ENCLOSURE 1

Evaluation of Proposed Changes

INTRODUCTION

The Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) system is part of the Emergency Core Cooling System (ECCS) network. LPCI acts, in conjunction with the Core Spray system, to provide a post loss of coolant accident (LOCA) source of cooling water when the Reactor Pressure Vessel is at low pressure conditions.

In performing this function, LPCI piping must penetrate the Primary Containment and Reactor Coolant Pressure Boundary. In accordance with Appendix A to 10CFR50, General Design Criteria (GDC) 54 and 55 primary containment isolation valves are provided. These valves are also included in the containment leakage rate testing program required by Appendix J to 10CFR50.

On each of two injection lines, three valves are currently specified as Containment Isolation Valves in Technical Specification (TS) Table 3.6.3-1, Primary Containment Isolation Valves. Outboard of containment is installed a motor operated gate valve, E11-F015. E11-F015 is remotely operated from the Control Room and is installed in the 24" injection line which connects to the Reactor Coolant System in the Reactor Recirculation Loop.

Also installed on the 24" injection line inside containment is a check valve, E11-F050. E11-F050 is installed to prevent reverse flow out of containment through the injection line. A 1-inch bypass line is installed around E11-F050 for the purpose of system warmup for the RHR - Shutdown Cooling Mode. A solenoid operated globe valve, E11-F610, is normally closed except during system warmup and is remotely operated from the control room.

The two divisional penetrations are designated as primary containment penetrations X-13A and X-13B. These penetrations are described in Fermi 2 UFSAR Table 6.2-2, "Summary of Primary Containment Penetrations and Associated Isolation Valves."

As Primary Containment Isolation Valves, these valves are subject to Type C Local Leak Rate Testing (LLRT) in accordance with Appendix J. In accordance with Surveillance 4.6.1.2.b this test is performed with gas at Pa, 56.5 psig.

E11-F015 and E11-F050 are also designated in Technical Specification Table 3.4.3.2-1, "Reactor Coolant System Pressure Isolation Valves," as Pressure Isolation Valves. As such, these valves form a boundary between the high pressure Reactor Coolant System piping and low pressure RHR piping.

The GDC 55 requires that each line that is part of the reactor coolant pressure boundary and that penetrates the primary containment be provided with containment isolation valves. The combination of valves is to be one inside and one outside containment and either automatic or locked closed. The outside containment valve may not be a simple check valve. The GDC does allow for a demonstration that acceptable containment isolation provisions have been provided on some other defined basis.

For penetration X-13, the use of a remote manual valve (E11-F015) was deemed acceptable since the RHR - LPCI mode is required for core cooling post-accident. The inside containment configuration meets the GDC explicitly since E11-F050 is automatically closed by reverse flow out of containment and E11-F610 is locked closed. No automatic containment isolation signal is used to isolate these valves due to the essential nature of the RHR - LPCI mode.

However, E11-F050 has repeatedly failed the Type C LLRT test requiring repair and retesting. Although this valve has been able to be repaired and successfully retested, the resultant radiological and safety concerns caused Detroit Edison to review the use of E11-F050 as a containment isolation valve under these circumstances.

E11-F050 is located within a hazardous area in the drywell. The area is also within a significant radiation field. For example, during the last refueling outage, 10.9 person-rem were expended in maintenance and testing on E11-F050 A and B. In addition, this type of work also extends the outage duration and increases the unavailability of one of the decay heat removal systems during the outage.

As a result of the review of the design basis for these penetrations, Detroit Edison has determined that there exists an alternative basis for meeting containment isolation provisions of the GDC 55 without the attendant valve testing problems associated with the current means of compliance with GDC 55. This alternative basis examines aspects of the RHR system which had not been included in previous evaluations. In addition, these aspects form a basis for changing the pressure isolation valve leakage test criteria. The new testing requirements will cause a reduction in occupational radiation exposure and improvements in industrial safety while continuing to assure containment integrity.

EVALUATION

The review of this penetration showed two aspects of the RHR system which had not been included in the previous evaluation of the plant's

conformance to GDC 55. One aspect is that, due to the system physical configuration and the LPCI Loop Selection feature of the RHR system, a water seal at a pressure sufficient to preclude containment atmospheric leakage will be present as a barrier to containment leakage. The second aspect is that the RHR system forms a closed system outside containment which is another barrier to containment leakage.

Detroit Edison has performed an engineering evaluation of the behavior of the water volume maintained between the inboard valves E11-F050 and E11-F610, and the outboard valve, E11-F015. The evaluation, which is summarized in Enclosure 2, concludes that a water seal will be maintained with the water volume to preclude containment atmospheric leakage through these penetrations. The water seal would exist for at least 30 days and is able to withstand a single active failure.

The second aspect of the RHR system design is that the system is a closed system outside containment. As such, the system can accommodate a single active failure and still maintain containment integrity. It is also protected against the effects of missiles and pipe whip. The system is designed to Category I standards, is classified as Quality Group B or better, and is designed to meet or exceed the maximum temperature and pressure of the containment.

The system is also included in the ASME Section XI In-Service Inspection Program and receives the required non-destructive examinations for Class 2 piping. These programs require the system be inspected at pressure and any visible leakage be promptly repaired.

The RHR system is also subjected to the inspections required by the Fermi 2 System Leakage Reduction program. This program is a commitment made by Detroit Edison in response to NUREG-0737 to conduct inspections to reduce and maintain leakage to as low as practical levels from systems outside of the primary containment that could or would contain highly radioactive fluids during or after a severe transient or accident. The inspections are done on an 18 month frequency and the acceptance criteria for the RHR system is 40 ml/min. external leakage for each RHR division.

Overpressure protection on the RHR system is provided with a relief valves set at 450 psig with a capacity of approximately 290 gpm. The relief setpoint is set sufficiently above the containment design pressure. Standard Review Plan (NUREG-0800) Section 6.2.4 paragraph II.6.g allows for the use of relief valves provided the setpoint is at least 150 percent of the containment design pressure. A pressure of 93 psig corresponds to 150 percent of the containment design pressure for Fermi 2.

