March 26, 1999

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Mr. Lew W. Myers
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY NUCLEAR POWER PLANT, UNIT 1 (TAC NO. MA3487)

Dear Mr. Myers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 105 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit 1. This amendment revises the Technical Specifications in response to your application dated September 9, 1998 (PY-CEI/NRR-2322L), as supplemented by submittals dated January 6 (PY-CEI/NRR-2352L), March 4 (PY-CEI/NRR-2370L), and March 18, 1999 (PY-CEI/NRR-2376L).

This amendment revises the design and licensing basis of containment isolation valves in the feedwater system. The amendment revises (1) Surveillance Requirement 3.6.1.3.11 of TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)" to exclude the feedwater check valves from the hydrostatic test program, (2) TS 5.5.2, "Primary Coolant Sources Outside Containment," to stipulate that water leakage past the feedwater motor-operated containment isolation valves and the reactor water cleanup system return to feedwater line is added to the program, and (3) TS 5.5.12, "Primary Containment Leakage Rate Testing Program," to state that the feedwater check valves will be tested in accordance with the Inservice Testing Program (TS 5.5.6).

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by:
Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 105 to License No. NPF-58
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 26, 1999

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Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

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Docket No. 50-440

Enclosures: 1. Amendment No.105 to License No. NPF-58
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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. NPF-58

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee, formerly The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, and Toledo Edison Company) dated September 9, 1998, as supplemented by submittals dated January 6, March 4, and March 18, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 105 are hereby incorporated into this license. The FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett

Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 26, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.6-19	3.6-19
5.0-7	5.0-7
5.0-15a	5.0-15a

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	------------------NOTES------------------ 1. Only required to be met in MODES 1, 2, and 3. 2. Feedwater lines are excluded. -- Verify combined leakage rate of 1 gpm times the total number of PCIVs through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify each outboard 42 inch primary containment purge valve is blocked to restrict the valve from opening > 50°.	18 months
SR 3.6.1.3.13	------------------NOTE------------------ Not required to be met when the Backup Hydrogen Purge System isolation valves are open for pressure control, ALARA or air quality considerations for personnel entry, or Surveillances or special testing of the Backup hydrogen Purge System that require the valves to be open. -- Verify each 2 inch Backup Hydrogen Purge System isolation valve is closed.	31 days

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the Low Pressure Core Spray System, High Pressure Core Spray System, Residual Heat Removal System, Reactor Core Isolation Cooling System, hydrogen analyzer portion of the Combustible Gas Control System, Post-Accident Sampling System, Reactor Water Cleanup System Return to Feedwater line, and Feedwater Leakage Control System, including the Feedwater System motor-operated containment isolation valves. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- BN-TOP-1 methodology may be used for Type A tests.
- The corrections to NEI 94-01 which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996, are considered an integral part of NEI 94-01.
- The containment isolation check valves in the Feedwater penetrations are tested per the Inservice Testing Program (Technical Specification 5.5.6).

The peak calculated primary containment internal pressure for the design basis loss of coolant accident P_a, is 7.80 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.20% of primary containment air weight per day at the calculated peak containment pressure (P_a).

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with this Program, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

 1) Overall air lock leakage rate is ≤ 2.5 scfh when tested at $\geq P_a$.

 2) For each door, leakage rate is ≤ 2.5 scfh when the gap between the door seals is pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.105 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated September 9, 1998, as supplemented by submittals dated January 6, March 4, and March 18, 1999, the FirstEnergy Nuclear Operating Company (the licensee, formerly The Cleveland Electric Illuminating Company and Centerior Service Company) proposed changes to the Perry Nuclear Power Plant, Unit 1 (PNPP) Technical Specifications (TSs) related to hydrostatic (water) testing of the containment isolation valves in the feedwater system. The proposed changes, which were submitted pursuant to 10 CFR 50.59 and 10 CFR 50.90, would revise the licensing and design basis of the feedwater isolation provisions.

The licensee has experienced extensive operational difficulties with regard to the containment isolation and leak rate testing of the feedwater system check valves. Past leak rate testing of these valves, pursuant to Appendix J to 10 CFR Part 50, have questioned whether adequate containment isolation would be attained following a postulated accident. The licensee has proposed changes to the licensing and design basis for the overall feedwater isolation system to improve and enhance the reliability of the containment isolation provisions.

The licensee's letter of January 6, 1999, included a proposed exemption to the leak rate testing requirements of Appendix J to 10 CFR Part 50, for the feedwater system check valves. The licensee proposed to conduct a visual examination of the check valves in lieu of leak rate testing. During the review process, the staff and the licensee concluded that performance of a leak rate test of the check valves was a superior method to demonstrate valve operability as opposed to a visual examination. Performance of a leak rate test would satisfy the requirements of Appendix J and preclude the need for an exemption. In the licensee's letter of March 4, 1999, the licensee committed to conduct leak rate testing of the check valves consistent with Appendix J requirements and acknowledged that an exemption to Appendix J was no longer necessary.

