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ASSESSMENT OF LWR PIPING DESIGN LOADING
BASED ON PLANT OPERATING EXPERIENCE

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EXECUTIVE SUMMARY

The objective of this study has been to: (1) identify current Light Water Reactor (LWR) piping design load parameters, (2) identify significant actual LWR piping loads from plant operating experience, (3) perform a comparison of these two sets of data and determine the significance of any differences, and (4) make an evaluation of the load representation in current LWR piping design practice, in view of plant operating experience with respect to piping behavior and response to loading.

Specific recommendations are as follows:

- a. Water hammer and steam hammer loading phenomena are not analyzed to a sufficient level of sophistication in the initial design phase. In particular, the main steam, feedwater, residual heat removal, core spray, and high pressure coolant injection systems are affected by recurring water/steam hammer events.
- b. The frequency of pipe failures caused by various categories of operational vibrations, e.g., flow- and equipment-induced, justifies a change in design approach. The recommended approach varies from the current reliance on visual detection of excessive vibrations during testing and operation, towards one of prediction and prevention based on rational criteria.
- c. In view of the failure and damage records for LWR piping and their underlying causes, such as high-cycle fatigue, thermo-hydraulic shock loading, stress corrosion cracking, thermal cycling, etc., it appears that the considerable amount

of design effort oriented towards low-probability postulated accident loads (earthquakes, LOCA's, etc.) is partially made at the expense of the consideration of higher-probability operational loads. This is not meant to imply that the design efforts associated with low-probability accident loads should be reduced, but rather that appropriate consideration be given to higher-probability operational loads that are known to result in damage to systems.

- d. The number and magnitude of thermal transient cycles have shown a tendency to increase over values anticipated in the initial design. Detailed on-line monitoring of PWR feedwater lines has, for example, indicated considerable amounts of spontaneous thermal cycling in nozzle regions during low flow conditions. This, in view of the frequent development of cracks in the pipe nozzle regions of the RPV (BWR's) and the steam generators (PWR's), indicates a need for a more detailed study of the thermal cycling phenomena in terms of amplitude and number of cycles.

In addition, the potential use of nuclear plants to satisfy peaks in power demand and changes in operating procedures, resulting in more frequent power fluctuations, raises questions regarding the adequacy of the thermal transient definitions in current design criteria, which need to be further investigated.

- e. A general feedback mechanism to the design process is needed to ensure that process parameters (temperature, pressure, flow rate), piping response parameters (displacements, reaction forces), and vibration parameters (such as accelerations, frequency content, and amplitudes) from a full range of

testing and operational conditions, be monitored and utilized for verification and/or adjustments to the analytical load idealizations used in the design process. This feedback is the only reliable vehicle for development of more realistic load-predicting capabilities.

The importance of realistic load predictions is obvious for the original piping design in future plants, but also, and perhaps even more so, for the upgrading of existing plants, since in those cases it is essential to limit the modification efforts to items that are truly relevant.

1. INTRODUCTION

The material presented in this report represents a study performed by EDS Nuclear Inc. for Sandia Laboratories as a part of the Light Water Reactor Safety Technology Program within the Nuclear Power Development Division of the Department of Energy (DOE).

The purpose of the study has been to: (1) identify current LWR piping design load parameters, (2) identify significant, actual LWR piping loads from plant operating experience, (3) perform a comparison of these two sets of data and determine the significance of any differences, and (4) make an evaluation of the load representation in current LWR piping design practice, in view of plant operating experience with respect to piping behavior and response to loading.

A number of representative, typical piping systems for PWR's as well as BWR's were selected for the study. The selection included some of the more essential systems, such as the primary coolant circulation systems, the mainsteam and feedwater systems, the decay heat removal systems, and some of the engineered safety systems. The scope of the study in terms of systems included is thereby considered sufficiently broad for generic conclusions to be drawn with respect to a typical LWR.

The types of pipe loadings discussed in this report can generally be grouped into four major categories, as follows:

i. Traditionally known, well-defined operational loads.

Typical examples are gravity loads, internal static pressure loads, quasi-static thermal expansion loads, and certain fundamental thermal transient loads.

- ii. Anticipated operational loads of cyclic or rapid transient nature.

Typical examples are thermal shock loading, water/steam hammer loads caused by valve actuation, flow- and equipment-induced vibrations, and controlled blowdown loads. Certain extreme environmental loads, such as Operating Basis Earthquake (OBE) loads, also fall into this category.

- iii. Postulated accident loadings.

These may include pipe rupture (blowdown) and pipe whip loads, jet impingement loads, Safe Shutdown Earthquake (SSE) loads, and various loads initiated by a Loss-of-Coolant Accident (LOCA).

- iv. Unanticipated operational loads.

Typical examples are water or steam hammer loads caused by water slug impact, water entrainment in steam lines, flow discharge into initially voided lines, etc.

This particular grouping of loads serves to define the different methods available and suitable for the verification of the analytical load idealizations using actual operational or testing experience. Thus, the loads in category (i), which are primarily sustained operational loads, can be confirmed by direct comparison of anticipated and actual values of the load parameters themselves, i.e., pressures, temperatures, weights, etc. Cyclic and fast transient loadings (category ii) can be verified primarily from operational conditions by observations and measurements of the piping response to the loading, but also to some extent by direct

measurements of the loading parameters, e.g., time histories and/or peak values of the internal pressure during a water hammer event. Piping loads (category iii) occurring during postulated accidents lend themselves to testing only in a very limited sense, as existing plant operating experience has not included any design basis accidents (DBA) of LOCA or SSE type.* The characteristics of piping loads in this category, therefore, are based almost entirely on analytical evaluations in addition to limited scope tests in laboratory environments. Finally, unanticipated operational loads (category iv) have mainly been identified through observed abnormal piping behavior or actual damage and failure. In cases where damage has occurred to the pipe, or pipe supports, an approximate lower bound evaluation can often be made of the load magnitude to which the components have been subjected. This magnitude is often so high that it is infeasible to design the affected systems and components to withstand these loads. Instead, the engineering efforts have been directed towards the elimination of the loads and assurance against recurrence. A typical example is water hammer loads of the water slug impact type, that occur in feedwater lines following the draining of feedwater sparger rings in steam generators. In this case, modifications such as relocated sparger outlets and reduced horizontal feedwater line portions upstream of the steam generator have reduced significantly the frequency of occurrence of the water slug impact phenomenon.

A five-task program was defined for the performance of the study. The five tasks defined correspond sequentially to Sections 2 through 6 of this report, and are briefly summarized in the following paragraphs.

*Except as noted briefly under Section 3.2.

- Tasks:
1. Identify and Categorize Significant Load Events.
 2. Identify Significant Piping Loads Experienced in Operating Plants.
 3. Identify Piping Loads Considered in Current Design Practice.
 4. Compare Actual Operating Loads with Design Loads.
 5. Evaluate Current Piping Design Loads in View of the Findings From Tasks 1 Through 4.

Task 1 (Section 2)

A summary was prepared of significant anticipated piping loads for a selection of representative systems in PWR's and BWR's, as compiled primarily from NSSS and plant SAR's. The purpose of this summary is to serve as a starting point and a frame of reference for the remaining four tasks of the study.

Task 2 (Section 3)

The identification of significant piping loads from operating plant experience has been based primarily on published accounts. The main general source used for this task has been "Nuclear Power Experience" (NPE), published by NPE Inc., Encino, California. This information system, which is updated monthly, provides descriptive accounts in a flexible format of significant issues related to nuclear power operating experience. Items included in the NPE system are selected subjectively by the NPE staff and are based primarily on correspondence between licensees and the NRC, and on published articles.

The possible incompleteness of the data sources for abnormal events does not constitute a significant limitation for the present study,

since a comprehensive, strict statistical evaluation is not attempted. The purpose of this study is, rather, to assess the load considerations in current piping design practices using operating experience as the benchmark, where any indication of insignificant discrepancy having a generic tendency gives rise to a concern and a potential need for adjustments in the design process to represent actual conditions more realistically. One limitation of the present study that is of significance is the absence of operational data to verify those aspects of load representation in current design practice that are adequate, conservative, or over- conservative. The lack of such data, i.e., typically routine monitoring during system operation, gives the study an unnecessarily negative leaning towards inadequacies in the designs, leading to abnormal events such as cracks, leaks, and complete failures, which are the topics concentrated upon, for example, in the NPE records.

The more important findings under Task 2 are associated with: thermal expansion and thermal transient effects, steam/water hammer loads, flow- and equipment-induced vibrations, and blowdown loads.

In an effort to determine the possible causes for the frequent crack development in feedwater piping near steam generators and auxiliary feedwater connections, a detailed temperature monitoring program was performed at a number of PWR's during 1979. The results from the monitoring program at Millstone 2 indicated that, during low flow conditions (auxiliary feed), significant thermal stratification, reaching temperature differentials of 400° F, could exist in the local region of the auxiliary feed connection towards the steam generator. These temperature differentials were, further, found to be of cyclic, transient nature with periods of somewhat less than one hour. It was determined analytically that

these particular cyclic temperature differentials, which typically are not considered in the design process, were sufficient to cause the cracks in question. This example shows the considerable practical value inherent in the "closed-loop" design concept, where detailed, well-planned on-line monitoring is used to obtain an optimal amount of relevant information, which is then fed back into the analytical evaluation for verification purposes.

With regard to water/steam hammer effects experienced in operating plants, it is noted that:

- Only a limited number of LWR piping systems are typically designed/analyzed explicitly for water/steam hammer effects, while in actual operating experience a considerable number of systems have been subjected to this type of loading.
- Water hammer events have often been identified first during routine inspection, long after the actual occurrence, when gross permanent pipe or support deformation has been found.
- The piping loads caused by certain categories of water hammer events, e.g., water-slugging phenomena, are of such high magnitude that it is not feasible to design the piping systems to accommodate their effects within code-allowable stress limits. Instead, the engineering efforts have been oriented towards the elimination or mitigation of the water slug creation by implementing changes to the design and operational procedures.

Floor- and equipment-induced vibrations leading to high cycle fatigue failures represent the dominating overall failure cause for LWR piping. The majority of these failures have occurred in small diameter piping. A study (Reference 5) performed on piping

failures indicates that approximately 25% of a total of 234 failures considered were caused by high cycle fatigue due to flow- and equipment-induced vibrations. More recent records (Reference 6) indicate that the same dominance of high cycle fatigue as a failure mode has continued to date. Excerpts from these records for a large number of individual cases are included in Appendices B and C to this report.

Task 3 (Section 4)

A considerable amount of data, representative for the types of load definitions typically considered in current LWR piping analysis, has been provided in Appendix D for purposes as outlined below:

- Provide a comprehensive basis for comparison between load definitions as used in typical current design approaches and indications of significant loads obtained from operating experience. (See Section 5.)
- Provide load data (e.g., with regard to thermal transients) in sufficient detail to show the types of simplification typically used in current analyses, for comparisons with the often very complex load phenomena experienced in actual operation. (See e.g., cases involving transient behavior of strong thermal qualifications, Section 3.2.2.)

Tasks 4 and 5

The comparison in Task 4 of current piping design loads versus piping loads experienced in actual operation has resulted in the following overall conclusions and recommendations:

- a. Water hammer and steam hammer loading phenomena are not analyzed to a sufficient level of sophistication in the initial design phase. In particular, the main steam, feedwater, residual heat removal, core spray, and high pressure coolant injection systems are affected by recurring water/steam hammer events.
- b. The frequency of pipe failures caused by various categories of operational vibrations, e.g., flow- and equipment-induced, justifies a change in design approach from the current reliance on visual detection of excessive vibrations during testing and operation, towards one of prediction and prevention based on rational criteria.
- c. In view of the failure and damage records for LWR piping and their underlying causes, such as high-cycle fatigue, thermo-hydraulic shock loading, stress corrosion cracking, thermal cycling, etc., it appears that the considerable amount of design effort oriented towards low-probability postulated accident loads (earthquakes, LOCA's, etc.) is partially made at the expense of the consideration of higher-probability operational loads. This is not meant to imply that the design efforts associated with low-probability accident loads should be reduced, but rather that appropriate consideration be given to higher-probability operational loads that are known to result in damage to systems.
- d. The number and magnitude of thermal transient cycles have shown a tendency to increase over values anticipated in the initial design. Detailed on-line monitoring of PWR feedwater lines has, for example, indicated considerable amounts of spontaneous thermal cycling in nozzle regions during low flow

conditions. This, in view of the frequent development of cracks in the pipe nozzle regions of the RPV (BWR's) and the steam generators (PWR's), indicates a need for a more detailed study of the thermal cycling phenomena in terms of amplitude and number of cycles.

In addition, the potential use of nuclear plants to satisfy peaks in power demand and changes in operating procedures, resulting in more frequent power fluctuations, raises questions regarding the adequacy of the thermal transient definitions in current design criteria, which need to be further investigated.

- e. A general feedback mechanism to the design process is needed to ensure that process parameters (temperature, pressure, flow rate), piping response parameters (displacements, reaction forces), and vibration parameters (such as accelerations, frequency content, and amplitudes) from a full range of testing and operational conditions, be monitored and utilized for verification and/or adjustments to the analytical load idealizations used in the design process. This feedback is the only reliable vehicle for development of more realistic load-predicting capabilities.

The importance of realistic load predictions is obvious for the original piping design in future plants, but also, and perhaps even more so, for the upgrading of existing plants, since in those cases it is essential to limit the modification efforts to items that are truly relevant.

2. CATEGORIZATION OF SIGNIFICANT PIPING LOAD EVENTS

Based primarily on NSSS Reference Safety Analysis Reports (SAR's), plant-specific SAR's, and, to a limited extent, on plant operating experience, a set of matrices has been developed for the identification of potentially significant loadings on representative Light Water Reactor piping systems.

The lists of significant plant events (Tables A.1.1 and A.2.1 of Appendix A) are based on NSSS Reference SAR's. The piping systems selected include, separately for PWR's and BWR's, representative portions of the primary coolant system, the emergency core coolant system, the reactor makeup water and purification systems, and the mainsteam and feedwater systems.

The matrices of load group vs. plant condition for individual piping systems (Tables A.1.2 - A.1.10, A.2.2 - A.2.8) were developed to have generic, i.e., not plant-specific, validity, which has necessitated multiple categorization with respect to plant conditions in some cases where differences exist between plants. Other reasons for multiple categorizations are explained as follows:

- a. Some load events, which viewed separately are placed in a plant condition category corresponding to a high frequency of occurrence, e.g., Upset, may occur in conjunction with lower frequency of occurrence events, e.g., Faulted. This situation may be the more limiting one for certain components despite the less restrictive acceptance criteria under Faulted conditions, and the high frequency of occurrence events have therefore been entered under Upset as well as Emergency and Faulted conditions.

- b. Loads such as flow- or equipment-induced vibrations which may be significant during steady-state operation have been entered under all potential plant conditions.
- c. Loads which normally are not taken into account (not required by code) in current piping qualification by analysis, such as displacement effects under Faulted conditions, have been included as significant loads despite their exclusion in code rules.

The loads found in the system matrices should be viewed as generically potential loads for typical piping systems in typical PWR and BWR plants. The magnitude, and thereby significance, of the individual loads in a particular application will be dependent on the piping configuration and function, and the operational procedures, at the particular plant. Loads showing significant variation between different plants are those due to equipment-induced vibration, water/steam hammer, and other thermal-hydraulic effects.

Piping loads due to jet impingement from a broken adjacent line have not been included in the load matrices. This is not meant to imply that this type of loading is not significant. Jet impingement loads are potentially significant for all pipe systems considered, but are entirely dependent on particular plant/system conditions, such as piping configuration, extent of physical separation, etc.

3. SIGNIFICANT LOAD EVENTS EXPERIENCED IN OPERATING PLANTS

3.1 Definition of Data Sources

The general source used in this section for the identification of significant loads on LWR piping systems as experienced in actual operation is "Nuclear Power Experience" (NPE) (Reference 6), published by Nuclear Power Experience Inc., Encino, California. This information system, which is updated monthly, provides verbal descriptions in a flexible format of significant issues related to nuclear power operating experience. Items included in the NPE system are selected subjectively by the NPE staff and are based primarily on correspondence between licensees and the NRC, and on published articles. The methods of selection, and the information sources available to the NPE staff, are of such a nature that the completeness of the items cannot be guaranteed, which thereby invalidates these records as a basis for a strict statistical evaluation. In addition, the NPE records suffer the same weakness as do the NRC "Gray Books," the Edison Electric Institute, and the Nuclear Plant Reliability Data System with regard to completeness. The shortcomings of the various existing data-gathering systems are discussed in great detail in Reference 13. Inconsistencies in the data base may, for example, occur due to differences in licensees' candor in reporting "significant events", in their interpretation of categorization of systems, and in their interpretation of the cause and effect of an individual event.

The possible incompleteness of the data sources for abnormal events does not constitute a significant limitation for the present study, since a comprehensive, strict statistical evaluation is not attempted. The purpose of this study is, rather, to assess the load considerations in current piping design practices using operating experience as the benchmark, where any indication of significant discrepancy having a generic tendency gives rise to a concern and a potential need for adjustments in the design process to represent actual conditions more realistically. The purpose thus defined leads up to a limitation of the present study that is of significance, namely the absence of operational data to verify those aspects of load representation in current design practice that are adequate, conservative, or overconservative. The lack of such data, i.e., typically routine monitoring during system operation, gives the study an unnecessarily negative leaning towards inadequacies in the designs, leading to abnormal events such as cracks, leaks, and complete failures, which are the topics concentrated upon, for example, in the NPE records.

It is anticipated that the installation of piping monitoring instrumentation, which has increased in recent years, will provide a data base for a more balanced assessment of pipe load parameters in the near future.

In addition to the NPE records, a number of specialized reports published by the NRC, EPRI, NSSS vendors, and others, which cover particular aspects of piping problems, have been used as a basis for the study; and these are listed in Section 7.

3.2 Significant Load Types Experienced in Operation

This subsection summarizes available published accounts of LWR operating experience with respect to the full range of structural loads to which piping systems can be subjected. The limitations lie, not in the intent, but in the availability of data and in the fact that certain postulated or otherwise anticipated loads have yet to occur to any significant extent. Examples of the latter are large Loss-Of-Coolant Accidents (LOCA) and strong-motion seismic excitations, and these accident-type loads are consequently not included in the summary. (We note that the Fukushima plant in Japan was subjected to significant seismic excitations during the Miyagi earthquake of June 12, 1978. It has been reported that the four 500 MW and one 800 MW BWR units, which have an OBE design level of 0.15 g, operated through the earthquake without tripping and without undergoing significant damage. The peak ground acceleration recorded at Fukushima was 0.12 g with about 30 seconds of strong motion. We have not, however, been able to locate any evidence of the piping loads resulting from this event and therefore exclude it from the study.)

3.2.1 Sustained Operational Loads

Prior to their application to nuclear power plants, the design of process piping (e.g., in fossil fuel plants, chemical plants, and in the petroleum industry) was based exclusively on the consideration of gravity loads, internal operating (static) pressure, and on thermal expansion loads. This is reflected also in the early

editions of the piping codes specified for use in nuclear application, which are direct outgrowths of the non-nuclear piping codes with very modest modifications.

The considerable amount of experience with regard to the significance and effects of these loads that has been accumulated over a period much longer than the existence of nuclear power, and the uncomplicated analytical methods required for an accurate prediction of the structural response, may suggest that no further improvements are needed, or can indeed be made, to the consideration of these loads in the design process. This is true to a large extent, but in view of certain questions that have arisen in connection with piping problems at operating plants and in view of the incorporation of these fundamental load cases in the overall design load combinations, the following items are of interest to note:

- a. The sustained operating loads, to which the piping systems are known to be subjected largely without interruption throughout the life of a plant, should be given particular attention with respect to accuracy in view of the load condition. Relatively minor discrepancies between actual and assumed load parameters (steady-state temperatures and pressures) can have significant effects due to the duration of the load and to the fatigue life dependence on the steady-state stress level upon which cyclic stresses are superimposed.

- b. It should be recognized that the tolerances that can be satisfied for loads of occasional nature, the magnitude of which can be estimated approximately at best, are of a different order of magnitude than those that should be imposed on steady-state loads that can be accurately determined. Long-term routine monitoring of basic parameters such as temperatures and pressures in actual operation is needed as a feedback verification of values used in the design process.

- c. As a result of detailed temperature measurements made at a number of PWR's (e.g., Millstone 2, Cook, Ginna, Robinson) in an effort to determine the possible causes for crack development in feedwater (FW) piping close to steam generators and auxiliary feedwater connections, it was found that significant thermal stratifications could exist in the piping during hot standby, auxiliary feed, main feed, and power escalation (Reference 6, March 1980). The stratification during low flow conditions was reported to range up to a value of 400°F, with the temperature increasing from the bottom of the piping towards the top. With increasing FW flow, the temperature gradually became uniform throughout the piping.