The piping from the inboard check valves E1100-F050A and B to valves E1150-F015A and B conforms to ASME Section III, Class I requirements. Since this piping fulfills the design requirements stipulated in NRC Branch Technical Position MEB 3-1 no pipe breaks or cracks are postulated.

The RHR system is currently considered an acceptable closed system outside containment for the purposes of meeting GDC 56 for penetrations X-223A through D. These penetrations are for the RHR pump suction from the suppression pool.

Detroit Edison is proposing to eliminate Type C LLRT testing requirements from E11-F050A and B and E11-F610A and B by removing the valves' designation as containment isolation valves. However, these valves will remain in place and will be tested as High/Low Pressure Isolation Valves in accordance with Specification 4.4.3.2.2.

Thus, the containment boundaries being utilized to conform through an alternative basis with GDC 55 are: the E11-F015 valve, the water seal provided inside E11-F015, and the closed system outside containment. These boundaries achieve sufficient redundant containment isolation capability to guard against offsite releases and, therefore, achieve the purpose of GDC 55.

Appendix J to 10CFR50 provides requirements for leak rate testing of the primary containment. Individual containment penetrations undergo what is termed a Type C Local Leak Rate Test (LLRT). Valve leakage can be categorized as either through the valve seat or external to the valve (such as a stem or bonnet leak).

For E11-F015, the operation of the RHR pumps, as further discussed in Enclosure 2, assures that through seat leakage is inwards towards the containment. External leakage remains of concern. Detroit Edison is proposing that acceptable external leakage be verified during the Pressure Isolation Valve (PIV) leakage testing performed in accordance with Specification 4.4.3.2.2. External leakage is proposed to be limited to 5 ml/min. This is consistent with the NUREG-0737 System Leakage Reduction Program previously described.

The 5 ml/min. external leakage criteria comes from the Enclosure 2 evaluation which determined that a 30-day water seal would be maintained. If no leakage is assumed into the water seal volume then 5 ml/min. is maximum leakage out of the volume that can be permitted.

PIV testing is performed at least every 18 months at 1045 psig which is much greater than any accident pressure. However, this is conservative since higher pressures tend to yield a greater external

leakage. The PIV test uses water as a test medium. This is acceptable since the water seal assures that gaseous leakage does not occur.

Since the PIV test will conservatively determine the acceptability of E11-F015 for containment isolation purposes, Detroit Edison is proposing to replace the normally required Type C LLRT with the described PIV test. This replacement requires a specific exemption to the requirements of Appendix J. The proposed exemption is specifically addressed in a later section of this enclosure.

PIV testing will still be required for E11-F050. E11-F610 will also be tested for PIV purposes. The system configuration is such that the two valves must be tested together with the total leakage required to be less than the acceptance criteria.

The current acceptance criteria for PIV testing is 1 gpm. This value is specified for all PIVs regardless of application. In order to avoid maintenance under hazardous conditions due to an overly strict acceptance criteria, Detroit Edison reviewed the 1 gpm acceptance criteria for E11-F050 in view of the nature of the RHR system as a closed system.

The Fermi 2 UFSAR Section 5.5.7.3.5 evaluates the provisions for overpressure protection of the RHR system. This evaluation is based upon a PIV leakage test acceptance criteria of 10 gpm. This is the limit proposed for E11-F050. The low pressure RHR pipe is protected from high pressure by the pressure relief valves with a relief capacity of 290 GPM at 450 psig. Therefore, the proposed 10 GPM leakage will be far below the relief capacity and will prevent the overpressurizing of the low pressure RHR pipe. The relief valves discharge to within primary containment.

For E11-F015 A and B, the 10 gpm UFSAR acceptance criteria and the current TS criteria of 1 gpm are being reduced to 0.4 gpm. This value is consistent with the assumptions used in the Enclosure 2 evaluation which determined that a water seal would remain inboard of E11-F015 for 30 days following an accident.

In summary, the proposed Technical Specification change:

- Eliminates the designation of E11-F050 A and B and E11-F610 A and B as containment isolation valves. They will retain their PIV function.

- Changes the containment leakage test methodology for E11-F015 A and B. These valves will be tested with water at 1045 ± 10 psig for external leakage only.

- Increases the allowable PIV test leakage to 10 gpm from 1 gpm for E11-F050, while reducing the allowable leakage for E11-F015 to 0.4 gpm.

The TS changes are based upon an alternative basis for meeting GDC 55 which takes into account the plant physical configuration and RHR system design.

The change allows a substantial reduction of occupational radiation exposure and avoids the risk of substantial valve rework in a hazardous area. While doing so, the change maintains an adequate assurance of primary containment integrity and that protection is maintained against off-site radiation releases in the event of an accident.

EXEMPTION TO APPENDIX J TO 10CFR50

In accordance with 10CFR50.12, the NRC may grant exemptions from the requirements contained in 10CFR50 under special circumstances. 10CFR50.12(a)(2) defines special circumstances. One situation considered a special circumstance is when application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The proposed testing requirements for E11-F015 is such a case where the underlying purpose of Appendix J is met in an alternative manner.

Appendix J requires in this case a direct pressure test in the accident direction at not less than 1.1 Pa, 62.2 psig, using water as a test medium. The use of water as a test medium would be based on reclassifying the valves as valves sealed with a fluid system as described in this proposal.

The through-seat leakage and external leakage would be determined and compared to a specific acceptance criteria contained in the Technical Specifications.

However, due to the RHR system configuration and the operation of the RHR pumps during an accident, the Appendix J testing requirements are not necessary to assure primary containment leakage is minimized during an accident. The alternative testing criteria is the Pressure Isolation Valve (PIV) hydrostatic test at 1045 ± 10 psig specified in

TS surveillance 4.3.3.2.2. During this PIV test external leakage would be verified to be less than 10 ml/min.

Due to the loop seal type piping configuration between E11-F050 and E11-F015 there will always be a volume of water at the inboard side of E11-F015. Additionally, due to the open cross-tie valve, E11-F010, in conjunction with the operation of one or more of the RHR pumps for 30 days or more, RHR pump pressure will be felt at the inlet of E11-F015 valves. Details of the evaluation which leads to these conclusions is included as Enclosure 2.

