, the use of one of these methods to prevent water hammer is required. In the event of failure of one pump due to water hammer, other fire pumps and the backup fire suppression systems would still be available.

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Opening of the quick-acting valve in the sprinkler system would send a slug of water down the dry-pipe, which would impact against the end of the pipe, causing a possible pipe jump. Most dry-pipe sprinkler systems also use air pressure in the dry-pipe side to keep the valve shut. Upon a decrease in the air pressure the valve opens letting in the water. The residual air that is left acts as a cushion to reduce the impact of the water slug. Even if a water hammer occurs and damages the sprinkler system and/or its adjacent piping, as happened at one plant, the damaged sections can be isolated from the rest of the fire protection system. The fire can be fought using hoses which are required as backups to the sprinkler system and are separated from the sprinkler systems (see Ref. 28).

Spent Fuel Pool Cooling System

The spent fuel pool cooling system is backed up by other safety systems which could provide water to maintain coverage of the spent fuel if the spent fuel pool cooling system fails.

Circulating Water System

A failure in the circulating water system could potentially inundate the auxiliary building and damage safety-related equipment. All plants have been reviewed to assure that essential equipment is not jeopardized by circulating water system failures.

Effluent Treatment System

The effluent treatment system is reviewed to assure that a major failure such as a tank rupture will not result in unacceptable dose consequences.

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Other Systems

Systems presented in Table B-1 were considered and judged not to represent safety concerns with respect to water hammer.

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TABLE B.1

SERVICE SYSTEMS NOT CONSIDERED FOR POTENTIAL WATER HAMMER WITH SAFETY SIGNIFICANCE

1. Diesel Generator Fuel Systems
2. Primary Makeup Water System
3. Condensate Makeup System
4. Condensate System
5. Process Sampling System
6. Drain System
7. Diesel Generator Lube Oil System
8. Potable Water System
9. Instrument Lines
10. Process Steam Lines
11. Extraction Steam Lines

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APPENDIX C:* TASK ACTION PLAN A-1 "WATER HAMMER"

1. DESCRIPTION OF PROBLEM

Since 1971 there have been about one hundred incidents involving water hammers in BWRs and PWRs that have been reported. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater and steam lines. The 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. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage.

No water hammer incident has resulted in the release of radioactivity outside of the plant. However, because of the continuing incidence of water hammer events, the number of phenomena, and the potential safety significance of the systems involved, systematic review procedures should be developed to ensure that water hammer is given appropriate consideration in CP and OL licensing reviews and in reviews of operating reactors. There is also a need for systematic investigations of potential water hammer phenomena to obtain information to be used in providing guidance for the licensing review process and developing NRC positions on water hammer for use in the SRP. These investigations will also provide guidance and methods for understanding and resolving water hammer problems in existing plants.

*This appendix contains the first three sections of TAP A-1 as presented in NUREG-0371.

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2. PLAN FOR PROBLEM RESOLUTION

The overall program for resolution of the water hammer issue is divided into four tasks.

Task 1 Water Hammer Summary Reports

Under this task the initial and final summary reports on water hammer will be prepared.

Task 1.1 Water Hammer Report by DOR/DSS Technical Review Group

An interdivisional (DOR/DSS) Technical Review Group on Water Hammer Phenomena was established on March 10, 1977. In accordance with its charter, this Group will prepare a report that will "review operating experience and analytical investigations to date, place the safety significance of water hammer phenomena in nuclear plants in perspective, and summarize the current staff position regarding water hammer phenomena for CP and OL reviews and reviews of operating plants." A draft of this report has been prepared. Extensive revisions of the draft will be made prior to its submission for approval at the Assistant Director and Director levels. The report will provide input for Tasks 2.0, 3.0 and 4.0.

Task 1.2 Final Summary Report on Water Hammer

A NUREG report will be prepared which summarizes the results of this Category A task on water hammer.

Task 2 Revision of CP and OL Review Procedures

The objective of this task is to develop systematic review procedures concerning water hammer for use in the CP and OL review process.

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The Standard Format and Standard Review Plan will be revised to require the applicant, as appropriate, to: (1) address potential water hammer problems in various systems; (2) demonstrate that there are adequate design features and operating procedures to prevent damaging water hammer events; and (3) expand the preoperational testing program to include verification that these design features and operating procedures do prevent damaging water hammer events. In addition, guidance for the licensing review process will be prepared in the form of Branch Technical Positions for steam generators, feedwater systems and other systems, where required.