Thermal stratification of this type introduces a bending mode of deformation on the piping, or, if restrained from bending, introduces a bending moment which is not taken into account in the analytical design phase due to the axisymmetric temperature

distribution typically assumed for steady-state thermal expansion load cases, as well as thermal transient evaluations. The unaccounted-for effects of thermal stratification could be considerable, especially when the magnitude is as high as in the above-quoted example. Thermal stratification effects could be potentially significant in other large-diameter water systems under low flow conditions. This indicates the need for more extensive and detailed system monitoring, as well as further evaluation of the effects of the stratification determined.

- d. In connection with the evaluation of the frequently occurring cracks and leaks in BWR recirculation valve bypass lines (e.g., Dresden, Quad Cities, Millstone 1, Monticello, Brunswick 2, Hatch 1), it has been hypothesized that one contributing factor may have been the thermal expansion stresses caused by the temperature differences between the main recirculation line and the bypass line when the system was operated with the bypass valve closed (Reference 6). It was recommended following the evaluation that the system be operated with open bypass valves, partly to reduce the thermal expansion stresses in the region, and partly to reduce the risk of corrosion in the stagnant water environment in the bypass lines.

In addition to a number of other possible factors contributing to the bypass line failures (e.g., stress-assisted corrosion cracking, high-cycle

fatigue caused by mechanical vibrations, low-cycle fatigue, residual stresses), the existence of significant fit-up stresses was suggested. A special study (Reference 12) was performed to determine the significance of the fit-up stresses based on actual strain measurements made during the removal (that is, release of fit-up forces) of a bypass line. The maximum fit-up stress value found, based on strain measurements, was 2.3 ksi and was located in the outlet side weldolet region. The analytical model that used the strain measurements to compute all unknown restraint forces in the bypass line, including gravity load effects, gave a maximum calculated stress of 4.3 ksi at the inlet side elbow. The study concluded that these stresses, which were lower than previously reported values from in-plant measurements but similar to values reported by General Electric, were insignificant compared to other stresses present.

3.2.2 Thermal Transient Loads

A thermal transient is, in the piping analysis context, an event involving a temperature change at a sufficiently high rate to invalidate the assumption of uniform temperature through the pipe wall. The true temperature distribution following a temperature change is non-linear, and the nonlinearity increases with rate of temperature change.

The expression "thermal cycling" is usually meant to describe a temperature oscillation where cold-hot-cold (C-H-C), for example, constitutes one cycle. A rapid, large amplitude, full thermal cycle, C-H-C, can potentially create a very large stress range on the inside surface, going from steady-state to high compression and then back down through zero to high tension. If the thermal cycle is rapid, typically only a very thin layer of the inside surface would be subjected to substantial stress variations, in the form of peak stresses. The main effect of this variation is potential fatigue damage.

The cycling of the type described has been shown to occur frequently in piping close to nozzles (e.g., RPV or steam generator nozzles) during periods of low flow, or no flow, in the branch if a large temperature difference exists between branch fluid and vessel fluid. This condition was monitored closely in feedwater lines in the regions of the steam generator nozzle, and close to the auxiliary feedwater connection to main feed in certain PWR's during 1979 (Reference 6, item PWR-2, VI.E.258; see also Section 3.2.1c of this report). Thermocouple instrumentation in conjunction with accelerometers and strain gauges had been installed for an on-line monitoring program at Millstone 2, in order to determine the cause for linear indications in the circumferential direction that had been detected in the Heat Affected Zone (HAZ) of the safe-end to pipe and pipe to elbow welds in the horizontal run of piping connected to the steam generator FW nozzles. The depth of the cracks was typically about 100 mils, which corresponded to the experience from seven other plants supplied by the same NSSS vendor. It was

determined from thermal fatigue analyses that the cracks were caused by the thermal cycling, but contended that the cracks would not propagate beyond 100 mils as a result of the rapid thermal cycling identified from the on-line monitoring. Plastic instability analyses were also performed with the inclusion of all other design loads. The results from these analyses showed that a through-wall crack longer than 25 inches, or a complete, circumferential crack penetrating 70% of the wall thickness, would be required to cause pipe rupture.

In addition to the purely technical items regarding crack growth, failure load, etc., one general conclusion was drawn from the experience of the cited study (as reported in Reference 6), namely that: "The implementation of on-line monitoring would provide detection of any unanticipated condition which would contribute to crack growth."

This conclusion can be generalized in validity for numerous applications of on-line monitoring of piping parameters, other than those indicated in this particular event.

This cited example of on-line monitoring, coupled with analytical evaluations, is of particular interest for the present study, primarily for the following reasons:

- a. It provides an excellent example of the practical value inherent in the "closed-loop" design concept where detailed, well-planned on-line monitoring is used to obtain an optimal amount of relevant

information which is then fed back into the analytical evaluation for verification purposes.

- b. The specific results with regard to the thermal cycling and its effect on crack development provide relevant and valuable information not only for the particular feedwater cracking problem, but also with regard to the enhancement of the understanding of this type of phenomenon.
- c. The information provided by the monitoring, with regard to considerable thermal stratification that was detected, is, again, valuable for the general understanding of thermal problems in piping systems. (See further Section 3.2.1.c in this report.)

Other reported abnormal events that have been associated with thermal transients provide much less detail about the event and conclusions than does the case cited above. An example from Reference 6 is item no. 105 in Table C.3 (Appendix C), which specifically states that thermal transient stresses caused a leak in a welding neck flange. This flange was located on the RPV head spray line close to the RPV nozzle.

It may be hypothesized, in view of the frequent occurrence of "stress corrosion cracks" in the regions of BWR vessel nozzles and connected piping, that severe thermal cycling may be of considerably greater importance for the crack development than previously has been believed. Further,

detailed temperature monitoring and stress evaluation, similar to what was cited above for PWR FW nozzle regions, are needed in order to refute or verify such a hypothesis.

One further concern of interest, that more and more frequently is expressed with regard to thermal transient effects, is the increasing tendency in number of cycles of postulated plant events, and, on a microscale, the indications, for example from monitoring programs such as that cited above, of more frequent, local thermal cycles in stagnant or near-stagnant branches near the nozzle regions. These tendencies may have substantial, detrimental effects on the theoretical fatigue life and indicate a need for further detailed assessment.

Reference 3 provides a comprehensive summary of statistical information with regard to observed frequency of occurrence of anticipated transients in domestic and foreign WR's.

3.2.3 Water/Steam Hammer Loads

A comprehensive review of water hammer and steam hammer events observed during testing or operation of LWR's through approximately mid-1978 has been made by the NRC (References 1, 2, and 4).

The scope of the review, as reported in Reference 2, was defined by the generalized use of the term "water hammer" to include transients involving steam flow (steam hammer) and two-phase flow (e.g., water entrainment in steam lines, steam bubble collapse) in addition to the

classical water hammer transients such as those involving valve closing and pump start-up in solid water systems.

A total of 119 water hammer events, reported to have occurred in various LWR piping systems prior to mid-1978, are listed in Reference 2. Ninety-six events (or 81%) were reported to have caused damage to the pipe, supporting system, or other components.

The structural loadings on piping systems resulting from water hammer events can be viewed as consisting of two parts, one of which is the local effect in terms of hoop stresses and longitudinal stresses in the pipe wall due to the internal pressure pulse. The second part is that caused by the rapid movement of the pressure pulse along the pipe, giving rise to net global forces on the pipe line at flow restrictions and changes of direction. It is this latter part of the loading which typically is the more significant one from a potential damage point of view. The magnitude of the dynamic forcing functions varies over a wide range, from insignificant to certain damage, depending on type of event (initiating cause), pipe size and configuration, initial pressure and temperature, valve characteristics, etc. Water hammer phenomena experienced in testing conditions or actual operation have been classified in accordance with the following list of potential initiating mechanisms (Reference 2):

1. Pump start-up with inadvertently voided discharge lines
2. Expected flow discharge into initially empty lines

3. Valve opening, closing, and instability
4. Check valve closure and delayed opening
5. Water entrainment in steam lines
6. Column separation
7. Steam bubble collapse and mixing of subcooled water and steam from interconnected systems
8. Slug impact due to rapid condensation
9. Pump start-up, stopping, and seizure

Table 3.1 contains a matrix of a total of 119 water hammer events, categorized with respect to the system in which they took place and with respect to type of phenomenon as defined above. The characterization of a particular event is in general based on a reconstruction after the fact. That is, based on visible or audible observations, if any, at the time of the event, recordings of plant parameters such as power level, water levels, flow values, valve operation, etc., and the amount and type of permanent damage to the system, an assumed sequence of events resulting in the water hammer may be reconstructed. In fact, a number of water hammer events have been identified first during routine inspections, long after the actual event, when gross permanent pipe or support deformation has been found. Reference 6 reports numerous instances of this type, where the main entry (typically LER) consists of reported damage to pipe, supports, valves, or

System Water Hammer Type	Main Steam	Main FW	Aux. FW	MS S/V	Press. Relief Valve disch.	RHR	Core Spray	HPCI	LPCI	SIS Accum.	Misc.	- SG Blow- down Line
Steam Bubble Collapse and mixing of sub- cooled Water Steam from interconnected systems		3				5		2		2	1 Note(2)	1
Pump Startup						1		1				
Check Valve Closure and delayed opening	4						1				4 Note(3)	
Water Entrainment in Steam Lines			2 (1)					7			2 Note(4)	
Valve Opening, Closing and Instability	10	21									1 Note(5)	
Water Slugs from FW Sparger Rings in PWR's		23										
Pump Startup with Inadvertantly Voided Discharge Lines		1				9	6	1	1		5 Note(6)	
Expected Transients in Initially Empty Lines				1	1							
Not Identified						1 Note(1)						1

Table 3.1 Number of Reported Water Hammer
Events Classified by Type and System

Notes:

- (1) Turbine drive steam lines.
- (2) Plant steam heating system.
- (3) Cooling system for Reactor Auxiliaries; Recirculation Spray; Containment Spray System.
- (4) RCIC system; Isolation Condenser.
- (5) CVC System
- (6) SSI Discharge; Emergency Equipment; Cooling Water; RHR Service Water; Service Water

Table 3.1 (cont'd.)

instruments, and where the cause for the damage is listed as "probable earlier water hammer", or similar. Examples of such instances included in Appendix C are: item 51 in Table C.2, and items 2, 9, 62, 71, and 136 in Table C.3.

In general, the normal plant monitoring systems are not sufficiently sensitive to provide an accurate picture of the pipe loading characteristics from the water hammer in terms of pressure pulse magnitudes and travelling speeds. Estimates of load magnitudes are instead often made based on the amount of permanent deformation of the pipe, supporting steel members, or external components impacted by the moving pipe. These estimates are by necessity approximate and provide at best a range of possible equivalent static forces at the location of the damage.

The most severe water hammer events have been those caused by water slugs created by rapid condensation of steam in the feedwater sparger rings during recovery of the sparger following situations with low steam generator water levels. A total of 17 events of this type, out of the 23 reported, was associated with damage, which in some cases was severe.

The water hammer event at Indian Point 2 in November, 1973, resulted in severe damage to the feedwater piping as evidenced by a permanent localized bulge in the pipe caused by overpressure, a 180° fracture of the pipe (18 inches normal diameter, 1-inch wall thickness) at the containment penetration end plate weld, permanent deformation of containment lines at penetrations, and

failure of several pipe supports. Based on the existence of a permanent bulge in the pipe, indicating hoop stresses in excess of the yield strength, a simple estimate of minimum required equivalent static pressure resulted in a value of 5000-6000 psi (Reference 2). This pressure value is consistent with the experience from water hammer tests performed at the Tihange plant in Belgium (Reference 10), which contains the only known pressure trace record from an actual plant event of the feedwater sparger water slugging type. This pressure trace, which is reproduced in Figure 3.1, shows pressure spikes of approximately 400 bars (5800 psi). Other events involving substantial amounts of damage are reported in References 2 and 4. The loads typically caused by the water slugging phenomenon are of such magnitude that it is not feasible to design the piping systems to accommodate their effects within code-allowable stress limits. Instead, the engineering efforts have been oriented towards the elimination or mitigation of the water slug creation by implementing changes to the design and operational procedures. Examples of modifications performed to date, with varying degrees of success, are:

- Change of bottom discharge from the sparger to top discharge via J-shaped tubes.
- Minimization of horizontal runs of feedwater pipe from the steam generator nozzle.
- Limitation of feedwater flow, while the feed ring sparger is uncovered, to certain critical threshold values.

FIGURE 3.1 Reference 10
Pressure Histories in Feedwater Pipe at Tihange

- Prompt reestablishment of feedwater flow following events which uncover the feed ring.

In addition to events involving feedwater water slugging as discussed above, the most serious water hammer concerns based on reactor operating experience are those caused by pump start-up with inadvertently voided lines in ECC and RHR systems of BWR's, and main feedwater line transients caused by flow control valves in BWR's and PWR's (Reference 2).

Water hammer events caused by pump start-up with inadvertently voided discharge lines due to air entrapment or draining have been experienced in a large number of instances, most notably in BWR core spray and RHR systems. Table 3.1 includes 20 events of this type, 15 of which were associated with core spray and RHR systems. Five additional events, associated with the same two systems, are included in Reference 6 as having occurred between 1977 and 1979. The latest event known to us at this time, assumed to be caused by pump discharge into a voided line, was identified in March, 1980, in the Reactor Water Cleanup System of a BWR Mark I facility. In this latter case, as in the majority of the 25 cases mentioned above, the water hammer event was identified by hardware damage found during routine inspections. The damage from essentially all of these water hammer events consisted of deformed or failed pipe supports and snubbers, and in some cases permanently deformed or cracked piping. No detailed pressure tracing has, to our knowledge, been recorded during the above types of events under actual plant conditions. Load estimates are

therefore limited to approximate calculations to determine the force magnitudes required to cause the damage observed. A large number of LWR piping systems are potentially susceptible to suffering water hammer loads due to voided pump discharge lines. Since it in general is infeasible to design all these lines to withstand the full range of water hammer forces, the approach has been to make design and operating procedure modifications to ensure that the discharge lines are properly filled and vented prior to pump start-up. Typically, jockey pump systems have been installed to maintain full lines by pumping suppression pool water to the RHR and core spray lines downstream of the pump discharge line check valves. Water hammer events caused by controlled valve opening and closing, e.g., turbine stop valve closure, feedwater regulating valve operation, mainsteam or feedwater isolation valve closure, safety and relief valve opening, etc., fall into the category of anticipated, and therefore designed-for, events. Despite the fact that this type of event generally is considered in the design, a number of damaging water hammers have occurred as a result of valve operation, primarily in mainsteam and feedwater systems. Table 3.1 contains a total of 32 events caused by valve opening, closing, or instability. All but one of these events were associated with the mainsteam and feedwater systems, with an approximately equal distribution between BWR's and PWR's. The majority of the mainsteam and feedwater events, 25 out of 31, caused some level of damage (Reference 2). The damage, as summarized in Reference 6, typically has

consisted of branch line cracks and failures, snubber failures (broken rods, permanent lock-ups), valve operation failures, and deformations to hangers and other supports.

Partly as a result of the considerable number of experienced water hammer type events, most of which involve certain amounts of unexplained, or insufficiently understood, physical phenomena, programs for instrumentation of critical lines have been initiated at some plants for future monitoring of piping behavior. The instrumentation includes accelerometers, support/snubber load cells, displacement transducers, etc. Reference 6 indicates that such instrumentation has been installed, or is being planned, on feedwater lines at, for example, Zion 2, Beaver Valley, Robinson 2, and Ginna. Despite the relatively high frequency of occurrence of water hammer type events caused by valve closure or valve instability in mainsteam and feedwater systems, no detailed actual plant records appear to be available of associated physical parameters such as pressure and temperature traces, and pipe response parameters such as displacements, pipe strains, and snubber/support reaction histories. One of the few examples available of this type of data is the pressure traces shown in Figure 3.2, which were obtained from a turbine trip test performed at Peach Bottom (BWR) in 1978 (Reference 7), and correspond to a relatively mild steam hammer type phenomenon following the closure of the turbine stop valves.

The explicit, detailed consideration of water hammer loading in current design practice is typically limited

FIGURE 3.2 Reference 7
Peach Bottom-2 EOC2 Test Turbine Trip Test TT2
Transient Pressure Measurements

CURVE NO.	MEASURED VARIABLE	PLOT SCALE (UNITS/DIV)	INITIAL VALUE
1	TURBINE STOP VALVE POSITION	100%	100%
2	TOTAL BYPASS POSITION	100%	0%
3	CORE EXIT PRESSURE	50 psi	993.0 psia
4	REACTOR VESSEL PRESSURE	50 psi	986.6 psia
5	STEAM LINE A PRESSURE	50 psi	977.0 psia
6	STEAM LINE D PRESSURE	50 psi	983.8 psia
7	TURBINE INLET PRESSURE A	50 psi	970.0 psia
8	TURBINE INLET PRESSURE D	50 psi	966.4 psia

to various valve closing events in mainsteam and feedwater systems, pressurizer safety/relief valve operation, and safety/relief valve operation for BWR mainsteam systems. One reason for the omission of explicit consideration of water hammer loadings in other systems that are potentially subjected to such loading (see Table 3.1) is the lack of adequate understanding and experience-based quantification of the physical phenomena involved. A relevant data base from plant operating experience, providing detailed information about the initiating transients, the pressure development during the event, and the pipe response generated by the event, is imperative for the development of a rational design approach. As mentioned before, it is recognized that the design approach may be aiming towards either the elimination of the potential loading, or, if this is not feasible, towards a pipe-restraint system which can withstand the loading. For these purposes, the special instrumentation added in recent years to systems with histories of problems will be invaluable for providing the operational data, so far lacking in sufficiently general validity to serve as a basis for the verification or adjustment of current design practices.

3.2.4 Flow- and Equipment-Induced Vibrations

According to a study performed in 1977 (Reference 5), the dominant mode of pipe failure for both BWR's and PWR's is high-cycle fatigue caused by mechanical vibrations.

(Note: The term "pipe failure" as used in Reference 5 is defined to include flaws which range in severity from local cracking, which may or may not involve leaks, to complete rupture of the pipe wall.)

This conclusion, which differs from earlier pipe failure studies in which low-cycle fatigue has been interpreted as the dominating cause, was based on the operating record of all LWR's in the U.S. over the 16-year period from August, 1960, to August, 1976.

Out of a total of 234 pipe failures considered in the study, 25.1% were stated to have been caused by high-cycle fatigue. For BWR's 19% of all failures were attributed to this cause, and for PWR's the corresponding percent was 34. Furthermore, the vast majority of the failures caused by vibration (high-cycle fatigue) in PWR's was associated with a single system, namely the Chemical and Volume Control System (CVCS).

Further statistics from Reference 5 indicate that:

- For PWR's 27% of the failures were associated with the primary plant charging system, 16% with the feedwater system, and 12% with the decay heat removal system

- For BWR's 22% of the failures occurred in the feedwater systems and 19% in the recirculation system.
- The majority of failures occurred in small diameter piping. For both PWR's and BWR's, approximately 70% of all failures were in pipes less than 6 inches in diameter. The percent of all failures that took place in pipes one inch or smaller in diameter were 51 and 29 for PWR's and BWR's respectively.

The sample statistics cited from Reference 5 are included here to stress the significance of flow- and equipment-induced vibrations as a piping failure cause (load case) in comparison to other failure causes (load cases).

Information about the significance of piping vibration as a failure mode for the period between 1976 and 1980, i.e., after the closing date for the quoted study, has been obtained from the NPE records (Reference 6), updated through April of 1980. (Note that, due to delays in reporting, etc., the April update contains only very few items more recent than January, 1980.)

An abbreviated summary of each individual issue (typically each individual failure or abnormal event), selected for this study from the NPE records, is tabulated in Appendices B (PWR's) and C (BWR's). These summaries include, in brief key word format, the following information: plant, unit, time of event, system and line affected, how event was identified, the assumed or determined cause for the event, and the action taken

after the event. In addition, the NPE identification code is included to provide traceability for additional, detailed information on individual issues.

For purposes of comparison with the data in Reference 5 and for an overview of the distribution in time of vibration-originated piping failures, or observations, associated specifically with two PWR piping systems, a summary of the information obtained from the NPE records is presented below:

Table 3.2
Vibration Failures; from NPE records (Reference 6)

System	(Prior to) '70)	'71	'72	'73	'74	'75	'76	'77	'78	'79	Total
RHR	--	--	--	2	4	3	1	2	--	1	13
CVCS	3	2	1	2	9	8	13	11	17	14	80

The high failure rate in the CVCS piping for the period covered by Reference 5, especially from 1973 onwards, is maintained or has actually increased in more recent years. The interpretation of the numbers in the above table should be made in view of the increasing number of plants that have gone into commercial operation in the middle and late seventies, and in view of the improved completeness in event reporting from licensees in recent years.

It is interesting to note that this category of pipe loading, i.e., vibration induced by active equipment or by hydraulic flow conditions, which constitutes the dominating overall cause for piping failures, falls into

the category of loads that are typically not taken into account explicitly in the analytical design phase. The background to this can be found partly in the treatment of this load category in the applicable piping code rules, which is typically limited to the statement that piping shall be designed and laid out in such a fashion that vibratory effects are minimized. The USNRC Regulatory Guide 1.68 specifies that it be verified as part of the initial test program that piping and component vibration are acceptable.