The combination of these factors, assures that E11-F015 will be pressurized with water during an accident and that any through seat leakage for E11-F015 will be towards containment. Therefore, measurement of through seat leakage is unnecessary to determine primary containment leakage during an accident. An indication of the leak-tightness of the E11-F015 valve seating-surface will be determined during the PIV test, however, the higher test pressure is not necessarily indicative of accident conditions.

External valve leakage is potential primary containment leakage in an accident. This is proposed to be determined by the 1045 $\pm$ 10 psig PIV test. The test pressure is substantially higher than the Appendix J test pressure of 62.2 psig. However, the higher pressure will conservatively cause greater leakage and therefore the PIV test is an acceptable method of determining external leakage.

The proposed limit of 5 ml/min. external leakage criteria for E11-F015 is less than the NUREG-0737 system leakage criteria of 40 ml/min. for each division. This criteria was developed to minimize any leakage outside primary containment.

Based upon the above, Detroit Edison has concluded that the alternative testing of E11-F015 previously described meet the underlying purpose of Appendix A to 10CFR50 to minimize the radiological consequences to the public from primary containment leakage under accident conditions. As such, special circumstances exist, as defined in 10 CFR50.12(a)(2), and the proposed exemption should be granted.

SIGNIFICANT HAZARDS CONSIDERATION

In accordance with 10CFR50.92, Detroit Edison has made a determination that the proposed amendment involves no significant hazards considerations. To make this determination, Detroit Edison must establish that operation in accordance with the proposed amendment

would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated, or 2) create the possibility of a new or different kind of accident from any accident previously evaluated, or 3) involve a significant reduction in a margin of safety.

The proposed change modifies Containment Local Leak Rate Testing (LLRT) and Pressure Isolation Valve (PIV) testing requirement for valves installed in the Residual Heat Removal (RHR) system Low Pressure Coolant Injection (LPCI) mode injection line. The change does not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change does not change the physical plant or the manner in which it is operated. The change eliminates certain tests while retaining those necessary to assure that containment systems perform as intended. The new basis for meeting GDC 55 provides an equivalent level of protection against off-site radiation release. Therefore, the probability or consequences of any previously analyzed accident has not significantly increased.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a change in the plant or the manner in which it is operated. The change eliminates certain tests while retaining those necessary to assure that containment systems perform as intended. No new accident modes are created.

3) Involve a significant reduction in a margin of safety. The change eliminates unnecessary testing which has caused significant radiation exposure and safety hazards. All systems and structures continue to operate in the same manner as before. Sufficient protection against radiological releases continues to be maintained.

Based on the above, Detroit Edison has determined that the proposed amendment does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT

Detroit Edison has reviewed the proposed Technical Specification changes against the criteria of 10CFR51.22 for environmental considerations. The proposed change does not involve a significant hazards consideration, nor significantly change the types or significantly increase the amounts of effluents that may be released

offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed Technical Specifications do meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

CONCLUSION

Based on the evaluation above: 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and 2) such activities will be conducted in compliance with the Commission's regulations and proposed amendments will not be inimical to the common defense and security or to the health and safety of the public.

In order to make procedural changes associated with the proposed changes, a thirty-day period is requested for implementation purposes.

ENCLOSURE 2

Engineering Summary of Evaluation of Primary Containment Penetration X-13A and X-13B

Introduction

The purpose of this evaluation is to examine penetrations X-13A and B for the Low Pressure Coolant (LPCI) mode of Residual Heat Removal. The goal is to determine the conditions (e.g., air or water) at the penetrations for the 30 day accident period in order to establish appropriate test requirements.

Functional and Regulatory Requirements

In the unlikely event of a postulated plant accident, coolant makeup to the Reactor Pressure Vessel (RPV) is required. The LPCI mode of the RHR provides this coolant. In doing so, the primary containment boundary is penetrated at penetrations X-13A and B. An inboard E11-F050A(B), and an outboard E11-F015A(B) valve is provided on each of the LPCI lines. Therefore, as opposed to other systems such as main steam, where the inboard and outboard valves are closed following an accident, the primary purpose of these valves is to open and not close in an accident so that the water inventory in the RPV can be made up. Two valves are provided to assure redundancy of isolation function, if required. This increases isolation reliability should one valve fail to close.

The basis for the pressure at which the leak rate of these valves is measured is the highest possible pressure that the primary containment would see under the worst accident scenario. A containment analysis was done for Fermi 2 and submitted for NRC approval as part of the Power Uprate Program. The Power Uprate Program has been approved by the NRC in Amendment 87 to the Fermi 2 Operating License. This analysis shows a peak containment pressure of 49.9 psig for a design basis loss of coolant accident (LOCA), a guillotine break of the recirculation line. This is the highest pressure that the Fermi 2 containment will see for any of the design basis events.

Salient Features of the Fermi 2 RHR System

Refer to Figure 1 for the following discussion on some of the salient features of the RHR system.

1. Each division of RHR has two pumps each of which are fed from a separate diesel generator.

2. The two divisions for RHR are cross-connected by a single header through the E11-F010 valve which is normally kept open.

3. Valves E11-F017A & B, E11-F015A & B, E11-F010 and the two recirculation pump discharge valves are electrically fed from the swing bus. The swing bus design assures power availability to

each of the above valves in case one of the divisional power supplies are lost.

4. One or more of the RHR pumps will be run following an accident for either LPCI injection or suppression pool cooling for at least a period of 30 days. In some cases, both functions will carried out simultaneously. The LPCI cross-tie valve will be closed in these cases; however, RHR pumps will be operating in both divisions.

5. To prevent any water hammer, the piping inside and outside the containment is kept full of water. A Keep Fill System is provided outside the containment that maintains the water pressure in the RHR piping at approximately 100 psig. Should this water pressure drop, the operator is alerted by a control room alarm set at 79.3 psig.

6. The RHR is a closed loop system taking suction from either the RPV (through recirc line) or suppression pool and discharging to the RPV (through recirc line) or suppression pool. The entire piping is confined to the primary containment or secondary containment. At Fermi, an added precaution of measuring all leaks into the secondary containment from each of the two RHR loops is taken. The leaks through RHR system leak paths into secondary containment, such as, isolation valves, vent and drain valves, restricting orifices, check valves, RHR pumps, instrument rack check points, instrument source valves, spare instrument taps, flow elements, thermal relief valves, blind flanges, etc., are recorded once every 18 months and maintained below 40 ml/min per division. This assures a virtually leak free closed loop RHR system.