Requests for preparation or modification of Regulatory Guides and changes to the Standard Technical Specifications will also be made under this task. In view of the relatively short time scale of the overall water hammer task, performance of the task objectives will not be keyed to the issuance of new or modified Regulatory Guides. However SD will be contacted at an early stage to permit them to make changes in manpower plans for work on the guides.

Work accomplished under this task will be based on the Task 1.1 report and information developed under Task 4.0. Branches assigned primary review responsibility in the SRP will have the responsibility for all revisions of a given section of the SRP and the corresponding section of the Standard Format. This will include the responsibility for obtaining concurrence of any other branch assigned a secondary review or coordination responsibility in the given section of the SRP.

Task 3 Water Hammer Positions for Operating Reactors

Task 3.1 Short-Term Position

The DOR/DSS technical review group concluded that continued short-term plant operation is justified in view of the low probability of a water hammer resulting in unacceptable consequences. However, the staff also

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concluded that a particular type of water hammer, namely that due to the rapid condensation of steam in feedwater lines of PWRs, represents the most immediate potential safety concern and that further actions by licensees were warranted to assure that an acceptably low risk to public safety is maintained. This is appropriate since steam generator feedwater line water hammers are well enough understood at this time to permit staff action. Accordingly, a generic position addressing this concern is being developed by DOR for operating plants and will be transmitted to affected licensees. A request for licensee proposed plant modifications to eliminate this concern and a more comprehensive reporting of water hammer events in the future will be included.

Task 3.2 Long-Term Position

Following completion of Tasks 2.0 and 4.0 and based on further data from operating plants, an assessment will be made of the need for any further requirements to be imposed on operating plants for other types of water hammer events. This assessment, which will include an impact/value appraisal, will consider all types of water hammers which are found to be significant to safety under Task 4.0.

Task 4 Water Hammer Safety Studies

The basic objective of this task is to obtain information and develop analytical methods and calculations regarding water hammer which will be used in completing the revisions of CP and OL review procedures under Task 2.0 and in implementing the long-term position paper of Task 3.0. The results of this task will also be used in implementing the revised CP and OL procedures developed under Task 2.0 and in the evaluation of water hammer incidents at operating reactors. The major part of the work will be done under technical assistance contracts.

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Task 4.1 Review and Evaluation of Potential Water Hammer Problems

This task, which will be completed under a technical assistance contract, will involve the review and evaluation of those actual and potential water hammer problems considered to be significant in the Task 1.1 report. The first objective is to identify typical scenarios (e.g., basic initiating mechanisms, design features, operating procedures, anticipated transients, and single failures) that could result in water hammer events. The safety significance of the water hammer events will then be assessed in terms of probability of occurrence and consequences. Where necessary, recommendations will be made on possible design or procedural changes to prevent the occurrence or minimize the consequences of the postulated water hammer. Recommendations will also be made on criteria to be used in the licensing process. The second objective is to evaluate design features, operating procedures and systems (e.g., BWR jockey pump system) which are used to prevent the occurrence of water hammer and to make recommendations on criteria to be used in the licensing process. This task will not be concerned with new PWR steam generators which are treated separately in Task 4.3. The interim and final reports on this task will be distributed to responsible branches for consideration in completion of Tasks 2.0 and 3.0.

Task 4.2 Development of Current Information on Water Hammer

The objectives of this task are (1) to provide a state-of-the-art review of experimental and analytical work reported in domestic and foreign literature which is pertinent to water hammer problems in nuclear plants, (2) to monitor Licensee Event Reports and experimental work on LOCA and ECC injection for information pertinent to water hammer, and (3) to ensure that information pertinent to water hammer which is obtained from licensees, vendors and architect-engineers under Task 3.0 and given in applicant responses to questions raised during current CP and OL reviews will be brought to the attention of all responsible branches in DOR and DSS. The

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state-of-the-art review will be accomplished under a technical assistance contract. In support of the review, the Office of International Programs will be requested to obtain information from foreign sources on analyses and tests pertinent to water hammer in nuclear plants. Interim and final reports on the review will be sent to responsible branches. Information from the monitoring functions will be distributed when received via memoranda to responsible branches. The information obtained from the licensees and applicants will be maintained in control files for use by all responsible branches.

