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BWR OWNERS' GROUP ASSESSMENT OF EMERGENCY CORE COOLING SYSTEM PRESSURIZATION IN BOILING WATER REACTORS

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EXECUTIVE SUMMARY

The Office for Analysis and Evaluation of Operational Data (AEOD) issued a case study report of operational events involving actual or potential overpressurizations of emergency core cooling systems (ECCS) in boiling water reactors (BWRs). The probability of an interfacing loss-of-coolant accident (LOCA) is the product of overpressurization event frequency and the probability of system rupture given the overpressurization. While the overpressurization event frequency is supported by actual industry occurrences, the probability of system boundary rupture stated in the AEOD report was "judgmentally assigned",* and believed by the participants in this study to be unrealistically high. This report was developed to present the results of a BWR Owners' Group (BWROG) assessment of the probability of ECCS failure due to overpressurization.

ECCS configurations for 19 domestic BWR plants were reviewed, from which ECCS piping system components subject to overpressurization were defined. Probabilistic methodology, developed by Lawrence Livermore National Laboratory (LLNL), was applied to the typical ECCS configuration to assess the probability of pipe rupture during overpressurization from the presence of latent circumferential weld defects. The rupture probabilities for other components, such as valves and heat exchangers, were approximated with reference to the piping probability. From these evaluations, the expected probability of an interfacing LOCA was assessed. Additionally, deterministic evaluations of safety margin during overpressurization were performed to show that these margins are greater than those specified by the ASME Code to provide assurance against gross rupture.

The BWR Owners' Group (BWROG) Committee has concluded that the conditional probability of BWR ECCS pressure boundary rupture during an overpressurization event is no greater than 3.0E-5 per event, two (2) orders of magnitude <u>less</u> than the stated "judgmental" AEOD probability. This report demonstrates that

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*Section 4.2 of Reference 1 has "judgmentally assigned" values to rupture probability due to uncertainties in undetected flaws, component leakage and waterhammer potential.

EXECUTIVE SUMMARY (Continued)

the resultant probability of an ECCS interfacing LOCA is 3.0E-7 per reactor year compared to the range of 1.0E-4 to 1.0E-5 as judged in the AEOD case study. Therefore, the expected probability of an interfacing LOCA is not significantly different from earlier industry assessments (References 4, 5, 6 and 7 of Reference 1), even though the frequency of overpressurization events may be greater than previously assessed.

This study was funded by the BWROG. A list of participating utilities in this activity is provided in Appendix E.

October 1987 Appendix R Audit

Open Items Not Requiring NRR Review

Fire Damper Operability (Unresolved Issue 84-40-01, 84-19-01 discussed on page 4 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30.)

A fire damper program is being formulated to evaluate existing test data and damper closure with air flow data and to address fire brigade and training procedures to provide reasonable assurance that the fire dampers will satisfactorily perform their design function. We plan to meet with Region I to discuss/formulate such a program by April 1988.

Incorporation of NRC Comments on Procedures (Page 11 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30

Procedure SE-10, Plant Shutdown from the Alternative Shutdown Panel, is currently being revised to reflect changes caused by the completion of Appendix R modifications. During this revision, operator, training, and NRC comments were reviewed and incorporated into the procedure. In addressing NRC comments, sign-off spaces have been added where needed, and the monitoring of the reactor cooldown rate has been enhanced.

<u>Accessibility of HPCI Inboard Steam Isolation Valve Panel</u>

(Page 11 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30)

The HPCI Inboard Steam Isolation Valve panel was originally provided with slotted screws which required tools for access. The slotted screws were changed to thumbscrews to allow an operator to access the panel without the use of tools. To address the NRC concern of overtightening, flat washers were added to compliment the thumbscrews. The washer addition will provide a smooth contact surface and enhance the operator's ability to loosen a tight thumbscrew.

<u>Fuse Replacement Controls</u> (Page 13 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30)

Administrative controls for fuse replacement are being actively reviewed. A modification has been initiated to generate a controlling document for fuse replacements. For the interim, a guideline document is being added to the operator's handbook to assist in the current practice of replacement in-kind.

1. INTRODUCTION

1.1 BACKGROUND

The Office for Analysis and Evaluation of Operational Data (AEOD) of the Nuclear Regulatory Commission (NRC) issued a Case Study Report AEOD/C502, "Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors", dated September 1985 (Reference 1). This report summarized the AEOD analysis of operational events involving actual or potential overpressurization of an emergency core cooling system (ECCS) in boiling water reactors (BWR) since 1975. The operating BWRs reviewed were product lines BWR/2 through BWR/6. Reference 1 focused on overpressurization or potential overpressurization events that occurred in BWRs due to testable isolation check valve failures in all ECCSs. It concluded that these overpressurization events indicate the likelihood of an interfacing loss-of-coolant accident (LOCA) to be higher, by two to several orders of magnitude, than had been previously assessed. The BWR Owners' Group (BWROG) authorized a committee activity to evaluate ECCS overpressurization and assess the capability of the ECCS components to withstand overpressurization. This BWROG report summarizes that assessment effort.

1.2 AEOD REPORT CONCLUSIONS

The AEOD review identified eight (8) events involving the failure of a testable isolation check valve, and expressed concern that these operational events should be considered as a precursor to an interfacing LOCA involving the reactor coolant system and an ECCS. The AEOD report states the event frequency to be 1.0E-2* per reactor year combined with a "judgmentally assigned" probability of ECCS boundary rupture of 1.0E-2 to 1.0E-3 per overpressurization event, resulting in an interfacing LOCA probability of 1.0E-4

*The BWROG Committee believes this event frequency to be overly conservative based on a continued operating data base without additional occurrences and an increased industry awareness of overpressurization events and the recognition of the need to reduce overpressurization event frequency.

to 1.0E-5 per reactor-year. This AEOD "judgmental" frequency is two (2) to three (3) orders of magnitude higher than the frequencies previously assessed by the industry.

1.3 BWR OWNERS' GROUP SCOPE OF EVALUATION

The objective of the BWR Owner's Group evaluation was to assess the failure potential of ECCS systems, piping and components when subjected to overpressurization. The objective did not include evaluation of the consequences of discharge of fluid from relief valves or leakage from pipe cracks, gaskets or flanged joints on the basis that these discharges, in most events, can be compensated for by increased feedwater system output due to the low fluid volume discharge rate from the sources relative to feedwater capacity as demonstrated by the events discussed in Reference 1. Additionally, it is judged that there is a very high probability that leakage from these sources can be isolated.

Stresses in the low-pressure side of ECCS piping and components were evaluated based on the system information received from the participating utilities and GE in-house information. Safety margins were evaluated and compared with ASME Code-specified values. Quantitative evaluation of rupture probability at a circumferential butt weld was evaluated and the rupture probabilities for other components were qualitatively evaluated. The system rupture probability was then assessed.

1.4 BWR OWNERS' GROUP COMMITTEE ACTIVITIES/OBJECTIVES/APPROACH

The BWR Owners' Group objective was to evaluate ECCS overpressurization and assess the BWR ECCS capability to withstand overpressurization without rupture. The frequency of ECCS overpressurization events in the BWR is well documented in the AEOD report. The BWROG response to the AEOD case study has focused principally on assessing the realistic probability of low design-pressure system pressure boundary rupture given an overpressurization occurrence.

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The overall ECCS rupture probability during the overpressurization event is the sum of the rupture probabilities of piping and the associated components such as valves and heat exchangers. It was judged that the most significant mode of rupture for the ECCS piping is that due to the presence of latent weld defects at the circumferential butt welds. Rupture probabilities for this mode were determined based on methods developed by the Lawrence Livermore National Laboratory (LLNL) and previously accepted by the NRC in the studies of pressurized water reactors (PWR). The rupture probabilities for other components such as valves and heat exchangers were approximated with reference to the preceding probability of rupture for the circumferential weld.

2. CONCLUSIONS AND RECOMMENDATIONS

2.1 CONCLUSIONS

Industry-accepted methodology from Reference 10 was applied to evaluate the probability of BWR ECCS failure due to overpressurization. The evaluation has led to the following conclusions:

- Deterministic evaluations of safety margins during overpressurization were performed to show that these margins are greater than those specified by the ASME Code to provide assurance against gross failure.

- The realistic conditional probability for BWR ECCS pressure boundary rupture during an overpressurization event has been estimated to be no greater than 3.0E-5 per event. This probability is two to three orders of magnitude <u>less</u> than the stated "judgmental" AEOD probability of 1.0E-2 to 1.0E-3.

- Assuming the AEOD event frequency, the realistic frequency of an ECCS interfacing LOCA caused by system overpressurization is 3.0E-7 per reactor-year compared to 1.0E-4 to 1.0E-5 claimed in the AEOD case study.

- The most probable result of overpressurization as indicated by events reported in Reference 1 and the evaluations in this report would be the discharge of fluid from relief valves and possibly leakage from bolted joints, and smoke generated by oxidizing paint on piping and equipment. The consequences of such discharges and leakage are expected to be minimal and will most likely result in early operator termination of an overpressurization condition due to activation of high area temperature alarms and/or visual observation of leakage by plant personnel as reported in the Reference 1 report. Activation of high line pressure alarms and smoke alarms and plant personnel observation of smoke are also likely to result in early operator termination of the overpressure condition.

The frequency of overpressurization events documented in the AEOD report is higher than estimated in previous studies. However, the probability of an interfacing LOCA determined from the BWROG evaluation does not justify these BWR operating events being classified as having "significant safety implications" as stated in the AEOD case study.

The AEOD recognizes that none of the documented overpressurization events has led to significant damage of the low design-pressure system piping, pumps or valves. However, the report cautions that future events may lead to failures caused by pre-existing flaws. The analysis performed as part of the BWROG activity confirmed that low design-pressure piping and system component failure due to overpressurization by full reactor pressure should not occur because of the design margins that protect against such failure. Furthermore, flaw analysis indicates that the limiting flaw length required to promote pressure boundary rupture would require a through-wall crack of nearly three (3) inches. Field experience has shown that the probability of incurring a crack of this size and having the crack go undetected is negligible. This provides further assurance that the low design-pressure BWR ECCS and RCIC System integrity would be maintained should overpressurization events occur.

2.2 RECOMMENDATIONS

a. The current frequency of overpressurization events is unnecessarily high. Corrective action should be implemented to reduce their occurrence. Individual utilities should consider specific actions to reduce overpressurization event frequency, including making plant operators more aware of these potential events and their causes as well as evaluating reducing isolation valve testing frequencies.

b. Due to the BWROG Committee's assessment of the low probability of low-pressure boundary rupture caused by overpressurization, the issue should not be classified as having "significant safety implications" and should be addressed accordingly by Industry and the NRC. The interfacing LOCA probabilities stated in this report are consistent with previous industry studies and should be considered as a more realistic assessment of BWR ECCS capability during overpressurization rather than those stated in Reference 1.

3. AEOD REPORT ASSESSMENT

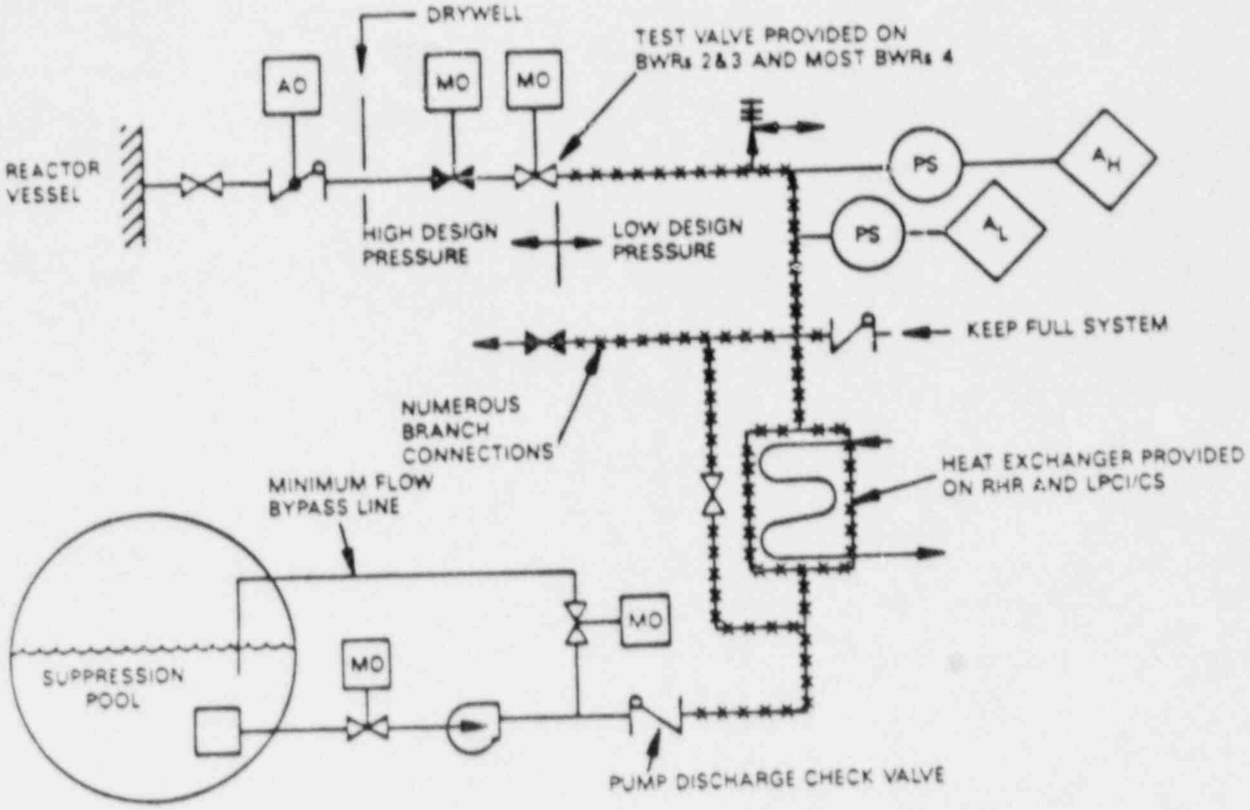
3.1 ECCS PIPING AND COMPONENTS AFFECTED BY OVERPRESSURIZATION

This section summarizes the ECCS and Reactor Core Isolation Cooling (RCIC) System piping and components subject to overpressurization by the types of events described in the Reference 1 AEOD report.

The systems potentially subject to overpressurization are: Core Spray (CS) (BWR/2 through BWR/4); Low Pressure Core Spray (LPCS) (BWR/5 and BWR/6); Residual Heat Removal (RHR) (all BWR/4 through BWR/6 and most BWR/3); Low Pressure Coolant Injection and Containment Cooling (LPCI/CC) (some BWR/3); High Pressure Coolant Injection (HPCI) (BWR/3 and BWR/4); and RCIC (all BWR/4 through BWR/6 and most BWR/3).

As illustrated in Figure 3-1, the RHR, LPCI/CC, CS and LPCS piping sections and components subject to overpressurization include the piping sections and components located downstream of the check valve(s) on the systems' main pump(s) discharge. The piping and components upstream of the main pump discharge check valve are not subject to overpressurization because, during normal power operation, the systems are required to be aligned with the suction valve and flow path from the suppression pool open. This alignment provides a flow path to the suppression pool that has a large cross sectional flow area upstream of the check valve. Therefore, the piping and components upstream of the check valve cannot be overpressurized.

There is a high level of assurance that the check valve on the pump discharge is closed and has a low leak rate due to the methods utilized to maintain the discharge line full of water. Some plants utilize the condensate transfer system to maintain the discharge line full. On these plants, excessive leakage of the check valve cannot be tolerated for an extended time period because of the excessive processing demand placed on the Radwaste System. The demand results from pumping the inleakage out of the suppression pool in order to maintain the suppression pool water level within acceptable limits. On other plants, a low flow capacity "keep full" pump is utilized to



—×—×— PIPING SUBJECT TO POTENTIAL OVERPRESSURIZATION

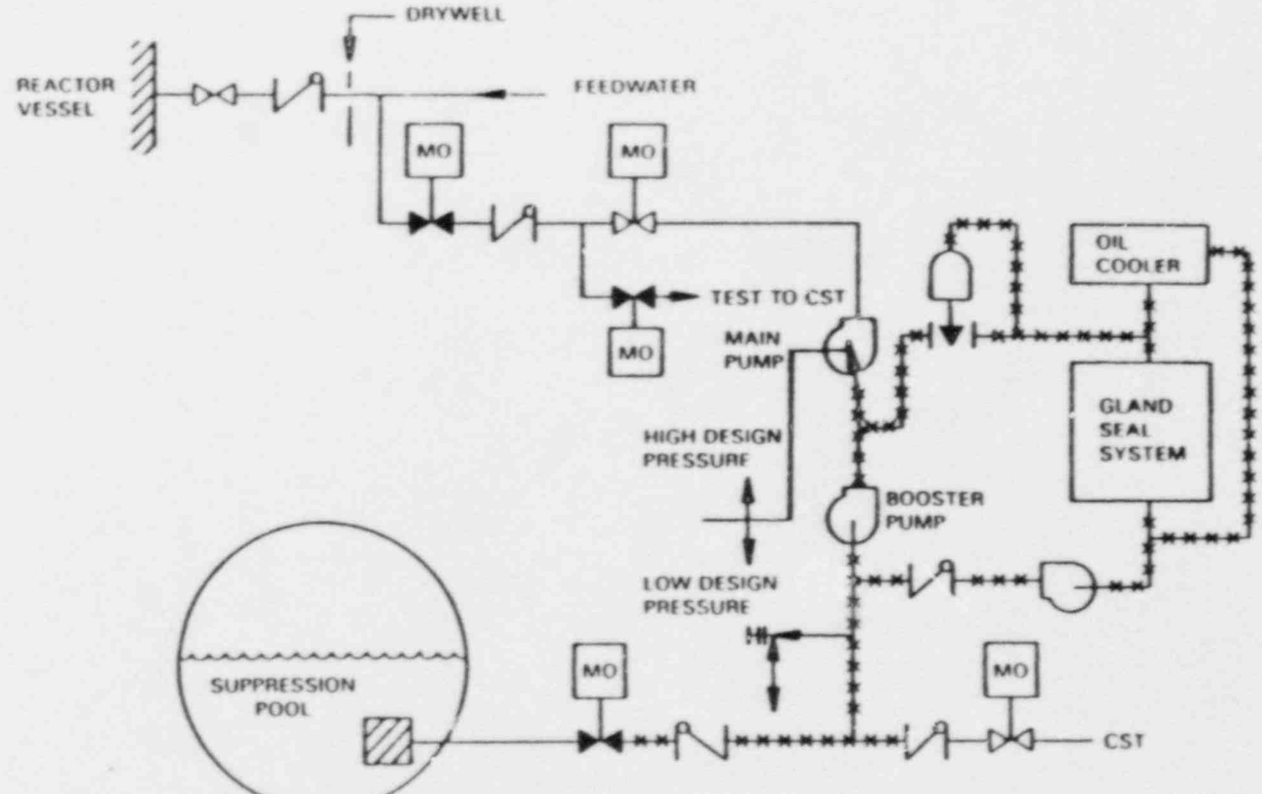
——— PIPING NOT SUBJECT TO OVERPRESSURIZATION

- OVERPRESSURIZATION POTENTIAL IS LIMITED TO PIPING/COMPONENTS DOWNSTREAM OF PUMP DISCHARGE CHECK VALVE
- PIPING/COMPONENTS UPSTREAM OF PUMP DISCHARGE CHECK VALVE CANNOT BE OVERPRESSURIZED BECAUSE OF THE OPEN FLOW PATH TO THE SUPPRESSION POOL (i.e., SUCTION LINE FROM SUPPRESSION POOL AND MINIMUM FLOW BYPASS LINE)
- PUMP DISCHARGE CHECK VALVE LEAKAGE RATE IS ASSURED TO BE LOW BECAUSE:
  - ON PLANTS THAT UTILIZE THE CONDENSATE TRANSFER SYSTEM TO KEEP THE DISCHARGE LINE FULL OF WATER EXCESSIVE CHECK VALVE LEAKAGE IS NOT TOLERABLE FROM PLANT OPERATIONAL CONSIDERATIONS BECAUSE OF THE RESULTING FILLING OF THE SUPPRESSION POOL THAT REQUIRES EXCESSIVE DISCHARGE OF WATER TO THE RADWASTE SYSTEM
  - ON PLANTS THAT UTILIZE LOW FLOW CAPACITY KEEP FULL PUMPS TO MAINTAIN THE DISCHARGE LINE FULL THE FILL PUMP CANNOT MAINTAIN PRESSURE ABOVE LOW PRESSURE ALARM SETPOINT IF THE CHECK VALVE LEAKAGE IS EXCESSIVE

Figure 3-1. Typical Configuration of RHR, CS and LPCI/CS Piping Sections Subject to Potential Overpressurization maintain the lines full. If the check valve leakage is excessive, the "keep full" pump cannot maintain the discharge line pressure above the low pressure alarm setpoint. Initiation of the alarm would result in operator action to reduce the check valve leakage in order to obtain an acceptable discharge line pressure.