The techniques commonly cited as corrective measures to preclude, or minimize, vibration-induced failures can be summarized as follows (Reference 5):

A. For mechanical vibration:

- increased pipe support rigidity;
- isolation of vibrating equipment from piping by flexible connections; and
- separation of the natural frequency of piping from the forcing frequencies.

B. For flow-induced vibration:

- reduced fluid velocities;
- incorporation of smooth flow transitions;

- mechanical designs that account for high fluid velocities; and
- avoidance of flow vortices.

Table B.2 in Appendix B includes two notable examples of flow-induced vibrations in PWR mainstream piping. At Turkey Point it was noted that the safety relief valve header lengths corresponded to approximately one-quarter of the dominating pressure pulse wavelength in the pipe and that the opposing header length was one-half of this wavelength, based on flow conditions at the power level at which the vibrations became excessive. It was further pointed out that the header acted essentially like organ pipes. Stress calculations based on vibration test results showed that the allowable pipe stress of 12,000 psi would be reached at a power level of 95% due to the vibrations. The design modifications made to reduce the vibrations to acceptable values consisted of the following:

- a. Header lengths were changed so that they were no longer one-quarter and one-half of the critical wavelength.
- b. Edges were rounded off at connections of safety relief lines to the main header.
- c. Flow spoilers, in the form of blind nipples penetrating about 6 inches into the main header, were installed upstream of each safety relief line connection.

At Fort Calhoun it was determined that the vibrations were caused by flow turbulence exciting acoustic (pulsation) resonances of the piping systems in the 3-10 Hz range, which in turn coupled to excite the mechanical natural frequencies of the various piping spans in that frequency range. The mechanical natural frequencies of the piping systems were 3, 5, 7, and 10 Hz.

In order to avoid the resonance problem, pipe supports were added to increase all lateral modal frequencies of the piping to a level above 15 Hz.

The detailed tests and calculations that were performed in connection with the two cases cited above have typically been carried out only following extreme vibrations in major systems where the consequences of a failure potentially would be extremely severe from a safety hazard, as well as a cost, point of view. The knowledge gained from these detailed tests could, however, be applied also to resolve the frequent vibration problems in smaller lines, the response of which in many respects follows the same physical ground rules.

3.2.5 Blowdown Loads

Blowdown loads discussed in this section will be divided into the following two categories:

- a. Loads in systems with steam blowdown to atmospheric pressure. Valve discharge piping is short with little or no flow restriction.

- b. Loads due to steam blowdown into closed or flow restricted systems.

Examples of blowdown systems from category (a) include safety, relief, and dump valves in PWR mainsteam systems. The primary forcing function in these blowdown cases is the jet force at the exhaust outlet from the valve. The axial reaction forces on the exhaust pipe are most effectively taken care of by pipe restraints placed as close as possible to the jet exhaust, in the direction opposite to the jet. This arrangement essentially isolates the remainder of the piping, upstream of the valve, from the load effects. The load magnitude is relatively well defined by the temperature/pressure condition of the steam and the surrounding atmosphere, the flow area at the exhaust outlet, and the valve opening characteristics. The most efficient way of confirming such analytical blowdown forces by test results would be to monitor the reaction forces at pipe supports upstream from the blowdown jet.

Design weaknesses in header/support systems associated with this category of blowdown loads were identified in a few isolated instances in the early seventies (Reference 6):

- Ginna -- March, 1972. Mainsteam safety relief valves. A stress analysis showed that the header had insufficient capacity within acceptable stress limits to carry the reaction loads during valve discharge.

Temporary supports, conservatively sized to carry downward reactions from blowdown valves, were installed.

- Turkey Point 3 -- December, 1971. Mainsteam safety valve header failure, causing equipment damage and personnel injury.
- Connecticut Yankee -- January, 1972. Following the incident at Turkey Point 3 (see above), a stress analysis was performed for the corresponding installation at Connecticut Yankee. The analysis showed that an undesirable stress condition would exist if the secondary safety valves actuated. Additional supports were installed to absorb the reaction from the thrust forces. The supports consisted of rods attached to trunnions, welded to the discharge elbow of each safety valve, and were designed for the full thrust force.

Blowdown loads of category (b), above, are of a considerably more complex nature than the direct thrust forces discussed thus far. The discharge of steam via a safety relief valve, with very short opening time, into a long discharge piping system, produces a complex cyclic force field at each pipe change of direction or flow restriction. This loading pattern is typically that of a steam hammer event, with pressure pulses travelling at the speed of sound along with the pipe, and can be further complicated by the presence of water seals that are sometimes added to protect valve seats.

Examples of piping systems with this type of blowdown discharge are:

PWR: Pressurizer safety relief lines from the pressurizer to the pressurizer relief tanks.

BWR: Mainsteam safety relief valve discharge lines. These lines go from the valves through the drywell to quenchers in the suppression pool.

Numerous instances of pressurizer relief valve actuation have occurred, with blowdown to the relief tanks. The absence of damage or abnormal piping behavior due to these actuations is taken as an indication of the adequacy, or conservatism, of the discharge line design.

The safety relief valve discharge in BWR's results not only in steam hammer of the type mentioned above for pressurizer relief lines, but also in considerable bubble drag loads on the submerged steam quenchers. In addition, a dynamic, structural response feedback is created from the steam outlet through the quenchers, via pressure waves in the suppression pool, to the torus wall (GE, Mark I) or to the drywell wall (GE, Mark II, III). A dynamic excitation is thereby fed back into the discharge lines at the attachment points to the torus or the drywell wall, respectively.

These suppression pool or torus-related loads on the S/RV discharge lines form part of an extensive BWR loading study being performed by General Electric in cooperation with BWR owners groups. This study includes extensive

large-scale testing, as well as analytical evaluations, based on which final design loads are defined (References 8 and 9).

3.2.6 Intergranular Stress Corrosion Cracking

The distinction between different load categories implied by the section headings in this report is not always well defined. A certain amount of overlap exists between categories such as water hammer, flow-induced vibration, equipment-induced vibration, blowdown loads, and stress-assisted corrosion cracking. For example, the vibrations caused by positive displacement charging pumps are equipment-induced in one sense, but they are often to a large extent transmitted through hydraulic pressure pulsations and are therefore flow-induced, and can for the same reason be compared to continuous, mild, harmonic water hammer effects. Though not strictly a load category, but rather a complex failure mode, the Intergranular Stress Corrosion Cracking (ISCC) phenomenon is included here since a number of the factors contributing to it can be attributed to structural loading effects. The ISCC phenomenon has been a major cause of the development of extensive cracking in essential, safety-related piping systems in BWR's, and, to a considerably smaller extent, in PWR's. The BWR piping systems primarily affected by this problem have been the primary recirculation system (in particular valve bypass lines), the core spray system, the reactor water cleanup system, and the coolant injection system (HPCI, LPCI).

A special study, that was performed as a result of the frequent cracking problems in BWR piping systems (Reference 6), identified several factors that were considered important for ISCC development:

A. Local stresses

- Weld shrinkage and fit-up stresses;
- Residual stresses resulting from weld preparation, machining, and grinding; and
- Operating stresses due to pressure, vibration, and/or thermal gradients.

B. Chemical agents

- High oxygen content in BWR primary water during hot standby from radiolytic decomposition of water. During standby operation, the oxygen concentration was expected to increase from the introduction of air-saturated makeup water and x-ray radiolysis (especially if mainsteam isolation valves are closed).
- Pipe ID surface contamination (chloride, fluoride, copper, etc.).

C. Sensitization in the heat-affected zones of welds due to excess time-temperature welding, or due to furnace heat treatment of susceptible material.

It is not clear at this point to what extent the individual factors contribute to the crack initiation and development. Recommended action for the particular

problems associated with BWR piping have included
(Reference 6, items BWR-I.B.19, 20):

- Replacement of susceptible piping with material that is less adversely affected by highly oxygenated water. Practical replacement materials have been ferritic steels, Inconel 600, stabilized SS Types 347 and 321, cast SS Types 304 and 316, and SS Type 304L or 316L.
- Increased inspection frequency of welds and adjoining areas of Types 304 and 316 austenitic SS piping within the reactor coolant pressure boundary that operate with no flow or low flow.
- Permanent removal of recirculation bypass lines.

It is also noted that it would be beneficial to reduce stresses and strains by improving the fit-up methods, the weld joint configuration, the control of welding parameters, and by modifying the piping layout in order to reduce the effects from thermal expansion, vibration, and other loads. However, no practical means have been readily available that would positively reduce piping stresses to acceptably low levels, and implementation has been considered too difficult.

4. LOADS TYPICALLY CONSIDERED IN CURRENT INDUSTRY PRACTICE

This section provides definitions of piping design loads that typically are explicitly taken into account in current design practice for a representative selection of LWR piping systems. For each selected system, the design loads, obtained directly from existing completed analysis reports, were summarized in a series of tables and figures which are presented in Appendix D.

A considerable amount of data, representative for the types of load definitions typically considered in current LWR piping analysis, has been provided in Appendix D for purposes as outlined below:

- Provide a comprehensive basis for comparison between load definitions as used in typical current design approaches and indications of significant loads obtained from operating experience. (See Section 5.)
- Provide load data (e.g., with regard to thermal transients) in sufficient detail to show the types of simplifications typically used in current analyses for comparisons with the often very complex load phenomena experienced in actual operation. (See e.g., cases involving transient behavior of strong thermal stratifications, Section 3.2.2.)

The information included in Appendix D has deliberately been provided in considerable detail for the selected systems, in order to give the reader a comprehensive picture of load idealizations in current LWR piping analysis, to an extent which goes somewhat beyond the immediate objectives of the current study.

For PWR's, the system/lines selected were: pressurizer surge line, pressurizer relief lines, pressurizer spray lines, residual heat removal and safety injection lines (ECCS), boron injection, letdown and charging system (CVCS), feedwater, and mainsteam systems. The BWR systems selected were the mainsteam safety/relief lines and the control rod drive hydraulic system (CRDHS), including insert and withdrawal hydraulic lines, scram discharge headers and volumes, and service water lines.

The information contained in Appendix D for each individual system is typically as follows:

- a. Summary table defining all load cases considered explicitly in the analysis. The load cases are given in a matrix format, defining them with respect to fundamental load category and plant condition. References are given to tables and graphs providing numeric load quantities.
- b. Simplified, schematic figure defining the boundaries of the piping considered.
- c. Schematic figure showing boundaries for temperatures and pressures for different operating (design) modes. This figure may be combined with item (b) in some cases.
- d. Table defining thermal transient events considered.
- e. Quantitative definitions of representative thermal transients in the form of Temperature-Time graphs.

- f. Further quantitative load descriptions as applicable to individual systems; for example, force time-history curves defining blowdown or water hammer loads.

The samples selected, being extracted from actual design cases, are by necessity plant-specific with regard to certain items. The operational pressure and temperature values used are, for example, specific for the particular NSSS design, and the force time-histories for blowdown and water hammer loads are specific to particular piping layout and valve characteristics. In addition, for the PWR plants used in the study, no emergency plant conditions had been defined, which explains the absence of this condition in the summary tables. If allowance is made for the relatively minor adjustments to actual load descriptions that are necessary for different plants of essentially the same type, the load definitions given can be considered typical and of generic validity.

5. COMPARISON OF ACTUAL VERSUS CURRENT DESIGN LOADINGS

The presentation in Section 4 and Appendix D shows that the piping loads considered in the analytical phase of the design process, for essentially all piping systems, typically include the following:

- Gravity Loads
- Thermal expansion and thermal anchor movement
- Thermal transients (Typically for Class 1 piping only)
- Seismic inertia and seismic anchor movements (OBE and SSE)
- Imposed displacements due to postulated accidents (DBE)

Examples of additional loads, explicitly included in the analyses of particular systems, are:

- Steam hammer and water hammer effects due to certain valve closure events in mainsteam and feedwater systems;
- Shock loading and thrust forces associated with safety/relief valve blowdown; and
- Poolswell effects and other suppression pool-related loadings on BWR systems such as safety/relief valve discharge lines and control rod drive hydraulic systems.

A comparison of the piping loads typically included in the design process as listed above with the findings from the review of plant operating experience presented in Section 3 has resulted in the following observations:

- a. Water hammer and steam hammer loads are inadequately considered in current designs.

The summary of reported water/steam hammer events, provided in Table 3.1, shows that the significance of this type of loading extends far beyond the phenomena initiated by the opening and closing of valves in mainsteam, feedwater, and pressurizer relief systems, which are the cases typically covered in current design. Table 3.1 includes primarily events that were associated with damage. (In fact, many were detected only from the permanent damage caused.) It can therefore be assumed that a much larger number of water hammer type events have occurred which were never reported, either because they were considered insignificant or simply because they were never identified.

The RHR, HPCI, and Core Spray systems in BWR's are particularly susceptible to water hammer occurrence, which typically is not taken into account in the design. Additional information, about individual water hammer events that have been reported for these three systems, is presented in Tables C.2, C.3 and C.4 in Appendix C.

In addition to those listed above, Table 3.1 includes 15 piping systems, the designs of which typically do not take water hammer loading into account, as having been subjected to these loads. Many of the instances have resulted in damage.

- b. Flow- and equipment-induced vibrations have typically not been considered in the design of piping systems, other than at the pre-operational and power escalation test stages which serve as design verifications.

During the test, vibrations in the piping system are judged to be excessive or acceptable based primarily on visual

observations. Quantitative, general criteria for what constitutes acceptable or excessive piping vibrations have not been identified. The absence of such criteria is likely to be part of the reason for the large number of failures reported to have been caused by vibrations during normal operation.

High-cycle fatigue, caused primarily by flow- and equipment-induced vibrations, has been reported (Reference 5) to be the most common failure mode in LWR piping systems. An illustration of this situation is given in Appendices B and C, which contain a total of 109 individual accounts of pipe failures caused by vibrations.

- c. Thermal transients, which are subject to some consideration in Class 1 piping and generally not considered in Class 2 and 3 piping analysis, are not adequately addressed.

Feedwater piping in PWR's, not being part of the primary coolant pressure boundary, is classified as Class 2 piping and is consequently not subject to detailed evaluation of thermal transient effects.

Recent, detailed temperature measurements in operating PWR feedwater piping in the regions of auxiliary feedwater connections and steam generator nozzles have identified unexpectedly large amounts of thermal stratification and thermal cycling. These factors are believed to have contributed to the extensive cracking that has been found in recent years in those particular areas of the feedwater piping.

In particular, it was found that the temperature fluctuations, involving full thermal cycles with high amplitudes, were

considerable in the nozzle regions of the piping under low flow conditions, e.g., during hot standby, auxiliary feed, and power escalation.

The significance of the stress portion in stress-assisted corrosion cracking phenomena has not been investigated in sufficient detail for conclusions to be drawn with respect to the relative importance of thermal transients, mechanical vibrations, and high steady-state stresses as contributors to this particular failure mode. The locations, primarily in BWR piping systems, that have been particularly susceptible to stress corrosion cracking, e.g., nozzle regions of piping connected to the RPV, are those where considerable thermal cycling can be expected.

This is suggestive of a major contribution from thermal transients in the complex causes for the crack development. Further detailed temperature monitoring in the affected areas during extended periods of plant operation is, however, required as a basis for an analytical evaluation before any conclusive statements can be made on this hypothesis.

In general, the number of thermal transient cycles tends to be higher than has previously been expected due to the increasing plant sensitivity and frequency of reactor trips, increasing numbers of heat-up/cool-down cycles, and phenomena such as those involving significant temperature stratification and cycling under low flow conditions as mentioned above in connection with PWR feedwater systems.

- d. A concern frequently expressed by the industry is that the use of design temperatures, for the evaluation of pipe stresses and

support loads resulting from thermal expansion effects, is excessively conservative, and may lead to an unnecessarily large number of snubbers in the piping systems. The "conservatism" in the use of design temperatures may thus result in unconservative system designs due to the potential loss in reliability inherent in the increased component complexity.

Due to the lack of long-term system monitoring data, it has not been possible in this study to verify or refute the alleged conservatism in the thermal expansion considerations. The lack of this data is actually also the reason for the use of design temperatures in the expansion evaluations, since substantial justification is required for the use of any lower temperatures.

6. CONCLUDING REMARKS AND RECOMMENDATIONS

From the findings in the present study, as presented in the previous sections of this report, conclusions and recommendations are summarized as follows:

- a. Water hammer and steam hammer loading phenomena are not analyzed to a sufficient level of sophistication in the initial design phase. In particular the mainsteam, feedwater, residual heat removal, core spray, and high pressure coolant injection systems are affected by recurring water/steam hammer events that have yet to be resolved.

Whether the design effort is aimed towards the elimination of load occurrence or towards pipe and restraint designs capable of withstanding the loads, it is equally important to enhance the understanding of the initiating conditions, the force magnitudes involved, etc., by detailed monitoring of system parameters during operation.

- b. The frequency of pipe failures caused by various categories of operational vibrations, e.g., flow- and equipment-induced, justifies a change in design approach from the current reliance on visual detection of excessive vibrations during testing and operation, towards one of prediction and prevention based on rational criteria.

The piping vibration problem is currently being addressed in the ANSI/ASME Operations and Maintenance Committee in their preparation of a standard entitled "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Plant Piping Systems".

The approach in this Standard, as the title suggests, is to use testing of completed systems as the primary design tool to incorporate vibration issues in the qualification of piping.

In order to minimize costly system modifications following the preoperational tests, however, it is recommended that the vibration loads be incorporated in the initial phase. To make this feasible it is recommended that a comprehensive study be performed, covering the following main issues:

- develop simple rules for the identification of piping susceptible to vibration problems.
 - identify general design rules for minimization of vibration effects.
 - define structural acceptance criteria for general vibratory loads.
 - develop simplified design methods suitable for practical use in showing compliance with the acceptance criteria.
- c. In view of the failure and damage records for LWR piping and their underlying causes, such as high-cycle fatigue, thermo-hydraulic shock loading, stress corrosion cracking, thermal cycling, etc., it appears that the considerable amount of design effort oriented towards low-probability postulated accident loads (earthquakes, LOCA's, etc.) is partially made at the expense of the consideration of higher-probability operational loads. This statement is not meant to imply that the design efforts associated with low-probability accident loads should be reduced, but rather that appropriate

consideration be given to higher-probability operational loads that are known to result in damage to systems.

The substantial amount of data defining the dynamic response of a piping system that is generated, for example in a seismic analysis, could be utilized also in a comprehensive evaluation of its response to general, operational dynamic excitations for systems that traditionally have had vibration problems. The mode shapes and modal frequencies that are determined for the performance of seismic response spectral analyses contain a large amount of valuable information for the prediction of system response to general dynamic excitation, such as forced vibration, shock loading, etc.

- d. The number and magnitude of thermal transient cycles have shown a tendency to increase over values anticipated in the initial design. Detailed on-line monitoring of PWR feedwater lines has, for example, indicated considerable amounts of spontaneous thermal cycling in nozzle regions during low flow conditions. This, in view of the frequent development of cracks in the pipe nozzle regions of the RPV (BWR's) and the steam generators (PWR's), indicates a need for a more detailed study of the thermal cycling phenomena in terms of amplitude and number of cycles.

In addition, the potential use of nuclear plants to satisfy peaks in power demand and changes in operating procedures, resulting in more frequent power fluctuations, raises questions regarding the adequacy of the thermal transient definitions in current design criteria, which need to be further investigated.

- e. A general feedback mechanism to the design process is needed to ensure that process parameters (temperature, pressure, flow rate) and piping response parameters (displacements, reaction forces) and vibration parameters (such as accelerations, frequency content, and amplitudes) from a full range of testing and operational conditions, be monitored and utilized for verification and/or adjustments to the analytical load idealizations used in the design process. This feedback is the only reliable vehicle for development of more realistic load-predicting capabilities.

The importance of realistic load predictions is obvious for the original piping design in future plants, but also, and perhaps even more so, for the upgrading of existing plants, since in those cases it is essential to limit the modification efforts to items that are truly relevant.

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APPENDIX A

Set of matrices developed for the identification of potential significant loadings on representative Light Water Reactor piping systems.

List of Tables

A.1.1	Significant PWR Plant Events
A.1.2	PWR: Significant Loading on Reactor Coolant Loop Piping
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A.2.7	BWR: Significant Loading on Control Rod Drive Hydraulic System Piping
A.2.8	BWR: Significant Loading on Emergency Core Cooling System Piping

Table A.1.1

Significant PWR Plant Events

1. Heat Up / Cool Down @ 100°F / hr.
2. Unit Loading / Unloading @ 5% / min.
3. Step Loading $\pm 10\%$
4. Large Step Load Decrease with Steam Dump
5. Fluctuations from Steady State
6. Feedwater Cycling at Hot Shutdown
7. Loop Out of Service
8. Unit Loading / Unloading between 0 + 15% Full Power
9. Boron Concentration Equalization
10. Refueling
11. Loss of Load (Without Immediate Reactor Trip)
12. Loss of Power
13. Partial Loss of Flow
14. Reactor Trip from Full Power
15. Inadvertent Reactor Coolant System Depressurization
16. Inadvertent Auxiliary Spray
17. Inadvertent Startup of an Inactive Loop
18. Control Rod Drop
19. Inadvertent Emergency Core Cooling System Actuation
20. OBE / $\frac{1}{2}$ Safe Shutdown Earthquake
21. Small LOCA
22. Small Steam Break
23. Complete Loss of Flow
24. Reactor Coolant Pipe Break (Large LOCA)
25. Large Steam Line Break
26. Feedwater Line Break
27. Reactor Coolant Pump Locked Rotor
28. Control Rod Ejection
29. Steam Generator Tube Rupture
30. Safe Shutdown Earthquake / Design Basis Earthquake

Table A.1.1 (Cont'd.)

