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POLICY ISSUE
(Information)

March 15, 1984 SECY-84-119

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: RESOLUTION OF UNRESOLVED SAFETY ISSUE A-1, "WATER HAMMER"

Purpose: To inform the Commissioners of the issuance of NUREG-0927, "Evaluation of Water Hammer in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1;" revisions to Standard Review Plan Sections 3.9.3, Rev. 1, "ASME Code Class 1, 2 and 3 Components Supports and Core Support Structures;" 3.9.4, Rev. 2, "Control Rod Drive Systems;" 5.4.6, Rev. 3, "Reactor Core Isolation Cooling System (BWR);" 5.4.7, Rev. 3, "Residual Heat Removal (RHR) System;" 6.3, Rev. 2, "Emergency Core Cooling System;" 9.2.1, Rev. 3, "Station Service Water System;" 9.2.2, Rev. 2, "Reactor Auxiliary Cooling Water Systems;" 10.3, Rev. 3, "Main Steam Supply System;" 10.4.7, Rev. 3, "Condensate and Feedwater System;" and NUREG-0993, the supporting Regulatory Analysis. Issuance of these documents completes the staff's technical resolution of Unresolved Safety Issue (USI) A-1.

Discussion: The central issue of USI A-1 dealt with consideration of the frequency and severity of water hammer events, the resulting loads on piping and equipment, and reported damage to determine whether such water hammer occurrences had resulted in unacceptable impairment of safety systems such that safety functions were unduly compromised; and whether significant accidental release of radionuclides to the environment would be expected as a result of water hammer occurrence.

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__Implementation:__ The revised Standard Review Plan Sections will be used only for review of "custom plant" Contruction Permit applications, and for Standard Plant applications docketed after the issuance of these Standard Review Plan Section revisions, which are intended for referencing in Construction Permit applications. These revisions represent current staff review practices (already used in current case reviews) and, as such, clarify staff review practices and ultimately reduce the burden of the regulatory process.

William J. Dircks
Executive Director for Operations

Enclosures: — IN BP
1. NUREG-0927, Staff's Technical Findings
2. SRP Section 3.9.3, Rev. 1
3. SRP Section 3.9.4, Rev. 2
4. SRP Section 5.4.6, Rev. 3
5. SRP Section 5.4.7, Rev. 3
6. SRP Section 6.3, Rev. 2
7. SRP Section 9.2.1, Rev. 3
8. SRP Section 9.2.2, Rev. 2
9. SRP Section 10.3, Rev. 3
10. SRP Section 10.4.7, Rev 3
11. NUREG-0993, Regulatory Analysis
12. Federal Register Notice

Enclosures 1 through 11 - Commissioners, SECY, OGC, OPE Only

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U. S. NUCLEAR REGULATORY COMMISSION

NOTICE OF ISSUANCE AND AVAILABILITY

NUREG-0927, "EVALUATION OF WATER HAMMER IN NUCLEAR POWER PLANTS-TECHNICAL FINDINGS RELEVANT TO UNRESOLVED SAFETY ISSUE A-1,"

STANDARD REVIEW PLAN (SRP) SECTIONS 3.9.3, 3.9.4, 5.4.6, 5.4.7, 6.3, 9.2.1, 9.2.2, 10.3, AND 10.4.7 (NUREG-0800)

AND THE SUPPORTING VALUE/IMPACT ANALYSIS

The U. S. Nuclear Regulatory Commission (NRC) staff has prepared the following documents: NUREG-0927 entitled, "Evaluation of Water Hammer in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1;" revisions to Standard Review Plan Sections 3.9.3, Rev. 1, "ASME Code Class 1, 2 and 3 Components Supports and Core Support Structures;" 3.9.4, Rev. 2, "Control Rod Drive Systems;" 5.4.6, Rev. 3, "Reactor Core Isolation Cooling System (BWR);" 5.4.7, Rev. 3, "Residual Heat Removal (RHR) System;" 6.3, Rev. 2, "Emergency Core Cooling System;" 9.2.1, Rev. 3, "Station Service Water System;" 9.2.2, Rev. 2, "Reactor Auxiliary Cooling Water Systems;" 10.3, Rev. 3, "Main Steam Supply System;" 10.4.7, Rev. 3, "Condensate and Feedwater System;" and the supporting Regulatory Analysis (NUREG-0993) which are included in this final issuance package.

These documents serve as the staff's technical resolution of the NRC's Task A-1, "Water Hammer." This issue was identified as an "Unresolved Safety Issue" in the 1978 Annual Report, pursuant to Section 210 of the Energy Reorganization Act of 1974.

NUREG-0927 provides the staff's technical evaluation of reported water hammer occurrences, attendant damage, underlying causes and corrective actions taken.

All changes to the SRP Sections resulting from the resolution of this Unresolved Safety Issue and any editorial changes are identified by a line in the margin of the revised SRP Section. NUREG-0993 contains the staff's Regulatory Analysis of actions being taken.

Comments were solicited from interested organizations, groups and individuals through a Federal Register notice of availability for comment in May 1983. The staff has evaluated the comments received, and addressed them, as appropriate, in the final documents. As a result of comments received, two SRP Sections (3.9.3 and 3.9.4) not previously published "for comment" have also been revised.

Copies of the documents will be available after ________ ______. Copies will be sent directly to utilities, utility industry groups and associations and environmental and public interest groups. Other copies will be available for review at the NRC Public Document Room, 1717 H Street, NW, Washington, D. C.; and the Commission's Local Public Document Rooms located in the vicinity of nuclear power plants. Addresses of these Local Public Document Rooms can be obtained from the Chief, Local Public Document Room Branch, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, telephone (301) 492-7536.

Dated at Bethesda, Maryland, this _____ day of 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

The staff's concerns were founded on the increasing frequency of water hammer occurrences in the early 1970's, and in particular the feedwater line rupture at Indian Point-2 in 1972 due to a steam generator water hammer.

In the process of the USI A-1 studies, the staff has evaluated reported water hammer occurrences in terms of: (a) reported occurrences documented by Licensee Event Reports (LERs), (b) damage incurred, (c) underlying causes and corrective actions that have been taken, and (d) safety systems (or functions) involved. The major findings can be summarized as follows:

1. Total elimination of water hammer occurrence is not feasible, because inherent in the design of nuclear power plants is the possible coexistence of steam, water, and voids in various nuclear plant systems. Experience shows that design inadequacies and operator- or maintenance-related actions have contributed about equally to initiating water hammer occurrences.

2. Since 1969, approximately 150 water hammer events have been reported through the NRC's Licensee Event Reports (LERs). Damage has been principally limited to pipe support systems. Approximately half of these events have occurred either in the preoperational phase or in the first year of commercial operation. This suggests a learning period existed in which design and procedural deficiencies were corrected and operating errors were reduced.

3. Water hammer frequency peaked in the mid-1970s, at a time when the rate of new plants beginning commercial operation was the highest. Experience led to corrective design changes (e.g., use of J-tubes to eliminate steam generator water hammer and "keep-full" systems, vacuum breakers, etc.) which reduced the frequency of occurrence.

4. Steam generator water hammer (SGWH) associated with top feedring SGs appears to have been corrected by the use of design features and the preoperational test requirements specified in the NRC's Standard Review Plan Branch Technical Position ASB 10-2, "Design Guidelines for Water Hammers in Steam Generators with Top Feedring Designs."

The staff has concluded that the frequency and severity of water hammer occurrences has been significantly reduced through (a) incorporation of design features such as keep-full systems, vacuum breakers, J-tubes, void detection systems and improved venting procedures, (b) proper design of feedwater valves and control systems and (c) increased operator awareness and training. Thus the water hammer issue at the present is less significant than was suggested by the water hammer occurrences in the early and mid-1970s, and the staff has concluded that this unresolved safety issue is technically resolved. Publication of the staff's technical findings in NUREG-0927, and revisions to the Standard Review Plan will provide added assurance of the continued use of plant design features that have been effective in reducing the frequency of water hammer occurrence.

The staff issued these documents and a supporting Regulatory Analysis (NUREG-0993) for a 60-day comment period which ended July 18, 1983. Following the comment period, the staff evaluated the comments received and has addressed them, as appropriate, in the final documents. As a result of comments received, revisions were made to two additional Standard Review Plan Sections, 3.9.3 and 3.9.4 (these were not in the for comment package). Changes were made in the other documents to reflect significant technical inputs received.

The revised documents have been reviewed and endorsed for issuance by the Committee to Review Generic Requirements (CRGR). These reports and Standard Review Plan revisions will be published and notification of availability will be made in the Federal Register (Enclosure 12). Appropriate Congressional Committees will be informed by letter.

Although the above actions provide technical resolution of USI A-1, the Commission should be aware that dynamic loads resulting from potential water hammer events remain an important consideration in the design of piping and piping support systems. NRC's Piping Review Committee is considering water hammer as one of several dynamic loads which must be taken into account in the development of recommendations regarding potential revisions to nuclear power plant piping design criteria. This committee will review and integrate water hammer experience into its review to assure that any proposed revisions to piping design criteria do not adversely affect the capability to withstand water hammer loads.

Part of Secy 84-119 3/15/84

NUREG-0927
Revision 1

Evaluation of Water Hammer Occurrence in Nuclear Power Plants

Technical Findings Relevant to
Unresolved Safety Issue A-1

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

A. W. Serkiz, Task Manager

NUREG-0927, Rev. 1 EVALUATION OF WATER HAMMER OCCURRENCE IN NUCLEAR POWER PLANTS FEBRUARY 1984

NUREG-0927
Revision 1

Evaluation of Water Hammer Occurrence in Nuclear Power Plants

Technical Findings Relevant to Unresolved Safety Issue A-1

Manuscript Revised: December 1983
Date Published: February 1984

A. W. Serkiz, Task Manager

Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

This report, which includes responses to public comments, summarizes key technical findings relevant to the Unresolved Safety Issue A-1, Water Hammer. These findings were derived from studies of reported water hammer occurrences and underlying causes and provide key insights into means to minimize or eliminate further water hammer occurrences. It should also be noted that this report does not represent a substitute for current rules and regulations.

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ACKNOWLEDGEMENTS

The technical findings relevant to the Unresolved Safety Issue A-1, Water Hammer, which are set forth in this report, represent the combined efforts of staff at the Nuclear Regulatory Commission, EG&G - Idaho, Inc., and Quadrex Corporation. The following persons deserve special mention for their participation and contributions:

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NOMENCLATURE

Acronyms

AE	Architect/Engineer
AFW	Auxiliary Feedwater
ALARA	As Low as Reasonably Achievable
ANSI	American National Standard Institute
ASB	Auxiliary Systems Branch
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BTP	Branch Technical Position
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
CE	Combustion Engineering
CRD	Control Rod Drive
CV	Valve Flow Coefficient
CVCS	Chemical and Volume Control System
ECCS	Emergency Core Cooling System
FCV	Feedwater (or Flow) Control Valve
HHSI	High Head Safety Injection
HPCI	High-Pressure Coolant Injection
LOCA	Loss-of-Coolant Accident
LPCI	Low-Pressure Coolant Injection
LWR	Light Water Reactor
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulation
PHSG	Preheat Steam Generator
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWCU	Reactor Water Cleanup
SGWH	Steam Generator Water Hammer
SRP	Standard Review Plan
SRV	Safety/Relief Valve
TAP	Task Action Plan
TSV	Turbine Stop Valve
USI	Unresolved Safety Issue

EXECUTIVE SUMMARY

This report presents the NRC staff's technical findings regarding the Unresolved Safety Issue (USI) A-1, Water Hammer, and presents the results of the concluding evaluations associated with resolving this safety issue and includes responses to public comments. The major findings can be summarized as follows:

1. Total elimination of water hammer occurrence is not feasible, due to the possible coexistence of steam, water, and voids in various nuclear plant systems. Experience shows that design inadequacies and operator- or maintenance-related actions have contributed about equally to initiating water hammer occurrences.

2. Since 1969, approximately 150 water hammer events have been reported through the NRC's Licensee Event Reports (LERs). Damage has been principally limited to pipe support systems. Approximately half of these events have occurred either in the preoperational phase or the first year of commercial operation. This suggests a learning period exists in which design deficiencies are corrected and operating errors are reduced.

3. Water hammer frequency peaked in the mid-1970s, at a time when the rate of introducing new plants into commercial operation was the highest. Experience led to corrective design changes (e.g., use of J-tubes to eliminate steam generator water hammer and "keep-full" systems, vacuum breakers, etc.) which reduced the frequency of occurrence.

4. Steam generator water hammer (SGWH) associated with top feedring SGs appears to have been corrected by the use of design features.

and the test requirements specified in the NRC's Branch Technical Position ASB 10-2, "Design Guidelines for Water Hammers in Steam Generators with Top Feedring Designs."

The major conclusions reached are that the frequency and severity of water hammer occurrence can be and to some extent have been significantly reduced through design features such as keep-full systems, vacuum breakers, J-tubes, void detection systems and improved venting procedures, proper design of feedwater valves and control systems and increased operator awareness and training; and that the current potential for significant damage as a result of water hammer events is less than it was in the early and mid 1970's.

Total elimination of water (steam) hammers is not feasible, due to various inherent features of plant design and operation. Therefore, currently accepted design practices for including anticipated water (steam) hammers as occasional mechanical loads in the design basis of piping and their supports systems should be maintained.

These topics are discussed in more detail in the report which follows.

Although publication of this technical findings report and of associated revisions to the Standard Review Plan complete the staff's work under the Task Action Plan for USI A-1, and constitute technical resolution of the issue as defined therein, the potential for water hammer loads remains an important consideration in the design and operation of nuclear power plants. The staff recognizes the continuing importance of ongoing activities aimed at further reductions in the frequency and possible magnitude of water hammer events and consideration of water hammer effects in nuclear power plant design. These include the activities of the NRC Piping Review Committee, which is currently reviewing regulatory practices related to Pipe Cracks, Pipe Breaks, Seismic Design and Dynamic Load/Load Combinations. Water hammer loads are one of several [illegible] which will be reviewed with respect to experience and current design practices. These activities are integrated with similar activities under the PVRC Technical Committee on Piping where the aim is to improve piping design with respect to dynamic loads. Finally, there are foreign programs aimed at correlating water hammer energy, with damage to piping systems. The joint efforts of these activities should continue.

1.0 INTRODUCTION

This report presents the results of concluding technical evaluations relevant to Unresolved Safety Issue (USI) A-1 Water Hammer, the safety significance of this issue as currently viewed, and potential methods to minimize the frequency and severity of water hammer occurrences.

Water hammer, as discussed in this report, encompasses more than the classical case in which pressure waves, caused by the sudden interruption of flow, are reflected through liquid-filled lines. Water hammers in nuclear power plants have been caused by voiding in normally water-filled lines, steam condensation in lines containing both steam and water, and the entrainment of water in steam-filled lines, as well as by rapid valve action, the classical cause. These underlying causes, the systems they affect, and means for their prevention or mitigation are discussed.

1.1 Safety Significance and Background

The safety significance of a water hammer in a particular system is related to the level of damage incurred (water hammer can introduce large hydraulic loads), the frequency of event occurrence and the safety function of the system.

During the early 1970s the number of water hammer events reported increased. This increase coincided with the increase in the number of plants starting up, as shown in figures 1-1 and 1-2. The staff's concern also increased and were set forth in NUREG-0582 (reference 1). As a result, water hammer was designated an Unresolved Safety Issue (USI) in late 1978.

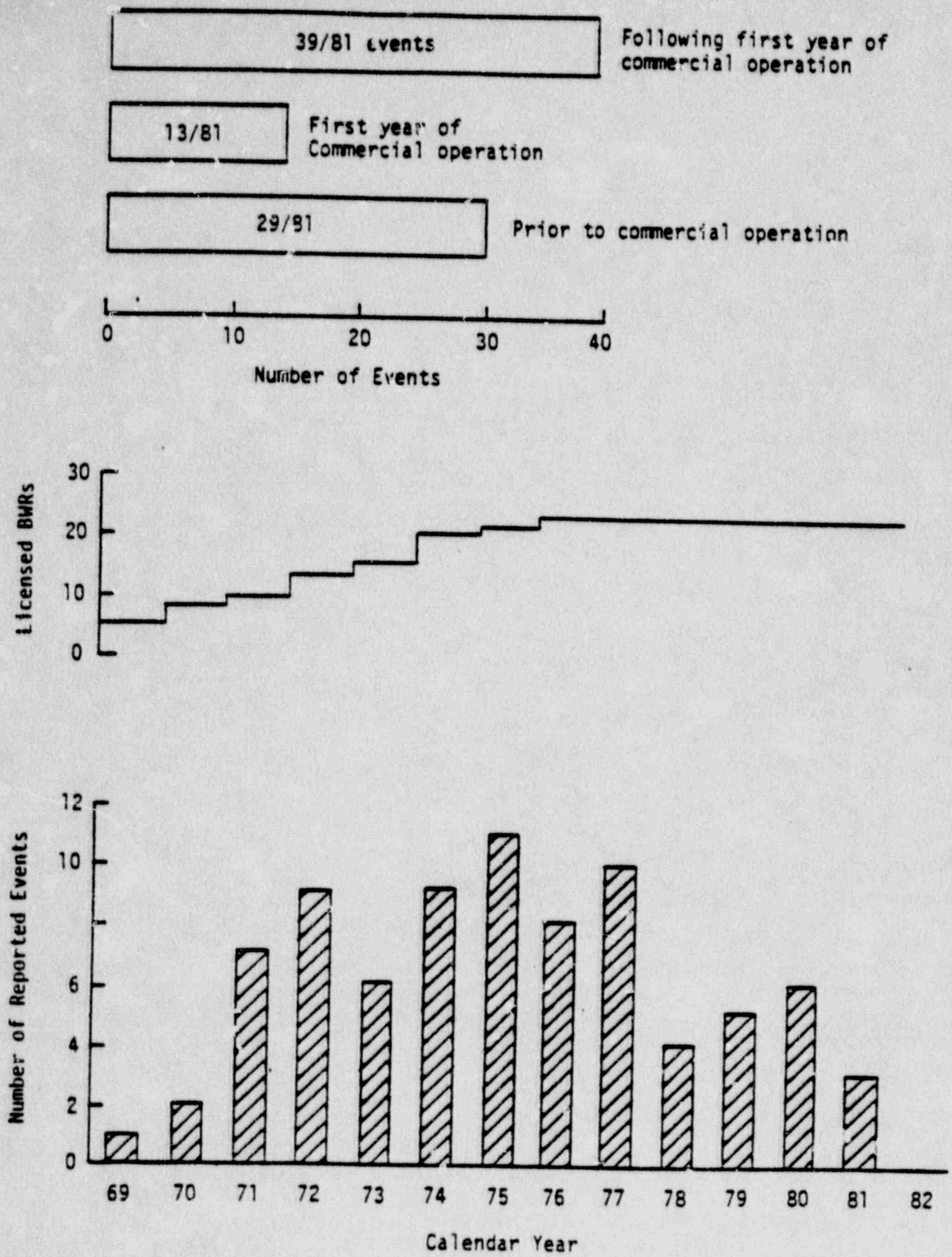

Figure 1-1 Reported Water hammer occurrences in US BWRs.

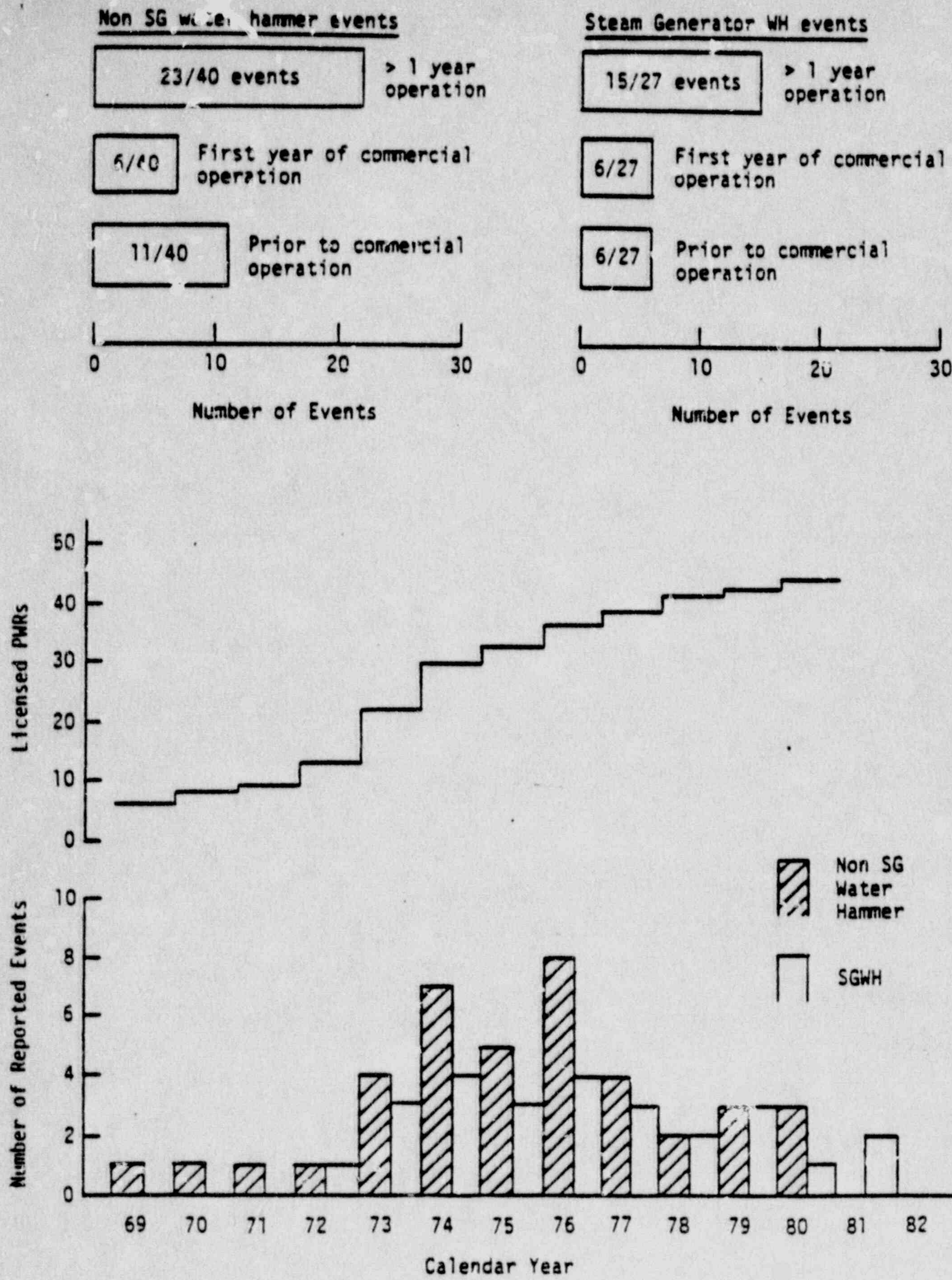

Figure 1-2 Reported water hammer occurrences in US PWRs.

Although much of the water hammer damage was limited to piping support systems, steam generator water hammer frequency increased sharply from 1973 to 1976 and was highlighted by the feedwater line crack occuring at Indian Point 2 in late 1973. An evaluation of PWR steam generator water hammer causes was undertaken in 1976, and the results are reported in NUREG-0291 (reference 2). Recommended design modifications to prevent or mitigate SGWH were embodied in the NRC's Branch Technical Position (BTP) ASB 10-2, "Design Guidelines for Water Hammers in Top Feedring Design," attached to Standard Review Plan (SRP) section 10.4.7 (reference 3).

In addition, efforts were undertaken to analyze water hammers and attempt to predict their occurrence and magnitude (references 4, 5, 6, and 7). Although underlying causes have been identified for sometime (reference 1), the analytical attempts were not successful due to computer code limitations in modeling actual physical phenomena (e.g., steam void collapse), modeling complexity of the interconnected subsystems, and the very large number of system alignments possible under various plant operating conditions. Simply stated, analysis could not provide adeqate information to eliminate water hammer.

Therefore, evaluations of reported occurrences were undertaken to carefully review levels of damage, underlying causes, frequency of occurrence, and system design or operational implications. A compilation of known or suspected water hammer events in nuclear power plants from 1969 to 1981 is contained in reference 8.

1.2 Current Safety Picture

The severity and frequency of reported water hammer events (reference 8) and evaluations of the damage and safety implications (reference 9) indicate that water hammer is not as significant a safety issue as previously thought for the following reasons:

a. The damage for most of the reported water hammer events has been limited to piping and equipment supports.

b. Many of the reported events were either not water hammers, or occurred in nonsafety systems. None of the water hammer events placed a plant in a faulted or emergency condition. None of the water hammer events resulted in a radioactive release.

c. About half the approximately 150 reported water hammer events since 1969 have occurred during preoperational testing and the first year of commercial operation (see figures 1-1 and 1-2). This suggests a learning process and increased operator awareness of the potential for water hammers.

d. Water hammer frequency has decreased. The frequency of water hammer events peaked during the mid-1970s. Experience has brought about design and operational modifications which have reduced the frequency and the potential severity of water hammer in many systems. Two examples are: a) the use of J-tubes in top feedring steam generators to increase drain time, thereby reducing the potential for steam collapse water hammer, and b) the use of jockey pumps in BWR keep-full systems to prevent pump startup into voided lines.

1.3 Key Findings

The key findings, upon which the recommended technical resolution of USI A-1 is based, are as follows:

a. Water hammers continue to occur, but at a low frequency. Total elimination of water hammer is not feasible due to design and operational conditions wherein steam, water, and voids can coexist within a system(s).

b. The overall incidence of water hammer in nuclear power plants has declined considerably in recent years. The decline in the incidence of water hammer events is due to the implementation of various design and operational modifications.

c. The most common cause of water hammer events is line voiding. Other significant causes include steam condensation, feedwater control valve instability, and steam water interfaces. Although these are the generic causes, many of the events have resulted from both design and operational deficiencies. Tables 1-1 and 1-2 summarize water hammer events in PWR and BWR systems, along with attendant safety significance and underlying causes.

d. BWRs continue to report a higher frequency of events than PWRs, primarily because of two factors. The first factor is the susceptibility of BWR ECCS lines to leakage-caused voiding because of the low elevation of the suppression pool which is the ECCS water source. The other factor is the presence of steam-water interfaces in BWRs.

e. Following the implementation of design features and testing contained in BTP ASB 10-2 (reference 3), the frequency of steam generator water hammer in top feedring design steam generators has been essentially eliminated. Additional review of water hammer potential for bottom feed (preheat) steam generators is in process.

f. The frequency and severity of water hammers can be significantly reduced through proper design features, such as keep-full systems, improved venting, void detection, feedwater control valve design verification and vacuum breakers.

g. The frequency and severity of water hammers can be significantly reduced by operator awareness and training, and by improving plant operating and maintenance procedures that cover features such as line warmup, proper valve usage, venting and draining, and void correction.

h. State-of-the-art mechanistic or quantitative two-phase analysis of water hammer phenomena is not a practical means of resolving all water hammer. Although, there are many water hammer events that can be analyzed, the extensiveness of possible plant conditions, alignments, and computer code calculational limits preclude analyzing all possible scenarios.

i. Anticipated water (steam) hammer events, caused by components performing in their intended manner should be included as occasional loads in the design basis of piping and their support systems.

1.4 Report Organization

Section 2.0 provides a more comprehensive technical discussion of reported events, underlying causes, and systems affected. Section 3.0 summarizes key technical findings and provides additional details on measures to prevent or mitigate water hammers.

TABLE 1-1. Overview of Reported PWR Water Hammer Events

System	Number of Events	Safety Significance*	Underlying Causes
Feedwater	13	Moderate	Control valve instabilities
ECCS Safety Injection	4	Moderate	Voided lines (3) Steam bubble collapse (1)
Reactor Primary System	5	Moderate	Relief valve discharge
Cooling Water	3	Moderate	Voided lines (2)
Steam Generators	27	High	Top feedring drainage followed by steam bubble collapse
Main Steam	8	Low	Valve closures/openings (4) Steam-water entrainment (3) Relief valve discharge (1)
Residual Heat Removal	1	Low	Incorrect valve alignment causing voided line
Chemical and Volume Control	2	Low	Voided line, steam bubble collapse
Condenser	4	Low	Design and procedures
TOTAL	67		

*Safety significance as used here is a relative rating based on severity of damage reported, frequence of occurrence, and the role of the particular system involved. These ratings are only relative to water hammer in the other systems listed below. The ratings are not the result of probabilistic risk assessments (PRA) and are not ratings of risk to plant personnel or the public.

TABLE 1-2. Overview of Reported BWR Water Hammer Events

System	Number of Events	Safety Significance*	Underlying Causes
Core Spray	9	High	Voided lines
Residual Heat Removal:			
Shutdown Cooling	7	High	Voided lines (11), steam bubble collapse (8), steam water entrainment (1), unknown (3)
Reactor Vessel Heat Spray	1	Low	
Containment Spray	4	Moderate	
Low-Pressure Coolant Injection	1	Low	
Fuel Pool Cooling	3	Low	
Steam Condensing	7	Moderate	
High-Pressure Coolant Injection	20	High	Steam water entrainment (12), steam bubble collapse (4), voided lines (3)
Cooling Water	9	High	Voided lines (5), design and procedures (2), water column separation (2)
Main Steam	6	Moderate	Steam water entrainment (2) Valve operation (4)
Isolation Condenser	4	Moderate	Steam water entrainment
Feedwater	3	Low	Valve controller instabilities
Condenser	3	Low	Maintenance and design errors
Reactor Core Isolation Cooling	2	Low	Steam water entrainment (1)
Reactor Water Cleanup	1	Low	Unknown
Plant Process Steam	1	Low	Steam bubble collapse
TOTAL	81		

*Safety significance as used here is a relative rating based on severity of damage reported, frequence of occurrence, and the role of the particular system involved. These ratings are only relative to water hammer in the other systems listed above. The ratings are not the results of probabilistic risk assessments (PRA) and are not ratings of risk to plant personnel or the public.

2.0 TECHNICAL DISCUSSION

2.1 Background and Technical Approach

2.1.1 Definition of Water Hammer

The definitions of water hammer types listed below are used in this document.

a. Water (Steam) Hammer. Water (steam) hammer is the change in the pressure of a fluid in a closed conduit caused by a rapid change in the fluid velocity. This pressure change is the result of the conversion of kinetic energy into pressure (compression waves) or the conversion of pressure into kinetic energy (rarefaction waves). Water hammer types are discussed in section 2.1.2.

b. Anticipated Water (Steam) Hammer. An anticipated water or steam hammer is one resulting from a component performing in the manner for which it has been designed and affecting the system in its expected manner. The pressure waves resulting from turbine stop-valve closure are an example of an anticipated event.

c. Unanticipated Water (Steam) Hammer. An unanticipated water or steam hammer is one that would not be expected from a component or system operating in the manner for which it was designed.

d. Nonwater (Non-Steam Hammer) Hammer Hydraulic Transients. Hydraulic transients that do not conform to definition a. above are not considered to be water hammers. Examples of nonwater (steam) hammer transients are steady-state pipe vibrations or oscillations, normal pressure transients and pump instabilities.

2.1.2 Water Hammer Types

The water hammers encountered in nuclear power plants encompass more than the classical case in which pressure waves, caused by the sudden interruption of flow, are reflected through liquid-filled lines. The majority of water hammers occurring in power plants have been caused by the entrainment of water in normally steam-filled lines, steam and water flow in the same line, and voiding in normally water-filled lines.

Water entrained in steam lines causes water hammers when the water slugs are stopped suddenly by obstructions such as closed valves. Further discussion of water entrainment is contained in section 2.4.4.

The presence of nonequilibrium steam and water flow in the same line can cause local steam condensation followed by large pressure drops and rapid slug acceleration. Water hammer forces are generated when the slugs impact a water column or other obstacle. Sections 2.4.2 and 2.4.5 contain detailed discussions of steam bubble collapse phenomena.

Voids can occur in normally water-filled lines for a variety of reasons. When water is pumped into the voided line, the water column accelerates through the void. When the column is suddenly stopped upon impact with an obstacle such as a valve or water column, water hammer forces are generated. A more detailed discussion of voiding is provided in section 2.4.1.

2.1.3 History of Evaluation

Because of the continuing incidence of water hammer events, the variety of phenomena, and the potential safety significance of the systems involved, water hammer was classified as USI A-1, and task action plan TAP A-1 was developed to provide a plan for resolving

USI A-1. However, even prior to the classification of water hammer as a USI, efforts were in process to prevent or mitigate water hammer.

NUREG-0291 (reference 2) presented the results of a study of the cause and effects of past PWR steam generator water hammer (SGWH) events. Recommendations were developed to prevent or mitigate SGWH.

In 1977 letters were issued to certain PWR licensees requiring submission of proposed plant design or procedural modifications to prevent damaging water hammers in the steam generators. Reviews of the licensee responses were made under the generic review program, "Steam Generator Feedwater Flow Instability."

Operating PWR plants having certain feedwater flow control valves were reviewed with respect to actions already taken or still needed to prevent damaging water hammer.

Following the classification of water hammer as a USI, NUREG-0582 (reference 1) was issued. NUREG-0582 was the first overall summary evaluation of water hammer in nuclear power plants. The staff reviewed information on water hammer events obtained primarily from licensee event reports and information requests to licensees. The staff concluded that continued plant operation and licensing was warranted, pending the evaluation of the water hammer issue, as outlined in TAP A-1. However, concurrently, the staff concluded that the overall frequency of water hammer events was unnecessarily large and that corrective steps in design and in plant operation should continue to be pursued through the licensing review process.

Numerous analytical studies were undertaken to analyze various water hammer phenomena. The phenomena analyzed included filling of voided systems (reference 4), fluid transient forcing functions for piping systems (reference 5), steam void collapse (reference 6), and check valve fluid transients (reference 7). The studies were undertaken to determine the limitations and the present state of existing computer codes. Table 2-1 presents the key findings of these studies, including code limitations and recommendations for experimental verification of analytical results.

Because of the analytical limitations, it was concluded that further development of analytical tools was not a feasible solution to the water hammer issue. As an example, even the advanced codes did not hold promise for calculating steam-water condensation phenomena, which would be required for analyzing steam void collapse, which is followed by water slug propagation and impact loads. It was also recognized that the potential combinations of system alignments and plant conditions conducive to water hammer were far too numerous to permit analysis of all potential water hammer scenarios.

Rather than continue analytical studies, emphasis was placed on evauation of events and plant design and operation to prevent or mitigate water hammer. The evaluations, discussed below, represent the major basis for the resolution of the water hammer safety issue.

NUREG-0291, discussed previously, was the basis for the initial evaluation of steam generator water hammers (SGWH). NUREG-0918 (reference 10) presents the plans for the prevention or mitigation of SGWH. NUREG-0291 summarizes causes of SGWH, various design and operating changes employed to prevent or mitigate SGWH, and implementation and status of modifications at each operating PWR plant.

TABLE 2-1

Water Hammer Analytical Studies

Study Ojective	Key Findings
Analyze rapid filling of voided piping systems (BWR core spray line-filling using SOLA-PLOOP computer code.) (reference 4)	1. Modified SOLA-PLOOP hydrodynamics code applicable. 2. Experimental verification of analytical tool recommended.
Formulate analytical procedure to predict structural sequences of fluid transients in nuclear piping systems, (reference 5)	1. Analytical procedure developed for: a. Sudden check valve closure of a BWR primary feedwater line, b. Simulated BWR core spray line experiencing an instantaneous valve opening. 2. Significant potential loads on piping systems.
Investigate steam-void-collapse water hammer initiating mechanisms, (reference 6)	1. K-FIX/MOD1 code judged inadequate due to to treatment of interphase heat transfer and mass transfer. 2. Analysis of experimental data with advanced codes like TRAC or THERMIT advised.
Construct analysis tool for analyzing fluid transients in piping systems having a check valve, (reference 7)	1. RELAP5 adaptable to model check valve, and perform BWR feedwater line transient calculations. 2. Future experimental verification desirable.

NUREG/CR-1606 (reference 11) documents work performed to evaluate condensation-induced water hammer in preheat steam generators. NUREG/ CR-1606 concluded that condensation-induced water hammers could occur in preheat steam generators and recommended each plant be reviewed separately and that appropriate preoperational testing be performed. Such testing is currently being implemented through the Operating License review process.

NUREG/CR-3090 (reference 16) evaluates the potential for water hammer occurrence during AFW operation of preheat steam generators (PHSG). The evaluation concludes that the likelihood of water hammer occurrence during PHSG AFW operation was extremely low. Furthermore, if an event did occur, it should have no adverse effects on AFW system operation or plant safety.

NUREG/CR-2059 (reference 8) presents a compilation of data for reported water hammer events occurring from January 1, 1969, through May 1, 1981. The compilation was performed to facilitate an understanding of the frequency and severity of damage from and the underlying causes of water hammer. For each reported event, available information concerning underlying causes, damage incurred, plant operating conditions and corrective actions taken were presented. NUREG/CR-2059 also provided cross compilations to permit statistical evaluations concerning plant state (e.g., preoperational, first year, or after the first of operation) when the occurrences took place, the systems affected, reactor type and water hammer types. The information presented in NUREG/CR-2059 was the data base used for the systems evaluations reported in NUREG/CR-2781 and for developing most of the findings presented in this report.

NUREG/CR-2781 (reference 9) presents the results of an evaluation of water hammer events in LWR power plants. The evaluation was based upon the data of reference 8, typical plant design drawings and operating procedures. The evaluation identified the susceptibility of plant systems to water hammer and the safety significance of water hammer in plant systems. Generic causes of water hammer were also identified.

Included in NUREG/CR-2781 are design and operating recommendations for the prevention or mitigation of water hammmer occurrence. Most of the findings of NUREG/CR-2781 are incorporated in this report.

2.2 Frequency and Severity of Water Hammer Events

No water hammer incidents have resulted in the loss of containment integrity or the release of radioactivity outside of the plant. The frequency and severity of events in PWR systems are low, with the exception of SGWH and feedwater-control-valve-induced water hammers. The most serious BWR water hammer concern is line voiding.

Water hammer frequency increased as the number of operational reactors increased. Figures 1-1 and 1-2 show the number of reported events and licensed reactors for BWRs and PWRs. This data base shows that approximately half the events occurred during preoperational testing or the first year of commercial operation. On the other hand, not all of the operating plants have reported water hammer. Reference 8 provides water hammer event summaries for reported occurrences from 1969 through mid-1981.

For ease of presentation the discussions contained in this section are divided into three groups, based upon the plants or systems in which they occur. These groups are:

- o PWR steam generator water hammers (SGWH)
- o PWR non-SGWH water hammers
- o BWR water hammers.

2.2.1 PWR SGWH

Thirteen plants reported 27 PWR steam generator events (reference 8). No water hammer event damaged the integrity of the reactor coolant boundary. No water hammer incidents resulted in the loss of containment integrity nor release of radioactivity outside of the plant. In most of the events, damage was nonexistent or limited to the piping support system. Many reported events actually represented a series of several events recorded during a single, short time span at the same plant. Many events were not observed at the time of occurrence, but the damage observed indicated that these events were caused by SGWH. SGWH events have varied greatly in magnitude and consequences. Effects reported have ranged from minor noises and feedwater piping vibration to major feedwater support damage and one feedwater piping through-the-wall crack.

SGWH has occurred in steam generators with top discharge feedring designs. Branch Technical Position ASB 10-2 has been issued by the NRC to complement the corrective and preventive measures for new plants. Safety evaluations of corrective measures were issued for operating Westinghouse and Combustion Engineering-designed systems. SGWH has been essentially eliminated in plants conforming to the measures contained in BTP ASB 10.2 (reference 3).

2.2.2 PWR Non-SGWH

Forty PWR non-steam generator water hammer events were reported in NUREG/CR-2059. None had any adverse safety effect on a plant. No water hammer event rendered a safety system inoperable or damaged the integrity of the reactor coolant boundary. No water hammer incidents resulted in the loss of containment integrity nor release of radioactivity outside of the plant. In most of the events, damage was limited to the piping support system. The frequency and severity of water hammer events, having the potential to affect safety systems, in PWR plants has been low, with the exception of

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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IMAGE EVALUATION TEST TARGET (MT-3)

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steam generator water hammers, which are previously discussed, and feedwater-control-valve-induced water hammers.

Eleven (28%) of the events occurred prior to the plant's commercial operation date, at seven different plants.