As illustrated in Figures 3-2 and 3-3, the HPCI and RCIC Systems' piping and components subject to overpressurization include the piping sections and components between the main pump suction inlet and the normally closed valve in the suction line from the suppression pool and the suction line check valve from the condensate storage tank.

The High Pressure Core Spray (HPCS) System (BWR/5 and BWR/6) low design pressure suction piping and components are effectively prevented from being overpressurized by the check valve on the HPCS pump discharge. There is high level of assurance that the check valve on the pump discharge is closed and has a low leak rate. This is because, normally, the leakage rate of the HPCS injection valve is much less than that of (1) the check valve on the pump discharge and (2) the valves in the HPCS suction lines. Thus, the line forward of the check valve would be maintained full and pressurized by a low flow capacity "keep full" pump as illustrated in Figure 3-4. If the check valve leakage is excessive, the "keep full" pump cannot maintain the discharge line pressure above the low discharge line pressure alarm setpoint. If the alarm is initiated, the operator would be required to take corrective action to reduce the check valve leakage in order to clear the low pressure alarm. If the pump's discharge check valve failed open and the leakage rate of the valves in the HPCS suction was less than that of the system injection valves, leakage back from the reactor would prevent the low discharge line pressure alarm and would result in a high suction pressure alarm. Operator action, to terminate the high suction pressure alarm by opening a vent to depressurize the line, would result in initiation of the low discharge line pressure alarm, thus alerting the operator to the check valve failed condition and the need for corrective action. The relief valve in the HPCS suction line would prevent overpressurization of the system suction piping in this event. Therefore, there is high level of assurance that the discharge line check



-x-x-x- PIPING SUBJECT TO OVERPRESSURIZATION

——— PIPING NOT SUBJECT TO OVERPRESSURIZATION

• POTENTIAL OVERPRESSURIZATION EXTENDS BACK TO SUPPRESSION POOL SUCTION MOTOR OPERATED (MO) VALVE AND CONDENSATE STORAGE TANK (CST) CHECK VALVE INCLUDING GLAND SEAL SYSTEM AND LUBE OIL COOLER

Figure 3-2. Typical HPCI Configuration - Piping Sections Subject to Potential Overpressurization

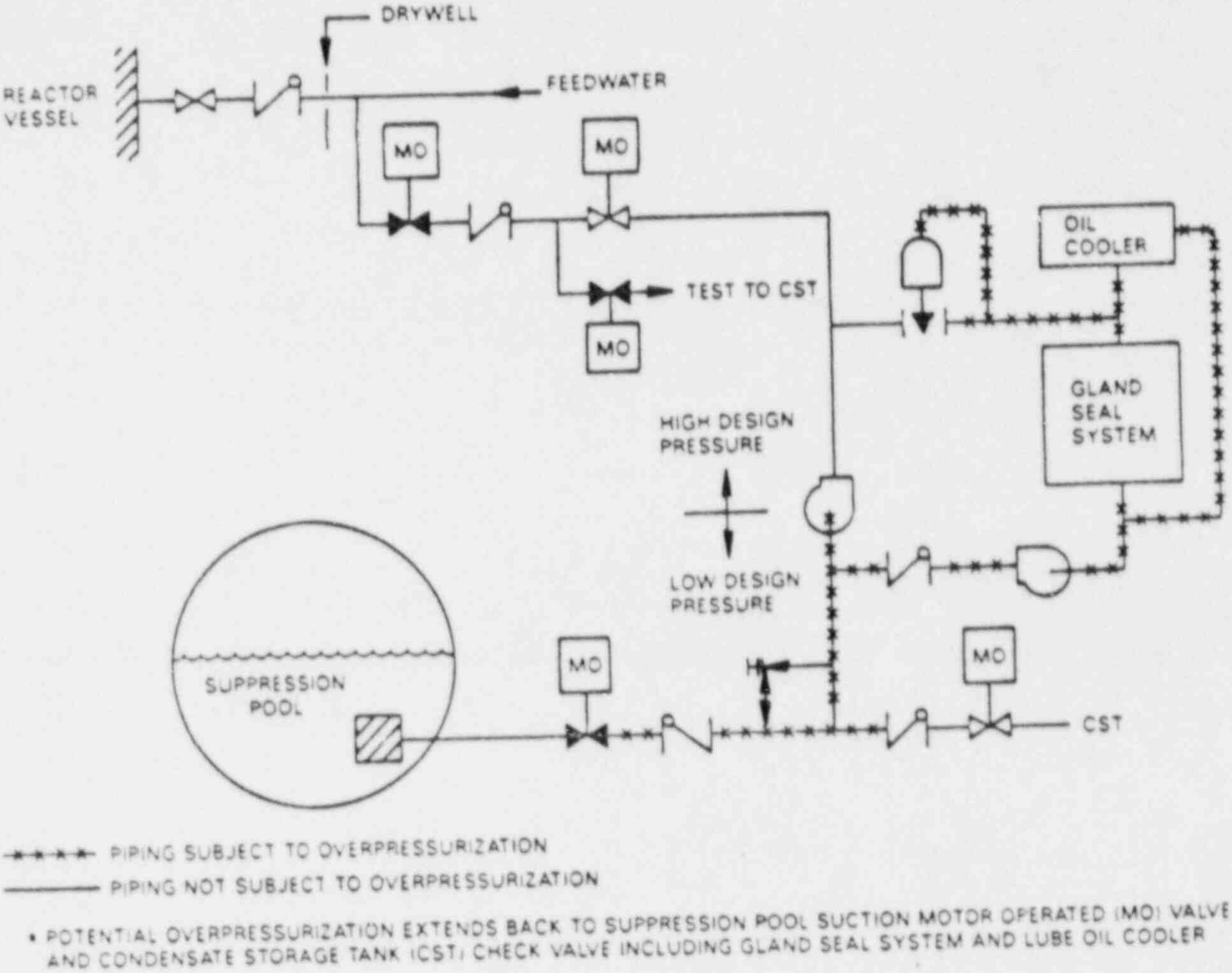
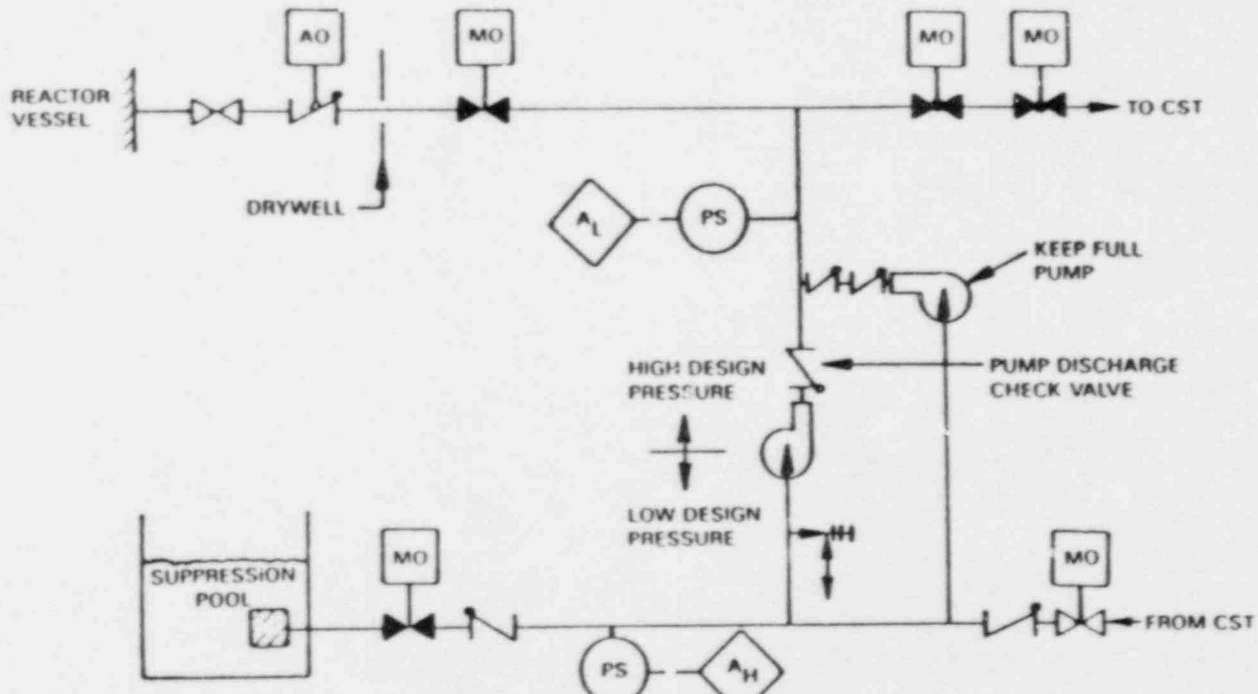


Figure 3-3. Typical RCIC Configuration - Piping Sections Subject to Potential Overpressurization



HPCS OVERPRESSURIZATION IS NOT CREDIBLE

<u>BASIS</u>

- THE HPCS PUMP DISCHARGE CHECK VALVE PREVENTS OVERPRESSURIZATION OF HPCS SUCTION PIPING
- THERE IS A HIGH LEVEL OF ASSURANCE THE HPCS PUMP CHECK VALVE IS CLOSED AND WELL SEATED. THIS IS BECAUSE THE KEEP FULL PUMP HAS A LOW PUMPING CAPACITY AND THE PUMP DISCHARGE CHECK VALVE LEAKAGE RATE MUST BE LOW TO AVOID A LOW DISCHARGE LINE PRESSURE ALARM
- THE RELIEF VALVE ON THE SUCTION PIPING PREVENTS OVERPRESSURIZATION DUE TO MINOR LEAKAGE PAST THE CHECK VALVE

Figure 3-4. Typical HPCS Configuration

valve is closed and is well seated. The probability of the injection and suction valves leaking at a rate that would result in preventing the low discharge line pressure and high suction pressure alarms coincident with the discharge line check valve failing open is judged not to be significant.

A more detailed discussion of the system's configuration and design data is presented in Appendix A.

3.2 PROBABILITY OF PRESSURE BOUNDARY RUPTURE DURING OVERPRESSURIZATION

Based on the ECCS piping configurations defined in Section 3.1, the system piping, valves and heat exchanger components were evaluated for potential rupture during an overpressurization event.

In evaluating the piping integrity during the overpressurization event, the following failure modes were considered: (1) burst due to high hoop stress; (2) rupture due to latent axial defects; and (3) rupture due to latent weld defects at circumferential butt welds. Each of these modes was evaluated as follows: (1) the hoop stress from overpressurization was calculated to compare with a conservative value of pipe burst hoop stress which was based on General Electric test data for burst hoop stress and a review of available technical literature; (2) through-wall flaw lengths that the ECCS low pressure system pipes can tolerate during the overpressurization event were determined (the purpose of this evaluation was to demonstrate that these limiting flaw lengths are large compared to the flaw lengths that are normally detected by normal in-service inspections); and (3) the probability of a double-ended pipe break (DEPB) during overpressurization resulting from latent defects at circumferential welds was calculated. The evaluations in (1) and (2) above are deterministic and, therefore, were not directly factored into the probabilistic evaluation performed for (3). A piping reliability model developed by Lawrence Livermore National Laboratory (LLNL), with appropriate modifications incorporated by General Electric for BWR applications, was used to calculate the probability of a DEPB at a circumferential weld.

Two types of deterministic evaluations were performed for the valves and RHR heat exchangers, which represent somewhat complex structures compared to the piping. The first evaluation consisted of examining the body or shell thickness and the hydrotest pressures to demonstrate that the rupture probability of a valve body or a RHR heat exchanger shell would be less than that of the connected piping. The second evaluation considered the bolted joints.

Potential dynamic loads such as those resulting from earthquakes and safety/relief valve discharges were not included in the analysis because it was concluded that the likelihood of their occurrence simultaneous with an overpressurization event is extremely small. Other potential dynamic loads during the overpressurization event, such as waterhammer caused by reactor water filling a partially voided ECCS line downstream of check valve, were also not included in the scope of this evaluation. The probability of partially voided lines is extremely low due to the "keep-full" systems. Therefore, it was concluded that there would be no significant dynamic loads being applied to the ECCS system during an overpressurization event.

3.2.1 Piping Integrity Evaluation

The pipe size, schedule and the material information defining the various BWR ECCS and RCIC piping systems surveyed were reviewed. From this information, the following general conclusions were drawn:

a. The ECCS piping is of seamless construction and the material is typically SA 106 Gr. B carbon steel.

b. The largest piping diameters in the surveyed core cooling systems range as follows:

Core Spray:	16-inch
RCIC:	6-inch
HPCI:	16-inch
RHR:	24-inch

c. The Code classification is generally Section III, Class 2 (Reference 2) or ANSI B31.1 (Reference 3).

The first step in assessing the rupture potential is to calculate the axial and circumferential stresses in these piping systems. Table 3-1 shows the calculated circumferential stresses for typical ECCS pipes when subjected to nominal reactor pressure of 1050 psi. A corrosion allowance of 0.08 inch was used. A review of the circumferential stresses in Table 3-1 shows that they range from a low value of 16.3 ksi for the 6-inch pipe in the RCIC System to a high value of 34.5 ksi for the 20-inch standard schedule RHR pipe.

The allowable stresses for various service conditions in the ASME Code are expressed as a constant times the Code specified allowable stress, denoted by symbol S. For Class 2 and 3 piping, it is the lesser of five-eighths of the yield stress or one-quarter of the ultimate stress. The allowable stress values, S, given in the ASME Code are essentially identical to those given in the older piping codes such as ANSI B31.1 used in the design of earlier BWR plants. The value of S for SA 106 Gr. B material is specified as 15 ksi for temperatures up to 600°F. This is 1/4 of the Code specified minimum ultimate stress of 60 ksi.

The Level D or faulted condition stress limits are relevant in this case, since these limits, while permitting some gross general deformation, still assure pressure-retaining capability of piping components. This is consistent with the requirement for pressure integrity of the ECCS piping during the overpressurization event. Therefore, the calculated stresses were compared to the allowable stresses for the faulted condition.

The Class 2 and 3 pipes are sized such that the hoop stress at design pressure is less than the ASME Code allowable stress, S. Since a peak pressure of two times the design pressure is permitted during Level D (faulted) conditions, the allowable circumferential stress during Level D conditions is 2 S. For SA 106 Gr. B, this allowable stress level is 30 ksi. An examination of the calculated hoop stresses in Table 3-1 shows that, except for the 20-inch pipe, all other hoop stresses are less than 30 ksi*. Therefore, it is

*Even though the calculated hoop stress of the 20-inch pipe exceeds the Level D allowable, further evaluation of limiting flaw size and burst data (Subsection 3.2.1.2) shows that a sizeable crack length would be required to cause pipe rupture.

Table 3-1

CALCULATED HOOP STRESSES DURING OVERPRESSURIZATION IN REPRESENTATIVE ECCS PIPING SIZES

PIPE SIZE (in.)	SCHEDULE	NOMINAL THICKNESS, T (in.)	HOOP STRESS* AT 1050 psi PRESSURE	LOWER BOUND BURST HOOP STRESS (ksi)	BURST** MARGIN
6	STD	0.28	16.3	54.0	3.31
14	STD	0.375	23.9	54.0	2.26
16	STD	0.375	27.4	54.0	1.97
20	STD	0.375	34.5	54.0	1.56
24	XS	0.500	28.9	54.0	1.87

*Thickness used for hoop stress calculation is (T - 0.08), where 0.08 inch is the corrosion allowance.

**Ratio of Lower Bound Burst Hoop Stress to Hoop Stress at Reactor Pressure (1050 psi).

550°F code minimum values for SA 106B

S_u = 60 ksi

S_y = 27 ksi

concluded that, generally, the hoop stresses in the ECCS piping during the overpressurization event will be less than the ASME Code specified limit for Level D conditions.

3.2.1.1 Hoop Stress Burst Margin

The safety margin relative to burst type of failure in ECCS piping system during an overpressurization event is the ratio of the expected hoop stress at burst and the calculated hoop stresses at reactor pressure (Table 3-1). The hoop stresses due to any thermal gradients are not included in this evaluation because such stresses are displacement controlled and, thus, do not directly contribute to burst failure or rupture.

Based on extensive sets of test data, Rodabaugh (Reference 8) noted that for seamless pipes the hoop stress at burst is essentially equal to the ultimate stress of the material. Burst test data reported by General Electric (Reference 9) specifically on seamless 106 Gr. B pipes were reviewed for this case. The pipe diameters in these tests ranged from 4 to 12 inches. The yield and ultimate strengths were also determined for each pipe tested. Figure 3-5 shows the hoop stress at burst as a fraction of the measured ultimate stress for that pipe material plotted versus the ultimate stress. It is seen from Figure 3-5 that the average burst hoop stress, is equal to approximately 90% of the ultimate stress. This conclusion is considered independent of the pipe size because the burst hoop stress, rather than the burst pressure, was used in the evaluation. The ASME Code-specified minimum value of ultimate stress for SA 106 Gr. B steel is 60 ksi to a temperature of 600°F. Therefore, the expected value of burst hoop stress is (60 x 0.9) or 54 ksi. Table 3-1 shows that the burst hoop stress margin ranges from 3.31 (6-inch) to 1.56 (20-inch). Even the minimum 1.56 margin is greater than the faulted or Level D safety margin of 1.4 of the ASME Code.