The supplemental information in the licensee's letters of January 6, March 4, and March 18, 1999, contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

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2.0 BACKGROUND

2.1 Current Feedwater Penetration Isolation Design

The feedwater penetration is a unique case for containment isolation. For the majority of system transients or loss of coolant accidents (LOCAs) other than a feedwater pipe break, feedwater flow will be maintained in order to get cooling water to the reactor vessel. However, there are certain conditions when feedwater flow must be isolated. Isolation provisions must (1) eliminate containment atmosphere leakage in the feedwater piping for LOCAs inside containment, and (2) isolate reactor coolant system leakage flowing in the reverse direction for feedwater line breaks outside containment.

The current Perry feedwater penetration and isolation provisions are shown in Figure 1. (It should be noted that there are two feedwater penetrations designated train A and B. However, for simplicity, Figure 1 only lists the valve numbers and does not provide the train A and B designations.) Each feedwater line penetrating containment has three containment isolation valves. Two piston lift-style check valves, located both immediately inside and outside the containment penetration, are in each feedwater line for isolation of significant flow from a feedwater line break outside containment. These anti-waterhammer check valves are not designed for air tests at low pressures. A third valve (a remote manual motor-operated gate valve), located outside containment upstream of the outside check valve, is provided for long-term, high integrity leakage protection when, in the judgment of the operator, continued make-up from feedwater is unnecessary or is not available. There is no automatic isolation of the feedwater lines based on accident signals, so that feedwater flow can be maintained to the reactor vessel. As noted in ANS-56.2/N271-76, "Containment Isolation Provisions for Fluid Systems," greater plant safety is maintained with a feedwater supply to the reactor.

The feedwater leakage control system (FWLCS) is designed to eliminate containment atmosphere through-line leakage in the feedwater piping for LOCAs inside containment by providing a positive water seal between the isolation valves. The FWLCS consists of two independent trains. As shown in Figure 1, Division 1 of the FWLCS furnishes sealing water from the suppression pool using the low-pressure core spray (LPCS) waterleg pump to an outboard volume (between the outboard feedwater isolation check valves, B21-F032, and the remote manual gate valves,motor operated valves (MOVs), B21-F065). The MOVs (B21-F065) are powered by Division 1 electrical power. Division 2 of the FWLCS furnishes sealing water from the suppression pool using the residual heat removal (RHR) waterleg pumps (B/C) to an inboard volume (between the two feedwater isolation check valves, N27-F559 and B21-F032). The FWLCS system is designed and installed as a single failure proof, safety-related, seismically qualified system that is designed to withstand the dynamic effects of postulated piping failures in the steam tunnel including protection from internally generated missiles.

The FWLCS is a manually activated system, effective within approximately 1 hour after the onset of a LOCA. When the operator has determined that feedwater is either unavailable or not necessary, the FWLCS is actuated to provide a water seal in the feedwater penetration line to prevent through-line leakage of the containment atmosphere to the environment. The current licensing basis assumes that operator action will be initiated within 20 minutes of a LOCA to

(1) close the motor operated gate valve, and (2) initiate the FWLCS such that a water seal between the isolation valves will be established within approximately 1 hour. (It should be noted that it takes approximately 40 minutes to fill the piping volume between the three containment isolation valves. Thus, a water seal is assumed to be established within one hour of a LOCA.)

The FWLCS includes interlocks to ensure that the outboard FWLCS (Division 1) is not initiated without the feedwater MOV being closed thereby preventing the inadvertent discharge of suppression pool water to the feedwater piping system. The inboard FWLCS (Division 2) system is not interlocked with the feedwater MOV.

2.2 Need for Change in Licensing Basis

The current licensing basis for feedwater isolation relies on the FWLCS to establish a water seal between the three containment isolation valves approximately one hour following an accident. Successful operation of the FWLCS relies upon operator action to close the MOV and initiate the FWLCS approximately 20 minutes following the accident. The feedwater check valves must be essentially leak tight so that the injected water from the FWLCS can establish a water seal. The check valves are leak tested during every refueling outage and have an acceptance criteria of 1 gpm water leakage. If leakage exceeds this limit, a water seal may not be established due to limitations of flow from the FWLCS makeup pumps.

Operational experience has not always demonstrated that the feedwater check valves are leak tight. "As-found" testing over the last several refueling outages has shown that the check valves leak in excess of 1 gpm and actual values have been on the order of 4-14 gpm. Whenever leakage exceeds allowable limits, the licensee is required to take corrective actions to restore valve integrity to the licensing limits. This requirement has led to excessive costs and man-rem exposures. During the sixth refueling outage, the licensee estimated that valve restoration cost approximately $880,000 and 5 man-rem exposure. More importantly, however, is the concern that due to excessive check valve leakage, the existing FWLCS may not be capable of performing its safety-related function.

The licensee explored a number of options to improve the reliability of the feedwater isolation system. As described in the next section, the licensee selected a design that would no longer inject the FWLCS in the piping volume between the containment isolation valves but would inject the FWLCS through the stem of the MOV and thus establish a water seal between the double discs of the MOV. Reliability of the MOV to close would be increased by introducing an alternate electrical power supply to the MOV. Finally, the licensee proposed to perform leak rate tests for the check valves pursuant to Appendix J to 10 CFR Part 50 to demonstrate continued valve integrity.