Task 4.3 Water Hammer in PWR Steam Generators

A. Current Steam Generator Designs

A number of damaging water hammer events have occurred which involve current steam generator designs with feedwater rings located near the top of the tube bundle and auxiliary feedwater lines connected to the main feedwater lines. A report (NUREG-0291) has been completed under a technical assistance contract in FY 1977 which deals with this water hammer problem.

B. New Steam Generator Designs

Some new steam generator designs incorporate bottom feed and preheater boxes. Recent tests have indicated that these designs may be susceptible to water hammer resulting from rapid steam condensation when cold auxiliary feedwater is added to the preheater. Potential water hammer problems for all new designs will be evaluated under this task. The major portion of the work will be done under a technical assistance program managed by the Auxiliary Systems Branch. Work during FY 1978 will cover review of scaling relatonships presently available and the applicability of 1/8-scale test data in predicting results for full-scale steam generators. The FY 1979 work will involve review and

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evaluation of vendor design changes intended to prevent water hammer and consideration of other possible design changes and operating procedures for preventing water hammer. The results of this task will be used in defining an NRC position on new PWR steam generator designs under Task 2.0.

Task 4.4 Water Hammer Calculations

There is a currently funded contract at Lawrence Livermore Laboratory, managed by the Engineering Branch, which is concerned with calculations of pressure transients and stresses in PWR feedwater lines, using forcing functions assumed to represent those resulting from rapid condensation in the steam generator feedring. A final report of this work is scheduled for the end of FY 1977.

For FY 1978 a new technical assistance program is scheduled. A major objective of the new program is to provide analytical methods and calculations to be used in the evaluation of water hammer incidents at operating reactors. Flow closure functions representing the various initiating events will be formulated. Existing computer programs will then be used to establish the system loading due to water hammer from various initiating events and to establish the sensitivity of these loads to system design parameters and operating procedures. The structural response to the water hammer will be calculated.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

Although water hammer can occur in any LWR and about one hundred incidents have been reported in 400 reactor-years of operation, none have resulted in the offsite release of radioactivity. The systems most frequently affected are the feedwater systems of PWRs. A few incidents have caused pipe failure, but none in a reactor coolant pressure boundary. Adequate protection from potential loss-of-coolant accidents, such as might be initiated by a water hammer event, is provided in plants by the emergency core cooling system. Since the probability of failure due to a water hammer is low and the consequences of postulated water hammer induced accidents would be adequately limited by currently installed redundant engineered safety features, continued operation and licensing of plants can proceed with reasonable assurance that the health and safety of the public is protected while this task is being conducted.

In the interim, means to prevent water hammer in feedwater systems in PWRs are being instituted. Applicants with new steam generator designs are being required to demonstrate that water hammer will not occur in these designs. Therefore, for construction permits, there is reasonable assurance that such a demonstration will be made prior to operation. Applicants with designs in which water hammer has been observed are being required to make appropriate modifications. Thus, plants with top feed steam generators are required to modify the feedring and test the system to assure water hammer will not occur.

Prior to start-up, tests to demonstrate that water hammer will not occur while entering the hot shutdown mode are being required.

Licensees of operating plants with steam generators having an internal feedring have been requested to evaluate the potential for water hammer in their feedwater systems and to make appropriate modifications to assure that water hammer will not occur. These modifications will provide additional assurance that continued operation will not present an undue risk to the health and safety of the public.

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The staff reviewed information on water hammer events obtained primarily from Licensee Event Reports and information requests to licensees. Approximately 30 fluid systems in each of four light water reactor designs were also reviewed to determine the potential for significant water hammer events. Water hammer, as used in this report, was generalized in meaning to include transients involving steam flow (steam hammer) and two-phase flow (e.g., water entrainment in steam lines, steam bubble collapse) in addition to the classical water hammer transients such as those involving valve closing and pump startup in solid water systems. Discussions of nine different types of water hammer problems, involving ten plant systems, are presented.

Completed or scheduled staff actions and staff recommendations concerning water hammer problems are summarized.

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