3.2.1.2 Limiting Axial Flaw Lengths in Piping During ECCS Overpressurization

A qualitative measure of the assurance of pressure integrity of ECCS piping during an overpressurization event is the length of the axial flaw (latent defect) that can be tolerated without rupture. Since SA 106 Gr. B

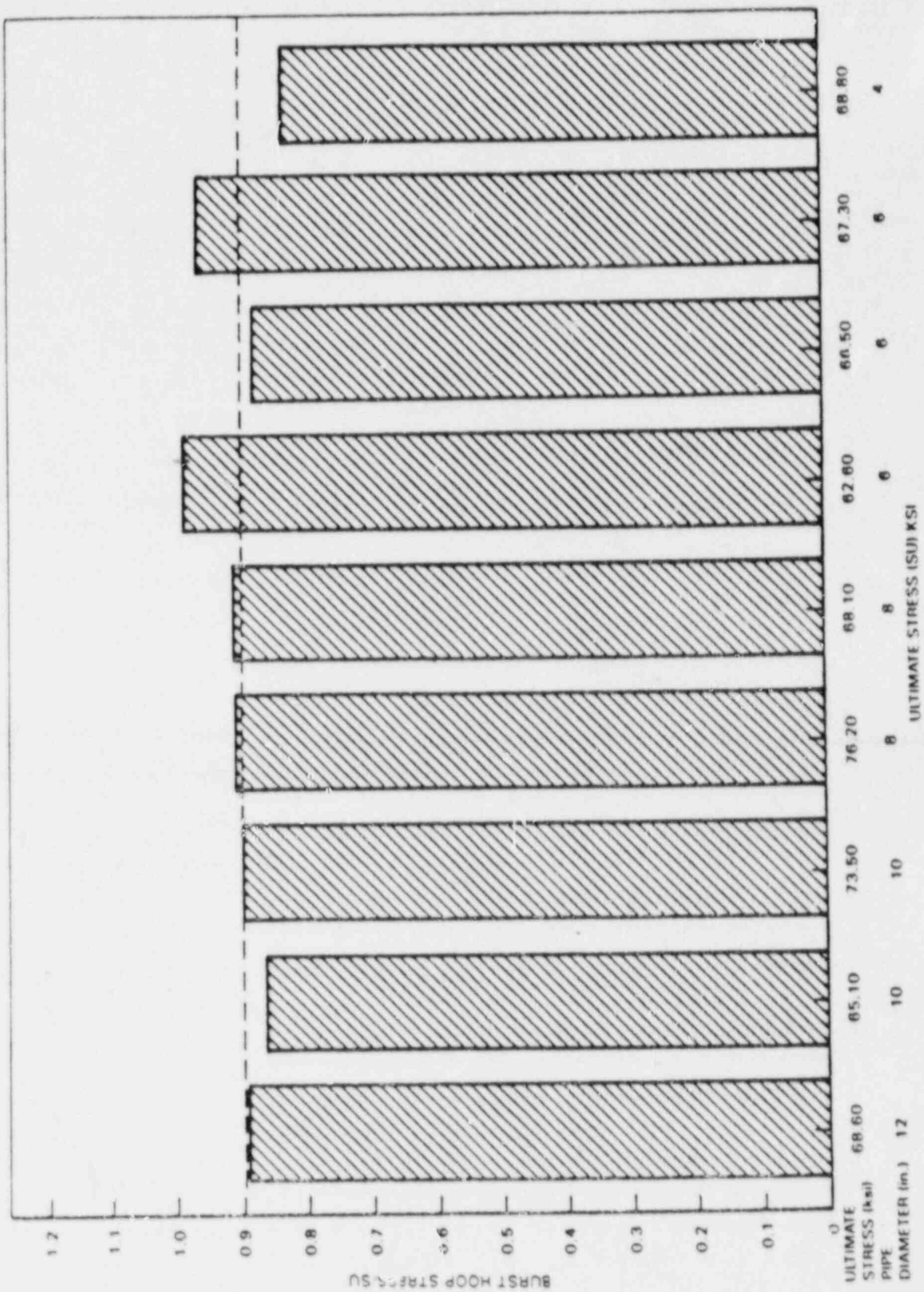


Figure 3-5. Burst Test Data from Reynolds (GEAP-5620). Pipe Diameters from 4 to 12 Inches; SA 106B Carbon Steel

carbon steel is expected to behave in essentially a ductile manner in the range of temperatures expected during overpressurization, either the elastic-plastic fracture mechanics (EPFM) analysis or the limit load approach is appropriate in such an evaluation. In comparing the two methods, the use of EPFM analysis requires information on the material stress-strain curve and material toughness in the form of a J-Resistance curve, while the only material parameter required in the limit load approach is the flow stress. Reference 4 indicates that both the EPFM analysis and the limit load predictions are in excellent agreement with the experimentally determined instability pressures of SA 106 Gr. B carbon steel pipes. The temperatures in these tests ranged from 538°F to 675°F, and the average value of the flow stress used was approximately 47 ksi. This empirical value of flow stress is applicable to the evaluation of axial flaws. A different value of flow stress ($2\ S_m$ or 36 ksi; page 3-19) is used for the evaluation of circumferential flaws. Since the flow stress values are empirical quantities backed out from corresponding test data, the use of different values of flow stresses for axial and circumferential flaw evaluations is justified.

The limit load approach is also the basis for the recently proposed ASME Code procedures for the evaluation of axial cracks in both the austenitic (Reference 5) and ferritic piping (Reference 6). Therefore, a limit load approach, with a conservative value of flow stress of 47 ksi, was used in the following evaluation to determine the largest tolerable axial crack length.

An empirical formula developed by Eiber, et al (Reference 7) relates the hoop stress, σ_h, at failure for pipes with axial through-wall flaws, to the flaw parameters as follows:

$$\sigma_h = \frac{\sigma_f}{M} \tag{3-1}$$

where σ_f = flow stress and, M is a curvature correction factor given by

$$M = [1 + 1.61\ \ell^2/4\ rt]^{\frac{1}{2}} \tag{3-2}$$

where:

ℓ = axial crack length

r = radius of the pipe

t = nominal thickness

An algebraic manipulation of Equations 3-1 and 3-2 yields the following equation for allowable axial crack length:

$$\ell = 2.48\,\sqrt{rt}\,\left[\left(\frac{\sigma_f}{\sigma_h}\right)^2 - 1\right]^{1/2} \qquad (3\text{-}3)$$

Table 3-2 shows the maximum tolerable crack length during the overpressurization event for various representative pipe sizes in ECCSs ranging from 2.4 inch to 5.8 inches. This indicates that large through-wall axial cracks would have to be present to cause piping rupture. Since a through-wall crack of such length would likely be detected and repaired, it is concluded that the probability of rupture of ECCS piping from unstable growth of latent axial defects during overpressurization is negligible.

Further evaluation of ECCS failure during overpressurization considering latent defects at circumferential welds is presented in Subsection 3.2.1.3.

3.2.1.3 Weld Evaluation of Probability of Pipe Rupture at a Circumferential Butt

This section evaluates the probability of failure during overpressurization due to a latent defect in a circumferential weld in the ECCS piping. Since the ECCS piping is seamless, the most likely locations where a latent defect may exist would be the circumferential butt welds. Probabilistic methodology, developed by Lawrence Livermore National Laboratory (LLNL), was used in this evaluation.

Table 3-2

LIMITING AXIAL THROUGH-WALL
FLAW LENGTHS IN BWR ECCS PIPING
AT REACTOR PRESSURE

PIPE SIZE (in.)	SCHEDULE	HOOP STRESS AT 1050 PSI (ksi)	LIMITING CRACK LENGTH (in.)
6	STD	16.3	3.3
14	STD	23.9	3.7
16	STD	27.4	3.2
20	STD	34.5	2.4
24	XS	28.9	5.8

As a part of the effort for the resolution of Unresolved Safety Issue A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant Systems", LLNL developed a probabilistic fracture mechanics methodology for the assessment of double-ended pipe break (DEPB) probability resulting from both direct and indirect causes (Reference 10). The DEPB probability assessment from direct causes considers the growth of as-fabricated surface flaws at welded joints, taking into account loads on the piping due to normal operating conditions and seismic events. Other factors, such as the capability to detect cracks by nondestructive examination and the capability to detect pipe leaks, are also modeled. Flaws which become through-wall but do not result in rupture may produce a detectable leak when the calculated leak rate is above the detection threshold. The ratio of the calculated DEPB probability to the detectable leak probability is a measure of the leak-before-break probability. LLNL has developed a computer code (PRAISE) which incorporates this methodology.

Even though the LLNL investigations were limited to pressurized water reactor (PWR) coolant piping, the techniques are sufficiently general for adaptation to all light water reactor piping systems. General Electric has modified the PRAISE code for BWR applications and has included a more general limit load-based failure criterion. Brief descriptions of the LLNL piping reliability model and the General Electric modifications are given in Appendix B.

The pipe rupture probabilities were calculated for a typical girth butt weld in the low pressure ECCS piping segments that would be pressurized during an overpressurization event. It was conservatively assumed that the probability of existence of a fabrication defect at a weld is 1.0. Further, no credit was taken for any preservice or inservice inspection.

The axial and bending stresses considered in the evaluation were those due to pressure, weight and thermal expansion. The axial membrane stress, which is essentially due to the reactor pressure of 1050 psi, is given in Table 3-3 for various pipe sizes. The bending stress, due to thermal expansion, is imposed on the subject piping as the system heats up during overpressurization. While an exact value of the weight combined with the

Table 3-3

AXIAL STRESSES USED IN PROBABILITY EVALUATION

Pipe Diameter (in.)	Stresses (ksi)	
	Membrane	Bending*
6	7.9	7.0
14	11.7	7.0
16	13.5	7.0
20	17.0	7.0
24	14.5	7.0

*Assumed (includes weight + thermal expansion) thermal expansion stress at a particular weld can be calculated from the information provided in piping system stress reports, a representative bounding value of 7 ksi was used in this evaluation. The PRAISE code evaluations need only nominal stresses (i.e., without any stress intensification factors normally used in code compliance evaluations). The calculated value of weight plus thermal expansion stresses at some locations in the ECCS piping systems may exceed the assumed representative upper bound value of 7 ksi. Nevertheless, it is judged that the following conservative assumptions still assure a bounding system rupture probability at circumferential welds:

a. The thermal expansion stresses are displacement controlled and, thus, are classified as secondary stresses in the ASME Code. Since only the primary (i.e., load-controlled) stresses such as pressure and weight stresses can cause pipe rupture, the inclusion of thermal expansion stresses in the failure criteria of the PRAISE code is conservative.

b. In calculating the system rupture probability, some bounding stress level is assumed at all of the welds in the piping system. It is judged that the increase in the calculated probability at some welds due to higher than 7 ksi bending stress will be more than offset by the lower calculated probability at a majority of the system welds where the stress is less than 7 ksi. In other words, the calculated system rupture probability, assuming all of the welds to be stressed at 7 ksi, is expected to bound the calculated probability in which actual bending stress levels are used.

Table 3-3 summarizes the stress magnitudes used for various pipe sizes in the calculation of circumferential weld rupture probabilities.

As described in Appendix B, the failure criterion used in the probability evaluation is based on the limit load approach. The material flow stress is a key parameter in this approach. Based on Reference 6, flow stress was conservatively assumed as 2 S_m, where S_m is the material design stress intensity specified in the ASME Code. Thus, the flow stress value of 36 ksi, based on the S_m value for SA 106 Gr. B at reactor temperature, was [illegible] the evaluation.

Table 3-4 shows the calculated rupture probabilities for each pipe size considered. The probabilities range from 5.4E-9 for the 24-inch pipe to 9.8E-8 or approximately 1.0E-7 for the 6-inch pipe. It is seen that the rupture probability is highest for the 6-inch pipe, although it has the lowest axial membrane stress. This is related to the differences in aspect ratios (half crack length/depth; β) of critical crack sizes (crack sizes for which failure is predicted) as a function of pipe diameter. The aspect ratios of critical cracks in larger diameter pipes are larger compared to those of the smaller diameter pipes. Since the rupture probabilities in the LLNL model are related to the inverse of the exponent of β (see Equation B-2), the larger diameter pipes have lower calculated rupture probabilities compared to the smaller diameter pipes at the same stress level. Based on the results shown in Table 3-4, it can be conservatively concluded that the conditional* rupture probability/weld for pipes greater than 6 inches in diameter is bounded at 1.0E-7.

Due to the assumed log normal characterization of the probability distribution for the crack aspect ratio, the LLNL piping reliability model is not appropriate for welds in pipes smaller than 6 inches in diameter. On the basis that the smaller pipes have lower pressure-induced stress for the same pressure level, it was judged that the rupture probability at circumferential butt welds in pipes smaller than 6 inches in diameter is no greater than that for the 6-inch diameter pipe.

3.2.1.4 Conclusions from Piping Integrity Evaluation

Based on the preceding evaluations, the following conclusions are drawn:

*"Conditional" probability means the probability assuming that system overpressurization has occurred.

Table 3-4

CALCULATED ECCS RUPTURE PROBABILITIES PER CIRCUMFERENTIAL WELD DURING OVERPRESSURIZATION

BASED ON PRAISE CODE

Pipe Size (in.)	Conditional Probability/Weld
6	9.8×10^{-8}
14	9.3×10^{-9}
16	6.3×10^{-9}
20	2.0×10^{-8}
24	5.4×10^{-9}

Note: Axial membrane and bending stresses listed in Table 3-3 were used in the calculation.

a. The calculated hoop stress burst margin in the ECCS piping for an overpressurization event is greater than the ASME Code specified safety margin for level D conditions which assure pressure integrity.

b. Through-wall axial cracks of significant length would have to be present to cause piping rupture. Since such through-wall cracks are likely to be detected and repaired, the probability of rupture of ECCS piping from unstable growth of latent axial defects during the overpressurization event is negligible.

c. Given that overpressurizatio⁻ has occurred, the probability of a rupture or a DEPB at any circumferential weld, in any ECCS system, is conservatively estimated at 1×10^{-7}.

3.2.2 Evaluation of Valve Integrity

A valve is an assemblage of several subcomponents including a body, stem, disc, bonnet, gland, yoke and operator. Therefore, the pressure integrity evaluation of a valve is more complex than piping. A quantitative probabilistic analysis method for the valves similar to the LLNL piping reliability model for piping is not currently available. Nevertheless, several inherent design features of the valves were examined to draw qualitative conclusions that the probability of pressure boundary rupture at a valve during the overpressurization event is expected to be no higher than that for the connected piping.

Figure 3-6 shows the cross-section view of an Anchor/Darling 150 pound-rated 14-inch motor-operated gate valve used on the low pressure side of a typical HPCI line. The two most likely failure modes by which a large break area could result during an overpressurization are: (1) rupture of the valve body and (2) failure of the body-to-bonnet joint. Each of these modes is evaluated separately below.

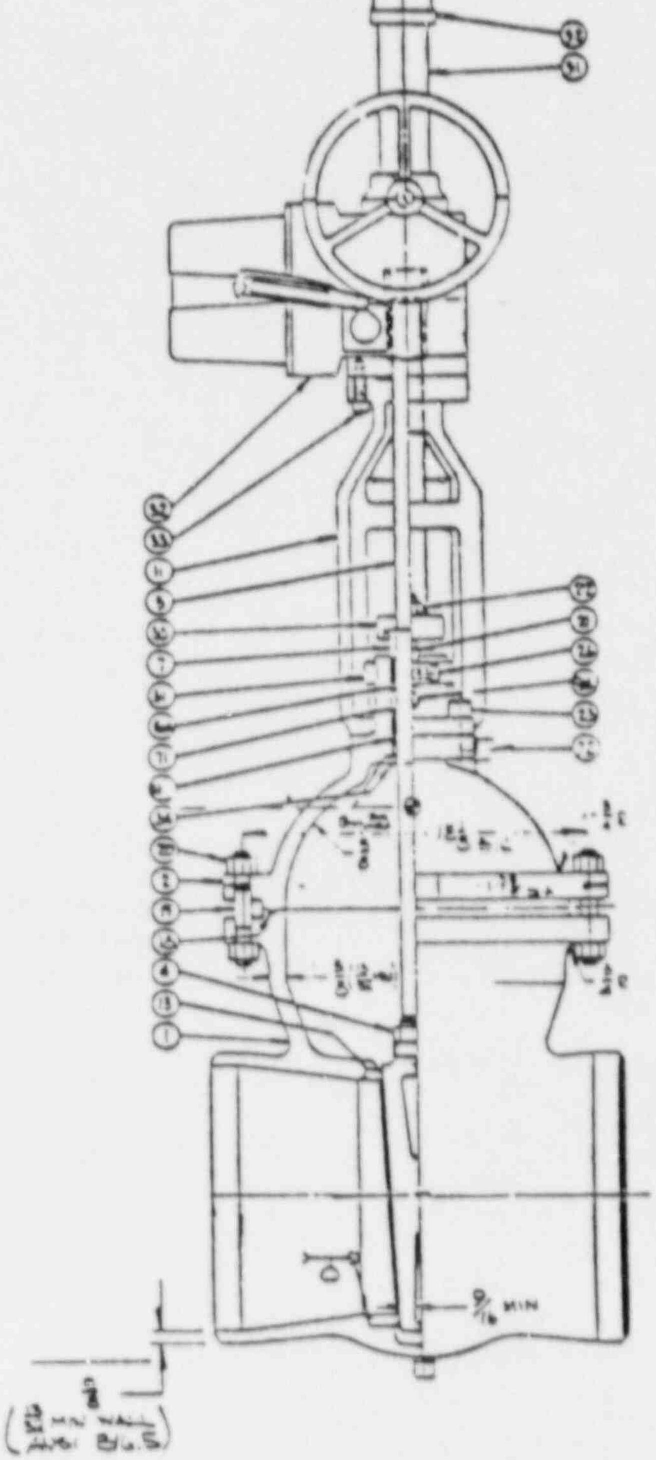


Figure 3-6. Schematic of 150-lb Pressure Rated HPCI Valve

3.2.2.1 Valve Body Integrity Evaluation

A review of the data in Appendix A indicates that the valves in the RHR and the core spray low-pressure systems which would be subjected to overpressurization are rated at 300 pounds. The HPCI/RCIC suction line valves are rated at 150 pounds. Table 3-5 (obtained from Reference 2) shows the required minimum body thicknesses for the valves of various sizes and ratings. It is seen that these minimum specified thicknesses are considerably higher than those specified for the same nominal diameter standard schedule pipes used typically on the low design pressure ECCS piping. For example, for a 14-inch valve, rated at 150 pounds, the specified minimum valve body thickness in Table 3-5 is 0.42 inches. (The specified minimum wall thickness for the valve of the same rating shown in Figure 3-6 is 5/8 inch or 0.625 inches for a margin of 1.5.) This also represents a greater than 10% margin above the nominal for that of a 14-inch standard schedule pipe whose thickness is 0.375 inches. The larger thicknesses in the valve bodies are intended to limit deformation to make valve function properly (e.g., to assure leak tightness at the valve seat). Rodabaugh (Reference 8, p. 11-1) has observed that because of this, the valve body is, in most cases, rigid to the extent that the pipe section attached to a valve will yield prior to its being able to impart sufficient forces to cause a pressure boundary failure of the valve. A review of the reported field failure incidents on piping systems also confirms this observation. Thus, the likelihood of valve body rupture during the overpressurization event is less than the rupture of the connected piping. Additionally, the valves also benefit from higher allowable stresses. The typical carbon steel material specified for valve bodies is SA216 WCB, which has a minimum room temperature ultimate strength of 70 ksi compared to 60 ksi for the carbon steel SA106 B used in piping.