31. Turbine Roll Test
32. Primary Side Leakage Test
33. Secondary Side Leakage Test
34. Tube Leakage Test
35. Primary Side Hydrostatic Test
36. Secondary Side Hydrostatic Test

Table A.1.2

PWR: Significant Loading on Reactor Coolant Loop Piping

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ②	1, 2, 3, 4, 5, 6, 7, 8, 9, & 10	11, 12, 13, 14, 15, 16, 17, 18 & 19	22, 22, & 23	24, 25, 26, 27 & 28
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: RCP, ST GEN, & RPV	TAM: RCP, ST GEN, & RPV SAM (OBE): ST GEN, RPV, RCP, & RB	TAM: RCP, ST GEN, & RPV	TAM: RCP, ST GEN, & PRV SAM (OBE): ST GEN, RPV, RCP, & RB
DBA Dynamic Loads	-	-	-	RCL Movements
Blowdown	-	-	21	-
Water Hammer	-	-	-	-
Other	Flow & RCP In- duced vibrations, flow induced drag, & surge nozzle shock.	Flow & RCP In- duced vibrations, flow induced drag, & surge nozzle shock.	Flow & RCP In- duced vibrations, flow induced drag, & surge nozzle shock.	Flow & RCP In- duced vibrations, flow induced drag, & surge nozzle shock.

Table A.1.3

PWR: Significant Loading on Pressurizer Surge Line

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ^②	1, 2, 3, 4 & 5	11, 12, 14 & 15	21 & 22 ^④	24 & 25
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: PRESS., & RCL	TAM: PRESS., & RCL SAM (OBE): RCL, RB, & PRESS.	TAM: PRESS., & RCL	TAM: PRESS., & RCL SAM (SSE): RCL, RB, PRESS. DBA (LOCA): RCL
DBA Dynamic Loads	-	-	-	RCL Movements
Blowdown	-	-	-	-
Water Hammer	-	-	-	-
Other	Flow induced vibrations.	Flow induced vibrations.	Flow induced vibrations.	Flow induced vibrations.

Table A.1.4

PWR: Significant Loading on Pressurizer Relief Lines

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ^②	1, 3, 4 & 5	11, 12, 13, 14 & 15	21 & 22	24 & 25
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: PRESS., & PRT	TAM: PRESS., & PRT SAM (OBE): RB, PRESS., & PRT	TAM: PRESS., & PRT	TAM: PRESS., & PRT SAM (SSE): RB, PRESS., & PRT DBA (LOCA): PRESS.
DBA Dynamic Loads	---	---	---	RCL & PRESS. Movements
Blowdown ⁷	PSV Actuation	PSV Actuation	PSV Actuation	PSV Actuation
Water & Steam Hammer	---	---	---	---
Other	---	---	---	---

Table A.1.5

PWR: Significant Loading on Pressurizer Spray Piping

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ② ⑦	1, 2, 3, 4, 5, 6, 7, 8, 9 & 10	11,12, 13, 14, 15, 16, 17, 18 & 19	21, 22 & 23	24, 25, 26, 27 & 28
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: PRESS., & RCL	TAM: PRESS., & RCL SAM (OBE): RCL, RB & PRESS.	TAM: PRESS., & RCL	TAM: PRESS., & RCL SAM (SSE): RCL, RB & PRESS. DBA (LOCA): RCL & PRESS.
DBA Dynamic Loads	-	-	-	RCL & PRESS. Movements
Blowdown	-	-	-	-
Water Hammer	-	-	-	-
Other	-	-	-	-

Table A.1.6

PWR: Significant Loading on Residual Heat Removal Piping

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ^②	1	③	③	③
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: RCL, RHR PPS, & Other Equipment	TAM: RCL, RHR PPS, & Other Equipment SAM (OBE): RCL, RB, BLD & RHR PPS	TAM: RCL, RHR PPS, & Other Equipment	TAM: RCL, RHR PPS, & Other Equipment SAM (OBE): RCL, RB, BLD & RHR PPS DBA (LOCA): RCL & RCO
DBA Dynamic Loads	-	-	-	RCL Movements
Blowdown	-	-	-	-
Water Hammer	-	-	-	-
Other	-	-	-	-

Table A.1.7

PWR: Significant Loading on Emergency Core Cooling Piping

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ^②	1 & 10	11, 14, 15 & 19	21 & 22	24 & 25
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: RCL, RHX & BIT	TAM: RCL, RHX & BIT SAM (OBE): RCL, RHX, BIT, RB, BLD & ACT	TAM: RCL, RHX & BIT	TAM: RCL, RHX & BIT SAM (OBE): RCL, RHX, BIT, RB, BLD & ACT
DBA Dynamic Loads	-	-	-	RCL Movements
Blowdown	-	-	-	-
Water Hammer	-	-	-	-
Other	Flow & PPS In- duced vibrations.	Flow & PPS In- duced vibrations.	Flow & PPS In- duced vibrations	Flow & PPS In- duced vibrations.

Table A.1.8

PWR: Significant Loading on Chemical and Volume Control System Piping

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ^②	1, 2, 3, 4 & 5	11, 12, 13, 14, 15 & 16 ^③	21 & 22	24 & 25
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: RCL, RCP, CHR, PPS, HX, ^⑧ VCT & DEM	TAM: RCL, RCP, CHR PPS, HX, ^⑧ VCR & DEM SAM (OBE): RB, RCL, RCP, CHR PPS, HX, ^⑧ VCT, DEM & AB	TAM: RCL, RCP, CHR PPS, HX, ^⑧ VCT & DEM	TAM: RCL, RCP, CHR PPS, HX, ^⑧ VCT & DEM SAM (SSE): RB, RCL, RCP, CHR PPS, HX, ^⑧ VCT, DEM & AB DBA (LOCA): RCL & RCO
DBA Dynamic Loads	-	-	-	RCL Movements
Blowdown	-	-	-	-
Water Hammer	-	-	-	-
Other	CHR PPS Induced vibrations.	CHR PPS Induced vibrations.	CHR PPS Induced vibrations.	CHR PPS Induced vibrations.

Table A.1.9

PWR: Significant Loading on Mainsteam Piping

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ② ⑤	(Test: Water Wt.) -	-	-	-
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support/ Anchor Movements	TAM: ST. GEN. & Turbine	TAM: ST. GEN. & Turbine SAM (OBE): ST. GEN. & RB	TAM: ST. GEN. & Turbine	TAM: ST. GEN. & Turbine SAM (SSE): RB & ST. GEN. DBA (LOCA): RCO & ST. GEN.
DBA Dynamic Loads	-	-	-	ST. GEN. Move- ments
Blowdown	-	SRV Actuation	SRV Actuation	SRV Actuation
Steam Hammer	-	TSV, IV & BV Actuation	TSV, IV & BV Actuation	TSV, IV & BV Actuation
Other	Flow induced vibration.	Flow induced vibration.	Flow induced vibration.	Flow induced vibration.

Table A.1.10

PWR: Significant Loading on Feedwater Piping

LOAD GROUP	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY ^①	FAULTED
Pressure	Design/Operating	Peak	Peak	Peak
Thermal Expansion	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating	Design/Maximum Operating
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients ② ⑤	-	-	-	-
Seismic Inertia	N/A	20 (OBE)	N/A	30 (SSE)
Support Anchor Movements	TAM: ST. GEN., FW PPS, FW HTR & Other Equipment	TAM: ST. GEN., FW PPS, FW HTR & Other Equipment SAM (OBE): ST. GEN & RB	TAM: ST. GEN., FW PPS, FW HTR & Other Equipment	TAM: ST. GEN., FW PPS, FW HTR & Other Equipment SAM (SSE): RB & ST. GEN. DBA (LOCA) ST. GEN.
DBA Dynamic	-	-	-	ST. GEN. Move- ments
Blowdown	-	-	-	-
Water Hammer	-	Feedwater Iso- lation valve act- uation.	Feedwater Iso- lation valve act- uation.	Feedwater Iso- lation valve act- uation.
Other	Flow induced vibrations.	Flow induced vibrations.	Flow induced vibrations.	Flow induced vibrations.

Notes: (Associated with Tables A.1.2 through A.1.10)

1. The Emergency Plant Condition is not used in some Safety Analysis Reports or Design Criteria.
2. Transients are due to pressure and temperature variations during the particular plant events listed.
3. All upset, emergency, and faulted plant events are followed by RHR System actuation with associated pressure and temperature transients.
4. These plant events per some design criteria have been evaluated under the upset plant condition.
5. Transients are not explicitly considered for fatigue on Class II / III pipe. Peak values are considered under design load conditions.
6. Blowdown loading is significant during all Pressurizer Relief Valve and Safety Relief Valve operations.
7. Temperature shock is significant during (Auxiliary Spray Operation) the following plant events: 1, 14, 16, 24, 25.
8. Regenerative, Letdown, Excess Letdown, and Seal Water Heat Exchangers.
9. Abbreviations:

N/A - Not Applicable

OBE - Operating Basis Earthquake

SSE - Safe Shutdown Earthquake

TAM - Thermal Anchor Movement

SAM - Seismic Anchor Movement

DBA - Design Basis Accident

LOCA - Loss of Coolant Accident

RCL - Reactor Coolant Loop

RB - Reactor Building

PRESS. - Pressurizer

ST. GEN. - Steam Generator

RHR PPS - Residual Heat Removal Pumps

PRV - Pressurizer Relief Valve

PSV - Pressurizer Safety Valve

CHR PPS - Charging Pumps

HX - Heat Exchanger

RCO - Reactor Containment Overpressurization

FW PPS - Feedwater Pumps

9. Abbreviations (cont'd.)

BIT - Boron Injection Tank
ACT - Accumulator Tanks
RHX - Residual Heat Exchanger
DEM - Demineralizers
VCT - Volume Control Tank
PRT - Pressurizer Relief Tank
AB - Auxiliary Building
RPV - Reactor Pressure Vessel
BLD - Building
TSV - Turbine Stop Valve
IV - Isolation Valve
BV - Bypass Valve

Table A.2.1

Significant BWR Plant Events

<u>Event</u>	
1.	Bolt Up
2.	Design Hydrostatic Test
3.	Startup (100°F/hr Heatup Rate)
4.	Daily Reduction to 75% Power
5.	Weekly Reduction to 50% Power
6.	Control Rod Pattern Change
7.	Loss of Feedwater Heaters
8.	50% Safe Shutdown Earthquake Event as Rated Operating Conditions
9.	Scram:
	A. Turbing Generator Trip, Feedwater On, Isolation Valves Stay Open
	B. Other Scrams
	C. Loss of Feedwater Pumps, Isolation Valves Closed
	D. Turbine Bypass, Single Safety or Relief Valve Blowdown
10.	Reduction to 0% Power, Hot Standby, Shutdown (100°F/hr Cooldown Rate)
11.	Unbolt
12.	Scram:
	A. Reactor Overpressure with Delay Scram, Feedwater Stays On, Isolation Valves Stay Open
	B. Automatic Blowdown

13. Improper Start of Cold Recirculation Loop
14. Sudden Start of Pump in Cold Recirculation Loop
15. Improper Startup with Reactor Drain Shut Off
Followed by Turbine Roll and Increase to Rated Power
16. Pipe Rupture and Blowdown (LOCA)
17. Safe Shutdown Earthquake at Rated Operating Conditions

Table A.2.2

BWR: Significant Loading on Reactor Coolant Recirculation Piping

LOAD TYPE	PLANT EVENT CATEGORY			
	NORMAL	UPSET	EMERGENCY	FAULTED
Static Int. Pressure	Operating	Operating	Peak	Peak
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Thermal Expansion	RPV temp. Differential temp. Main Line/bypass	RPV temp. Differential temp. Main Line/bypass	RPV temp. Differential temp. Main Line/bypass	RPV temp. Differential temp. Main Line/bypass
Seismic (Inertia)	-----	OBE	-----	SSE
Seismic (Rel. Displ.)	-----	OBE	-----	SSE
Seismic (other)	-----	-----	-----	-----
Thermal Transient	Events: 2, 3, 4, 5, 6,	Events: 3, 9	Events: 12, 13, 14, 15	Events: 16, 17
Pressure Transient	7, 10			
Blowdown	-----	-----	-----	-----
Steam/Wtr. Hammer	-----	-----	-----	-----
Other Dyn Loads	Flow and pump induced vibrations	Flow and pump induced vibrations	Flow and pump induced vibrations	Flow and pump induced vibrations
Other	-----	-----	-----	-----

Table A. 2.3

BWR: Significant Loading on Mainsteam Piping

LOAD TYPE	PLANT EVENT CATEGORY			
	NORMAL	UPSET	EMERGENCY	FAULTED
Static Int. Pressure	Design/Oper.	Oper.	Peak	Peak
Gravity	Live + Dead (Test: Water Wt.)	Live + Dead	Live + Dead	Live + Dead
Thermal Expansion	Design/Oper. temp.	Design/Oper. temp.	Design/Oper. temp.	Design/Oper. temp.
Seismic (Inertia)	-----	OBE	-----	SSE
Seismic (Rel. Displ.)	-----	OBE	-----	SSE
Seismic (other)	-----	-----	-----	-----
Thermal Transient				
Pressure Transient	Events 2-7, 10	Events 8, 9	Events 12-15	Events 16, 17
Blowdown	-----	S/RV Actuation	S/RV Actuation	S/RV Actuation
Steam/ Wtr. Hammer	-----	Actuation of Turbine Con- trol valves, Isolation valves and Bypass valves	Actuation of Turbine Con- trol valves, Isolation valves and Bypass valves	Actuation of Turbine Con- trol valves, Isolation valves and Bypass valves
Other Dyn. Loads	Flow induced vibrations	Flow induced vibrations	Flow induced vibrations	Flow induced vibrations
Other	-----	-----	-----	-----

Table A.2.4

BWR: Significant Loading on Feedwater Lines

LOAD TYPE	PLANT EVENT CATEGORY			
	NORMAL	UPSET	EMERGENCY	FAULTED
Static Int. Pressure	Design/Oper.	Peak	Peak	Peak
Gravity	Dead + Live	Dead + Live	Dead + Live	Dead + Live
Thermal Expansion	Design/Oper. Temp.	Design/Oper. Temp.	Design/Oper. Temp.	Design/Oper. Temp.
Seismic (Inertia)	---	OBE	---	SSE
Seismic (Rel. Displ.)	---	OBE	---	SSE
Seismic (other)	---	---	---	---
Thermal Transient	Events: 2, 3, 4, 5, 6,	Events: 8, 9	Events: 12, 13, 14, 15	Events: 16, 17
Pressure Transient	7, 10			
Blowdown	---	---	---	---
Water Hammer	---	Isolation valve operation	Isolation valve operation	Isolation valve operation
Other Dyn. Loads (1), (2)	Flow and Pump induced vibrations	Flow and Pump induced vibrations	Flow and Pump induced vibrations	Flow and Pump induced vibrations
Other	---	---	---	---

Table A.2.5

BWR: Significant Loading on Safety/Relief Valve Discharge Piping

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	EMERGENCY	FAULTED
Static Int. Pressure	Design Open	Peak	Peak	Peak
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Thermal Expansion	Design/Oper.	Design/Oper. (Fluid-Envir.)	Design/Oper.	---
Seismic (Inertia)	---	OBE	---	SSE
Seismic (Rel. Displ.)	---	OBE	---	---
Seismic (other)	---	Slosh on submerged parts	Slosh on submerged parts	Slosh on submerged parts
Thermal Transient	---	---	---	---
Pressure Transient		See LOCA below	See LOCA below	(See LOCA below)
Blowdown	---	SRV Discharge	SRV Discharge	SRV Discharge
Steam/Wtr. Hammer	---	---	---	---
Other Dyn. Loads (1) (2)	---	SBA, IBA (LOCA) -Cond. Oscill. -Chugging	IBA, SBA (LOCA) -Cond. Oscill. -Chugging	DBA (LOCA) -Pool Swell -Cond. Oscill. -Chugging
Other	---	---	---	---

Table A.2.6

BWR: Significant Loading on Drywell/Wetwell Vent System Piping

LOAD TYPE	PLANT EVENT CATEGORY			
	NORMAL	UPSET	EMERGENCY	FAULTED
Static Int. Pressure	Containment environment	Containment environment	Containment environment	Containment environment
Gravity	Dead + Live Load	Dead + Live Load	Dead + Live Load	Dead + Live Load
Thermal Expansion	Containment environment	Containment environment	Containment environment	Containment environment
Seismic (Inertia)	-----	OBE	-----	SSE
Seismic (other)	-----	Suppression pool wave slosh on submerged piping	Suppression pool wave slosh on submerged piping	Suppression pool wave slosh on submerged piping
Thermal Transient	-----	-----	-----	-----
Pressure Transient	-----	-----	-----	-----
Blowdown	-----	-----	-----	-----
Steam/WTR Hammer	-----	-----	-----	-----
Other Dyn. Loads (1), (2)	-----	S/RV Disch. -Jet, Drag Loads SBA, IBA (LOCA) -Cond. oscill. -Chugging	S/RV Disch. -Jet, Drag Loads SBA, IBA (LOCA) -Cond. oscill. -Chugging	S/RV Disch. -Jet, Drag Loads DBA (LOCA) -Cond. oscill. -Chugging -Pool swell

Table A.2.7

BWR: Significant Loading on Control Rod Drive Hydraulic System Piping

LOAD TYPE	PLANT EVENT CATEGORY			
	NORMAL	UPSET	EMERGENCY	FAULTED
Static Int. Pressure	Design/Oper.	Peak	Peak	Peak
Gravity	Dead + Live	Dead + Live	Dead + Live	Dead + Live
Thermal Expansion	Design/Oper. Temp.	Design/Oper. Temp.	Design/Oper. Temp.	Design/Oper. Temp.
Seismic (Inertia)	-----	OBE	-----	SSE
Seismic (Rel. Displ.)	-----	OBE	-----	SSE
Seismic (other)	-----	-----	-----	-----
Thermal Transient	-----	-----	-----	-----
Pressure Transient	-----	-----	-----	-----
Blowdown	-----	-----	-----	-----
Water Hammer	-----	Fast Valve Actuation	Fast Valve Actuation	Fast Valve Actuation
Other Dyn. Loads (3), (4)	-----	S/RV Dis-charge Excitation of Supporting structures. Suppr. Pool froth impact.	S/RV Dis-charge Excitation of Supporting structures. Suppr. Pool froth impact.	DBA (LOCA) Pool swell impact on supporting structures
Other	-----	-----	-----	-----

Table A.2.8

BWR: Emergency Core Cooling System Piping

LOAD TYPE	PLANT EVENT CATEGORY			
	NORMAL	UPSET	EMERGENCY	FAULTED
Static Int. Pressure	Design/Oper.	Peak	Peak	Peak
Gravity	Dead + Live	Dead + Live	Dead + Live	Dead + Live
Thermal Expansion	Design/Oper. Load	Design/Oper. Load	Design/Oper. Load	Design/Oper. Load
Seismic (Inertia)	---	OBE	---	SSE
Seismic (Rel. Displ.)	---	OBE	---	(SSE)
Seismic (other)	---	---	---	---
Thermal Transient	Events: 2, 3, 4, 5, 6,	Events: 8, 9	Events: 12, 13, 14, 15	Events: 16, 17
Pressure Transient	7, 10			
Blowdown	---	---	---	---
Steam/Wtr. Hammer	---	Potential WH at HPCI actuation	Potential WH at HPCI actuation	Potential WH at HPCI actuation
Other Dyn. Loads	---	Potential flow and pump in- duced vibra- tions	Potential flow and pump in- duced vibra- tions	Potential flow and pump in- duced vibra- tions
Other	---	---	---	---

Notes: (Associated with tables A.2.1 through A.2.8)

1. A. Poolswell Loads:
 - a. Pool Impact & Drag
 - b. Froth Impingement
 - c. Pool Fall back
 - d. LOCA Jet
 - e. LOCA Bubble Drag
- B. Condensation Oscillation
 - a. Load on submerged piping (Water-Air-Steam transfer from drywell to wetwell through vents).
- C. Chugging
 - a. Load on submerged pipes (Oscillating water column in vent down comers due to intermittent condensation)
2. SBA = Small Break Accident

IBA = Intermediate Break Accident

DBA = Design Basis Accident
3. Depending on the exact location and supporting system of CRDHS piping in a particular plant, this piping may be subjected to the loads defined in Notes 1, 2 above.
4. Dynamic excitation of the drywell wall, containment and other internal structures as transmitted through the suppression pool motion during S/RV discharge, generates dynamic loading on CRDHS piping supported off these structures. Pool swell impact and froth impact and drag loads can be transmitted either directly to the piping or indirectly via supporting structures.