2.3 Proposed Feedwater Penetration Isolation Design

In the proposed design, as shown in Figure 2, both independent trains of the FWLCS will be routed to the bonnet area of each existing feedwater MOV. To allow the FWLCS to seal the 20" gate valves, the FWLCS will be supplied to an existing ½"-threaded connection in the packing area of the valve bonnet. Once in the bonnet area of the 20" gate valve, the FWLCS seal water flows around the wedge-shaped gate into the area between the two hardfaced mainseats in the valve body forming a water seal. Since FWLCS water is supplied at higher than accident pressure (P_a), any air leakage path past the check valves would be sealed by the FWLCS.

The rerouted FWLCS subsystems will continue to be designed and installed as single failure proof, safety-related, seismically qualified systems and will be designed to withstand the dynamic effects of postulated piping failures in the steam tunnel including protection from internally generated missiles.

In the revised configuration, the Division I train FWLCS interlocks will be maintained to prevent actuation of the Division 1 LPCS waterleg pump unless the feedwater MOVs are closed. The Division II FWLCS RHR (B/C) waterleg pump operation will be governed by plant procedures and instructions.

The proposed design change includes provisions for providing alternate power from Division III to the feedwater MOVs if Division I power is lost. This reduces the possibility of a FWLCS failure upon concurrent loss of offsite power and loss of Division I power. The alternate power design approach to be taken is similar to that taken at PNPP for station blackout where Division III is backup for Division II through procedures. This design change enhances the likelihood of MOV closure within a 1-hour time frame if Division I power is lost. The 1-hour time frame for MOV closure and the establishment of the water seal in the feedwater line penetration is consistent with the current licensing base.

2.4 Proposed Technical Specification Changes

The licensee's proposal includes the following changes to the TSs:

(1) A note will to be added to Surveillance Requirement (SR) 3.6.1.3.11 of TS 3.6.1.3, "Primary Containment Isolation Valves," that excludes the feedwater check valves from the hydrostatic test program. This change will relieve the licensee from conducting leak rate tests of the feedwater check valves with a 1 gpm acceptance criterion.

(2) TS 5.5.2. "Primary Coolant Sources Outside of Containment," is a program which provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, will be modified to include two new piping systems. The two new piping systems will be the feedwater system motor operated isolation valves and the reactor water cleanup system return to feedwater lines. These changes are needed due to the rerouting of the FWLCS piping.

(3) TS 5.5.12, "Primary Containment Leakage Rate Testing Program," will be modified to state that the containment isolation check valves in the feedwater penetrations will be tested pursuant to the Inservice Testing Program (TS 5.5.6). This change documents that leak rate testing of the feedwater check valves will be performed. However, the acceptance criterion for these tests will be relaxed.

3.0 EVALUATION

The staff's review focused on the following five areas:

(1) Impact on Containment Isolation Provisions - By rerouting the FWLCS piping from the volume between the isolation valves to the stem of the MOV, leak testing requirements pursuant to Appendix J to 10 CFR Part 50 may change due to changes in the test medium. A determination must be made regarding the adequacy of the licensee's leak rate testing program.

(2) Impact on Existing Pipe Break Analysis - A determination on whether the proposed changes to the feedwater isolation provisions impact any of the existing pipe break analysis.

(3) Feedwater Leakage Control System Reliability - The proposed design change is not single failure proof and the mechanical failure of the MOV to close on demand could compromise containment isolation. The licensee's risk-informed discussion must support the proposed design and licensing changes.

(4) Electrical Interface for Proposed Alternate Power Supply - The MOVs of both feedwater trains are powered from Division I Power. The proposal enhances the reliability of electrical power by introducing an alternate power supply fed from Division III. The electrical interface must be accomplished in an acceptable manner.

(5) Evaluation of Manual Operator Actions - Similar to the existing licensing basis, operator actions are relied upon to initiate the FWLCS. An evaluation of the new operator actions for the proposed design changes must be found acceptable.

3.1 Impact on Containment Isolation Provisions

Leak Rate Testing of the Feedwater Check Valves

The proposed testing change is based on design and licensing basis changes proposed for implementation to improve functioning of the FWLCS. Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," noted that tests need to be performed on check valves that perform a safety function in the closed position to prevent reverse flow as stated in Position 3 of Attachment 1 to GL 89-04, "Back Flow Testing of Check Valves." Category C tests on such "safety function check valves" were described as needing to prove that the disc closes promptly on its seat on cessation or reversal of flow. As stated in the GL, verification that a Category C valve is in the closed position can be done by visual observation, by an electrical signal indicated by a position-indicating device, by observation of appropriate pressure indication in the system, by leak testing, or by other positive means. Main feedwater header check valves were listed in GL 89-04 as an example of ASME code class check valves that are frequently not tested.

The licensee proposes that the feedwater containment isolation check valves be Category C tested for their safety function at an appropriate frequency as determined in the Inservice Testing Program. To be consistent with GL 89-04, as discussed in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," and as addressed in Supplement 1 to GL 89-04, the test interval for check valves verified closed by leak testing may be extended to the refueling outage. Therefore, the Inservice Testing Program test interval is consistent with the current Appendix J test interval. Testing will meet the "exercised closed" test and the "exercised open" test. The "exercised closed" test will require a hydrostatic (water) leak rate test, with an acceptance criterion of ≤200 gallons per minute (gpm) per feedwater penetration, when tested at ≥1.1 P_a. This test is to be identified in TS 5.5, "Programs and Manuals," section 5.5.12, "Primary Containment Leakage Rate Testing Program," by reference to the Inservice Testing Program (TS 5.5.6) for the containment isolation check valves in the feedwater penetrations.