Another feature in the valve design that provides additional assurance of pressure integrity during the overpressurization is high rating of hydrotest pressures. Table 3-6 (obtained from Reference 2) shows the specified hydrotest pressures for typical valves. Table 3-6 data indicate that 150-pound carbon steel valves are subjected to hydrotest pressures of 425 psi, while 300-pound valves are tested to withstand pressures of 1125 psi. Thus, 300-

Table 3-5

VALVE BODY MINIMUM THICKNESS

(Table NC-3511-1 of Ref. 2)

Inside Diameter d_m, in.	Minimum Wall Thickness, t_m, in.						
	Primary Pressure Rating P_r, lb.						
	150	300	400	600	900	1500	2500
0.1	.10	.10	.10	.10	.10	.10	.10
0.2	.10	.10	.10	.10	.10	.10	.12
0.3	.10	.10	.10	.10	.12	.12	.18
0.4	.10	.10	.13	.13	.16	.16	.23
0.5	.10	.10	.16	.16	.19	.19	.27
0.6	.10	.12	.16	.16	.21	.21	.30
0.7	.11	.14	.16	.16	.22	.22	.33
0.8	.13	.16	.17	.17	.24	.24	.36
0.9	.15	.17	.18	.18	.26	.26	.40
1	.16	.19	.19	.19	.28	.28	.44
2	.16	.25	.25	.25	.46	.46	.79
3		.28	.31	.31	.42	.66	1.14
4		.31	.38	.38	.51	.83	1.47
5		.38	.44	.44	.63	1.02	1.81
6	...	.38	.44	.50	.74	1.21	2.15
7	.30	...	.50	.57	.83	1.41	2.51
8	.31	...	.56	.63	.93	1.59	2.83
9	.33	.47	.63	.70	1.03	1.76	3.17
10	.34	.50	.69	.77	1.13	1.94	3.51
11	.36	.53	.72	.85	1.24	2.12	3.85
12	.38	.56	.75	.92	1.35	2.31	4.19
13	.40	.61	.81	.97	1.46	2.50	4.52
14	.42	.65	.84	1.03	1.56	2.69	4.86
15	.43	.68	.88	1.11	1.67	2.88	5.20
16	.45	.71	.91	1.18	1.77	3.06	5.54
17	.46	.75	.94	1.25	1.86	3.24	5.88
18	.48	.78	1.00	1.31	1.96	3.42	6.22
19	.50	.81	1.07	1.39	2.07	3.61	6.55
20	.51	.84	1.10	1.46	2.17	3.79	6.89
21	.53	.88	1.13	1.53	2.28	3.97	7.23
22	.54	.91	1.17	1.59	2.38	4.15	7.57
23	.56	.94	1.20	1.66	2.48	4.33	7.91
24	.57	.97	1.24	1.72	2.59	4.51	8.25
25	.59	1.00	1.28	1.79	2.69	4.69	8.59
26	.61	1.04	1.32	1.85	2.79	4.87	8.92
27	.62	1.07	1.36	1.91	2.89	5.05	9.26
28	.64	1.10	1.39	1.98	2.99	5.24	9.60
29	.66	1.14	1.43	2.04	3.09	5.42	9.94
30	.67	1.17	1.47	2.11	3.19	5.60	10.28

Table 3-5 (Continued)
VALVE BODY MINIMUM THICKNESS
(Table NC-3511-1 of Ref. 2)

Inside Diameter d_m, in.	Minimum Wall Thickness, t_m, in. Primary Pressure Rating p_r, lb						
	150	300	400	600	900	1500	2500
31	.69	1.20	1.51	2.17	3.30	5.78	10.62
32	.71	1.23	1.54	2.23	3.40	5.96	10.95
33	.72	1.27	1.58	2.30	3.50	6.14	11.29
34	.74	1.30	1.62	2.36	3.60	6.32	11.63
35	.75	1.33	1.65	2.43	3.70	6.50	11.97
36	.77	1.37	1.69	2.49	3.80	6.68	12.31
37	.79	1.40	1.73	2.55	3.90	6.87	12.65
38	.80	1.43	1.77	2.62	4.01	7.05	12.98
39	.82	1.47	1.81	2.68	4.11	7.23	13.32
40	.84	1.50	1.84	2.75	4.21	7.41	13.66
41	.85	1.53	1.88	2.81	4.31	7.59	14.00
42	.87	1.56	1.92	2.88	4.41	7.77	14.34
43	.88	1.60	1.96	2.94	4.51	7.95	14.68
44	.90	1.63	1.99	3.00	4.61	8.13	15.01
45	.92	1.66	2.03	3.07	4.72	8.32	15.35
46	.93	1.70	2.07	3.13	4.82	8.50	15.69
47	.95	1.73	2.11	3.20	4.92	8.68	16.03
48	.97	1.76	2.14	3.26	5.02	8.86	16.37
49	.98	1.80	2.18	3.32	5.12	9.04	16.71
50	1.00	1.83	2.22	3.39	5.22	9.22	17.04

Table 3-6

HYDROSTATIC SHELL TEST PRESSURES

(Table NC-3512(c)-2 of Ref. 2)

(All Pressures are in Pounds per Square Inch Gage-psig)

	MATERIAL												
	Ferritic Steel						Austenitic Steel						
							Types						
Class	Carbon Steel	Carbon Steel (Low Temp)	Carbon Moly	1 Cr-$\frac{1}{2}$ Mo	1$\frac{1}{4}$ Cr-$\frac{1}{2}$ Mo	2$\frac{1}{4}$ Cr-1 Mo	321	304	347	316	310	304L	316L
150	425	425	425	425	425	425	425	425	425	425	425	425	425
300	1125	975	1050	1125	1125	1125	1125	1125	1125	1125	1125	1000	1000
400	1500	1275	1400	1500	1500	1500	1500	1500	1500	1500	1500	1325	1325
600	2250	1925	2075	2250	2250	2250	2250	2250	2250	2250	2250	2000	2000
900	3375	2900	3125	3375	3375	3375	3375	3375	3375	3375	3375	3000	3000
1500	5625	4825	5200	5625	5625	5625	5625	5625	5625	5625	5625	5025	5025
2500	9375	8025	8675	9375	9375	9375	9375	9375	9375	9375	9375	8350	8350

pound valves have already been hydrotested at a pressure greater than that expected during the overpressurization event (i.e., 1050 psi).

From the preceding discussion, three conclusions are drawn. First, the likelihood of rupture of 150-pound valve bodies during an overpressurization event appears no greater than that of the corresponding diameter standard schedule pipe. Second, for the same stresses, valves have a higher design margin due to higher allowable stresses compared to piping. Third, because of high hydrotest pressures, the rupture of 300-pound (or larger) valve bodies during overpressurization also appears highly unlikely.

3.2.2.2 Body-to-Bonnet/Cover Bolted Joint Evaluation

Another part of the valve whose failure could lead to a breach of the pressure boundary is the bolted joint between the body and bonnet or cover (Figure 3-6).

As noted in the preceding subsection, the 300-pound pressure rated valves are required to be hydrotested at 1125 psi. Therefore, only the bolted joints in the 150-pound pressure rated valves were evaluated. This included valves in the 6, 14 and 16-inch sizes. The body-to-bonnet joint in the gate valve shown in Figure 3-6 was also determined to be the bounding case and therefore was used in this evaluation.

The bonnet in Figure 3-6 is attached to the valve body by sixteen 3/4-inch 10NC-2 studs on an 18-1/8 inch bolt circle diameter. Appendix C describes the theoretical relationship between the bolt stress, pre-load and the pressure loading. Based on Equation C-1, the pre-load stress in the studs was estimated as $45{,}000/\sqrt{0.75}$ or ≈52,000 psi. The average stress in the studs, due to internal pressure of 1050 psi and no pre-load stress, was calculated as ≈51,000 psi. When the effect of pre-load is taken into account (using Equation C-2), the bolt stress with 1050 psi internal pressure is calculated as 57,700 psi. This represents a small increase (approximately 11%) from the stress experienced by the studs under pre-load alone. This confirms that, in most flanged joints, the major stress applied to the studs or bolts is that applied in tightening the nuts. It follows that, if a bolt or stud did not fail during tightening, then it is not likely to fail during service.

If the stress in the studs due to pre-load is less than that estimated by Equation C-1, the calculated stress during the pressurization to 1050 psi will be even lower. Thus, the average stress in the studs during an overpressurization event is estimated to range from 51,000 psi to 57,700 psi. Since the ASME Code implied minimum yield strength at 550°F for the stud material (SA 193 B7) is ≈87,000 psi, the minimum margin for the calculated stress in the studs during an overpressurization event measured against the ASME Code is 1.5. (The ASME Code implied factor of safety is 1.4 for Level D conditions in which the pressure integrity of a component is the only concern.)

The inherent structural redundancy of the bolted joint provides additional assurance that, during an overpressurization event, loss of coolant would more likely result from valve leakage than rupture, which is likely to be detected long before valve integrity would be compromised. Results of finite element analysis, on a sample bolted joint given in Reference 11, are discussed in Appendix C. Such analysis clearly illustrates the load shedding and redistribution characteristics when complete degradation of one or more adjacent studs in the joint is assumed. Reference 11 shows that the stress increase in the stud next to the failed studs is small. It should, however, be noted that a review of General Electric service experience data base on BWR pressure boundary materials has indicated no reported incidents of degradation in the low alloy steel (SA 193 B7) bolting used in ECCS piping system valves and heat exchangers. This was not surprising, since most of the factors identified in Reference 11 (i.e., the presence of borated water, stress corrosion cracking and fatigue) are not likely to be associated with the typical operating conditions in the parts of BWR ECCSs being considered in this evaluation.

The preceding discussions lead to the qualitative conclusion that the yielding of the bolted joints in the ECCS valves on the low design pressure side during an overpressurization event are more likely to result in a leak rather than a gross rupture.

3.2.2.3 Conclusions from Valve Integrity Evaluation

Based on the evaluations presented in the preceding subsections, it is concluded that:

a. The likelihood of rupture of the body of a 150-pound pressure-rated valve during an overpressurization event is less than that of the corresponding diameter standard schedule pipe. In the case of valves rated 300 pounds or greater, the prescribed hydrotest pressure of 1125 psi assures that its probability of rupture during an overpressurization event is negligible.

b. The likelihood of gross rupture at bolted joints in the low pressure side ECCS valves is extremely small. Leakage of fluid through the bolted joints is the more likely consequence during an overpressurization event. The potential effects of leakage is discussed in Section 3.2.5.

c. The overall rupture probability of a low pressure side ECCS valve was judged to be no greater than that at the circumferential butt weld between the valve and the connecting pipe.

3.2.3 Evaluation of Heat Exchanger Integrity

Several RHR heat exchanger designs were reviewed with the most limiting one selected for this evaluation. The shell inside diameters of the heat exchanger designs reviewed ranged from 40 inches to 63 inches. The typical design pressure for the RHR heat exchangers was 450 psi. The 63-inch inside diameter design was found to be most limiting and Figure 3-7 shows the outline of this heat exchanger. During RHR system operation, reactor water enters the heat exchanger at opening A and exits at B. The shell tube sheet and the channel are connected together through a flanged joint. The process cooling water from the channel side circulates through the tube bundle situated inside. The three parts of the heat exchanger that are stressed during the RHR overpressurization event are: (1) the shell; (2) shell-to-tube-sheet

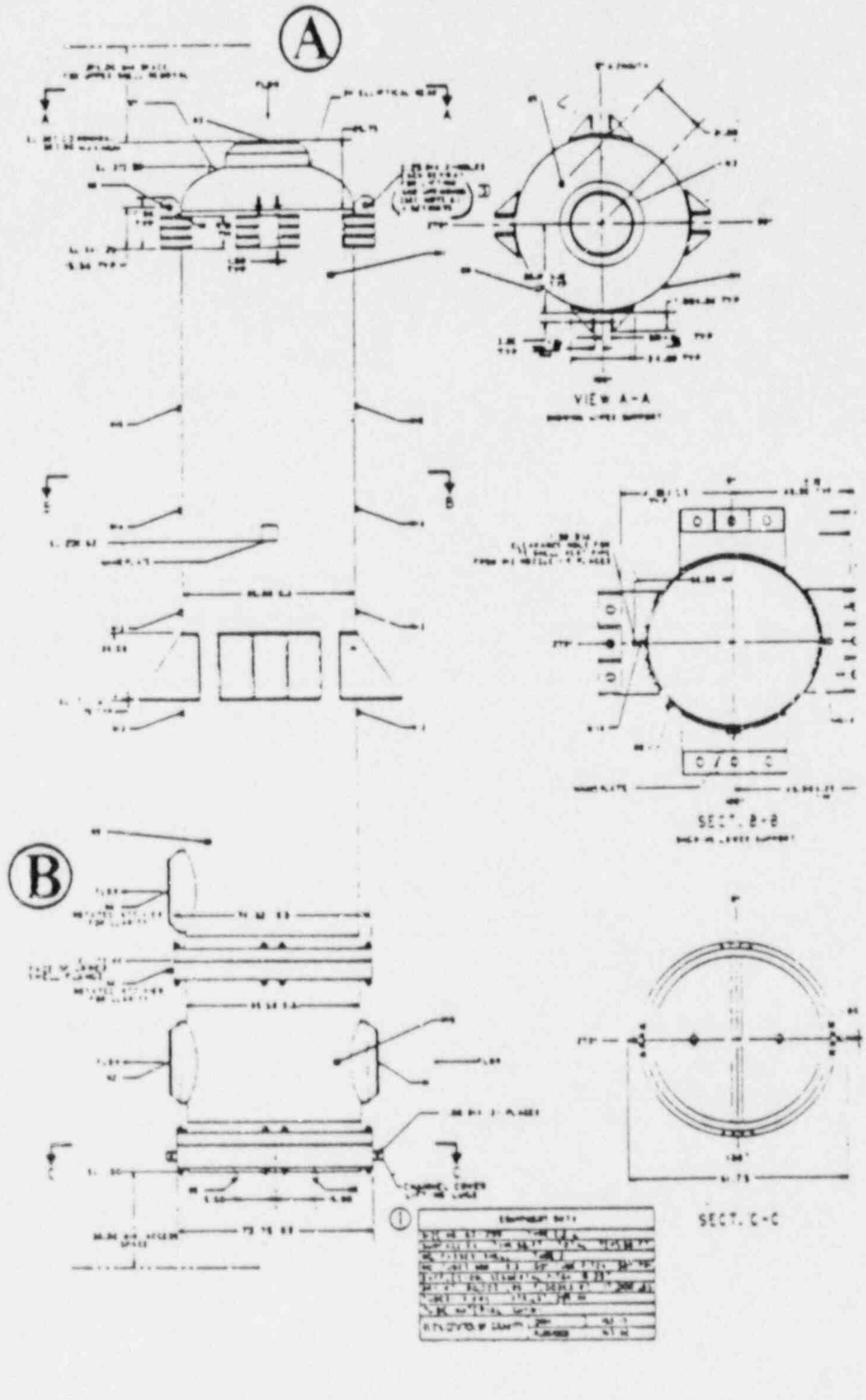


Figure 3-7. Schematic of 63-Inch I.D. RHR Heat Exchanger

Why or why not?

14. What Steps are Planned to Prevent Recurrence?

Prepared by: ____________________ Date: __________

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flanged joint; and (3) the tubes. Each of these was evaluated for large break potential during overpressurization.

3.2.3.1 Heat Exchanger Shell Evaluation

The heat exchanger shell is a cylindrical structure with a top elliptical head. The hoop stress in the cylindrical portion is expected to be governing. The nominal shell thickness for the example case is 1.0 inch. For an internal pressure of 1050 psi, assuming a standard corrosion allowance of 0.08 in., the hoop stress was calculated as 36 ksi. The material specified for the heat exchanger shell is SA516 Gr. 70. The faulted condition or Level D allowable pressure hoop stress 2 S for this material is approximately the same value as the calculated hoop stress. Furthermore, it should be noted that the calculated hoop stress of 36 ksi is for the limiting case. For most of the other RHR heat exchangers, the calculated hoop stress during overpressurization is less than 30 ksi. The primary membrane stress in the other regions of the shell is expected to be less than that calculated for the cylindrical region. Overall, the calculated hoop stresses, relative to Level D allowable values, fall essentially in the same range as those for the piping (Table 3-1). Thus, it can be qualitatively concluded that the burst failure of the worst case heat exchanger shell during an overpressurization event is no more likely than failure of the connecting piping (Section 3.2.1).

3.2.3.2 Shell-to-Tube-Sheet Bolted Joint Evaluation

The heat exchanger shell, tube sheet and channel are held together by a flanged joint. A review of such joints in various RHR heat exchanger designs indicated that the flanged joint in the 63-inch shell I.D. heat exchanger, considered in Subsection 3.2.3.1, was limiting and, therefore, was selected for evaluation. The sealing of this flange joint is assured by 68 1-3/8-inch diameter bolts equally spaced in a circular pattern. The same procedure used for valve body-to-bonnet joint evaluation in Subsection 3.2.2.2 was used to calculate the bolt stresses in the heat exchanger flange joint. Based on Equation C-1, the minimum pre-load stress in the bolts was estimated as $45,000/\sqrt{1.375}$ or ≈ 38 ksi. The average stress in the bolts, due to internal pressure of 1050 psi and assuming no pre-load stress, was calculated as

54 psi. With the bolt stresses expected during overpressurization being greater than the pre-load stresses, some leakage of reactor coolant at this joint is expected during overpressurization. In fact, in the Vermont Yankee incident (Reference 1) the RHR System overpressurization did result in leakage of steam and water mixture from the heat exchanger tube sheet-to-shell flange area. While some leakage may occur at this joint, the overall integrity of this joint is assured as demonstrated by comparing overpressurization stresses with the bolt yield stress. The ASME Code minimum specified yield stress at 550°F for bolt material SA 197 B7 is ≈87 ksi. Thus, the margin against flange bolt failure is 87/54 or 1.61.

It is concluded, therefore, that the tube-sheet-to-shell flange joint is more likely to leak rather than fail during an overpressurization event.

3.2.3.3 Tube Integrity Evaluation

Heat exchanger tubes are subjected to reactor pressure on the outside surface during an overpressurization event. Therefore, the principal failure mode would be tube buckling due to the external pressure. The tube materials range from type 304L stainless steel to copper-nickel alloy. Because of the lower yield strength, 304L stainless steel tubes were evaluated as the limiting material. The evaluated tubes have a 1-inch O.D. with a thickness of 0.049 in. (corresponding to 18 BWG). The collapse pressure of these tubes can be estimated using the procedures given in Paragraph ND-3100 of the ASME Code (Reference 2).

Figure 3-8 (Figure ND-3133.8-1 of the ASME Code) can be used to graphically determine the collapse pressure of heat exchanger tubing. Figure 3-8 illustrates design pressure as a function of design stress for various heat exchanger tube thickness-to-diameter (T/D_o) ratios. The design pressure has a built-in factor of safety of 3.0. Thus, the expected collapse pressure will be three times the design pressure determined from Figure 3-8. The design stress for 304L stainless steel at reactor temperature is 14 ksi. For a typical T/d_o of ≈0.05, the resulting design pressure from Figure 3-8 is 550 psi. Therefore, the expected collapse pressure (i.e., 3 x design

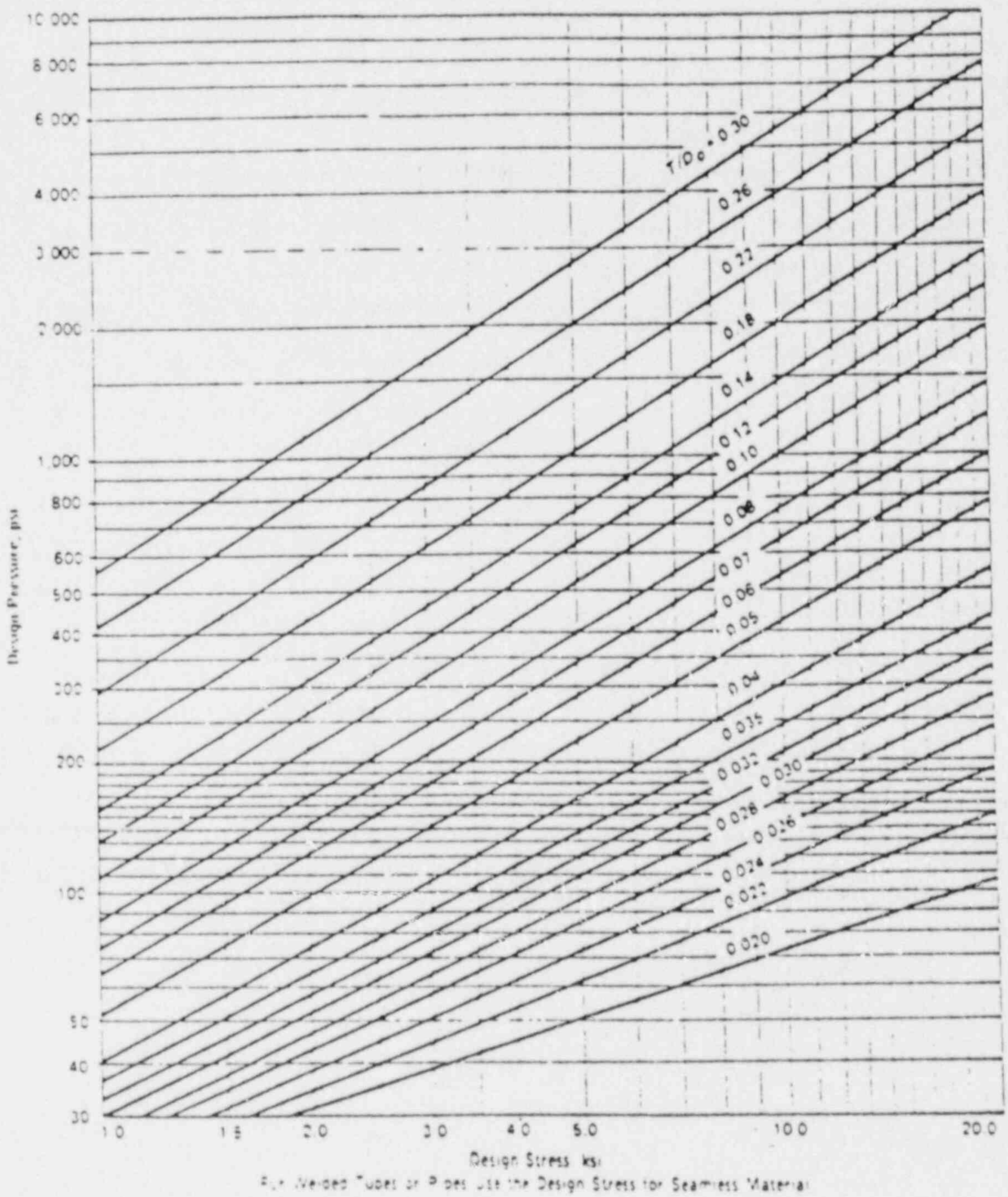


Figure 3-8. Chart-Tube Wall Thickness Versus External Pressure
(Figure ND-3133.8-1 of Reference 2)

pressure) is 1650 psi, yielding a margin against collapse failure during overpressurization of 1.57.