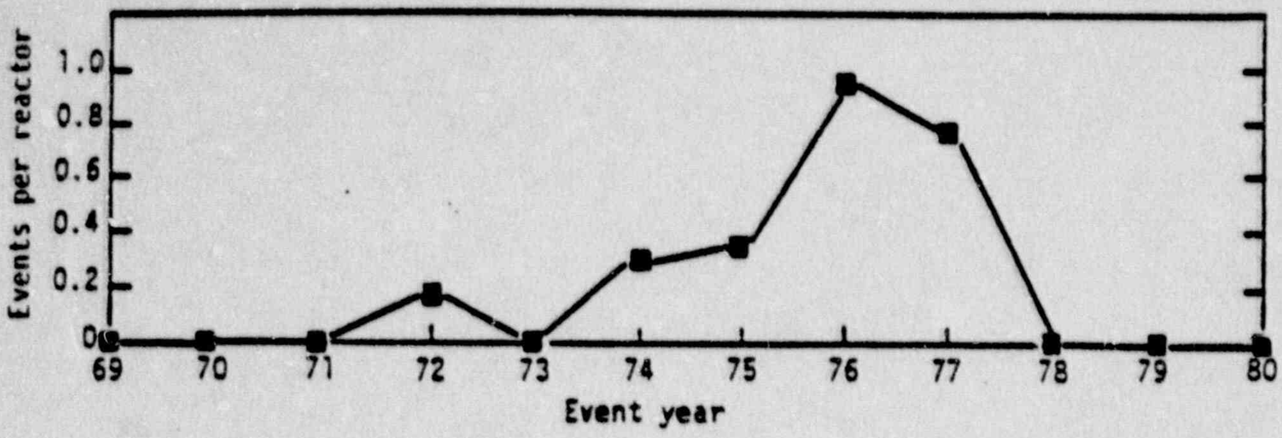
Six (15%) of the events occurred within one year after the plant's commercial operation date, at six different plants.

Twenty-three (58%) of the events occurred after the first year of plant's commercial operation, at 16 different plants.

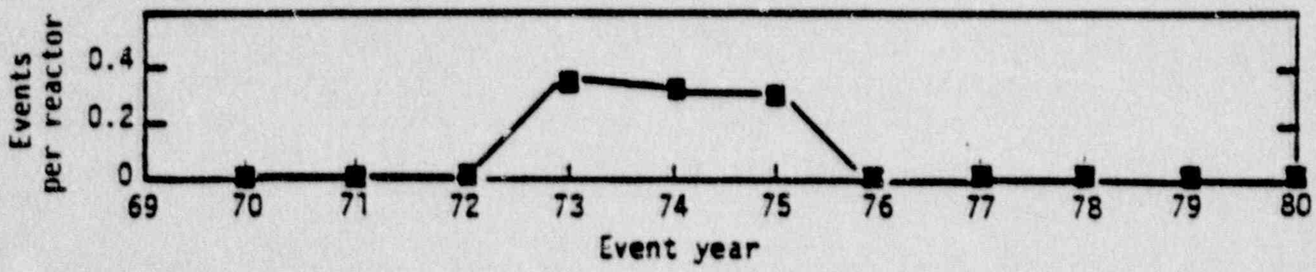
The incident rates (events per year per reactor) for events which occurred after the first year of a plant's commercial operation are lower than the rates of events which occurred prior to that time (see figure 2-1). The average incident rate for all plants after the first year of commercial operation is 0.09 events per year per reactor.

This data illustrates that there is a learning curve in which operational and design deficiencies are corrected. However, water hammers will continue to occur at a relatively low frequency.

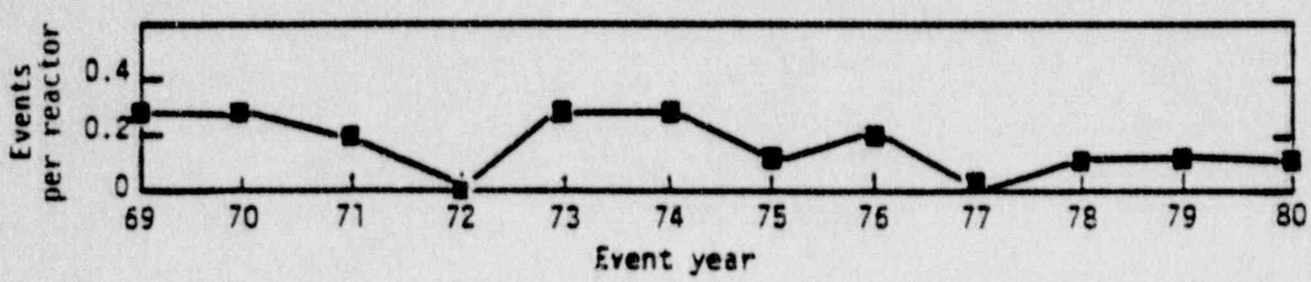
Of the 40 reported PWR non-steam generator events, NUREG/CR-2781 considered only 24 to be unanticipated water hammer events having the potential to affect safety systems. The other 16 events were either not water hammer, did not have the potential to affect safety systems, or were anticipated events that should have been considered in the system design basis. Of the 24 water hammers, that had potential to affect safety systems, 12 occurred in the feedwater system. Eight of the feedwater system water hammers were related to the feedwater control valve. The damage reports indicate



Incident rate prior to commercial operation.



Incident rate during the first year of commercial operation.



Incident rate after the first year of commercial operation.

PWR normalized incident rate

Figure 2-1

that the greatest forces were generated by events occurring in the feedwater system. This is to be expected due to the large line size and the high fluid velocities and high fluid density in the feedwater system.

Of the 16 non-FCV events in the various systems (including FW) in the PWR plants, seven involved line voiding, two involved improper valve usage, one involved a drain malfunction, and one involved a design error. The causes of five of the events are unknown.

2.2.3 BWR Water Hammer Events

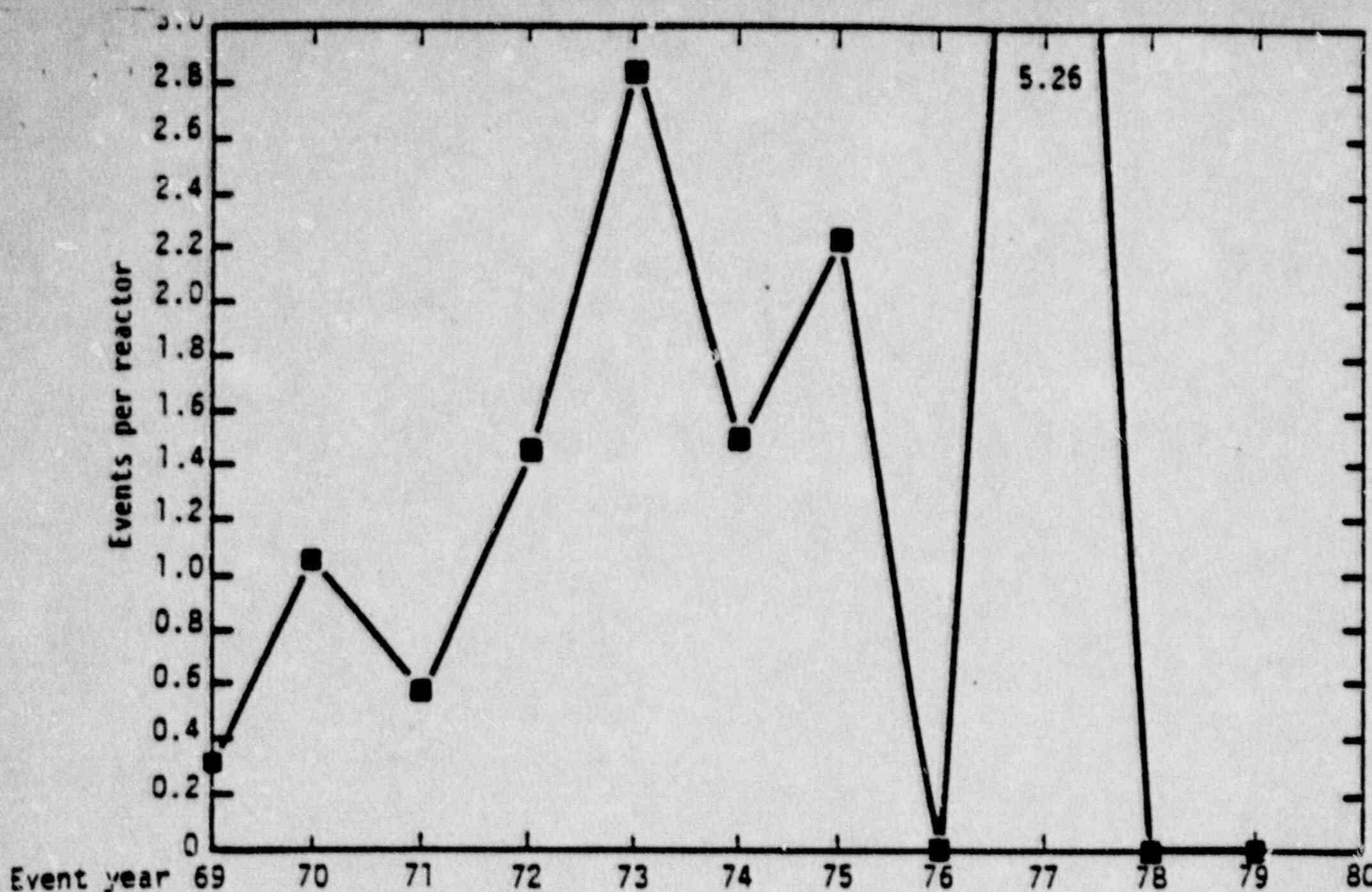
There were 81 BWR water hammer events reported in NUREG/CR-2059. None of the water hammer events placed a plant in a faulted or emergency condition. For most of the 81 events, damage was limited to the piping support system. For some events there was no reported damage. However, 18 of the water hammer events rendered a train of a safety system inoperable. These included two events in which flooding, caused by water hammers in nonsafety systems, rendered a safety system inoperable. No events damaged the integrity of the reactor coolant boundary. No events resulted in the loss of containment integrity nor release of radioactivity outside of the plant.

Twenty-nine (36%) of the events occurred prior to the plant's commercial operation date, at 15 different plants.

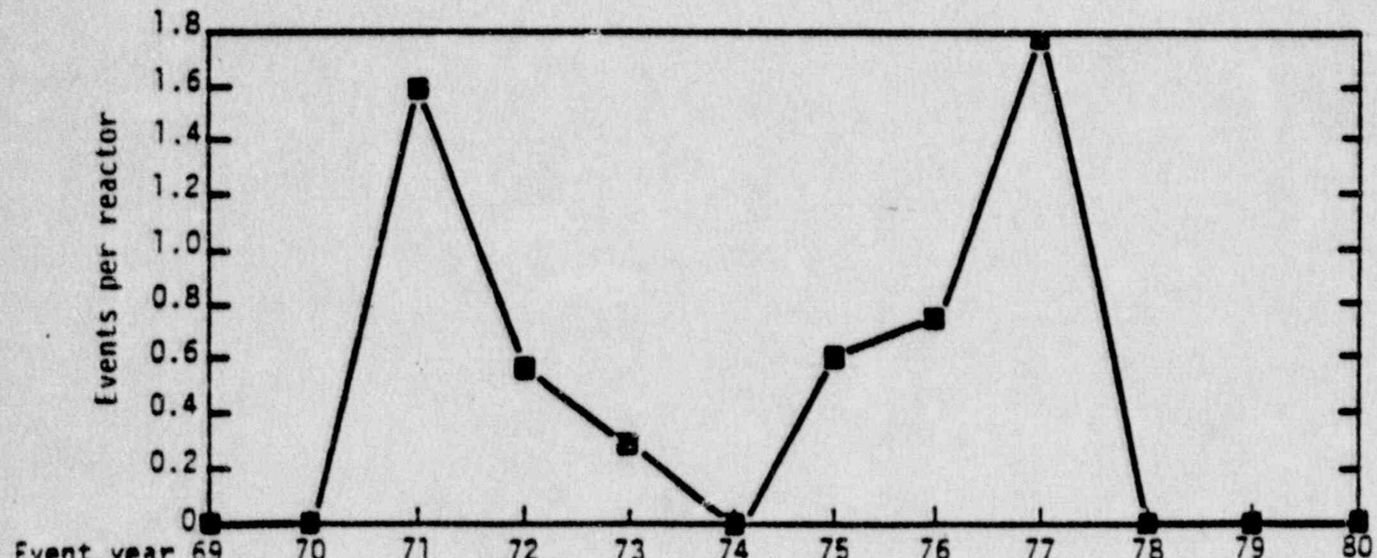
Thirteen (16%) of the events occurred within one year after the plant's commercial operation date at eight different plants.

Thirty-nine (48%) of the events occurred after the first year of the plant's commercial operation, at 17 different plants.

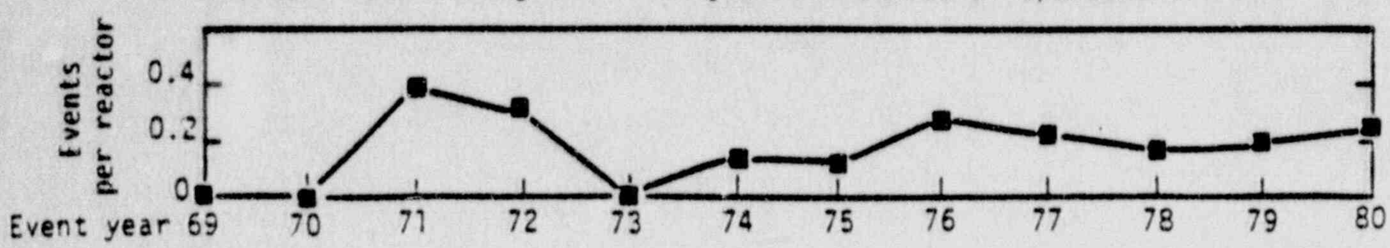
The incident rates (events per year per reactor) for events which occur after the first year of a plant's commercial operation are



Incident rate prior to commercial operation.



Incident rate during the first year of commercial operation.



Incident rate after the first year of commercial operation.

BWR normalized incident rate

82-268

Figure 2-2

lower than rates of events which occur during the first year of a plant's commercial operation. Furthermore, the incident rates for events which occur during the first year of a plant's commercial operation are lower than the rates for events which occurred prior to a plant's commercial operation (see figure 2-2). The average incident rate for all plants after the first year of commercial operation is 0.22 events per year per reactor. Although the frequency of events is higher for BWR plants than PWR plants, a similar learning curve appears to exist for BWRs as does for PWRs.

Of the 81 reported events in the BWR plants, NUREG/CR-2781 considered only 69 to be unanticipated water hammer events that had the potential to affect safety systems. The other 12 events were either not water hammers, did not have the potential to affect safety systems, or were predictable events that should have been considered in the system design basis. Fifty-nine of the 69 water hammer events, having the potential to affect safety systems, occurred in four systems, namely, RHR (23), HPCI (20), core spray (9), and service cooling water (7). Other systems in which water hammer events occurred include isolation condenser (four), RCIC (one), main steam (two), and feedwater (three).

The most serious BWR water hammer concern is line voiding, which is discussed in more detail in section 2.4.1. It was the largest single cause of BWR water hammers and was responsible for at least 39 events. This generic cause includes flow into voided lines, steam-bubble collapse, and possibly some of the unknown events.

Other causes of water hammers having the potential to affect safety systems in BWR plants are: HPCI turbine steam line drain pot failure (seven), improper HPCI turbine steam line warmup (five), improper main steam line warmup (one), feedwater valve controller have been prevented by adequate support design. The loads from these events should have been incorporated into the design basis of the piping support systems in accordance with references 12, 13, 14, and 15. The only unanticipated water hammer event was caused by steam water entrainment duc to improper operation of the MSIVs during line warmup.

There were five RCS pressurizer events reported in reference 1. These events should be considered anticipated hydraulic transients, in which forces are generated by a pressure wave passing through the discharge pipe following relief valve opening. The effect of pressurizer SRV actuation should have been incorporated into the component and pipe support system design basis in accordance with references 12, 13, 14, and 15.

There were four events in the emergency core cooling system (ECCS) at four different plants. Three of the events occurred in active safety injection subsystems during testing or plant operation and were classified as flow-into-voided-line events. Voiding is discussed in section 2.4.1. The fourth event was a steam-bubble collapse water hammer in an accumulator discharge line that occurred during testing while the plant was shutdown.

Two flow-into-voided-line events also occurred in essential cooling water systems.

Four events, occurring in the condenser system, did not have potential to affect safety systems and were possibly not water hammers.

2.3.2 BWR Systems

Twenty-three BWR RHR water hammer events were reported in NUREG/CR-2059. Flow into a voided line caused 16 events. The elevations of the RHR pump discharge lines are higher than the elevation of the pump suction source. This makes the system susceptible to leakage-induced voiding. Further discussion of voiding is presented in section 2.4.1. Six events were steam-bubble collapse events. Steam bubbles in the RHR pump discharge line to the RHR heat exchanger were caused by steam leakage through the steam isolation cooling line isolation valves. Steam-bubble collapse in water-filled lines is similar to line voiding, as discussed in section 2.4.1 One RHR event was a water entrainment water hammer, caused by improper warmup of the HPCI turbine inlet line, which is connected to the RHR steam condensing line.

A total of 20 HPCI system water hammer events were reported in NUREG/CR-2059. The cause listed for most events (12 of 20) is steam-water entrainment. The other events were caused by steam-bubble collapse (four), flow into voided line (three) and unknown (one). Nine steam-water entrainment events occurred in turbine inlet lines and were caused by improper isolation valve usage (four events) and drain pot malfunctions (five events). HPCI isolation valves and drain pots are discussed in sections 2.4.4.1 and 2.4.4.2 respectively. Two steam water entrainment events in the turbine exhaust line were caused by drain level switch malfunctions. The twelfth steam-water entrainment events occurred in a gland seal condenser inlet line and was caused by operator error. The four steam-bubble collapse events were caused by vacuum, occurring in turbine exhaust lines (see section 2.4.5). The three flow-into-voided-line events and the one unknown event occurred in pump discharge lines.

Eight of the nine core spray events were caused by flow into a voided line. The other event was caused by steam-bubble collapse. The core spray system is highly susceptible to leakage-caused voiding, because its discharge lines are at higher elevations than its pump suction lines. Section 2.4.1 discusses voiding.

There were seven essential cooling water system water hammer events reported in reference 8. Two events also occurred in nonessential cooling water systems. Four of the essential system events were caused by flow into a voided line, two were caused by column separation and the cause of one event is unknown.

There were three water hammer and seven vibration incidents in BWR feedwater systems caused by regulating valves. Feedwater valves are discussed in more detail in section 2.4.3.

Six BWR main steam events were reported. One event was caused by an MSIV suddenly opening into an inadequately warmed up line. The other events were anticipated valve closure or relief valve discharge events or in nonsafety-related portions of the system. References 12, 13, 14, and 15 require that the design basis of the piping and support system include these anticipated loads.

There were four steam water entrainment events in the isolation condenser inlet line. One was caused by inadequate line drainage provisions. The other three were caused by high reactor water levels that permitted water to enter the inlet line (refer to section 2.4.4.3).

2.4 Generic Evaluations

The evaluations of references 8 and 9 determined that there are several generic causes of water hammer in nuclear plants. Some of these causes are peculiar to particular systems and others occur in several systems. These generic causes are discussed below.

2.4.1 Line Voiding

NUREG/CR-2781 has identified line voiding as the single greatest cause of water hammer events. Forty-nine percent (46 of 93) of the unanticipated non-SGWH water hammer events having the potential to affect safety systems reported in NUREG/CR-2059 were caused by pumping water into a line containing voids. Voids can occur through many means, including improper line filling, during maintenance, gas evolvement, improper venting, out-leakage of water, in-leakage of steam, and column separation following pump stoppage or valve closure. The generic line-voiding causes discussed in this section include flow into voided lines and steam-bubble collapse. Possibly, some of the unknown events were caused by line voiding.

Generally, voiding occurs in standby systems that are normally idle. Systems that are continually operating, such as feedwater, are started slowly and kept full by continuous operation. BWR systems are more prone to voiding than PWR systems. There are two main reasons for the differences between the BWR and PWR voiding frequency. The first is the elevation of the safety system's water source. The PWR pumps are supplied by the refueling water storage tank, which is maintained at an elevation above the pump discharge lines. The BWR safety systems most prone to line voiding, RHR and core spray, receive their supply from the suppression pool, which is maintained at a level below the elevation of the pump discharge lines. This elevation difference permits fluid in the discharge lines to leak back into the suppression pool. The open loop service water systems for both BWR and PWR plants are also supplied by sources below the level of the system lines. Other systems which experience less voiding are supplied by the condensate storage tank, which in many plants is maintained at a level above the pump discharge lines. The second difference between BWR and PWR plants is the presence of steam-water interfaces in BWRs, permitting the leakage of steam bubbles into the water lines.

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

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

2.4.2 Steam Generator Water Hammer

Steam generator water hammer (SGWH) is defined as a steam-condensation-induced water hammer occurring in the secondary side of a PWR steam generator and the connecting feedwater line. Twenty-seven SGWH events were reported in NUREG/CR-2059, making SGWH the second most common type of water hammer. The mechanism for SGWH is described below (references 2 and 10).

Steam generators in most plants using Westinghouse and Combustion Engineering steam generators have a top feedring through which the feedwater is injected into the downcomer between the baffle and the outer shell. The top feedrings in the Westinghouse and Combustion Engineering designs originally had bottom holes to discharge the feedwater.

During certain plant transients, which occur as a part of plant shutdown operations, the SG water level may drop below the bottom of the feedring sparger. A bottom discharge feedring can be drained of water and filled with steam within 1 or 2 minutes after the feedring is uncovered if feedwater flow has been terminated. As the feedwater (usually highly subcooled auxiliary feedwater) enters the horizontal pipe run into the feedring, it flows under the steam blanket, as depicted in figure 2-3(a). Rapid steam condensation can occur at the interface between the steam and the subcooled feedwater, causing a countercurrent flow of steam over the top of the feedwater. Interaction forces between the steam and water can create enough turbulence to seal off a pocket of steam, as depicted in figure 2-3(b). Continued rapid condensation of steam in the pocket accelerates the slug of water into the void, as depicted in figure 2-3(c).

Acceleration forces on the water slug can be very large, because the pressure on one side is at steam generator pressure, initially in excess of 750 psi, while the pressure on the trapped vapor side can be greatly reduced, depending on condensation rate. As a result, the water slug can have a high velocity when it impacts against the incoming water column, and a pressure pulse is produced (figure 2-3(d). This constitutes one possible explanation of a steam generator water hammer. The magnitude of the pressure pulse and its propagation through the feedwater line depend on many factors. These include the steam void condensation rate, the initial volumes of the void and water slug, steam pressure in the steam generator, sonic velocity in the feedwater line, and piping geometry and layout (references 2 and 10). In a severe SGWH the pressure pulse may be as high as thousands of psi (references 2 and 10).

Most Babcock & Wilcox (B&W) steam generators have a top-discharge externally mounted main feedring (reference 10). Auxiliary feedwater

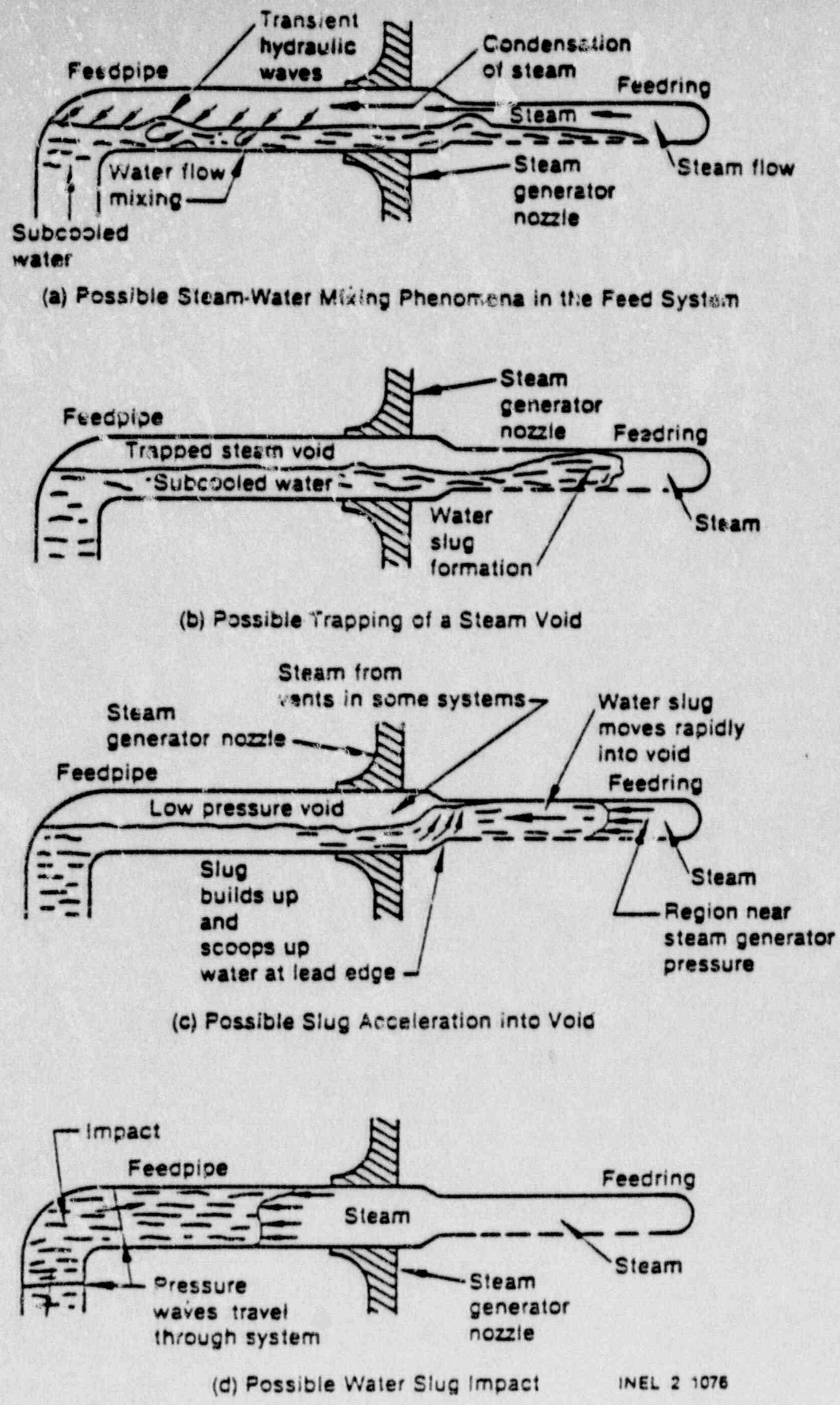


Figure 2-3 Possible sequential events leading to steam generator water hammer

is injected through a separate smaller diameter top-discharge externally mounted auxiliary feedring. B&W steam generators with externally mounted feedrings have not experienced damaging SGWH.

A different steam generator, called the preheat steam generator (PHSG) has recently been introduced. There are two feedwater nozzles in a preheat steam generator (references 11 and 16). The lower (main) nozzle is located at the preheat section and is used for feedwater supply to the steam generator during power operations when hot main feedwater is available. The upper (auxiliary) nozzle is located at the upper section of the steam generator and is used for supplying the feedwater when main feedwater is unavailable or is below a specified minimum temperature. The main nozzle is not used during low power operation because injecting cold feedwater through the main nozzle might cause steam bubble collapse in the preheat section of the steam generator if cold feedwater were injected into the preheat section (reference 11). In the Westinghouse PHSG design, neither the main feedwater nor the AFW line utilize a sparger. In the Combustion Engineering (CE) PHSG design, only the AFW line utilizes a sparger. The AFW sparger in later CE PHSG designs has a loop seal to preclude sparger draining. Many PHSGs also include tempering or bypass flow lines to keep the AFW line full during normal plant operation.

Generic and in-plant testing has shown that PHSG water hammer can be avoided during main feed flow through the use of appropriate procedures, that ensure only hot water is supplied through the main nozzle. Evaluations, reported in NUREG/CR-3090, show that the occurrence of SGWH during AFW operation of PHSG designs is highly unlikely. The occurrence of an SGWH event in a PHSG would require multiple component failures (including several check valves and operator errors). Even if such an event occurred, it is not expected to have an adverse effect on plant safety or AFW system operability.

2.4.3 Feedwater Control Valves

2.4.3.1 PWR Feedwater Control Valves

The major cause of non-SGWH water hammer events in PWR feedwater systems is feedwater control valve (FCV) instability. FCVs contributed to eight of the ten feedwater system events for which a cause could be identified.

The FCV instabilities resulted from such deficiencies as over-sizing of the valve, improper adjustment of the control circuitry, unbalanced valve trim and damage to the valve internal components.

Generally, the NSSS vendor supplies and specifies FCVs. The AE designs the remainder of the condensate/feedwater system, from the condensate pumps to the steam generator. Failure to verify FCV compatibility with the feedwater system has resulted in several designs in which the FCV is incompatible with the remainder of the feedwater system. The most common incompatibility has been valve oversizing. The incompatibility problem can be especially severe for systems containing motor-driven feed pumps, because such systems have very high FCV pressure drops at reduced plant loads. The high pressure drops at low flows tend to decrease valve stability. Systems containing turbine-driven feed pumps are more stable because feedwater flow is partially controlled by varying turbine speed.

2.4.3.2 BWR Feedwater Regulating Valves

Feedwater regulating valve instability and malfunctions caused three water hammers and seven events involving abnormal feedwater line vibration. A possible cause of the feedwater regulating valve instability and malfunctions is valve operator and controller deterioration due to excessive cycling. Additionally, the older designs of the valve operator and controller may have been inadequate. Plants had experienced excessive control system hunting and continuous valve cycling for many years.

Early feedwater regulating valves had an anticipatory control system with an internal feedback loop. This control system was characterized by continuous cycling. After 1976, the loop control system was replaced by one that uses a three-element (water level, steam flow, and feedwater flow) controller at high loads and does not contain an internal feed back loop. Single-element (water level) control is used at low loads, because the three-element controller causes valves cycling at low loads. Cycling occurs because the steam flow signal is not accurate enough at low flow, causing instability in three-element control. The valve actuators were also strengthened to improve their ability to withstand cycling. There have been no BWR feedwater regulating valve incidents reported since the above modifications were made.

2.4.4 Steam Line Water Entrainment

2.4.4.1 Isolation Valve Operation

Improper operation of steam line isolation valves can cause water entrainment in steam lines. One BWR and one PWR main steam water hammer were caused by opening a main steam isolation valve (MSIV) too rapidly. The rapid valve opening in lines that were not completely warmed up caused steam condensation. The water slugs, formed by the condensate, caused water hammers upon impact with closed turbine stop valves.

Four HPCI turbine inlet line water hammers were caused by isolation valve operation. There are no provisions for draining the HPCI turbine steam supply line upstream of the outboard isolation valve. Therefore, if an isolation valve is closed, water will accumulate in the line upstream of the valve. Normally, the outboard valve is opened; then the inboard isolation valve is opened slowly for gradual admission of steam. The outboard isolation valve often has a "seal-in" control feature that causes the valve to open fully in a noninterruptible manner; thus, the valve cannot be opened gradually. When the outboard valve is opened with the inboard valve fully open, the steam flow rate builds up rapidly. Liquid that was in the line between the valves can flow rapidly through the line and suddenly stop at the turbine stop valve, generating a water hammer.

2.4.4.2 Drain Pot Operation

Five water hammer events were caused by steam trap failures in the HPCI turbine inlet line. Two similar events occurred in the HPCI turbine exhaust line. HPCI is the only system in which water hammers caused by steam trap failures have been reported.

In the HPCI turbine steam supply line, the drain pot can fail to drain through the outlet steam trap because of plugging of the steam trap orifice. If the drain pot high level switch fails to open the steam trap bypass valve, water will accumulate in the drain pot and steam line. Under these conditions, initiation of steam flow can cause a steam-water entrainment water hammer. During normal HPCI standby conditions, the drain pot will be nearly empty. The level switch and bypass valve are rarely cycled. Such infrequent usage is conducive to the level switch or valve sticking. If the level switch is inoperative, a high water level can occur in the drain pot without opening the bypass valve or providing any indication to the operator.

2.4.4.3 Isolation Condenser Inlet Line

Four water hammers occurred in isolation condenser inlet lines. One event was caused by improper line slope that did not permit condensate drainage. The other three events were caused by high reactor water levels that allowed water to be drawn into the normally steam-filled isolation condenser inlet line.

2.4.5 Turbine Exhaust Line Steam-Bubble Collapse

Five water hammers were caused by turbine exhaust line steam condensation. Four were in the HPCI system and one was in the RCIC system.

Unless the turbine exhaust line contains vacuum breakers, rapid steam condensation in the exhaust line can create a vacuum, drawing a water slug from the suppression pool into the exhaust line. The water slug, traveling at a high velocity, impacts the check valve disc, resulting in a fast valve closure that can cause a water hammer. Short operational periods that can occur during testing are particulary conducive to condensation, because the turbine housing and exhaust line inside walls can remain cool and provide a subcooled condensing surface for the stagnant steam remaining in the pipe and turbine after shutdown.

2.4.6 Operator Training

Most of the reported water hammer events involved plant operators and maintenance personnel to a varying degree. They frequently write the plant operating procedures, and ultimately approve them. The operators start the pumps, open the valves and place systems in operation, test them, and maintain them.

Over 50% of the events occurred during plant startup and in the twelve months following commercial operation. This indicates there is a learning period during which plant personnel and management become familiar with system operations, change procedures, correct design errors, modify equipment such as vents and drains, and reduce their errors. To be most effective, efforts to reduce water hammer events should start before plant operation and the learning-by-experience period begins.

NUREG/CR-2781 reported the following general causes of operator-involved events:

- o A lack of awareness often exists among plant operators concerning the possibility of water hammer events occurring in a particular system or subsystem, their causes, and what the results of those events would be. Plant operators know from experience that water hammers occur, but they have not had specific training as to why or where they happen, what systems are susceptible, or what corrective actions are possible.
- o Sufficient information is often unavailable to the operators concerning the conditions in the system before the water hammer events occur.
- o Equipment malfunctions and maintenance-related failures of components, such as shutoff valves, steam traps, and check valves, are often not fully considered by designers and plant operators with respect to causing water hammer events.

Many water hammer events can be eliminated by design changes that provide the operator with more information (e.g., void detection and improved steam drain pot level indicators), preclude adverse conditions (e.g., vacuum breakers and keep-full systems) and minimize the potential for operator error (e.g., valve interlocks and operability requirements). However, there are many operations, such as line warmup and venting, that require operator knowledge of system conditions. Therefore, it is important that plant operators, including personnel responsible for writing maintenance instructions and supervising maintenance activities, receive training in the causes and prevention of water hammer.

2.4.7 Operating and Maintenance Procedures

Many the water hammer events were reportedly caused by inadequate operating and maintenance procedures. Additionally, other events might have been avoided had different procedures been available. Because required operator actions are controlled by procedures, more adequate operating and maintenance procedures would aid in reducing the frequency of water hammer events.

Certain good practices that aid in preventing water hammer, such as gradual line warmup, controlled valve opening, draining, and venting, are usually covered by procedures. However, the potential for water hammer is generally not considered in either procedure writing or review (reference 9), although most procedures require line venting where appropriate.

2.4.8 Anticipated Loads

Certain loads, such as steam hammer due to rapid valve closure or forces caused by safety and relief valve actuation, are to be expected and are predictable. As an example, turbine stop valves typically close in approximately 0.1 to 0.2 seconds, causing steam hammers. Other anticipated loads include sudden pump startup and filling of an empty piping system that cannot be kept full, such as an open loop cooling water system. The forces generated by these loads should be considered in determining the design basis for the piping, its support system, and other components, such as valves. The inclusion of these loads in the design basis for piping is required by ASME B&PV Code section III, ANSI B31.1 and SRP 3.9.3 (references 12, 13, and 14).

2.4.9 Control Rod Drive (CRD) Hydraulic Lines

No water hammer events have been reported in CRD systems. However, analyses (reference 17) have shown that transient piping loads can be generated during scram CRD hydraulic system actuation. The results of these analyses were submitted as public comment to this document. The forces generated by these loads should be considered in determining the design basis for the piping, its support system, and other components, such as valves. The inclusion of these loads in the design basis for piping is required by ASME B&PV Code section III, ANSI B31.1 and SRP 3.9.3 (references 12, 13, and 14).

2.5 Corrective Actions

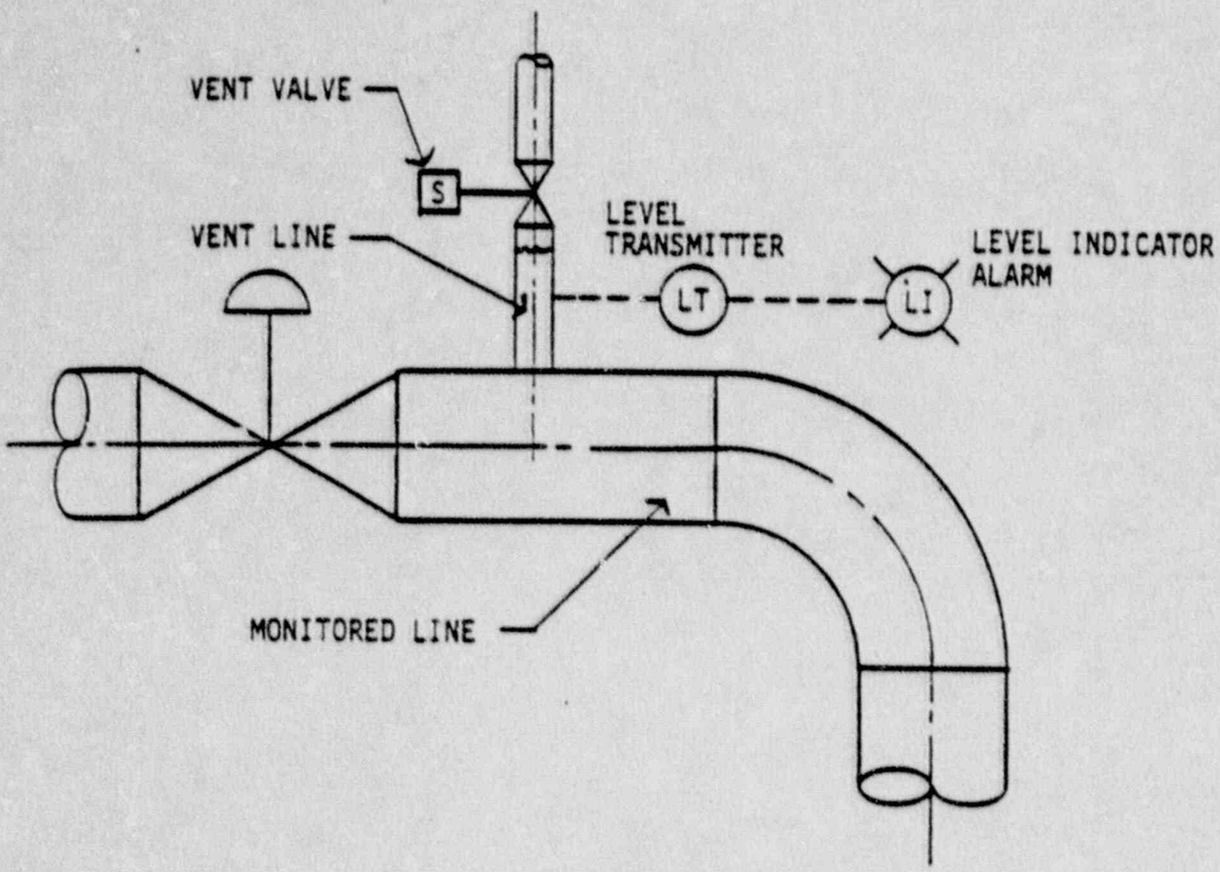
The corrective actions discussed below provide means to prevent or mitigate water hammers.

2.5.1 BWR Plants

2.5.1.1 Design Features

a. Void Detection. Void detection and alarm can be provided for the applicable systems. Void detection mechanisms could be located at those points in the normally liquid-filled lines where voids or steam bubbles could form or collect and have the potential to cause a damaging water hammer in a safety system. All void points that have the potential to cause damaging water hammers in a safety system could be monitored. No specific void detection designs are suggested; however, figures 2-4 and 2-5 show possible void detection design concepts.