The limit of ≤200 gpm per feedwater penetration, when tested at ≥1.1 P_a, is used as the method to test for proper check valve closure (Category C "exercised closed") and will also ensure no "significant leakage" (Category A leak testing). Hydrostatic testing is acceptable to the staff since that is the medium expected to be acting on these check valves when they are performing their safety function to prevent a feedwater line break outside containment from becoming an uncontrolled LOCA. A specific leak rate limit, established by the licensee to be ≤200 gpm per feedwater penetration, is consistent with the Appendix J acceptance criterion (Option A, III.C.3(a)) that the fluid leakage rates do not exceed those specified in the technical specifications or associated bases. In addition, the Inservice Testing Program test ensures no "significant leakage" for the feedwater line break outside containment and therefore the staff concludes that the Appendix J acceptance criterion (Option A, III.C.3(b)) concerning an adequate valve seal-water inventory is also met. In this case, the reactor coolant makeup is sufficient to maintain the sealing function for at least 1 hour, by maintaining the feedwater line full, at which time credit for remote-manual operator closure of the MOV has been previously accepted by the staff. For this case, the feedwater break outside containment, the specific requirement for a 30-day inventory is not considered to be necessary as the accident may be terminated by closure of the MOV within 1 hour.

While the licensee's acceptance criterion for the feedwater check valves will be ≤200 gpm per feedwater penetration, it should be noted that the current configuration of the PNPP feedwater penetration line and the available taps into the piping limit the range of the leak rate tests that can be performed. Based on discussions with the licensee, it was determined that the existing taps in the feedwater line cannot pass more than 19 gpm at the expected test pressure of ≥1.1 P_a. Therefore, for the upcoming refueling outage, the acceptable leak rate for which no check valve inspection or refurbishment is needed will be less than 19 gpm for each check valve. The licensee is considering changes to the feedwater penetration line to allow for tests at higher leak rates in the future, or to continue to pursue the visual inspection option by developing a means to ensure that the leakage is within the 200 gpm allowable.

As described above, the staff finds the proposed leak rate testing of the feedwater check valves acceptable because it is consistent with the requirements of Appendix J to 10 CFR Part 50.

Isolation Provisions for Branch Lines into the Feedwater Pipe

Once the FWLCS is rerouted to the stem and bonnets of the MOVs in lieu of the feedwater piping volume, branch lines off of the feedwater line need a different licensing basis for leakage mitigation (See Figure 2). These lines no longer have a water seal since the FWLCS will no longer be used to fill the feedwater piping between the isolation valves.

The RHR branch line off of the feedwater line will be treated as a closed system outside of containment similar to the lines discussed in Note 4 to USAR Table 6.2-33 and Note 7 to Table 6.2-40. These notes explain why leakage in these lines is not considered to be bypass leakage. Leakage from the systems listed in USAR Notes 4 and 7 as "closed" are controlled by the Primary Coolant Sources Outside Containment program. A safety-related globe valve (1E12-F053) in this branch line will be treated as a high integrity containment isolation valve, similar to the feedwater MOVs. The 1E12-F053 valves will be added to the containment isolation valve listings. These valves meet the qualifications of a containment isolation valve. An air test will be performed on 1E12-F053 and the air leakage will be added into the Type C totals and limited by 0.60 L_a in accordance with Appendix J to 10 CFR Part 50. Also, the leakage from the F053 valves will be added into the Type A integrated leak rate test (ILRT) since the feedwater penetrations will not be drained during the ILRT. This RHR branch pathway will consist of an air leak rate tested containment isolation valve and a closed system outside of containment. In addition, a high-to-low-pressure interface water test is performed on the 1E12-F053 globe valve and the check valve inboard of the F053 (1E12-F050) in accordance with ASME Section XI. These valves are tested to water leakage limits of ≤5 gpm.

A reactor water cleanup (RWCU) branch line also exists. This line returns the filtered RWCU water to the reactor vessel through the feedwater lines. The piping "outboard" of the RWCU branch line check valve (1G33-FO52) leads directly back to containment penetration, and is ASME Code Class 2, Seismic Category I, protected from pipe whip, missiles and jet forces, and analyzed for "break exclusion." This closed system outside containment contains only mechanical joints, including the packing on the outboard containment isolation valve (1G33-F039). This outboard valve, including the stem and bonnet, is already part of the air leak rate test program. The remainder of the RWCU line between the feedwater line and the containment penetration will be added to the TS 5.5.2 Primary Coolant Sources Outside Containment program with a specific leakage acceptance limit of zero (0) leakage when tested at RWCU operating pressures (>1,000 psig). Zero water leakage outside the piping when operating at over 1,000 psig ensures that there will be no air leakage from those mechanical joints at P_a (7.8 psig for PNPP) and the RWCU branch line check valves (1G33-FO52) are not added into the leak rate testing program. This approach meets Branch Technical Position CSB 6-3 "Determination of Bypass Leakage Paths in Dual Containment Plants," Item B.9. Item B.9 specifies the criteria for when a closed system may be used as a leakage boundary to preclude bypass leakage. This approach is also confirmed by PNPP leak test program results where joints that showed water leakage at full system operating pressures did not exhibit measurable air leakage when tested at P_a.