7. The inboard isolation valves E11-F050A and B are Anchor Darling 24" air operated swing check valves. The outboard isolation valves E11-F015A and B are 24" motor operated Powell flexible wedge gate valves. The inboard isolation bypass valves E11-F610 A and B are 1" solenoid operated valves.

Analysis

To evaluate the behavior of the water volume between valves E11-F050 and E11-F015 a limiting accident was used. The rupture of a recirculation line most directly affects the LPCI injection line since LPCI injects into the recirculation line. The rupture of the Division I was found to be more limiting due to the specific geometry of the piping. An additional failure of the Division II batteries was assumed. This gives the largest containment pressure and the greatest radiological consequences.

The postulated rupture of the recirculation line results in an instantaneous rise of drywell pressure. This would start the permissives for the Division I Diesels which energize the diesel bus in less than 13 seconds. Once the diesel buses are energized, the two Division I RHR pumps would start. The coolant injection to the RPV through E11-F015B valve happens in approximately 46 seconds based on RHR valve pressure permissive. Though the Division II Battery is lost, the swing bus design assures power to the E11-F010, E11-F017B, and E11-F015B valves of the RHR system.

Between the time when the RHR pumps have started (less than 13 seconds) and when LPCI injection happens, the RHR pumps run in the minimum flow mode taking suction from the suppression pool and discharging back to the suppression pool.

Following the LPCI injection (the Division I Core Spray would be injecting in the vessel also) the operator may switch one of the RHR pumps to the suppression pool cooling mode. This would happen at operator discretion, once the level in the vessel is restored, but not later than 20 minutes into the accident. This limits the suppression pool temperature rise so that the NPSH limit for the low pressure ECCS is not exceeded. Therefore, one or two of the RHR pumps will be in constant use either for suppression pool cooling or LPCI injection or both.

For the purpose of this analysis, the accident scenario has been divided into three different time periods:

A) 0-13 seconds: The containment sees a pressure rise from 0-49.9 psig in approximately 4-5 seconds, and then the pressure starts dropping. RHR pumps have not started at this time and both the LPCI injection valves are closed.

B) 13-46 seconds: RHR Pumps A and C have started and pressurize the RHR piping to approximately 115 psig at the inlet of valves E11-F015A and B at approximately 13 seconds plus. Valve E11-F017A starts closing at approximately 5 seconds based on LPCI loop select logic and is fully closed in approximately 29 seconds. Valve E11-F015B starts opening at approximately 18 seconds (based on pressure permissive of approximately 450 psig) and is fully open at approximately 42-46 seconds. During this period, the containment pressure is slowly declining.

C) 46 seconds-30 days: One or two RHR pumps (A, C) are running either for LPCI injection or suppression pool cooling. Ten minutes after the accident, the operator has the option to open valve E11-F017A.

Immediately preceding the accident, i.e., at time -0, the RHR lines are kept filled with water that is maintained at an approximate pressure of 100 psig using a pressure control valve in the condensate system. Any low pressure in the condensate line or RHR lines is alarmed in the control room. A number of options are available to the control room operator to restore the keep fill pressure. In essence, prior to any accident, the RHR keep fill system will maintain the pressure in the RHR lines at approximately 100 psig, and adequate alarms and means are available to restore this pressure.

The keep fill system assures water up to the inlet of valve E11-F015. The line between E11-F015 and F050 is also kept filled with water.

At time zero, a DBA LOCA is assumed with loss of offsite power. The loss of offsite power renders the keep fill system inoperable. Therefore, post accident at time 0+, no credit is taken for any water replenishment using the keep fill system.

Refer to Figure 2 for piping isometrics between valve E11-F015 and E11-F050. If we assume a continuous leak of reactor coolant through this pipe, a temperature gradient will be established. For the purpose of this evaluation, this leak is conservatively assumed to be 0.4 gpm. A conservative steady state temperature profile in the pipe was established. Some of these assumptions used to establish this profile are:

- The outside convective heat transfer coefficient was calculated assuming the entire RHR line is horizontal. This is conservative as over 30% of the line is vertical and the vertical heat transfer coefficient is almost 20% higher than the coefficient for horizontal pipes.

- The outside convective heat transfer coefficient was calculated assuming zero air flow across the piping. This is conservative as engineering references predict a 6ft/sec. average face velocity.

- The pre LOCA drywell temperature is constant and equal to the maximum Technical Specification limit of 145°F.

Calculation shows that at least a 27 foot section of pipe that is downstream of valve E11-F015A(B), i.e., the entire horizontal run and part of the vertical run in the immediate vicinity of the penetration, has water at or below 212°F. This approximate 27 foot subcooled section will be hereafter referred to as control water volume (C.V.) or water slug. See Figure 2A and 2B.

Water temperature beyond the CV will be greater than 212°F.

Time Period A 0-13 Seconds

Following a recirculation line break at time 0.0, the fluid depressurizes, beginning at the break. The depressurization wave starts at the break location and travels towards valve E11-F015. The loss of water inventory during this period will be due to depressurization and flashing. The water inventory loss due to depressurization is directly dependent upon the initial pressure in the pipe.

In order to maximize inventory loss (for purpose of analysis) due to depressurization, a water pressure of 1233 psia is assumed in the entire inboard RHR piping. The 1233 psia is based on reactor dome pressure of 1060 psia (Tech Spec max) +23 psid static head of water + 150 psid recirc pump developed head. This 1233 psia pressure and temperature profile based on a 0.4 gpm continuous leak yields the worst combination for water inventory loss following an accident. These are the conditions assumed in the calculations. Some other conservative assumptions are:

- Wall friction is ignored. This is conservative as the friction will help to physically stop the water slug and thereby minimize C.V. loss during the depressurization.

- Drywell pressure is a constant 14.7 psia. This is conservative as drywell pressure rises quickly following a LOCA. Increased drywell pressure helps stop the water slug and thus help reduce the C.V. inventory loss.