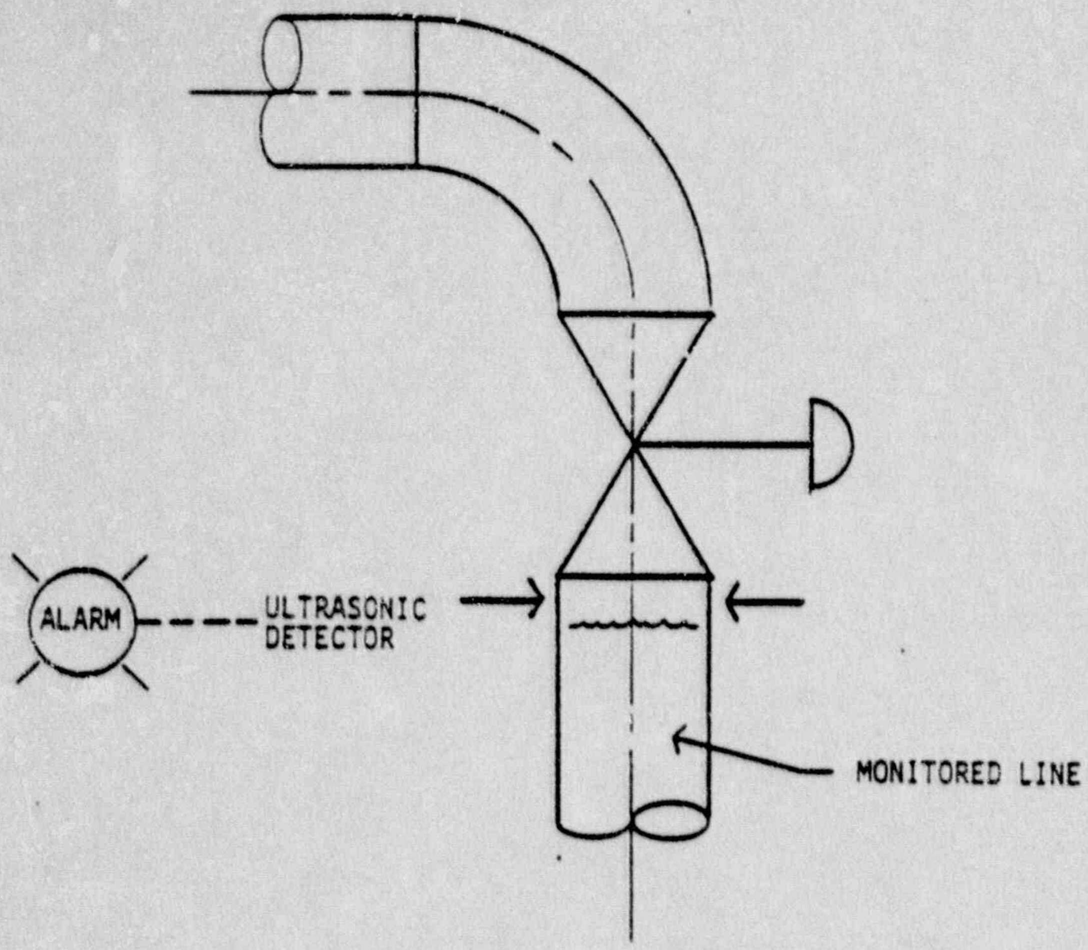
   The operability requirements for the applicable system should require rapid correction of voids having the potential to cause damaging safety-related water hammers. Use of the system other than for emergency use could be prohibited until these voids are vented and filled.



LEVEL TRANSMITTER IN VENT LINE DETECTS THE INCIPIENCE OF VOIDING AND PROVIDES ALARM.

FIGURE 2-4

CONCEPTUAL DESIGN OF POSSIBLE VOID DETECTION SYSTEM



ULTRASONIC DETECTOR, DETECTS VOID IN VERTICAL LINE AND PROVIDES ALARM.

FIGURE 2-5

CONCEPTUAL DESIGN OF POSSIBLE VOID DETECTION SYSTEM

It is difficult to define a maximum acceptable void size. Such a definition would require extensive case-specific transient analysis. Given the current state of the art, the results of such analysis would have limited credibility. Furthermore, the accurate determination of void size in a horizontal or sloped line would require a sophisticated measuring system. It may, therefore, be desirable to eliminate voids as soon after their inception as possible, rather than quantatively define and determine an acceptable size.

Voiding in open-loop service water systems may be considered acceptable if analysis has been performed to demonstrate that there will be no adverse effects if the system is started with voids present.

Applicable systems:

- o Residual heat removal
- o Core spray
- o High pressure coolant injection
- o Essential service water.

b. <u>Keep-Full Systems</u>. Continuously operating keep-full systems should be provided for filling voids in normally water-filled lines in the systems listed below. A jockey pump or a storage tank at a higher elevation than the lines of concern may be considered to be an adequate keep-full system.

Applicable systems:

- o Core spray
- o High-pressure coolant injection
- o Reactor core isolation cooling
- o Residual heat removal.

c. Filling Safety-Related, Open-Loop Service Water Systems. One of the following should be demonstrated for open-loop service water systems:

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

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

Applicable systems:

- o Residual heat removal
- o Core spray
- o High-pressure coolant injection
- o Essential cooling water
- o Reactor core isolation cooling.

e. Turbine Exhaust Line Vacuum Breaker. Vacuum breakers should be provided in the turbine exhaust lines that have a liquid interface.

Applicable Systems:

- o High-pressure coolant injection
- o Reactor core isolation cooling.

f. HPCI Steam Line Drain Pot.

1. The adequacy of the sizing of the HPCI drain pot system should be verified.
2. Those systems in which operational verification and maintenance of level switches cannot be performed while the system is in service should be modified to permit such verification and maintenance.

g. HPCI Turbine Inlet Line Isolation Valves. Neither valve should contain a seal-in feature on opening when in the manual mode. The inboard valve should be designed to permit gradual line warm up.

h. Feedwater Control Valve. The mutual compatibility of the feedwater control valve and the feedwater system should be verified. Valve design parameters, including actuator, flow coefficient (CV), and trim, should be compatible with all final designed operating conditions of the condensate and feedwater system. Furthermore, the valve and its control system should be designed to minimize the potential for oscillation instability, vibrations, and water hammer.

Valve design features that minimize instability include balanced trim design for all pressure drop and flow configurations, stiff actuators, moderate rate of operator response, long valve strokes and minimal pressure drop. These features, however, should be designed to be compatible with achieving proper control.

i. <u>Steam Hammer</u>. The design basis for the main steam system components and piping supports should consider steam hammer resulting from the most rapid anticipated closure of all system valves, including the turbine stop valves.

j. <u>Relief Valve Discharge</u>. The design basis for the main steam system components and piping supports should consider fluid forces resulting from safety and relief valve operation.

k. <u>Control Rod Drive</u>. The design basis for the control rod drive (CRD) hydraulic system should include water hammer loads resulting from the worst case CRD actuation.

2.5.1.2 <u>Operational Features</u>

a. <u>Operator Training</u>. Plant operators, including personnel responsible for writing maintenance instructions and supervising maintenance activities, should receive training on the causes and prevention of water hammer. The training will make it possible for the operators to become aware of a potential water hammer situation and take preventive or corrective measures.

b. <u>Operating and Maintenance Procedures</u>. The licensee should review all operating, maintenance and testing procedures for the systems listed below for their appropriateness in preventing water hammer.

Applicable systems:

- o Residual heat removal
- o High-pressure coolant injection
- o Core spray

- o Essential service and cooling water
- o Isolation condenser
- o Feedwater
- o Main steam.

c. Voiding. All potential void points which can cause a damaging water hammer in a safety system could be monitored. These systems should not be used when voids are present, other than for emergency use. Voids should be corrected as soon as possible.

Applicable systems:

- o Residual heat removal
- o High-pressure coolant injection
- o Core spray
- o Reactor core isolation cooling
- o Essential service and cooling water.

Voiding in open-loop service water systems may be considered acceptable if analysis has been performed to demonstrate that there will be no adverse effects if the system is started with voids present.

d. HPCI Steam Line Drain Pot. The level indicators on the HPCI drain pot system should be checked for operability periodically and repaired if necessary.

e. HPCI Turbine Inlet Line Isolation Valves. Procedures should prohibit both opening the inboard isolation valve unless the normally open outboard isolation valve is fully open and closing the normally open outboard valve unless the inboard valve is fully closed. The provisions should apply when the valves are in the manual mode, for all operating conditions except cold shutdown.

2.5.2 PWR Plants

2.5.2.1 Design Features

a. Void Detection. The actions are the same as those listed for BWR plants in section 2.5.1.1.a.
Applicable systems:
- o Emergency core cooling (safety injection)
- o Essential service water.

b. Filling Safety-Related Open-Loop Service Water Systems. The actions are the same as those listed for BWR plants in section 2.5.1.1.c.

c. Venting. The actions are the same as those listed for BWR plants in section 2.5.1.1.d.
Applicable systems:
- o Emergency core cooling (safety injection)
- o Essential service water.

d. Feedwater Control Valves. The actions are the same as those listed for BWR plants in section 2.5.1.1.h.

e. Steam Hammer. The actions are the same as those listed for BWR plants in section 2.5.1.1.i.

f. Relief Valve Discharge. The actions are the same as those listed for BWR plants in section 2.5.1.1.j. Applicable systems:
- o Main steam
- o Reactor coolant (pressurizer).

g. Steam Generator Water Hammer

1. Top Feed Designs
   - o The feedring should incorporate top discharge J-tubes.
   - o The horizontal feedwater pipe entering the steam generator should be as short as possible, preferably less than seven feet long.
   - o Automatic initiation of AFW should be provided.

2. Preheater Designs
   - o Minimize the horizontal lengths of feedwater piping between the steam generator and the vertical run of piping by providing downward turning elbows immediately upstream of the main and auxiliary feedwater nozzles.
   - o Provide a check valve upstream of the auxiliary feedwater connection to the top feedwater line.
   - o Provide for maintaining the top feedwater line full at all times.
   - o Automatic initiation of AFW should be provided.

3. Once-Through Designs
   - o Provide auxiliary feedwater to the steam generator through an external header.
   - o Automatic initiation of AFW should be provided.

2.5.2.2 Operational Features

a. Operator Training. The actions are the same as those listed for BWR plants in section 2.5.1.2.a.

b. Operating and Maintenance Procedures. The actions are the same as those listed for BWR plants in section 2.5.1.2.b. Applicable systems:
   - o Emergency core cooling (safety injection)
   - o Feedwater
   - o Main steam
   - o Essential service and cooling water.

c. <u>Steam Generator Water Hammer</u>

1. <u>Top Feed Designs</u>

   - Auxiliary feedwater flow should be initiated as soon as possible following loss of main feed flow. This will prevent draining of the feedring.
   - Tests should be performed to verify that procedures for recovering steam generator water level will not result in SGWH.

2. <u>Preheater Designs</u>

   - Auxiliary feedwater flow should be initiated as soon as possible following loss of main feed flow. This will prevent draining of the upper feed line.
   - Tests should be performed to verify that procedures for recovering steam generator water level will not result in SGWH.
   - Tests should be performed to verify that the procedures for switching from the AFW nozzle to the main feed nozzle will not result in water hammer.
   - Maintain the top feedwater line full at all times.

3. <u>Once-Through Designs</u>

   - Auxiliary feedwater flow should be initiated as soon as possible following loss of main feed flow.
   - Tests should be performed to verify that procedures for recovering steam generator water level will not result in SGWH.

3.0 WATER HAMMER PREVENTION AND MITIGATION MEASURES

Section 3.0 provides additional details concerning the measures for water hammer prevention and mitigation contained in section 2.5. Because these measures may be applied to more than one system, the following generic subjects are identified:

- o Void Detection Systems
- o Keep Full Systems
- o Venting
- o Filling Safety-Related, Open-Loop Service Water Systems
- o HPCI Turbine Inlet Line Valve
- o Feedwater Control Valve and Controller
- o HPCI and RCIC Turbine Exhaust Line Vacuum Breakers
- o Steam Hammer
- o Relief Valve Discharge
- o HPCI Turbine Inlet Line Drain Pot Level Detection
- o Operator Training
- o Operating and Maintenance Procedures.

Water hammer assessments are generally performed during systems evaluations rather than during generic evaluations. Therefore, a systematic approach has been developed for using these findings. Table 3-1 summarizes for each BWR system the water hammer events that have occurred in the system and lists both design and operational means for water hammer prevention and mitigation. It is suggested that when reviewing a BWR system for water hammer considerations, the reviewer look up the system in table 3-1 to determine the appropriate review considerations and and then review the topics presented in sections 3.1 through 3.11.

Table 3-2 presents similar information for PWR systems and should be used in a similar manner to that described for BWR systems.

Tables 3-3 and 3-4 (for PWR and BWR) respectively identify which operating and maintenance procedure considerations discussed in section 3.12 are applicable to each plant system. Thus, for example, table 3-3 shows that the operating and maintenance procedures for PWR ECCS should address prevention of rapid valve motion, avoidance of voids in water-filled lines and components, and proper filling and venting of water-filled lines and components.

Finally, it should be noted that the findings set forth below represent the results of over 10 years' accumulated experience, design changes, etc., and therefore should benefit new designs.

3.1 Void Detection Systems

- o A void detection system could be provided to detect voids at all high points in liquid-filled normally idle piping, where voids or steam bubbles could form through maintenance, operation, draining, out-leakage, gas evolvement, or in-leakage of steam of flashing fluid; and the potential for damaging water hammer events in safety systems exist.
- o The void detection system could provide detection at all high points, including components and portions of lines isolated from other high points by valves.
- o Void detection instrumentation can provide indication in the control room or locally. If local indication is provided, it should be demonstrated that radiation exposure to personnel will be As Low As Reasonably Achievable (ALARA).
- o Void detection systems should be provided with means to test or verify the operability of the system.
- o Void detection system displays and controls, if added to the control room, should not increase the potential for operator error. A human factors analysis should be performed, as required, taking into consideration the use of the information by an operator during both normal and abnormal plant conditions.

- Portions of piping or components that can only become voided during maintenance operations need not be continuously monitored, provided design or procedural measures have been implemented which will ensure that all voids will be eliminated prior to returning the piping or components to service.
- Operating procedures should incorporate a requirement that any void detected in the applicable systems (tables 3-1 and 3-2) shall be corrected as sooon as practical.

3.2 Keep-Full Systems

- A keep-full system should be provided as to prevent void formation in normally idle water-filled lines for the systems listed in tables 3-1 and 3-2.
- The keep-full system should be designed to operate continuously. An acceptable design is a continuously operating jockey pump or a storage tank with a water level at a higher elevation than the lines which it services.
- The keep-full system should be adequately sized to meet head and flow demands. The design of the keep-full system should account for line elevation, friction, and any pressure increases induced by valve in-leakage. The flow capacity of the keep-full system should accommodate the combined maximum leak rate of all the systems it services. Justifiable safety margins should be demonstrated in establishing head and flow demand values.
- The keep-full system should have the capability to provide detection of significant leakage in piping systems it services.
- Keep-full instrumentation should provide indication in the control room or locally. If local indication is provided, it should be demonstrated that radiation exposure to personnel will be As Low As Reasonably Achieveable (ALARA).
- The keep-full system should be provided with means to test or verify the operability of the system.

- The keep-full system displays and controls should not increase the potential for operator error. A human factors analysis should be performed, taking into consideration the use of the information by an operator during both normal and off-normal plant conditions.

3.3 Venting

- Vents should be provided to vent components and piping at all points specified in section 3.1.
- Vents in locations where voiding may occur during normal operation should be automatic, remotely operated, or designed for easy access and operator usage. Manual locally operated vents can be used for those locations where voiding can occur only during maintenance operations.
- Operating areas, where manual vents are located, shall be analyzed to assure adequate human operator environmental conditions including light, heat, access and radiation levels.
- The vent system design should provide the operator with the ability to determine the adequacy of a venting operation.
- The design and location of either remote or local controls for nonautomatic vents should be such that radiation releases will be As Low As Reasonably Achieveable (ALARA).
- The size of vent lines connected to the reactor coolant boundary should be kept smaller than the size corresponding to the definition of a LOCA (10 CFR part 50, appendix A) to avoid unnecessary challenges to the ECCS.
- Venting system displays and controls, if added to the control room, should not increase the potential for operator error. A human factors analysis should be performed, as required, taking into consideration the use of the information by an operator during both normal and abnormal plant conditions.
- Means should be provided to test or verify the operability of the vent system on a regular schedule.

3.4 Filling Safety-Related, Open-Loop Service Water Systems

One of the following should be met by safety-related, open-loop service water systems:

- o For manually started systems, voids can be filled within the required start time. This provision concerns manually started systems only.

- o Neither column separation nor voiding can occur during standby or following pump shutdown.

- o The system is designed with a startup mode that slowly fills and vents the discharge lines in such a manner as to prevent water hammer on pump startup.

- o The system is designed to maintain function following a postulated water hammer event.

3.5 HPCI Turbine Inlet Line Valve

- o The HPCI turbine inlet line inboard or outboard isolation valves should not contain a "seal in" feature on opening when the valves are in the manual mode.
- o The design of the The HPCI turbine inlet line inboard valve should permit gradual opening of the valve, as required, to permit acceptable line warmup.
- o The operating procedures for the HPCI turbine inlet line should incorporate a requirement that prohibits closing the outboard isolation valve unless the inboard isolation valve is fully closed and opening the inboard isolation valve unless the outboard isolation valve is fully open, when the valves are in manual mode (for systems in which the outboard valve is normally open).

3.6 Feedwater Control Valve and Controller

- The feedwater control valve design should be reviewed to verify that the valve design parameters, including actuator, flow coefficient (CV), and trim, are compatible with all final-designed operating conditions of the condensate and feedwater system. Specifically, the following concerns should be addressed:
  - The feedwater control valve calculated or actual pressure drop should be compared with the valve specification to verify valve compatibility for the entire operating range of the valve.
  - If low flow bypass valves are used, the feedwater bypass valve and the feedwater control valve interaction should be reviewed to ensure that the lower end of the feedwater controllable operating range is below the level at which the low-flow bypass valves are used in place of the feedwater control valves.
  - Feedwater control valve stroke versus load characteristics, minimum steady-state operating loads, valve stability lower limit ranges, and other information affecting valve stability should be reviewed to verify that instability will not occur at the least open-valve steady-state operating point.
  - Values of the minimum static pressure in the feedwater control valve, the vapor pressure of the fluid, and other information affecting flashing should be reviewed to ensure that flashing will not occur at steady-state or normal transient conditions.
  - The sizing of the feedwater control valve should be compatible with the remainder of the feedwater and condensate systems. The valve should not be excessively oversized.
  - Feedwater control valve testing or operating experience data should be reviewed to verify that valve stroke characteristics will provide stable operation.

- Feedwater control valve design should be reviewed to verify that valve trim imbalance has been considered. Minimizing imbalance generally tends to increase valve stability.

o The feedwater controller valve control design should be reviewed to verify that the design precludes rapid valve actuation motion under any planned or spuri..us signal.

o The feedwater controller valve control design should be reviewed to ensure that it has been designed to prevent excessive ocillations and hunting.

o Test, analytical, or operating experience data should be reviewed to ascertain that neither valve nor control system components will fail suddenly due to normal usage or fatigue, resulting in rapid valve motion or flow rate changes.

3.7 HPCI and RCIC Turbine Exhaust Line Vacuum Breakers

o The HPCI and RCIC turbine exhaust lines should be provided with vacuum breakers to prevent vacuum formation in any portion of the exhaust line due to steam condensation.

o The vacuum breaker design and location should be reviewed to determine that all requirements for the reactor coolant pressure boundary are met.

o The vacuum breaker design should be reviewed to determine that the design precludes the introduction of water slugs from the suppression pool and rapid check valve closure.

o Vacuum breaker sizing should be reviewed to determine that the design accounts for the effects of condensation caused by an unwarmed exhaust line and water backflow.

3.8 HPCI Turbine Line Drain Pot Level Detection

o Drain systems should be provided for HPCI turbine lines to drain all condensate from the line low spots.

- The HPCI turbine lines' configurations and slopes should be reviewed to verify that all low spots drain to the drain system and that sufficient slope is provided in the configuration to ensure complete drainage.
- HPCI line drain systems should be reviewed to verify that the drain pots have been adequately sized to handle all expected condensate.
- The HPCI turbine inlet line drain system should be reviewed to verify that the design permits testing of the drain system with the turbine inlet line isolation valves open.
- HPCI turbine line drain systems should provide high-level indication in the control room. The systems' displays and controls should not increase the potential for operator error. A human factors analysis should be performed, as required, taking into consideration the use of the information by an operator during both normal and abnormal plant conditions. Means should be provided to test the operability of the HPCI line drain systems' level indicators and bypass valves.
- The procedures for the HPCI system should incorporate a requirement that the level indicators on the HPCI drain pot system should be checked on a regular schedule for operability.

3.9 Steam Hammer

The design bases for the main steam components and pipe supports should consider steam hammer forces resulting from the most rapid anticipated closure of all system valves, including the turbine stop valves.

3.10 Relief Valve Discharge

The design basis for the components and pipe supports of the applicable systems should consider fluid forces resulting from safety and relief valve operation, including those loads from water slugs where water seals are used or the valve lines normally contain subcooled or saturated liquid.

Applicable systems:

- o BWR
  - Main steam
- o PWR
  - Main steam
  - Reactor coolant system (pressurizer)

3.11 Plant Personnel Training

- o Training in the cause, effect, and prevention of water hammer should be provided to
  - licensed and nonlicensed operating personnel
  - maintenance personnel who maintain plant fluid systems where water hammer can occur
  - personnel who directly supervise these operating and maintenance personnel.
- o The training content should be relevant to the specific plant systems
- o Training should ensure that operating information pertinent to water hammer, originating both wthin and outside the utility organization, is continually supplied to oeprators and other personnel and is incorporated into training and retraining programs.

3.12 Operating and Maintenance Procedures

Operating and maintenance procedures for systems in which water hammer can occur should take into consideration the potential for water hammer. Operating and maintenance procedures should address:

- o Prevention of rapid valve motion
- o Introduction of voids into water-filled lines and components
- o Proper filling and venting of water-filled lines and components
- o Introduction of steam or heated water that can flash into water-filled lines and components

- o Introduction of water into steam-filled lines or components
- o Proper warmup of steam-filled lines
- o Proper drainage of steam-filled lines
- o The effects of valve alignments on line conditions.

3.13 Steam Generator Water Hammer (SGWH)

The following provisions of Branch Technical Position ASB 10-2 (reference 3) should be followed.

a. "Top Feed Designs

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 seven 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.

b. Preheater Designs

1. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Also perform a water hammer test at *% of power by using feedwater through the auxiliary feedwater (top) nozzle at the lowest feedwater

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*The power level at which feedwater flow is transferred from the auxiliary feedwater nozzle to the main feedwater nozzle.

temperature that the plant standard operating procedure (SOP) allows and then switching the feedwater at that temperature from the auxiliary feedwater nozzle to the main feedwater (bottom) nozzle by following the SOP.

2. Minimize the horizontal lengths of feedwater piping between the steam generator and the vertical run of piping by providing downward turning elbows immediately upstream of the main and auxiliary feedwater nozzles.
3. Provide a check valve upstream of the auxiliary feedwater connection to the top feedwater line.
4. Maintain the top feedwater line full at all times.

c. Once-Through Designs (B&W)

1. Provide auxiliary feedwater to the steam generator through an external header.
2. 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. Provide the procedures for these tests for approval before conducting the tests."

d. NUREG 0737, paragraph II.E.1.2 (reference 15), requires automatic auxiliary feedwater system initiation. The requirements for the AFW system automatic initiation are contained in reference 15.

3.14 Control Rod Drive (CRD) Hydraulic Lines

The design basis for CRD hydraulic lines should consider the transient forces resulting from the worst case CRD system actuation.

TABLE 3-1. BWR System Water Hammer Causes and Preventive Measures.

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES (*)	
		DESIGN	PLANT OPERATION
RHR	Voiding, Steam-Bubble Collapse	Void Detection (3.1), Keep-full (3.2), Venting, (3.3)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)
HPCI	Steam Water Entrainment, Turbine Inlet Valve Operation	No Opening Seal-in in Manual Mode (3.5a), Gradual Opening (3.5b)	Valve Opening Sequence (3.5c), Operator Training (3.11), Operating Procedures (3.12)
	Steam Water Entrainment Drain Pot Malfunction	Proper Drain System Including Drain Pot Sizing and Level Verification (3.8)	Verification of Drain Pot Level (3.8), Operating Procedures (3.12)
	Turbine Exhaust Line Bubble Collapse	Exhaust Line Vacuum Vacuum Breakers	
	Pump Discharge Line Voiding	Void Detection (3.1), Keep-Full System (3.2), Venting (3.3)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)
Core Spray	Voiding Steam-Bubble Collapse	Void Detection (3.1), Keep-Full System (3.2), Venting (3.3)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)
Essential Service Water	Voiding Column Separation	Void Detection (3.1), Keep-Full System (3.2), Venting (3.3), Open Loop Line Analysis (3.4)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)

(*)Refers to section of this report providing details of preventive measures.

TABLE 3-1. Continued

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES	
		DESIGN	PLANT OPERATION
Main Steam	Steam Hammer Relief Valve Discharge	Valve Closure (3.9) and Relief Valve Discharge Loads (3.10)	
	Steam Water Entrainment		Operating Procedures (3.12), Operator Training (3.11)
Feedwater	Feedwater Control Valve Instability	Feedwater Controller Design Verification 3.6a, b, and c	
RCIC	Exhaust Line Steam Bubble Collapse	Exhaust Line Vacuum Breakers (3.7)	
Isolation Condenser	High Reactor Water Level		Operating Procedures (3.12), Operator Training (3.11)
**Control Rod Drive	Rapid Valve motion	Actuation Loads (3.14)	

**Control Rod Drive events have not been reported, but have been anlytically postulated.

TABLE 3-2. PWR System Water Hammer Causes and Preventive Measures.

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES (*)	
		DESIGN	PLANT OPERATION
Feed-water	Feedwater Control Valve (FCV) Over-sizing & Instability	FCV Design Veri-fication (3.6)	
	Unknown and Operator Error Induced Steam Bubble Collapse		Operating Procedures (3.12), Operator Training (3.11)
Main Steam	Steam Hammer (Valve Closure)	Include Valve Closure Loads in Pipe Support and Component Design Basis (3.9)	
	Relief Valve Discharge	Include Relief Valve Discharge Loads in Pipe Sup-port and Components Design Basis (3.10)	
	Steam Water Entrain-ment, Unknown		Operating Procedures (3.12), Operator Train-ing (3.11)
Reactor Coolant (Pres-surizer)	Relief Valve Discharge	Include Relief Valve Discharge Loads in Pipe Support and Components Design Basis (3.10)	
RHR	Voiding	Venting (3.3)	Operating Procedures (3.12), Operator Training (3.11)
ECCS	Voiding	Venting (3.3), Void Detection(3.1)	Operating Procedures (3.12), Operator Training (3.11)
CVCS	Steam Bubble Col-lapse or Vibration		Operating Procedures (3.12), Operator Training (3.11)

TABLE 3-2. Continued

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES (*)	
		DESIGN	PLANT OPERATION
Essential Cooling Water	Voiding	Venting (3.1), Filling Essential Cooling Water (3.4), Analysis (3.4)	Filling Essential Cooling Water (3.4), Operating Procedures (3.12), Operator Training (3.11)
Steam Generator	Line Voiding Followed by Steam Bubble Collapse	BTP ASB 10-2 Provisions (3.13): Top Discharge, Short Line Lengths, External Header (B&W Only)	BTP ASB 10-2 Provisions (3.13): Testing, Keeping Line Full. Automatic AFW Initiation

(*)Refers to section of this report providing details of preventive measures.

TABLE 3-3

PWR Operating and Maintenance Procedure Water Hammer Considerations

System	Water Hammer Consideration	Rapid Valve Motion	Intro- duction of Voids	Filling and Venting	Steam or Hot Water in Water-Filled Lines & Components	Water Into Steam Lines	Steam Line Warmup	Steam Line Drainage	Valve Alignment
Feedwater		X			X				X
Main Steam						X	X	X	X
RHR		X	X	X					X
ECCS		X	X	X					X
CVCS		X	X	X	X				X
Cooling Water		X	X	X					X

TABLE 3-4

BWR Operating and Maintenance Procedure Water Hammer Considerations

System	Water Hammer Consideration							
	Rapid Valve Motion	Introduction of Voids	Filling and Venting	Steam or Hot Water in Water-Filled Lines & Components	Water Into Steam Lines	Steam Line Warmup	Steam Line Drainage	Valve Alignment
Core Spray		X	X					X
RHR	X	X	X	X	X	X		X
Isol. Cond.					X			
HPCI		X	X	X	X	X	X	X
RCIC		X	X	X	X	X	X	X
Main Steam					X	X	X	X
Feedwater	X			X				X
Cooling Water	X	X	X					X

4.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Water Hammer in Nuclear Power Plants," NUREG-0582, July 1979. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

2. J. A. Block, et al. "An Evaluation of PWR Steam Generator Water Hammer." NUREG-0291, Creare, Incorporated, for U.S. Nuclear Regulatory Commission. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

3. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC Report NUREG-0800, July 1981, Branch Technical Position ASB 10-2, attached to section 10.4.7. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

4. R. L. Williamson, "An Analysis Tool for Predicting the Transient Hydrodynamics Resulting from the Rapid Filling of Voided Piping Systems," EG&G Interim Report RE-E-79-009, February 1979 (prepared for NRC Internal Use).

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9. R. A. Uffer, et al., "Evaluation of Water Hammer Events in Light Water Reactor Plants," NUREG/CR-2781, QUAD-1-82-018, Quadrex Corporation, June 1982. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

10. J. T. Han, and N. Anderson, "Prevention and Mitigation of Steam Generator Water Hammer in PWR Plants," NUREG-0918, U.S. Nuclear Regulatory Commission, November 1982. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

11. P. Saha, et al., "An Evaluation of Condensation-Induced Water Hammer in Preheat Steam Generators," NUREG/CR-1606, Brookhaven National Laboratory, September 1980. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

12. "Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code, section III. Available from American Society of Mechanical Engineers, New York, New York.

13. "American National Standard Code for Pressure Piping, Power Piping," ANSI/ASME B31.1, 1980 edition. Available from American Society of Mechanical Engineers, New York, New York.

14. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC Report NUREG-0800, July 1981, section 3.9.3 "ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures." Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

15. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980, paragraphs II.D.1 and II.E.1.2. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

16. D. E. Sexton, M. Kasahara and R. A. Uffer, "Evaluation of Water Hammer Potential in Preheat Steam Generators," NUREG/CR-3090, QUAD-1-82-243, Quadrex Corporation, December 1983. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

17. Letter to A. W. Serkiz, U.S. NRC, from F. R. Seddiqui, Reactor Controls, Inc. dated July 29, 1983.

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
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Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

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The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

APPENDIX A

STANDARD REVIEW PLAN SECTION 3.9.3
STRESS LIMITS FOR ASME CLASS 1, 2, AND 3 COMPONENTS AND COMPONENT SUPPORTS OF SAFETY-RELATED SYSTEMS AND CLASS CS CORE SUPPORT STRUCTURES UNDER SPECIFIED SERVICE LOADING COMBINATIONS

A. INTRODUCTION

Nuclear power plant components and supports are subjected to combinations of loadings derived from plant and system operating conditions, natural phenomena, postulated plant events, and site-related hazards. Section III, Division 1 of the ASME Code (hereafter referred to as the Code) provides specific sets of design and service stress limits that apply to the pressure retaining or structural integrity of components and supports when subjected to these loadings. The design and service stress limits specified by the Code do not assure, in themselves, the operability of components, including their supports, to perform the mechanical motion required to fulfill the component's safety function. Certain of the service stress limits specified by the Code (i.e., level C and D) may not assure the functional capability of components, including their supports, to deliver rated flow and retain dimensional stability. Since the combination of loadings, the selection of the applicable design and service stress limits appropriate to each load combination and the proper consideration of operability is beyond the scope of the Code; and the treatment of functional capability, including collapse and deflection limits, is not adequately treated by the Code for all situations, such factors must be evaluated by designers and appropriate information developed for inclusion in the Design Specification or other referenced documents.

Applicants require guidance with regard to the selection of acceptable design and service stress limits associated with various loadings and combinations thereof, resulting from plant and system operating conditions and design basis events, natural phenomena, and site-related hazards. The relationship and application of the terms "design conditions," "plant operating conditions," "system operating conditions," and the formerly used term "component operating conditions," now characterized by four levels of service stress limits, have not been clearly understood by applicants and their subcontractors.

For example, under the "faulted plant or system condition" (e.g., due to LOCA within the reactor coolant pressure boundary), the emergency core cooling system (ECCS) should be designed to operate and deliver rated flow for an extended period of time to assure the safe shutdown of the plant. Although the "plant condition" is termed "faulted," components in the functional ECCS must perform the safety function under a specified set of service loadings which includes those resulting from the specified plant postulated events. The selection of level "D" (related to the "faulted" condition) service stress limits for this system, based solely on the supposition that all components may use this limit for a postulated event resulting in the faulted plant condition cannot be justified, unless system operability is also demonstrated.

This appendix is necessary to improve consistency and understanding of the basic approach in the selection of load combinations applicable to safety-related systems and to establish acceptable relationships between plant postulated events, plant and system operating conditions, component and component support design, and service stress limits, functional capability, and operability.

B. DISCUSSION

Current reviews of both standardized plants and custom plants have indicated the need for additional guidance to reach acceptable design conclusions in the following areas:

(1) Relationship between certain plant postulated events, plant and system operating conditions, resulting loads and combinations thereof, and appropriate design and service stress limits for ASME Class 1, 2 and 3 components and component supports and Class CS core support structures.

(2) Relationship of component operability assurance, functional capability, and allowable design and service stress limits for ASME Class 1, 2, and 3 components and component supports.

The Code provides five categories of limits applicable to design and service loadings (design, level A, level B, level C, and level D). The Code rules provide for structural integrity of the pressure retaining boundary of a component and its supports, but specifically exclude the subject of component operability and do not directly address functional capability. The types of loadings to be taken into account in designing a component are specified in the Code, but rules specifying how the loadings, which result from postulated events and plant and system operating conditions, are to be combined and what stress level is appropriate for use with loading combinations are not specified in the Code. It is the responsibility of the designer to include all this information in the Code required Design Specification of each component and support.

C. POSITION

Effective with the 1977 Edition, the Code provides design stress limits and four sets of service stress limits for all classes of components, component supports, and core support structures. The availability of such design and service stress limits within the Code requires that the MEB review and determine maximum acceptable design and service stress limits which may be used with specified loads, or combinations thereof, for components and component supports of safety-related systems (refer to definition in Table III) and core support structures.

This appendix provides guidance for dealing with the components and component supports of safety-related systems and core support structures in the following areas:

(1) Consideration of design loadings and limits.

(2) Consideration of service loading combinations resulting from postulated events and the designation of acceptable service limits.

(3) Consideration of piping functional capability and operability of active pumps and valves under service loading combinations resulting from postulated events.

(4) Applicability of the appendix to components, component support structures, and core support structures and procedures for compliance.

1.0 ASME CLASS 1, 2, AND 3 COMPONENTS AND COMPONENT SUPPORTS OF SAFETY-RELATED SYSTEMS AND CLASS CS CORE SUPPORT STRUCTURES

1.1 Design Considerations and Design Loadings

ASME Code Class 1, 2, and 3 components, component supports, and class CS core support structures shall be designed to satisfy the appropriate subsections of the Code in all respects, including limitations on pressure, and the requirements of this appendix. Component supports that are intended to restrain either force and displacement or anchor movement shall be designed to maintain deformations within appropriate limits as specified in the component support Design Specifications.

Design loadings shall be established in the Design Specification. The design limits of the appropriate subsection of the Code shall not be exceeded for the design loadings specified.

1.2 Service Loading Combinations

The identification of individual loads and the appropriate combination of these loads (i.e., sustained loads, loads due to system operating transients SOT, OBE, SSE, LOCA, DBPB, MS/FWPB and their dynamic effects) shall be in accordance with Section 1.3. The appropriate method of combination of these loads shall be in accordance with NUREG-0484, "Methodology for Combining Dynamic Loads" (Reference 9).

1.3 Service Conditions

1.3.1 Service Limit A

Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service limit not greater than Level A when subjected to sustained loads resulting from normal plant/system operation.

1.3.2 Service Limit B

Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service limit not greater than Level B when subjected to the appropriate combination of loadings resulting from (1) sustained loads, (2) specified plant/system operating transients (SOT), and (3) the OBE.

1.3.3 Service Limit C

(a) Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service limit not greater than Level

C when subjected to the appropriate combination of loadings resulting from (1) sustained loads, and (2) the DBPB.

(b) The DBPB includes loads from the postulated pipe break, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

1.3.4 Service Limit D

(a) Class 1, 2, and 3 components, component supports, and Class CS core support structures shall meet a service stress limit not greater than Level D when subjected to the appropriate combination of loadings resulting from (1) sustained loads, (2) either the DBPB, MS/FWPB, or LOCA, and (3) and SSE.

(b) The DBPB, MS/FWPB, and LOCA include loads from the postulated pipe breaks, themselves, and also any associated system transients or dynamic effects resulting from the postulated pipe breaks. Asymmetric blowdown loads on PWR primary systems shall be incorporated per NUREG-0609 (Reference 10).

2.0 OPERABILITY AND FUNCTIONAL CAPABILITY

2.1 Active Pumps and Valves

SRP Section 3.10 (Reference 4) shall demonstrate that the pump or valve, as supported, can adequately sustain the designated combined service loadings at a stress level at least equal to the specified service limit, and can perform its safety function without impairment. Loads produced by the restraint of free end displacement and anchor point motions shall be included.

2.2 Snubbers

The operability requirements specified for mechanical and hydraulic snubbers installed on safety-related systems is subject to review by the staff. When snubbers are used, their need shall be clearly established and their design criteria presented.

2.3 Functional Capability

The design of Class 1, 2, and 3 piping components shall include a functional capability assurance program. This program shall demonstrate that the piping components, as supported, can retain sufficient dimensional stability at service conditions so as not to impair the system's functional capability. The program may be based on tests, analysis, or a combination of tests and analysis.

3.0 TABLES

3.1 Table I summarizes the requirements of this appendix for use with ASME Class 1, 2, and 3 components, component supports, and Class CS core support structures. The table illustrates plant events, system operating conditions, service loading combinations, and service stress limits and should always be used in conjunction with the text of this appendix.

3.2 Table II defines all the terms used in this appendix.

4.0 PROCEDURES FOR COMPLIANCE

4.1 Design Specification and Safety Analysis Report

(a) The design options provided by the Code and related design criteria specified in the Code required Design Specification for ASME Class 1, 2, and 3 components, component supports, and Class CS core support structures should be summarized in sufficient detail in the Safety Analysis Report of the application to permit comparison with this Appendix.

(b) The presentation in the PSAR should specify and account for all design and service loadings, method of combination, the designation of the appropriate design and service stress limits (including primary and secondary stresses, fatigue consideration, and special limits on pressure when appropriate) for each loading combination presented, and the provisions for functional capability.

(c) The presentation in the FSAR should indicate how the criteria in Sections 1 and 2 of this appendix have been implemented.

(d) The staff may request the submission of the Code required Design Documents such as Design Specifications, Design Reports, Load Capacity Data Sheets, or other related material or portions thereof to establish that the design criteria, the analytical methods, and functional capability satisfy the guidance provided by this appendix. This may include information provided to, and received from, component and support manufacturers. As an alternative to the applicant submitting these documents, the staff may require them to be made available for review at the applicant's or vendor's office.

4.2 Use with Regulatory Guides

The information and requirements contained in this appendix supersede those in the October 1973 version of Regulatory Guide 1.67 and the May 1973 version of Regulatory Guide 1.48. Regulatory Guides 1.124 and 1.130 on Class 1 linear and Class 1 plate and shell component support structures are to be supplemented by this appendix.

TABLE I

Allowable Service Stress Limits for Specified Service Loading Combinations for ASME Class 1 Components and Class CS Support Structures

Plant Event[2]	System Operating Conditions	Service Loading Combination[1,4]	Service Stress Limit
1. Normal Operation	Normal	Sustained Loads	A
2. Plant/System Operating Transients (SOT) + OBE	Upset	Sustained Loads + SOT + OBE	B[3]
3. DBPB	Emergency	Sustained Loads + DBPB	C[3]
4 MS/FWPB	Faulted	Sustained Loads + MS/FWPB	D[3]
5. DBPB or MS/FWPB + SSE	Faulted	Sustained Loads + DBPB or MS/FWPB + SSE	D[3]
6. LOCA	Faulted	Sustained Loads + LOCA	D[3]
7. LOCA + SSE	Faulted	Sustained Loads + LOCA + SSE	D[3]

NOTE: [1]The appropriate method of combination is subject to review and evaluation. Refer to Section 1.2.

[2]Refer to Table II for definition of terms.

[3]In addition to meeting the specified service stress limits for given load combinations, operability and functional capability must also be demonstrated as discussed in Subsection 2.0 of this appendix and in SRP Section 3.10.