The piping of each FWLCS subsystem, which will connect to the bonnets and seats of the MOVs, currently contains two existing isolation valves. These valves receive a high-to-low pressure interface water test since they connect back to the RHR/LPCS waterleg pumps. These tests will continue to be performed.

As described above, the staff has concluded that containment isolation valves in branch lines leading to the feedwater piping will be appropriately treated through Appendix J to 10 CFR Part 50 and the licensee's Primary Coolant Sources Outside Containment program. Therefore, the staff concludes that the containment isolation provisions for the branch lines are acceptable.

3.2 Impact on Existing Pipe Break Analysis

Feedwater Line Break Outside Containment

For a feedwater line break outside containment, there will be a 1,000 psid pressure acting to close the feedwater check valves to prevent significant reverse flow through the line. The analysis of this accident is presented in Section 15.6.6, "Feedwater Line Break - Outside Containment," of the PNPP USAR. Closure of the feedwater check valves is assumed to occur shortly after the postulated break and 1.454×10^6 lbm of condensate comprise the inventory used for the radiological consequences analysis. The resulting doses are calculated to be well within 10 CFR Part 100 guidelines and are bounded by the doses resulting from either the main steam line break outside containment or the feedwater line break inside containment.

At PNPP, the current FWLCS leakage test performed at $\geq 1.1\ P_a$ is used by the licensee to demonstrate that closure of the check valves at reactor coolant pressure (1,000 psi) would occur, consistent with the USAR 15.6.6 analysis for a feedwater line break outside containment. Hydrostatic leak rate testing at low pressures (i.e., $1.1 P_a$) will continue to be used at PNPP to demonstrate proper closure. A sensitivity study was performed by the licensee to determine the amount of leakage from the feedwater penetrations that would result in consequences similar to the limiting main steam line break (MSLB) outside containment. It was determined that 200 gpm per line (400 gpm total) leakage for 2 hours would have to be exceeded for the consequences to exceed the current value in USAR Table 15.6-11. The results of this study will be included in USAR Section 15.6.6.5.2.4, "Sensitivity Analysis," as identified in the licensee's letter dated January 6, 1999. The "exercised closed" test, a hydrostatic (water) leak rate test, with an acceptance criterion of ≤ 200 gpm per feedwater penetration when tested at $\geq 1.1\ P_a$, will verify proper closure of these valves to prevent significant leakage of this order of magnitude.

Therefore, based upon the above information, the staff concludes that the proposed changes concerning the FWLCS and the leak testing of the feedwater penetration do not affect the licensing basis for the feedwater line break outside containment.

Feedwater Line Break (LOCA) Inside Containment

For a feedwater line LOCA inside containment, the operator first verifies feedwater unavailability through low feedwater pressure (approximately 30 psig), then closes the outboard MOVs with keylock switches, and opens the motor operated FWLCS valves from the control room. The current licensing basis assumes that this operator action will take place within the first

20 minutes following an accident. Sealing water is provided from the suppression pool via the residual heat removal (RHR) and the low-pressure core spray (LPCS) waterleg pump(s). Since the source of sealing water is the suppression pool, a 30-day water supply is assured. When the FWLCS is initiated following a LOCA, there should be no demand for keep-fill water in the RHR and LPCS systems since these systems will be operating. Therefore, the waterleg pumps should be totally dedicated to provide sealing water to the FWLCS.

Operator actions to recognize feedwater unavailability and initiate the FWLCS remain unchanged under the proposed design. A water seal in the feedwater piping (i.e., the double disc of the gate valves) will be established and a 30-day supply from the suppression pool will still be available. Under the proposed design changes, the time necessary for the FWLCS to establish the water seal is approximately 9 minutes as opposed to the previous 44 minutes thus allowing additional time for control room operators to take action. Therefore, the staff concludes that the proposed changes concerning the FWLCS do not affect the licensing basis for the feedwater line break inside containment.

3.3 Feedwater Leakage Control System Reliability

The licensee discussed the risk impact of the proposed change and provided adequate information for comparing the proposed change in risk to acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in Regulatory Guide (RG) 1.174 entitled, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." An acceptable approach to risk-informed decision making is to show that the proposed change to the licensing basis meets several key principles (RG 1.174). One of these principles is to show that the proposed change results in an increase in risk, in terms of core damage frequency (CDF) and large early release frequency (LERF), which is small and consistent with the Commission's Safety Goal Policy Statement. Information submitted by the licensee indicated that the plant CDF would not change and the plant LERF would actually decrease once the proposed modification is implemented.