- Following the break, water above 212°F flashes into a two phase mixture and most of the inventory between the break and C.V. will flow out of the break. However, some of the mixture will fall towards the C.V. regions. For conservatism, all of the fluid between the break and C.V. is assumed to be lost.

During the time period of 0-13 seconds, a number of phenomena occur. Besides the RHR pipe depressurization wave (< 1 second) phenomenon, all water initially above 212° is conservatively assumed to be lost due to flashing (< 1 second). In the meanwhile, the drywell pressure and temperature rise and they reach their peak in approximately 5 seconds. Further conservatism is added by assuming the C.V. temperature follows the drywell temperature and peaks in approximately 5 seconds and the second reflashing occurs as a result of drywell pressure reduction after 5 seconds. As a result, approximately 26.8 ft of C.V. is left after 13 seconds.

During the initial 13 seconds post LOCA, the water pressure in the RHR lines on the secondary containment side of the primary containment penetration was computed again using very conservative assumptions. For instance, it was assumed that the drywell pressure remains

constant at 49.9 psig; the volume of keep fill is 2674 ft^3 (as opposed to actual 2711 ft^3), no makeup water is supplied to keep fill; each of the RHR pumps check valves leak 1 gpm to torus; each of the secondary containment RHR loops (this includes all valves, pump seals, heat exchanger, blind flanges, restricting orifices, etc.) leak 40 ml/min. The ISI test program assures that the RHR pump check valve leakage and the total RHR secondary containment leakage are less than or equal to that assumed in the computation.

Even with the conservative assumptions, the RHR line pressure (keep fill pressure) is higher in the secondary containment compared to the primary containment. The primary containment piping still has a 26.8 foot section filled with water. Based on this, no airborne activity will be released from the primary containment to the secondary containment.

As most of the loss of water inventory happens during the initial few seconds, an independent verification of the blowdown phenomenon was made. A RELAP 5 MOD 3 version 5m5 computer code model was set up to simulate the RHR pipe blowdown using the calculated temperature gradient as an input. The results showed that an approximate 32 ft section of pipe will remain filled with water as opposed to approximately 27 feet section predicted in the calculation. All present and following discussions will reflect the more conservative calculation results.

Time Period B 13-46 Seconds

During this period, RHR pumps A and C get started at less than or equal to 13 seconds. Partial LPCI injection starts through RHR Loop B (E11-F015B) when the RPV pressure drops to approximately 450 psig, this occurs approximately 18 seconds into the accident. Full LPCI injection will happen at approximately 42-46 seconds. As RHR loop B will be injecting into the RPV (for this accident scenario), it need not be considered for the containment isolation perspective.

RHR loop A valve E11-F017A gets a signal to close at approximately 5 seconds (LPCI Loop Select Logic). However, this valve will not fully close until 29 seconds into the accident. Since RHR pumps get started at approximately 13 seconds, the RHR pipe up to E11-F015A gets pressurized. The net effect is that the pressure at the inlet of valve E11-F015A is greater than 115 psig.

During this period, the drywell pressure is dropping (see Figure 3A and B) and this will result in some more water inventory loss due to depressurization related flashing of the water slug. At the end of 46 seconds, the drywell pressure is approximately 41 psia and drywell temperature is approximately 268°F and an additional approximate .7 feet of water inventory is lost.

Therefore, as stated in time period A at the end of 46 seconds, the RHR line pressure on the secondary containment side is higher than in primary containment and the primary containment has an approximate 26.1 foot section filled with water. Based on this, no airborne activity will be released from the primary containment to the secondary containment.

Time Period C 46 Seconds - 30 Days

This case covers the longest time period that is 46 seconds-30 days following the accident. During this period for LPCI injection and/or suppression pool cooling, one or both of the RHR pumps (A/C) will be operational. RHR Loop B will be either injecting or pressurized at the inlet of E11-F015B valve. As this is not the broken loop and as it is filled with water inside the primary containment, and as the pressure on the inlet of E11-F015B is higher than the primary containment pressure, no airborne activity release is possible (through this loop) to the secondary containment.

Once the broken RHR Loop A is isolated by closing valve E11-F017A, E11-F017A cannot be opened for the first 10 minutes of the accident. This will prevent the RHR pump head from directly reaching the valve E11-F015A. Valve E11-F017A (globe) has a bypass valve E11-F611A (gate) across it. Globe valves have a relatively poor leak performance compared to the gate valve such as E11-F015A. The RHR pump pressure will be exercised under the seat of the globe valve E11-F017A, which will further inhibit its leak tightness. The net effect will be to pressurize the water volume between valves E11-F015A and E11-F017A.

Therefore, as discussed above during this time period (46 seconds-30 days) the operator has an option to open the valve E11-F017A. Even if this valve is not opened, the leakage from E11-F017A will pressurize the water volume between valves E11-F015A and E11-F017A. For the purposes of this analysis, we have added another magnitude of conservatism by assuming that leakage part valve E11-F017A and its bypass valve E11-F611A will not add to any water inventory to the water volume between valves E11-F015A and E11-F017A. Some other conservative assumptions are:

- Condensed steam from drywell atmosphere does not add to the slug mass.

- No mass addition to the initial volume from the higher pressure water in the RHR secondary containment pipe. In addition, up to 5 ml/min. can leak out of the CV through external leakage from E11-F015 and the water seal will be maintained for the 30 day period.

- The drywell pressure and temperature response assumed are based on the GE Power Uprate Submittal to the NRC (and approved by the NRC for Fermi 2). This analysis included some conservative assumptions such as 102% power, and higher dome pressure 1063 psia (licensed dome pressure is 1060 psia).

During the 30 day period, as the ambient drywell pressure and temperature decrease, the control volume is conservatively assumed to undergo a constant energy expansion to saturation pressure corresponding to 212°F. This causes partial flashing of the control volume leaving an approximate 24.4 foot section of RHR pipe filled with water. As the drywell temperature drops to 70°F, a further drop in volume occurs due to changing density leaving at least a 23.4 foot section of RHR pipe filled with water at the end of 30 days.

Results

It is concluded that even with conservative assumptions, the primary containment side of the RHR pipe will have at least a 23.4 foot section filled with water, 30 days after the accident. This water will completely fill the horizontal leg and part of the vertical leg (see Figures 2a and 2b). Also, any leakage from E11-F015 (a flexible wedge gate valve) will be into the primary containment.