[4]These events must be considered in the pipe stress analysis and pipe support design process when specified in the ASME Code-required Design Specification. The Design Specification shall define the load and specify the applicable Code Service Stress Limit. For clarification, it should be noted that the potential for water hammer and water (steam) hammer occurrence should also be given proper consideration in the development of Design Specifications.

TABLE II

DEFINITION OF TERMS

Active Pumps and Valves - A pump or valve which must perform a mechanical motion in order to shut down the plant or mitigate the consequences of a postulated event. Safety and relief valves are specifically included

Component and Support Functional Capability - Ability of a component, including its supports, to deliver rated flow and retain dimensional stability when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

Component and Support Operability - Ability of an active component, including its support, to perform the mechanical motion required to fulfill its designated safety function when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

DBPB - Design Basis Pipe Breaks - Those postulated pipe breaks other than a LOCA or MS/FWPB. This includes postulated pipe breaks in Class 1 branch lines that result in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system.

This condition includes loads from the postulated pipe breaks, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

Design Limits - The limits for the design loadings provided in the appropriate subsection of Section III, Division 1, of the ASME Code.

Design Loads - Those pressures, temperatures, and mechanical loads selected as the basis for the design of a component.

Functional System - That configuration of components which, irrespective of ASME Code Class designation or combination of ASME Code Class designations, performs a particular function (i.e., each emergency core cooling system performs a single particular function and yet each may be comprised of some components which are ASME Class 1 and other components which are ASME Code Class 2).

LOCA - Loss-of-Coolant Accidents - Defined in Appendix A of 10 CFR Part 50 as "those postulated accidents that result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system."

This condition includes the loads from the postulated pipe break, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

<u>MS/FWPB - Main Steam and Feedwater Pipe Breaks</u> - Postulated breaks in the main steam and feedwater lines. For a BWR plant this may be considered as a LOCA event depending on the break location.

This condition includes the loads from the postulated pipe break, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break.

<u>OBE - Operating Basis Earthquake</u> - Defined in Section III (d) of Appendix A of 10 CFR Part 100 as "that earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant, necessary for continued operation without undue risk to the health and safety of the public, are designed to remain functional."

This condition includes the loads from the postulated seismic event, itself, and also any associated system transients or dynamic effects resulting from the postulated seismic event.

<u>Piping Components</u> - These items of a piping system such as tees, elbows, bends, pipe and tubing, and branch connections constructed in accordance with the rules of Section III of the ASME Code.

<u>Postulated Events</u> - Those postulated natural phenomena (i.e., OBE, SSE), postulated site hazards (i.e., nearby explosion), or postulated plant events (i.e., DBPB, LOCA, MS/FWPB) for which the plant is designed to survive without undue risk to the health and safety of the public. Such postulated events may also be referred to as design basis events.

<u>SSE - Safe Shutdown Earthquake</u> - Defined in Section III(c) of Appendix A of 10 CFR Part 100 as "that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is the earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to assure:

(1) The integrity of the reactor coolant pressure boundary.

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline."

This condition includes the loads from the postulated seismic event, itself, and also any associated system transients or dynamic effects resulting from the postulated seismic event.

Service Limits - The four limits for the service loading as provided in the appropriate subsection of Section III, Division 1, of the ASME Code.

Service Loads - Those pressure, temperature, and mechanical loads provided in the Design Specification.

SOT - System Operating Transients - The transients and their resulting mechanical responses due to dynamic occurrences caused by plant or system operation.

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The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

If other types of CRDS are proposed or if new features that are not specifically mentioned here are incorporated in CRDS of current types, information should be supplied for the new systems or new features similar to that described below.

1. The descriptive information, including design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives, is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly.

2. A review is performed of information pertaining to design codes, standards, specifications, and standard practices, as well as to General Design Criteria, regulatory guides, and branch positions that are applied in the design, fabrication, construction, and operation of the CRDS.

   The various criteria, described in general terms above, should be supplied along with the names of the apparatus to which they apply. Pressurized portions of the system which are a part of RCPB are reviewed to determine the extent to which the applicant complies with the Class 1 requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the Code"). Those portions which are not part of the RCPB are reviewed with other specified parts of Section III, or other sections of the Code. The MEB reviews the non-pressurized portions of the control rod drive system to determine the acceptability of design margins for allowable values of stress, deformation, and fatigue used in the analyses. If an experimental testing program is used in lieu of analysis, the program is reviewed to determine whether it adequately covers the areas of concern in stress, deformation, and fatigue.

3. Information is reviewed which pertains to the applicable design loads and their appropriate combinations, to the corresponding design stress limits, and to the corresponding allowable deformations. The deformations are of interest in the present context only in those instances where a failure of movement could be postulated due to excessive deformation and such movement would be necessary for a safety-related function.

   If the applicant selects an experimental testing option in lieu of establishing a set of stress and deformation allowables, a detailed description of the testing program must be provided for review.
   In the preliminary safety analysis report (PSAR), the load combinations, design stress limits and allowable deformations criteria should be provided for review.

   In the final safety analysis report (FSAR), the actual design should be compared with the design criteria and limits to demonstrate that the criteria and limits have not been exceeded.

   Loadings imposed during normal plant operation and startup and shutdown transients include but are not limited to pressure, deadweight, temperature effects, and anticipated operational occurrences. Loadings associated with specific seismic and other dynamic events are then combined with the above plant-type loads. For BWRs only, the CRDS is reviewed to verify that the system is capable of withstanding adverse dynamic loads such as water hammer. The response to each set of combined loads has a selected stress or deformation limit. The selection of a specific limit is influenced by the probability of the postulated event occurring and the need to assure operation during and after the event.

4. The portion of the SAR is reviewed that describes plans for the conduct of an operability assurance program or that references previous test programs or standard industry procedures for similar apparatus. For example, the life cycle test program for the CRDS is reviewed. The operability assurance program is reviewed to ascertain coverage of the following:

   a. Life cycle test program.

   b. Proper service environment imposed during test, including appropriate anticipated normal operational occurrences, seismic, and postulated accident conditions.

   c. Mechanism functional tests.

   d. Program results.

In addition, the MEB will coordinate other branches' evaluations that interface with the overall review of the CRDS as follows:

The Core Performance Branch (CPB) will verify fuel system design, including effects of the CRDS on fuel behavior in meeting the requirements of the reactor core design under various normal and accident operating conditions in SRP Section 4.2. The Materials Engineering Branch (MTEB) will review the material aspects of CRDS in SRP Section 4.5.1.

For those areas of review identified above as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

MEB acceptance criteria are based on meeting the requirements of the following regulations:

1. GDC 1 and and 10 CFR Part 50,§50.55a, as its relates to CRDS, requires that the CRDS be designed to quality standard commensurate with the importance of the safety functions to be performed.

2. GDC 2, as it relates to CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

3. GDC 14, as it relates to CRDS, requires that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

4. GDC 26, as it relates to CRDS, requires that the CRDS be one of the independent reactivity control systems which is designed with appropriate

margin to assure its reactivity control function under anticipated normal operation condition.

5. GDC 27, as it relates to CRDS, requires that the CRDS be designed with appropriate margin, and in conjunction with the emergency core cooling system, be capable of controlling reactivity and cooling the core under postulated accident conditions.

6. GDC 29, as its relates to CRDS, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.

Specific criteria necessary to meet the relevant requirements of the regulations identified above are as follows:

1. The descriptive information is determined to be sufficient provided the minimum requirements for such information meet Section 3.9.4 of Reference 11.

2. Construction (as defined in NCA-1110 of Section III of the ASME Code, Reference 7) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:

   a. Pressurized Portions of Equipment Classified as Quality Group A, B, C (Regulatory Guide 1.26)

      Section III of the ASME Code, Class 1, 2, or 3 as appropriate (Ref. 7).

   b. Pressurized Portions of Equipment Classified as Quality Group D (Regulatory Guide 1.26)

      (1) Section VIII, Division 1 of the ASME Code for vessels and pump casings (Ref. 7).

      (2) Applicable to Piping Systems (American National Standards Institute, ANSI):[1]

B16.5	Steel Pipe Flanges and Flanged Fittings (Ref. 13).
B16.9	Steel Butt Welding Fittings (Ref. 14).
B16.11	Steel Socket Welding Fittings (Ref. 15).
B16.25	Butt Welding Ends (Ref. 16).
B31.1	Piping (Ref. 17).
SP-25	Standards (Ref. 18).
B16.34	Valves (Ref. 19).

   c. Nonpressurized Equipment (Non-ASME Code)

      Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than those for other plants of

[1]This list can be extended by a staff review and acceptance of other ANSI and MSS standards in the piping system area.

similar design having a period of successful operation. Justification of any decreases should be provided.

3. For the various design and service conditions defined in NB-3113 of Section III of the ASME Code (Ref. 7), load combination sets are as given in Standard Review Plan Section 3.9.3 (Ref. 12). The stress limits applicable to pressurized and nonpressurized portions of the control rod drive systems should be as given in Reference 12 for the response to each loading set. The CRDS design should adequately consider water hammer loads to assure that system safety functions can be achieved.

4. The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and overcoming a stuck rod meet system design requirements.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

1. The objectives of the review are to determine that design, fabrication, and construction of the control rod drive mechanisms provide structural adequacy and that suitable life cycle testing programs have been utilized to prove operability under service conditions.

   In the construction permit (CP) review, it should be determined that the design criteria utilize proper load combinations, stress and deformation limits, and that operability assurance is provided by reference to a previously accepted testing program or that a commitment is made to perform a testing program which includes the essential elements listed below. In the operating license (OL) review, the results of any testing program not previously reviewed should be evaluated.

2. The design criteria presented should be evaluated for both the internal pressure-containing portions and other portions of the CRDS. These include the CRDM housing, hydraulic control unit, condensate supply system and scram discharge volume, and portions such as the cylinder, tube, piston, and collect assembly.

   Of particular concern are any new and unique features which have not been used in the past. Pressure-containing components are checked to ensure that they meet the design requirements of the codes and criteria which have been accepted by the Mechanical Engineering Branch, and are identified in Standard Review Plan Section 3.2.2. The review of the functional design of reactivity control systems, including control rod drive systems, is the responsibility of the Reactor Systems Branch (RSB) (see SRP Section 4.6). The loading combinations for the various plant operating conditions are checked for consistency with Reference 12; given these loading combinations, the stress limits of the appropriate code should not be exceeded, or the limits in Reference 12 should not be exceeded if not specified in the listed design code. Exceptions taken by the applicant to any of the accepted codes, standards, or NRC criteria must be identified and the basis clearly justified so that evaluation is possible. Engineering judgment, experience, comparisons with earlier cases and design margins, and consultation with supervisors permit the reviewer to reach a decision on the acceptability of any exceptions posed by the applicant.

The choice of structural materials of construction for the CRDS is reviewed by the MTEB in SRP Section 4.5.1.

3. Loading combinations are defined as those loadings associated with plant operations which are expected to occur one or more times during the lifetime of the plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power, combined with loadings caused by natural or accident events including, for BWRs, water hammer loads. The load combinations which are postulated to occur are specified for each of the design and service conditions as defined in Paragraph NB-3113 of the ASME Code (Ref. 7). These load combinations are defined in Reference 12 and are compared by the reviewer with those provided by the applicant.

   The design stress limits, including fatigue limits, and deformation limits as appropriate to the components of the control rod drive mechanism are compared by the reviewer with those of specified codes, previously designed and successfully operating systems, or with the results of scale model and prototype testing programs.

4. The control rod drive mechanisms of a new design or configuration should be subjected to a life cycle test program to determine the ability of the drives to function during and after normal operating occurrence, seismic, and postulated accident condition over the full range of temperatures, pressures, loadings, and misalignment expected in service. The tests should include functional tests to determine times of rod insertion and withdrawal, latching operation, scram operation and time, system valve operation and scram accumulator leakage for hydraulic CRDS, ability to overcome a stuck rod condition, and wear. Rod travel and number of trips expected during the mechanism operational life should be duplicated in the tests.

   The reviewer checks the elements of the test program to be sure all required parameters have been included and finally reviews the test results to determine acceptability. Excessive wear, malfunction of components, operating times beyond determined limits, scram accumulator leakage, etc., all would be cause for retesting.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this SRP section and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the design of the control rod drive system is acceptable and meets the requirements of General Design Criteria 1, 2, 14, 26, 27, and 29, and 10 CFR Part 50, §50.55a. This conclusion is based on the following:

1. The applicant has met the requirement of GDC 1 and 10 CFR Part 50, §50.55a, with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the control rod drive system are in conformance with the requirements of appropriate ANSI and ASME Codes.

2. The applicant has met the requirements of GDC 2, 14, and 26 with respect to designing the control rod drive system to withstand effects of earthquakes and anticipated normal operation occurrences with adequate margins to assure its reactivity control function and with extremely low probability of leakage or gross rupture of reactor coolant pressure boundary. The CRDS design capabilities include the ability to accommodate water hammer dynamic loads resulting from rapid opening of the scram insert and withdraw valves and closure of the hydraulic buffer under the worst case loading condition without compromising the safety functions of the system. The specified design transients, design and service loadings, combination of loads, and limiting the stresses and deformations under such loading combinations are in conformance with the requirements of appropriate ANSI and ASME Codes and acceptable regulatory positions specified in SRP Section 3.9.3.

3. The applicant has met the requirements of GDC 27 and 29 with respect to designing the control rod drive system to assure its capability of controlling reactivity and cooling the reactor core with appropriate margin, in conjunction with either the emergency core cooling system or the reactor protection system. The operability assurance program is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and implementation of acceptance criterion associated with water hammer loads in BWRs, subsection II.3, is as follows.

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

4. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

5. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

6. 10 CFR Part 50, Appendix A, General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

7. ASME Boiler and Pressure Vessel Code, Sections III and VIII, American Society of Mechanical Engineers.

8. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

9. Regulatory Guide 1.29, "Seismic Design Classification."

10. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

11. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

12. Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

13. ANSI B 16.5, "Steel Pipe Flanges and Flanged Fittings," American National Standard Institute.

14. ANSI B 16.9, "Wrought Steel Butt Welding Fittings," American National Standard Institute.

15. ANSI B 16.11, "Steel Fittings Steel Welding and Threaded," American National Standard Institute.

16. ANSI B 16.25, "Butt Welding Ends - Pipe, Valves, Flanges, and Fittings," American National Standard Institute.

17. ANSI B 31.1, "Power Piping," American National Standard Institute.

18. MSS-SP-25, "Marking for Valves, Fittings, Flanges, and Unions," Manufacturers Standardization Society.

19. ANSI B 16.34, "Steel Valves with Flanged and Butt Welding Ends," American Society of Mechanical Engineers.

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

Section No. 5.4.7
Revision No. 3

Appendix No. N/A
Revision No. N/A

Branch Tech. Position RSB 5-1
Revision No. 2

Date Issued April 1984

FILING INSTRUCTIONS

PAGES TO BE REMOVED			NEW PAGES TO BE INSERTED		
PAGE NUMBER		DATE	PAGE NUMBER		DATE
5.4.7-1 thru 5.4.7-11	Rev. 2	July 1981	5.4.7-1 thru 5.4.7-12	Rev. 3	April 1984
BTP RSB 5-1 to SRP Section 5.4.7	Rev. 2	July 1981	BTP RSB 5-1 to SRP Section 5.4.7	Rev. 2	July 1981 (No change except for page numbers)
5.4.7-12 thru 5.4.7-18			5.4.7-13 thru 5.4.7-19		

The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

5.4.7 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

The residual heat removal (RHR) system is used in conjunction with the main steam and feedwater systems (main condenser), or the reactor core isolation cooling (RCIC) system in conjunction with the safety/relief valves in a boiling water reactor (BWR), or auxiliary feedwater sytem in conjunction with the atmospheric dump valves in a pressurized water reactor (PWR) to cool down the reactor coolant system following shutdown. Parts of the RHR system also act to provide low pressure emergency core cooling and are reviewed as described in SRP Section 6.3. Some parts of the RHR system also provide containment heat removal capability and are reviewed as described in SRP Section 6.2.2. The review by RSB is to ensure that the design of the RHR system is in conformance with General Design Criteria 2, 4, 5, 19, and 34.

Both PWRs and BWRs have RHR systems which provide long-term cooling once the reactor coolant temperature has been decreased by the main condenser, RCIC, or auxiliary feedwater systems. In both types of plants, the RHR is typically a low pressure system which takes over the shutdown cooling function when the reactor coolant system (RCS) temperature is reduced to about 300°F. Although the RHR system function is similar for the two types of plants, the system design are different.

The RHR system in PWRs takes water from the RCS hot legs, cools it, and pumps it back to the cold legs or core flooding tank nozzles. The suction and discharge lines for the RHR pumps have appropriate valving to assure that the low pressure RHR system is always isolated from the RCS when the reactor coolant pressure is greater than the RHR system design pressure. The heat removed in the heat exchangers is transported to the ultimate heat sink by the component cooling water or service water system. In PWRs, the RHR system is also used to fill, drain, and remove heat from the refueling canal during refueling operations, to circulate coolant through the core during plant startup prior to RCS pump operation, and in some to provide an auxiliary pressurizer spray.

The RHR system in BWRs is typically composed of four subsystems. The containment heat removal and low pressure emergency core cooling subsystems are discussed in SRP Sections 6.2.2 and 6.3. The shutdown cooling and steam condensing (via RCIC) subsystems are covered by this SRP section. These subsystems make use of the same hardware, consisting of pumps, piping, heat exchangers, valves, monitors, and controls. In the shutdown cooling mode, the BWR RHR system can also be used to supplement spent fuel pool cooling. As in the PWR, the low pressure RHR piping is protected from high RCS pressure by isolation valves.

The steam condensing mode of RCIC operation in BWRs (when included in the plant design) provides an alternative to the main condenser or normal RCIC mode of operation during the initial cooldown. Steam from the reactor is transferred to the RHR heat exchangers where it is condensed. The condensate is piped to the suction side of the RCIC pump. The RCIC pump returns the condensate to the reactor vessel. The heat removed in the heat exchangers is transported to the ultimate heat sink by the service water system.

Other means of removing decay heat in the event that the RHR system is inoperable have been proposed for some BWRs. These approaches use some of the piping that is used for the steam condensing mode of RCIC. These approaches are also covered by this SRP section.

The reactor coolant temperatures and pressure must be decreased before the low pressure RHR system can be placed in operation; therefore, the review of the decay heat removal function must consider all conditions from shutdown at normal reactor operating pressure and temperature to the cold depressurized condition. RSB reviews the requirements for reliability and capability of removing decay heat identified in NUREG-0660 (II.E.3.2 and II.E.3.3), NUREG-0718 (II.B.7), and NUREG-0737 (III.D.1.1). With respect to the staff review for compliance with Branch Technical Position RSB 5-1 (Ref. 5), the Auxiliary Systems Branch (ASB), Chemical Engineering Branch (CMEB), and RSB effort is divided as follows:

1. For BWRs, the RSB reviews the processes and systems used in the cooldown of the reactor for the entire spectrum of potential reactor coolant system pressures and temperatures during decay heat removal.

2. For PWRs, the RSB reviews the approach used to meet the functional requirements of BTP RSB 5-1 with respect to cooldown to the conditions permitting operation of the RHR system. Since an alternate approach to that normally used for cooldown may be specified, the reviewers identify all components and systems used. The CMEB has primary review responsibility for the review of the pertinent portions of the CVCS (SRP Section 9.3.4). The ASB, as part of its primary review responsibility for SRP Sections 10.3 and 10.4.9 reviews the atmospheric dump valves and the source for auxiliary feedwater, respectively, for conformance to BTP RSB 5-1. The RSB reviews the pressurizer relief valve and ECCS, if used. In addition, the RSB reviews the tests and supporting analysis concerning

mixing of borated water and cooldown under natural circulation as required in BTP RSB 5-1.

3. For both PWRs and BWRs, the ASB reviews the component cooling or service water systems that transfer decay heat from the RHR system to the ultimate heat sink as part of its primary review responsibility for SRP Sections 9.2.1 and 9.2.2

4. The RSB reviews the design and operating characteristics of the RHR system with respect to its shutdown and long-term cooling function. Where the RHR system interfaces with other systems (e.g., RCIC system, component cooling water system) the effect of these systems on the RHR system is reviewed. Overpressure protection provided by the valving between the RCS and RHR system is also reviewed.

In addition, the Reactor Systems Branch will coordinate evaluations of other branches that interface with the overall review of the RHR system as follows: The Containment Systems Branch verifies that portions of the RHR system penetrating the containment barrier are designed with acceptable isolation features to maintain containment integrity for all operating conditions including accidents as part of its primary review responsibility for SRP Section 6.2.4; The Structural and Geotechnical Engineering Branch (SGEB) determines the acceptability of the design analysis, procedures and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles as part of its primary review responsibility for SRP Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 thru 3.7.4, 3.8.4 and 3.8.5. The Materials Engineering Branch (MTEB) verifies that inservice inspection requirements are met for system components as part of its primary review responsibility for SRP Section 6.6 and, upon request, verifies the compatibility of the materials of construction with service conditions as part of its primary review responsibility for SRP Section 6.1. The Mechanical Engineering Branch (MEB) determines that the components, piping and structures are designed and tested in accordance with applicable codes and standards as part of its primary review responsibility for SRP Sections 3.9.1 through 3.9.3. The MEB also determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2 The effects of pipe breaks inside and outside of containment, such as pipe whip and jet impingement, are reviewed by MEB and ASB as part of their primary review responsibilities for SRP Sections 3.6.2 and 3.6.1, respectively. The MEB also reviews adequacy of the inservice testing program of pumps and valves as part of its primary review responsibility for SRP Section 3.9.6. The Procedures and Systems Review Branch (PSRB) reviews the proposed preoperational and startup test programs to confirm that they are in conformance with the intent of Regulatory Guide 1.68 as part of its primary review responsibility for SRP Section 14.2. The PSRB also has primary review responsibility for Task Action Plan items II.K.1 (C.1.10) of NUREG-0737 (OLs only) and I.C.6 of NUREG-0718 (CPs only) regarding procedures to ensure that system operability status is known. The ASB reviews flood protection as part of its primary review responsibility for SRP Section 3.4.1. The ASB identifies the structures systems and components to be protected against externally generated missiles and reviews the adequacy of protection against such missiles as part of its primary review responsbility for SRP Section 3.5.1.4 and 3.5.2. The ASB also reviews protection against internally generated missiles both inside and outside of containment as part of its primary review responsibility for SRP Sections 3.5.1.1 and 3.5.1.2. The Power Systems Branch (PSB) identifies the safety-related electrical loads and determines that power systems supplying motive or control power for the RHR system meet acceptable criteria and will perform these intended functions during all plant operating and accident conditions as part of its primary review responsibility for SRP Sections 8.1, 8.2, 8.3.1, and 8.3.2. The Instrumentation and Control Systems Branch (ICSB), as part of its primary review responsibility for SRP Sections 7.1 and 7.4 reviews the instrumentation and control systems for the RHR system to determine that it will perform its design function as required and conform to all applicable acceptance criteria. The ICSB also reviews the provisions taken to meet GDC 19 with respect to equipment outside of the control room for hot and cold shutdown. The Radiological Assessment Branch (RAB) has primary review responsibility for SRP Section 12.1 through 12.5 including Task Action Plan items II.B.2 of NUREG-0737 and NUREG-0718 which involve a radiation and shielding design review and corrective actions taken to ensure adequate access to vital areas and protection of safety equipment (CPs and OLs). The review for Fire Protection, Technical Specifications, and Quality Asurance are coordinated and performed by the CMEB, Standardization and Special Projects Branch (SSPB) and Quality Assurance Branch (QAB) as part of their primary review responsibility for SRP Sections 9.5.1, 16.0 and 17.0, respectively.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP Section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

The Reactor Systems Branch acceptance criteria are based on meeting the requirements of the following regulations:

A. General Design Criterion 2 with respect to the seismic design of systems, structures and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system. Acceptability is based on meeting position C-2 of Regulatory Guide 1.29 or its equivalent.

B. General Design Criterion 4, as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

C. General Design Criterion 5 which requires that any sharing among nuclear power units of structures, systems and components important to safety will not significantly impair their safety function.

D. General Design Criterion 19 with respect to control room requirements for normal operations and shutdown, and;

E. General Design Criterion 34 which specifies requirements for a residual heat removal system.

Specific criteria necessary to meet the requirements of General Design Criteria 2, 4, 5, 19, and 34 are as follows:

1. The system or systems are to satisfy the functional, isolation, pressure relief, pump protection and test requirements specified in Branch Technical Position RSB 5-1.

2. In order to meet the requirements of General Design Criterion 4 (Ref 11), design features and operating procedures shall be provided to prevent damaging water hammer due to such mechanisms as voided pump discharge lines, water entrainment in steam lines and steam bubble collapse.

3. Interfaces between the RHR system and RCIC and component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHR system must conform to GDC 5.

4. The requirements for the reliability and capability of removing decay heat under the following Task Action Plan items must also be satisfied:

   a. Meeting Task Action Plan item II.E.3.2 of NUREG-0660 which involves systems reliability. NRR will conduct a generic study to assess the capability and reliability of shutdown heat removal systems under various transients and degraded plant conditions including complete loss of all feedwater. Deterministic and probabilistic methods will be used to identify design weaknesses and possible system modifications that could be made to improve the capability and reliability of these systems under all shutdown conditions. (CPs and OLs). Specific requirements will be based on the results of this study.

   b. Meeting Task Action Plan item II.E.3.3 of NUREG-0660 which involves a coordinated study of shutdown heat removal requirements. An effort to evaluate shutdown heat removal requirements in a comprehensive manner is required, thereby permitting a judgment of adequacy in terms of overall system requirements. As part of this project, NRR will conduct a study to assess the desirability of and possible requirement for a diverse heat-removal path, such as feed and bleed, particularly if all secondary-side cooling is unavailable. The NRC staff will work with the recently established ACRS Ad Hoc Subcommittee on this matter to develop a mutually acceptable overall study program. (CPs and OLs). Specific requirements will be based on the results of this study.

   c. Meeting Task Action Plan item II.B.8 of NUREG-0718 (Ref. 7) which involves description by the applicants of the degree to which the designs conform to the proposed interim rule on degraded core accidents. (CPs only)

   d. Meeting Action Plan item III.D.1.1 of NUREG-0737 (Ref. 8) and NUREG-0718 (Ref. 7) which involves primary coolant sources outside of containment (CPs and OLs).

5. When the RHR system is used to control or mitigate the consequences of an accident, it must meet the design requirements of an engineered safety

feature system. This includes meeting the guidelines of Regulatory Guide 1.1 regarding net positive suction head.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the Preliminary Safety Analysis Report meet the acceptance criteria given in subsection II.

For operating license (OL) reviews, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the Final Safety Analysis Report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

As noted in subsections I and II, the RSB review for PWRs is limited to the low pressure - low temperature RHR system. For BWRs, the review is to include all of the systems used to transfer residual heat from the reactor over the entire range of potential reactor coolant temperatures and pressures. The following steps are to be applied by the reviewer for the appropriate systems, depending on whether a PWR or BWR is being reviewed. These steps should be adapted to CP or OL reviews as appropriate.

1. Using the description given in the applicant's Safety Analysis Report (SAR), including component lists and performance specifications, the reviewer determines that the system(s) piping and instrumentation are such to allow the system(s) to operate as intended, with or without offsite power and given any single active component failure. This is accomplished by reviewing the piping and instrumentation diagrams (P&IDs) to confirm that piping arrangements permit the required flow paths to be achieved and that sufficient process sensors are available to measure and transmit required information. A failure modes and effects analysis (or similar system safety analysis) provided in the SAR is used to determine conformance to the single failure criterion.

2. Using the comparison tables of SAR Section 1.3, the RHR system is compared to designs and capacities of such systems in similar plants to see that there are no unexplained departures from previously reviewed plants. Where possible, comparisons should be made with actual performance data from similar systems in operating plants.

3. From the system description and P&IDs, the reviewer determines that the isolation requirements of Branch Technical Position RSB 5-1 (Ref. 5) are satisfied.

4. The reviewer determines that the RHR system design has provisions to prevent damage to the RHR pumps in accordance with Branch Technical Position RSB 5-1 (Ref. 5). The reviewer checks the isolation valves in the suction line for potential closure, NPSH requirements, pump runout, and potential loss of miniflow line during pump testing. If operator action is required to protect the pumps, the reviewer evaluates the instrumentation required to alert the operator and the adequacy of the time frame for operator action.

5. The RHR systems is reviewed to evaluate the adequacy of design features that have been provided to prevent damaging water (steam) hammer due to such mechanisms as voided discharge lines, water entrainment in steam lines and steam bubble collapse. For systems with a water supply above the discharge lines, voided lines are prevented by proper vent location and filling and venting procedures. The vents should be located for ease of operation and testing on a periodic basis. If the normal alignment of suction valves is to a source below the highest level of the pump discharge lines (e.g., the suppression pool for RHR systems of BWRs) back leakage through the pump discharge check valves will result in line voiding.

   Proper vent location and filling and venting procedures are still needed. In addition, a special keep-full system with appropriate alarms is needed to supply water to the discharge lines at sufficiently high pressure to prevent voiding. Operating and maintenance procedures shall be reviewed by the applicant to assure that adequate measures are taken to avoid water hammer due to voided line conditions.

   For RHR systems of BWRs which use the steam condensing mode of operation, the evaluation should include consideration of water hammer due to (a) water entrainment in the steam supply line during startup, (b) formation of steam bubbles in the RHR system pump discharge lines and heat exchangers resulting from leakage past valves in the steam supply line, and (c) water entrainment in the discharge line of the pressure relief valve used to prevent overpressurization of the system during operation in the steam condensing mode.

6. Using the system process diagrams, P&IDs, failure modes and effects analysis, and component performance specifications, the reviewer determines that the system(s) has the capacity to bring the reactor to conditions permitting operation of the RHR system in a reasonable period of time, assuming a single failure of an active component with only either onsite or offsite electric power available. For the purposes of this review, 36 hours is considered a reasonable time period. The ASB is responsible for the review of the initial cooldown phase for PWRs. Therefore, this review effort is to be coordinated with that branch. For the purposes of the review of both PWRs and BWRs, only the operation of safety grade equipment is to be assumed.

7. The cooldown function is to be reviewed to determine if it can be performed from the control room assuming a single failure of an active component, with only either onsite or offsite electric power available. Any operation required outside of the control room is to be justified by the applicant. Like item 5, the initial cooldown for PWRs is to be reviewed by ASB.

8. By reviewing the system description and the P&IDs, the reviewer confirms the RHR system satisfies the pressure relief requirements of Branch Technical Position RSB 5-1 (Ref. 5).

9. By reviewing the piping arrangement and system description of the RHR system, the reviewer confirms that the RHR system meets the requirements of GDC 5 (Ref. 2) concerning shared systems.

10. The RSB reviewer contacts the ASB reviewer in conjunction with his review of the RHR system heat sink and refueling system interaction to interchange information and assure that the reviews are consistent with regard to the interfacing parameters. For example, the ASB review determines the maximum service or component cooling water temperature. The RSB reviewer then reviews the RHR system description to determine that this maximum temperature has been allowed for in the RHR system design.

11. The RSB reviewer contacts his counterpart in the ICSB to obtain any needed information from their review. Specifically, ICSB confirms that automatic actuation and remote-manual valve controls are capable of performing the functions required, and that sensor and monitoring provisions are adequate. The instrumentation and controls of the RHR system are to have sufficient redundancy to satisfy the single failure criterion.

12. The RSB reviewer contacts his counterpart in CSB so that the information needed concerning their reviews will be interchanged.

13. The RSB reviewer contacts his counterpart in PSRB to discuss any special test requirements and to confirm that the proposed preoperational test program for the RHR system is in conformance with the intent of Regulatory Guide 1.68.

14. The proposed plant technical specifications are reviewed to:

    a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

    b. Verify that the frequency and scope of periodic surveillance testing is adequate.

15. The reviewer contacts the SGEB reviewer to confirm that the systems employed to remove residual heat are housed in a structure whose design and design criteria provide adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

16. For PWRs, the reviewer confirms that the auxiliary feedwater supply satisfies the requirements of Branch Technical Position RSB 5-1.

17. The RSB reviewer provides information to other branches in those areas where the RSB has a review responsibility that is not explicitly covered in steps 1-15 above. These additional areas of review responsibility include:

    a. Identification of engineered safety features (ESF) and safe shutdown electrical loads, and verification that the minimum time intervals for the connection of th ESF to the standby power systems are satisfactory.

    b. Identification of vital auxiliary systems associated with the RHR system and determination of cooling load functional requirements and minimum time intervals.

c. Identification of essential components associated with the main steam supply and the auxiliary feedwater system that are required to operate during and following shutdown.

18. The RSB review evaluates the applicant responses to the following Task Action Plan items:

   a. II.E.3.2 of NUREG-0660 (CPs and OLs)

   b. II.E.3.3 of NUREG-0660 (CPs and OLs)

   c. II.B.8 of NUREG-0718 (CPs only)

   d. III.D.1.1 of NUREG-0737 and NUREG-0718 (CPs and OLs)

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's Safety Evaluation:

For PWRs

The residual heat removal function is accomplished in two phases: the initial cooldown phase and the residual heat removal (RHR system) operation phase. In the event of loss of offsite power, the initial phase of cooldown is accomplished by use of the auxiliary feedwater system and the atmospheric dump valves. This equipment is used to reduce the reactor coolant system temperature and pressure to values that permit operation of the RHR system. The review of the initial cooldown phase is discussed in Section _____ of the SER. The review of the RHR system operational phase is discussed below. The residual heat removal (RHR) system removes core decay heat and provides long-term core cooling following the initial phase of reactor cooldown. The scope of review of the RHR system for the _____plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHR system and his analysis of the adequacy of those criteria and bases and the conformance of the design to these criteria and bases.

The staff concludes that the design of the Residual Heat Removal System is acceptable and meets the requirements of General Design Criteria 2, 4, 5, 19, and 34. This conclusion is based on the following:

(1) The applicant has met the General Design Criterion 2 with respect to position C-2 of Regulatory Guide 1.29 concerning the seismic design of systems, structures and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.

(2) The applicant has met the General Design Criterion 4 with respect to dynamic effects associated flow instabilities and loads (e.g., water hammer).

(3) The applicant has met the requirements of General Design Criterion 5 with respect to sharing of structure, systems and components by demonstrating that such sharing does not significantly impair the ability of the Residual Heat Removal System to perform it safety function including in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(4) The applicant has met General Design Criterion 19 with respect to the main control room requirements for normal operations and shutdown and General Design Criterion 34 which specifies requirements for the residual heat removal system by meeting the regulatory position in Branch Technical Position RSB 5-1.

In addition, the applicant has met the requirements of the following Task Action Plan Items:

(1) Task Action Plan item II.E.3.2 of NUREG-0660 (Ref. 10) as it relates to systems capability and reliability of shutdown heat removal systems under various transients.

(2) Task Action Plan item II.E.3.3 of NUREG-0660 (Ref. 10) as it relates to a coordinated study of shutdown heat removal requirements.

(3) Task Action Plan item II.B.8 of NUREG-0718 (Ref. 7) as it relates to description by the applicants of the degree to which the designs conform to the proposed interim rule on degraded core accidents (CPs only).

(4) Task Action Plan item III.D.1.1 of NUREG-0737 (Ref. 8) and NUREG-0718 (Ref. 7) as they relate to primary coolant sources outside of containment (CPs and OLs).

For BWRs

The residual heat removal function is accomplished in two phases: the initial cooldown phase and a low pressure-temperature operation phase. In the event of loss of offsite electrical power, the initial cooldown phase is accomplished using the reactor core isolation cooling (RCIC) system and the safety/ relief valves. The low pressure-temperature mode of operation is usually accomplished by the residual heat removal (RHR) system. However, certain single failures can render the RHR system inoperative. In that event, two alternate systems that use components of the RCIC and RHR system are available to bring the reactor to cold shutdown conditions.

The scope of review of these systems for the ________ plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for these systems and his analysis of the adequacy of those criteria and bases and of the conformance of the design to these criteria and bases.

The staff concludes that the design of the Residual Heat Removal System is acceptable and meets the requirements of General Design Criteria 2, 4, 5, 19, and 34. This conclusion is based on the following:

(1) The applicant has met General Design Criterion 2 with respect to position C-2 of Regulatory Guide 1.29 concerning the seismic design of systems, structures and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.

(2) The applicant has met the General Design Criterion 4 with respect to dynamic effects associated flow instabilities and loads (e.g., water hammer).

(3) The applicant has met the requirements of General Design Criterion 5 with respect to sharing of structures, systems, and components by demonstrating that such sharing does not significantly impair the ability of the Residual Heat Removal System to perform its safety function including in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(4) The applicant has met General Design Criterion 19 with respect to the main control room requirements for normal operations and shutdown and General Design Criterion 34 which specifies requirements for the residual heat removal system by meeting the regulatory position in Branch Technical Position RSB 5-1.

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(1) Task Action Plan item II.E.3.2 of NUREG-0660 as it relates to systems capability and reliability of shutdown heat removal systems under various transients.

(2) Task Action Plan item II.E.3.3 of NUREG-0660 as it relates to a coordinated study of shutdown heat removal requirements.

(3) Task Action Plan item II.B.8 of NUREG-0718 (Ref. 7) as it relates to description by the applicants of the degree to which the designs conform to the proposed interim rule on degraded core accidents (CPs only).

(4) Task Action Plan item III.D.1.1 of NUREG-0737 (Ref. 8) and NUREG-0718 (Ref. 7) as they relate to primary coolant sources outside of containment (CPs and OLs).

In addition to the above criteria, the acceptability of the RHR system may be based on the degree of design similarity with previously approved plants. Deviations from these criteria from other types of RHR systems (e.g., systems that are designed to withstand reactor coolant system operating pressure or systems located entirely inside containmnt) will be considered on an individual basis.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced BTP RSB 5-1, regulatory guides, NUREGs and implementation of acceptance criterion subsections II.B and II.2 is as follows:

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP aplicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems and Components."

3. 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

4. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

5. Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," attached to SRP Section 5.4.7.

6. Regulatory Guide 1.29, "Seismic Design Classification."

7. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."

8. NUREG-0737, "Clarification of TMI Action Plan Requirements."

9. Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Systems."

10. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident."

11. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

BRANCH TECHNICAL POSITION RSB 5-1
DESIGN REQUIREMENTS OF THE RESIDUAL HEAT REMOVAL SYSTEM

BACKGROUND

GDC 19 states that, "A control room shall be provided from which actions can be taken to operate the nuclear power unit under normal conditions. . ."

Normal operating conditions including the shutting down of a reactor; therefore, since the residual heat removal (RHR) system is one of several systems involved in the normal shutdown of all reactors, this system must be operable from the control room.

GDC 34 states that "Suitable redundance. . .shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure."

In most current plant designs the RHR system has a lower design pressure than the reactor coolant system (RCS), is located outside of containment and is part of the emergency core cooling system (ECCS). However, it is possible for the RHR system to have different design characteristics. For example, the RHR system might have the same design pressure as the RCS, or be located inside of containment. Plants which may have RHR systems that deviate from current designs will be reviewed on a case-by-case basis. The functional, isolation, pressure relief, pump protection, and test requirements for the RHR system are included in this position.

BRANCH POSITION

A. Functional Requirements

The system(s) which can be used to take the reactor from normal operating conditions to cold shutdown* shall satisfy the functional requirements listed below.

1. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. These systems shall satisfy General Design Criteria 1 through 5.

2. The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system function can be accomplished assuming a single failure.

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*Processes involved in cooldown are heat removal, depressurization, flow circulation, and reactivity control. The cold shutdown condition, as described in the Standard Technical Specifications, refers to a sub critical reactor with a reactor coolant temperature no greater than 200°F for a PWR and 212°F for a BWR.

3. The system(s) shall be capable of being operated from the control room with either only onsite or only offsite power available. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified.

4. The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure.

B. RHR System Isolation Requirements

The RHR system shall satisfy the isolation requirements listed below.

1. The following shall be provided in the suction side of the RHR system to isolate it from the RCS.

   (a) Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.

   (b) The valves shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply shall not cause any valve to change position.

   (c) The valves shall have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system.

2. One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS:

   (a) The valves, position indicators, and interlocks described in item 1(a) thru 1(c) above,

   (b) One or more check valves in series with a normally closed power-operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function, the power-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

   (c) Three check valves in series, or

   (d) Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.

C. Pressure Relief Requirements

The RHR system shall satisfy the pressure relief requirements listed below.

1. To protect the RHR system against accidental overpressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the ASME Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressure relieving capacity of the RHR system. For example, during shutdown cooling in a PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in selection of the design bases.

2. Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not:

   (a) Result in flooding of any safety-related equipment.