The proposed feedwater line isolation change does not have any impact on the plant's CDF because it is related to containment isolation following core damage (Level 2 PRA). Since PNPP has not performed a Level 2 PRA, the results of the Level 1 PRA were used in conjunction with a reliability study (which compares the reliability of the current feedwater line isolation design to the reliability of the proposed modification) to show that the proposed modification would most likely decrease the already low contribution to the plant LERF associated with a feedwater line isolation failure. Even though the plant CDF is not affected, a Level 1 CDF discussion is relevant in determining the overall acceptability of the proposed modification because the feedwater line penetration does not need to be sealed by the FWLCS unless core damage has occurred. Furthermore, the feedwater line penetration will most likely be sealed even without crediting the FWLCS if the core damage scenario does not involve a feedwater line break at a low elevation of the system inside containment but is associated with a pressurized vessel providing a strong seating force on the check valves in the feedwater line.

Based on the PRA submitted by the licensee as part of its individual plant examination (IPE) and the current "living" PRA, the CDF from internal events for the PNPP is less than 2×10^{-5} per year.

This CDF is dominated by sequences which do not involve a feedwater line break and are associated with a pressurized vessel at the time the feedwater line stops feeding the vessel (for example, sequences initiated by various transients, ATWS events, loss of offsite power and station blackout). This pressure provides a strong seating force on the check valves in the line at the beginning of the event, which is how the valves are designed to seal well. Many of these events would also provide the water seal on the feedwater penetration since the feedwater line would not be broken, and water would remain from the initial injection feedwater or the reflood water.

The only core damage scenarios requiring successful operation of the FWLCS to seal the feedwater penetration lines involve a feedwater line break at a low elevation inside containment. The frequency of such core damage scenarios is a small portion of the CDF from loss of coolant accidents (LOCAs) which is about $2x10^{-7}$ per year based on the IPE results and $5x10^{-8}$ per year based on the current "living" PRA results. This shows that the contribution of the FWLCS in preventing large release following a core damage event is very small (the LERF would most likely increase by less than $1x10^{-7}$ per year if the FWLCS was assumed to always be unavailable). A comparison of the reliability of the current feedwater line isolation design to the reliability of the proposed modification showed that the proposed modification would most likely decrease the already low contribution to the plant LERF associated with a feedwater line isolation failure.

Impact of the Proposed Change on the Reliability of the Containment Isolation Provision

The licensee submitted information from a reliability study which compared the reliability of the current feedwater line isolation design to the reliabilities of two alternative designs. One of the alternative designs was the proposed modification whereas the second alternative design was the two MOV alternative. The two MOV alternative assumed a second MOV gate valve would be installed in series with each of the existing outboard gate valves and each division of FWLCS would also be routed to each division's respective MOV. Although this modification is "single failure proof" with respect to active component failures, manual action is still required, and preferred, to close the MOVs and initiate the FWLCS. The time needed to pump in the water seal is less than 9 minutes (as for the proposed design), allowing a much longer time for operator diagnosis and response. The staff review of the manual operator actions is provided below.

PNPP assessed the conditional probability of failure to provide the required water seal in both feedwater lines including the current design as well as the two alternatives for two cases: one assumed no loss of offsite power (LOOP), the other assumed a LOOP. In comparing these probabilities, it was conservatively assumed that the current design worked as assumed in the licensing basis (i.e., the check valves close and establish a water seal).

Conditional probability of failure of feedwater leakage control				
FWLCS design	Without LOOP		With LOOP assumed	
	Total	HEP*	Total	HEP*
Current design	0.267	0.26	0.28	0.260
Proposed design	0.0419	0.036	0.0689	0.067
Two MOV alternative	0.0362	0.036	0.0491	0.036

* Human Error Probability

The results for the case which assumed availability of offsite power indicate the following:

1. The relatively low reliability of the current design (i.e., the relatively high failure probability) is mainly due to the relatively high probability of operator failure to close the MOV and initiate the FWLCS (human error probability [HEP] 0.26). This is due to the relatively short time (20 minutes) available for recognizing the need to isolate the feedwater system and for taking appropriate operator actions.

2. The fact that the "current design" (assuming it works as assumed in the licensing basis) is "single failure proof" with respect to active component failures is not significant in terms of reliability. This is due to the fact that the human error probability dominates the reliability of the current design (0.26 human error probability versus $6x10^{-3}$ hardware failure probability).

3. The improved reliability of the "proposed design," as compared to the "current design," is primarily due to the reduced human error probability because of the significant increase in the time available for operator diagnosis and response (the human error probability would decrease from 0.26 to 0.036 while the hardware failure probability would remain essentially the same).

4. There is no significant increase in reliability for the "two MOV alternative" as compared to the "proposed design" (the failure probability decreases from 0.0419 to 0.0362). This is due to the fact that the human error probability remains essentially unchanged (0.036) while the hardware failure probability of the "proposed design" is already low (about $6x10^{-3}$).

Similar insights were drawn about the reliability of these designs assuming no offsite power is available.

The staff reviewed the reliability study performed by the licensee and found it to be reasonable. In addition, a sensitivity study performed by the staff, indicated that the reliability of the "proposed design" remains comparable to the reliability of the "two MOV alternative" even when the human error probability values are significantly smaller (up to an order of magnitude) than those assessed by the licensee.

Conclusions Regarding the Licensee's Risk and Reliability Assessments

The staff reviewed PNPP's submittal which included a discussion of the risk impact of the proposed change as well as information from a reliability study which compared the reliability of the current feedwater line isolation design to the reliabilities of two alternative designs, one of which was the proposed modification. The major findings of the staff's review are summarized below:

1. The proposed feedwater line isolation change does not have any impact on the plant's CDF because it is related to containment isolation following core damage (Level 2 PRA).