FIGURE 1

SIMPLIFIED DIAGRAM RHR SYSTEM

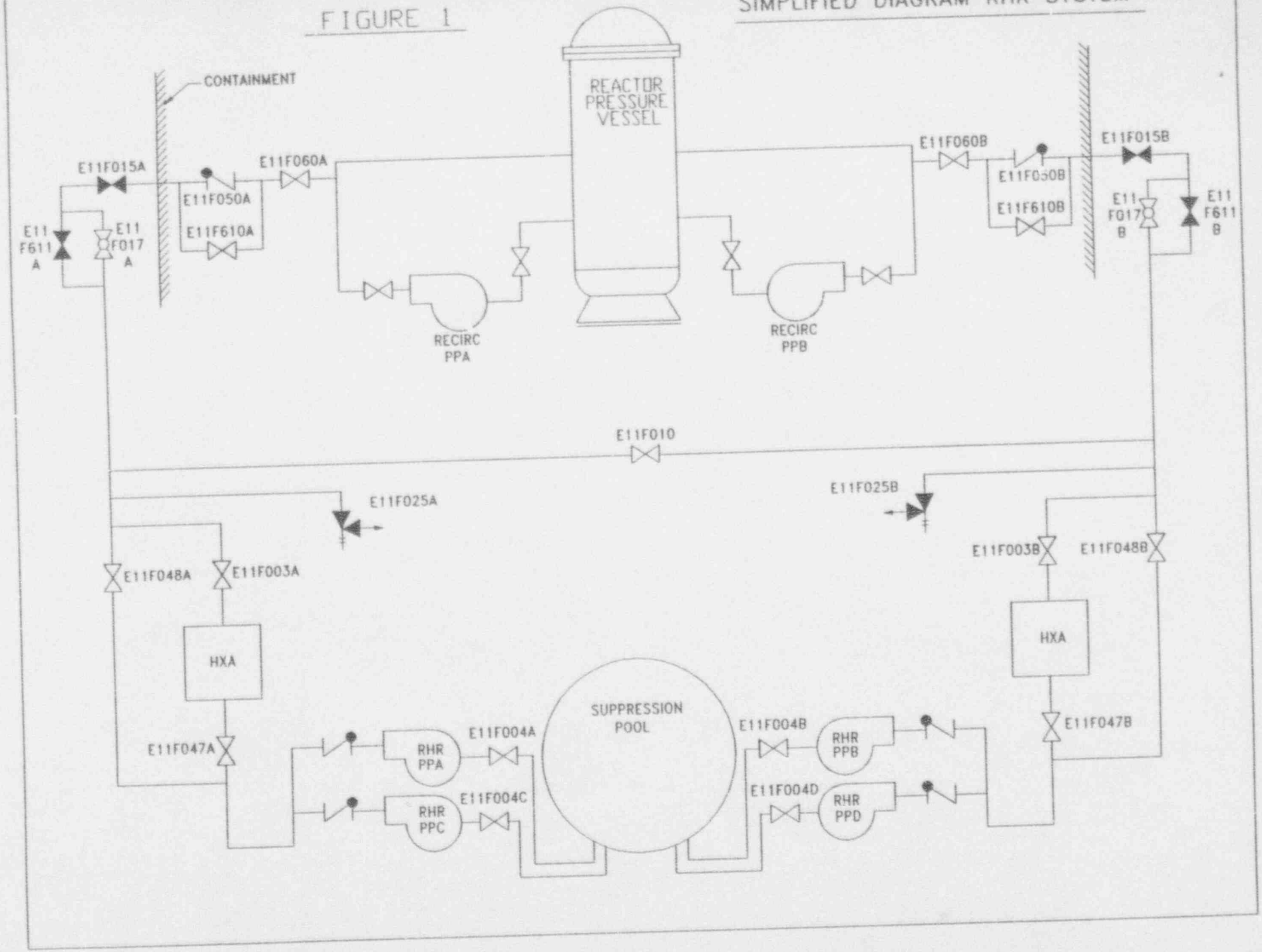