   (b) Reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA.

   (c) Result in a non-isolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of the containment.

3. If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing.

D. Pump Protection Requirements

The design and operating procedures of any RHR system shall have provisions to prevent damage to the RHR system due to overheating, cavitation or loss of adequate pump suction fluid.

E. Test Requirements

The isolation valve operability and interlock circuits must be designed so as to permit on line testing when operating in the RHR mode. Testability shall meet the requirements of IEEE Standard 338 and Regulatory Guide 1.22.

The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.68. The programs for PWRs shall include tests with supporting analysis to (a) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (b) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with performance of previously tested plants of similar design may be substituted for these tests.

F. Operational Procedures

The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For pressurized water reactors, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions.

G. Auxiliary Feedwater Supply

The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure.

H. Implementation

For the purposes of implementing the requirements for plant heat removal capabilitity for compliance with this position, plants are divided into the following three classes:

Class 1 - Full compliance with this position for all plants (custom or standard) for which CP or PDA applications are docketed on or after January 1, 1978. See Table 1 for possible solutions for full compliance.

Class 2 - Partial implementation of this position for all plants (custom or standard) for which CP or PDA applications are docketed before January 1, 1978, and for which an OL issuance is expected on or after January 1, 1979. See Table 1 for recommended implementation for Class 2 plants.

Class 3 - The extent to which the implementation guidance in Table 1 will be backfitted for all operating reactors and all other plants (custom or standard) for which issuance of the OL is expected before January 1, 1979, will be based on the combined I&E and DOR review of related plant features for operating reactors.

TABLE 1. POSSIBLE SOLUTION FOR FULL COMPLIANCE WITH BTP RSB 5-1 AND RECOMMENDED IMPLEMENTATION FOR CLASS 2 PLANTS

Design Requirements of BTP RSB 5-1	Process and [System or Component]	Possible Solution for Full Compliance	Recommended Implementation for Class 2 Plants (see Note 1)
I. Functional Requirement for Taking to Cold Shutdown a. Capability Using Only Safety Grade Systems b. Capability with either only onsite or only offsite power and with single failure (limited action outside CR to meet SF) c. Reasonable time for cooldown assuming most limiting SF and only offsite or only onsite power.	Long-term cooling [RHR drop line]	Provide double drop line (or valves in parallel) to prevent single valve failure from stopping RHR cooling function. (Note: This requirement in conjunction with meeting effects of single failure for long-term cooling and isolation requirements involve increased number of independent power supplies and possibly more than four valves).	Compliance will not be required if it can be shown that correction for single failure by manual actions inside or outside of containment or return to hot standby until manual actions (or repairs) are found to be acceptable for the individual plant.
	Heat removal and RCS circulation during cooldown to cold shutdown (Note: Need SG cooling to maintain RCS circulation even after RHR in operation when under natural circulation [steam dump valves].)	Provide safety-grade dump valves, operators, and power supply, etc. so that manual action should not be required after SSE except to meet single failure.	Compliance required.
	Depressurization (Pressurizer auxiliary spray or power-operated relief valves).	Provide upgrading and additional valves to ensure operation of auxiliary pressurizer spray using only safety-grade subsystem meeting single failure. Possible alternative may involve using pressurizer power-operated relief valves which have been upgraded. Meet SSE and single failure without manual operation inside containment.	Compliance will not be required if a) dependence on manual actions inside containment after SSE or single failure or b) remaining at hot standby until manual actions or repairs are complete are found to be acceptable for the individual plant.

TABLE 1. POSSIBLE SOLUTION FOR FULL COMPLIANCE WITH BTP RSB 5-1
AND RECOMMENDED IMPLEMENTATION FOR CLASS 2 PLANTS

Design Requirements of BTP RSB 5-1	Process and [System or Component]	Possible Solution for Full Compliance	Recommended Implementation for Class 2 Plants (see Note 1)
	Boration for cold shutdown [CVCS and boron sampling].	Provide procedure and upgrading where necessary such that boration to cold shutdown concentration meets the requirements of I. Solution could range from (1) upgrading and adding valves to have both letdown and charging paths safety grade and meet single failure to (2) use of backup procedures involving less cost. For example, boration without letdown may be acceptable and eliminate need for upgrading letdown path. Use of ECCS for injection of borated water may also be acceptable. Need surveillance of boron concentration (borunometer and/or sampling). Limited operator action inside or outside of containment if justified.	Same as above.
II. RHR Isolation	RHR System	Comply with one of allowable arrangements given.	Compliance required. (Plants normally meet the requirement under existing SRP Section 5.4.7).
III. RHR Pressure Relief			
Collect and contain relief discharge	RHR System	Determine piping, etc., needed to meet requirement to provide in design.	Compliance will not be required, if it is shown that adequate alternate methods of disposing of discharge are available.

TABLE 1. POSSIBLE SOLUTION FOR FULL COMPLIANCE WITH BTP RSB 5-1 AND RECOMMENDED IMPLEMENTATION FOR CLASS 2 PLANTS

Design Requirements of BTP RSB 5-1	Process and [System or Component]	Possible Solution for Full Compliance	Recommended Implementation for Class 2 Plants (see Note 1)
V. Test Requirement			
Meet R.G. 1.68. For PWRs, test plus analysis for cooldown under natural circulation to confirm adequate mixing and cooldown within limits specified in EOP.		Run tests confirming analysis to meet requirement.	Compliance required.
VI. Operational Procedure			
Meet R.G. 1.33. For PWRs, include specific procedures and information for cooldown under natural circulation.		Develop procedures and information from tests and analysis.	Compliance required.
VII. Auxiliary Feedwater Supply			
Seismic Category I supply for auxiliary FW for at least four hours at hot shutdown plus cooldown to RHR cut-in based on longest time for only onsite or only offsite power and assumed single failure.	Emergency Feedwater Supply	From tests and analysis obtain conservative estimate of auxiliary FW supply to meet requirement and provide seismic Category I supply.	Compliance will not be required, if it is shown that an adequate alternate seismic Category I source is available.

Note 1: The implementation for Class 2 plants does not result in a major impact while providing additional capability to go to cold shutdown. The major impact results from the requirement for safety-grade steam dump valves.

NUREG-0800
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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

Section No. 5.4.6
Revision No. 3

Appendix No. N/A
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Branch Tech. Position N/A
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FILING INSTRUCTIONS

PAGES TO BE REMOVED			NEW PAGES TO BE INSERTED		
PAGE NUMBER		DATE	PAGE NUMBER		DATE
5.4.6-1 thru 5.4.6-9	Rev. 2	July 1981	5.4.6-1 thru 5.4.6-10	Rev. 3	April 1984

The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

The reactor core isolation cooling (RCIC) system in a boiling water reactor (BWR) is a safety system which serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. Abnormal events which could cause such a situation to arise include an inadvertent isolation of all main steam lines, loss of condenser vacuum, pressure regulator failures, loss of feedwater, and the loss of offsite power. Each of these transients is analyzed in Chapter 15 of the applicant's safety analysis report (SAR). For each of these events, the high pressure part of the emergency core cooling system (ECCS) provides a backup function to the RCIC system. This review of the RCIC is performed to assure conformance with the requirements of General Design Criteria 4, 5, 29, 33, 34 and 54.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel and supplying steam to and removing condensate from the RCIC steam turbine where applicable. Fluid removed from the reactor vessel following a shutdown from power operation is normally made up by the feedwater system, supplemented by inleakage from the control rod drive system. If the feedwater system is inoperable, the RCIC turbine-pump unit starts automatically or is started by the operator from the control room. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool.

The review of the RCIC system includes the system design bases, design criteria, description, and the points noted below.

The RSB is responsible for performing the technical review of the RCIC system in the following areas:

1. The piping and instrumentation diagrams are reviewed to determine that the system is capable of performing its intended function and of being preoperationally and operationally tested.

2. The degree of separation of the RCIC system from the high pressure core spray (HPCS) system, or high pressure coolant injection (HPCI) system is reviewed for protection against common mode failure of redundant systems.

3. The process flow diagram is reviewed to confirm that the RCIC system design parameters are consistent with expected pressures, temperatures and flow rates.

4. The complete sequence of operation is reviewed to determine that the system can function as intended and that the system is capable of manual operation.

5. The system is reviewed for compliance with the applicable requirements of NUREG-0737 (Ref. 1).

In addition, the RSB will coordinate other branch evaluations that interface with the overall review of the system as follows: Auxiliary Systems Branch (ASB) reviews the RCIC and HPCI (or HPCS) systems for protection against common mode failures from missiles as part of its primary review responsibility for Standard Review Plan (SRP) Sections 3.5.1.1 and 3.5.1.2. Protection against flooding of RCIC and redundant equipment is reviewed by ASB as part of its primary review responsibility for SRP Section 3.4.1. Protection against damage from pipe whip and jet impingement is reviewed by the Mechanical Engineering Branch (MEB) as part of its primary review responsibility for SRP Sections 3.6.1 and 3.6.2. The Standardization and Special Projects Branch (SSPB) reviews the proposed technical specifications as part of its primary review responsibility for SRP Section 16.0. The Procedures and Systems Review Branch (PSRB) reviews the proposed preoperational and critical startup test programs as part of its primary review responsibility for SRP Section 14.2. The MEB reviews the RCIC system to assure that it has the proper seismic and quality group classification as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2. The RCIC is to be enclosed in a seismic Category I structure or building. The design adequacy of this structure or building is evaluated by the Structural and Geotechnical Engineering Branch (SGEB) as part of its primary review responsibility for SRP Sections 3.3, 3.4, 3.5, 3.7, and 3.8. The Containment Systems Branch (CSB) reviews the RCIC system, as part of its primary review responsibility for SRP Sections 6.2.4 and 6.2.6 to confirm that the design is compatible with the containment system and can be isolated. The Instrumentation and Control Systems Branch (ICSB), as part of its primary review responsibility for SRP Section 7.4, evaluates the adequacy of controls and instrumentation of the RCIC system with regard to the required features of automatic actuation, remote sensing and indication, and remote control. The Power Systems Branch (PSB), as part of its primary review responsibility for SRP Section 8.3, evaluates the adequacy of emergency onsite power, sufficiency of battery capacity, and the use of d-c power only. The MEB, as part of its primary review responsibility for SRP Section 3.9.3, ensures that the design and installation of the RCIC system meet applicable codes and are adequate for its proper functioning. The Equipment Qualification Branch (EQB) reviews RCIC system equipment to determine that it is seismically and environmentally qualified for its intended use as part of its primary review responsibility for SRP Sections 3.10 and 3.11.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

RSB acceptance criteria are based on meeting the relevant requirements of General Design Criteria 4, 5, 29, 33, 34 and 54. Specific criteria to meet the requirements of the above GDCs are as follows:

A. General Design Criteria 4, as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

B. General Design Criterion 5 as it relates to structures, systems and components important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function.

C. General Design Criterion 29 as it relates to the system being designed to have an extremely high probability of performing its safety function in the event of anticipated operational occurrences.

D. General Design Criterion 33 as it relates to the system capability to provide reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary so the fuel design limits are not exceeded.

E. General Design Criterion 34 as it relates to the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or reactor coolant pressure boundary overpressurization.

F. General Design Criterion 54 as it relates to piping systems penetrating primary containment being provided with leak detection and isolation capabilities.

Specific acceptance criteria, Regulatory Guides, and Task Action Plan items that provide information, recommendations and guidance and in general describe a basis acceptable to the staff that may be used to implement the requirements of the Commission regulations identified above are as follows:

1. The general objective of the review is to determine that the RCIC system, in conjunction with the HPCS (or HPCI) system, the safety/relief valves, and the suppression pool cooling mode of the residual heat removal system meets the requirements of General Design Criterion 34 (Ref. 2) by providing the capability for decay heat removal to allow complete shutdown of the reactor under conditions requiring its use. It must maintain the reactor water inventory above the top of the active fuel until the reactor is depressurized sufficiently to permit operation of the low pressure

cooling systems. The RCIC system, in conjunction with the HPCS (or HPCI) system, the safety/relief valves, and the suppression pool cooling mode of the RHR system must be capable of removing fission product decay heat and other residual heat from the reactor core following shutdown so as to preclude fuel damage or reactor coolant pressure boundary overpressurization. Since RCIC in conjunction with HPCS (or HPCI) is used to provide makeup inventory in some modes of residual heat removal, these systems should jointly meet the guidelines of BTP RSB 5-1, attached to SRP Section 5.4.7.

2. The RCIC system is also used to supply reactor coolant makeup for small leaks. Accordingly, the systems must meet the requirements of General Design Criterion 33 (Ref. 4) in this regard.

3. Historically, credit has been taken for RCIC system capability to mitigate the consequences of certain abnormal events; however, since the cooling function is redundant to the HPCI or HPCS system, the RCIC system itself is not required to meet the single failure criterion, but in conjunction with HPCS (or HPCI) must satisfy the single failure criterion in this regard. In addition, the RCIC system is to perform its function without the availability of any a-c power per the requirements of General Design Criterion 34 (Ref. 2), and in conjunction with HPCS (or HPCI) must be designed to assure an extremely high probability of accomplishing its safety function as required by General Design Criterion 29 (Ref. 6).

4. As a system which must respond to certain abnormal events, the RCIC system must be designed to seismic Category I standards (discussed in SRP Section 3.2.1) and must not be shared among nuclear power units except as permitted by General Design Criterion 5 (Ref. 7).

5. The RCIC and HPCS (or HPCI) systems must be protected against natural phenomena, external or internal missiles, pipe whip, and jet impingement forces so that such events cannot fail both systems simultaneously. Acceptance criteria for these are discussed in SRP Sections 3.3.1 through 3.6.2. Acceptance criteria for RCIC instrumentation are described in SRP Section 7.4.

6. The RCIC system must meet the requirements of General Design Criterion 54 (Ref. 8) with regard to leak detection and isolation provisions for lines passing through the primary containment. Other containment isolation criteria for RCIC are described in SRP Sections 6.2.4 and 6.2.6.

7. The RCIC system must meet the recommendations of Task Action Plan items II.K.1.22, II.K.3.13, II.K.3.15, II.K.3.22, II.K.3.24, and III.D.1.1 of NUREG-0737 (Ref. 1) and NUREG-0718 (Ref. 11) with regard to actions needed for operation, system initiation setpoint and automatic restart capability, break detection provisions, automatic suction switchover to the suppression pool, adequacy of space cooling, and leakage minimization, respectively.

8. If the RCIC system is used to control or mitigate the consequences of an accident, either by itself or as a backup to another system, it must meet the requirements of an engineered safety feature. The RCIC system must

meet the guidelines of Regulatory Guide 1.1 (Ref. 9) regarding net positive suction head.

9. In order to meet the requirements of General Design criterion 4 (Ref. 12) design features and operating procedures, designed to prevent damaging water hammer due-to such mechanisms as voided discharge lines, steam bubble collapse and water entrainment in steam lines, shall be provided.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II.

For the operating license (OL) review, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

1. Using the RCIC operating requirements specified in SAR Section 5.4.6 and Chapter 15, the reviewer confirms that the RCIC system can maintain coolant inventory in the reactor vessel to keep the core covered and assure cladding integrity. This determination is based on engineering judgment and independent calculations (where deemed necessary), using information as specified in steps 2 and 3 below. The reviewer consults with the CPB to assure that the decay heat loads used in the RCIC analyses are applicable and suitably conservative.

2. Using the description given in Section 5.4.6 of the SAR, including component lists and performance specifications, the reviewer determines that the RCIC system piping and instrumentation are such as to allow the system to operate as intended. This is accomplished by reviewing the piping and instrumentation diagrams to confirm that piping arrangements permit the required flow paths to be achieved and that sufficient process sensors are available to measure and transmit required information.

3. Using the comparison tables of SAR Section 1.3, the RCIC system is compared to designs and capacities of such systems in similar plants to see that there are no unexplained departures from previously reviewed plants. Where possible, comparisons should be made with actual performance data from similar systems in operating plants.

4. The reviewer checks the piping and instrumentation diagrams and equipment layout drawings for the RCIC and HPCS (or HPCI) systems to see that the systems are physically separated and can function independently.

5. The reviewer examines the system design in SAR Section 5.4.6 to verify that the capability for automatic switchover of suction from the condensate storage tank to the suppression pool has been provided per the requirements of item II.K.3.22 of NUREGs-0737 and 0718 (Ref. 1 and 11). The reviewer also judges whether adequate control and monitoring infor-

mation is available to allow the operator to actuate the system manually or to realign the RCIC system manually within the time allowed (i.e., change the RCIC system suction from the condensate storage tank to the suppression pool, or to the steam condensing mode of the residual heat removal system).

6. The reviewer contacts ICSB to confirm that automatic actuation and remote-manual valve controls are capable of performing the functions required and that sensor and monitoring provisions are adequate. The instrumentation and controls of the RCIC system, in conjunction with the HPCS (or HPCI) system are to have sufficient redundancy to satisfy the single failure criterion.

7. The reviewer contacts PSB to ascertain that the RCIC system operation is not dependent on a-c power sources, and that there is sufficient battery capability to permit operation of the RCIC for a period of two hours without the availability of a-c power.

8. The reviewer checks with MEB to verify that essential RCIC system components are designated seismic Category I.

9. The reviewer contacts PSRB to verify that the applicant's proposed preoperational and initial startup test programs are in compliance with Regulatory Guide 1.68 (Ref. 10). At the OL stage, the reviewer confirms with PSRB that sufficient information is provided by the applicant to identify the test objectives, methods of testing, and test acceptance criteria (see par. C.2.b of Regulatory Guide 1.68). PSRB also verifies that the proposed test programs will provide reasonable assurance that the RCIC system will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the RCIC system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the RCIC system. If the RCIC system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are reviewed at the CP stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

10. The SSPB is contacted in regard to the proposed plant technical specifications to:

    a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

    b. Verify that the frequency and scope of periodic surveillance testing is adequate.

11. The reviewer confirms that the RCIC is housed in a structure whose design and design criteria have been reviewed by other branches (i.e., ASB, SGEB, MEB) to assure that it provides adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

12. Upon request from the primary reviewer, other branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

13. The reviewer checks the automatic and manual actions necessary for proper functioning of the RCIC system (in conjunction with the HPCS or HPCI, the safety relief valves and the suppression pool cooling mode of RHR) for completeness and practicality when used for residual heat removal per the requirements of item II.K.1.22 of NUREGs-0737 and 0178 (Ref. 1 and 11).

14. The reviewer checks the RCIC system break detection provisions to see that the system is protected against spurious trip signals per the requirements of item II.K.3.15 of NUREGs-0737 and 0718 (Ref. 1 and 11).

15. The reviewer confirms, in conjunction with ASB as necessary, that the RCIC system can withstand a loss of offsite power to its support systems, including space coolers, for at least two hours per the requirement of item II.K.3.24 of NUREGs-0737 and 0718 (Ref. 1 and 11).

16. The reviewer confirms per the requirements of item II.K.3.13 of NUREGs-0737 and 0178 (Ref. 1 and 11) that analyses have been provided or referenced to determine the need to separate the RCIC and the HPCS (or HPCI) initiation levels. Based on these study results, the reviewer checks the RCIC design for appropriate provisions. In addition, the reviewer checks to see that automatic restart capability is provided for RCIC.

17. The reviewer checks (by calculation as necessary) to see that adequate net positive suction head is available for RCIC suction from all potential sources (i.e., condensate storage tank, suppression pool, or RHR steam condensing mode discharge).

18. The reviewer examines the RCIC in conjunction with the HPCS or HPCI, the safety/relief valves and the suppression pool cooling mode of RHR for conformance to the recommendations of BTP RSB 5-1 to SRP Section 5.4.7 regarding residual heat removal.

19. The RCIC system is reviewed to evaluate the adequacy of design features that have been provided to prevent damaging water (steam) hammer due to such mechanisms as voided discharge lines, water entrainment and steam bubble collapse. If the normal water supply is above the discharge lines, voided lines are prevented by proper vent location and filling and venting procedures. The vents should be located for ease of operation and testing on a periodic basis. If the normal alignment of the suction valves is to a source below the highest level of the pump discharge lines (e.g., the suppression pool,) back leakage through the pump discharge check valves will result in line voiding. Proper vent location and filling and venting procedures are still needed. In addition, a special keep-full system with appropriate alarms is needed to supply water to the discharge lines at sufficiently high pressure to prevent voiding. Operating and maintenance procedures shall be reviewed by the applicant to assure that adequate measures are taken to avoid water hammer due to voided line conditions.

The RCIC system uses a steam-driven turbine. Typical design features for the steam supply line include (a) drain pots, (b) sloped lines, and (c) limitations on opening and closing sequences and-seal-ins for manual operation of the isolation valves to preclude introducing water slugs into the line. The turbine exhaust line features include sloped lines and vacuum breakers.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

> The reactor core isolation cooling (RCIC) system includes the piping, valves, pumps, turbines, instrumentation, and controls used to maintain water inventory in the reactor vessel whenever it is isolated from the main feedwater system. Certain engineered safety features (HPCS or HPCI) provide a redundant backup for this function. The scope of review of the RCIC system for the ________ plant included piping and instrumentation diagrams, equipment layout drawings, and functional specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RCIC system, his analysis of the adequacy of the criteria and bases, and the conformance of the design to these criteria and bases.

The staff concludes that the reactor core isolation cooling system design is acceptable and meets the requirements of General Design Criteria 4, 5, 29, 33, 34 and 54. This conclusion is based on the following:

1. The applicant has met the requirements of (cite Reg.) with respect to (state limits of review) by: (Use one or more of the following as applicable)

   a. meeting the regulatory position in Regulatory Guide ________,

   b. providing and meeting an alternative method to the regulatory position in Regulatory Guide ________, that the staff has reviewed and found to be acceptable,

   c. meeting the regulatory position in BTP__________.

   d. The calculational method used by the applicant for (state) has been previously reviewed by the staff and found acceptable; the staff has reviewed the key parameters in this case and found them to be suitably conservative.

   e. The applicant has met the requirements of (industry standard - number and title) that has been reviewed by the staff and determined to be appropriate for this application.

2. Repeat the above discussion for each GDC listed.

In addition, conformance with General Design Criterion 55, 56, and 57 regarding containment isolation is discussed in Section 6.2 of this report. Conformance with General Design Criterion 2 and 4 for protection against natural phenomena, environmental hazards and potential missiles is discussed in Sections 3.3 through 3.6 of this report.

The RCIC and HPCS (or HPCI) systems, in conjunction with the safety/relief valves and the suppression pool cooling mode of the residual heat removal system, have been found capable of removing core decay heat following feedwater system isolation and reactor shutdown so that sufficient coolant inventory is maintained in the reactor vessel to keep the core covered and ensure cladding integrity. This capability has been found to be available even with a loss of offsite power and with a single active failure.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides, NUREGs and implementation of acceptance criterion subsections II.A and II.9 is as follows:

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. NUREG-0737, "Clarifiation of TMI Action Plan Requirements," November 1980.

2. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

3. Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," attached to SRP Section 5.4.7.

4. 10 CFR Part 50, Appendix A, General Design Criterion 33, "Reactor Coolant Makeup."

5. Regulatory Guide 1.53, "Single Failure Criterion."

6. 10 CFR Part 50, Appendix A, General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

7. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

8. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

9. Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Systems."

10. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

11. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."

12. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases".

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Standard Review Plan for the
Review of Safety Analysis Reports
for Nuclear Power Plants

Section No. 6.3
Revision No. 2

Appendix No. N/A
Revision No. N/A

Branch Tech. Position RSB 6-1
Revision No. 1

Date Issued April 1984

FILING INSTRUCTIONS

PAGES TO BE REMOVED			NEW PAGES TO BE INSERTED		
PAGE NUMBER		DATE	PAGE NUMBER		DATE
6.3-1 thru 6.3-12	Rev. 1	July 1981	6.3-1 thru 6.3-13	Rev. 2	April 1984
BTP RSB 6-1 to SRP Section 6.3	Rev. 1	July 1981	BTP RSB 6-1 to SRP Section 6.3	Rev. 1	July 1981 (No change except for page numbers)
6.3-13 thru 6.3-14			6.3-14 thru 6.3-15		

The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

6.3 EMERGENCY CORE COOLING SYSTEM

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

The RSB reviews the information presented in the applicant's safety analysis report (SAR) regarding the emergency core cooling system (ECCS). The major elements of the review are:

1. Design Bases

The design bases for the ECCS are reviewed to assure that they satisfy applicable regulations, including the general design criteria and the amendments to 10 CFR Part 50 regarding ECCS acceptance criteria issued by the Commission on December 28, 197_ (Ref. 1).

2. Design

The design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases.

3 Test Program

The preoperational and initial startup test programs for the ECCS are reviewed by the Procedures and Systems Review Branch (PSRB) to determine if they are sufficient to confirm the performance capability of the ECCS. RSB reviews the need for special design features to permit the performance of adequate test programs.

4. Technical Specifications

The proposed technical specifications are reviewed to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The ability of the ECCS to mitigate the consequences of a spectrum of loss-of-coolant accidents is reviewed by RSB under SRP Section 15.6.5.

In addition the RSB will coordinate with other branches evaluations that interface with the overall ECCS review as follows: Auxiliary Systems Branch (ASB), as part of its primary review responsibility for SRP Sections 9.2.1, 9.2.2, 9.2.5, and 9.2.6, reviews those auxiliary systems essential for ECCS operation (service water system, component cooling system, ultimate heat sink, and condensate storage facility) and assesses the capability of these systems to perform all functions required by the ECCS. The ASB will supply, on request, evaluations of portions of the power conversion systems (e.g., steam supply lines, steam generators, feedwater systems) which interface with the reactor coolant system in such a way as to influence the course of a loss-of-coolant accident (LOCA) for a particular plant. The ASB also reviews the effects of pipe breaks outside containment on ECCS. This review includes the effect of pipe whip, jet impingement forces, and environmental conditions created as part of its primary review responsibility for SRP Section 3.6.1. Instrumentation and Control Systems Branch (ICSB), as part of its primary review responsibility for SRP Section 7.3, reviews the adequacy of ECCS-associated controls and instrumentation with regard to the features of automatic actuation, remote sensing and indication, and remote control. The Containment Systems Branch (CSB) verifies that portions of the ECCS penetrating the containment barrier are designed with acceptable isolation features to maintain containment integrity for all operating conditions, including accidents, as part of its primary review responsibility for SRP Section 6.2.4. The Power Systems Branch (PSB) as part of its primary review responsibility for SRP Sections 8.1, 8.2, 8.3.1, and 8.3.2, reviews the adequacy of the power supply for the ECCS. The Mechanical Engineering Branch (MEB), as part of its primary review responsibility for SRP Section 3.9.3, reviews the loading combinations (operational, LOCA, and seismic) and the associated stress limits. In addition, the MEB, as part of its primary review responsibility for SRP Section 3.6.2, reviews the criteria used for postulating the effects of pipe breaks both inside and outside containment on ECCS. This review includes criteria used for postulating the effects of pipe whip, jet impingement forces, and any related environmental conditions. The ECCS is also reviewed by MEB to assure that system and components have the proper seismic and quality group classifications. This aspect of the review is performed as part of its primary review responsibility for SRP Sectons 3.2.1 and 3.2.2. The Structural and Geotechnical Engineering Branch (SGEB) reviews the structures housing the ECCS for the proper seismic classification as part of its primary review responsibility for SRP Sections 3.8.1, 3.8.2, and 3.8.3. The Materials Engineering Branch (MTEB), on a generic basis, reviews the thermal shock effect of water injected into the primary coolant system from the ECCS. The Procedures and Systems Review Branch (PSRB) reviews the proposed preoperational and initial startup test programs to determine that they are consistent with the intent of Regulatory

Guides 1.68 and 1.79 as part of its primary review responsibility for SRP Section 14.2.

The PSRB also has primary review responsibility for Task Action Plan items II.K.1 (C.1.10) of NUREG-0694 (OLs only) and I.C.6 of NUREG-0718 (CPs only) regarding procedures to ensure that system operability status is known. The Radiological Assessment Branch (RAB) has primary review responsibility for SRP Sections 12.1 through 12.5 including Task Action Plan items II.B.2 of NUREG-0694 and NUREG-0718 which involve radiation and shielding design review to take corrective actions to ensure adequate access to vital areas and protection of safety equipment (CPs and OLs). The review for Technical Specifications and Quality Assurance are coordinated and performed by the Standardization and Special Projects Branch and Quality Assurance Branch as part of their primary review responsibility for SRP Sections 16.0 and 17.0, respectively.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

The RSB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 2 as it relates to the seismic design of structures, systems, and components whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function. Acceptability is based on meeting position C2 of Regulatory Guide 1.29.

B. General Design Criterion 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

C. General Design Criterion 5 as it relates to structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.

D. General Design Criterion 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions.

E. General Design Criterion 27 as it relates to the system design having the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

F. General Design Criteria 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.

G. 10 CFR Part 50, §50.46, and Appendix K to 10 CFR Part 50 as it relates to the ECCS being designed so that its cooling performance is in accordance with an acceptable evaluation model.

Specific acceptance criteria, Regulatory Guides, and Task Action Plan items that provide information, recommendations, and guidance and in general describe a basis acceptable to the staff that may be used to implement the requirements of the Commission regulations identified above are as follows:

In regard to the ECCS acceptance criteria (Ref. 1), the five major performance criteria deal with:

1. Peak cladding temperature.
2. Maximum calculated cladding oxidation.
3. Maximum hydrogen generation.
4. Coolable core geometry.
5. Long-term cooling.

These areas are reviewed as a part of the effort associated with the LOCA analysis (SRP Section 15.6.5). However, the impact of various postulated single failures on the operability of the ECCS is evaluated under this SRP section.

The ECCS must meet the requirements of GDC 35 (Ref. 6). The system must have alternate sources of electric power, as required by GDC 17 (Ref. 4), and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident.

The ECCS must be designed to permit periodic inservice inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, piping, pumps, and valves in accordance with the requirements of GDC 36 (Ref. 7). The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by GDC 37 (Ref. 8).

The combined reactivity control system capability associated with ECCS must meet the requirements of GDC 27 (Ref. 5) and should conform to the recommendation of Regulatory Guide 1.47 (Ref. 11). The primary mode of actuation for the ECCS must be automatic, and actuation must be initiated by signals of suitable diversity and redundance. Provisions should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room.

The design of the ECCS should conform to the recommendations of Regulatory Guide 1.1 (Ref. 9).

Design features and operating procedures, designed to prevent damaging water hammer due to such mechanisms as voided discharge lines and water entrainment in steam lines shall be provided, in order to meet the requirements of General Design Criterion 4 (Ref. 17).

The design of those portions of the system which are not safety related, whose failures could have an adverse effect on the ECCS system, must be in accordance with GDC 2 (Ref. 2), and acceptance is based on meeting Position C2 of Regulatory Guide 1.29 (Ref. 10).

Interfaces between the ECCS and component or service water systems must be such that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems, e.g., residual heat removal (RHR) and containment heat removal systems, the ECCS must conform to GDC 5 (Ref. 3).

The requirements of the following Task Action Plan items must also be satisfied:

1. Task Action Plan Item II.B.8 of NUREG-0718 (Ref. 14) which involves description by the applicants of the degree to which the designs conform to the proposed interim rule on degraded core accidents (CPs and OLs).

2. Task Action Plan Item III.D.1.1 of NUREG-0694 and NUREG-0718 which involves primary coolant sources outside of containment (CPs and OLs).

3. Task Action Plan Item II.E.2.1 of NUREG-0737 which involves reliance on ECCS.

4. Task Action Plan Item II.K.3(10) of NUREG-0737 and NUREG-0718 which involves final recommendations by B&O task force regarding applicant's proposal of use of anticipatory trips only at high power for selected plants.

5. Task Action Plan Item II.K.3(15) of NUREG-0737 and NUREG-0718 which involves isolation of HPCI and RCIC for BWR plants.

6. Task Action Plan Item II.K.3(18) of NUREG-0737 and NUREG-0718 involving ECCS outages for all plants.

7. Task Action Plan Item II.K.3(21) of NUREG-0737 and NUREG-0718 which involves a study evaluating restart of LPCS and LPCI after manual trip for BWR plants.

8. Task Action Plan Item II.K.3(39) of NUREG-0660 which involves evaluation of effects of water slugs in piping caused by HPI and CFT flows in B&W plants.

In addition to the above criteria, the acceptability of the ECCS may be based on the degree of design similarity with previously approved plants.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II of this SRP section.

For operating license (OL) reviews, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

Much of the review described below is generic in nature and is not performed for each plant. That is, the RSB reviewer compares the ECCS design and parameters to those of previously reviewed plants and then devotes the major portion of the review effort to those areas where the application is not identical to previously reviewed plants. The following steps are taken by the RSB reviewer to determine that the acceptance criteria of subsection II have been met. These steps should be adapted to CP or OL reviews as appropriate.

1. The relationship of the system under review to other previously approved plants is established. Systems or design features claimed to be identical or equivalent to those of previously approved plants are confirmed to be identical or equivalent.

2. Piping diagrams are reviewed to evaluate the functional reliability of the system in the event of single failures. That is, by referring to piping and instrumentation diagrams, the existence of the redundancy required by the criteria is confirmed.

3. The significant design parameters (e.g., pump net positive suction head, pump head vs. flow, accumulator volume and pressure, water storage volume, system flow rate and pressure, etc.) are examined for each component to confirm that these parameters satisfy operating requirements and the recommendations of Regulatory Guide 1.1 (Ref. 9).

4. The piping and instrumentation diagrams are checked in consultation with MEB to see that essential ECCS components are designated seismic Category I and Safety Class II (the cooling water side of heat exchangers can be Safety Class III).

5. The ECCS design is reviewed to confirm that the system can function in postaccident environments, considering possible mechanical effects, missiles, and the pressure, temperature, moisture, radioactivity, and chemical conditions resulting from LOCA. Protection against valve motor flooding should be confirmed by the RSB reviewer. Regarding the effects of pressure, temperature, etc., the RSB reviewer should confirm that accident conditions are specified which provide the basis for proof tests for environmental qualification of ECCS components.

6. The criteria, supporting analyses, plant design provisions, and operator actions that will be taken are reviewed to ensure that there will not be unacceptably high concentrations of boric acid in the core region (resulting in precipitation of a solid phase) during the long-term cooling phase following a postulated LOCA.

7. The ECCS design is reviewed to confirm that there are provisions for maintenance of the long-term coolant recirculation and decay heat removal systems, e.g., pump or valve overhaul, in the post-LOCA environment (including consideration of radioactivity).

8. The availability of an adequate source of water for the ECCS is confirmed, and the source volume, location, and susceptibility to failure (e.g., freezing) are evaluated. (RSB will request ASB review as required.) In PWRs, the piping from the water source to the ECCS safety injection pumps is evaluated for conformance with RSB 6-1 (Ref. 13).

9. The ECCS flow paths are reviewed to determine the extent to which flow from the ECCS pumps is diverted as a backup feature to other safeguards equipment (e.g., RHR, containment spray). The reviewer should confirm that the remaining portion of the flow provides abundant core cooling, despite the most severe single failure that affects ECCS flow.

10. For a boiling water reactor (BWR), the reactor coolant automatic depressurization system is reviewed to confirm the capability to satisfy LOCA pressure relief functions, including consideration of a single failure.

11. The design of ECCS injection lines is reviewed to confirm that the isolation provisions at the interface with the reactor coolant system are adequate. The number and type of valves used to form the interface between low pressure portions of the ECCS and the reactor coolant system must provide adequate assurance that the ECCS will not be subjected to a pressure greater than its design pressure. This may be accomplished by any of the following provisions:

    a. One or more check valves in series with a normally closed motor-operated valve. The motor-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

    b. Three check valves in series.

    c. Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leaktightness and the testing is performed at least annually.

12. The reviewer should identify those portions of nonsafety-related systems which could have an adverse effect on ECCS and should ensure that modifications are in place to correct these situations.

13. Motor-operated isolation valves in ECCS lines connecting the accumulators to the reactor coolant system in a pressurized water reactor (PWR) are reviewed to ensure that adequate provisions are made against inadvertent isolation.

14. The capacity and settings of relief valves provided for the ECCS to satisfy system overpressure protection requirements are reviewed. In particular, for PWRs, the reviewer confirms that the accumulator relief

valves have adequate capacity so that leakage from the reactor coolant system will not jeopardize the integrity of the accumulators.

15. The ECCS is reviewed to evaluate the adequacy of design features that have been provided to prevent damaging water (steam) hammer due to such mechanisms as voided discharge lines, water entrainment in steam lines and steam bubble collapse. For systems with a water supply above the discharge lines, voided lines are prevented by proper vent location and filling and venting procedures. However, for the core spray and low pressure coolant injection systems of BWRs, the low elevation of the suppression pool will result in line voidage because of back leakage through pump discharge check valves and leaking valves in the full flow test line. Proper vent location and filling and venting procedure are still needed. In addition, a special keep-full system with appropriate alarms is needed to supply water to the discharge lines for any system which has a water source below the level of the highest pump discharge lines and at sufficiently high pressure to prevent voiding.

    For the High Pressure Coolant Injection (HPCI) system of BWRs which uses a steam-driven turbine, typical design features for the steam supply line include (a) drain pots with testable drain pot level switches, (b) sloped lines, and (c) limitations on opening and closing sequences and seal-ins for manual operation of the isolation valves to prevent introducing water slugs into the line. The turbine exhaust line features include sloped lines and vacuum breakers.

16. The reviewer confirms that no component or feature of the ECCS in one reactor facility on a multiple plant site is shared with the ECCS in another facility, or that shared features clearly meet the requirements of GDC 5 (Ref. 3).

17. The reviewer confirms that within an individual reactor facility, any components shared between the ECCS and other systems (e.g., coolant makeup systems, residual heat removal systems, containment cooling systems) satisfy engineered safeguard feature design requirements and that the ECCS function of the shared component is not diminished by the sharing.

18. The reviewer confirms that ECCS components located exterior to the reactor containment are housed in a structure which, in the event of leakage from the ECCS, permits venting of releases through iodine filters designed in accordance with Regulatory Guide 1.52.

19. The complete sequence of ECCS operation from accident occurrence through long-term core cooling is examined to see that a minimum of manual action is required and, where manual action is used, a sufficient time (greater than 20 minutes) is available for the operator to respond.

20. The reviewer confirms that long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. If an intermediate heat transport system, such as the component cooling water system, is used to provide long-term cooling capability, the system must be designed and constructed to an appropriate group classification, must be seismic Category I, and must be capable of sustaining a single active or passive failure without loss of function.

21. The RSB reviewer consults with the ICSB reviewer to:

a. Confirm that the power requirements of the ECCS, including the timing of electrical loads, are compatible with the design of onsite emergency power systems, both a-c and d-c.

b. Confirm that there are sufficient instrumentation and controls available to the reactor operator to provide adequate information in the control room to assist in assessing post-LOCA conditions, including the more significant parameters such as coolant flow, coolant temperature, and containment pressure. If ECCS flow is diverted as a backup to other safeguards systems, the reviewer confirms that instrumentation and controls are available to provide sufficient information in the control room to determine that adequate core cooling is being provided.

c. Confirm that automatic actuation and remote-manual valve controls are capable of performing the functions required, that suitable interlocks are provided, which do not impair separation of power trains or inhibit the required valve motions, and that instrumentation and controls have sufficient redundancy to satisfy the single failure criterion.

22. Analyses are provided by the applicant in Chapter 15 of the SAR to assess the capability of the ECCS to meet functional requirements. These analyses are reviewed by the RSB, as described in SRP Section 15.6.5, to determine conformance to the acceptance criteria for ECCS. However, the following portions of the review of ECCS response in loss-of-coolant accidents are performed by the RSB reviewer under this SRP section:

a. The lower limit of break size for which ECCS operation is required is established; i.e., the maximum break size for which normal reactor coolant makeup systems can maintain reactor pressure and coolant level is determined. The capability of the ECCS to actuate and perform at this lower limit of break size is confirmed.

b. The reviewer confirms that the analyses take into account a variety of potential locations for postulated pipe breaks, including ECCS injection lines.

c. The reviewer confirms that the analyses take into account a variety of single active failures. The reviewer should keep in mind that different single failures may be limiting, depending on the particular break location and break size postulated.

d. The ECCS component response times (e.g., for valves, pumps, power supply) are reviewed to confirm that they are within the delay times used in the accident analyses.

e. The ECCS design adequacy for all modes of reactor operation (e.g., full power, low power, hot standby, cold shutdown, partial loop isolation) is confirmed.