3.2.3.4 Conclusions from Heat Exchanger Integrity Evaluation

The preceding evaluations lead to the following qualitative conclusions:

a. The likelihood of rupture of a RHR heat exchanger shell during an overpressurization event is of the same order of magnitude as the connected piping (discussed in Section 3.2.1).

b. The tube sheet-to-shell bolted joint, which would be stressed during an overpressurization event, is likely to leak rather than experience a gross rupture.

c. Heat exchanger tubes have an inherent safety margin of three against collapse during an overpressurization event.

d. The overall rupture probability of an RHR heat exchanger was judged to be no greater than that at the circumferential butt weld in the connecting RHR piping.

3.2.4 Overall System Rupture Probability Evaluation

The overall ECCS low pressure piping system rupture probability during an overpressurization event is equal to the sum of the rupture probabilities of the piping and the piping components such as valves and heat exchangers. In the case of piping, it was judged that the rupture probability at the circumferential butt welds was the dominant contributor. Thus, the rupture probability in the piping was defined as the product of the per-circumferential butt weld rupture probability (conservatively estimated as 1.0E-7 from Subsection 3.2.1.3) and the number of circumferential butt welds in the system. The rupture probabilities for valves and heat exchangers was approximated as equal to that at a circumferential butt weld as discussed in Subsections 3.2.2.3 and 3.2.2.4 (e.g., each valve and heat exchanger counts as an additional circumferential butt weld in the system).

The number of welds in the portions of ECCS piping affected by overpressurization depends upon the system configuration. In a limited plant survey, the number of welds in an RHR system were determined as 112 for pipe sizes three inches and larger. Similarly, the number of valves 3/4 inches and larger in a typical RHR system was determined to be 91. On this basis, the number of "equivalent circumferential welds" was conservatively assumed as 300. Thus, the ECCS low pressure system piping rupture probability is estimated as (300)x(1.0E-7) or 3.0E-5. This expected probability is at least two orders of magnitude lower than the range of 1.0E-2 to 1.0E-3 stated in the AEOD case study.

3.2.5 Probable Results of Overpressurization

As indicated by the above analysis discussion, failure of system components due to overpressurization is not expected to occur.

The results of such an overpressurization event would most likely be limited to:

(1) Discharge of fluid and two-phase fluid from systems relief valves.

(2) Leakage of fluid and two-phase fluid from bolted joints and possibly failure of some gasket(s).

(3)* Discharge of fluid and two-phase fluid from small undetected cracks in welds.

*It is judged that only small cracks in welds may fail to be undetected during normal plant surveillance and that the number of such small cracks would be few. The effective flow area of these cracks would be small. For example, the effective flow area of such cracks would be much less than the two-square-inch effective flow area of a large 25-inch long crack of critical size in a 24-inch diameter pipe weld. (A crack of critical size is the length of crack that can be present without resulting in a guillotine failure of the pipe during an overpressurization event.)

(4) Activation of line high pressure alarms.

(5) Activation of smoke alarms due to oxidation of paint on piping and components.

(6) Activation of area high temperature alarms due to steam that results from the discharge of two-phase fluid.

The volume of fluid discharged from bolted joints would be largely limited by the minimal flexure of the joint and bolts. Following termination of the overpressurization condition, the joint would likely reseat and stop leaking unless gasket rupture had occurred. If gasket failure occurs, additional fluid discharge could result and may continue after termination of the overpressurization until the system sufficiently depressurizes.

The volume discharged from relief valves, bolted joints, undetected cracks, and gaskets if they fail, is judged to be small compared to the volume of associated equipment rooms such that the leakage would not be expected to result in substantial flooding consequences for overpressurization events of durations similar to those reported in Reference 1.

The most significant consequences to be expected would be the additional hazard to plant personnel from discharged fluid and the potential that some equipment may be rendered inoperable due to spray from leaking components. The equipment that may be rendered inoperable due to spray effects is expected to be limited to equipment in the same division as the overpressurized system because of equipment divisional separation.

Activation of line high pressure alarms, smoke alarms, high area temperature alarms and personnel observation of smoke, spray and steam discharge from relief valves and leaking components would likely result in decreasing the duration of the overpressurization event by alerting operators to the overpressurization condition. These phenomena resulted in alerting operators to the overpressurization condition in the events reported in Reference 1.

There is a high likelihood that operators would be able to isolate the discharge from relief valves and leaking components by closing the system injection valves. The HPCI and RCIC Systems' injection valves are generally located outside the room in which the systems' low design pressure components are located. This further improves the likelihood of being able to isolate discharges and leaks from HPCI and RCIC Systems because of the reduced potential environmental effects of the discharges and leaks on the system injection/isolation valves. The likelihood of operators being able to isolate such discharges and leaks is, to some degree, demonstrated by the fact that the operators were able to isolate the discharges and leaks reported in Reference 1.

3.? PROBABILITY OF AN INTERFACING LOCA

The expected frequency of a BWR interfacing LOCA involving the reactor coolant system (RCS) and the ECCS and RCIC system is determined as follows:

$$P_{LOCA} = P_{Press} \times P_{Rupture}$$

where:

P_{LOCA} = Probability of an interfacing LOCA between the RCS and ECCS.

P_{Press} = Probability of overpressurizing the low pressure ECCS and RCIC system piping.

$P_{Rupture}$ = Conditional probability of a rupture in the ECCS piping given an overpressurized condition.

Substituting the values of the probability of overpressurization and the conditional probability of ECCS rupture, the probability of an interfacing LOCA is, therefore, determined to be:

$$P_{LOCA} = (1.0E-2)^{*} \times (3.0E-5)$$

$$P_{LOCA} = 3.0E-7 \text{ per reactor-year}$$

The value of 3.0E-7 per reactor-year for the expected frequency of an interfacing LOCA is acceptably low, nearly three (3) orders of magnitude lower than the expected frequency of a large break LOCA described in the WASH-1400 Reactor Safety Study. It is, therefore, concluded that an ECCS overpressurization event poses no significant threat to the safety of the BWR.

*For purposes of calculation, the frequency of system overpressurization is assumed to be that stated in Reference 1. However, it is believed that this event frequency is overly conservative based on a continued operating data base without additional occurrences since the issuance of Reference 1 and the increased industry awareness of these occurrences and the need to reduce event frequency.

4. REFERENCES

1. Lam, P., "Overpressurization of Emergency Core Cooling System in Boiling Water Reactors", Report AEOD/C502, September 1985.

2. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant components, American Society of Mechanical Engineers.

3. USA Standard Code for Pressure Piping, Section I, Power Piping Systems, B31.1.

4. Wilkowski, et al, "Summary of NRC Phase I Degraded Piping Program - Instability Analyses and Review of Experimental Programs to Evaluate J/T Instability Predictions", ASME PVP - Vol. 95, 1984.

5. S. Ranganath, H.S. Mehta and D.M. Norris, "Structural Evaluation of Flaws in Power Plant Piping", ASME PVP - Vol. 95, 1984.

6. A Zahoor, H.S. Mehta, S. Yukawa, R.M. Gamble and S. Ranganath, "Flaw Evaluation Procedures and Standards for Ferritic Piping", EPRI Draft Final Report on Research Project RP 1757-51, July 1985.

7. "Investigation of the Initiation and Extent of Ductile Pipe Rupture", R.J. Eiber, et al, Battelle Columbus Laboratories, Report BMI-1908, June 1971.

8. E.C. Rodebaugh, et al, "Survey Report on Structural Design of Piping Systems and Components", TID-25553, December 1970, National Technical Information Service.

9. M.B. Reynolds, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws", General Electric Report GEAP-5620, April 1968.

10. "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant", NUREG/CR-2189, Vol. 1 through 9.

11. Nickel, R.T., R.C. Cipolla and E.A. Merrick, "The Use of Leak-Before-Break Criteria and Assessment of Margins in Addressing Closure Integrity Issues", SMIRT-8, Paper D6/3, August 19, 1985.

APPENDIX A

BWR PLANT DATA SUMMARY

APPENDIX A
PLANT DATA SUMMARY

The data received from nineteen (19) BWR plants that were utilized in the study to evaluate the probability of an interfacing LOCA are summarized in this Appendix.

Table A-1 lists the utilities, plants, BWR type, and containment type for the plants considered in the study.

Table A-2 summarizes the system design pressure, hydro test pressure, largest pipe diameter and the largest diameter pipe radius (r) to pipe thickness (t) ratio for the piping sections identified in Paragraph 3.1 of this report as being subject to potential overpressurization.

The maximum pipe hoop stress due to pressurization is equal to the r/t ratio multiplied by the pressure to which the piping is exposed.

Table A-2 and Figures A-1 through A-31 summarize the reactor vessel isolation valve configuration for each system. The figures do not differentiate between gate and globe valves (i.e., the same symbol is used for both). The piping and valve configuration upstream of the reactor pressure vessel (RPV) isolation valves are plant-specific and vary in detail. However, the piping sections subject to potential overpressurization are, in all cases, the same as illustrated in Figures 3-1, 3-2 and 3-3. RHR RPV isolation valves that are utilized for the low pressure coolant injection (LPCI) function are presented in these figures. Other RHR RPV isolation valves that are utilized for the reactor shutdown cooling function only (such as those on the RHR suction line from the RPV, RPV head spray and shutdown return to the recirculation system on or feedwater line) are not presented because these valves are interlocked to prevent their opening when reactor pressure is above the design pressure of the connected low design pressure piping, they do not receive any automatic opening signal and these valves are not tested for operability during normal power operation.

All piping material is sealess A 106 Grade B or A 333 Grade 1 or 6 carbon steel.

All valve body material is A 216 WCB carbon steel.

All RHR, LPCI/CS, CS and LPCS valves downstream of the pump discharge check valves, with the exception of the RPV isolation valves, have a minimum pressure rating of 300 lbs. All HPCI and RCIC system valves upstream of the main pump suction inlet have a minimum pressure rating of 150 lbs.

Table A-1

UTILITY, PLANT NAME, BWR AND CONTAINMENT TYPE

BWR OWNER	PLANT	BWR	CONTAINMENT TYPE (MARK I, II OR III)
Boston Edison Company	Pilgrim	3	I
Carolina Power & Light Co.	Brunswick 1&2	4	I
Commonwealth Edison Co.	Dresden 2&3	3	I
	Quad Cities 1&2	3	I
	LaSalle 1&2	5	II
Georgia Power Co.	Hatch 1&2	4	I
General Public Utilities Nuclear	Oyster Creek	2	I
Gulf States Utilities	River Bend 1	6	III
Illinois Power Company	Clinton 1	6	III
Iowa Electric Light & Power Co.	Duane Arnold	4	I
Nebraska Public Power District	Cooper	4	I
Niagara Mohawk Power Corp.	Nine Mile Point 1	2	I
	Nine Mile Point 2	5	II
Northern States Power Co.	Monticello	3	I
Pennsylvania Power & Light Co.	Susquehanna 1&2	4	II
Philadelphia Electric Co.	Peach Bottom 2&3	4	I
	Limerick 1&2	4.5	II
Public Service Electric & Gas Co.	Hope Creek 1	4.5	I
Tennessee Valley Authority	Browns Ferry 1,2&3	4	I
Washington Public Power Supply System	Hanford 2	5	II

Table A-2

BWROG PARTICIPATING UTILITY PLANT DATA

PLANT	SYSTEM	ISOLATION VALVE CONFIG. (Fig.)*	DESIGN PRESS. (psig)	HYDRO PRESS. (psig)	LARGEST PIPE DIA. (in.)	LARGEST PIPE (r/t)
Browns Ferry 1,2&3	RHR	A-1	450		24	27.6
	CS	A-1	500		14	22.7
	HPCI	A-12	150		16	26.1
	RCIC	A-12	150		6	15.6
Brunswick 1 & 2	RHR	A-1	460	690	24	23.9
	CS	A-1	460	690	12	24.5
	HPCI	A-13	150		16	26.1
	RCIC	A-14	150		6	15.6
Clinton	RHR	Later	500	750	18	
	LPCS	Later	600	900	14	
	RCIC	Later	75	113	6	
Cooper	RHR	A-1	450	450	24	23.9
	CS	A-1	500	500	12	20.6
	HPCI	A-15	150		16	26.1
	RCIC	A-15	150		6	15.6
Dresden 2&3	LPCI	A-1	350		18	24.1
	CS	A-1	350		12	20.6
	HPCI	A-16	150		16	26.1
Duane Arnold	RHR	A-1	375		20	32.9
	CS	A-1	355		10	17.9
	HPCI	A-12	125		14	22.7
	RCIC	A-12	125		6	15.6
Hanford 2	RHR	A-2	500	625	20	22.8
	LPCS	A-2	470	588	16	26.1
	RCIC	A-17	125	110	6	15.6
Hatch 1 & 2	RHR		375/400	469	24	23.9
	CS					
	HPCI		125/140	156	16	26.1
	RCIC		100/125 /140	156	6	15.6
Hope Creek 1	RHR	A-2	410	615	20	22.8
	CS	A-9/A-3	500	750	14	23.9
	HPCI	A-18	105	158	16	26.1
	RCIC	A-19	105	158	6	15.6

*Identifies the appropriate figure number contained in this Appendix.

Table A-2 (Continued)

BWROG PARTICIPATING UTILITY PLANT DATA

PLANT	SYSTEM	ISOLATION VALVE CONFIG. (Fig.)*	DESIGN PRESS. (psig)	HYDRO PRESS. (psig)	LARGEST PIPE DIA. (in.)	LARGEST PIPE (r/t)
LaSalle 1 & 2	RHR	A-4	500	750	18	17.7
	LPCS	A-4	550	825	16	18.0
	RCIC	A-31	100	150	8	16.8
Limerick 1 & 2	RHR	A-4	420	630	20	32.9
	CS	A-9/A-5	500	750	14	22.7
	HPCI	A-20	125	182	16	26.1
	RCIC	A-21	125	182	6	15.6
Monticello	RHR	A-6	275	413	16	26.1
	CS	A-6	303	455	10	17.9
	HPCI	A-22	23/30	90	14	22.7
	RCIC	A-23	50	75	6	15.6
Nine Mile Point 1	CS	A-7	470		12	20.6
Oyster Creek	CS	A-8	300		10	17.9
Peach Bottom 2 & 3	RHR	A-9	450	675	24	27.6
	CS	A-9	450	675	14	22.7
	HPCI	A-24	150	225	16	26.1
	RCIC	A-24	150	225	6	15.6
Pilgrim	RHR	A-10	500	535	18	29.5
	CS	A-11	300	550	10	17.9
	HPCI	A-25	80/60	110	16	26.1
	RCIC	A-25	80		6	15.6
Quad Cities 1 & 2	RHR	A-1	400	510	18	24.1
	CS	A-1	500	594	12	20.6
	HPCI	A-27	150	188	16	26.1
	RCIC	A-26	150		6	15.6
River Bend	RHR	A-2	500	750	18	29.5
	LPCS	A-2	600	900	14	22.7
	RCIC	A-28	90	135	6	15.6
Susquehanna 1 & 2	RHR	A-9	450	563-670	24	18.7
	CS	A-9	500	635-660	14	22.7
	HPCI	A-29	100	165	16	26.1
	RCIC	A-30	150	200	6	15.6