APPENDIX B

Summary of items excerpted from NPE (Reference 6), and of relevance to the present study (PWR issues).

List of Tables

- B.1 PWR Piping Vibration Issues; RCL Branch Piping
- B.2 PWR Piping Vibration Issues; Mainsteam Piping
- B.3 PWR Piping Vibration Issues; RHR and Auxiliary Cooling Piping
- B.4 PWR Piping Vibration Issues; Chemical and Volume Control Systems

Table B.1 Summary of NPE Excerpts (PWR-2, V.B.). RCL Piping Vibration

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
7	Millstone 2	9,10/'5	RCL/ Instrument. line	elbow socket weld leaked	-fatigue failure mechanism with relative low levels of alternate stress. RCP running frequency corresponded closely with assembly natural frequency.	-repaired welds -modified pipe to increase assembly natural frequency & reduced torque moment on elbow socket weld
8	Arkansas 1	10/74	RCL/Cold leg drain line	socket weld cracked & leaked	-fatigue fracture mechanism -preliminary investigation speculated that alternating mechanical loading was the cause	-changed pipe configuration to change natural frequency & reduce mass in plane of maximum acceleration
22 (V.F)	Calv. Cliffs 2	12/76	RCL/ Pressure transmitter piping	cracks in heat-affected zone of welds	-excessive vibration	-shimmed hanger to prevent movement

Table B.2 Summary of NPE Excerpts (PWR-2, VI.D.). PWR MS Piping Vibration

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
61	Turkey Pt. 3 & 4	9/73	MS/Downstream of safety relief valve header	piping vibrated	pipe resonance with header & particular pipe configuration (acoustic pulsations)	-shortened header -reduced flow obstruction (header nozzle)
64	Ft. Calhoun	11/73	MS/Downstream of containment isolation valves to turbine	piping vibrated	flow turbulence exciting acoustic (pulsation) resonances of piping in 3-10 Hz range	modify pipe supports to increase piping resonances above 15 Hz

Table B.3 Summary of NPE Excerpts (PWR-2, VIII.B.). PWR Auxiliary Coolant Piping Vibration Issues

(sheet 1 of 3)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
15	Indian Pt. 2	5/73	RHR/ RHR exchange discharge vent line	3/4" socket weld cracked & leaked	fatigue failure vibration thought unlikely until 10/73 incident	repaired weld & replaced vent assembly
15	Indian Pt. 2	10/73	RHR/ RHR exchange discharge vent line	3/4" socket weld cracked & leaked	fatigue failure probably caused by excessive pipe vibration	replaced vents with socket weld caps
34	Arkansas 1	6/74	Decay Heat System/ Loop A discharge pipe	crack in heat- affected zone of weld	fatigue caused by excessive vibration in system piping, resulting from cavitation of pump discharge valves	added flow orifice
52	Oconee 3	10/74	LPI/ LPI discharge header sample line	leak	failed due to vibration of LPI discharge header	- repaired sample line - coil added to absorb vibrational stress
70	Oconee 3	10/75	LPI/ LPI discharge header sample line	weld failure	cavitation occur- ring at high flow rates causing line to vibrate exces- sively	?

Table B.3 (sheet 2 of 3)

NFE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
96	Arkansas 1	9/76	Decay Heat System/ Vent line	cracked weld with leak	excessive vibration	weld repaired
102	Cook 1	11/74	RHR/ 2" inlet line to valve	leaking weld	weld fatigue due to excessive pipe vibrations	weld repaired
102	Cook 1	12/74	RHR/ Instrument line	leaking socket weld	weld fatigue due to excessive pipe vibrations	weld repaired
102	Cook 1	3/75	RHR/ Equalizing line	cracked welds	weld fatigue due to excessive pipe vibrations	weld repaired
102	Cook 1	5/75	RHR/ Equalizing line	cracked weld	weld fatigue due to excessive pipe vibrations	program of redesign & modification of these lines in progress
128	Indian Pt. 2	4/77	RHR/ Seal water piping	leaking in heat-affected areas assoc. with socket weld connections	excessive vibration	rigid pipe replaced with SS tubing & Swagelok fittings
144	Indian Pt. 3	?/76	RHR/ 3/4" vents	general system modification	fatigue cracking	modified vents by replacing flanges & nipples with SS tube & caps to reduce weight

Table B.3 (sheet 3 of 3)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
171	Arkansas 1	12/77	Decay heat cooler inlet vent line	weld leaked	failure due to vibration	3/4" half coupling & vent valve re- placed with sockolet & plug
226	Three Mile Is. 2	1/79	Decay heat pipe	weld cracked & leaked	result of vibration	- weld repaired - evaluated to determine if additional pipe hangers were required

Table B.4 Summary of NPE Excerpts (PWR-2, VIII.A.). PWR CVCS Piping Vibration Issues (sheet 1 of 14).

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
4	Palisades	3/70 3/72	Charging discharge	leaking cracks at SW fittings	vibration from positive displacement discharge pumps	- added Green Accumulator - replaced SW elbows by sweeping elbows
11	Beznau	1969	Charging line	failed socket welds	?	?
2	Pt. Beach 1	1970	Charging/discharge branches	cracks	- vibrations from reciprocating charging pumps (pulse frequencies) - installation faults	- shortened branches - repaired welds
44 (3 instances)	Kewanee	2,3,4/74	Charging/system sensing lines 3/8"	cracks, leaks	- fluid pulsations - vibrations (positive displacement pumps)	- shortened branches - added acoustic filter to screen out high-frequency vibrations
63	Kewanee	1/75	3/4" branch to 2" charging line	cracks, leaks	- fluid pulsations - vibrations (Positive displacement pumps)	- shortened branches - added acoustic filter to screen out high-frequency vibrations - initiated further studies
34	Zion 1	6/73	3/4" branch from positive displacement pump discharge	360° cracks, leaks (@ 200 gal.)	- vibration/fatigue from positive displacement charging pumps (cont.)	- replaced spool & elbow

Table B.4 (sheet 2 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
34 (cont.)	Zion 1				- detailed examination showed cyclic tensile stress at low magnitude to be the cause	
56	Ginna	5/71	3/4" drain @ inlet manifold to pump	leaks at SW	intergranular corrosion due to installation method	replaced pipe portion
56	Ginna	6/74	3/4" branch connection	leaks	corrosion	replaced
56	Ginna	7/74	3/4" weldolet at vent	leaks, cracks	installation (weld) errors (Further investigation showed <u>fatigue</u> to be the cause of the above 3 events)	
58	Palisades	8/74	Charging/ 3/4" vent on letdown line	cracks, leaks (75% of circumference)	fatigue failure & possible design inadequacy	removed & plugged
63	Kewanee	1/75	Charging/ 3/4" vent on charging line	cracks, leaks	fatigue due to fluid pulsations	reduced vent line length & increased wall thickness
64	Zion 1	11/73	Charging/ 3/4" instrument line connected to flow meter	leakage from socket weld	- fatigue due to mechanical vibrations - piping not according to design thickness	- repaired weld - replaced piping with correct thickness

Table B.4 (sheet 3 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
64	Zion 2	10/74	Charging/ 3/4" pressure indicator line on discharge piping	leak, crack	weld failure, believed to have occurred before present damping was added	repaired weld
64	Zion 2	11/74	Charging/ 3/4" pressure indicator line on suction piping	leak, crack	weld failure due to pump pulsations	- repaired weld - investigated feasibility of installing pres- sure pulsation damping on pump
64	Zion 2	12/74	Charging/ 3/4" pressure indicator line on suction piping	leak, crack next to weld	fatigue due to wall thickness change resulting from welding	replaced piping
76	Zion 2	4/75	Charging/ 3/4" pressure indicator line on discharge piping	weld leak	fatigue due to pump-induced piping vibrations	- rerouted pressure indicator - added additional restraints
81	Three Mile Is. 1	7/75	Make-up/ 3/4" vent on suction line	2 weld leaks	cyclic fatigue failure	- repaired welds - modified supports to increase dis- tance from dis- charge pump to first support

Table B.4 (sheet 4 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
93	Indian Pt. 2	7/75	Charging/ Vent connec- tion on charging line	Weld leak	vibration-induced stress	reduced vent pipe length
98	Three Mile Is. 1	8/75	Make-up/ 3/4" vent on suction line	leak, crack in weld	fatigue due to make-up pump vibration	- repaired weld - replaced portion of vent line - pipe guides rebuilt to reduce vibration
120	Yankee Rowe	8/75	Charging/ Relief valve nozzle on charging pump	crack in flange	vibration from charging pumps	replaced flanged relief valve
120	Yankee Rowe	2/76	Charging/ Relief valve discharge line	leakage from welds	vibration-induced fatigue failure	replaced flange & tee
120	Yankee Rowe	2/76	Charging/ Discharge header	leak in weld of reducer	vibration-induced fatigue failure	- repaired weld - prepared design changed to install pulsation dampers on charging pump discharge

Table B.4 (sheet 5 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
130	Yankee Rowe	5/76	Charging/ Pump relief valve	weld crack, leakage	vibration-induced fatigue failure	- replaced pipe section - prepared design change to install pulsation dampers & sweeping elbows on charging pump discharge
144	Pt. Beach 2	8/76	Charging/ Pump suction header	weld crack, leakage	unsubstantial branch connection	repaired weld
159	Millstone 2	1/76	Charging/ 3/4" discharge relief valve inlet line	2 weld leaks	fatigue	repaired welds
159	Millstone 2	8/76	Charging/ 3/4" discharge relief valve outlet line	weld leak	fatigue	- repaired weld - relocated exist- ing hanger
164	Millstone 2	10/76	Volume control outlet line	pipe saddle crack, leakage	fatigue due to vibration	support configura- tion changed, removing pipe welds
166	Palisades	11/76	Charging/ Pump suction header	pipe leak	- pipe vibration - field-fabricated tee	conducted review of possible tee replace- ment & adequacy of pipe clamp

Table B.4 (sheet 6 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
175	Millstone 2	12/76	Charging/Discharge relief valve	weld leak, crack	fatigue due to vibration	repaired weld
176	Salem 1	1/77	3/4" letdown vent line	pipe break, leakage	excessive vibration caused pipe shear	- removed vent line - socket plugged & welded
184	Calv. Cliffs 2	12/76	Charging/Relief valve line	weld crack	possible pump/suction line resonance	repaired weld
184	Calv. Cliffs 2	1/77	Charging/Suction & discharge lines	socket weld cracks	possible pump/suction line resonance	redesigned suction line restraints
188	Indian Pt. 2	2/77	Seal injection header	socket weld connection leak	vibration-induced stress	vent assembly removed & capped nipple installed
196	Indian Pt. 2	3/77	Charging/Seal injection header	socket weld leaks	excessive vibration	- repaired weld - study undertaken to provide damping of the charging system
204	Indian Pt. 2	3/77	Charging/Relief valve line	socket weld leak	excessive vibration	study undertaken to provide damping of the charging system

Table B.4 (sheet 7 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
205	Calv. Cliffs 2	4/77	Volume control discharge line	weld crack	excessive vibration induced by a possible pump/suction line resonance	redesigned suction line restraints
209	Yankee Rowe	2/77	Charging/ Discharge line, pump relief line	weld leak	high stress on weld & charging pump nipple	replaced pipe section
238	Calv. Cliffs 1	7/76	Charging/ Inlet line, relief valve line	pinhold weld leak	fatigue resulting from pump operation during heat-up & cool-down	repaired weld
238	Calv. Cliffs 1	11/76	Charging/ Discharge line, relief valve line	weld leak	vibration fatigue	study initiated on the charging system vibration problem
238	Calv. Cliffs 1	8/77	Charging/ Suction line, relief valve line	weld leak, crack	vibration fatigue	study initiated on the charging system vibration problem
238	Calv. Cliffs 1	9/77	Charging/ Discharge line	weld leak	vibration fatigue	study initiated on the charging system vibration problem
243	Indian Pt. 3	9/76	Charging/ SI header vent line	socket weld leak	fatigue induced by high moment force of heavy flange on vent line	- repaired weld - removed flange & replaced with cap

Table B.4 (sheet 8 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
253	Pt. Beach 2	12/71	RHR/ Suction line to RHR pump	vibration on valve during RCP operation of RHR pumps off	unknown	Westinghouse investi- gated but could not duplicate vibration
259	F. Calhoun 1	1,2/78	Charging/ Discharge relief valve line	weld crack, leak	vibration	- repaired weld - initiated design modification to add pump suction & discharge sup- pressor
264	Calv. Cliffs 2	1/78	Charging/ Suction line test connec- tion	weld crack, leak	vibration	repaired weld
272	Calv. Cliffs 1	4/78	Charging/ Suction relief valve line	weld leaks	vibration	added pipe support to eliminate vibrations
279	Calv. Cliffs 1	9/75	Charging/ Charging header instru- ment line	socket weld crack	weld examined metalurgically to identify failure mode; no definite cause given	repaired weld
280	Ft. Calhoun 1	4/78	Charging/ Suction header relief valve line	weld crack, leak	pipe vibration	repaired weld

Table B.4 (sheet 9 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
283	Ft. Calhoun 1	5/78	Charging/ Charging header elbow	weld crack	vibration	- repaired weld - vibration damping modification to be installed
235	Calv. Cliffs 1	5/78	Charging/ Discharge relief valve line	pinhole weld leak	fatigue resulting from pump operation during heat-up & cool-down and/or vibration	repaired weld
288	Turkey Pt. 4	3/78	Charging/ Relief line	weld leak	pump operational pressure pulsation & vibration effects	repaired weld
290	Calv. Cliffs 2	6/78	Charging/ Suction header	weld crack with pinhole leak	fatigue resulting from pump operation during heat-up & cool-down and/or vibration	repaired weld
295	Calv. Cliffs 2	4/78	Charging/ Suction header	weld crack on half- coupling	piping vibration	request submitted for additional pipe support & piping misalignment
296	Calv. Cliffs 2	8/78	Charging/ Suction line	weld crack, leak	cracking found in heat-affected zone of previous weld repair	- weld soft-patched, then encapsulated - pipe section to be replaced

Table B.4 (sheet 10 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
297	Turkey Pt. 3	6/78	Charging/ Line to cold leg	evidence of excessive motion	2 adjacent pipe supports were inoperable, permit- ting excessive pipe motion	- supports repaired - engineering sup- port evaluation initiated - installation of pulsation dampers planned
302	Indian Pt. 2	5/77	Charging/ Drain line of suction header	elbow leak	excessive vibration of charging system	- repaired ? - studies continued to determine modification
308	Trojan	9/77	Charging/ Suction line	pipe crack, leak (cir- cumferential)	vibration-induced fatigue	- replaced pipe - removed pipe- welded supports
310	Kewannee	9/78	Charging/ Discharge line	weld crack, leak	operational stress & vibration	- repaired weld - aligned & re- welded discharge line flange
317	Turkey Pt. 3	5/78	Charging/ 3/4" vent line off charging line	weld crack & leakage; short circumferen- tial crack	cyclic stress due to vibration & pulsation	- replaced nipple - installation of pulsation dampers planned
319	Indian Pt. 2	11/78	Seal injection header drain line	socket weld leak	vibration-induced stresses	modified drain valve assembly

Table B.4 (sheet 11 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
329	Calv. Cliffs 2	12/78	Charging/ Suction relief valve line	socket weld crack	fatigue resulting from pump operation during heat-up & cool-down and/or vibration	- repaired weld - initiated design modification to add desurgers to the charging pumps to alleviate vibration
332	Davis Besse 1	1/79	Make-up/ Discharge of tube oil pumps	nipple oil spray leak	fatigue failure caused by vibration	replaced nipple & bushing
333	Ft. Calhoun 1	12/78	Charging/ Pump suction branch line	weld crack, leak	pipe vibration	- repaired weld - modification to reduce pump vibra- tion under evalua- tion
338	Turkey Pt. 4	3/79	Charging/ Suction lines	pipe cracks adjacent to weld between suction pipe & drain line	pump operational pressure pulsation & vibration effects	- repaired weld - modifications to reduce vibrations to be added
364	Crystal River 3	6/79	Make-up/ Discharge drain line	pinhole weld area leak	cyclic vibration fatigue	- repaired pipe - incorporated additional support

Table B.4 (sheet 12 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
374	Calv. Cliffs 1	9/79	Charging/ Discharge drain line	socket weld crack	vibrations due to flow pulsations	design modifications under review to install larger pump discharge damper & suction stabilizer
374	Calv. Cliffs 1	10/79	Charging/ Discharge relief valve line	weld crack, leak	vibrations due to flow pulsations	review of design modifications underway
382	PWR's in general (NRC Review)	5/78	Charging/ Suction & discharge lines	pipe cracks	high-cycle fatigue	NRC believes pipe that absorbs vibra- tory loads originat- ing in the pump should adequately endure maximum alternating stress for ∞ number of cycles
388	Crystal River 3	10/79	Make-up/ Discharge vent valve line	weld area leak	cyclic vibration fatigue	repaired weld
390	Calv. Cliffs 1 & 2	10/79	Charging/ Suction & Discharge	general system modi- fication for cracks	vibration-induced by operation of positive displace- ment pumps	For suction lines: - changed pipe sch. 10 to 40 on line to 1st suction valve (cont.)

Table B.4 (sheet 13 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
390 (cont.)	Calv. Cliffs 1 & 2					<ul style="list-style-type: none"> - changed socket fittings to butt welded fittings - installed suction stabilizers <p>For discharge lines:</p> <ul style="list-style-type: none"> - installed discharge dampers - changed operational procedures
391	Crystal River 3	9/79	Make-up/ Pump discharge drain line	pinhole leak in weld area	cyclic vibration fatigue	repaired weld
393	Salem 1	?/79	Charging/ Pump suction header	Failure?	Vibration	<ul style="list-style-type: none"> - added leaf spring dampers on SI line - added friction plate vibration eliminators on discharge & suction lines - rerouted suction line to eliminate gas pockets - increased pipe schedule - strengthened supports

Table B.4 (sheet 14 of 14)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
397	Ft. Calhoun 1	11/79	Charging/ Discharge & suction piping	failures (addendum to item no. 333)	vibration-induced failures	added accumulators on discharge & suction lines
400	Rancho Seco	11,12/79	Make-up/ Discharge header drain line	pinhole socket weld leak	stress cracking due to vibration	removed pipe & replaced with new pipe nipple & socket weld cap
403	Crystal River 3	9/79	Make-up/ Pump casing drain line	weld crack	cyclic stress fatigue	repaired weld
403	Crystal River 3	12/79	Make-up/ Discharge drain line	leak in weld area of pipe elbow	cyclic stress fatigue	- repaired weld - installed pipe restraints on discharge lines
407	Indian Pt. 3	1/80	Charging/ Charging line vent	socket weld leak	?	repaired weld

APPENDIX C

Summary of items excerpted from NPE (Reference 6), and of relevance to the present study (BWR issues).