23. The proposed plant technical specifications are reviewed to:

a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when ECCS equipment is inoperable due to repairs and maintenance. The means of indicating that safety systems have been bypassed or are inoperable should be in accordance with Regulatory Guide 1.47 (Ref. 11).

b. Confirm that the limiting conditions of operation ensure that the specified operating parameters (minimum poison concentrations, minimum coolant reserve in storage, etc.) are within the bounds of the analyzed conditions.

c. Verify that the frequency and scope of periodic surveillance testing is adequate.

24. The reviewer confirms that the design provides the capability for periodically demonstrating that the system will operate properly when an accident signal is received. That is, it should be demonstrated by an applicant that pumps and valves operate on normal and emergency power and that water pressure and flow are as designed when the plant is operating (periodic system surveillance). When the plant is shut down for refueling, the system should be tested for delivery of coolant to the vessel.

25. The RSB reviewer contacts his counterpart in PSRB to discuss any special test requirements and to confirm that the proposed preoperational test program for the ECCS is in conformance with the intent of Regulatory Guide 1.68 (Ref. 12). |

26. The RSB review evaluates the applicant responses to the following Task Action Plan items:

(a) II.B.8 of NUREG-0718 (CPs only)
(b) III.D.1.1 of NUREG-0737 and NUREG-0718 (CPs and OLs)
(c) II.E.2.1 of NUREG-0660
(d) II.K.3(10) of NUREG-0660
(e) II.K.3(15) of NUREG-0660
(f) II.K.3(18) of NUREG-0660
(g) II.K.3(21) of NUREG-0660
(h) II.K.3(39) of NUREG-0660

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions which should be included in the staff's safety evaluation report. (For completeness, this evaluation finding includes the RSB review effort described in SRP Section 15.6.5.)

The emergency core cooling system (ECCS) includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transfer heat from the core following a loss-of-coolant accident. The scope of review of the ECCS for the ___________ plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

The staff concludes that the design of the Emergency Core Cooling System is acceptable and meets the requirements of General Design Criteria 2, 4, 5, 17, 27, 35, 36, and 37. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 2 with regard to the seismic design of nonsafety systems or portions thereof which could have an adverse effect on ECCS by meeting position C.2 of Regulatory Guide 1.29.

(2) The applicant has met the requirements of GDC 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

(3) The applicant has met the requirements of GDC 5 with respect to sharing of structures, systems, and components by demonstrating that such sharing does not significantly impair the ability of the ECCS to perform its safety function including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(4) The applicant has met the requirements of GDC 17 with regard to providing sufficient capacity and capability to assure that (a) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (b) the core is cooled and vital functions are maintained in the event of postulated accidents.

(5) The applicant has met the requirements of GDC 27 with regard to providing combined reactivity control system capability to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained and the applicant's design meets the guidelines of Regulatory Guide 1.47.

(6) The applicant has met the requirements of GDC 35 to provide abundant cooling for ECCS by providing redundant safety-grade systems that meet the recommendations of Regulatory Guide 1.1.

(7) The applicant has met the requirements of GDC 36 with respect to the design of ECCS to permit appropriate periodic inspection of important components of the system.

(8) The applicant has met the requirements of GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation.

(9) The applicant has provided an analysis of the proposed ECCS relative to the acceptance criteria of 10 CFR Part 50, §50.46, and Appendix K to demonstrate that their ECCS designs for peak cladding temperature, maximum calculated cladding oxidation, maximum hydrogen generation, coolable core geometry, and long-term cooling are in accordance with the acceptable evaluation model.

In addition, the applicant has met the requirements of the following Task Action Plan items:

(1) Meeting Task Action Plan item II.B.8 of NUREG-0718 (Ref. 14) which involves description by the applicants of the degree to which the designs conform to the proposed interim rule on degraded core accidents (CPs only).

(2) Meeting Task Action Plan item II.D.1.1 of NUREG-0737 (Ref. 15) and NUREG-0718 (Ref. 14) which involves primary coolant sources outside of containment (CPs and OLs).

(3) Meeting Task Action Plan item II.E.2.1 of NUREG-0660 (Ref. 16) which involves reliance on ECCS.

(4) Meeting Task Action Plan item II.K.3(10) of NUREG-0660 which involves applicant's proposal to limit anticipatory trip to high power for selected plants.

(5) Meeting Task Action Plan item II.K.3(15) of NUREG-0660 which involves isolation of HPCI and RCIC for BWR plants.

(6) Meeting Task Action Plan item II.K.3(18) of NUREG-0660 which involves ECCS outages for all plants.

(7) Meeting Task Action Plan item II.K.3[illegible] of NUREG-0660 which involves restart of LPCS and LPCI for BWR plants.

(8) Meeting Task Action Plan item II.K.3(3a) of NUREG-0660 which involves evaluation of effects of water slugs in piping caused by HPI and CFT flows in B&W plants.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides, NUREGs, BTP RSB 6-1 and implementation of acceptance criterion subsection II.B is as follows:

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. 10 CFR Part 50, §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," issued by the Commission December 28, 1973; Federal Register, Vol. 39, No. 3, January 4, 1974.

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."

5. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control System Capability."

6. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

7. 10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System."

8. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

9. Regulatory Guide 1.1, "Net Position Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

10. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

11. Regulatory Guide 1.47, "Bypass and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

12. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

13. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

14. Branch Technical Position RSB 6-1, "Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps," attached to SRP Section 6.3.

15. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

16. NUREG-0737, "Clarification of TMI Action Plan Requirements."

17. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Basis."

BRANCH TECHNICAL POSITION RSB 6-1

PIPING FROM THE RWST (OR BWST) AND CONTAINMENT SUMP(S) TO THE SAFETY INJECTION PUMPS

A. Background

Current PWRs utilize the refueling water storage tank (RWST) or the borated water storage tank (BWST) as the sole source of water for the safety injection pumps during the first 20 to 40 minutes of any accident that trips a safety injection signal. Since acceptable results of safety analyses of the accidents are based on the operation of a minimum number of these pumps, interruption of this water supply for even a short period of time could result in unacceptably high fuel and cladding temperatures if the safety injection pumps fail because of cavitation or overheating.

General Design Criteria 35 requires that the emergency core cooling system have suitable redundancy in components and features and suitable interconnections to assure the system safety function can be accomplished assuming a single failure. The principal problem appears to be a definition of single failure. A recent draft of ANSI N658, "Single Failure Criteria for PWR Fluid Systems," defines an active failure as:

(a) "An active failure is a malfunction, exceeding passive failures, of a component which relies on mechanical movement to complete its intended function upon demand."

(b) "Spurious action of a powered component originating within its actuation system shall be regarded as an active failure unless specific design featur or operating restrictions preclude such spurious action."

This branch position on the availability of the RWST is based on the above criteria and the recognition that water supplied from the RWST system to the ECCS system is absolutely essential in the event of a LOCA.

B. Branch Position

1. The single active failure criterion defined in (a) and (b) above will be applied in evaluating the design of the piping systems that connect the safety injection pumps to the RWST (BWST) and the containment sumps.

2. The piping systems, including valves, shall be designed to satisfy the requirements listed below without the need to disconnect the power to any valve.

3. The valves and piping between the RWST (or BWST) and the safety injection pumps must be arranged so that no single failure will prevent the minimum flow to the core required to satisfy 10 CFR Part 50, §50.46.

4. The valves and piping between the RWST (or BWST) and safety injection pumps must be arranged so that no single active failure will result in damage to pumps such that the minimum flow requirements for long-term core and containment cooling after a LOCA are not satisfied.

5. The valves and piping that connect the RWST (or BWST) and the containment sumps(s) to the safety injection pumps must be arranged so as not to preclude automatic switchover from the injection mode of ECCS operation to recirculation cooling from the sump. These piping systems must be arranged so that the differential pressure between the sump and the RWST (or BWST), even if there is a single active failure, will not result in a loss of core cooling or a path that permits release of radioactive material from the containment to the environment.

C. Implementation

1. CPs Under Review and Future CP Reviews

The proposed position will be applied to all CP reviews for which an SER was not published prior to April 16, 1975. It is expected that all of the events of the proposed position will be applied for such reviews. Taking this position on CPs would eliminate the need for various schemes such as locking out power to valves located in the line between the various ECCS pumps and refueling water storage tank.

2. OLs Under Review

For operating licenses that are presently under review and OLs to be reviewed in the future that are not covered by item 1, the proposed position will not be completely applied. Specifically, locking out power to valves will be permitted. For most plants it is expected that this will be sufficient to meet the single failure criteria. However, in other plants changes to the piping and valving arrangements may be required to satisfy the single failure criteria.

3. Plants Under Construction

These plants will be handled as discussed in item C.2. It is expected, however, that we will discuss the proposed position with each of the applicable PWR vendors. It will be obvious to the vendors which plants now under construction may have a problem. Then a generic review may be conducted for those plants that have a severe problem.

4. Operating Plants

All of the operating plants are being evaluated as an ongoing part of the current ECC review. The review should be conducted as discussed in item C.2 to assure that these plants meet the essential parts of the proposed position.

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

Section No. 9.2.1
Revision No. 3

Appendix No. N/A
Revision No. N/A

Branch Tech. Position N/A
Revision No. N/A

Date Issued April 1984

FILING INSTRUCTIONS

PAGES TO BE REMOVED			NEW PAGES TO BE INSERTED		
PAGE NUMBER		DATE	PAGE NUMBER		DATE
9.2.1-1 thru 9.2.1-8	Rev. 2	July 1981	9.2.1-1 thru 9.2.1-9	Rev. 3	April 1984

The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

9.2.1 STATION SERVICE WATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - None

I. AREAS OF REVIEW

The service water system (SWS) provides essential cooling to safety-related equipment and may also provide cooling to nonsafety-related auxiliary components that are used for normal plant operation. The ASB reviews the system from the service water pump intake to the points of cooling water discharge to assure conformance with the requirements of General Design Criteria 2, 4, 5, 44, 45, and 46. The ultimate heat sink (reviewed under SRP Section 9.2.5) provides the intake source of water to the SWS for long-term cooling of station features required for plant shutdown and also any special equipment required to prevent or mitigate the consequences of postulated accidents and as such is an interface system to the SWS. The SWS pump performance characteristics will be compared to the high and low water levels of the ultimate heat sink to assure that pumping capability can be provided for extended periods of operation following postulated events.

1. The ASB reviews the characteristics of the SWS components (pumps, heat exchangers, pipes, valves) with respect to their functional performance as affected by adverse operational (i.e., water hammer) and environmental occurrences including cold weather protection, by abnormal operational requirements, and by accident conditions such as a loss-of-coolant accident (LOCA) with the loss of offsite power. Since the SWS normally has requirements that relate to cooling functions during normal plant operation as well as for safety functions, the review will include an evaluation of the capability of the system to perform these multiple functions.

2. The ASB also reviews the design of the SWS with respect to:

a. The capability for detection, control, and isolation of system leakage including the capability for detection and control of radioactive leakage into and out of the system and prevention of accidental releases to the environment.

b. Measures to preclude long-term corrosion and organic fouling that would tend to degrade system performance.

c. Provisions for system and component operational testing, including the instrumentation and control features that determine and verify that the system is operating in a correct mode (i.e., valve position, pressure and temperature indication).

d. The effects of the failure of nonseismic Category I equipment, structures or components of safety-related portions of the SWS are taken into account in the design.

3. The ASB reviews the SWS capability to flood the reactor containment should this be required in a post-accident recovery situation.

4. The ASB reviews the system to determine that a malfunction, a failure of a component, or the loss of a cooling source will not reduce the safety-related functional performance capabilities of the system. Specifically, ASB performs the following reviews under the SRP sections indicated:

a. Review for flood protection is performed under SRP Section 3.4.1.

b. Review of the protection against internally-generated missiles is performed under SRP Section 3.5.1.1.

c. Review of the structures, systems and components to be protected against externally-generated missiles is performed under SRP Section 3.5.2.

d. Review of high and moderate energy pipe breaks is performed under SRP Section 3.6.1.

In addition, the ASB will coordinate other branches evaluations that interface with the overall review of the system as follows: The Reactor Systems Branch (RSB) identifies essential components associated with the reactor coolant system and the emergency core cooling systems that are required for operation during normal operations or accident conditions. The RSB establishes accident cooling load functional requirements and minimum time intervals. The RSB performs these reviews as part of its primary review responsibility for SRP Sections 5.4.7, 5.4.8, 6.0 and 15.0. The Structural and Geotechnical Engineering Branch (SGEB) determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), probable maximum flood (PMF), and tornado missiles as part as its primary review responsibility for SRP Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5. The Mechanical Engineering Branch (MEB) determines that the components, piping and structures are designed in accordance with applicable codes and standards as part of its primary review responsibility for SRP Sections 3.9.1 through 3.9.3. The MEB also determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for SRP Sections 3.2 1 and 3.2.2. The MEB also reviews the adequacy of the inservice testing program of pumps and valves as part of its primary review responsibility for SRP Section 3.9.6. The Materials Engineering Branch (MTEB) verifies that inservice inspection requirements are met for system components as part of its primary review responsibility for SRP Section 6.6 and, upon request, verifies the compatibility of the materials of construction with service conditions. The Instrumentation and Control Systems Branch (ICSB) and Power Systems Branch (PSB) will evaluate the system controls, instrumentation, and power sources with respect to capabilities, capacity, and reliability for supplying power during normal and emergency conditions to safety-related pumps, valves and other components as part of their primary review responsibility for SRP Sections 7.1 and 8.1, respectively. The reviews for Fire Protection, Technical Specifications and Quality Assurance are coordinated and performed by the Chemical Engineering Branch, Standardization and Special Projects Branch and Quality Assurance Branch as part of their primary review responsibility for SRP Sections 9.5.1, 16.0, and 17.0, respectively.

For those areas of review identified above as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections identified as the primary review responsibility of those branches.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the service water system, as described in the applicant's safety analysis report (SAR), including related sections of Chapters 2 and 3 of the SAR is based on specific general design criteria and regulatory guides. Listed below are specific criteria as they relate to the SWS.

The design of the service water system is acceptable if the integrated system design is in accordance with the following criteria:

1. General Design Criterion 2, as related to structures housing the system and the system itself being capable of withstanding the effects of earthquakes. Acceptance is based on meeting the guidance of Regulatory Guide 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions.

2. General Design Criterion 4, as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety being capable of performing required safety functions.

4. General Design Criterion 44, as related to transferring heat from structures systems and components important to safety, to an ultimate heat sink. Acceptance is based on the following:

a. The capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under both normal operating and accident conditions.

b. Component redundancy so that the safety function can be performed assuming a single active component failure coincident with the loss of offsite power.

c. The capability to isolate components, subsystems, or piping if required so that the system safety function will not be compromised.

d. Meeting task action plan item II.K.1-C.1.22 of NUREG-0694 for boiling water reactors regarding automatic and manual actions necessary when the main feedwater system is not operable.

e. Meeting task action plan item II.K.1.22 of NUREG-0718 for B&W plants regarding automatic and manual actions for proper functioning of the auxiliary heat removal systems when the main feedwater system is not operable.

5. General Design Criterion 45, as related to design provisions to permit inservice inspection of safety-related components and equipment.

6. General Design Criterion 46, as related to design provisions to permit operational functional testing of safety-related systems and components.

III. <u>REVIEW PROCEDURES</u>

The procedures set forth below are used during the construction permit (CP) application review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II. For review of operating license (OL) applications, the review procedures and acceptance criteria are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report.

Upon request from the primary reviewer, the coordinating review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

As a result of the various SWS designs provided, there will be variations in system requirements. For the purpose of this SRP section, a typical system is assumed which has fully redundant systems, with each of the systems having an identical essential (safety features) portion and an identical non-essential portion (used for normal operation). For cases where there are variations from the typical arrangement, the reviewer will adjust the review procedures given below. However, the system design will be required to meet the acceptance criteria given in subsection II. Also, the reviewer will need to refer to SRP sections for other systems that would interface with the SWS, depending upon the nature and conditions of the ultimate heat sink cooling water (e.g., salt water).

1. The SAR is reviewed to determine that the system description and piping and instrumentation diagrams (P&IDs) show the SWS equipment that is used for normal operation, and the minimum system heat transfer and flow requirements for normal plant operation. The system performance requirements will also be reviewed to determine that it describes component allowable operational degradation (e.g., pump leakage) and describes the procedures that will be followed to detect and correct these conditions when they become excessive.

2. The reviewer, using the results of failure modes and effects analyses as appropriate, comparisons with previously approved systems, or independent calculations, determines that the system is capable of sustaining the loss of any active component and meeting minimum system requirements (cooling load and flow) for the degraded conditions. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed for the following points:

    a. Essential portions of the SWS are correctly identified and are isolable from the non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical division between each portion and indicate the required classification changes. System drawings are also reviewed to see that they show the means for accomplishing isolation and the system description is reviewed to identify minimum performance requirements for the isolation valves. The drawings and descriptions are reviewed to verify that automatically operated isolation valves separate non-essential portions and components from the essential portions.

    b. Essential portions of the SWS, including the isolation valves separating essential and non-essential portions, are classified Quality Group C and seismic Category I. Components and system descriptions in the SAR that identify mechanical and performance characteristics are reviewed to verify that the above seismic and safety classifications have been included, and that the P&IDs indicate any points of change in piping quality group classification.

    c. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It will be acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around pumps or isolation valves that would be required by this program.

    d. The review of seismic design is performed by SGEB and the review for seismic and quality group classification is performed by MEB as indicated in subsection I of this SRP section.

3. The reviewer determines that the safety function of the system will be maintained, as required, in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. The reviewer uses engineering judgment, the results of a failure mode and effects analyses, and the results of reviews performed under other SRP sections to verify the following:

a. The failure of portions of the system or of other systems not designed to seismic Category I and located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the SWS, will not preclude operation of the essential portions of the SWS. Reference to SAR Chapter 2 describing site features and the general arrangement and layout drawings will be necessary as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable. (CP)

b. The essential portions of the SWS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the Section 3 series of the SRP. The reviewer will utilize the procedures identified in these SRP sections to assure that the analyses presented are valid. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable. The location and the design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate.

c. The SWS pumps will have sufficient available net positive suction head (NPSH) at the pump suction locations, considering low water levels. Reference to SRP Section 2.4, which indicates the lowest probable water level of the heat sink, and to drawings indicating the elevation of service water pump impellers will be necessary. An independent calculation verifying the applicant's conclusion will be necessary for acceptance.

d. Provisions are made in the system to detect and control leakage of radioactive contamination into and out of the system. It will be acceptable if the system P&IDs show radiation monitors located on the system discharge and at components susceptible to leakage, and these components can be isolated by one automatic and one manual valve in series.

e. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the SWS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and the procedures for reviewing this information are given in the corresponding SRP sections.

f. Essential components and subsystems necessary for safe shutdown can function as required in the event of loss of offsite power. The system design will be acceptable if the SWS meets minimum system requirements as stated in the SAR assuming a concurrent failure of a single active component, including a single failure of an auxiliary electric power source. The SAR is reviewed to determine that for each SWS component or subsystem affected by the loss of offsite power, system flow and heat transfer capability meet or exceed minimum requirements. The results of failure modes and effects analyses are considered in assuring that the system meets these requirements. This will be an acceptable verification of system functional reliability.

g. Provisions are made for protection of the essential service water supply from potential failures or malfunctions caused by freezing, icing, and other adverse environmental conditions. Statements in the SAR that would indicate that safety grade heating sources will be used for this purpose, considering the equipment necessary for safe shutdown, will be acceptable.

4. The descriptive information, P&IDs, SWS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system can function following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the failure mode and effects analysis presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system flow and heat transfer requirements are met for each accident situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

5. The SAR is reviewed to assure that the applicant has described all the automatic and manual actions necessary for proper functioning of the service water system when the main feedwater system is not operable. The design will be acceptable in this regard if sufficient detail is presented to provide reasonable assurance that the requirements of items II.K.1.22 of NUREG-0718 and II.K.1-C.1.22 of NUREG-0694 are property implemented.

6. The SAR is reviewed to assure that the applicant has committed to address the potential for water hammer in open loop systems and will provide for venting and filling of such systems, operating procedures for avoidance of water hammer, and that the system is designed to maintain functioning following an inadvertent water hammer occurrence.

IV. EVALUATION FINDINGS

The reviewer determines that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The service water system (SWS) includes all components and piping from the SWS pump intake to the points of cooling water discharge. Portions of the SWS that are necessary for safe shutdown accident prevention, or accident mitigation are designed to seismic Category I, Quality Group C requirements. Based on the review of the applicant's proposed design criteria, design bases and safety classification for the service water system regarding the requirements for continuous cooling of safety-related components necessary for a safe plant shutdown, the staff concludes that the design of the service water system is acceptable and meets the

requirements of General Design Criteria 2, 4, 5, 44, 45, and 46. This conclusion is based on the following:

1. The applicant has met the requirements of General Design Criterion 2 with respect to safety-related portions of the system being capable of withstanding the effects of earthquakes. Acceptance is based on meeting Regulatory Guide 1.29 position C.1 for the safety-related portions and position C.2 for the nonsafety-related portions.

2. The applicant has met the requirements of GDC 4 with respect to dynamic effects associated with flow instabilities (i.e., water hammer loads) with respect to impairment of the required service water systems during normal plant operations, and under upset or accident conditions. Acceptance is based on the following:

   a. Vents shall be provided for venting of components and piping at high points in liquid filled, but normally idle piping (or systems) where voiding can occur. These vents should be designed for ease of operational testing on a periodic basis.

   b. Consideration will be given to voiding which can occur following pump shutdown, or during standby. If the system design is such that voiding could occur, means should be provided for a slow system fill upon pump start for avoidance of water hammer or that the system be designed to maintain functioning following an inadvertent water hammer occurrence.

   c. Operating and maintenance procedures will be reviewed by the applicant to assure that sufficient measures have been taken for avoiding water hammer (e.g., rapid fill due to pump start, periodic fill and vent checks, avoidance of sudden valve movement, or realignment).

3. The applicant has met the requirements of General Design Criterion 5 with respect to sharing of structures, systems and components by demonstrating that such sharing does not significantly impair the ability of the service water system to perform its safety function, including in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

4. The applicant has met the requirements of General Design Criterion 44 with respect to cooling water by providing a system to transfer heat from structures, systems and components important to safety to an ultimate heat sink. The applicant has demonstrated that the service water system can transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions assuming loss of offsite power and a single failure and that portions of the system can be isolated so that the safety function of the system will not be compromised. The applicant has also met task action plan items II.K.1-C.1.22 of NUREG-0694 and II.K.1.22 of NUREG 0718 in meeting General Design Criterion 4.

5. The applicant has met the requirements of General Design Criterion 45 with respect to inspection of cooling water systems by providing a

service water system design which permits inservice inspection of safety-related components and equipment.

6. The applicant has met the requirements of General Design Criterion 45 with respect to testing of cooling water systems by providing a service water system design which permits operational functional testing of the system and its components.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's Regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission Regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced Regulatory Guide, NUREGs and implementation of acceptance criterion subsection II.2 is as follows:

(a). Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

3. 10 CFR Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

4. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

5. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water Systems."

6. Regulatory Guide 1.29, "Seismic Design Classification."

7. NUREG-0694, "TMI-Related Requirements for New Operating Licenses."

8. NUREG-0718, "Proposed Licensing Requirements for Pending CP's and Manufacturing License."

9. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

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The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

9.2.2 REACTOR AUXILIARY COOLING WATER SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - None

I. AREAS OF REVIEW

The ASB reviews reactor auxiliary cooling water systems (CWS) that are required for safe shutdown during normal, operational transient, and accident conditions and for mitigating the consequences of an accident or preventing the occurrence of an accident. These include closed loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system (ECCS).

The review of these systems includes components of the system, valves and piping, and points of connection or interfaces with other systems. Emphasis is placed on the CWS for safety-related components such as ECCS equipment, ventilation equipment, and reactor shutdown equipment. The ASB reviews reactor auxiliary cooling water systems to ensure conformance with the requirements of General Design Criteria 2, 4, 5, 44, 45, and 46.

1. The ASB reviews the capability of the auxiliary cooling systems to provide adequate cooling water to safety-related ECCS components and reactor auxiliary equipment for all planned operating conditions. The review includes the following points:

   a. The functional performance requirements of the system including the ability to withstand adverse operational (i.e. water hammer) and environmental occurrences, operability requirements for normal operation, and requirements for operation during and subsequent to postulated accidents.

b. Multiple performance functions (if required) assigned to the system and the necessity of each function for emergency core cooling and safe shutdown.

c. The capability of the system surge tank to perform its intended function.

d. The capability of the system to provide adequate cooling water during all operating conditions.

e. The sizing of the system for core cooling and decay heat loads and the associated design margin.

2. Other system aspects that are reviewed include:

a. The effects of non-seismic Category I component failures on the seismic Category I portion of the system.

b. The provisions for detection, collection, and control of system leakage and the means provided to detect leakage of activity from one system to another and preclude its release to the environment.

c. The requirements for operational testing and inservice inspection of the system.

d. The capability of the system to provide adequate cooling to the seals and bearings of all reactor coolant pumps.

e. Instrumentation and control features necessary to accomplish design functions, including isolation of components to deal with leakage or malfunctions and actuation requirements for redundant equipment.

f. A simplified reliability analyses using event-tree and fault-tree logic techniques.

3. ASB also performs the following reviews under the SRP sections indicated:

a. Review of flood protection is performed under SRP Section 3.4.1,

b. Review of the protection against internally-generated missiles is performed under SRP Section 3.5.1.1,

c. Review of the protection of structures, systems and components against the effects of externally-generated missiles is performed under SRP Sections 3.5.1.4 and 3.5.2, and

d. Review of high and moderate energy pipe breaks is performed under SRP Section 3.6.1.

In addition, the ASB will coordinate other branches evaluations that interface with the overall review of the system as follows. The Reactor Systems Branch (RSB) will identify engineered safety feature components associated with the reactor coolant system and the emergency core cooling systems that are required for operation during normal operations, transients, and accident conditions.

RSB will establish cooling load functional requirements and minimum time intervals associated with safety-related components. The RSB performs these reviews as part of its primary review responsibility for SRP Sections 5.4.7, 5.4.8, 6.0, and 15.0. The Structural and Geotechnical Engineering Branch (SGEB) will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of Category I structures that house the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles as part of its primary review responsibility for SRP Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1, 3.7.4, 3.8.4 and 3.8.5. The Mechanical Engineering Branch (MEB) determines that the components, piping and structures are designed in accordance with applicable codes and standards as part of its primary review responsibility for SRP Sections 3.9.1 and 3.9.3. The MEB also determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for SRE Sections 3.2.1 and 3.2.2. The MEB also reviews the adequacy of the inservice testing program of pumps and valves as part of its primary review responsibility for SRP Section 3.9.6. The Material Engineering Branch (MTEB) verifies that inservice inspection requirements are met for system components as part of its primary review responsibility for SRP Section 6.6 and, upon request, verifies the compatibility of the materials of construction with service conditions. The Instrumentation and Control Systems Branch (ICSB) and Power Systems Branch (PSB) will determine the adequacy of the design, installation, inspection, and testing of all essential electrical components, system controls, and instrumentation required for proper operation as part of their primary review responsibilities for SRP Sections 7.1 and 8.1, respectively. The review for Fire Protection, Technical Specifications, and Quality Assurance are coordinated and performed by the Chemical Engineering Branch (CMEB), Standardization and Special Projects Branch (SSPB) and Quality Assurance Branch (QAB) as part of their primary review responsibility for SRP Sections 9.5.1, 16.0, and 17.0, respectively.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

Acceptability of the designs of cooling water systems as described in the applicant's Safety Analysis Report (SAR), including related sections of Chapters 2 and 3 of the SAR, is based on specific general design criteria and regulatory guides, and on independent calculations and staff judgments with respect to system functions and component selection. The design of a CWS is acceptable if the integrated system design is in accordance with the following requirements and recommendations:

1. General Design Criterion 2, as related to structures housing the system and the system itself being capable of withstanding the effects of earthquakes. Acceptance is based on meeting the guidance of Regulatory Guide 1.29, Position C.1 for safety-related portions and Position C.2 for non-safety-related portions.

2. General Design Criterion 4, as related to dynamic effects associated with flow instabilities and attendant loads (i.e., water hammer) during normal plant operation as well as during upset or accident conditions.

3. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

4. General Design Criterion 44, as its relates to:

   a. The capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under both normal operating and accident conditions.

   b. Component redundancy so that safety functions can be performed assuming a single active component failure coincident with the loss of offsite power.

   c. The capability to isolate components, systems, or piping, if required, so that the system safety function will not be compromised.

   d. Task Action Plan items II.K.2.16 and II.K.3.25 of NUREGs-0718 and 0737 as they related to loss of cooling water to reactor coolant pump (RCP) seals.

   e. A single failure in the CWS does not result in fuel damage or reactor coolant leakage in excess of normal coolant-makeup capability. Single failure includes but is not limited to operator error, spurious activation of a valve operator, and loss of a cooling water pump.

   A moderate-energy leakage crack or an accident that is initiated from a failure in the CWS piping does not result in excessive fuel damage or reactor coolant leakage in excess of normal coolant-makeup capability. A single active failure is considered when evaluating the consequences of this accident. Moderate leakage cracks are determined in accordance with the guidelines of Branch Technical Position ASB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment."

   It has been demonstrated by testing that the reactor coolant pumps will withstand a complete loss of cooling water for 20 minutes, and instrumentation in accordance with IEEE 279 that alarms in the control room is provided to detect a loss of cooling water to ensure a period of 20 minutes is available so that the operator would have sufficient time to initiate manual protection of the plant. Alternatively, if it is not demonstrated by the necessary pump testing that the reactor coolant pumps will operate for 20 minutes without operator corrective action:

   1. Instrumentation in accordance with IEEE 279 is provided consistent with the criteria for the protection system to initiate automatic protection of the plant upon loss of cooling water to a pump. For this case, the component cooling water supply to

the seal and bearing of the pump may be designed to nonseismic Category I requirements and Quality Group D, or

2. The component cooling water supply to each pump is designed to be capable of withstanding a single active failure or a moderate-energy line crack as defined in Branch Technical Position ASB 3-1 and to seismic Category I, Quality Group C, and ASME Section III Class 3 requirements.

5. General Design Criterion 45, as related to the design provisions to permit inservice inspection of safety-related components and equipment.

6. General Design Criterion 46, as related to the design provisions to permit operational functional testing of safety-related systems or components to ensure:

   a. Structural integrity and system leak tightness.

   b. Operability and adequate performance of active system components.

   c. Capability of the integrated system to perform required functions during normal, shutdown, and accident situations.

III. REVIEW PROCEDURES

The procedures set forth below are used during the construction permit (CP) application review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II of this SRP section. For the review of operating license (OL) applications, the review procedures and acceptance criteria given in subsection II will be used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report.

One of the main objectives in the review of a CWS is to determine its function with regard to safety. Some cooling systems are designed as safety-related systems in their entirety, others have only portions of the system that are safety-related, and others are classified as nonsafety-related because they do not perform any safety function. To determine the safety category of a CWS, the ASB will evaluate its necessity for achieving safe reactor shutdown conditions or for accident prevention or accident mitigation functions. The safety functions to be performed by these systems in all designs are essentially the same, however, the method used varies from plant to plant depending upon the individual designer.

Upon request from the primary reviewer, the coordinating review branches will provide input for the areas of review stated in subsection I of this SRP section. The primary reviewer obtains and uses such input as required to ensure that this review procedure is complete.

In view of the various designs provided, the procedures set forth below are for a typical CWS designed entirely as a safety-related system. Any variance of the review procedures to take account of a proposed unique design will be such as to ensure that the system meets the criteria of subsection II. The reviewer will select and emphasize material from this SRP section, as may be appropriate for a particular case.

1. The information provided in the SAR pertaining to the design bases and design criteria, and the system description section are reviewed to verify that the equipment used and the minimum system heat transfer and flow requirements for normal plant operations are identified. A review of the system piping and instrumentation diagrams (P&IDs) will show which components of the system are used to:

   a. Remove heat from the reactor primary coolant system necessary to achieve a safe reactor shutdown.

   b. Provide essential cooling for containment components or systems such as the sprays, ventilation coolers, or sump equipment.

   c. Provide cooling for decay heat removal equipment.

   d. Provide cooling for emergency core cooling pump bearings or other emergency core cooling equipment necessary to prevent or mitigate the consequences of an accident.

2. The system performance requirements section is reviewed to determine that it describes allowable component operational degradation (e.g., pump leakage) and describes the procedures that will be followed to detect and correct these conditions when degradation becomes excessive.

3. The reviewer, using the results of failure-modes and -effects analyses, determines that the system is capable of sustaining the loss of any active component and, on the basis of previously approved systems or independent calculations, that the minimum system requirements (cooling load and flow) are met for these failure conditions. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed for the following points:

   a. Essential portions of the CWS are correctly identified and are isolable from the nonessential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical division between each portion and indicate required classification changes. System drawings are reviewed to see that they show the means for accomplishing isolation and the SAR description is reviewed to identify minimum performance of the isolation valves. The drawings and description are reviewed to verify that automatically operated isolation valves separate nonessential portions and components from the essential portions.

   b. Essential portions of the CWS, including the isolation valves separating seismic Category I portions from the nonseismic portions, are Quality Group C and seismic Category I. System design bases and criteria, and the component classification tables are reviewed to verify that the heat exchangers, pumps, valves, and piping of essential portions of the system will be designed to seismic Category I requirements in accordance with the applicable criteria. The review of seismic design is performed by SGEB and the review for seismic and quality group classification is performed by MEB as indicated in subsection I of this SRP section.

c. The system is designed to provide water makeup as necessary. Cooling water systems that are closed loop systems are reviewed to ensure that the surge tanks have sufficient capacity to accommodate expected leakage from the system for seven days or that a seismic source of makeup can be made available within a time frame consistent with the surge tank capacity (time zero starts at low level alarm). The surge tank and connecting piping are reviewed to ensure that makeup water can be supplied to either header in a split header system. Redundant surge tanks (one to each header) or a divided surge tank design are acceptable to ensure that in the event of a header rupture, the loss of the entire contents of the surge tank will not occur.

d. The system is designed for removal of heat loads during normal operation and of emergency core cooling heat loads during accident conditions, with appropriate design margins to ensure adequate operation. A comparative analysis is made of the system flow rates, heat levels, maximum temperature, and heat removal capabilities with similar designs previously found acceptable. To verify performance characteristics of the system, an independent analysis may be made.

e. Design provisions are made that permit appropriate inservice inspection and functional testing of system components important to safety. The applicant should ensure that the SAR information delineates a testing and inspection program and the system drawings show the necessary test recirculation loops around pumps or isolation valves necessary for this program.

f. Essential portions of the system are protected from the effects of high-energy and moderate-energy line breaks. The system description and layout drawings will be reviewed to ensure that no high- or moderate-energy piping systems are close to essential portions of the CWS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR, and the procedures for reviewing this information are given in the corresponding SRP sections.

g. Essential components and subsystems (i.e., those necessary for safe shutdown) can function as required in the event of a loss of offsite power and instrument air systems. The system design will be acceptable in this regard if the essential portions of the CWS meet minimum system requirements as stated in the SAR assuming a concurrent failure of a single active component, including a single failure of any auxiliary electric power source. The SAR is reviewed to determine that for each CWS component or subsystem affected by the loss of offsite power or instrument air systems, system flow and heat transfer capability exceed minimum requirements. The results of failure-modes and -effects analyses are considered in ensuring that the system meets these requirements. This will be an acceptable verification of system functional reliability. The effects of loss of cooling water to RCP seals as a result of loss of power will be reviewed as indicated in Task Action Plan items II.K.2.16 and II.K.3.25 of NUREGs-0718 and 0737.

4. The system design information and drawings are analyzed to ensure that the following features will be incorporated.

   a. A leakage detection system is provided to detect component or system leakage. An adequate means for implementing this criterion is to provide sumps or drains with adequate capacity and appropriate alarms in the immediate area of the system.

   b. Components and headers of the system are designed to provide individual isolation capabilities to ensure system function, control system leakage, and allow system maintenance.

   c. Design provisions are made to ensure the capability to detect leakage of radioactivity or chemical contamination from one system to another. Radioactivity monitors and conductivity monitors should be located in the system component discharge lines to detect leakage. An alternative means is to prevent leakage from occurring by operating the system at higher pressure to ensure that leakage is in the preferred direction.

   d. The system is designed to provide cooling to the reactor coolant pump seals and bearings during normal plant operating conditions, anticipated transients, and following postulated accidents. Instrumentation in accordance with IEEE 279 with alarms in the control room should be provided to detect a loss of cooling water in order to ensure that a period of 20 minutes is available to the operator to initiate manual protection of the plant, if necessary. It has been demonstrated by testing that the reactor coolant pumps could potentially operate with loss of cooling water for 20 minutes without the need for operator action.

      As an alternative to pump testing, the reviewer verifies that:

      (1) Instrumentation in accordance with IEEE 279 is provided consistent with the criteria for the protection system to initiate automatic protection of the plant upon loss of water to a pump. For this case, the component cooling water supply to the seal and bearing of the pump may be designed to nonseismic Category I requirements and Quality Group D, or

      (2) The component cooling water supply to each pump is designed to be capable of withstanding a single active failure or a moderate-energy line crack as defined in Branch Technical Position ASB 3-1 and to seismic Category I, Quality Group C, and ASME Section III, Class 3 requirements.

5. The reviewer verifies that the system has been designed so that system functions will be maintained as required in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods. The reviewer evaluates the system using engineering judgment and the results of failure-modes and -effects analyses to determine the following:

a. The failure of portions of the system or of other systems not designed to seismic Category I standards and located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the CWS, will not preclude essential functions. The review will identify these nonseismic category components or piping and ensure that appropriate criteria are incorporated to provide isolation capabilities in the event of failure. Reference to SAR Chapter 2, describing site features, and the general arrangement and layout drawings will be necessary as well as the SAR tabulation of seismic design classifications for structures and systems.

b. The essential portions of the CWS are protected from the effects of floods, hurricanes, tornadoes, and internally- or externally-generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the SRP sections for Chapter 3 of the SAR. The reviewer will use the procedures identified in these SRP sections to ensure that the analyses presented are valid. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable. The location and design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate.

6. The descriptive information, P&IDs, CWS drawings, and failure-modes and -effects analyses in the SAR are reviewed to ensure that essential portions of the system will function following design basis accidents assuming a concurrent single, active component failure. The reviewer evaluates the information presented in the SAR to determine the ability of required components to function, traces the availability of these components on system drawings, and checks that the SAR information contains verification that minimum system flow and heat transfer requirements are met for each accident situation for the required time spans. For each case, the design will be acceptable if minimum system requirements are met.

7. The SAR is reviewed to assure that the applicant has committed to address the potential for water hammer in the auxiliary cooling water systems and will provide means for prevention, or avoidance, such as venting and filling capability and operating procedures for avoidance of water hammer.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type to be included in the staff's safety evaluation report:

The reactor auxiliary cooling water systems include pumps, heat exchangers, valves and piping, expansion tanks, makeup piping, and the points of connection or interfaces with other systems. Portions of the reactor auxiliary cooling water systems that are necessary for safe shutdown, accident prevention or accident mitigation are designed to seismic Category I and Quality Group C requirements. Based on the review of the applicant's proposed design criteria, design bases, and safety classification for the reactor auxiliary cooling water systems with regard to the requirements for providing adequate cooling water for the safety-related ECCS components and reactor auxiliary equipment for all conditions of plant operation, the staff concludes that the design of the reactor auxiliary cooling water systems is acceptable and meets the requirements of General Design Criteria 2, 4, 5, 44, 45, and 46. This conclusion is based on the following:

1. The applicant has met the requirements of General Design Criterion 2 with respect to safety-related portions of the systems being capable of withstanding the effects of earthquakes. Acceptance is based on meeting Regulatory Guide 1.29, Position C.1 for the safety-related portions and position C.2 for the nonsafety-related portions.

2. The applicant has met the requirements of GDC 4 with respect to dynamic effects associated with flow instabilities and attendant loads (i.e., water hammer) with respect to impairment of the required functions of auxiliary cooling systems during normal plant operations, and under upset or accident conditions. Acceptance will be based on the following commitments by the applicant:

   a. Vents shall be provided for venting components and piping at high points in liquid filled systems which is normally idle and in which voids could occur. These vents should be located for ease of operation and testing on a periodic basis.

   b. Operating and maintenance procedures shall be reviewed by the applicant to assure that adequate measures are taken to avoid water hammer due to voided line conditions.

3. The applicant has met the requirements of General Design Criterion 5 with respect to sharing of structures, systems and components by demonstrating that such sharing does not significantly impair the ability of the reactor auxiliary cooling water systems to perform their safety function, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

4. The applicant has met the requirements of General Design Criterion 44 with respect to cooling water by providing a system to transfer heat from structures, systems and components important to safety to an ultimate heat sink. The applicant has demonstrated that the reactor auxiliary cooling water systems can transfer the combined heat load of these structures, systems and components under normal operating and accident conditions assuming loss of offsite power and a single failure, and that portions of the system can be isolated so that the safety function of the system will not be compromised.