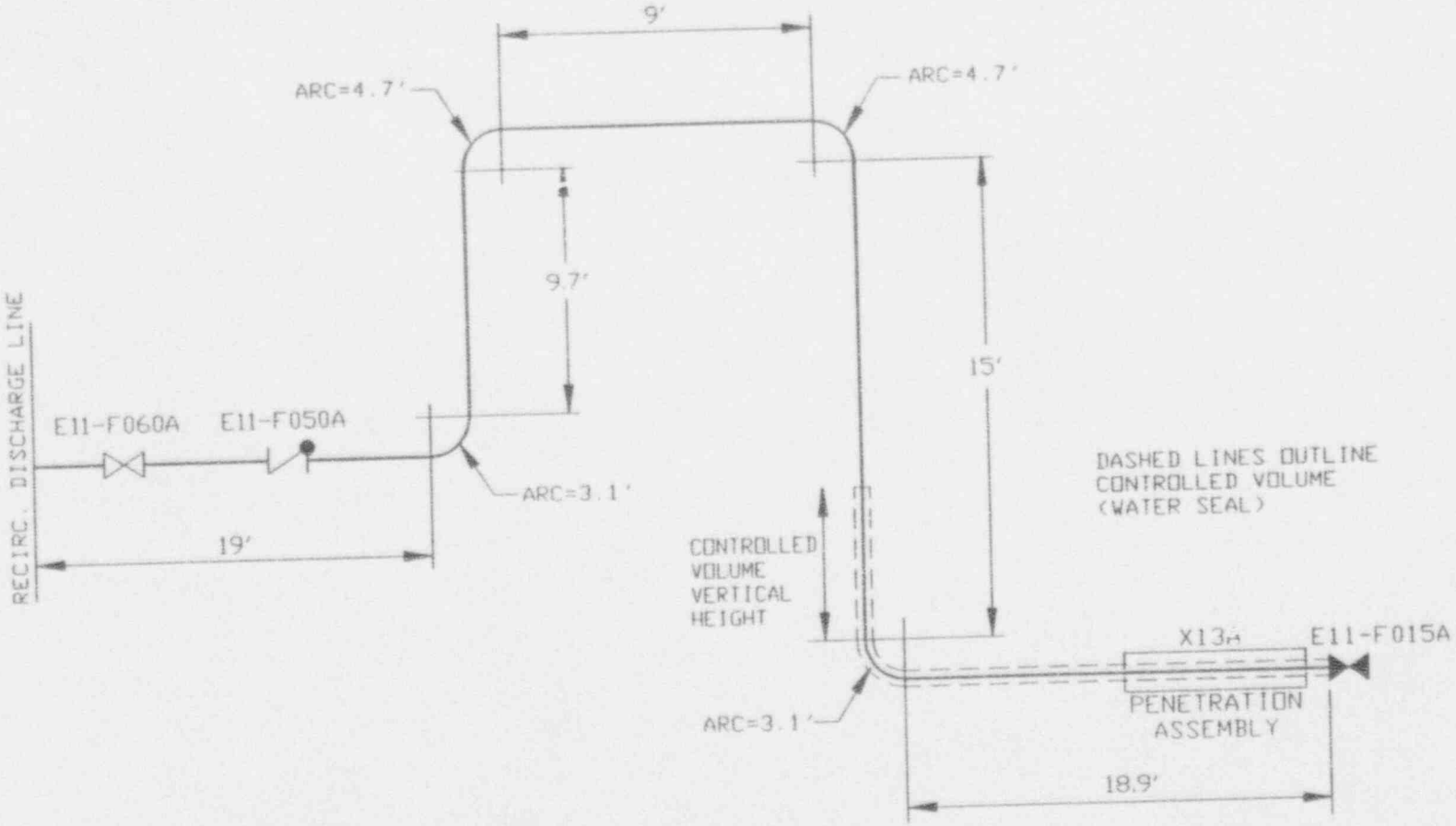
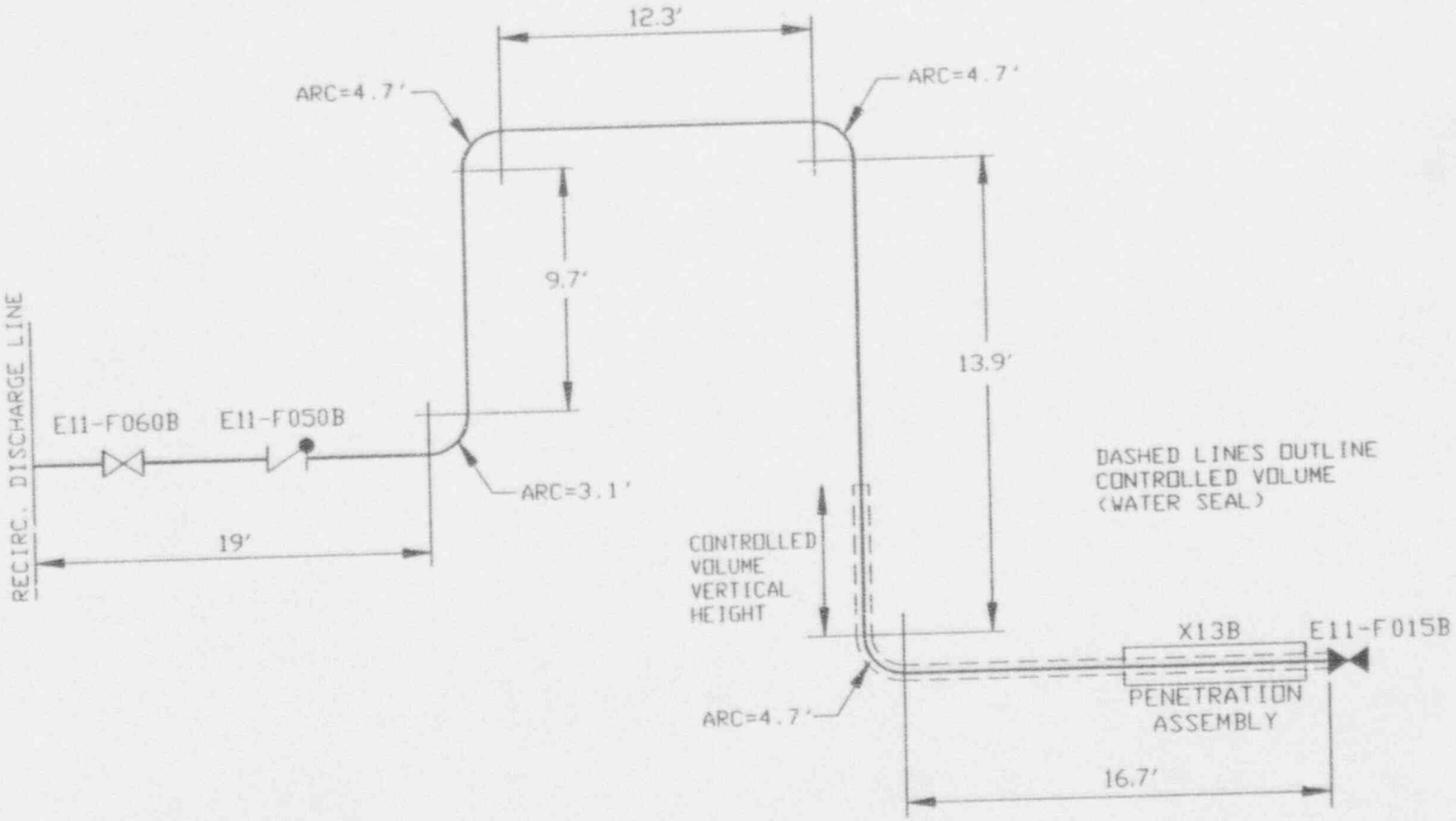