2. A comparison of the reliability of the current feedwater line isolation design to the reliability of the proposed modification shows that the proposed modification would most likely decrease the already low contribution to the plant LERF associated with a feedwater line water seal failure.

3. The frequency of core damage scenarios requiring a feedwater line water seal is very small (most likely smaller than $1x10^{-8}$ per year).

4. All three feedwater line water seal designs require operator action. The values of the human error probability dominate the reliability of all three designs. This implies that none of the three designs is "single failure proof" with respect to human error.

5. There is no significant difference in reliability between the "two MOV alternative," which is a fully "single failure proof" design with respect to active component failures, and the "proposed design."

A comparison of the reduction in the conditional failure probability to establish feedwater leakage control (establish a water seal) for the proposed design change to a change which would add an additional, independent MOV to each feedwater penetration line shows that the proposed design is comparable to a design which would add a second barrier to containment atmosphere leakage to the environment through the feedwater penetration line. The proposed design accounts for more than 90% of the available reduction in conditional failure probability, based on operator remote-manual closure of the MOVs and initiation of the FWLCS.

The likelihood of establishing a water seal in the feedwater penetration to prevent containment atmosphere leakage from a LOCA inside containment to the environment for the proposed design, based on a single barrier, is comparable to that of a design which would include a second MOV in the line as a second barrier. The proposed design also improves the likelihood of establishing the water seal, when compared to the current design, as a result of the increased time available for the operator actions needed to start the FWLCS and establish the seal within 1 hour.

Therefore, the staff concludes that the licensee's risk-informed discussion is consistent with the Commission's Safety Goal Policy Statement as documented in Regulatory Guide 1.174 and supports the proposed modifications.

3.4 Electrical Interface for Proposed Alternate Power Supply

The motor operated gate valves in each of the feedwater trains are currently powered from the Division I electrical power supply. In order to provide greater assurance that these MOVs will be available for closure following a LOCA and a total loss of both the normal and emergency Division I electric power supplies, the licensee proposed to install an alternate power supply from the Division III electrical power supply. Operator actions would be relied upon to manually connect the Division III power supply to the MOVs.

Evaluation of Electrical Connections

The licensee proposed to provide this alternate power supply by installing a power cable between Division III (motor control center EF1E1, Compartment V) and Division I (MCC EF1A07, Compartment XV). The cable will be terminated at the load side of a fusible disconnect switch in the MCC compartments at each end.

These disconnect switches will remain open and the fuses will remain out of their holders until such time as the alternate power supply is needed. The fuses will be stored in the bottom of the MCC compartments so as to be accessible as needed. Labels will also be applied to each compartment's door describing the purpose of the compartment and directing that the fuses are not to be removed from the compartment.

Prior to installing the fuses and closing the breakers for the alternate power supply, operators will be required to open MCC EF1A07's feed breaker at Bus EF-1-A, and all the breakers on MCC EF1A07. This will prevent potential back feed of power to other circuits. Specific plant procedures cover how to implement this alternate power supply. Only the feedwater MOVs will be operated using this alternate power supply. The cable and its conduit will be seismically qualified and classified as safety-related. The licensee has determined that the additional load for Division III under a LOCA condition is within the capabilities of the Division III diesel generator.

The staff was concerned about the potential of losing both electrical divisions when the Division III diesel is being used to close the MOVs and the offsite power was restored thus potentially powering all of Division I. The licensee states that once it has been determined that the Division I diesel is inoperable, procedures include steps to disconnect all loads upstream of the valves in Division I. After the lines have been disconnected from Division I power, the valves will be connected to Division III to close. This will provide electrical independence between Divisions I and III. If offsite power is restored during the 68-second period when the MOVs are being closed, it will not cause an adverse impact since the valves have been electrically isolated from Division I.

Conclusions Regarding Electrical Connections

The staff has evaluated the proposed design and procedure changes as follows:

1. During power operation, the power cable between Division III and Division I will not be used. This addresses any concerns regarding the potential for losing both divisions if they were tied together.

2. This special supply of Division III power will be limited only to circuits for MOVs B21-F065A/B and then only for approximately 68 seconds.

3. Alternate power supply (Division III) will be connected to the MOVs only when there is a complete loss of Division I offsite and onsite power sources. If the offsite power is restored during the 68-second period when the MOVs are being closed, it will not cause an adverse impact since the valves have been electrically isolated from the rest of Division I.

4. Existing physical separation and electrical independence between Divisions I and III will be maintained.

5. Additional loading for Division III under a LOCA condition is within the capabilities of the Division III diesel generator.

The staff concludes that connecting Division III to Division I can provide greater assurance that the feedwater motor-operated valves will be available for closure in the long term following a LOCA and it can be accomplished in an acceptable manner. Therefore, the staff finds the electrical interface acceptable.

3.5 Evaluation of Manual Operator Actions

The staff used the following guidance on manual operator actions and the time required to perform those actions to complete its evaluation:

1. Generic Letter (GL) 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability (1991)."

2. American National Standards Institute/American Nuclear Society (ANSI/ANS)-58.8, "Time Response Design Criteria for Safety-Related Operator Actions (1984)."