5. The applicant has met the requirements of General Design Criterion 45 with respect to inspection of cooling water systems by providing reactor auxiliary cooling water systems design features which permit inservice inspection of safety-related components and equipment.

6. The applicant has met the requirements of General Design Criterion 46 with respect to testing of cooling water systems by providing reactor auxiliary cooling water systems design features which permit operational functional testing of the system and its components.

7. Also in meeting the requirements of General Design Criterion 44, the applicant has demonstrated that the system can withstand a loss of power without damage to RCP seals in accordance with items II.K.2.16 and II.K.3.25 of NUREGs-0718 and 0737.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's Regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission Regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced Regulatory Guide, NUREGs and implementation of acceptance criterion subsection II.2 is as follows:

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR Part 50.

2. General Design Criterion 5, "Sharing of Structures, Systems, and Components," of Appendix A to 10 CFR Part 50.

3. General Design Criterion 44, "Cooling Water," of Appendix A to 10 CFR Part 50.

4. General Design Criterion 45, "Inspection of Cooling Water System," of Appendix A to 10 CFR Part 50.

5. General Design Criterion 46, "Testing of Cooling Water System," of Appendix A to 10 CFR Part 50.

6. Regulatory Guide 1.29, "Seismic Design Classification."

7. NUREG-0718 "Proposed Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."

8. NUREG-0737 "Clarification of TMI Action Plan Requirements."

9. General Design Criterion 4, "Environmental and Missile Design Basis."

U.S. NUCLEAR REGULATORY COMMISSION

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U.S. NUCLEAR REGULATORY COMMISSION
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10.3 MAIN STEAM SUPPLY SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)
Power Systems Branch (PSB)

Secondary - None

I. AREAS OF REVIEW

The main steam supply system (MSSS) for both boiling water reactor (BWR) and pressurized water reactor (PWR) plants transports steam from the nuclear steam supply system to the power conversion system and various safety-related or non-safety-related auxiliaries. Portions of the MSSS may be used as a part of the heat sink to remove heat from the reactor facility during certain operations and may also be used to supply steam to drive engineered safety feature pumps. The MSSS may also include provisions for secondary system pressure relief in PWR plants.

The MSSS for the BWR direct cycle plant extends from the outermost containment isolation valves up to and including the turbine stop valves, and includes connected piping of 2-1/2 inches nominal diameter and larger up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of reactor operation. The MSSS for the PWR indirect cycle plant extends from the connections to the secondary sides of the steam generators up to and including the turbine stop valves, and includes the containment isolation valves, safety and relief valves, connected piping of 2-1/2 inches nominal diameter and larger up to and including the first valve that is either normally closed or capable of automatic closure during all modes of operation and the steam line to the auxiliary feedwater pump turbine. The ASB is responsible for the review of the MSSS from the containment up to and including the outermost isolation valve. The PSB is responsible for the review of the remainder of the MSSS. (The turbine stop valve review is included in SRP Section 10.2.) The PSB also determines the adequacy of the design, installation, inspection, and testing of the electrical power supplies for essential components required for proper operation of the MSSS. The design of the MSSS must be in accordance with General Design Criteria 2, 4, 5, and 34.

1. The ASB and PSB review the MSSS to determine which, if any, portions of the system are essential for safe shutdown of the reactor or for preventing or mitigating the consequences of accidents. The system is reviewed to verify that:

   a. A single malfunction or failure of an active component would not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.

   b. Appropriate quality group and seismic design classification are met for safety-related portions of the system.

   c. Failures of nonseismic Category I equipment or structures, or pipe cracks or breaks in high- and moderate-energy piping will not preclude essential functions of safety-related portions of the system.

   d. The system is capable of performing multiple functions such as transporting steam to the power conversion system, providing heat sink capacity or pressure relief capability, or supplying steam to drive safety system pumps (e.g., turbine-driven auxiliary feedwater pumps), as may be specified for a particular design.

   e. The design of the MSSS includes the capability to operate the atmospheric dump valves remotely from the control room following a safe shutdown earthquake coincident with the loss of offsite power so that a cold shutdown can be achieved with dependence upon safety-grade components only.

   f. The system design capability can withstand adverse dynamic loads, such as steam hammer resulting from rapid valve closure and relief valve fluid discharge loads.

2. The ASB reviews the MSSS with regard to measures provided to limit blowdown of the system in the event of a steam line break.

3. The ASB and PSB also review the design of the MSSS with respect to the following:

   a. The functional capability of the system to transport steam from the nuclear steam supply system as required during all operating conditions.

   b. The capability to detect and control system leakage, and to isolate portions of the system in case of excessive leakage or component malfunctions.

   c. The capability to preclude accidental releases to the environment.

   d. Provisions for functional testing for safety-related portions of the system.

4. ASB also performs the following reviews under the SRP sections indicated:

a. Review for flood protection is performed under SRP Section 3.4.1.

b. Review of the protection against internally generated missiles is performed under SRP Section 3.5.1.1.

c. Review of the structures, systems, and components to be protected against externally generated missiles is performed under SRP Section 3.5.2.

d. Review of high- and moderate-energy pipe breaks is performed under SRP Section 3.6.1.

In the review of the main steam supply system, the ASB and PSB will coordinate other branches' evaluations that interface with the overall review of the system as follows: The Reactor Systems Branch (RSB) identifies essential components associated with the portion of the MSSS inside the primary containment that are required for normal operations and accident conditions, establishes shutdown cooling load requirements versus time, and verifies the design transient used in establishing the flow capacity and setpoint(s) of steam generator relief and safety valves as part of its primary review responsibility for SRP Section 5.2. The Structural and Geotechnical Engineering Branch (SGEB) determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles as part of its primary review responsibility for SRP Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5. The Equipment Qualification Branch (EQB) reviews the seismic and environmental qualification of components under SRP Sections 3.10 and 3.11. The Mechanical Engineering Branch (MEB) determines that the components, piping, and supports are designed in accordance with applicable codes and standards as part of its primary review responsibility for SRP Sections 3.9.1 through 3.9.3. The MEB determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2. The MEB also reviews the adequacy of the inservice testing program of the system valves as part of its primary review responsibility for SRP Section 3.9.6. The Materials Engineering Branch (MTEB) verifies, upon request, the compatibility of the materials of construction with service conditions. The Instrumentation and Control Systms Branch (ICSB) reviews portions of the MSSS with respect to the adequacy of design, installation, inspection, and testing of essential components necessary for instrumentation and control functions as part of its primary review responsibility for SRP Sections 7.1, 7.4, 7.5, and 7.7. The Procedures and Systems Review Branch (PSRB) determines the acceptability of the preoperational and startup tests as part of its primary review responsibility for SRP Section 14.0. The reviews for fire protection, technical specifications, and quality assurance are coordinated and performed by the Chemical Engineering Branch, Standardization and Special Projects Branch (SSPB), and Quality Assurance Branch as part of their primary review responsibility for SRP Sections 9.5.1, 16.0, and 17.0, respectively.

For those areas of review identified above as being part of the primary review responsiblity of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP sections of the corresponding primary branches.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the MSSS, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides.

The design of the MSSS is acceptable if the integrated design of the system is in accordance with the following criteria:

1. General Design Criterion 2, as related to safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, and the positions of the following:

   a. Regulatory Guide 1.29, as related to the seismic design classification of system components, Positions C.1.a, C.1.e, C.1.f, C.2, and C.3.

   b. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles, Appendix Positions 2 and 4.

2. General Design Criterion 4, with respect to safety-related portions of the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks, and the position of Regulatory Guide 1.115 as related to the protection of structures, systems, and components important to safety from the effects of turbine missiles, Position C.1.

   The system design should adequately consider steam hammer and relief valve discharge loads to assure that system safety functions can be achieved and should assure that operating and maintenance procedures include adequate precautions to avoid steam hammer and relief valve discharge loads. The system design should also include protection against water entrainment.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety to perform required safety functions.

4. General Design Criterion 34, as related to the system function of transferring residual and sensible heat from the reactor system in indirect cycle plants, and the following:

   a. The positions in Branch Technical Position RSB 5-1 as related to the design requirements for residual heat removal.

   b. Issue Number 1 of NUREG-0138 as related to credit being taken for all valves downstream of the main steam isolation valves (MSIV) to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSIV.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II of this SRP section. For review of operating license (OL) applications, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report.

The procedures for OL applications include a determination that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements for system testing, minimum performance, and surveillance, developed as a result of the SSPB review, as indicated in subsection I of this SRP section.

The primary reviewers, will coordinate this review with the other branches' areas of review as stated in subsection I of this SRP section. The primary reviewers obtain and use such input as required to assure that this review procedure is complete.

The review procedures below are written for typical MSSSs for both direct and indirect cycle plants. The reviewer will select and emphasize material from this SRP section, as may be appropriate for a particular case.

1. There are significant differences in the design of the MSSS for an indirect cycle (PWR) plant as compared to that for a direct cycle (BWR) plant. Further, different portions of the MSSS are safety-related in different plant designs, although the safety functions of the system are much the same in all PWR plants, and also in all BWR plants. The first step in the review of the MSSS, then, is to determine which portions are designed to perform a safety function. For this purpose, the system is evaluated to determine the components and subsystems necessary for achieving safe reactor shutdown in all conditions or for performing accident prevention or mitigation functions.

2. The reviewer determines that essential (safety-related) portions of the MSSS are correctly identified and are isolable to the extent required from nonessential portions of the system. The system description and piping and instrumentation diagrams (P&IDs) are reviewed to verify that they clearly indicate the physical division between each portion. System arrangement drawings are reviewed to identify the means provided for accomplishing system isolation.

3. The SGEB reviews the seismic design bases and MEB reviews the quality and seismic classification as indicated in subsection I of this SRP section. The SAR is reviewed by ASB and PSB to verify that essential portions of the MSSS are designed to Quality Group B and/or seismic Category I requirements, and to verify that the design classifications specified meet the acceptance criteria specified in subsection II of this SRP section. In general:

   a. The main steam lines from the steam generators to the containment isolation valves in PWR plants are classified seismic Category I and Quality Group B.

   b. The main steam lines in BWR plants extending from the outermost containment isolation valve and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operations but not including the turbine stop and bypass valves are classified seismic Category I and a quality group classification in accordance with BTP RSB 3-1.

Alternatively, for BWRs containing a shutoff valve (in addition to the two containment isolation valves) in the MSSS, seismic Category I and a quality group classification in accordance with BTP RSB 3-2 should be applied to that portion of the MSSS extending from the outermost containment isolation valves up to and including the shutoff valve.

4. The SAR is reviewed to assure that design provisions have been made to permit appropriate functional testing of system components important to safety. It is acceptable if the SAR delineates a testing and inspection program and the system drawings show any test recirculation loops or special connections around isolation valves that would be required by this program.

5. The system description, safety evaluation, component table, and P&IDs are reviewed to verify that the system has been designed to:

   a. Provide the necessary quantity of steam to any turbine-driven safety system pumps. The reviewer verifies that the design is capable of providing the required steam flow to the turbine so that an adequate supply of water can be pumped. (OL)

   b. Assure safe plant operation by including appropriate design margins for pressure relief capacity and setpoints for the secondary system, and for removal of decay heat during various accident conditions, as may be applicable in a particular case. The review is done on a case-by-case basis, and system acceptability is based on a comparison of system flow rates, heat loads, maximum temperatures, and heat removal capabilities to those of similarly designed systems for previously reviewed plants. For PWRs the design is reviewed to verify system capability for controlled cooldown to about 350°F to allow actuation of RHR system.

   c. Provide leakage detection means for steam leakage from the system in the event of a steam line break. Temperature or pressure sensors are acceptable means for initiating signals to close the main steam line isolation valves and/or turbine stop valves to limit the release of steam during a steam line break accident.

   d. Assure that in the event of a postulated break in a main steam line in a PWR plant, the design will preclude the blowdown of more than one steam generator, assuming a concurrent single active component failure. In this regard, all main steam shut-off valves downstream of the MSIVs, the turbine stop valves, and the control valves are considered to be functional. The reviewer should verify that the main steam isolation valves, shut-off valves in connecting piping, turbine stop valves, and bypass valves can close against maximum steam flow. The reviewer verifies that the SAR provides a tabulation and descriptive text of all flow paths that branch off the main steam lines between the MSIVs and the turbine stop valves. The descriptive information shall include the following for each flow path:

(1) System identification

(2) Maximum steam flow in pounds per hour

(3) Type of shut-off valve(s)

(4) Size of valve(s)

(5) Quality of the valve(s)

(6) Design code of the valve(s)

(7) Closure time of the valve(s)

(8) Actuation mechanism of the valve(s) (i.e., solenoid operated, motor operated, air operated diaphragm valve, etc.)

(9) Motive or power source for the valve actuating mechanism.

e. In the event of a main steam line break, termination of steam flow from all systems identified in d, above, except those that can be used for mitigation of the accident, is required to bring the reactor to a safe cold shutdown. For these systems the reviewer verifies that the SAR describes what design features have been incorporated to assure closure of the steam shut-off valve(s) and what operator actions, if any, are required. If the systems that can be used for mitigation of the accident are not available, or the decision is made to use other means to shut down the reactor, the reviewer verifies that the SAR decribes how these systems are secured to assure positive steam shut-off and what operator actions, if any, are required.

f. Assure that in the event of a postulated safe shutdown earthquake in a PWR plant, the design includes the capability to operate atmospheric dump valves remotely from the control room so that cold shutdown can be achieved using only safety-grade components, assuming a concurrent loss of offsite power (refer to Branch Technical Position RSB 5-1 attached to SRP Section 5.4.7).

6. The reviewer verifies that the system is designed so that essential functions will be maintained, as required, in the event of adverse environmental phenomena, certain pipe breaks, or loss of offsite power. The reviewer uses engineering judgment and the results of failure modes and effect analyses to determine that:

a. Failure of nonseismic Category I portions of the MSSS or of other systems located close to essential portions of the system, or of nonseismic Catagory I structures that house, support, or are close to essential portions of the MSSS, do not preclude operation of the essential portions of the MSSS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that confirm that the above conditions are met are acceptable.

b. Essential portions of the MSSS are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Flood protection and missile protection criteria are evaluated under the SRP Section 3 series. The locations and the design of the system and structures are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of winds, flooding, and tornado missiles is acceptable.

c. Essential portions of the MSSS are protected from the effects of high and moderate energy line breaks and cracks, including pipe whip, jet forces, and environmental effects. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in SRP Section 3.6.

d. Essential components and subsystems necessary for safe shutdown can function as required in the event of loss of offsite power. The SAR is reviewed to verify that for each MSSS component or subsystem affected by a loss of offsite power, the system functional capability meets or exceeds minimum design requirements. Statements in the SAR and results of failure modes and effects analyses are considered in assuring that the system meets these requirements. This is an acceptable verification of system functional reliability.

7. The descriptive information, P&IDs, MSSS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system will function following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum requirements are met for each accident situation for the required time spans. For each case the design is acceptable if minimum system requirements are met.

8. The SAR is reviewed to assure that the applicant has committed to address the potential for steam hammer and relief valve discharge loads, and will take adequate procedures action to minimize such occurrences. Drain pots, line slope and valve operators should be addressed.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

The main steam supply system (MSSS) includes all components and piping from the outermost containment isolation valves (for BWRs) [from the steam generator connection (for PWRs)] up to and including the turbine stop valves. The essential portions of the MSSS are designed to quality Group B [for PWRs, from the steam generator to the containment isolation valves, and connected piping up to and including the first valve that is normally closed] [for BWRs, from the outermost containment isolation valves and connecting piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation, but not including the turbine stop and bypass valves]. Those portions of the MSSS necessary to mitigate the consequences of an accident such as a steam line break are designed to the quality standards commensurate with the importance to its safety function, and are designed to the following standards: ________________________. The scope of review of the MSSS for the ____________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system.

The basis for acceptance of the MSSS in our review was conformance of the applicant's design criteria and bases to the Commission's regulations as set forth in the General Design Criteria (GDC) of Appendix A to 10 CFR Part 50. The staff concludes that the plant design is acceptable and meets the requirements of GDC 2, 4, 5, and 34. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," with respect to the ability of structures housing the safety-related portion of the system and the safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods and GDC 4 "Environmental and Missile Design Bases" with respect to structures housing the safety-related portions of the system and the safety-related portions of the system being capable of withstanding the effects of external missiles, and internally-generated missiles, pipe whip and jet impingement forces associated with pipe breaks. The essential portions of the MSSS (as identified in the above discussion) are designed Seismic Category I and housed in a Seismic Category I structure which provides protection from the effects of tornadoes, tornado missiles, turbine missiles, and floods. This meets the positions of Regulatory Guide 1.29, "Seismic Design Classification," Position C.1.a, C.1.e, C.2 and C.3 or C.1.f, C.2 and C.3; Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles," Position C.1; and Regulatory Guide 1.117, "Tornado Design Classification," Appendix Positions 2 and 4.

   In addition, the system design capabilities should include the capability to accommodate steam hammer dynamic loads resulting from rapid closure of systems valves (including turbine bypass and stop valves), and safety/relief valve operation without compromising required safety functions. Water entrainment considerations should include provisions for drain pots, line sloping and valve operation. Operating and maintenance procedures are to be reviewed by the applicant to alert plant personnel to the potential for such occurrences and means to minimize such occurrences. This commitment should be stated in the applicants' SAR.

2. The applicant has met the requirements of GDC 5, "Sharing of Structures, Systems, and Components with Respect to the Capability of Shared Systems and Components," important to safety to perform required safety functions. We have reviewed the interconnections from the MSSS of each unit to ___________. The interconnections are designed so that the capability to mitigate the consequences of an accident in either unit and achieve safe shutdown in that unit is retained without reducing the capability of the other unit to achieve safe shutdown.

   or

   Each unit of the ___________ plant has its own MSSS with no interconnections between the safety-related and/or nonsafety- related portions.

3. The applicant has met the requirements of GDC 34, "Residual Heat Removal," with respect to the system function of transferring residual and sensible heat from the reactor system in PWR plants. The MSSS is capable of providing heat sink capacity and pressure relief capability and supplying steam to the steam driven safety-related pumps necessary for safe shutdown. The MSSS is also designed to include the capability to operate the atmospheric pump valves remotely from the control room following a safe shutdown earthquake coincident with the loss of offsite power so that a cold shutdown can be achieved with dependence upon safety-grade components only. This meets the positions in Branch Technical Position RSB 5-1, "Design Requirements of Residual Heat Removal System," and in Issue 1 of NUREG-0138.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implemenation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides, NUREGs and implementation of acceptance criterion subsection II.2, associated with water hammer loads, is as follows:

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures Systems and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

5. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

6. Regulatory Guide 1.29, "Seismic Design Classification."

7. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

8. Regulatory Guide 1.117, "Tornado Design Classification."

9. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1, Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

10. Branch Technical Position RSB 3-1, "Classification of Main Steam Components Other than the Reactor Coolant Pressure Boundary for BWR Plants," attached to SRP Section 3.2.2.

11. Branch Technical Position RSB 3-2, "Classification of BWR/6 Main Steam and Feedwater Components Other Than the Reactor Coolant Pressure Boundary," attached to SRP Section 3.2.2.

12. Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," attached to SRP Section 5.4.7.

13. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, memorandum from Director NRR to NRR Staff."

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Standard Review Plan for the
Review of Safety Analysis Reports
for Nuclear Power Plants

Section No. 10.4.7
Revision No. 3

Appendix No. N/A
Revision No. N/A

Branch Tech. Position ASB 10-2
Revision No. 3

Date Issued April 1984

FILING INSTRUCTIONS

PAGES TO BE REMOVED			NEW PAGES TO BE INSERTED		
PAGE NUMBER		DATE	PAGE NUMBER		DATE
10.4.7-1 thru 10.4.7-6	Rev. 2	July 1981	10.4.7-1 thru 10.4.7-7	Rev. 3	April 1984
BTP ASB 10-2 to SRP Section 10.4.7	Rev. 2	July 1981	BTP ASB 10-2 to SRP Section 10.4.7	Rev. 3	April 1984
10.4.7-7 thru 10.4.7-8			10.4.7-8 thru 10.4.7-10		

The U.S. Nuclear Regulatory Commission's
Standard Review Plan, NUREG-0800, prepared by the
Office of Nuclear Reactor Regulation, is available
for sale by the National Technical Information
Service, Springfield, VA 22161.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

10.4.7 CONDENSATE AND FEEDWATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary Systems Branch (ASB)

Secondary - None

I. AREAS OF REVIEW

The condensate and feedwater system (CFS) 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.

ASB reviews the CFS from the condenser outlet to the connection with the nuclear steam supply system and to the heater drain system to assure conformance to General Design Criteria 2, 4, 5, 44, 45 and 46. For indirect cycle plants, there are also interfaces with the secondary water makeup system and the auxiliary feedwater system. The CFS is used for normal shutdown. The only part of the CFS classified as safety-related, i.e., required for safe shutdown or in the event of postulated accidents, is the feedwater piping from the steam generators for PWRs and from the nuclear steam supply system for BWRs, up to and including the outermost containment isolation valve.

1. The ASB reviews the characteristics of the CFS with respect to the capability to supply adequate feedwater to the nuclear steam supply system as required for normal operation and shutdown.

2. The ASB review determines that an acceptable design has been established for:

   a. The interfaces of the CFS with the auxiliary feedwater system (PWR), the reactor core isolation cooling system (BWR), and the condensate cleanup system with regard to functional design requirements and seismic design classification.

b. The feedwater system (PWR), including the auxiliary feedwater system piping entering the steam generator, with regard to possible fluid flow instabilities (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.

c. The detection of major system leaks that could affect the functional performance of safety-related equipment.

3. ASB also performs the following reviews under the SRP sections indicated:

(a) Review for flood protection is performed under SRP Section 3.4.1,

(b) Review of the protection against internally generated missiles is performed under SRP Section 3.5.1.1,

(c) Review of the structures, systems, and components to be protected against externally generated missiles is performed under SRP Section 3.5.2, and

(d) Review of high- and moderate-energy pipe breaks is performed under SRP Section 3.6.1.

The ASB will coordinate evaluations performed by other branches that interface with the overall evaluation of the system as follows:

The Reactor Systems Branch (RSB) determines that transients resulting from feedwater flow control malfunctions will not violate the primary system pressure boundary integrity criterion as part of its primary review responsibility for SRP Sections 15.1.1 through 15.1.4, and that the loss of normal feedwater flow will not violate the fuel damage criterion or the system pressure boundary integrity criterion as part of its primary review responsibility for SRP Section 15.2.7.

The Power Systems Branch (PSB) evaluates the system power sources with respect to their capability to perform safety-related functions during normal, transient, and accident conditions as part of its primary review responsibility for SRP Section 8.3.1. The Structural and Geotechnical Engineering Branch (SGEB) determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles as part of its primary review responsibility for SRP Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5. The Mechanical Engineering Branch (MEB) determines that the components, piping and structures are designed in accordance with applicable codes and standards as part of its primary review responsibility for SRP Sections 3.9.1 through 3.9.3. The MEB determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2. The MEB also reviews the adequacy of the inservice testing program of pumps and valves as part of its primary review responsibility for SRP Section 3.9.6. Upon request, the MEB determines the acceptability of design analyses, procedures, and criteria used to establish the adequacy of devices or restraints as they may relate to significant water hammers in system piping and the MEB reviews test programs of components that may be affected by water hammers. The Materials Engineering Branch (MTEB) verifies that inservice inspection requirements are met for system components as part of its primary review responsibility for SRP Section 6.6 and, upon request, verifies the compatibility of the materials of construction with service conditions. The review for Fire Protection, Technical Specifications, and Quality Assurance are coordinated and performed by the Chemical Engineering Branch, Standardization and Special Projects Branch, and Quality Assurance Branch as part of their primary review responsibility for SRP Sections 9.5.1, 16.0, and 17.0, respectively. The Equipment Qualification Branch (EQB) reviews the seismic qualification of Category I instrumentation and electrical equipment and the environmental qualification of mechanical and electrical equipment as part of its primary review responsibility for SRP Sections 3.10 and 3.11, respectively. Upon request, the Instrument and Control Systems Branch (ICSB) will review the instrumentation and controls associated with the feedwater control system (BWR) or steam generator level control system (PWR).

For those areas of review identified above as being part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP sections of the corresponding primary branches.

II. ACCEPTANCE CRITERIA

Acceptability of the condensate and feedwater system, as described in the applicant's safety analysis report (SAR), is based on the specific requirements of General Design Criteria and the positions of regulatory guides. Listed below are the specific criteria as they relate to the CFS.

1. General Design Criterion 2, as related to the system being capable of withstanding the effects of earthquakes. Acceptance is based on meeting the guidance of Regulatory Guide 1.29, Position C.1 for safety-related portions, and Position C.2 for nonsafety-related portions.

2. General Design Criterion 4, as related to the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation as well as during upset or accident conditions. Acceptance is based on meeting the guidance contained in the attached Branch Technical Position ASB 10-2 for reducing the potential for water hammers in steam generators and on meeting the guidance related to feedwater control induced water hammer.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety to perform required safety functions.

4. General Design Criterion 44, as it relates to:

   a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.)

c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

5. General Design Criterion 45, as related to design provisions to permit periodic inservice inspection of system components and equipment.

6. General Design Criterion 46, as related to design provisions to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II of this SRP section. For the review of operating license (OL) applications, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report.

The primary reviewer will coordinate this review with the areas of review of interfacing branches as stated in subsection I of this SRP section. The primary reviewer obtains and uses such inputs as required to assure that this review procedure is complete.

The reviewer will select and emphasize material from this SRP section as may be appropriate for a particular case.

The SAR is reviewed to determine that the system description and diagrams delineate the function of the condensate and feedwater system under normal and abnormal conditions. The reviewer verifies the following:

1. The system has been designed to function as required for all modes of operation. The results of failure modes and effects analyses presented in the SAR, if any, are used in making this determination.

2. The system piping is designed to preclude hydraulic instabilities from occurring in the piping for all modes of operation. As appropriate, the reviewer evaluates the results of model tests and analyses that are relied on to verify that water hammer will not occur, or proposed tests of the installed system that are intended to verify design adequacy. Steam generators are reviewed in accordance with Branch Technical Position ASB 10-2.

   The feedwater control valve and controller design shall be verified to be stable and to be compatible with system(s), imposed operating conditions (e.g., control functions required, range of control and pressure drop characteristics, valve stroke, trim, etc.). Test data or operating experience data shall be used where available. In addition, the applicant has committed to review plant operating and maintenance procedures to assure that precautions for avoidance of steam/water hammer and water hammer occurrences have been provided.

3. The outermost containment isolation valves and all downstream piping to the nuclear steam supply system are designed in accordance with seismic Category I requirements. The review for seismic design is performed by SGEB and the review for seismic and quality group classification is performed by MEB as indicated in subsection I of this SRP section.

4. The CFS design is such that the plant can be safely shut down using the auxiliary feedwater system or the reactor core isolation cooling system, if required.

5. The CFS design, or other plant systems, provide the capability to detect and control leakage from the system.

6. The reviewer verifies that the essential portion of the system has been designed so that system function will be maintained as required in the event of adverse environmental phenomena or loss of offsite power. The review for protection against natural phenomena is performed in the Chapter 3 SRP sections. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses, to determine that the failure of nonessential portions of the system or of other systems not designed to seismic Category I standards and located close to essential portions of the system, or of nonseismic Category I structures that house, support, or are close to essential portions of the CFS, will not preclude operation of the essential portions of the CFS.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The condensate and feedwater system includes all components and equipment from the condenser outlet to the connection with the nuclear steam supply system and to the heater drain system [secondary water makeup system, and auxiliary feedwater system interfaces. (PWRs only)]. Based on the review of the applicant's proposed design criteria, the design bases, and safety classificatic for the safety-related portions of the condensate and feedwater system and the requirements for system performance for all conditions of plant operation, the staff concludes that the design of the condensate and feedwater system and supporting systems is in conformance with the Commission regulations as set forth in General Design Criterion 2, 4, 5, 44, 45 and 46. This conclusion is based on the following:
>
> 1. The applicant has met the requirements of General Design Criterion 2 with respect to safety-related portions of the system being capable of withstanding the effects of earthquakes by meeting Regulatory Guide 1.29 Position C.1 for the safety-related portions and Position C.2 for the nonsafety-related portions.

2. The applicant has met the requirements of General Design Criterion 4 with respect to the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) by having the feedwater system designed in accordance with the guidance contained in Branch Technical Position ASB 10-2 and thereby eliminating or reducing the possibility of water hammers in steam generators (PWRs only).

   That the applicant has adequately addressed feedwater control valve and controller designs with respect to water hammer potential and the applicant has committed to review operating and maintenance procedures to assume that precautions taken will minimize, or avoid, water hammers.

3. The applicant has met the requirements of General Design Criterion 5 with respect to the capability of shared systems and components important to safety to perform required safety functions. We have reviewed the interconnections of the CFS between each unit. The interconnections are designed so that the capability to mitigate the consequences of an accident in either unit and achieve safe shutdown in that unit is retained without reducing the capability of the other unit to achieve safe shutdown.

4. The applicant has met the requirements of General Design Criterion 44 with respect to cooling water by providing a redundant and isolable system capable of transferring heat loads from the reactor system to a heat sink under both normal operating and accident conditions. The applicant has demonstrated that the condensate and feedwater system can provide sufficient cooling water to transfer the heat load of the reactor system under normal operating conditions and accident conditions assuming loss of offsite power and a single failure and that portions of the system can be isolated so that the safety function of the system will not be compromised.

5. The applicant has met the requirements of General Design Criterion 45 with respect to inspection of cooling water systems by providing a feedwater system design that permits inservice inspection of safety-related components and equipment.

6. The applicant has met the requirements of General Design Criterion 45 with respect to testing of cooling water systems by providing a feedwater system design that permits operational functional testing of the safety-related portion of the system and its components.

The staff concludes that the design of the CFS conforms to all applicable GDCs and positions of the regulatory guide cited and is, therefore, acceptable.

V. IMPLEMENTATION

The following is intended to provide guidance to all applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guide and implementation of acceptance criterion subsection II.2, associated with water hammer loads, is as follows:

(a) Operating plants and OL applicants need not comply with the provisions of this revision.

(b) CP applicants will be required to comply with the provisions of this revision.

(c) It should be noted that steam generators in operating plants and NTOL's where a SER has been issued, now comply with the revised BTP ASB 10-2.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures Systems and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

7. Regulatory Guide 1.29, "Seismic Design Classification."

8. Branch Technical Position ASB 10-2, "Design Guidelines for Avoiding Water Hammer in Steam Generators."

BRANCH TECHNICAL POSITION ASB 10-2

DESIGN GUIDELINES FOR AVOIDING WATER HAMMERS IN STEAM GENERATORS

BACKGROUND

Plant operational experience has shown that top-feed steam generators containing feedwater spargers with bottom drain holes incur steam condensation induced water hammers. This type of water hammer has frequently occurred after the feedwater sparger was uncovered (due to some plant transient) and cold auxiliary feedwater flow was subsequently initiated. The initiation of the auxiliary feedwater flow into the steam generator produces a water slug in the sparger or feedwater piping, which is then accelerated by the unbalanced pressures produced by the condensation of a steam pocket in the line. The resultant impulse could be of a sufficient magnitude to cause damage to the steam generator internal components and feedwater systems piping. The most damaging of such water hammer incidents occurred at Indian Point No. 2 in 1973, where the water hammer loads resulted in rupture of an 18-inch feedwater pipe and damage to the containment inner liner. The repeated occurrence of such water hammers and potential severity such flow instabilities resulted in the NRC in engaging Creare Inc. in 1976 to evaluate causes and effects, and to develop recommendations for avoidance of top feed steam generator water hammer, and design methods minimize associated dynamic loads.

The underlying causes of water hammer in top-feed steam generators were extensively studied by Creare, Inc. who reported findings and recommended design modifications to minimize or preclude such water hammer occurrence in NUREG-0291 (1977). These recommendations called for: (a) use of J-tubes on the topside of the feedring to minimize loss of water when uncovered, (b) early initiation of auxiliary feedwater to keep piping and feedring full of water, (c) short horizontal FW pipe lengths at the SG nozzle to reduce magnitude of slug formation and impact, (d) limit FW recovery flow rates to less than 150 gpm/SG to minimize steam-water entrainment and subsequent formation of a water slug. The use of top discharge feed (i.e., tubes) makes flow rate limits practical because the limit only has to be imposed until the piping is full, regardless of steam generator water level. The design and operational modifications were implemented by plants experiencing SG water hammer and appear to have essentially eliminated SGWH. NUREG-0918 details plant specific modifications which were made. In addition, experience sustains maintaining preoperational tests to verify the absence of SGWH.

More recently, Westinghouse and Combustion Engineering have introduced steam generators of the preheat type, wherein the majority of feedwater enters the steam generator at the bottom through a preheater section. The potential for condensation-induced water hammer in preheat steam generators was studied by BNL and reported in NUREG/CR-1606, "An Evaluation of Condensation-Induced Water Hammer in Preheat Steam Generators," June 1980. This report, citing the lack of definitive experimental and analytical results, recommended full scale verification tests to demonstrate the absence of damaging water hammer in preheat steam generators and connecting feedwater piping (i.e., preoperational tests).

B&W steam generators, which are a "once through" flow design, have generally not reported water hammer occurrence. However, in May 1982, several B&W plants (following inservice inspection) reported damaged internal auxiliary feedwater headers and support structures. The cause was attributed to steam pocket collapse. The internal auxiliary feedring design concept is similar to CE & W top feedring concepts which have experienced water hammer before corrective design measures were implemented. For these B&W plants, the OTSG's are being modified to return to the previous design using auxiliary feedwater injection manifolds which are external to the steam generator.

The staff believes that SGWH evidence and studies performed to date warrant the establishment of design guidelines for steam generators and the associated piping. Guidelines have been developed that may be used to reduce the probability of a damaging steam condensation induced water hammer, particularly for the Westinghouse and Combustion Engineeering PWR designs which use top-feed steam generators.

BRANCH TECHNICAL POSITION

In CP and OL application reviews, the staff requires the applicant to provide the following design capability and verification:

Top-Feed Steam Generator Designs

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

a. Prevent or delay water draining from the feedring following a drop in steam generator water level by means such as top discharge J-Tubes and limitng feedring seal assembly leakage.

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

c. 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 and submit the results from such tests.

d. Implement pipe refill flow limits where practical.

Preheat Steam Generator Designs

1. Minimize the horizontal lengths of feedwater piping between the steam generator and the vertical run of piping by providing downward turning elbows immediately upstream of the main and auxiliary feedwater nozzles.

2. Provide a check valve upstream of the auxiliary feedwater connection to the top feedwater line.

3. Maintain the top feedwater line full at all times.

4. Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Also perform a water hammer test at *% of power by using feedwater through the auxiliary feedwater (top) nozzle at the lowest feedwater temperature that the plant standard operating procedure (SOP) allows and then switching the feedwater at that temperature from the auxiliary feedwater nozzle to the main feedwater (bottom) nozzle by following the SOP, and submit the results of such tests.

Once Through Steam Generator (OTSG) Designs

a. Provide auxiliary feedwater to the steam generator through an externally mounted supply top discharge header.

b. 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. Provide the procedures for these tests for approval before conducting the tests, and submit the results of such tests.

REFERENCES

(1) Block, J. A. et.al., "An Evaluation of PWR Steam Generator Water Hammer," NUREG-0291, June 1977.

(2) Chapman, R. L., et.al., "Compilation of Data Concerning Known and Suspected Water Hammer Events in Nuclear Power Plants," NUREG/CR-2059, May 1982.

(3) Anderson, N. and Han, J. T., "Prevention and Mitigation of Steam Generator Water Hammer Events in PWR Plants," NUREG-0918, December 1982.

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*The power level at which feedwater flow is transferred from the auxiliary feedwater nozzle to the main feedwater nozzle.

NUREG-0993, Rev. 1

REGULATORY ANALYSIS FOR USI A-1, "WATER HAMMER"

FEBRUARY

Revision 1

Regulatory Analysis for USI A-1, "Water Hammer"

(Formerly *Value-Impact Analysis for USI A-1, "Water Hammer"*)

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

A. W. Serkiz

NUREG-0993
Revision 1

Regulatory Analysis for USI A-1, "Water Hammer"

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ABSTRACT

NUREG-0993, Revision 1 is the staff's regulatory analvsis dealing with the resolution of the Unresolved Safety Issue A-1, Water Hammer. This report contains the value-impact analysis for this issue, public comments received, and staff response, or action taken, in response to those comments. The staff's technical findings regarding water hammer in nuclear power plants are contained in NUREG-0927.

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REGULATORY ANALYSIS
FOR
USI A-1, WATER HAMMER

I. The Recommended Actions

A. Summary of Problem and Recommended Actions

The Unresolved Safety Issue (USI) A-1 deals with safety concerns related to water hammer occurrence in nuclear power plants. The staff's concerns were prompted by the increasing frequency of water hammer occurrence (see Figures 1 and 2) in the mid-1970's as new plants were coming on line, and, in particular, by the feedwater line rupture at Indian Point 2 in 1973 (attributed to water hammer induced by steam-void collapse). Principal concerns were: the potential for inadequate dynamic load design, disabling of safety systems, and the release of radioactivity. The staff's views were set forth in NUREG-0582 (Ref. 1), and water hammer was designated a USI in 1979.

Historically, nearly 150 water hammer events have been reported since 1969; 81 have occurred in boiling water reactors (BWRs) and 67 have occurred in pressurized water reactors (PWRs). (Twenty-seven of the PWR water hammers have occurred in steam generators.) With the exception of the Indian Point 2 event in 1973, reported damage has been principally confined to pipe hangers, snubber systems, and equipment-mounting structures. Furthermore, approximately half of these water hammers occurred in the plant preoperational phase or first year of commercial operation (which indicates a plant operational learning process). Also, only about half of the operating plants have reported water hammer occurrences. A compilation of reported water hammer occurrences, underlying causes and plant corrective actions taken is provided in NUREG/CR-2059 (Ref. 2).

As noted above, the increasing frequency of occurrence drew both staff and utility attention to water hammer, and corrective actions were implemented in the mid-1970's. Steam generator (top feedring design) water hammer was studied (Ref. 3) and eliminated through NRC-initiated design retrofits calling for J-tubes, shortened piping, and controlling auxiliary feedwater flow rates (Ref. 4). Design corrective actions were also initiated by the industry and implemented for BWRs (e.g., "keep-full" systems, vacuum breakers, etc.). The net result of the corrective plant design modifications has been a reduced frequency of water hammer occurrence.