FIGURE 2A



PIPING ISOMETRIC - LPCI "A" INJECTION LINE (INB'D)

FIGURE 2B



PIPING ISOMETRIC - LPCI "B" INJECTION LINE (INB'D)

ENCLOSURE 3

Proposed Technical Specification Changes

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24-hour period.

d. Leakage specified in Table 3.4.3.2-1 at a reactor coolant system pressure of 1045 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check* valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

TABLE 3.4.3.2-1
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	VALVE DESCRIPTION	MAXIMUM LEAKAGE (gpm)
1. RHR System		
E11-F015A	LPCI Loop A Injection Isolation Valve	0.4[a]
E11-F015B	LPCI Loop B Injection Isolation Valve	0.4[a]
E11-F050A	LPCI Loop A Injection Line Testable Check Valve	10
E11-F050B	LPCI Loop B Injection Line Testable Check Valve	10
E11-F008	Shutdown Cooling RPV Suction Outboard Isolation Valve	1
E11-F009	Shutdown Cooling RPV Suction Inboard Isolation Valve	1
E11-F608	Shutdown Cooling Suction Isolation Valve	1
2. Core Spray System		
E21-F005A	Loop A Inboard Isolation Valve	1
E21-F005B	Loop B Inboard Isolation Valve	1
E21-F006A	Loop A Containment Check Valve	1
E21-F006B	Loop B Containment Check Valve	1
3. High Pressure Coolant Injection System		
E41-F007	Pump Discharge Outboard Isolation Valve	1
E41-F006	Pump Discharge Inboard Isolation Valve	1
4. Reactor Core Isolation Cooling System		
E51-F012	Pump Discharge Isolation Valve	1
E51-F013	Pump Discharge to Feedwater Header Isolation Valve	1

(a) External Leakage from this valve shall be limited to 5 ml/min.

TABLE 3.4.3.2-2
REACTOR COOLANT SYSTEM INTERFACE VALVES LEAKAGE PRESSURE MONITORS

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E11-F015A & B, E11-F050A & B	RHR LPCI	≤ 449
E11-F008, F009, F608	RHR Shutdown Cooling	≤ 135
E21-F005A & B, E21-F006A & B	Core Spray	≤ 452
E41-F006, F007	HPCI	≤ 71
E51-F012, F013	RCIC	≤ 71

TABLE 3.6.3-1 (Continued)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves[e]	
1. Main Steam Isolation Valves (MSIV) Leakage Control Valves	NA
B21-F434	
2. RHR Shutdown Cooling Suction Inboard Isolation Valve Bypass Valve[q]	NA
E11-F608	
3. LPCI Inboard Isolation Valves[f][s]	NA
Loop A: E11-F015A Loop B: E11-F015B	
4. RHR Pumps Recirculation Motor Operated Valves[b][g]	NA
Pumps A/C: E11-F007A Pumps B/D: E11-F007B	
5. Warmup and Flush Line Isolation Valve[b]	NA
E11-F026B	
6. Reactor Protection System Instrumentation Isolation Valves	NA
Division I: E11-F412 E11-F413 Division II: E11-F414 E11-F415	
7. RHR Pump Torus Suction Isolation Valves[b]	NA
Pump A: E11-F004A Pump B: E11-F004B Pump C: E11-F004C Pump D: E11-F004D	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves	
1. Main Feedwater Reverse Flow Check Valves B21-F010A B21-F010B B21-F076A B21-F076B	NA
2. Deleted	
3. RHR Heat Exchanger Relief Valves(b) E11-F001A E11-F001B	NA
4. RHR Heat Exchanger Outlet Line Relief Valves(b)(p) E11-F025A E11-F025B	NA
5. RHR Pump Suction From Recirc Piping Reverse Flow Check Valve E11-F408	NA
6. RHR Shutdown Cooling Suction Relief Valve(b)(p) E11-F029	NA
7. RHR Pump Torus Suction Relief Valves(b)(p) E11-F030A E11-F030B E11-F030C E11-F030D	NA

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS (Continued)

(k) Will automatically close when a) RCIC Turbine Steam Stop Valve E51-F045 closes or b) RCIC Turbine Governor Trip and Throttle Valve E51-F059 closes.

(l) Will automatically close as a result of the conditions listed in Note (k) above, as well as when RCIC flow is greater than 130 gpm.

(m) These valves are actuated by remote manual key-locked switches and will cut the TIP cable and seal off the TIP guide tube when actuated. These valves are squib-fired.

(n) May be closed remotely as a secondary actuation mode to reverse flow.

(o) Valves realign automatically on a reactor scram signal.

(p) Thermal relief valves.

(q) Locked closed.

(r) Not subject to Type C leakage tests.

(s) Hydrostatically tested in accordance with Specification 4.4.3.2.2 in lieu of the reqirements of Specification 4.6.1.2.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that contain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.

The purpose of the RCS interface valves leakage pressure monitors (LPMs) is to provide assurance of the integrity of the Reactor Coolant System pressure isolation valves which form a high/low pressure boundary. The LPM is designed to alarm on increasing pressure on the low pressure side of the high/low pressure interface to provide indication to the operator of abnormal interface valve leakage.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

A reduced leakage acceptance criteria and an external leakage acceptance criteria are specified for the LPCI Injection Isolation Valve, E11-F015 A and B, to assure adequate water is maintained inboard of these valves such that the associated primary containment penetration can be classified as a water tested penetration under Appendix J to 10CFR50.

CHEMISTRY 3/4.4.4

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is demonstrated by leak rate testing and by verifying that all primary containment penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by locked valves, blank flanges or deactivated automatic valves secured in the closed position. For test, vent and drain connections which are part of the containment boundary, a threaded pipe cap with acceptable sealant in addition to the containment isolation valve(s) provides protection equivalent to a blank flange.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 56.5 psig, P_a. Updated analysis demonstrates maximum expected pressure is less than 56.5 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing, testing the airlocks after each opening, testing the Low Pressure Coolant Injection Inboard Isolation Valves, and analyzing the Type A test data.

Appendix J to 10 CFR Part 50, Paragraph III.A.3, requires that all Type A tests be conducted in accordance with the provisions of N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." N45.4-1972 requires that Type A test data be analyzed using point-to-point or total time analytical techniques. Specification 4.6.1.2a. requires use of the mass plot analytical technique. The mass plot method is considered the better analytical technique, since it yields a confidence interval which is a small fraction of the calculated leak rate; and the interval decreases as more data sets are added to the calculation. The total time and point-to-point techniques may give confidence intervals, which are large fractions of the calculated leak rate, and the intervals may increase as more data sets are added.