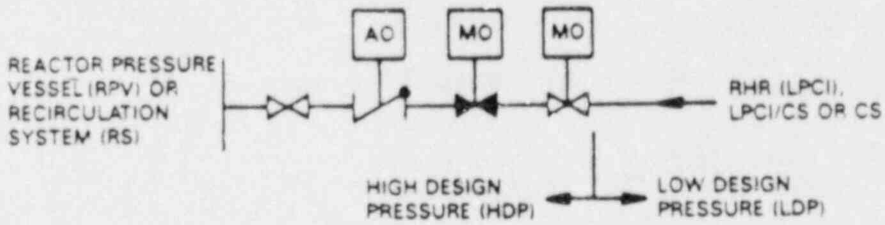


FIGURE A-1

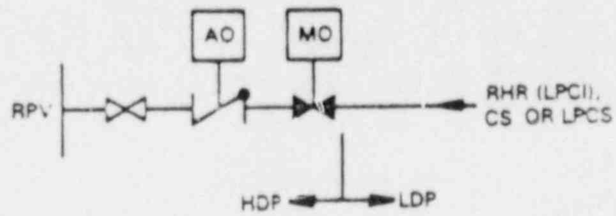


FIGURE A-2

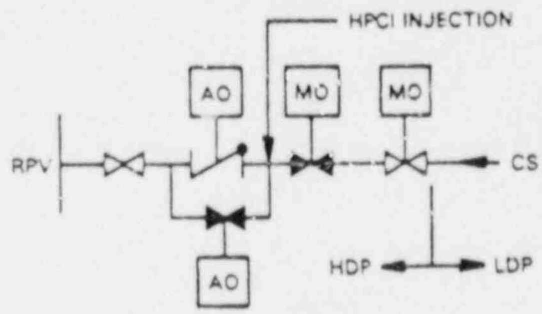


FIGURE A-3

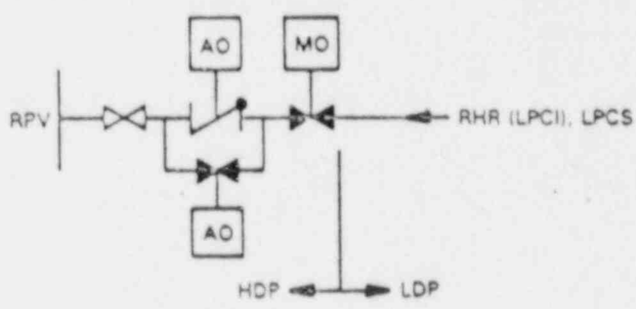


FIGURE A-4

FIGURE A-5

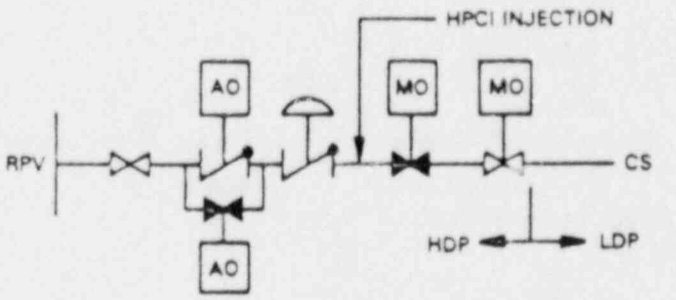


FIGURE A-6

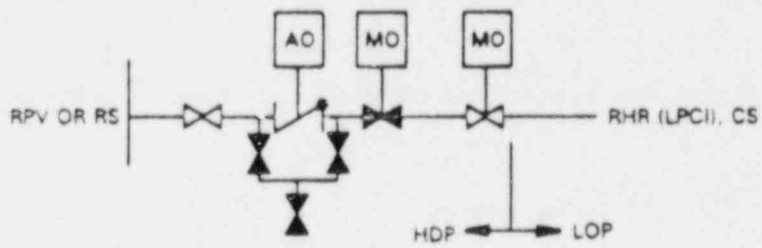


FIGURE A-7

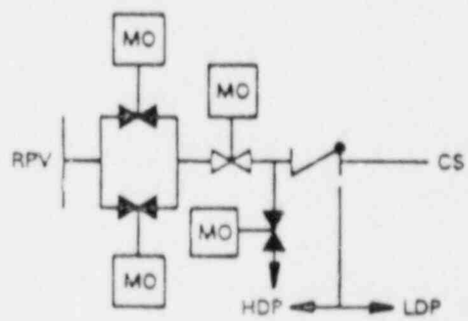


FIGURE A-8

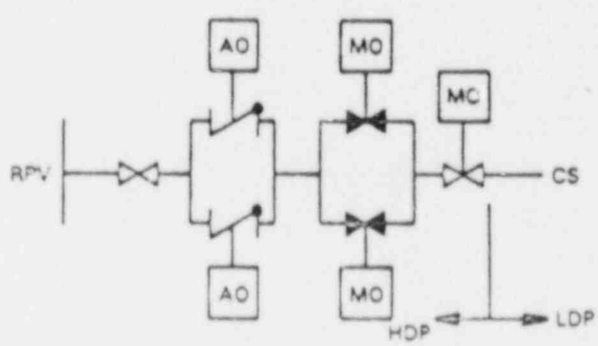


FIGURE A-9

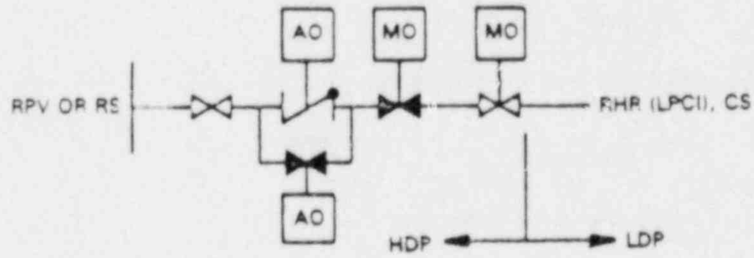


FIGURE A-10

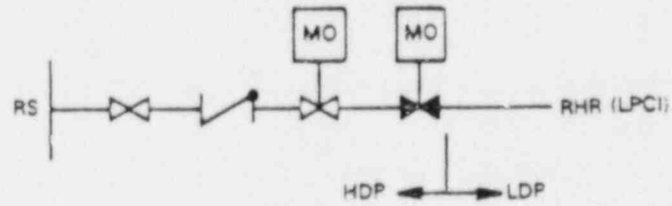


FIGURE A-11

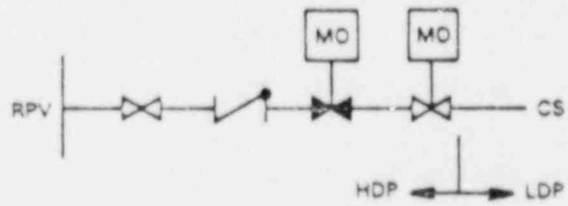


FIGURE A-12

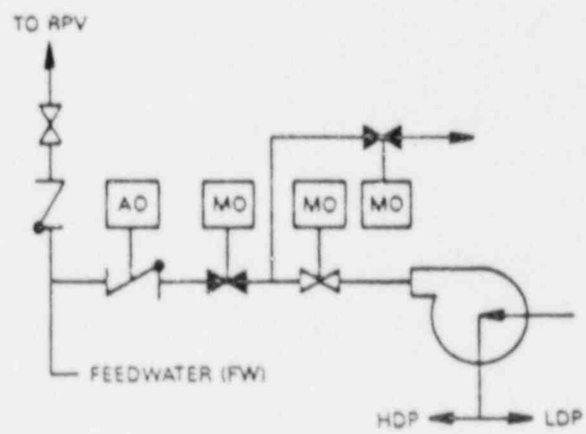


FIGURE A 13

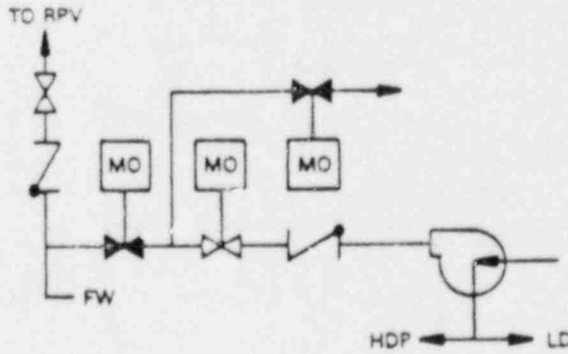


FIGURE A-14

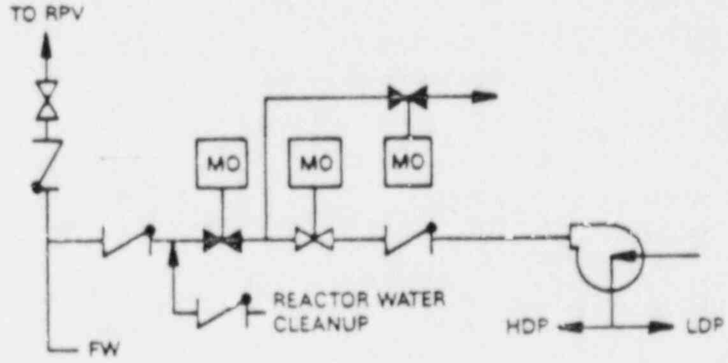
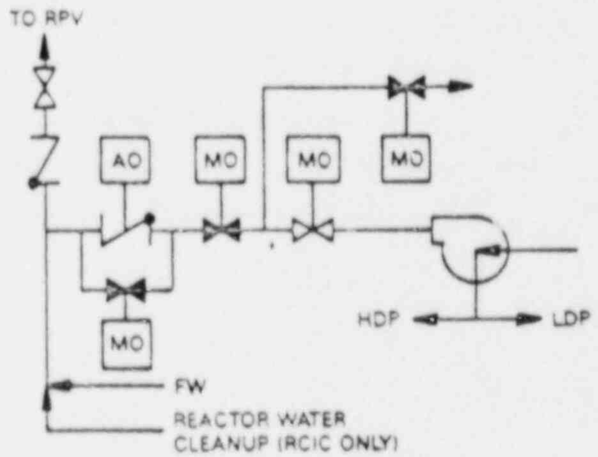


FIGURE A-15



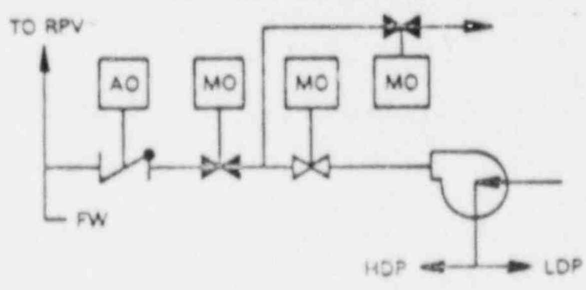
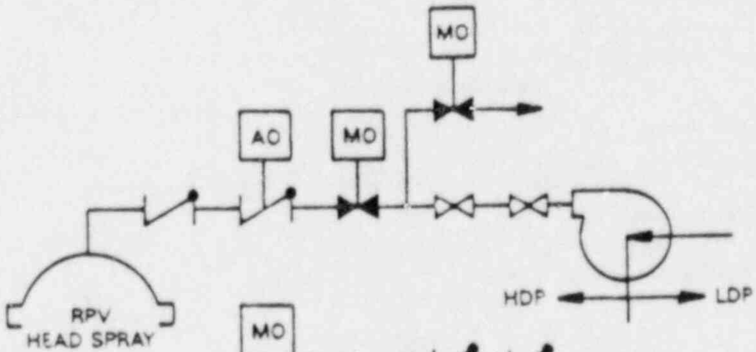


FIGURE A-16

FIGURE A-17



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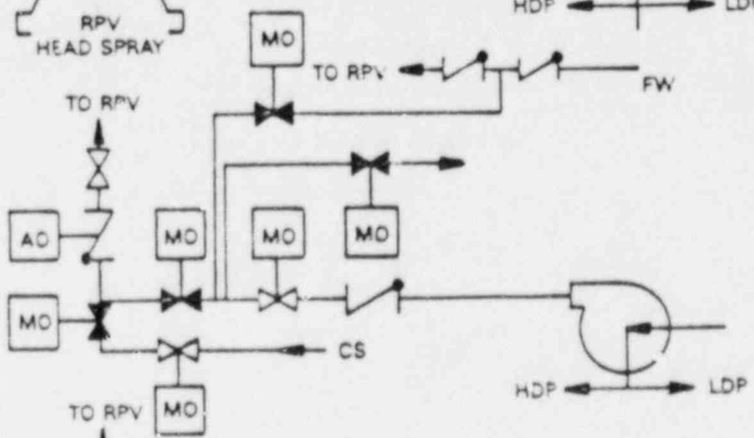

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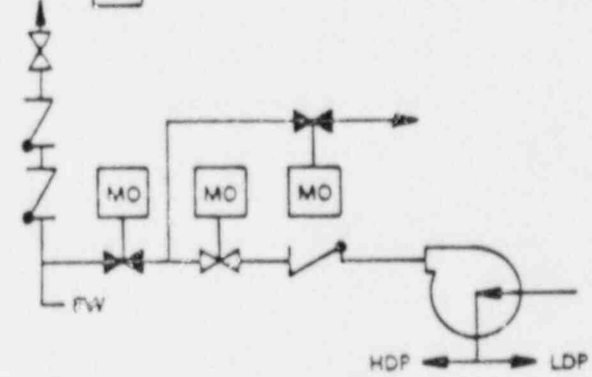


FIGURE A-20

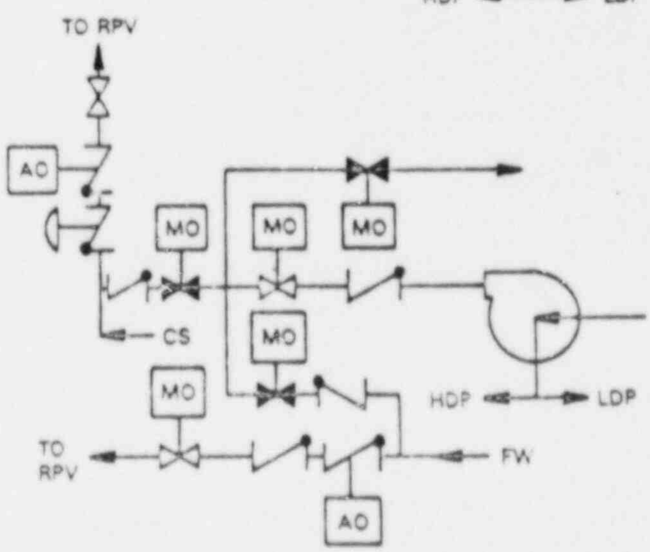


FIGURE A-21

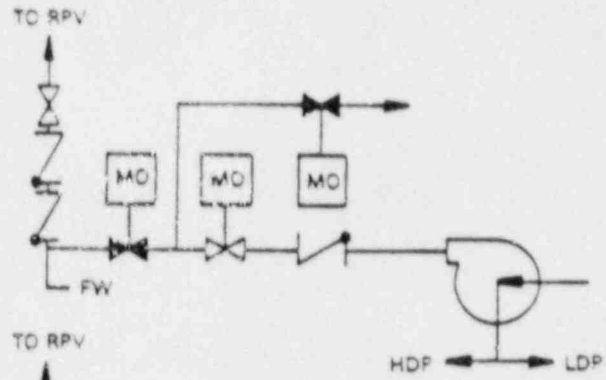


FIGURE A-22

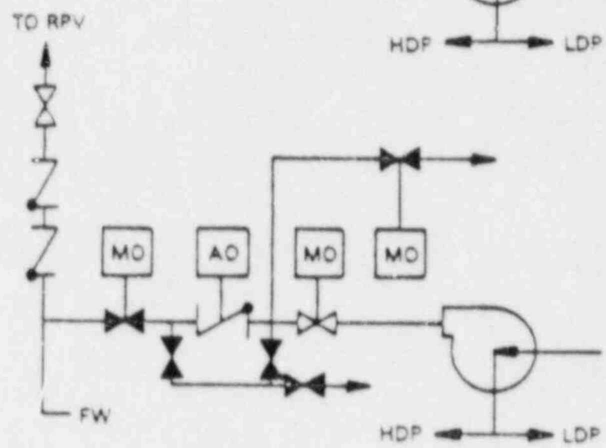


FIGURE A-23

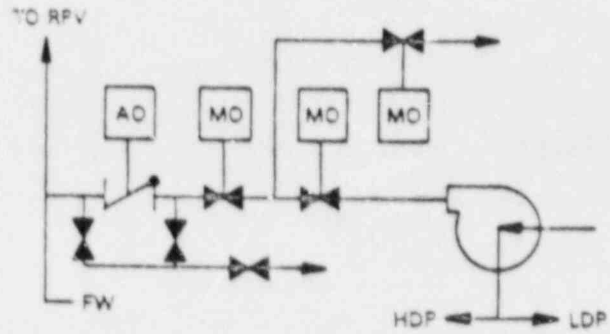


FIGURE A-24

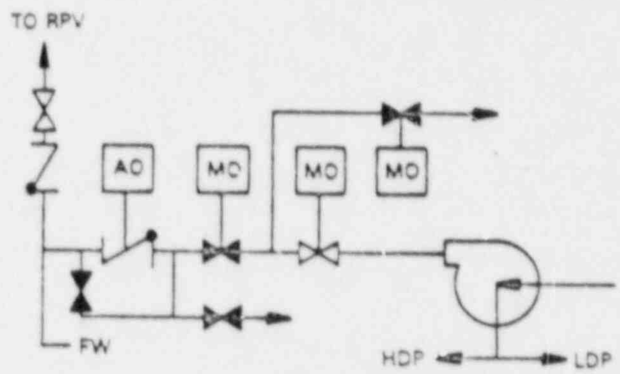


FIGURE A-25

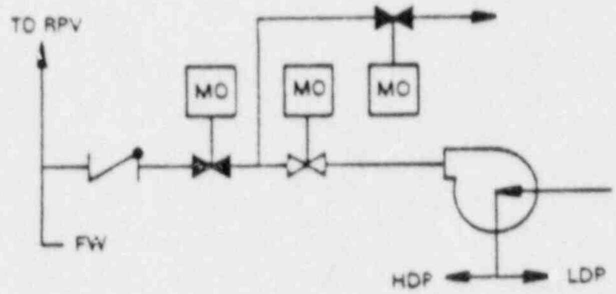


FIGURE A-26

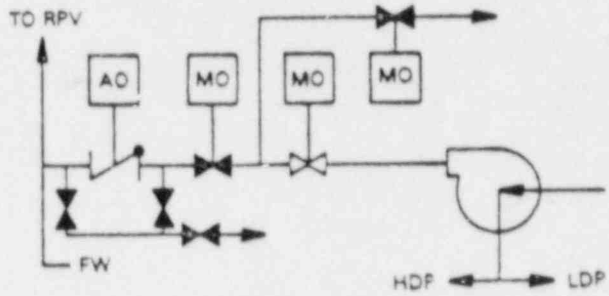


FIGURE A-27

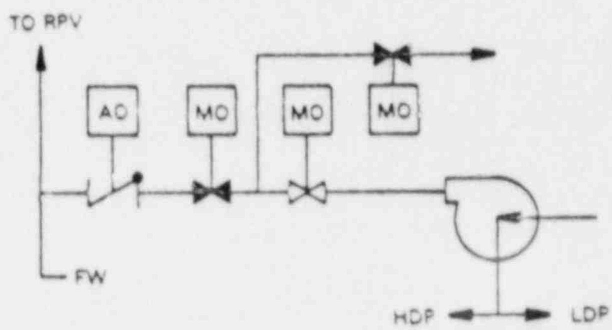
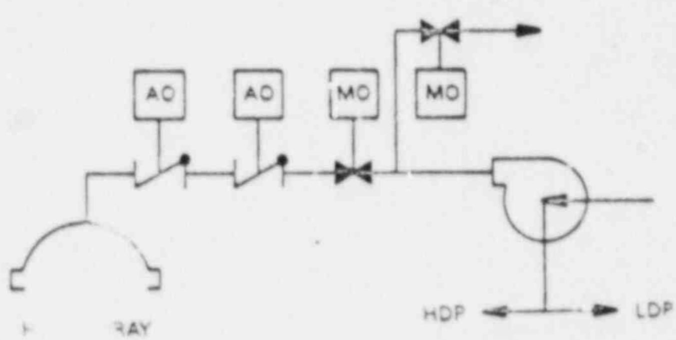


FIGURE A-28



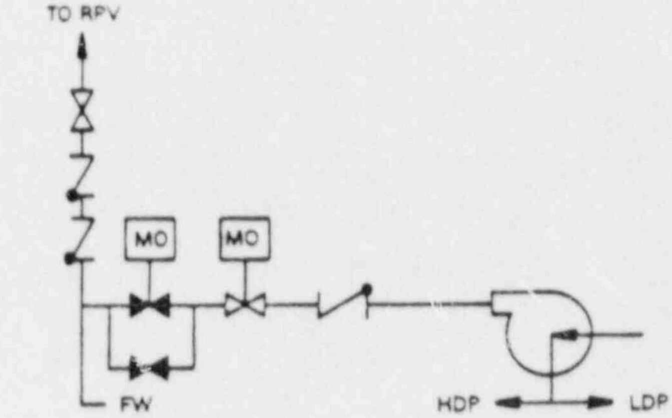


FIGURE A-29

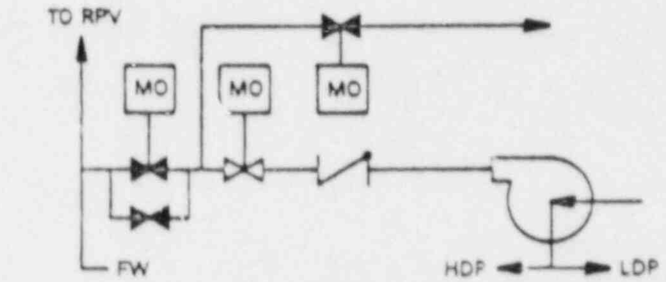


FIGURE A-30

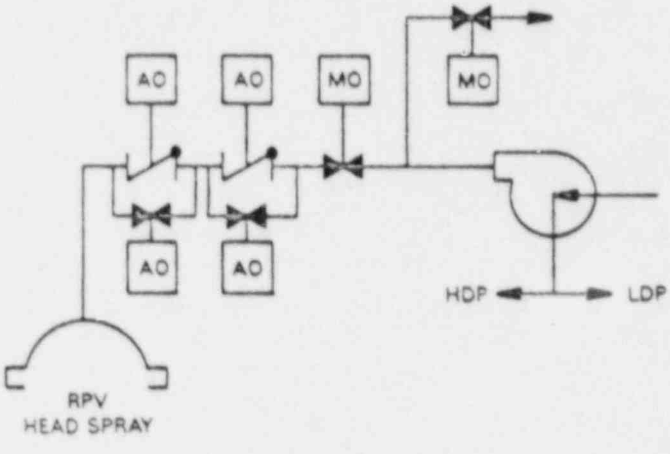


FIGURE A-31

APPENDIX B

DESCRIPTION OF LAWRENCE LIVERMORE NATIONAL LABORATORY (LLNL) PIPING RELIABILITY MODEL AND GENERAL ELECTRIC MODIFICATIONS

B.1 GENERAL

Figure B-1 provides a representation of the Lawrence Livermore National Laboratory (LLNL) piping reliability model (B.13). An initial population of circumferential cracks is considered. The depths and lengths of these initial cracks are described by appropriate probability distributions. The initial cracks are expected to be detected with certain probability during the pre-service and/or in-service inspections. Cracks that escape detection and repair are modeled to grow subcritically due to fatigue and/or stress corrosion mechanisms.

The crack growth calculation is based on the stress history induced by the normal and abnormal operating transients, earthquakes, and other cyclic loadings. The parameters defining the assumed fatigue and stress corrosion crack growth laws are characterized as normally distributed random variables.

The critical crack size is determined using a net section stress criterion in which the material flow stress is assumed to be a random variable. The probability of failure of a pipe is the probability of a crack growing to the corresponding critical size. The model also includes the influence of a leak detection system and a hydrostatic proof test.

B.2 PROBABILITY DISTRIBUTIONS AND FAILURE CRITERION

The probability distributions and the values of the parameters used in the LLNL model, as they pertain to carbon steel piping, are described below.

B.2.1 Initial Crack Size Distribution

The specification of circumferential surface crack geometry requires two parameters (see Figure B-2): depth (a) and surface length (ℓ). Therefore, the initial crack size distribution is characterized by a bivariate distribution.