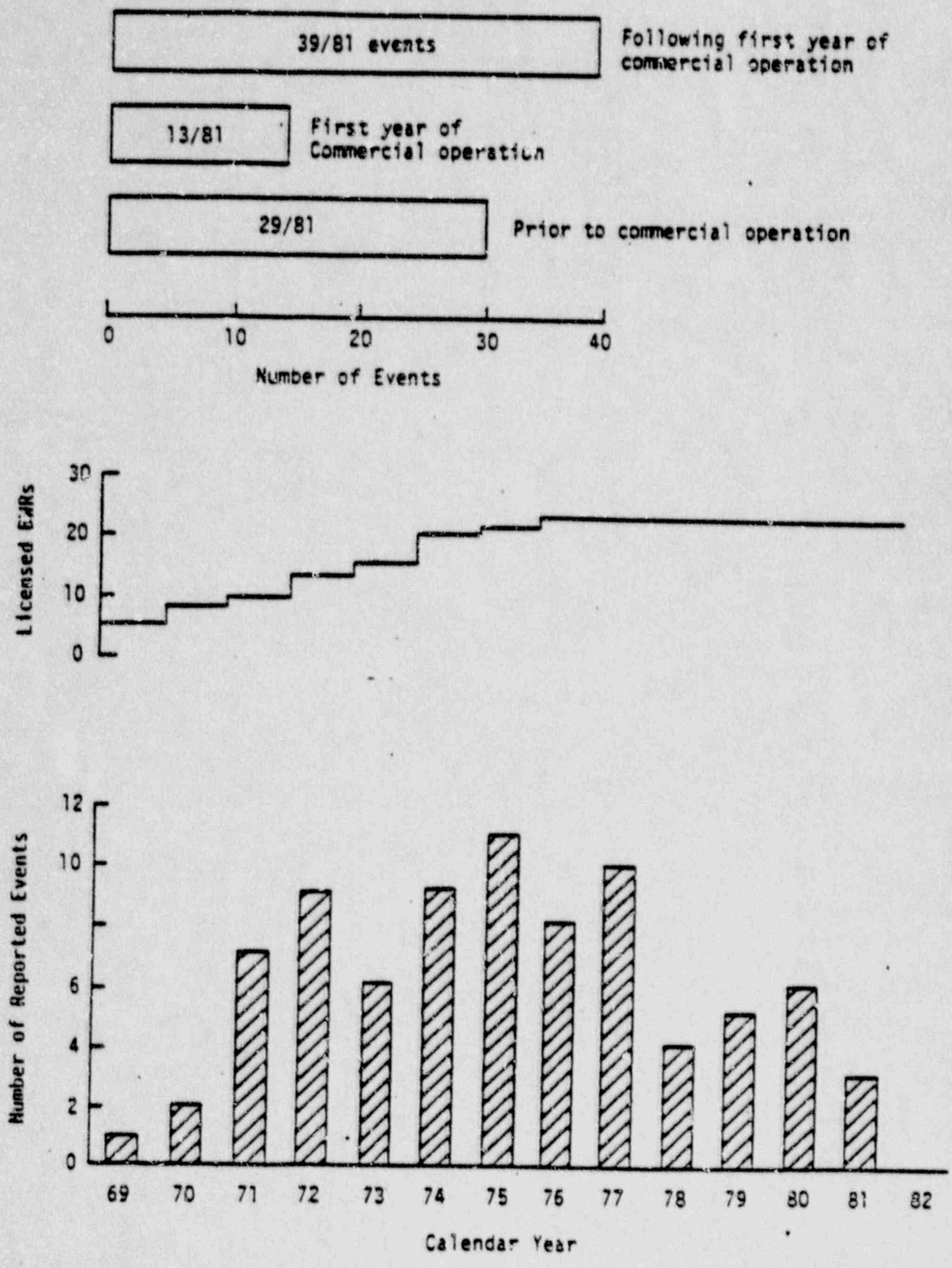

Figure 1 Reported Water hammer occurrences in US BWRs

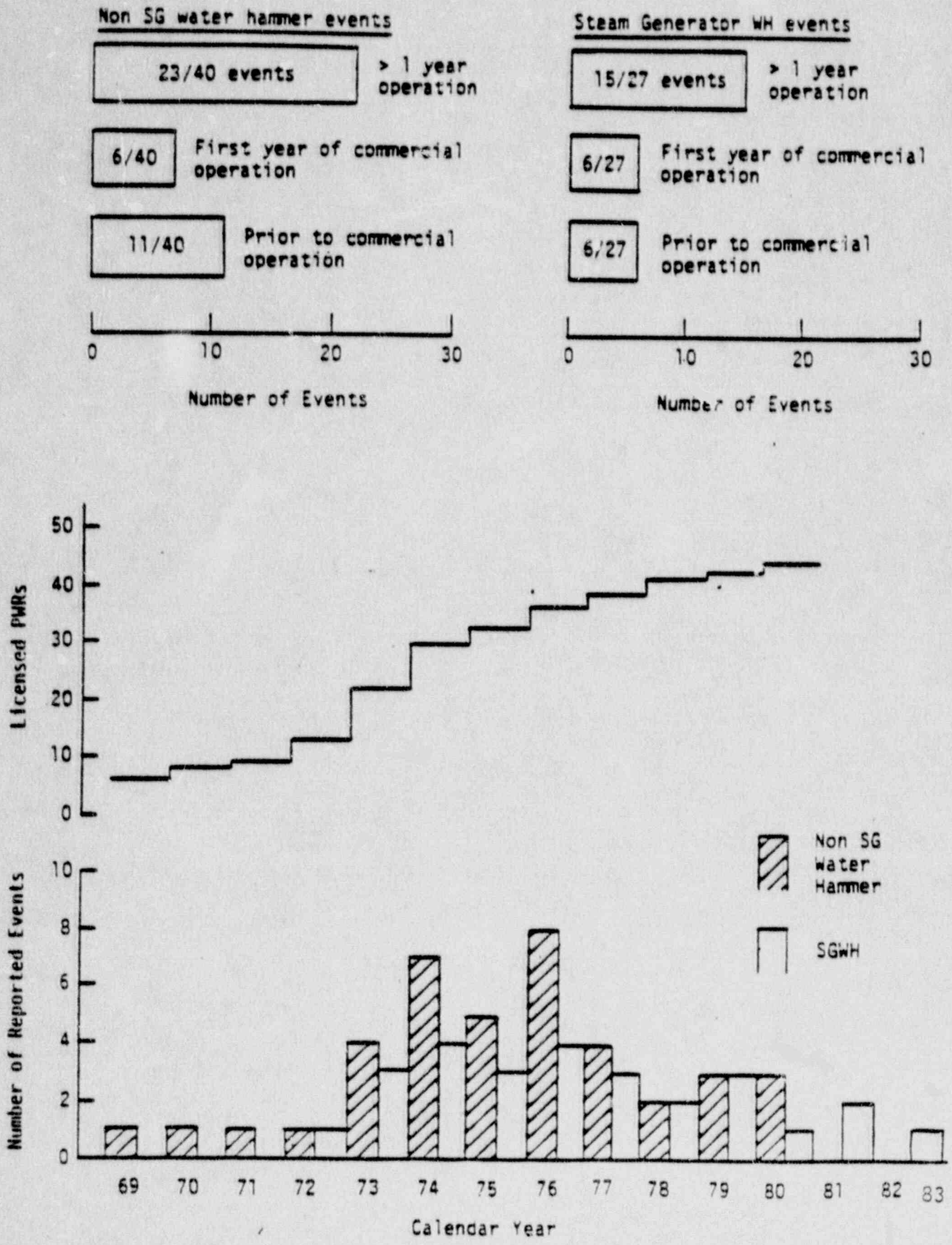

Figure 2 Reported water hammer occurrences in US PWRs.

The water hammer issue has recently been further studied (Ref. 5), and the technical conclusions derived reveal a significantly lesser safety concern than previously hypothesized. These results can be summarized as follows:

(1) Total elimination of water hammer is not feasible due to the possible coexistence of steam, water, and voids in various subsystems. Experience shows that design inadequacies and operator- or maintenance-related actions have contributed about equally to water hammer occurrences. BWRs are intrinsically more susceptible to water hammer occurrence.

(2) Reported damage has been principally confined to piping support systems, and none of the reported water hammer occurrences has resulted in any radioactive release.

(3) Frequency and severity of water hammer can be reduced and maintained low through the continued use of the types of design features discussed above.

(4) Additional operator awareness and training could lead to a further reduction of water hammer occurrence. Use of void detection instrumentation to alert operators to voided conditions would also help.

The staff's current technical findings relative to the water hammer issue are set forth in NUREG-0927 (Ref. 9). These findings are based on water hammer evaluations; References 2, 4, 5 and 8; and public comments received (see Appendix A).

The following actions are recommended:

(1) Issue the staff's water hammer technical findings (NUREG-0927) as an informational document for use by the industry for feedback of design and operating experience to plant staff. NUREG-0927 reviews water hammer occurrences, underlying causes, and systems affected, and sets forth potential means for avoiding water hammer.

(2) Ensure operator awareness and training (for avoiding water hammer) through the implementation of TMI Task Action Plan, Part I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff," and Part I.A.2.3, "Administration of Training Programs." The NRC Office of Inspection and Enforcement (IE) has verified, through its inspection program, that general procedures for implementing Part I.C.5 have been established. The Licensee Qualifications Branch of the NRC Division of

Human Factors Safety (DHFS) is developing guidelines and criteria to evaluate and upgrade utility training programs (per Part I.A.2.3) and will include water hammer as one of a number of safety issues currently identified. Since activities for implementing applicable sections of the TMI Task Action Plan are under way, and since the safety significance of water hammer is less than previously viewed, no special action to implement findings presented in NUREG-0927 is necessary.

(3) Issue the following revisions to Standard Review Plan (SRP) Sections: 3.9.3, ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures; 3,9.4, Control Rod Drive Systems; 5.4.6, Reactor Core Isolation Cooling System (BWR); 5.4.7, Residual Heat Removal (RHR) System; 6.3 Emergency Core Cooling System; 9.2.1 Station Service Water System; 9.2.2, Reactor Auxiliary Cooling Water Systems; 10.3, Main Steam Supply Systems; and 10.4.7, Condensate and Feedwater Systems reflect current water hammer findings and will ensure continued use of design features which have eliminated or minimized water hammer occurrence. Public comments received have been reflected in these SRP revisions (see Appendix A). The revised SRPs would be used for reviews of "custom plant" Construction Permit (CP) applications and for reviews of Standard Plant applications docketed after issuance of the revision and which are intended for referencing in CP applications.

B. <u>Need for Recommended Actions</u>

The need for the recommended actions is as follows:

(1) Make use of experience gained regarding design features and operating experience which have shown a capability to eliminate or minimize water hammer occurrence to ensure that future plant designs utilize design features proved effective in eliminating water hammer.

(2) Clarify current staff review practices to ensure that the review process is more predictable and thus reduce the burden of the regulatory process.

C. Value-Impact Data on the Recommended Actions

1. Risk Analysis Results

A risk assessment study (Ref. 6) was performed to assess the significance of risk from water hammer occurrence with respect to overall plant risk. Water hammer frequencies were derived from reported occurrences (Ref. 2), and component or system failure models were developed from system assessments (Ref. 5). For example, if only piping support damage was reported, then the assumption was made that the system would still function. If water hammer occurrence resulted in disabling the system, then models were constructed for modifying failure event networks (i.e., failure-on-demand frequency). A more detailed discussion of the derived frequencies and failure models is contained in Reference 7.

Three specific nuclear plants for which probabilistic risk assessment (PRA) models were available (namely Millstone 1 (BWR-3), Browns Ferry 2 (BWR-4), and Sequoyah 1 (PWR)) were selected for this risk study as representative of operating reactors. Since reported water hammer experience reveals a higher frequency of occurrence in BWRs and a dependence on different BWR designs, the emphasis was directed at potential BWR risks.

The release categories and associated public dose estimates employed are shown in Table 1. These public dose values were derived using the CRAC code and assuming the guidelines and quantities of radioactive isotopes used in the Reactor Safety Study (WASH-1400), the meteorology at a typical mid-west site (Byron-Braidwood), a uniform population density of 340 people per square mile (which is an average of all U.S. nuclear power plant sites), and no evacuation of population. They are also based on a 50-mile-release-radius model (see also Ref. 7). The release categories shown in Table 1 correspond to radiological release causes described in WASH-1400 (e.g., steam explosion with containment rupture, core melt, etc.).

The estimated public dose due to water hammer was derived from the "base case" PRA results versus calculated increases in core melt frequencies and increases in the respective release category frequencies for the plants noted above (see Table 1). Basically, the calculations provide a means to compare calculated risk results with and without water hammer.

The results of these risk assessments are summarized in Table 2, where both calculated public doses and core melt frequencies are shown. The differences shown in the third column are the calculated change due to inclusion of water hammer-induced failures in the event trees. These calculations are conservative since the assumption was made that safety systems were disabled as a result of a frequency-of-failure or demand model as derived from reported water hammer events. (Refer to Reference 6 for a more detailed analysis.)

The results in Table 2 can be summarized as follows:

(1) Water hammer effects on PWR risk are negligible.
(2) Water hammer effects on BWR risk are negligible or small.

As part of the risk analyses performed for BWRs, BWR plants with isolation condensers (ICs) were evaluated in some detail because: (a) Millstone 1 has repeatedly incurred water hammer in the IC, and (b) should a water hammer fail the pressure boundary of the IC, a direct release pathway is opened to the environment. This type of failure model in a risk analysis using the Millstone Integrated Reliability Evaluation Program model resulted in a significant dose and consequent risk from potential IC failure by water hammer.

When the risk model was modified to include a feedwater pump trip on high reactor vessel water level, the risk from water hammer in the isolation condenser was virtually eliminated. The risk analyses, therefore, showed that a high reactor vessel water level feedwater pump trip, which removes the conditions for water carryover into the IC, is a generic resolution to the problem.

Operating experience data support this conclusion. Plants that have a feedwater pump trip (Dresden 2 and 3) have not reported water hammer in the IC. Some plants without such a trip (Millstone 1 and Nine Mile Point) have reported IC water hammer events. Millstone 1 has not reported an IC water hammer since installation of the feedwater pump trip about 10 months ago.

Table 3 provides an overview of all operating BWR plants with ICs. Only Oyster Creek and Big Rock Point have not installed or have not committed to have installed a high reactor vessel water level feedwater pump trip. Neither plant has reported any water hammer experience with its IC. As noted in the

TABLE 1, PUBLIC DOSE VALUES UTILIZED FOR USI A-1 RISK STUDY

Release Category	Release Category Dose Multiplier (man-rem)
PWR-1	5.4E+6
PWR-2	4.8E+6
PWR-3	5.4E+6
PWR-4	2.7E+6
PWR-5	1.0E+6
PWR-6	1.5E+4
PWR-7	2.3E+4
PWR-8	7.5E+4
PWR-9	1.2E+2
BWR-1	5.4E+6
BWR-2	7.1E+6
BWR-3	5.1E+6
BWR-4	6.1E+5
BWR-5	2.0E+1

*Values from NUREG/CR-2800 (7); (man-rem) x (probability of occurrence) = public dose resulting from the release category noted. Total release obtained by summing the categories.

TABLE 2 SUMMARY OF RISK ASSESSMENT CALCULATIONS

Calculated Public Dose (man-rem/plant-year)

Type of Plant	Base Case (w/o W.H.)	With Water Hammer	Calculated Increase (due to W.H.)
BWR-3s	No calculated change due to Water Hammer.		
BWR-4s	1147	1169	22
PWRs	No calculated change due to Water Hammer.		

Calculated Core Melt Frequency (1/plant-yrs)

Type of Plant	Base Case (w/o W.H.)	W/Water Hammer	Change in Core Melt Frequency (due to W.H.)
BWR-3s	No calculated change due to Water Hammer.		
BWR-4s	2.0E-4	2.1E-4	1.0E-5
PWRs	No calculated change due to Water Hammer.		

footnote to the table, Oyster Creek is expected to install such a trip for other reasons. Big Rock Point is an older, smaller plant whose overall safety design is being addressed in the Systematic Evaluation Program. Therefore, no special or additional action for BWRs with ICs is contemplated as a part of the resolution for A-1.

It should be clearly recognized that the dose and risk attributable to IC water hammer are calculated values. None of the reported water hammer events has resulted in a significant release of radioactive material to the environment.

Using the incremental dose due to water hammer shown in Table 2, and assuming an "average" outstanding plant life of 25 years, the following change in public dose can be calculated:

BWR-4 Averted Public Dose = (22 man-rem/Rx-yr) (25 yrs)
= 550 man-rem/Rx

BWR-3s = No calculated change due to water hammer
PWRs = No calculated change due to water hammer.

These increases in public dose can be viewed as "averted public dose" (presupposing that corrective action is taken to avoid water hammer) for value-impact discussions. Thus, the very low values (0-550 man-rem/reactor) calculated for both PWRs and BWRs, do not support any special hardware backfit actions for operating plants.

2. Industry Impact

No new plant hardware or design changes are being recommended as a result of the USI A-1 resolution evaluations. The feedwater pump trip (noted previously as providing a generic resolution to BWR isolation condenser water hammers) is either in place or is being installed for other reasons in BWR-3s with ICs. Therefore, plant impacts are judged to be minimal or nonexistent.

TMI Task Action Plan I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff," requires that procedures be developed for feedback of operating experience to plant staff. Several groups within the industry (e.g., Institute of Nuclear Power Operations and reactor owners groups) have taken the lead in providing collective competance for meeting I.C.5 requirements. IE has verified the establishment of general guidelines for implementing I.C.5. Issuance of NUREG-0927 for

TABLE 3 BWRs with Isolation Condensers

Plant	Type	Number of Independent Isolation Condenser Loops	Installed Pump Trip On High Vessel Water Level
Millstone 1	BWR-3	1	Yes
Dresden 2	BWR-3	1	Yes
Dresden 3	BWR-3	1	Yes
Oyster Creek	BWR-2	2	No*
Nine Mile Point 1	BWR-2	2	No*
Big Rock Point	BWR-1	2	No

*These plants will in all likelihood need to install vessel overfill protection to reference Generic GE Safety/Relief Valve testing in their responses to TMI Action Plan Part II.D.1. Nine Mile Point is now committed to installing the trip in 1984.

informational purposes will assist industry activities currently under way. A more comprehensive set of guidelines and criteria to evaluate and upgrade utility training programs (per TMI Task Action Plan, I.A.2.3) is being developed by the Licensee Qualifications Branch. Water hammer is one of a number of safety issues being identified in the Licensee Qualification Branch plan. Thus issuance of NUREG-0927 will provide information which can be used with both I.C.5 and I.A.2.3.

With respect to forward fits (i.e., implementation of the revised SRP Sections), the impact should also be minimal. The proposed changes reflect design changes which have come about to remedy specific water hammer occurrences (i.e., fixes for top feedring steam generators, etc.) as problems arose and therefore represent proven design concepts. Since the proposed SRP revisions reflect the current state-of-knowledge concerning water hammer occurrences and systems which can prevent or minimize water hammer, the designer/operator can incorporate these revisions of proven system design changes.

In addition, the SRP has not previously contained specific guidance for reviewers with respect to water hammer considerations, with the exception of review guidance for water hammer in top feedring steam generators (ASB BTP 10-2). Thus the depth and scope of staff review have varied with individual reviewer experience and insights; however, this is consistent with the audit nature of the staff's review function. These changes do identify water hammer review areas that should be addressed on the basis of prior water hammer occurrences, design changes implemented by industry, and precautionary measures indicated by operating experience. Thus revising these SRP sections to include specific guidance on water hammer will clarify staff review practices and ultimately reduce the burden of the regulatory process. NUREG-0927 (which summarizes findings based on water hammer experience) can be used as a reference technical report.

Thus industry impact is judged to be minimal. Design costs associated with avoiding water hammer could be on the order of $50,000-100,000 (0.5-1.0 year of engineering time). The cost of new systems such as keep-full systems ($200,000-400,000), vacuum breakers ($100,000-300,000), and feedwater control systems ($100,000-200,000) are not insignificant, but they do not constitute major plant equipment costs. Operator training for water hammer avoidance is estimated to be on the order of $25,000-50,000 per plant. (The preceding cost estimates are based on discussions with vendors in the nuclear industry and should be viewed as preliminary estimates. Firm cost estimates will require plant-specific design take-offs for estimating actual equipment and associated installation costs.)

3. NRC Operations

The impact on NRC operations (or the review process) is negligible. The proposed SRP changes will reflect licensing review positions for new construction permit applications, since only a "forward fit" is recommended. NUREG-0927, the technical findings report, will be of use to both the reviewer and applicant. The estimated impact for reviewer attention to water hammer using the proposed SRP revisions is 4 weeks of engineering effort (or $15,000/plant).

With respect to followup to TMI Task Action Plan, Part I.C.5, each utility is required to conduct an internal audit to ensure that the feedback program functions at all levels. The Licensee Qualifications Branch is currently developing guidelines and criteria to evaluate and upgrade utility training programs to implement TMI Task Action Plan, Part I.A.2.3. Water hammer (as well as other safety issues) will be included in these guidelines. The Offices of Nuclear Reactor Regulation and Inspection and Enforcement will monitor effectiveness of this approach following implementation.

4. Other Government Agencies

No impact on other government agencies is projected.

5. Public

The value to the public would be the avoidance of public dose associated with water hammer events leading to core melt and offsite releases. As discussed in Section I.C.1 (see also Table 2), the calculated additional releases are zero for PWRs and approximately 550 man-rem/plant for BWRs. Since averted dose estimates are negligible, the principal value to the public would be to provide feedback to the industry in experience gained and in maintaining proven design concepts for future plants.

II. Regulatory Resolution

A. Regulatory Alternatives

(1) Issue NUREG-0927 for information only, and follow up on the feedback of operational experience through implementation of the TMI Task Action Plan, Part I.C.5.

(2) Issue NUREG-0927 for information, and issue for comment proposed revisions to SRP Sections 5.4.6, 5.4.7, 6.3, 9.2.1, 9.2.2, 10.3, and 10.4.7. These revisions reflect design changes and operating procedures which have proven effective in preventing water hammer. The revised SRP sections would apply to new custom plant CP reviews.

B. Discussion and Comparison of Regulatory Alternatives

(1) Issuance of NUREG-0927 for information only would have a zero impact. Followup on incorporation of water hammer operating experience into training models by the NRC Regional staff, per TMI Task Action Plan, Part I.C.5 would be a minimal impact since such inspections are normally done.

(2) Option (2), revision of the SRP sections noted and issuance of NUREG-0927 (for information) is a minimum impact and a forward fit approach. This option would ensure that future CP reviews will consider water hammer findings and design features currently proven effective to avoid water hammer. This is the recommended course of action.

III. Recommended Implementation Plan

A. Safety or Environmental Significance of Proposed Actions

The principal safety significance rests primarily on continuance of established plant design and operational procedures that have demonstrated the capability to minimize or avoid water hammer severity and damage, thereby avoiding damage leading to radioactive release. The recommended approach (Option 2) provides for continuance of proven design and operational considerations.

B. Recommended Resolution Action

(1) Issue NUREG-0927 containing the staff's technical findings as a technical information document.

(2) Issue and implement the revised SRP Sections 3.9.3, ASME Code Class 1, 2, and 3, Components Supports and Core Support Structures; 3.9.4, Control Rod Drive Systems; 5.4.6, Reactor Core Isolation Cooling System (BWR); 5.4.7, Residual Heat Removal (RHR) System; 6.3, Emergency Core Cooling System; 9.2.1, Station Service Water System; 9.2.2 Reactor Auxiliary Cooling Water Systems; 10.3, Main Steam Supply Systems; and 10.4.7, Condensate and Feedwater Systems, which are based on the findings reported in NUREG-0927, public comments received, and concluding staff evaluations.

Implementation of these revised SRP sections will apply to the review of custom plant CP applications and standard plant applications that are docketed after issuance of the SRP revisions.

These revised SRP sections incorporate licensing review guidance which would maintain use of experience gained regarding plant design features proved effective in eliminating water hammer, and which also clarify current staff review practices to ensure that the review process is more predictable and definitive.

(3) Issue NUREG-0993 (formerly issued for public comment as the value-impact analysis) as the Regulatory Analysis in support of the SRP revisions noted above.

IV. Statutory Considerations

A. NRC Authority

The recommended changes to SRP Sections 3.9.3, 3.9.4, 5.4.6, 5.4.7, 6.3, 9.2.1, 9.2.2, 10.3, 10.4.7 are within the statutory authority of the NRC. Also, plant-specific safety assessments are within the statutory authority of the NRC.

B. Need for National Environmental Policy Act (NEPA) Statement

The proposed changes do not warrant a NEPA statement.

V. Summary and Conclusions

Based on the above discussion, the following actions are recommended:

(1) Issue the revised SRP Sections for forward-fit implementation.

(2) Issue NUREG-0927 as a technical findings document. This staff report summarizes the staff's assessment of water hammer in nuclear power plants.

(3) Ensure operator awareness and training with respect to avoiding water hammer through the use of the TMI Task Action Plan, Part I.C.5 and Part I.A.2.3, oprator training evaluation criteria under current development by the Licensee Qualifications Branch.

(4) Conclude current Operating License reviews through staff evaluations in progress.

REFERENCES

(1) U. S. Nuclear Regulatory Commission, NUREG-0582, "Water Hammer in Nuclear Power Plants," July 1979.

(2) Chapman, R. L. et al., "Compilation of Data Concerning Known and Suspected Water Hammer Events in Nuclear Power Plants," report prepared by EG&G Inc. for the NRC, NUREG/CR-2059, May 1982.

(3) Block, J. A. et al., "An Evaluation of PWR Steam Generator Water Hammer," report prepared by Creare Inc. for the NRC, NUREG-0291, June 1977.

(4) Anderson, N. and J. T. Han, "Resolution of SGWH in Operating PWR Plants - Partial Resolution of USI A-1," NUREG-0918, November 1982.

(5) Uffer, R. A. et al., "Evaluation of Water Hammer Events in Light Water Reactor Plants," report prepared by Quadrex Corp. for the NRC, NUREG/CR-2781, July 1982.

(6) Amico, P. and W. Ferrell, "Probabilistic Assessment of Unresolved Safety Issue A-1: Water Hammer," report prepared by Science Applications, Inc., September 1982.

(7) Andrews, W. B. et al., "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," NUREG/CR-2800, February 1983.

(8) Uffer, R. A. et al., "Evaluation of Water Hammer Potential in Preheat Steam Generators," report prepared by Quadrex Corp. for the NRC, NUREG/CR-3090, December 1982.

(9) Serkiz, A. W., "Evaluation of Water Hammer Experience in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1," NUREG-0927, February 1984.

APPENDIX A

SUMMARY OF PUBLIC COMMENTS RECEIVED

AND ACTION TAKEN

Appendix A Public Comments Received and Action Taken

Comment	NRC Staff Response
<u>Stone & Webster Engineering Corp. (S&W) (7/15/83):</u>	
The practice of including classical water hammers as occasional mechanical loads (per ASME B&PV Code, Section III) in piping stress analyses and piping system design support requirements should be maintained. Include a statement to this effect in the revised SRP sections, which had been issued for comment. Issue NUREG-0927 for information, along with the SRP revisions proposed.	The principal thrust of the S&W recommendations to maintain the practice of designing for water hammer loads in piping stress analysis and piping supports has been incorporated into NUREG-0927, and SRP 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," has been revised to reflect a need to consider water hammer in development of Design Specifications under the ASME Code requirements.
<u>Westinghouse Electric Corp. (W) (7/20/83):</u>	
Westinghouse recommended that the concept of design qualification of qualified designs within specified operating limits without requiring in-plant testing for each application be reflected in NUREG-0927 and the SRP. Westinghouse goes on to make the point that once a design has been qualified within specified operating parameter limits to preclude or minimize water hammer, adequate assurance exists that repeated applications of the same design within the same limits will have minimum potential for water hammer.	Westinghouse's points regarding verification of design adequacy relative to water hammer qualification testing (lab and first plant) as the basis for acceptance of repeated applications of the same design in subsequent plants is technically sound; it also is representative of the means by which new designs are introduced into SAR submittals. Although the staff agrees that qualified standard designs for steam generators may be available, the main feedwater and auxiliary feedwater systems and their controls are, in general, plant-specific and their design repeatability has been limited to identical plants at one site, or to standardized plants, if these systems are all within the scope of the Preliminary Design Approval. For these cases, it has been the practice of the NRC staff to <u>not</u> require that the testing in BTP ASB 10-2 be repeated on the identical plant. However, in general, the steam generator main and auxiliary feedwater system and controls and their operating procedures are

Comment	NRC Staff Response
	are not the same and even modest changes have resulted in steam generator water hammer (such as the Maine Yankee water hammer occurrence in 1983). The staff has concluded, therefore, that it is prudent to maintain BTP ASB 10-2 to demonstrate for new plants the absence of a proclivity to water hammer by preoperational testing.
W also commented on the feedwater control valve relative to feedwater line water hammer in PWRs and recommended that compatability of feed system components should be verified by the system designer.	The W comments regarding verification that the feedwater control valve is compatable with other systems and control requirements have been incorporated in NUREG-0927.
W made several editorial comments (see W comments Nos. 3, 4, 5, 6, and 7 in Enclosure 3).	NUREG-0927 has been modified as appropriate.
Reactor Controls, Inc. (RCI) (7/29/83):	
RCI addressed concerns regarding water hammer load during the scram function of the control rod drive (CRD) hydraulic system. RCI cited testing testing and analysis of the CRD system that has shown that water hammer loads require consideration. RCI recommended that either SRP 3.9.4 or SRP 4.6 be revised to include specifically the need to consider adverse dynamic loads such as water hammer in the CRD system.	RCI's recommendation to address potential water hammer in BWR CRD systems has been incorporated into SRP 3.9.4, "Control Rod Drive Systems," I&E will handle matters related to CRD water hammer in operating plants, and NRR will continue licensing review activities (per TIA 82-59).

Comment	NRC Staff Response
J. F. Doherty (8/7/83):	
Mr. Doherty commented that NUREG-0927 should address the likelihood of pipe break on certain pipes (particularly in the BWR) with given cracks, instead of assuming completely intact pipe and assessing damage to pipe supports and restraints.	Mr. Doherty's concerns related to BWR pipe cracks and pipe failures are of a broader nature than USI A-1 and are being addressed by current regulatory reviews related to reactor piping. NRC's Piping Review Committee is currently reviewing regulatory practices related to pipe cracks, pipe breaks, seismic design, and dynamic load/load combinations. Water hammer loads are one of several dynamic loads that will be reviewed with respect to experience and current design practices. Any recommendations for changes in regulatory requirements for the design of piping and piping supports will include consideration of their effect on the capability of piping systems to withstand water hammer loads.
Fluid Components, Inc. (FCI) (8/9/83):	
FCI raised questions regarding water hammer in BWR piping that has incurred intergranular stress corrosion. FCI also pointed out problems in the use of level sensors, as shown in Figure 2.4 and 2.5, and recommends the use of a thermal dispersion level switch. The thermal dispersion principle also can warn the operator of incipient voiding.	FCI's comments regarding water hammer in degraded BWR piping are similar to Mr. Doherty's. BWR pipe cracks and pipe failures are undergoing a major regulatory review from a more comprehensive safety viewpoint. See also the response above.
Review and Synthesis Assoc., S. H. Bush (11/1/83):	
Mr. Bush commented: "In my opinion, the Staff position with regard to water hammer is unrealistically optimistic. It works on the a priori assumption that previous water hammers in nuclear systems represent bound energy levels. This	The staff has neither assumed that previous water hammers represent upper bound energy levels, nor has the staff based its concluding evaluations on calculated energy loads. Rather, the probabilistic risk assessment (PRA) performed for USI A-1

Appendix A Continued

Comment	NRC Staff Response
is based on after-the-fact calculations of energy levels for a limited number of water hammer or water slugging events. These calculation yielded values that are a small fraction of the theoretical energy bound. While I do not anticipate cases of water hammer near the theoretical upper bound, I would not be surprised if some were to occur that are several times the existing calculated levels."	utilized a failure model developed from reported water hammer occurrences (or frequency) and water hammer occurrences (or frequency) and underlying causes. The PRA conservatively assumed that a predicted limiting event (such as pipe rupture or loss of safety system function) would occur even for those systems in which no disabling event was reported. The calculated change in public dose due to water hammer induced failures was negligible or very low (i.e., less than 25 man-rem/rx-yr). In addition, it should be noted that design basis accidents (DBAs) currently assume double ended guillotine breaks (DEGBs) for analyzing accident scenarios and the design of safety systems. Regardless of the postulated upper bound loads, pipe breaks more severe than a DEGB are not likely to occur.
Mr. Bush also stated: "The Staff also labors under some misconceptions that aren't necessarily valid. There is a concern for the global effect of seismic events while dynamic events such as water hammer are dismissed on the basis of redundancy, etc. In the following, I shall attempt to cite some positive aspects, followed with points where I find the Staff position unrealistically optimistic. I shall use an example, not necessarily valid that attacks their rather cavalier dismissal of such events....	Water hammer has not been dismissed on the basis of redundancy. The staff has carefully reviewed water hammer occurrences, their underlying causes, and corrective actions taken, and finds that certain design fixes (e.g., J-tubes in top feedring steam generators, keep-full systems and vacuum breakers in BWR systems) and operational awareness to avoid conditions leading to water hammer have significantly reduced the frequency of water hammer. The staff's findings are reported in NUREG-0927. Also, there is an important difference between global events (such as earthquakes) and water hammer events. An earthquake event can affect many lines and systems simultaneously. The effect of a water hammer occurrence is generally limited to a

Appendix A Continued

Comment	NRC Staff Response
"Events such as the H. B. Robinson and Turkey Point failures where dynamic loads resulting from relief valve closure blew the valves off the header were corrected by modifying an admittedly lousy weld joint design. Apparently, the industry learned a lesson because we haven't had any more such failures.	Inadequate designs are prone to failure. Corrective measures have been taken and proved to be effective.
"In the early 1970's, there were a large number of water slugging events where a valve was opened into a voided line and pipes were bent, hangers pulled out of the wall, etc. Techniques such as jockey pumps to fill the voided lines have markedly minimized such events. I'm doubtful they have totally eliminated them.	Keep-full systems have essentially eliminated water hammer occurrence in BWRs. Before jockey pumps were used, line voiding was a major cause of water hammer in BWRs. The staff supported industry's introduction of such systems and has recommended continued use of this design feature. The staff has stated on numerous occasions that total elimination of water hammer is not possible.
"Steam-water reactions, particularly those induced from the steam generator, have been virtually eliminated through installation of J-tubes. Again, I am not sure that steam-water reactions have been eliminated."	J-tubes have been shown effective in minimizing steam generator water hammer (SGWH) in some PWRs. It should be noted that not all PWRs have installed J-tubes and that some plants have operated without J-tubes since commencement of commercial operation without water hammer occurrence. NRC's plant review actions and conditional approvals for continued plant operation are summarized in NUREG-0918. Maine-Yankee was one such plant granted approval to operate. Following the January 1983 water hammer occurrence at Maine Yankee, J-tubes have been installed, operating procedures revised as needed, and a pre-operational test run to demonstrate absence of water hammer under loss of feedwater conditions. Again, total elimination of SGWH is not claimed.

Comment	NRC Staff Response
Mr. Bush went on to state: "As indicated, I am doubtful we have experienced feasible, more energetic water hammers. An examination of the industrial literature will reveal water hammers that catastrophically failed piping. These are still possible."	The staff is aware of deaerator systems in fossil fuel plants that have experienced numerous large water hammers. These events have not resulted in catastrophic piping failures even though severe damage has occurred in piping support systems and building structures. In these systems, piping cracks have occurred near piping anchors as a result of frequent large bending moments introduced by long segments of flexible piping systems. However, nuclear power plants are built to more stringent codes and standards than are normally employed in many industrial piping systems. The water hammer data base for USI A-1 represents more than 600 reactor-years of operating experience, over which time fewer than 150 water hammers have occurred. Since 1982 only two water hammers have occurred; one of these occurred in pre-operational testing. With the exceptions of the IP-2 feedwater line rupture in 1972 and the Maine Yankee feedwater nozzle crack in 1983, all other water hammer damage has been primarily to piping and equipment supports. Catastrophic water hammers have not occurred in nuclear power plants.
"Earlier incidents of water hammer or water slugging tended to be dismissed they 'only' pulled out all supports for a hundred feet or more of piping rather than failing the pipe. These support failures served as excellent energy absorbers minimizing damage to piping. Since then, we have gone in the wrong direction: namely, using large embedment plates, larger bolts, bigger lugs on the piping, etc. These measures almost certainly transfer the energy absorption to the pipe. In	Water hammer occurrences were not dismissed because they only pulled out all supports for a few hundred feet...," or for any other reason. The staff's concerns regarding water hammer occurrences in the early and mid 1970s are well documented in NUREG-0582 (July 1979) and resulted in water hammer being designated in unresolved safety issue (USI A-1). As noted above, design changes and operation awareness of water hammer potential have significantly reduced occurrence frequency (see also NUREG-0927).

Comment	NRC Staff Response
the opinion of the PVRC Steering Committee and some of the prestigious consultants for the NRC Task Group on Seismic Design, ASME III has gone in the wrong direction. The piping supports are too strong and the equipment supports too weak. Bosnak's group feels the same. Hopefully, we can change it in time, but that probably will be for new plants, not a backfit requirement. We may wish to make it a requirement along with requiring removal of excessive supports based on increased damping values.	
"Another problem pertains to the BWR IGSCC in larger pipes. The new appendix to ASME XI addresses the seismic case. I'm not sure we will see the same margins for more severe dynamic events. Ev Rodabaugh raised this concern and I intend to take it to ASME XI for consideration. There are other concerns we need to address with regard to this appendix. We probably acted too precipitiously, but there was a very real need.	Postulated changes in ASME codes should be addressed by the appropriate ASME Committee, not USI A-1. NRC staff members and consultants are active members of such Committees.
"Let me address another concern. The Staff dismisses water hammer on the basis of redundant systems. Let me postulate a relatively unlikely, but not impossible, scenario. It we had a steam-water reaction at the feedwater-steam generator interface, we could get a shock wave traversing the pipe. We may get one in the steam generator that could break several tubes. This is a classic initiator for pressurized thermal shock and all the redundant systems available won't help in a	The staff does not dismiss water hammer on the basis of redundant systems (see NUREG-0927). Moreover, the steam generator scenarior presented by Mr. Bush is not possible. The occurrence of a steam-slugging water hammer in the feedwater line requires that the feedwater sparger be uncovered. Under such circumstances, any pressure wave emanating from the feedwater line would be reduced to an insigificant magnitude upon entering the steam medium. Even if the sparger

Comment	NRC Staff Response
reactor pressure vessel with a high transition temperature. While I admit this scenario is unlikely, it points out the weakness in the staff position.	were not uncovered, the pressure wave would be greatly reduced by the orifice effect of the sparger holes or J-tubes, the extreme area difference between the steam generator and the feedwater line, and the air chamber effect of the steam generator tubes should the water. The effects on the steam generator tubes should be of the same order of magnitude or less than those resulting from the bubble collapse phenomenon, that occurs in steam generators following load rejection or turbine trip. These events occur far more frequently than steam generator water hammer and do not fail tubes.
"With regard to the ACRS questions, I suspect a meaningful PRA would be extremely expensive and much more difficult than the LLNL PRA's on pipe failure. Inputs would be virtually non-existent with the exception of events such as turbine trip and valve closure. Furthermore, the upper bound values would be virtually impossible to live with.	A PRA analysis dealing with the safety significance of water hammer has been performed (see SAI's report: "Probabilistic Assessment of Unresolved Safety Issue A-1, Water Hammer," January 1983), and the results have been summarized in NUREG-0993. As noted above, evaluation of water hammer events resulted in a negligible change in calculated risk.
"Summarizing, water hammer problems have been reduced but not eliminated; the Staff position strikes me as unduly optimistic; positive action may be necessary to correct the multiple problem of too many supports and too strong supports.	The staff's position as reported in NUREG-0927 and NUREG-0993 is not unduly optimistic; the staff makes no claim to total elimination of water hammer. The staff does support continued use of proven design concepts and operational procedures that have significantly reduced water hammer occurrence, and such water hammer considerations are reflected in the revised sections of the Standard Review Plan. In addition, an NRC Piping Review Committee is currently reviewing regulatory practices related to pipe cracks, pipe breaks, seismic design, and dynamic load/load combinations. Water hammer loads are one of several dynamic loads that will be reviewed with respect to experience and current

Comment	NRC Staff Response
	design practices. Thus, any recommendations for changes in regulatory requirements for the design of piping and piping supports will include consideration of their effect on the capability of piping systems to withstand water hammer loads.
E. C. Rodabaugh Associates Inc. (11/10/83):	
Mr. Rodabaugh referred to discussions with S. Bush and J. O'Brien and offers his opinion that water hammer is more of a concern than earthquake. Mr. Rodabaugh stated also that piping in newer plants is more restrained than in older plants that this may be of concern where the restraint-pipe attachment involves lugs welded to the pipe pressure boundary, should water hammer occur. He recommends additional research to better define the water hammer problem to (1) identify potential water hammers too severe to "design against"; (2) identify the role of plasticity in evaluation of pressure boundary failure; and (3) determine if water hammer tests could feasibly be added to dynamic-loading-of-piping programs.	Mr. Rodabaugh's concerns related to stiffer piping restraints and ASME Code Section XI, IWB-3640, will be considered by NRC's Piping Review Committee. His views related to water hammer concerns outweighing earthquake concerns derive from to his participation in structural analysis working groups (in the mid-West), while similar opinions have been expressed. Although he has recommended additional research, he also states that great urgency does not exist for water hammer research. NRC-RES will review his recommendations.

PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE SECRETARY

November 17, 1989

MEMORANDUM FOR: Teresa Neville, Acting Chief
Public Document Room

THRU: Sandy Showman, Chief
Correspondence and Records Branch

FROM: Andrew Bates, Chief
Operations Branch

SUBJECT: RELEASE OF DOCUMENTS TO PDR

Attached for placement in the PDR are copies of:

- SECY-89-122 - Resolution of Unresolved Safety Issue (USI) A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment"
- SECY-88-272 - Technical Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity
- SECY-84-119 - Resolution of Unresolved Safety Issue A-1, "Water Hammer"

These documents are being placed in the PDR at the EDO's request with concurrence by Commissioners' offices.

Attachments:
As stated

cc: EDO
GPA
DCS - P1-124

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