List of Tables

- C.1 BWR Feedwater Piping Issues
- C.2 BWR Core Spray Piping Issues
- C.3 BWR Residual Heat Removal Piping Issues
- C.4 BWR High Pressure Coolant Injection Piping Issues

Table C.1 Summary of NPE Excerpts (BWR-2, VI.E.). BWR FW Piping Issues

(sheet 1 of 2)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
4	Nine Mile Pt. 1	1969-70 (test)	FW, CRDH	failure of small diameter piping & control components	high-frequency vibration "almost beyond sensory perception" at high-speed operation of pumps ("... seemed to be produced by hydraulics...")	modified bypass designs
27	Dresden 3	4/74	FW/ Test line 3/4"	leaks	- weld strike arc gouge initiated crack - relative thermal expansion - header vibrations	removed rigid, independent support
30	Dresden 2	7/74	FW/ 10" drain	cracks at support point	- vibration of main header - fatigue	- repaired - added to study of unit 3 vibration problems
45	Quad Cities 2	6/74	FW/ 3/4" test connection	cracks	fatigue failure due to unsupported cantilever arrangement	added support
45	Quad Cities 2	7/75	FW/ 4" flushing	leaks	flow-induced vibration	replaced pipe segment
45	Quad Cities 2	8/75	FW/ Regulating station	vibrations, cracks, leaks	flow induced at valve station	- added snubbers - made various other modifications to reduce vibrations

Table C.1 (sheet 2 of 2)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
50	Quad Cities 2	10/75	FW/ Flush line	vibrations, cracks, leaks	flow-induced vibrations	- added supports - added reinforcing sleeve - further study of vibration problem
92	Dresden 1,2,3 Quad Cities 1, 2	(early operation)	Condensate & FW	pump surging, vibrations	pump problems	modified pumps
103	Dresden 3	9/78	FW/ Low flow lines	leaking cracks	excessive flow	resized orifices for flow reduction

Table C.2 Summary of NPE Excerpts (BWR-2, VII.E.). BWR Core Spray Piping Issues

(sheet 1 of 3)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
1	Oyster Creek 1	6/71 (test)	Core Spray	- water hammer - large pipe movement (2.5")	pump discharge into empty line	installed jockey pumps
3	Dresden 2	3/71 (test)	Core Spray	severely damaged seismic restraints & hangers (loads approx. 10 times design loads)	water hammer due to pump discharge into empty line	- replaced cast han- ger parts by forged ones - installed jockey pumps
18	Arnold	4/74	Core Spray	- seismic restraint anchor pulled out of con- crete - max. pressure 370 psig (cf. tech. specif. limit of 250 psig)	water hammer due to pump discharge into empty line (inadvertant)	upgraded control room procedures
22	Dresden 3	11/74	Core Spray	core spray valve limit switches damaged	water hammer	- repaired damage - detailed inspec- tion
51	Dresden 2	7/76	Core Spray	valve limit switches broken	series of earlier water hammers	repaired

Table C.2 (sheet 2 of 3)

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NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
63	Arnold	2/77 (test)	Core Spray	- loud noise - valve damage	water hammer due to inadvertent core spray initiation	repaired & inspected
25	Dresden 2	1/75	Core Spray	small cracks in 10" core spray lines near RPV	stress-assisted corrosion (?)	- NRC shut down all BWR's for inspec- tion - SS changed to CS for portions of pipe
26	Brown's Ferry 1	2/75	Core Spray/ 3/4" test connection	leaks	metal fatigue due to mechanical vibration	added supports
28	Dresden 2	2/75	Core Spray (same as item 25)	cracks	(see item 25)	- replaced pipe portion - examined welds UT - inspected Unit 3 (no cracks found)
33	Tsuruga (Japan)	3/75	Core Spray	(same as item 25)	(see item 25)	replaced SS by CS
40	Fukushima 1	Spr/75	Core Spray	(same as item 25)	(see item 25)	replaced SS by CS
49	Oyster Crack 1	4/76	Core Spray sensing line	leaks	fatigue failure	- replaced line - inspected all other similar lines

Table C.2 (sheet 3 of 3)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
66	Peach Bottom 2	5/77	Core Spray	crack in HAZ of 12" elbow near RPV covered 215° of pipe ID	stress-assisted corrosion	replaced pipe portion
67	Peach Bottom 3	1/77	Core Spray	cracks in HAZ of reducer welds	oxygen stress corrosion	replaced spool pieces
74	Brunswick 2, Pilgrim 1, Vermont Yankee	10/77 10/77 8/77	Core Spray	cracks	stress corrosion	replaced SS material with CS
85	Brown's Ferry 3	12/79	Core Spray	cracks close to safe end	stress corrosion (?)	replaced SS pipe with CS
88	Quad Cities	3/80	Core Spray	leaking cracks in elbow welds closest to RPV; axial cracks extending through weld from base metal to base metal	unknown, but not indicative of intergranular stress corrosion	replaced SS pipe with CS

Table C.3 Summary of NPE Excerpts (BWR-2, VII.D). BWR RHR Piping Issues

(sheet 1 of 5)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
2	Quad Cities 1	4/72	RHR	Found in routine inspection - 4 damaged seismic restraints - 2 bottomed & 4 damaged spring hangers - valve inoperable	water hammer due to improper venting	- replaced jockey pump system with condensate transfer system for filling - repaired damage - reevaluated piping system by walk-down & analysis
6	Dresden 2	9/71	Shutdown cooling	During test at 384 MW _e of isolation valves: - vibrational shock felt - recirc. pump vibration alarm - dislocated isolation - abnormal pipe movement of 5/8"	water hammer	- revised test procedures - stress analysis concluded no sustained damage to piping
9	Millstone 1	6/72	LPCI	Found in routine inspection: - 11 dislocated supports - 10" pipe displacement	early water hammer (before "stay-full" system was in operation)	- replaced supports - added lugs

Table C.3 (sheet 2 of 5)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
37	Brown's Ferry 1	3/74	Torus Spray header	inspection foll. tests: 15 of 16 hangers failed	Torus vibration due to extensive relief valve discharge tests	repaired with stronger hangers
58	FitzPatrick	1/75	LPCI drain	crack 70% around pipe	vibration	replaced shortened piece of pipe to reduce vibrations
62	FitzPatrick	3/75	Containment Spray	damaged pipe restraints found in rou- tine inspec- tion	air/water hammer due to inadequate "keep-full" system	- planned to modify "keep-full" system - NDT & stress anal- ysis to verify affected pipe
71	FitzPatrick	7/75	RHR steam supply	damaged pipe restraint found in rou- tine inspec- tion	water hammer during warm-up of HPCI steam supply	replaced restraint
75	Brunswick 2	9/75	RHR	3 instances of water hammer (noise)	water hammer	revised procedures for proper venting & filling
77	Peach Bottom 2	11/75	RHR/ 3/4" vent	leak	flow-induced vibration due to throttled valve	repaired

Table C.3 (sheet 3 of 5)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
78	Monticello	11/75	LPCI/ 1" relief line	leaking crack	excessive cyclic movement due to unrestrained relief valve discharge line	modified line to reduce stresses
105	Millstone 1	11/76	RPV head spray	leak in 6" welding neck flange close to RPV pene- tration	thermal transient stresses during cool- ing spray mode, aggravated by stress corrosion	- replaced spool piece - planned replace- ment with non- stress corroding material
112	Brunswick 2	12/76	RHR	damaged snubber	water hammer caused by steam bubble in heat exchanger at pump start-up	- replaced snubber - procedural modi- fications
123	Hatch 1	12/74	RHR head spray	leak	water hammer due to voids in line (?)	- replaced pipe with modified design - transient tests to determine cause
124	Brown's Ferry 1	9/74	RHR pump seal injec- tion lines	cracks	vibration of pumps during high flow	- replaced Sch 40 with SCH 40 CS - CS nipples later failed - replaced with Sch 80
126	Brunswick 1	3/77	RHR hx/ Steam cond. inlet	damaged snubber & hanger	water hammer	- repaired components - modified venting procedures

Table C.3 (sheet 4 of 5)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
127	Brunswick 2	4/77	RHR/ Torus cooling mode	damaged snubber	water hammer	- repaired components - modified venting procedures
131	Cooper	2/77	RHR/ 1" drain	leak	high-cycle fatigue (mechanical vibrations)	- repaired weldolet - inspection planned for similar connections elsewhere in dry well
133	Brunswick 2	7/77	RHR/ Pump suction at 4" clean-out line	crack at weldolet	crack propagation from previously repaired weld	- replaced 4" weldolet with 8" weldolet plus 8x4 reducer - hydrotested
136	Arnold	9/77	RHR/ Inlet	indication of excessive pipe movement (analysis concluded piping had been overstressed)	water hammer that occurred in August, 1974 (probable cause)	?
139	Monticello	10/77	RHR/ 2" boss connection to relief valve	leaking crack	excessive cyclic movement due to unrestrained vibration	- replaced boss with 3" weldolet & 3x2 reducer - installed relief valve restraint
149	Arnold	1,4/77	RHR/ Fuel pool cooling	damaged snubber	water hammer at each RHR start due to inadequate venting	- repaired damage - modified venting procedures

Table C.3 (sheet 5 of 5)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
150	Brunswick 1	12/77	RHR/ Steam cond. line	damaged snubber	water hammer due to valves leaking steam into hx	- replaced snubber - repaired leaking valves
172	Arnold	12/78	RHR	2 snubbers damaged	water hammer due to inadequately filled system (defective procedure)	- modified procedure - repaired snubbers
176	Brunswick 1	11/77	RHR/ Steam cond. line	inoperable snubber	water hammer due to valves leaking steam into hx	- replaced snubber - repaired leaking valves
179	Quad Cities 1	2/79	RHR/ Vent	crack in 3/4" vent socket	unknown	repaired
180	Dresden 2	3/79	LFCI/ Drain	crack in elbow weld (360°)	improper installa- tion (bottomed out in socket)	repaired
197	Dresden 3	10/79	LFCI	supports damaged	water hammer	repaired

Table C.4 Summary of NPE Excerpts (BWR-2, VII.E). BWR HPCI Piping Issues

(Sheet 1 of 2)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
2	Dresden 2	5/70 (test)	HPCI/ Steam turbine inlet	"dented" pipe section 10' in length - damaged supports & snubbers	steam driven water slug from reactor following high RPV water level	- repaired or replaced components - modified procedures
18	Vermont Yankee	1971 (test)	HPCI/ Turbine exhaust	--	water hammer: water slug impact at check valve due to vacuum in exhaust	modified HPCI & RCIC systems
23	Peach Bottom 2	12/73 (test)	HPCI/ Turbine oil line	oil line ruptured at bursting	- insufficient supports - stresses due to weight and/or vibration	- repaired - added support
25	Brown's Ferry 1	10/72 (test)	HPCI/ Turbine exhaust	"violent movement" of the line, despite 5 kip, and later, 10 kip snubber	water hammer	installed condensing sparger in suppression pool
30	Brown's Ferry 1	4/74	HPCI/ Turbine steam supply	piping & numerous restraints severely damaged	steam driven water slug (water hammer)	- corrected operational instructions - pipe design problem expected to be corrected by pipe designers

Table C.4 (sheet 2 of 2)

NPE Item No.	Plant/Unit	Time of Event	System/Line	Observation	Cause as described	Corrective Measures
36	Arnold	6/74 (test)	HPCI/ Turbine steam supply	pipe hangers & snubbers damaged	steam driven water slug (water hammer)	- reinstructed operators - corrected pro- cedures
57	Arnold	4/75	HPCI Test line 3/4"	test line fracture at weld to weldolet	unknown (possibly due to "external lateral force")	repaired
39	Dresden 2	3/76	HPCI/ 14" safe end	cracks	intergranular stress corrosion	replaced safe end
98	Brown's Ferry 3	1/77	HPCI/ Pump discharge	damaged supports	surging during pump start-up	repaired supports
104	FitzPatrick	9/75	HPCI/ Steam line	damaged supports	water hammer	effected design changes to allow draining of the line
136	Brunswick 2	3/78	HPCI	damaged snubber	water hammer due to sticking check valve	replaced snubber
186	Brown's Ferry 1	1/80	HPCI/ Sensory line	hangers broken	water hammer during system warm-up (?)	repaired damage

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APPENDIX D

Tables & Figures identifying loads typically considered
in current industry design practice.

List of Summary Tables

- D.1.1 PWR: Current Design Loading on the Pressurizer Surge Line.
- D.2.1 PWR: Current Design Loading on Pressurizer Safety/Relief Piping.
- D.3.1 PWR: Current Design Loading on Pressurizer Spray Piping.
- D.4.1 PWR: Current Design Loading on the Residual Heat Removal System and Hot Leg Safety Injection Lines.
- D.5.1 PWR: Current Design Loading on Boron Injection Piping.
- D.6.1 PWR: Current Design Loading on the Safety Injection System
- D.7.1 PWR: Current Design Loading on the Charging System.
- D.8.1 PWR: Current Design Loading on the Letdown System.
- D.9.1 PWR: Current Design Loading on the Mainsteam Piping.
- D.10.1 PWR: Current Design Loading on the Feedwater Piping.
- D.11.1.1 BWR: Current Design Loading on the Safety/Relief Valve Discharge Lines (Wetwell Piping)
- D.11.2.1 BWR: Current Design Loading on Safety/Relief Valve Discharge (Lines Drywell Piping)
- D.12.1.1 BWR: Current Design Loading on the Control Rod Drive Hydraulic System (Insert & Withdraw Lines)
- D.12.2.1 BWR: Current Design Loading on the Control Rod Drive Hydraulic System (Scram Header and Volume Tank)
- D.12.3.1 BWR: Current Design Loading on the Control Rod Drive Hydraulic System (Exhaust, Drive, Charging, and Cooling Lines).

List of Supporting Tables & Figures

Tables and Figures providing supporting information for Summary Tables, included where applicable. Designated as D.X.1, D.X.2, . . . etc. for PWR's and as D.X.X.1, D.X.X.1, D.X.X.2, . . . etc. for BWR's.

- a. Representative Simplified Schematic
- b. Representative Table/Schematic of Thermal Expansion Modes.
- c. Table of Thermal Transient Events Analyzed.
- d. Representative Transients, Plotted as Temperature - Time, Pressure-Time, and Flow-Time.
- e. Representative Force-Time-History Curves for Blowdown or Water/Steam Hammer, at Selected Pipe Segments.

Table D.1-1

PWR: Current Design Loading on the Pressurizer Surge Line

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1)	FAULTED
Pressure	Design: 2485 psig Operating: 2235 psig	Operating: 2235 psig	Test: 3106 psig	Faulted: 2485 psig
Thermal Expansion	Design: 680 ^o F 3 Operating Modes: Table D.1.2	3 Operating Modes: Table D.1.2	N/A	Faulted: 680 ^o F
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Transients: Plant/System Events with Significant Loads. (2)	Line Segments 40, 41, & 42: Table D.1.3 and Figure D.1.4 (3)	Line Segments 40, 41, & 42: Table D.1.3 and Figure D.1.4 (3)	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE=function of OBE
Support/Anchor Movements	TAM: RCL, PRESS.	TAM: RCL, PRESS. SAM: RCL, BLD	N/A	TAM: RCL, PRESS. SAM: RCL, BLD DBA (LOCA): RCL

FIGURE D.1.2 PRESSURIZER SURGE LINE

Thermal Expansion Modes

- (1.) 653°F for the entire piping.
- (2.) As shown at indicated points.
- (3.) As shown at indicated points.

Note: Jurisdictional limits of report are between
and including weld point symbol ● . (Nozzle Safe End Weld)

TABLE D.1.3
CONDITIONS ANALYZED FOR SEGMENT 40 (4)

Transient Event Name	Description	No. of Occurrences
H-1	Heatup-Reduced Spray (35 GPM)	200
H-2	Heatup-Reduced Spray (60 GPM)	600
H-3	Heatup-Full Spray (880 GPM)	800
H-5	Heatup-Full Spray (880 GPM)	400
H-6	Heatup-Full Spray (880 GPM)	200
C-1	Cooldown-Full Spray (880 GPM)	200
C-7	Cooldown-Auxiliary Spray (85 GPM)	200
N-3	Unit Loading at 5%/Minute	18,320
N-4	Unit Unloading at 5%/Minute	18,300
N-8H	Steady State Fluctuations	3,154,325
U-3	Reactor Trip with Cooldown and Safety Injection	10
U-4A	Inadvertent RCS Depressurization and Loss of Flow	105
U-4B	Inadvertent RCS Depressurization and Loss of Loss of Flow	505
U-1	Loss of Load and Loss of Power	120
N-1	Zero Stress (except for Gravity)	250

CONDITIONS ANALYZED FOR SEGMENT 41 (4)

Transient Event Name	Description	No. of Occurrences
H-1	Heatup-Reduced Spray (35 GPM)	200
H-2	Heatup-Reduced Spray (60 GPM)	600
H-3	Heatup-Full Spray (880 GPM)	800
H-5	Heatup-Full Spray (880 GPM)	400
H-6	Heatup-Full Spray (880 GPM)	200
C-1	Cooldown-Full Spray (880 GPM)	200
C-7	Cooldown-Auxiliary Spray (85 GPM)	200
N-7	Large Step Load Decrease	38,845
N-8D	Steady State Fluctuations	3,152,400
U-1	Loss of Load and Loss of Power	120
U-4	Inadvertent RCS Depressurization and Loss of Flow	110
U-5	Small Loss of Coolant	5
U-6	Small Steam Break	5
N-1	Zero Stress	250

TABLE D.1.3 (Continued)
CONDITIONS ANALYZED FOR SEGMENT 42 (4)

<u>Transient Event Name</u>	<u>Description</u>	<u>No. of Occurrences</u>
H-1	Heatup-Reduced Spray (35 GPM)	200
H-2	Heatup-Reduced Spray (60 GPM)	600
H-3	Heatup-Full Spray (880 GPM)	300
H-3	Heatup-Full Spray (880 GPM)	400
H-6	Heatup-Full Spray (880 GPM)	200
C-1	Cooldown-Full Spray (880 GPM)	200
C-7	Cooldown-Auxiliary Spray (85 GPM)	200
N-3	Unit Loading at 5%/Minute	18,310
N-5	Step Load Increase at 10%	20,815
N-7	Large Step Decrease with Steam Dump	2,320
N-8D	Steady State Fluctuations	3,150,005
U-1	Loss of Load and Loss of Power	120
U-3	Reactor Trip with Cooldown and Safety Injection	10
U-4	Inadvertent RCS Depressurization and Safety Injection	100
U-5	Small Loss of Coolant	5
N-1	Zero Stress	250

FIGURE D.1.4

EVENT NAME: PLANT HEATUP - REDUCED SPRAY OPERATION
H₁

SEGMENT NO.: 40, 41, 42

NO. OCCURRENCES: 200

Notes:

Pressure is constant at 325 psia

FIGURE D.1.4 (Continued)

EVENT NAME: REDUCED SPRAY OPERATION , HZ, C4, C5

SEGMENT NO.: 40, 41, 42

NO. OCCURRENCES :

Notes: Pressure is constant at 325 PSIA

FIGURE D.1.4 (Continued)

EVENT NAME: FULL SPRAY OPERATION, (H₀, H₃, H₄, H₅, C₁, C₃, C₂)

SEGMENT NO.: 40, 41, 42

NO. OCCURRENCES: 1400
TOTAL

FIGURE D.1.4 (Continued)

EVENT NAME: UNIT LOADING @ 570/MIN (NORMAL CONDITION)

SEGMENT NO. N-3

NO. OCCURRENCES 13,300

FIGURE D.1.4 (Continued)

EVENT NAME: UNIT UNLOADING @ 5% / MIN. (NORMAL CONDITION)

SEGMENT NO. 11-1

NO. OCCURRENCES 3,500

FIGURE D.1.4 (Continued)

EVENT NAME: *STEP LOAD INCREASE OF 10% (NORMAL CONDITION)*

SEGMENT NO. *N-5*

NO. OCCURRENCES *2000*

FIGURE D.1.4 (Continued)

EVENT NAME: LOAD STEP DECREASE WITH STEAM DUMP
(NORMAL CONDITION)
SEGMENT NO. 11-7 NO. OCCURRENCES 200

FIGURE D.1.4 (Continued)

EVENT NAME: STEADY STATE FLUCTUATIONS A-INITIAL
(NORMAL CONDITIONS)
SEGMENT NO. A-8D NO. OCCURRENCES 5000

FIGURE D.1.4 (Continued)

EVENT NAME: STEADY STATE FLUCTUATIONS B-2 RANDOM
(NOZZLE CONDITIONS)

SEGMENT NO. 4-54

NO. OCCURRENTES 500,000

FIGURE D.1.4 (Continued)

EVENT NAME: LOSS OF LOAD & LOSS OF POWER (UPSET CONDITION)

SEGMENT NO. 11-1

NO. OCCURRENCES 120

FIGURE D.1.4 (Continued)

EVENT NAME: LOSS OF LOAD & LOSS OF POWER (UPSET CONDITION)

SEGMENT NO. 41-1

NO. OCCURRENCES 120

EVENT NAME: REACTOR TRIP W/ COOLDOWN & SAFETY INJECTION
 (UPSET CONDITION) FIGURE D.1.4 (Continued)
 SEGMENT NO. 4-3 NO. OCCURRENCES 10

FIGURE D.1.4. (Continued)

EVENT NAME: INADVERTENT RCS DEPRESSURIZATION & LOSS OF FLOW
(UPSET CONDITION)
SEGMENT NO. 4-4 NO. OCCURRENCES 100

FIGURE D.1.4 (Continued)

EVENT NAME: *SMALL LOSS OF COOLANT (UPSET CONDITION)*

SEGMENT NO. *4-5*

NO. OCCURRENCES *5*

EVENT NAME: SMALL STEAM BREAK (UPSET CONDITION)

FIGURE D.1.4 (continued)

SEGMENT NO. 11-6

NO. OCCURRENCES 5

Table D. 2.1

PWR: Current Design Loading on Pressurizer Safety/Relief Piping

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1)	FAULTED
Pressure	Design: 2485 psig Operating: 2235 psig	Operating: 2235 psig	Test: 3106 psig	Faulted: 2560 psig
Thermal Expansion	Design: 680 ^o F 8 Operating Modes: Figure D.2.5	8 Operating Modes: Figure D.2.5	N/A	Faulted: 680 ^o F
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Transients: Plant/System Events with significant loads. (2)	N-1, N-2, N-89A, N-89B, N-8A, & N-8B: Table 4.2.3, Figure D.2.4 (3)	U - 3B: Table D.2.3 & Figure D.2.4(3)	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE = function of OBE
Support/Anchor Movements	TAM: PRESS.	TAM: PRESS. SAM: BLD, PRESS.	N/A	TAM: PRESS SAM: BLD, PRESS.
Blowdown	PSV Actuation: Table D.2.6	PSV Actuation: Table D.2.6	N/A	PSV Actuation: Table D.2.6

FIGURE D.2.2 PRESSURIZER SAFETY AND RELIEF LINE

Note: Jurisdictional boundary is between and including weld symbols ●, but not including valve and flange bodies.