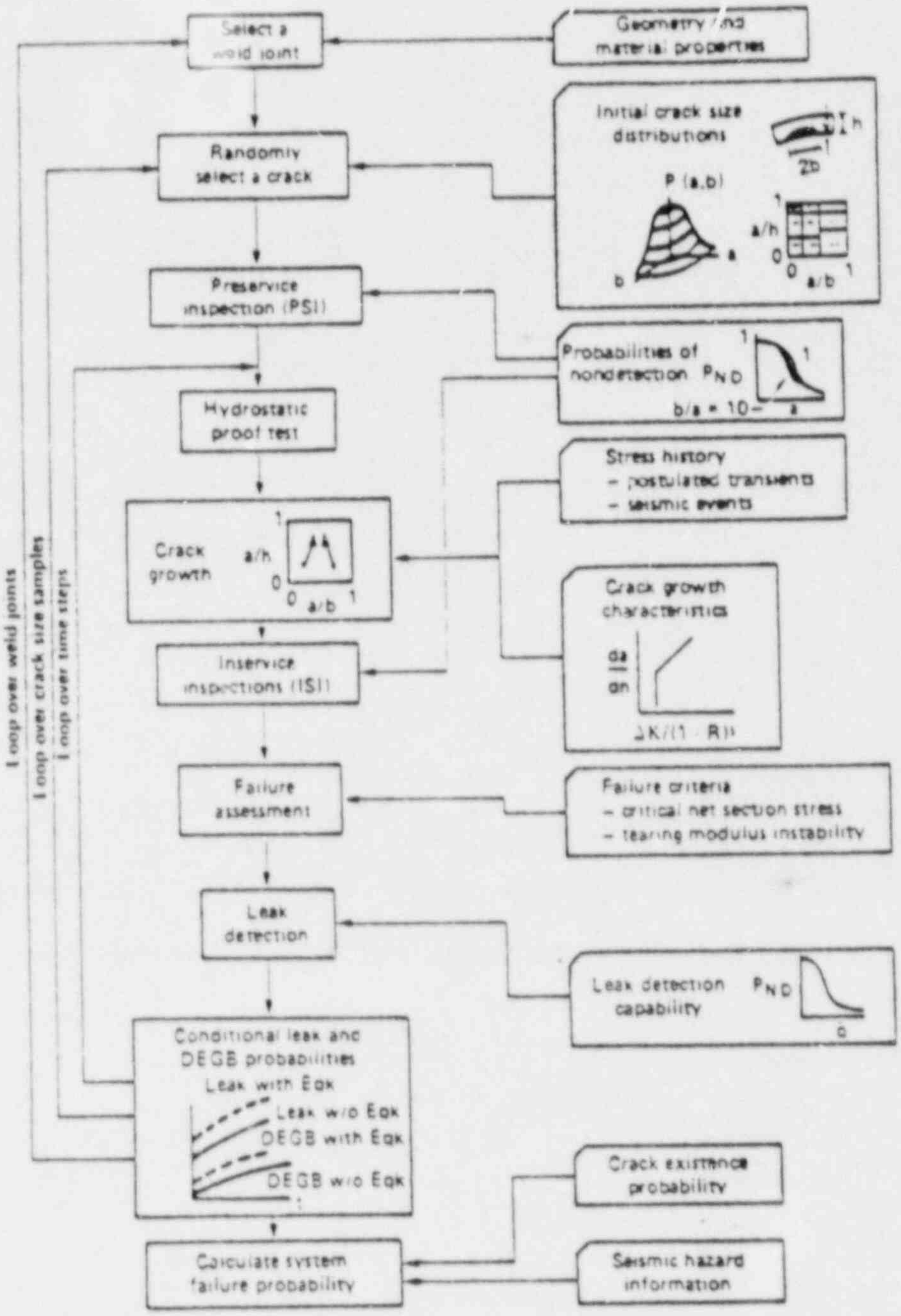


Figure B-1. Flowchart of Probabilistic Fracture Mechanics Model Implemented in PRAISE Computer Code

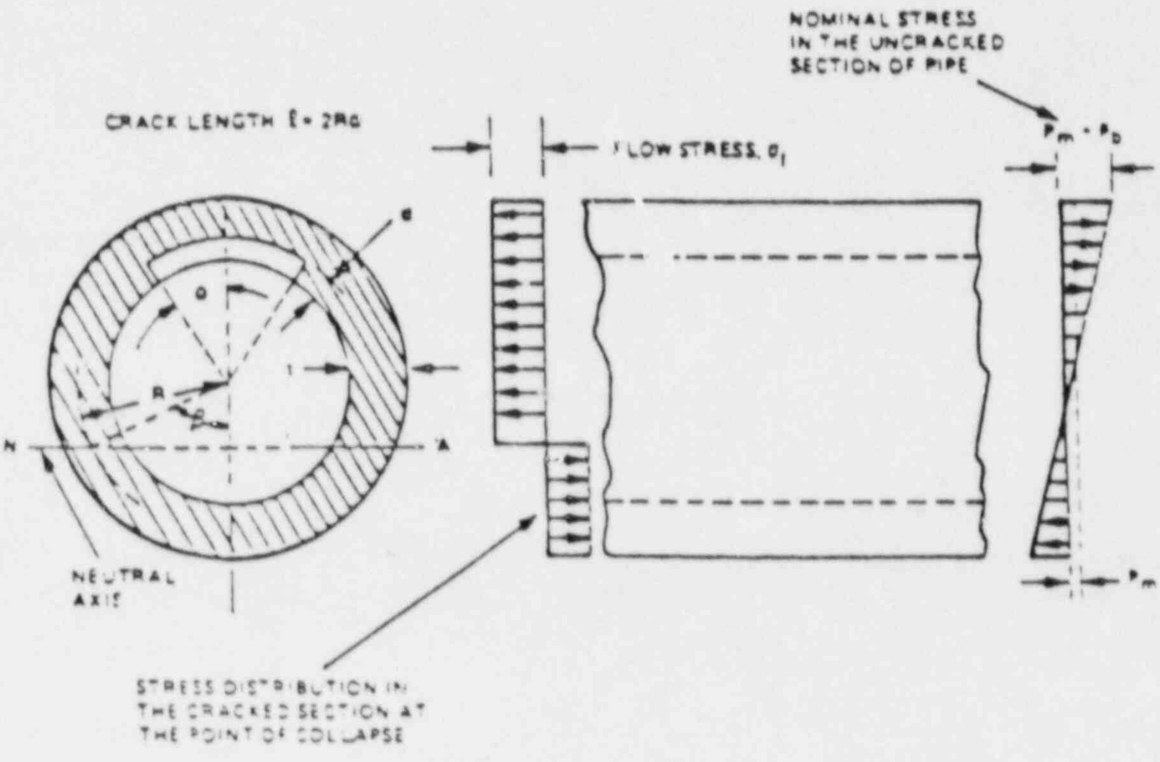


Figure B-2 Schematic Showing Stress Distribution at the Cracked Location in a Pipe at Limit Load

The crack depth is expressed by the exponential distribution:

$$P_a(a) = \frac{e^{-a/\mu}}{\mu(1-e^{-h/\mu})} \qquad \text{(B-1)}$$

where h is the wall thickness and $\mu = 0.246$ in., a constant. An option to specify lognormal distribution is also provided in the PRAISE code.

The initial crack length is expressed in terms of an aspect ratio (β) equal to $\ell/2a$. A modified lognormal probability density function is assumed for β:

$$P_\beta(\beta) = \begin{cases} 0 & \beta < 1 \\ \dfrac{C_\beta}{\lambda\beta(2\pi)^{\frac{1}{2}}} \, e^{-(\ln \frac{\beta}{\beta_m})^2/(2\lambda^2)} & \beta \geq 1 \end{cases} \qquad \text{(B-2)}$$

where constants $\beta_m = 1.336$, $C_\beta = 1.419$ and $\lambda = 0.5382$

B.2.2 Crack Existence Probability

It is assumed that an initial crack is produced by a fabrication process such as welding. Therefore, the crack existence probability is related to the volume of weld and heat-affected zone, $V = 2\pi Dh^2$, where D is the inside diameter and h is the pipe thickness. The probability of having an initial crack in weld volume (V) is estimated as:

$$P(1) = VP_{v*}\, e^{-VP_{v*}} \qquad \text{(B-3)}$$

where P_{v*} is the rate of cracks per unit volume.

B.2.3 Crack Detection Probability

The probability of not detecting a flaw (P_{ND}) is a function of the flaw size, the material inspected, and the instrumentation characteristics. For the case of wrought austenitic stainless steel and flaws that are long (i.e., $\xi \geq 5$), the following formula is suggested:

$$P_{ND}(a) = 0.5 \text{ erfc } (\nu \ln \frac{a}{a^*}) \quad \text{(B-4)}$$

where a* and ν are constants and erfc is the complementary error function.

B.2.4 Fatigue Crack Growth Parameters

The fatigue crack growth is characterized using the equations:

$$\frac{da}{dN} = C\left[\frac{\Delta K}{(1-R)^{1/2}}\right]^m \quad \text{(B-5)}$$

where:

ΔK = range of applied stress intensity factor (ksi $\sqrt{\text{in.}}$)

R = load ratio defined as K_{min}/K_{max} in which K_{min} and K_{max} represent the minimum and maximum stress intensity factors, respectively.

a = crack length (in.)

N = number of cycles

C,m = constants related to the material and environment.

The exponent, m, is treated as a constant, while the constant C can be specified as a constant or as a random variable with specified median and 90th percentile values.

B.2.5 Failure Criterion

A net section criterion is used. This criterion is stated as follows:

$$\sigma_{LC} A_p = \sigma_o (A_p - A_{crack}) \tag{B-6}$$

where:

σ_{LC} = axial component of the load controlled stress.

A_p = metal cross-sectional area of the pipe.

A_{crack} = cross-sectional area of the crack.

σ_o = material flow stress, which is a function of the yield and ultimate stress.

The load control stress, σ_{LC}, includes the longitudinal membrane stress due to pressure, and the membrane and bending stresses due to weight and seismic inertia loadings.

B.2.6 Leak Rate Calculations

The leak rate calculations are based on an assumed single-phase flow model of initially subcooled liquids through narrow passages and is described in Volume 5 of Reference B.1. The mathematical relationship between crack length and leak rate built into the PRAISE code is based on typical PWR coolant loop pressure and temperature conditions.

B.2.7 Numerical Simulation

Because many parameters are treated as random variables in the LLNL study, the Monte Carlo Simulation technique is used to evaluate the leak and double-ended guillotine break (DEGB) probabilities. A stratified sampling scheme is used to increase the accuracy and computational efficiency of the numerical simulation.

B.3 MODIFICATIONS INCORPORATED BY GENERAL ELECTRIC

The modifications made to the PRAISE code by General Electric are in the areas of failure criteria and leak rate evaluation.

B.3.1 Improved Failure Criteria

An examination of the net-section failure criterion, as represented by Equation B-6, indicates that it does not include induced bending at the cracked section due to the eccentricity produced by the presence of the crack. The following limit load equations developed in Reference B.2 include this effect:

$$\beta = \frac{(\pi-\alpha d/t) - (P_m/\sigma_f)\pi}{2} \tag{B-7}$$

$$P = \frac{2\sigma_f}{\pi}\,(2 \sin \beta - d/t \sin \alpha) \tag{B-8}$$

where:

- t = pipe thickness
- α = half crack angle as shown in Figure B-2
- β = angle that defines location of neutral axis
- σ_f = material flow stress
- P_m = applied membrane stress in the uncracked section
- P_b = applied bending stress in the uncracked section

The resulting failure criterion line is schematically shown in Figure B-3. The key input in Equations B-6 and B-7 is the material flow stress, σ_f. The lower the assumed value of flow stress, the more conservative (i.e., failure predicted at smaller crack depth and shorter crack length for the same stress level) the resulting failure criterion. Therefore, a requisite level of conservatism can be assumed in the failure criterion by selecting an appropriate value of flow stress.

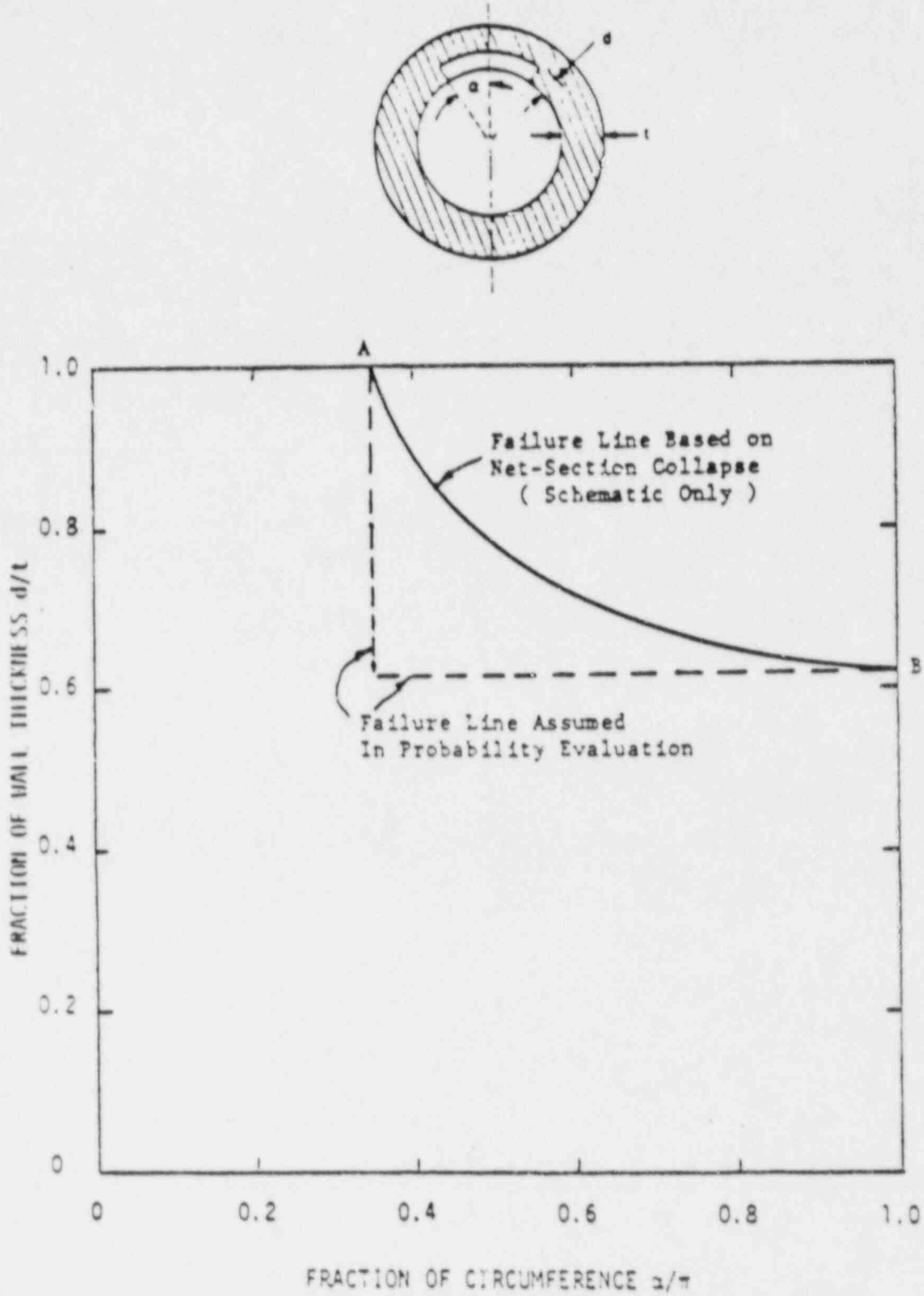


Figure B-3 Schematic Illustration of Failure Criterion Used in Piping Rupture Probability Evaluation

Since a large number of repetitive evaluations of failure criteria under the Monte Carlo Simulation scheme are involved in the PRAISE code, the failure criterion line schematically shown in Figure B-3 was simplified as shown by the dotted line in the same figure. A key advantage of this simplification is that it can also be used with the methodologies that provide information on the failure flaw sizes only at the ends of the failure line (points A and B in Figure B-3). For example, the elastic-plastic fracture mechanics based methods can readily treat only the flaws that are either through-wall or 360° part-through.

B.3.2 Leak Rate Prediction

The leak rate model currently built in the PRAISE code corresponds to PWR main coolant loop conditions. Therefore, the code was modified to accept an externally supplied crack length versus leak rate relationship. This provides complete flexibility to the user in the selection of appropriate leak rate calculation model.

B.4 REFERENCES

B.1 "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant", NUREG/CR-2189, Vol. 1 through 9.

B.2 Ranganath, S. and Mehta, H.S., "Engineering Methods for the Assessment of Ductile Fracture Margin in Nuclear Power Plant Piping", Elastic-Plastic Fracture: Second Symposium, Volume II-Fracture Resistance Curves and Engineering Applications, ASTM STP 803, C.F. Shih and J.P. Gudas, Eds., American Society for Testing and Materials, 1983, pp. II-309-II-330.

APPENDIX C

BEHAVIOR OF FLANGED JOINTS DURING PRESSURIZATION

C.1 GENERAL

Bolted connections, whose failure during an ECCS overpressurization event could lead to a large leakage area, are a part of the valves and heat exchanger assemblies on the low-pressure side of the ECCS piping systems. The questions that need to be addressed in the evaluation of the integrity of such bolted connections during the overpressurization are: (1) what is the maximum stress in the bolt during this event, and (2) what is the impact of the failure of one or two bolts in the overall integrity of the bolted connection? This appendix briefly reviews the theoretical background and the necessary equations to address these questions.

C.2 BOLT STRESS

The bolt stress in a typical flanged joint is primarily a function of the preload and depends, to a lesser extent, on the applied load and the relative stiffness of the bolt and the flange. In most flanged joints, the major stress applied to the bolts is the preload that is applied in tightening the nuts. Reference C.1 states that the following empirical formula provides a fair estimate of bolt stress, S_i, due to preload:

$$S = \frac{45000}{\sqrt{d}} \qquad \text{(C-1)}$$

where d is nominal bolt diameter in inches.

The total stress, S_T, in the bolt is given by Equation C-2:

$$S_T = S_i + \frac{F}{A_{net}} \cdot \frac{K_b}{K_f + K_b} \qquad \text{(C-2)}$$

where:

F = applied load/bolt

A_{net} = net cross-sectional area of bolt

K_b = bolt stiffness

K_f = flange stiffness

Typically, the flange stiffness is eight times the bolt stiffness (Equation C-3).

Accordingly,

$$S_T = S_i + \frac{F}{A_{net}} \cdot \frac{1}{9} \qquad \text{(C-3)}$$

Now consider an example where the bolt diameter is 1 inch. Equation C-1 would indicate a bolt pre-stress of 45,000 psi for this case. Assume that the applied load/bolt is such that F/A_{net} is 36,000 psi. This means that the applied load would produce a stress of 36,000 psi in a bolt with no preload. On the other hand, the preloaded stress in the bolt based on Equation C-3 would only increase from 45,000 psi to 49,000 psi. This represents an increase in bolt stress of only ≈9%. This confirms that, when a flanged joint is subjected to pressurization, the bolts experience only a small increase in the stress over and above that induced by the preload.

When the applied load or the flanged joint is such that the bolt preload is completely overcome, the bolt stress is then simply given by

$$S_T = \frac{F}{A_{net}} \qquad \text{(C-4)}$$

This occurs when

$$F = (S_i \times A_{net}) \frac{(K_f + K_b)}{K_f} \qquad \text{(C-5)}$$

C.3 IMPACT OF BOLT DEGRADATION ON FLANGED JOINT INTEGRITY

A review of General Electric service experience data base on BWR pressure boundary materials indicated no reported incidents of degradation in carbon steel (SA193 B7) bolting used in ECCS piping system valves and heat exchangers. This was not surprising, since most of the factors (identified in Reference C-3), such as the presence of borated water, stress corrosion cracking and fatigue, are not likely to be associated with the operating conditions in the parts of the BWR ECCS systems being considered in this evaluation. Therefore, the only possible scenario for the failure of a bolt during the pressurization event is the following: an undetected defect or crack at the thread root exists such that the bolt failure does not result during the preloading but occurs during the small incremental loading during pressurization to 1050 psi. The analytical results presented in Reference C.4 are helpful in assessing the impact of failure of one or more bolts on the overall integrity of a bolted joint.

As a part of an effort to establish the leak-before-break margins in a steam generator manway closure, Reference C.4 reported the results of finite element analysis on the load shedding and redistribution characteristics of this bolted joint when the failure of a number of studs was modeled. Figure C-1 from Reference C.4 shows the load redistribution curve for three adjacent studs as a function of a number of failed contagious studs. It is seen that even if three contagious studs were to fail in this joint, the stresses in the nearest stud would only increase by 22%. This clearly illustrates the redundant nature of a bolted connection. Furthermore, failure of a number of studs would lead to leakage which is likely to be detected.

Based on the preceding discussion it is concluded that:

a. There are no inherent environmental or other mechanisms present which could cause degradation of bolting in the ECCS system valves and heat exchangers.

b. Highly redundant nature of bolted joints will result in leakage rather than failure in the unlikely event that one or two bolts were to fail.