FIGURE D.2.2 (Continued)

<u>Line Segment</u>	<u>Line Segment Boundaries</u>
45	From Pressurizer connection to upstream side of 1NC1, 1NC2, and 1NC3.
47a	From Pressurizer connection to the first 6 x 3-inch tee. Note: Temperatures and Pressures for 47a are identical to temperatures and pressures for 47.
47	From the first 6 x 3-inch tee to upstream side of Power Relief Valves 1NC34A, 1NC32B, and 1NC36B.

TABLE D.2.3 TRANSIENT EVENTS ANALYZED FOR SAFETY/RELIEF PIPING (4)

Transient Event Name	Description	No. of Occurrences
N-1	Ambient	250
N-2	Steady State	100000
U-3B	Inadvertent RCS Depressurization	10
N-89A	Safety and Relief Valve Operation (Start)	40
N-89B	Safety and Relief Valve Operation (End)	40
N-8A	Relief Valve Operation (Start)	45
N-8B	Relief Valve Operation (End)	45

FIGURE D.2.4.

EVENT NAME: PRESSURIZER RELIEF VALVE OPERATION (N-8)

SEGMENT NO.: 47

NO. OCCURRENCES: 85

Notes: Above is the highest thermal shock obtained from the steady state temperature distribution study. Note that due to that distribution no thermal shock is experienced in segment 47a.

FIGURE D.2.4 (Continued)

EVENT NAME: PRESSURIZER SAFETY VALVE OPERATION (N-9)

SEGMENT NO.: 45

NO. OCCURRENCES: 40

Notes: Above is the highest thermal shock obtained from the steady state temperature distribution study

FIGURE D.2.5 THERMAL EXPANSION MODES

Note: ○ - Thermal Boundary Point

Jurisdictional boundary is between and including weld symbols ●, but not including valve and flange bodies.

Figure D.2.5 (Continued)

Thermal Operating Mode Temperatures

Thermal Boundary Point	Operational Modes ($^{\circ}$ F)							
	1	2	3	4	5	6	7	8
A	200	200	200	200	200	200	ENTIRE LINE AT 680 $^{\circ}$ F	200
B	680	580	680	680	680	680		680
C	200	200	200	200	200	200		200
D	680	680	680	680	680	680		680
E	680	680	680	680	680	680		680
F	200	200	200	200	200	200		200
G	680	200	200	200	680	680		200
H	680	70	680	120	680	680		120
I	680	70	680	680	680	680		120
J	680	70	680	680	680	680		680
K	680	70	120	680	680	120		680
L	680	200	200	680	680	200		680
M	200	200	200	200	200	200		200
N	120	70	120	120	120	120		120
O	680	70	120	680	680	680		680
P	680	70	120	680	680	680		680
Q	680	70	680	680	680	680		680
R	680	200	680	680	200	680		200
S	680	200	200	680	200	680		680
T	680	70	120	680	120	680		120
U	680	200	200	680	200	680		200
V	680	70	120	120	200	680		120
W	680	200	200	200	680	680		200

Figure D.2.6 PRESSURIZER SAFETY VALVE ACTUATION

Force History on First
Straight Line Segment
past Safety Valve

FIGURE D.2.6 (continued) PRESSURIZER SAFETY VALVE ACTUATION

Table D.3.1

PWR: Current Design Loading on Pressurizer Spray Piping

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1.)	FAULTED
Pressure	Design: 2485psig Operating: 2350 psig	Operating: 2350 psig	Test: 3106psig	Faulted: 2550psig
Thermal Expansion	Design: 558 ^o F 4 Operating Modes: Figure D.3.2	4 Operating Modes: Figure D.3.2	N/A	Faulted: 557 ^o F
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Transients: Plant/System Events with significant loads. (2)	N-1, N-90, & N-9U (3.) : Table D.3.3 & Figure D.3.4	U-2U, U-5, U-6D, U-6U, U-7A, U-7B, U-8B: Table D.3.3 & Figure D.3.4	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE=function of OBE
Support/ Anchor Movements	TAM: RCL, PRESS.	TAM: RCL, PRESS. SAM: RCL, BLD	N/A	TAM: RCL, PRESS. SAM: RCL, BLD DBA(LOCA):RCL

FIGURE D.3.2 SPRAY LINE (5)

To Regenerative Heat Exchanger

Thermal Expansion Modes:

- (1.) All segments except 24 are at 592°F, and as shown on the figure.
- (2.) All segments are at 32°F, and as shown on the figure.
- (3.) As shown on the figure.
- (4.) All segments are at 212°F, and as shown on the figure.

Note: Jurisdictional boundary is between and including weld symbols •, but not valve bodies.

FIGURE D.3.2 (Continued)

<u>Line Segment</u>	<u>Line Segment Boundaries</u>
50a	The 6-inch spray line from the pressurizer to the connection with the auxiliary spray line.
50b	The 6-inch spray line from the connection with the auxiliary spray line to the 6 x 6 x 6-inch tee connection.
50c	The loop 2,4-inch spray line from the connection with the spray valve bypass line to the 6 x 6 x 6-inch tee connection.
50d	The loop 1,4-inch spray line from the connection with the spray valve bypass line to the 6 x 6 x 6-inch tee connection.
50e	The loop 2,4-inch spray line from the downstream side of 1NC29 to the connection with the spray valve bypass line.
50f	The loop 1,4-inch spray line from the downstream side of 1NC27 to the connection with the spray valve bypass line.
51a	The loop 2,4-inch spray line from the connection with the spray valve bypass line to the upstream side of 1NC29.
51b	The loop 1,4-inch spray line from the connection with the spray valve bypass line to the upstream side of 1NC27.
51c	The loop 2,4-inch spray line from the connection with the loop 2 cold leg to the connection with the spray valve bypass line.
51d	The loop 1,4-inch spray line from the connection with the loop 1 cold leg to the connection with the spray valve bypass line.

TABLE D.3.3 TRANSIENT EVENTS ANALYZED FOR SPRAY LINE (4)

Transient Event Name	Description	No. of Occurrences
N-1	Ambient	250
N-9D	Aux Spray during Cooldown (Start)	200
N-9U	Aux Spray during Cooldown (End)	200
U-2U	Loss of Load	80
U-5	Loss of Power	40
U-6D	Inadvertent Auxiliary Spray (Start)	10
U-6U	Inadvertent Auxiliary Spray (End)	10
U-7A	Reactor Trip with Cooldown (Start)	230
U-7B	Reactor Trip with Cooldown (End)	90
U-8B	Inadvertent RCS Depressurization	20

FIGURE D.3.4

EVENT NAME: Auxiliary Spray Actuation During Cooldown, N-9

SEGMENT NO.: 50A THRU 51C AND 51D

NO. OCCURRENCES: 200

Notes:

Flow for Segs. 50C, 50D, 52A, 52B, 52C, 52D, 51C, 51D is 1 GPM while no flow is defined for 51A & 51B.

FIGURE D.3.4 (Continued)

EVENT NAME: LOSS OF LOAD) WITHOUT IMMEDIATE TURBINE OR REACTOR
TRIP U-2
SEGMENT NO. 50 A TO 50 F & 51 A TO 51 D NO. OCCURRENCES 30

- NOTES:
- 1) PRESSURE DIAGRAM: SOLID LINE IS USED FOR 50 A TO 50 F
BROKEN LINE IS USED FOR 51 A TO 51 D
 - 2) FLOW DIAGRAM: SOLID LINE IS USED FOR 50 A & 50 B
BROKEN LINE IS USED FOR 50 C - 50 F & 51 A - 51 D
 - 3) THE FLOW IS DIFFERENT BY 1 GPM, THE HIGHER FLOW IS USED.

FIGURE D.3.4 (Continued)

EVENT NAME: LOSS OF POWER (BLACKOUT WITH NATURAL CIRCULATION) U-5

SEGMENT NO. 50A TO 50F AND 51A TO 51C NO. OCCURRENCES 40

- NOTES: 1) FOR SEGMENT 50A TO 50F PRESSURE IS .53 PSI LOWER AT TIME 0, AND BETWEEN 10,760 AND 11,450 SECONDS
2) FOR SEGMENT 50A AND 50E FLOW IS DOUBLE

FIGURE D.3.4 (Continued)

EVENT NAME: INADVERTENT AUXILIARY SPRAY U-6

SEGMENT NO. 50A → 50F & 51A → 51D NO. OCCURRENCES 10

NOTES:

- 1) NO TEMPERATURE CHANGE FOR ALL SEGMENTS EXCEPT 50A
- 2) NO FLOW CHANGE FOR ALL SEGMENTS EXCEPT FOR 50A
- 3) THE FLOW IS DIFFERENT BY 102 GPM, THE HIGHER FLOW IS USED

FIGURE D.3.4 (Continued)

EVENT NAME: REACTOR TRIP WITH COOL DOWN AND INADVERTENT
SI ACTUATION U-7

SEGMENT NO. 50A → 50F & 51A → 51D NO. OCCURRENCES 10

NOTES:

- 1) PRESSURE DIAGRAM SOLID LINE IS USED FOR 50A → 50F
BROKEN LINE IS USED FOR 51A → 51D
- 2) FLOW: SEG. 50A, B HAVE 2 GPM FLOW.
SEG. 50C, D, 51C, D HAVE 1 GPM FLOW.
SEG. 50E, F, 51A, B HAVE 0 FLOW

TABLE D.4.1

PWR: Current Design Loading on the Residual Heat Removal System
and Hot Leg Safety Injection Lines

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1)	FAULTED
Pressure	Design: 2485 psig Operating: 2235 psig	Operating: 2235 psig	Test: 3106 psig	Faulted: 2235 psig
Thermal Expansion	Design: 650°F 4 Operating Modes for Sec- tions ND-01 & ND-09: Table D.4.4 1 Operating Mode for Section NI-08: Figure D.4.3	4 Operating Modes for Sec- tions ND-01 & ND-09: Table D.4.4 1 Operating Mode for Section NI-08: Figure D.4.3	N/A	Faulted: 650°F
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients: Plant/System Events with significant loads (2)	Seg. 1, 2 & 15: N-1, N-2 & N-3 Table D.4.5 Figure D.4.6	Seg. 1, 2 & 15: U-2, U-5, U-6, & U-7 (3) Table D.4.5 Figure D.4.6	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE = function (OBE)
Support/Anchor Movements	TAM: RCL	TAM: RCL SAM: RCL, BLD & RB	N/A	TAM: RCL SAM: RCL, ELD, & RB LOCA (DBA): RCL

(5)

Ⓐ - Thermal Boundary Points Table D.4.4

Jurisdictional boundary is between and including weld symbols , but not including valve bodies.

FIGURE D.4.2 (continued)

<u>Line Segment</u>	<u>EDS Problem No.</u>	<u>Line Segment Boundaries</u>
1	ND-01	From connection at H. L. 3 to upstream side of valve 1ND1B.
2	ND-01	From downstream side of valve 1ND1B to upstream side of valve 1ND2A
15	ND-01, NI-09	From downstream side of valves 1NI157, 160 and 134 to connection with Hot Legs 1, 2 and 4 and from downstream of valve 1NI126 to connection with 14" RHR Line.
16	NI-09	From downstream side of valves 1NI156 and 1NI159 to upstream side of valves 1NI157 and 1NI160.
17	ND-01, NI-08	From downstream side of valves 1NI125, 129, 126 and 134 to upstream side of valves 1NI128 and 1NI124.
20	ND-01	From reactor containment to upstream side of valves 1NI125 and 1NI129.

FIGURE D.4.3
THERMAL OPERATING MODE FOR SECTION NI-08

Table D.4.4

Thermal Operating Mode Temperatures

Thermal Boundary Point	Thermal Operating Modes ($^{\circ}$ F)			
	1	2	3	4
RCL	No Load	No Load, Leg 350° F	No Load, Leg 350° F	Full Load, Hot Leg 618° F
A	557	350	350	618
B	120	120	120	120
C	557	350	350	618
D	120	120	120	618
E	70	70	70	70
F	557	350	350	618
G	120	120	120	120
H	557	350	350	618
I	120	120	120	120
J	557	350	350	618
K	300	350	70	120
L	300	350	70	120
M	70	70	70	70

TABLE D.4.5
EVENTS ANALYZED FOR SEGMENTS 1, 2 and 15 (4)

<u>Transient Event Name</u>	<u>Description</u>	<u>Number of Occurrences</u>
AMB	Ambient Conditions	280
N1	Heatup	200
N2	Cooldown	200
N3	Plant Loading at 5% per Minute	20,300
U-2	Loss of Load W/O Immediate Turbine or Reactor Trip	80
U-5	Loss of Power	20,540
U-6	Reactor Trip with Cooldown and Inadvertant SI Actuation	679
U-7	Inadvertant RCS Depressurization	24

FIGURE D.4.6

Reference: "Transient Parameter Analysis"

Safety Injection System, Line Segment Number (15)

Residual Heat Removal System, Line Segment Number (1)

Event: HEATUP/STARTUP
Normal Condition

Number of Occurrences= 200

FIGURE D.4.6 (Continued)
Reference: "Transient Parameter Analysis"

Safety Injection System, Line Segment Number (15)

Residual Heat Removal System, Line Segment Number (1)

Event: SHUTDOWN/COOLDOWN
Normal Condition

Number of Occurrences= 200

FIGURE D.4.6 (Continued)

Reference: "Transient Parameter Analysis"

Safety Injection System, Line Segment Number (15)

Residual Heat Removal System, Line Segment Number (1)

Event: PLANT LOADING AT 5% PER MINUTE
Normal Condition

Number of Occurrences= 18,300

FIGURE D.4.6 (Continued)

Reference: "Transient Parameter Analysis"

Safety Injection System, Line Segment Number(15)

Residual Heat Removal System, Line Segment Number(1)

Event: LOSS OF LOAD W/O IMMEDIATE TURBINE
OR REACTOR TRIP

Upset Condition

Number of Occurances= 30

FIGURE D.4.6 (Continued)

Reference: "Transient Parameter Analysis"

Safety Injection System, Line Segment Number (15)

Residual Heat Removal System, Line Segment Number (1)

Event: LOSS OF POWER
 BLACKOUT WITH NATURAL CIRCULATION
 Upset Condition
 Number of Occurrences= 40

FIGURE D.4.6 (Continued)

Reference: "Transient Parameter Analysis"

Safety Injection System, Line Segment Number (15)

Residual Heat Removal System, Line Segment Number (1)

Event: REACTOR TRIP WITH COOLDOWN AND
INADVERTANT S.I. ACTUATION

Upset Condition

Number of Occurrences= 10

FIGURE D.4.6 (Continued)

Reference: "Transient Parameter Analysis"

Safety Injection System, Line Segment Number (15)

Residual Heat Removal System, Line Segment Number (1)

Event: INADVERTANT RCS DEPRESSURIZATION
Upset Condition

Number of Occurances= 20

Reference: FIGURE D.4.6 (Continued)
"Transient Parameter Analysis"

Residual Heat Removal System
Line Segment Number (2)

Event: *HEATUP/STARTUP*
Normal Condition

Number of Occurrences= 200

FIGURE D.4.6 (Continued)

Reference: "Transient Parameter Analysis"

Residual Heat Removal System

Line Segment Number (2)

Event: SHUTDOWN/COOLDOWN
Normal Conditions

Number of Occurrences= 200

FIGURE D.4.6 (Continued)

EVENT NAME: INADVERTANT RCS DEPRESSURIZATION, UPSET CONDITIONS, (U7-6)

NO. OF OCCURRENCES: 20

SEGMENT NO. 15

NOTE: Flow is zero for both cases.

Table D.5.1

PWR: Current Design Loading on Boron Injection Piping

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1.)	FAULTED
Pressure	Design: 2485psig Operating: 2550 psig	Operating: 2550 psig	Test: 3106 psig	Faulted: 2300 psig
Thermal Expansion	Design: 650 ^o F 1 Operating Mode: Table D. 5.2	1 Operating Mode: Table D. 5. 2	N/A	Faulted: 558 ^o F
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Transients: Plant/System Events with significant loads. (2)	N-1A, N-1B, & N-9A (3.): Table D. 5.3	U-3A, U-3B, & U-6A (3.): Table D. 5.3 & Figure D. 5.4	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE = function of OBE
Support/ Anchor Movements	TAM: RCL	TAM: RCL SAM: RCL, BLD	N/A	TAM: RCL SAM: RCL, BLD DBA(LOCA):RCL

FIGURE D.5.2 BORON INJECTION SYSTEM (5)

NOTE:

Jurisdictional boundary is between and including weld symbols • , but not including valve bodies.

TABLE D.5.3 CONDITIONS ANALYZED (4)

Transient Event Name	Description	No. of Occurrences
N-1A	Heat-up	250
N-1B	Cooldown	684
N-9A	Refueling	84
U-3A	Boron Injection	50
U-3B	Boron Injection	50
U-6A	Loss of Flow in One Loop (Temps for the Loop that Lost Flow)	80

FIGURE D.5.4

EVENT NAME: BORON INJECTION U-3

SEGMENT NO.: //

NO. OCCURRENCES: 50

FIGURE D.5.4 (Continued)

EVENT NAME: LOSS OF FLOW IN ONE LOOP (TEMPERATURE FOR THE
LOOP THAT LOST FLOW) U-6

SEGMENT NO.: 11

NO. OCCURRENCES: 83

TABLE D.6.1

PWR: Current Design Loading on the Safety Injection System

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1.)	FAULTED
Pressure	Design: 2485 psig Operating: -Seg. 3: 2300 psig -Seg. 4A, 4B, 4D & 4E: 1520 psig -Seg.: 4C, 14: 650 psig (see table D.6.2)	Operating: -Seg. 3: 2300 psig -Seg. 4A, 4B, 4D, 4E: 1520 psig -Seg.: 4C, 14: 650 psig (see table D.6.2)	Test: 3106 psig	Faulted: -Seg. 3: 2300 psig -Seg. 4 & 14: 650 psig (see table D.6.2)
Thermal Expansion	Design: 650°F 2 Operating Modes Figure D.6.2	2 Operating Modes: Figure D.6.2	N/A	Faulted: -Seg. 3: 558°F -Seg. 4A, 4B, 4C 4D, 4E: 300°F -Seg. 14: 120°F (see table D.6.2)
Gravity	Live & Dead	Live & Dead	Live & Dead	Live & Dead
Transients: Plant/System Events with Significate System Loads (2)	Segments 3, 4 & 14: N-1A, N-2D, N-2U, N-7 & N-9 Seg. 4C: N-1A, N-2A, N-2B, N-7, & N-9 Table D.6.3 Figure D.6.4 (3)	Segs. 3, 4 & 14: U-5B, U-6, & U-9 Seg. 4C: U-5, U-9A Table D.6.3 Figure D.6.4 (3)	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE = function (OBE)
Support/ Anchor Movements	TAM: RCL	TAM: RCL SAM: RCL, BLO	N/A	TAM: RCL SAM: RCL, BLO LOCA (DBA): RCL

FIGURE D.6.2

SAFETY INJECTION SYSTEM (5)

NOTE:

1. Jurisdictional Boundary is between and including weld symbols, but not including valve bodies.
2. All segments and lines are identical for Loops 2, 3 and 4, but are not shown for simplicity.

FIGURE D.6.2 (Continued)

<u>Segment</u>	<u>EDS Problems</u>	<u>Boundaries</u>
3	NI-01, 02, 03, 04	From RCL Cold Leg 1, 2, 3 and 4 to the 10 inch valve 1NI60, 1NI71, 1NI82 and 1NI94, respectively.
4A	NI-01, 02, 03, 04	From 10 inch line at valve 1NI60, 1NI71, 1NI82, and 1NI94 to 6 inch lines.
4B	NI-01, 02, 03, 04	6 inch lines from connection with 2 inch lines (NI-11, 12) to connection with 10 inch lines.
4C	NI-11, 12	2 inch lines from valves 1NI171, 169, 167, 165 to 6 inch lines
4D	NI-01, 02, 03, 04	10 inch lines from valves 1NI59, 70, 81, 93 to 6 inch lines.
4E	NI-01, 02, 03, 04	6 inch lines from valves 1NI175, 176, 180, 181 to 2 inch lines.
14	NI-01, 02, 03, 04	10 inch lines from valves 1NI54A, 65B, 76A, 88B to valves 1NI59, 70, 81, 93 respectively.

TABLE D.6.3(4)
EVENTS ANALYZED FOR SEGMENTS 3, 4, and 14
(EXCEPT 4C)

Transient Event Name	Description	No. of Occurrences
N-1A	Ambient	200
N-2D	Cooldown (Downshock)	200
N-2U	Cooldown (Upshock)	200
N-7	Large Step Load Decrease	141460
N-9	Refueling	80
U-5B	Loss of Load	101
U-6	Loss of Flow in One Loop	80
U-9	Inadvertent RCS Depressurization	41

TABLE D.6.3 (Continued) EVENTS ANALYZED FOR SEGMENT 4C

Transient Event Name	Description	No. of Occurrences
N-1A	Ambient Conditions	255
N-2A	Cooldown - Part 1	200
N-2B	Cooldown - Part 2	200
N-7	Large Step Load Decrease	141460
N-9	Refueling	80
U-5	Loss of Load	97
U-9A	Inadvertent RCS Depressurization	24

FIGURE D.6.4

EVENT NAME: SHUTDOWN / COOLDOWN N-2

SEGMENT NO. SEG 3

NO. OCCURRENCES 200

FIGURE D.6.4 (Continued)

EVENT NAME: SHUTDOWN COOLDOWN N-2

SEGMENT NO. SEG 4A TO 4E

NO. OCCURRENCES 200

NOTES: NO FLOW FOR SEG 4C AND 4D

FIGURE D.6.4 (Continued)

EVENT NAME: REFUELING T-N-1

SEGMENT NO. 3, 4A TO 4E

NO. OCCURRENCES 80

NOTES:

NO FLOW FOR SEG 4C, 4D

FIGURE D.6.4 (Continued)

EVENT NAME: LOSS OF FLOW IN 1 LOOP (TEMPERATURE FOR
THE LOOP THAT LOST FLOW) U-6

SEGMENT NO: 3

NO. OCCURRENCES 80

FIGURE D.6.4 (Continued)

EVENT NAME: INADVERTENT RCS DEPRESSURIZATION V-9

SEGMENT NO. 3

NO. OCCURRENCES 20

FIGURE D.6.4 (Continued)

EVENT NAME: IN ADVERTENT RCS DEPRESSURIZATION U-9

SEGMENT NO. 4A TO 4E

NO. OCCURRENCES 20

NOTES: NO FLOW FOR SEG 4E

Table D.7.1

PWR: Current Design Loading on the Charging System

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1)	FAULTED
Pressure	Design: -Segments 23& 24: 2720psig -Segments 26, 28, 29: 2485psig (see figure D.7.2)	Design: -Segments 23& 24: 2720 psig -Segments 26, 28, 29: 2485psig (see figure D.7.2)	Test: 3106psig	Faulted: -Segment 24: 0 psig. -Segments 26, 28, 29: 2485psig (see figure D.7.2)
Thermal Expansion	Design: 558 ^o F 8 Operating Modes: Table: D.7.3, Figure D.7.2	8 Operating Modes: Table D.7.3, Figure D.7.2	N/A	Faulted: -Segments 26, 28, 29: 557 ^o F -Segments 24: 120 ^o F (see figure D.7.2)
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Transients: Plant/System Events with significant loads (2.)	Segments: 26, 28, 29a & 29b: N-1A, N-3, N-5A, N-5B, N-6B, N-7, N-8A, N-8B, N-10A, N- 12A, N-12B. Segment 24: N-3, N-2A, N-11 Table D.7.4 (3)	Segments 26, 28, 29a, & 29b: U- 4B, U-7A, U-7B, & U-8B. Segment 24: U- 2A, U-2B. Table D.7.4 Figure D.7.5(3)	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE=function(OBE)
Support/ Anchor Movements	TAM: RCL, HX	TAM: RCL, HX SAM: RCL, BLD	N/A	TAM: RCL, HX SAM: RCL, BLD LOCA(DBA):RCL

FIGURE D.7.2 CHARGING LINES (5)

Note: (A) - Thermal Boundary Points Table D.7.3.
Jurisdictional boundary is between and including weld symbols $\oplus$, but not including valve bodies.

FIGURE D.7.2 (Continued)

<u>Line Segment</u>	<u>Line Segment Boundary</u>
23	From downstream side of INV22 to normal spray line connection.
24	From downstream side of INV21A to upstream side of INV22 and INV841.
25	From downstream side of Regenerative Heat Exchanger to upstream sides of INV21A, INV14 and INV16A.
26	From downstream side of INV15 to connection at cold leg of loop 1.
27	From downstream side of INV16A to upstream side of INV17.
28	From downstream side of INV18 to connection at cold leg of loop 4.
29 a	From downstream side of INV14 to upstream side of INV15.
29 b	From downstream side of INV17 to upstream side of INV18.

Table D.7.3

Thermal Operating Mode Temperatures

Thermal Boundary Point	Thermal Operating Modes ($^{\circ}$ F)							
	1	2	3	4	5	6	7	8
A	560	557	560	170	558	558	558	558
B	100% Load	100% Load	560 No Load	120 No Load	558 100% Load	558 100% Load	558 100% Load	558 100% Load
C	100% Load	100% Load	560 No Load	170 No Load	558 100% Load	558 100% Load	558 100% Load	558 100% Load
D	120	528	560	70	100	70	220	380
E	120	528	560	70	100	70	220	380
F	560	528	560	70	480	70	220	380
G	100	528	560	70	100	70	220	380
H	560	528	560	70	100	70	220	380
I	100	528	560	70	100	70	220	380
J	560	569	560	70	100	70	220	380
K	120	120	120	170	100	70	220	380
L	120	120	120	170	120	120	120	120
M	120	120	560	70	100	70	220	380
N	120	120	120	120	100	70	220	380
O	120	528	560	70	100	70	220	380
P	100	528	560	70	100	70	220	380

TABLE D.7.4
EVENTS ANALYZED FOR SEGMENTS 26, 28, 29a AND 29b
(NORMAL AND ALTERNATE CHARGING LINES)

<u>Transient Event</u>	<u>Description</u>	<u>Number of Occurrences</u>
N-1A	Ambient	250
N-3	Normal Operation	125,360
N-5A	Letdown Trip (Prompt Return)	220
N-5B	Letdown Trip (Prompt Return)	200
N-6B	Letdown Trip (Delayed Return)	20
N-7	Charging Trip (Prompt Return)	20
N-8A	Charging Trip (Delayed Return)	100
N-8B	Charging Trip (Delayed Return)	100
N-10A	Charging Flow 50% Increase	50,080
N-12A	Letdown Flow 60% Increase	50,000
N-12B	Letdown Flow 60% Increase	24,000
U-4B	Loss of Load	100
U-7A	Reactor Trip with Cooldown	10
U-7B	Reactor Trip with Cooldown	10
U-8B	Inadvertent RCS Depressurization	220

TABLE D.7.4 (Continued)
EVENTS ANALYZED FOR SEGMENT 24
(AUXILIARY SPRAY LINE)

<u>Transient Event</u>	<u>Description</u>	<u>Number of Occurrences</u>
N-3	Normal Operation (Ambient)	3,375
N-2A	Shutdown/Cooldown	200
N-11	Letdown Flow 40% Decrease and Return to Normal	2,000
U-2A	Inadvertent Auxiliary Spray	10
U-2B	Inadvertent Auxiliary Spray	10
T-3B	Primary Side Leak Test	155

FIGURE D.7.5

EVENT NAME: LETDOWN TRIP WITH PROMPT RETURN TO SERVICE
(N-5)

SEGMENT NO.: 26, 28, 29a, 29b

NO. OCCURRENCES: 200

TRANSIENT REPORT REFERENCE NO.: GS-3.0 . REV. 3

FIGURE D.7.5 (Continued)

EVENT NAME: LETDOWN TRIP WITH DELAYED RETURN TO SERVICE,
(N-6)

SEGMENT NO.: 26, 28, 29a, 29b

NO. OCCURRENCES: 20

FIGURE D.7.5 (Continued)

EVENT NAME: CHG TRIP WITH PROMPT RETURN TO SERVICE
(N-7)

SEGMENT NO.: 26, 28, 29a, 29b

NO. OCCURRENCES: 20

FIGURE D.7.5 (Continued)

EVENT NAME: CHG. TRIP WITH DELAYED RETURN TO SERVICE,
(N-3)

SEGMENT NO.: 26, 28, 29a, 29b

NO. OCCURRENCES: 20

FIGURE D.7.5 (Continued)

EVENT NAME: CHG FLOW 50% INCREASE, (N-10)

SEGMENT NO.: 26, 28, 29a, 29b

NO. OCCURRENCES: 24,000

FIGURE D.7.5 (Continued)

EVENT NAME: LET DOWN FLOW 60% INCREASE (N-12)

SEGMENT NO.: 26, 28, 29a, 29b

NO. OCCURRENCES: 24,000

FIGURE D.7.5 (Continued)

EVENT NAME: LOSS OF LOAD W/O IMMEDIATE TURBINE OR REACTOR
TRIP U-2

SEGMENT NO.: 23

NO. OCCURRENCES: 80

Table D.8.1

PWR: Current Design Loading on the Letdown System

LOAD TYPE	PLANT CONDITION			
	NORMAL	UPSET	TEST (1.)	FAULTED
Pressure	Design: 2485 psig	Design: 2485psig	Test: 3106psig	Faulted: 2385psig
Thermal Expansion	Design: 558°F 2 Operating Modes: Figure D. 8.2	2 Operating Modes Figure D. 8.2	N/A	Faulted: 557°F
Gravity	Live + Dead	Live + Dead	Live + Dead	Live + Dead
Transients: Plant/System Events with significant loads (2.)	N-1A, N-13B Table D. 8.3 Figure D. 8.4(3)	U-5, U-7B, U-8B Table D. 8.3 Figure D. 8.4(3)	N/A	N/A
Seismic Inertia	N/A	OBE	N/A	SSE=function(SSE)
Support/ Anchor Movements	TAM:RCL,HX	TAM:RCL,HX SAM:RCL,BLD	N/A	TAM:RCL,HX SAM:RCL,BLD LOCA(DBA);RCL

TABLE D.8.3
EVENTS ANALYZED FOR SEGMENTS 21 AND 60
(LETDOWN LINE)

<u>Transient Event</u>	<u>Description</u>	<u>Number of Occurrences</u>
N-1A	Ambient	250
N-13B	Letdown Shutoff	100
STEADY	Normal Operation	570
U-5	Loss of Flow in One Loop	80
U-7B	Reactor Trip with Cooldown	10
U-8B	Inadvertent RCS Depressurization	20

FIGURE D.8.4

EVENT NAME: LOSS OF FLOW IN ONE LOOP (TEMP FOR THE LOOP THAT LOST FLOW), U-5

SEGMENT NO.: 21

NO. OCCURRENCES: 100

FIGURE D.8.4 (Continued)

EVENT NAME: LET DOWN SHUTOFF (N-13)

SEGMENT NO.: 21

NO. OCCURRENCES: 100

Table D.9.1

PWR: Current Design Loading on the Mainsteam Piping

PLANT CONDITION	LOAD TYPE
NORMAL	Steam Generator Pressure: 849 psia Gravity Thermal Expansion: 600°F TAM: Steam Generator
UPSET	Steam Generator Pressure: 849 psia: Gravity Thermal Expansion: 600°F Support Anchor Movements TAM: Steam Generator SAM: Steam Generator, RB, BLD OBE: Inertia Steam hammer = Fast Valve Closure (10.) see figures D.9.2 & D.9.3 Blowdown: Safety Valve Opening (see Faulted Condition)
FAULTED	Steam Generator Pressure: 849 psia Gravity Support Anchor Movements SAM: Steam Generator, RB, BLD DBA (LOCA): Steam Generator Steam hammer: Fast Valve Closure (10.), Mainsteam rupture see figures D.9.4 & D.9.5 Blowdown: Safety Valve Opening Thrust Force of 23,320 lbs.

FIGURE D.9.2

MAINSTREAM

FIGURE D. 9. 2 (Continued)

MAIN STEAM

D-93

Detail Near
Stop Valves

FIGURE D.9.3UPSET STEAM HAMMERNEAR STEAM GENERATORFORCING FUNCTION

FIGURE D.9.4 FAULTED STEAM HAMMER

FIGURE D. 9. 5 FAULTED STEAM HAMMER
FORCING FUNCTION
NEAR STEAM GENERATOR

Table D.10.1

PWR: Current Design Loading on the Feedwater Piping

PLANT CONDITION	LOAD TYPE
NORMAL	Steam Generator Pressure: 849 psia Gravity Thermal Expansion: 446°F TAM: Steam Generator
UPSET	Steam Generator Pressure: 849 psia Gravity Thermal Expansion: 446°F Support Anchor Movements TAM: Steam Generator SAM: Steam Generator, RB, BLD OBE: Inertia Waterhammer: Fast Valve Closure (9.) see figures. D10.2 & D10.3
FAULTED	Steam Generator Pressure: 849 psia Gravity Support Anchor Movements SAM: Steam Generator, RB, BLD DBA(LOCA): Steam Generator Waterhammer: Fast Valve Closure (9.), Feedwater Rupture

FIGURE D.10.2

CONDENSATE AND FEEDWATER SYSTEM MODEL

FIGURE D.10.3 FORCING FUNCTION UPSTREAM THE
REGULATING VALVE UPSET WATERHAMMER

FIGURE D.10.3 (Continued)
FORCING FUNCTIONS NEAR THE STEAM GENERATOR UPSET WATERHAMMER

TABLE D.11.1.1

BWR: Current Design Loading on Safety/Relief
Valve Discharge Lines (Wetwell Piping)

PLANT CONDITION	LOAD TYPE
Service Level A (6)	Pressure (625 psi design pressure) Gravity
Service Level B (6)	Pressure (625 psi design pressure) Thermal Expansion (500°F design temp.) TAM: Torus, Drywell/Wetwell Vent Lines S/RV Inertia S/RV Discharge: Peak Pressure loads (due to water clearing transient), Thrust Loads (Water and Pressure Wave), Air-Bubble Induced Drag Loads, & Air Clearing Differential Loads) SAM (OBE): BLD, Torus/Vent Lines S/RV Anchor Motion Gravity
Service Level C (6)	Pressure (625 psi design pressure) S/RV Discharge: Peak Pressure loads (due to water clearing transient), Thrust Loads (Water and Pressure Wave), Air-Bubble Induced Drag Loads & Air-Clearing Differential Loads) SSE: Inertia LOCA (SBA): Condensation Oscillation, Chugging, Inertia S/RV Inertia Gravity
Service Level D (6)	Pressure (625 psi design pressure) S/RV Discharge: Peak Pressure Loads (due to water clearing transient), Thrust Loads (Water and Pressure Wave), Air-Bubble Induced Drag Loads & Air Clearing Differential Loads) SSE: Inertia S/RV Inertia Gravity LOCA (DBA): Pool Swell Impact, Air-Bubble Induced Drag Loads, Condensation Oscillation, Chugging, & Inertia

FIGURE D.11.1.2
S/RV WETWELL

FIGURE D.11.1.2 (Continued)
S/RV WETWELL PIPING

PLAN-TORUS SEGMENTS WITH TYPICAL LINE B

SECTION A-A

FIGURE D.11.1.2 FORCING FUNCTION ON SUBMERGED
S/RV DISCHARGE LINE (AT WATER LEVEL)

FIGURE: D.11.1.2 FORCING FUNCTION ON SUBMERGED S/RV
DISCHARGE LINE (DISCHARGE NOZZLE)

TABLE D.11.2.1

BWR: Current Design Loading on Safety/Relief Valve
Discharge Lines (Drywell Piping)

PLANT CONDITION	LOAD TYPE
Service Level A (6)	Pressure (625 design pressure) Gravity
Service Level B (6)	Pressure (625 design pressure) Gravity S/RV Discharge: Peak Pressure Loads (due to water clearing trans- ient) & Thrust Loads (Water and Pressure Wave) S/RV Inertia TAM: MS Line SAM (OBE): MS Line, BLD S/RV Anchor Motion Thermal Expansion (500°F design temp.)
Service Level C (6)	Pressure (625 design pressure) Gravity S/RV Discharge: Peak Pressure Loads (due to water clearing trans- ient) & Thrust Loads (Water and Pressure Wave) S/RV Inertia SSE: Inertia LOCA (SBA): Inertia
Service Level D (6)	Pressure (625 design pressure) Gravity S/RV Discharge: Peak Pressure Loads (due to water clearing trans- ient) & Thrust loads (Water and Pressure Wave) S/RV Inertia LOCA (DBA): Inertia

FIGURE D.11.2.2
S/RV DRYWELL PIPING
LINE A

FIGURE D.11.2.2 (Continued)
S/RV DRYWELL PIPING LINE B

FIGURE D.11.2.2 FORCING FUNCTION FIRST STRAIGHT
LINE SEGMENT PAST S/RV

FIGURE D.11.2.2 FORCING FUNCTION SECOND STRAIGHT
LINE SEGMENT PAST S/RV

Table D.12.1.1
BWR: Current Design Loading on the Control Rod
Drive Hydraulic System (CRDHS)
(Insert & Withdraw Lines)

PLANT CONDITION	LOAD TYPE
NORMAL	Pressure: 2000 psi design pressure Gravity Thermal Expansion: 150°F Insert Lines 546°F Withdraw Lines TAM: Hydraulic Control Unit, Control Rod Drive Housing flange.
UPSET	Pressure: 2000 psi operating pressure (8.) Gravity Thermal Expansion: 150°F Insert Lines 546°F Withdraw Lines Displacement of the Hydraulic Control Unit Floor due to Pool Swell. 1/2 SSE: Inertia (7.) Water Hammer: Scram Valve Opening
EMERGENCY	Pressure: 2000 psi operating pressure (8.) Gravity 1/2 SSE: Inertia (7.) Water Hammer: Scram Valve Opening
FAULTED	Pressure: 2000 psi operating pressure(8.) Gravity SSE: Inertia Inertia on the Hydraulic Control Unit Floor due to Pool Swell.

FIGURE D.12.1.2

Insert and Withdraw Lines - Inside Drywell

The 270° - 0° quadrant is shown; other quadrants are similar.

Table D.12.2.1
BWR: Current Design Loading on the Control Rod
Drive Hydraulic System (CRDHS)
(Scram Header and Volume Tank)

PLANT CONDITION	LOAD TYPE
NORMAL	Pressure: 1250 psi design pressure Gravity Thermal Expansion: 280°F
UPSET	Pressure: 1250 psi operating pressure (8.) Gravity Thermal Expansion: 280°F. Displacement of the Hydraulic Control Unit Floor due to Pool Swell 1/2 SSE: Inertia (7.)
EMERGENCY	Pressure: 1250 psi operating pressure(8.) Gravity 1/2 SSE: Inertia(7.)
FAULTED	Pressure: 1250 psi operating pressure(8.) Gravity SSE: Inertia

FIGURE D.12.2.2
SCRAM HEADERS

Table D.12.3.1
 BWR: Current Design Loading on the Control Rod
 Drive Hydraulic System (CRDHS)
 (Exhaust, Drive, Charging & Cooling Lines)

PLANT CONDITION	LOAD TYPE
NORMAL	Pressure: 2000 psi design pressure Gravity Thermal Expansion: 140°F TAM: Hydraulic Control Unit
UPSET	Pressure: 2000 psi operating pressure (8.) Gravity Thermal Expansion: 140°F TAM: Hydraulic Control Unit 1/2 SSE: Inertia (7.)
EMERGENCY	Pressure: 2000 psi operating pressure (8.) Gravity 1/2 SSE: Inertia (7.)
FAULTED	Pressure: 2000 psi operating pressure (8.) Gravity SSE: Inertia

FIGURE D.12.3.2
Exhaust, Charging and Cooling Lines

Notes: (Associated with Tables & Figures D.1.1 thru D.13.4.1)

1. The test condition has a significant design load since the hydrostatic test pressure exceeds the design pressure.
2. Those transient events with less than 2°F/second temperature change or a temperature range of less than 20°F are considered insignificant, and are not included. Transient events which have been conservatively enveloped are not analyzed.
3. Symbol for Plant/System Events:

N-2B

↑ Load Case Number - Indicates case within an event. The load case is defined as, that point in an event when the stress level is at a maximum value. Two load cases are separated by a steady state condition:

U: Up Load Case
D: Down Load Case
A: Part A of the Transit
B: Part B of the Transit
Blank: Event contains one load case only.

— Load Event Number

— Load Category: N-Normal or U-Upset.

4. Transient Events are those derived from 'Plant Events' tables A.1.1 & A.2.1.
5. Class II pipe for the Pressurizer Safety/Relief Lines, Pressurizer Spray Lines, RHR & Hot Leg SI Lines, Boron Injection, Safety Injection, Charging and Letdown Lines have the same loads as shown for the Class I pipe. The exception is the transients listed, which are not applicable for Class II pipe.
6. Service Levels A, B, C, & D correspond to the Normal, Upset, Emergency, & Faulted Plant Conditions.
7. The Operating Basis Earthquake in this case has been defined as 1/2 of Safe Shutdown Earthquake.

8. Operating Pressure has been conservatively defined as Design Pressure.

9. The following valve closures are considered for water hammer:

Regulating Valve
Quick Check Valve
Isolation Valve
Non-Return Check
Manual Bypass
Bypass Regulating
Bypass Quick Check

10. The following valve closures are considered for steam hammer:

Turbine Stop Valve
Regulating Valve

11. Abbreviations:

N/A - Not Applicable
OBE - Operating Basis Earthquake
SSE - Safe Shutdown Earthquake
TAM - Thermal Anchor Movement
SAM - Seismic Anchor Movement
DBA - Design Basis Accident
LOCA - Loss of Coolant Accident
RCL - Reactor Coolant Loop
RB - Reactor Building
PRESS.- Pressurizer
PSV - Pressurizer Safety Valve
BLD - Building
HX - Heat Exchanger

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