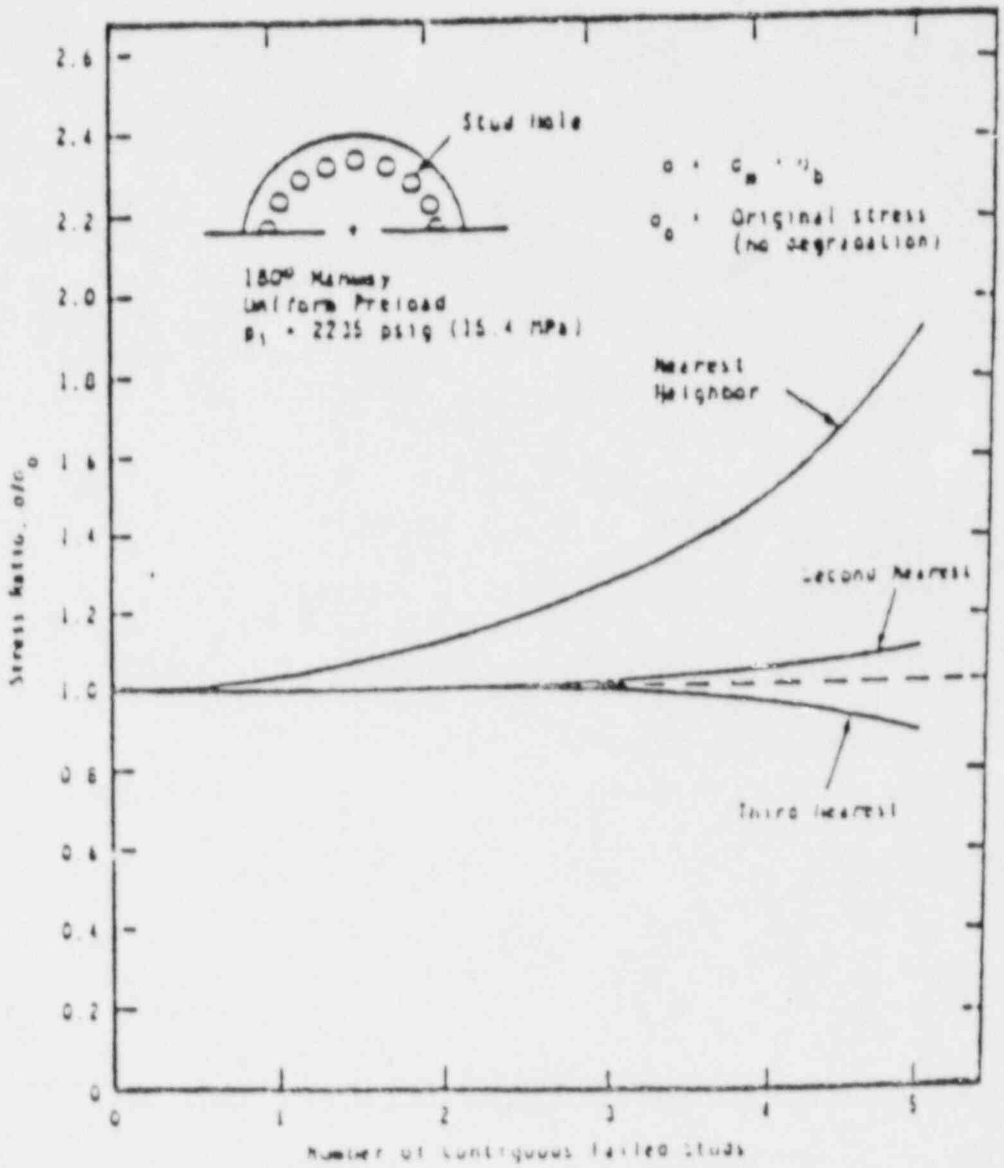


Figure C-1 Load Redistribution in the Three Nearest Studs due to Stud Degradation

C.4 REFERENCES

C.1 Rodabaugh, E.C., et al, "Survey Report on Structural Design of Piping Systems and Components", TID-25553, December 1970.

C.2 Shigley, J.E., "Mechanical Engineering Design", 2nd Edition, McGraw Hill Book Co., 1972.

C.3 NUREG-0933, "A Prioritization of Generic Safety Issues", Safety Program Evaluation Branch, Division of Safety Technology, U.S. NRC, New Generic Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants", December 1983.

C.4 Nickel, R.E., R.C. Cipolla and E.A. Merrick, "The Use of Leak-Before-Break Criteria and Assessment of Margins in Addressing Closure Integrity Issues", SMIRT-8, Paper D 6/3, August 1985.

APPENDIX D

SAMPLE PRAISE CODE RUN

```
     ECCS OVERPRESS. FAILURE PROBAB. 14 INCH STD SCH PIPE

     ---- CIRCUMFERENTIAL CRACK ANALYSIS ----

     TRANSDUCER DIAMETER = 1.00000

     ASTAR =      .350

     EPS = .00000

FATIGUE CRACK GROWTH ONLY

PIPE DIMENSIONS
          INSIDE RADIUS =       6.71  INCHES
          WALL THICKNESS          .30  INCHES

INITIAL CRACK SIZE DISTRIBUTION
CRACK DEPTH IS EXPONENTIAL
          PARAMETER =          4.0700
ASPECT RATIO IS LOG-NORMAL
          MEDIAN =             1.3360
          SHAPE PARAMETER       .5380
          NORMALIZATION CONSTANT =       1.4187

CRACK GROWTH LAW PARAMETERS
          EXPONENT =                     4.00
          GROWTH LAW CONSTANT IS CONSTANT
          CONSTANT =               .5300E-10
          THRESHOLD =                    4.00
          FLOW STRESS IS CONSTANT
          CONSTANT=                .3600E+02

CRITICAL CRACK SIZES ARE INPUT BY USER AS FOLLOWS:
          CRITICAL LENGTH (B) FOR PROOF TEST        13.19
          CRITICAL DEPTH (A) FOR PROOF TEST           .27
          CRITICAL LENGTH (B) FOR NORM. OPER.        6.70
          CRITICAL DEPTH (A) FOR NORM. OPER.          .17
          B & A FOR SEISMIC STRESS INPUT WITH EARTHQUAKE DATA

PIPE STRESS VALUES
          CSD DEADWT STRESS (KSI)    1.00
          NOT UNIFORM STRESS (KSI)  18.93
          PRESSURE STRESS (KSI)     11.93
          PROOF TEST STRESS(KSI)     3.56

LEAK DETECTION AND DEFINITION PARAMETERS
          DETECTABLE LEAK(GPM)          5.0
          BIG LEAK(GPM)                 5.0

THE CRACK-LENGTH-LEAK-RATE RELATION IS INPUT BY USER:
          TOTAL NUMBER OF DATA PAIRS =     10 :

           0.            0.                5.00000E-01  2.00000E-01
           1.50000E+00  2.40000E+00        2.50000E+00  9.60000E+00
           3.50000E+00  2.50000E+01        4.50000E+00  5.26000E+01
           5.50000E+00  1.02200E+02        6.50000E+00  1.82100E+02
           7.50000E+00  3.11300E+02        8.50000E+00  5.22200E+02

NO RESIDUAL STRESSES ARE MODELED
```

```
NO PRE-SERVICE ULTRASONIC INSPECTION

TIME INTERVALS
          PLANT LIFETIME        40.0  YEARS
          ENDPOINTS OF INTERVALS AT   0.0  10.0  20.0  30.0  40.0

NO IN-SERVICE INSPECTIONS ARE MODELED

                    - - - NO SEISMIC EVENTS EVALUATED - - -

SKIP PARAMETER FOR INDICATOR FUNCTION PRINTOUT IS     5

                    - - -  NORMAL OUTPUT REQUESTED - - -
```

```
   NUMBER OF TRANSIENT TYPES =    1

TYPE  1        HEAT UP FROM COLD SHUTDOWN AND BACK
STOCHASTIC WITH MEAN TIME          .200  YRS . VENT
MAX DELTA TEMP =480.0  BLOCKING FACTOR =    1
```

--- DEFINITIONS OF CELLS IN STATE SPACE ---

*** USER SPECIFIED MESH ***

CELL	AOH1	AOH2	AOB1	AOB2	PROBABILITY	SAMPLES
1	.9800	1.0000	.0137	.0500	.3644295E-08	1000
2	.9800	1.0000	.0500	.1000	.1354566E-05	100
3	.9800	1.0000	.1000	.2000	.1037903E-03	100
4	.9800	1.0000	.2000	.4000	.1707275E-02	100
5	.9800	1.0000	.4000	.6000	.3243403E-02	100
6	.9800	1.0000	.6000	1.0000	.5409612E-02	100
7	.9400	.9800	.0132	.0500	.7556451E-08	1000
8	.9400	.9800	.0500	.1000	.2808694E-05	100
9	.9400	.9800	.1000	.2000	.2152095E-03	100
10	.9400	.9800	.2000	.4000	.3540038E-02	100
11	.9400	.9800	.4000	.6000	.6725201E-02	100
12	.9400	.9800	.6000	1.0000	.1121684E-01	100
13	.9000	.9400	.0126	.0500	.7928214E-08	1000
14	.9000	.9400	.0500	.1000	.2946076E-05	100
15	.9000	.9400	.1000	.2000	.2257973E-03	100
16	.8000	.9000	.0112	.0500	.2156925E-07	1000
17	.8000	.9000	.0500	.1000	.8017176E-05	100
18	.8000	.9000	.1000	.2000	.6142970E-03	100
19	.7000	.8000	.0098	.0500	.2432084E-07	1000
20	.7000	.8000	.0500	.1000	.9039928E-05	100
21	.5000	.7000	.0070	.0500	.5834532E-07	1000
22	.5000	.7000	.0500	.1000	.2168665E-04	100
23	.5000	.8000	.1000	.2000	.2354351E-02	100

SUM OF CELL PROBABILITIES = .3540178955E-01

- - FAILURE DATA WITHOUT EARTHQUAKES - -

CELL	TIME	SUM LEAK	SUM BIG LEAK	SUM LOCA	SUM2 LEAK	SUM2 BIG LEAK	SUM2 LOCA
1	0.0	.81400E+03	.81400E+03	.50400E+03	.81400E+03	.81400E+03	.50400E+03
2	0.0	.60000E+02	.60000E+02	0.	.60000E+02	.60000E+02	0.
3	0.0	.39000E+02	.90000E+01	0.	.39000E+02	.90000E+01	0.
4	0.0	.17000E+02	0.	0.	.17000E+02	0.	0.
5	0.0	.12000E+02	0.	0.	.12000E+02	0.	0.
6	0.0	.12000E+02	0.	0.	.12000E+02	0.	0.
7	0.0	.35900E+03	.35900E+03	.35900E+03	.35900E+03	.35900E+03	.35900E+03
8	0.0	0.	0.	0.	0.	0.	0.
9	0.0	0.	0.	0.	0.	0.	0.
10	0.0	0.	0.	0.	0.	0.	0.
11	0.0	0.	0.	0.	0.	0.	0.
12	0.0	0.	0.	0.	0.	0.	0.
13	0.0	.21400E+03	.21400E+03	.21400E+03	.21400E+03	.21400E+03	.21400E+03
14	0.0	0.	0.	0.	0.	0.	0.
15	0.0	0.	0.	0.	0.	0.	0.
16	0.0	.11100E+03	.11100E+03	.11100E+03	.11100E+03	.11100E+03	.11100E+03
17	0.0	0.	0.	0.	0.	0.	0.
18	0.0	0.	0.	0.	0.	0.	0.
19	0.0	.25000E+02	.25000E+02	.25000E+02	.25000E+02	.25000E+02	.25000E+02
20	0.0	0.	0.	0.	0.	0.	0.
21	0.0	0.	0.	0.	0.	0.	0.
22	0.0	0.	0.	0.	0.	0.	0.
23	0.0	0.	0.	0.	0.	0.	0.

- - INDICATOR FUNCTIONS WITHOUT EARTHQUAKES - -

CELL	TIME	SUM LEAK	SUM BIG LEAK	SUM LOCA	SUM2 LEAK	SUM2 BIG LEAK	SUM2 LOCA
1	0.0	.81400E+03	.81400E+03	.50400E+03	.81400E+03	.81400E+03	.50400E+03
2	0.0	.60000E+02	.60000E+02	0.	.60000E+02	.60000E+02	0.
3	0.0	.39000E+02	.90000E+01	0.	.39000E+02	.90000E+01	0.
4	0.0	.17000E+02	0.	0.	.17000E+02	0.	0.
5	0.0	.12000E+02	0.	0.	.12000E+02	0.	0.
6	0.0	.12000E+02	0.	0.	.12000E+02	0.	0.
7	0.0	.35900E+03	.35900E+03	.35900E+03	.35900E+03	.35900E+03	.35900E+03
8	0.0	0.	0.	0.	0.	0.	0.
9	0.0	0.	0.	0.	0.	0.	0.
10	0.0	0.	0.	0.	0.	0.	0.
11	0.0	0.	0.	0.	0.	0.	0.
12	0.0	0.	0.	0.	0.	0.	0.
13	0.0	.21400E+03	.21400E+03	.21400E+03	.21400E+03	.21400E+03	.21400E+03
14	0.0	0.	0.	0.	0.	0.	0.
15	0.0	0.	0.	0.	0.	0.	0.
16	0.0	.11100E+03	.11100E+03	.11100E+03	.11100E+03	.11100E+03	.11100E+03
17	0.0	0.	0.	0.	0.	0.	0.
18	0.0	0.	0.	0.	0.	0.	0.
19	0.0	.25000E+02	.25000E+02	.25000E+02	.25000E+02	.25000E+02	.25000E+02
20	0.0	0.	0.	0.	0.	0.	0.
21	0.0	0.	0.	0.	0.	0.	0.
22	0.0	0.	0.	0.	0.	0.	0.
23	0.0	0.	0.	0.	0.	0.	0.

- - - ECCS OVERPRESS FAILURE PROBAB 14 INCH STD SCH PIPE

- - RESULTS WITHOUT EARTHQUAKES - - -

TIME	AVG LEAK	AVG BIG LEAK	AVG LOCA	SIGMA LEAK	SIGMA BIG LEAK	SIGMA LOCA
0.0	1.36990E-03	1.01642E-05	9.24834E-09	2.15907E-04	2.99600E-06	2.95714E-10
10.0	8.42761E-03	7.80782E-05	9.26382E-09	2.02539E-04	9.88486E-06	2.95801E-10
20.0	1.10444E-02	9.21290E-05	9.27968E-09	1.98735E-04	1.06245E-05	2.95922E-10
30.0	1.32535E-02	1.03009E-04	9.27968E-09	4.04762E-04	1.14233E-05	2.95922E-10
40.0	1.46834E-02	1.19464E-04	9.27968E-09	4.99530E-04	1.27219E-05	2.95922E-10

APPENDIX E

PARTICIPATING UTILITIES - BWR OWNERS' GROUP
ECCS PRESSURIZATION COMMITTEE

APPENDIX E

PARTICIPATING UTILITIES - BWR OWNERS' GROUP ECCS PRESSURIZATION COMMITTEE

This report applies to the following plants, whose owners participated in the report's development:

BWR OWNER	PLANT
Boston Edison Company	Pilgrim
Carolina Power & Light Company	Brunswick 1 & 2
Commonwealth Edison Company	Dresden 2 & 3
	Quad Cities 1 & 2
	LaSalle 1 & 2
Detroit Edison Company	Fermi 2
Georgia Power Company	Hatch 1 & 2
General Public Utilities Nuclear	Oyster Creek
Gulf States Utilities	River Bend 1
Illinois Power Company	Clinton 1
Iowa Electric Light & Power Company	Duane Arnold
Nebraska Public Power District	Cooper
Niagara Mohawk Power Corporation	Nine Mile Point 1 & 2
Northern States Power Company	Monticello
Pennsylvania Power & Light Company	Susquehanna 1 & 2
Philadelphia Electric Company	Peach Bottom 2 & 3
	Limerick 1 & 2
Public Service Electric & Gas Company	Hope Creek 1
Tennessee Valley Authority	Browns Ferry 1, 2 & 3
Washington Public Power Supply System	Hanford 2

ATTACHMENT 3

October 1987 Appendix R Audit
Open Items Not Requiring NRR Review

1. Fire Damper Operability (Unresolved Issue 84-40-01, 84-19-01 discussed on page 4 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30.)

NRC Comment:

The licensee's Technical Specifications require that fire dampers be inspected visually. The NRC raised the concern that a visual damper inspection does not provide assuranc that the fire dampers will be able to function properly during a fire. This concern was raised because: A) The licensee could not provide Q.C. records indicating that the fire dampers were drop tested after installation, as called for in the engineering packages; and, B) a recently issued 10CFR21 letter highlighted the concern that the type of fire dampers used by the licensee may not close under air flow conditions.

The licensee addressed this concern by revising the fire fighting strategy procedures giving the fire brigade the option to de-energize the ventilation systems involved. With no air flow presumably the fire dampers will close. The licensee's actions did not satisfy the original NRC concern for the following reasons:

1. The inspector observed a fire brigade drill. Although an attempt was made to verify whether the fire jumped to areas above the hypothetical fire scene, no attempt was made to find and isolate the ventilation equipment.

2. Assuming that the brigade does turn off the air handling units there is no assurance that the dampers will fully close after the air handling units are turned off. This is because the dampers may drop and bind in a partially open position before the air flow is cut-off. To assure that the dampers close, the licensee must provide assurance that the dampers will close under air flow or that the air handling units are de-energized prior to dropping of the dampers.

This item continues to be unresolved. Considering the above concerns the inspector questioned the operability of the dampers.

Response:

A fire damper program has been formulated to evaluate existing test data and damper closure with air flow data and to address fire brigade and training procedures to provide reasonable assurance that the fire dampers will satisfactorily perform their design function. The evaluation program will be completed by August 1988.

2. Incorporation of NRC Comments on Procedures (Page 11 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30

NRC Comment:

Procedure SE-10 "Plant Shutdown from the Alternative Shutdown Panel" was reviewed and found to be adequate. However, the team commented that some steps in the procedure may need signature checks to assure control. For instance the steps monitoring the reactors' cooldown rate and other steps that operators perform in the attachments to the procedure do not have sign-off blocks that the operation was performed. The licensee in subsequent discussions committed to review the procedure and add sign-off spaces where needed.

Response:

Procedure SE-10 is currently being revised to reflect changes caused by the completion of Appendix R modifications. During this revision, operator, training, and NRC comments were reviewed and incorporated into the procedure. In addressing NRC comments, sign-off spaces have been added where needed, and the monitoring of the reactor cooldown rate has been enhanced.

3. Accessibility of HPCI Inboard Steam Isolation Valve Panel (Page 11 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30)

NRC Comment:

During the walkdown of procedure SE-10, it was observed that the breaker panel for the inboard steam isolation valve of the HPCI system has a cover fastened on with wing nuts. The team observed that if the wing nuts are too tight the operators may not be able to open the panel. The licensee stated that either bigger wing nuts or a tool will be provided to assure panel access.

Response:

The HPCI Inboard Steam Isolation Valve panel was originally provided with slotted screws which required tools for access. The slotted screws were changed to thumbscrews to allow an operator to access the panel without the use of tools. To address the NRC concern of overtightening, flat washers were added to compliment the thumbscrews. The washer addition will provide a smooth contact surface and enhance the operator's ability to loosen a tight thumbscrew.

4. <u>Fuse Replacement Controls</u> (Page 13 in Inspection Report Nos. 50-277/87-30 and 50-278/87-30)

NRC Comment:

During the review of the licensees circuit coordination study it was identified that the licensee does not have administrative control procedures in place to control future fuse replacement activities. The licensee stated that a procedure to control fuse replacement is currently in the process of being written and implemented. The licensee further explained that fuse replacement is currently performed by either "replace in kind" or using the Control Room mark up drawings which call for the type of fuses to be used.

Response:

The following administrative controls for fuse replacement are being initiated. A modification has been started by the Nuclear Engineering Department to generate a controlling document for fuse replacements. Additionally, a guideline document will be added to the watch standards guide to assist the operators in the practice of fuse replacement. The guide will be revised by December 1988 to reflect this practice.