GL 91-18 states: "The consideration of manual action . . . must include the ability and timing in getting to the area, training of personnel to accomplish the task, and occupational hazards to be incurred such as radiation, temperature, chemical, sound, or visibility hazards." ANSI/ANS-58.8 provides guidance on estimating response times for operator actions and allows licensees to use time intervals derived from independent sources, provided they are based on task analyses or empirical data. Based on these guidelines, the NRC staff evaluated the licensee's evaluation of the new operator actions, as detailed below. Operator actions previously approved under the current licensing basis (i.e., closing the MOV and initiating the FWLCS) are still considered acceptable and are not being further addressed.

Specific Operator Actions Required

Operator action would be required to provide a method to use Division III power to operate the feedwater MOV's (normally powered by Division I) in the event that Division I power is lost following a LOCA. This alternative power supply is provided by the temporary installation of a power cable between Division III and Division I. This action was evaluated against the considerations in ANSI/ANS 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," to verify that the proposed contingency action can be accomplished. A draft procedure was prepared by the licensee, developed from several existing procedures to conduct a walkdown.

The walkdown showed that the time to perform all of the steps necessary to provide Division III power to the MOVs and to start the FWLCS without Division I power is approximately 19.5 minutes. This included the time for the operator to travel to the required area, obtain the required tools, and perform the required actions, plus time for the control room actions to take place. A Shift Supervisor reviewed the results of the walkdown and all the assessed times. The operator action was assumed to begin 30 minutes after the start of the design basis LOCA, based on the guidance in ANSI/ANS 58.8-1984. The operator action was then estimated to take approximately 19.5 minutes to complete. Therefore, the action of initiating FWLCS and establishing a water seal at the MOV can be completed within the current licensing basis period of 64 minutes, following the occurrence of a design basis LOCA.

Potentially Harsh or Inhospitable Environmental Conditions Expected

Environmental conditions in the area in which the operator actions will occur are not expected to be inhospitable or harsh. The time considered for the dose evaluation used the time to perform the entire evolution, even though some of the actions take place in the control room. The areas accessed are on elevations other than the control room. All areas requiring access are outside the Radiologically Restricted Area. Therefore, there are no components in the travel path containing radioactive materials that would result in radiation levels that would preclude access to the areas required to perform the proposed action.

Ingress/Egress Paths Taken by Operators to Perform their Functions

The areas required to be accessed to perform this action are readily accessible. The areas where the actions are being performed all have adequate normal lighting and also have battery backed emergency lighting.

Procedural Guidance for Required Actions

Discussions with the licensee indicated that the final plant procedure for the proposed manual operator action will be developed and verified prior to conducting operator training.

Specific Operator Training Necessary to Carry Out Actions Including any Operator Qualifications Required to Carry Out Actions

The licensee stated that operator training on the proposed manual operator action for all operators will be completed prior to the end of the outage in which this design change will be installed.

Additional Support Personnel and Equipment Required by the Operator to Carry Out Actions

The licensee stated that fuses needed for the proposed manual action will be accessible because they are stored in the bottom of the MCC compartments. Labels will also be applied to each compartment's door describing the purpose of the compartment and directing that the fuses are not to be removed from the compartment.

Description of Information Required by the Control Room Staff to Determine Such Operator Action is Required, Including Qualified Instrumentation Used to Diagnose the Situation and to Verify that the Required Action has been Successfully Taken

The licensee stated that successful diagnosis of the event could include either high radiation alarms or low feedwater pressure indication. The operators have 30 minutes to reach this diagnosis.

The staff concludes that the information discussed above is acceptable because it is consistent with Standard Review Plan guidance, ANSI/ANS 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," and Generic Letter 91-18. On the basis of the above information, the licensee has provided assurance that the required operator actions can be performed and therefore, the staff concludes that the licensee's responses related to the newly proposed operator actions, are acceptable.

4.0 SUMMARY

The licensee has determined that enhancements to the licensing and design basis of the feedwater isolation provisions are necessary to improve the reliability of the FWLCS to perform its safety-related function following a postulated accident.

The staff has determined that the proposed changes to the FWLCS to provide a water seal on the MOV seat to eliminate air leakage would perform the same function as the current FWLCS which is to fill a portion of the feedwater piping between the containment isolation valves. While the proposed design relies upon closure of the MOV, the licensee has shown that the reliability of the proposed design is comparable to a design which would include a second MOV in each feedwater penetration line. The proposed design also improves the likelihood of establishing a water seal within 1 hour as a result of an increase in the time available for the operator to take action.

In conclusion, the staff finds the proposed changes to the licensing and design basis of the feedwater isolation provisions to be an improvement over the existing design. The staff has conducted an extensive review that focused on the physical modifications and continued compliance with all applicable regulations. As previously discussed, the staff has concluded that the proposed modifications meet the appropriate acceptance criteria with respect to feedwater pipe breaks, containment isolation, and leak rate testing. In addition, the staff reviewed and approved the licensee's risk-informed discussion supporting the proposed modifications, the introduction of an alternate electrical supply for the MOVs, and the additional operator actions. Therefore, based on this review, the staff concludes that the proposed changes are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (63 FR 56262). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

7.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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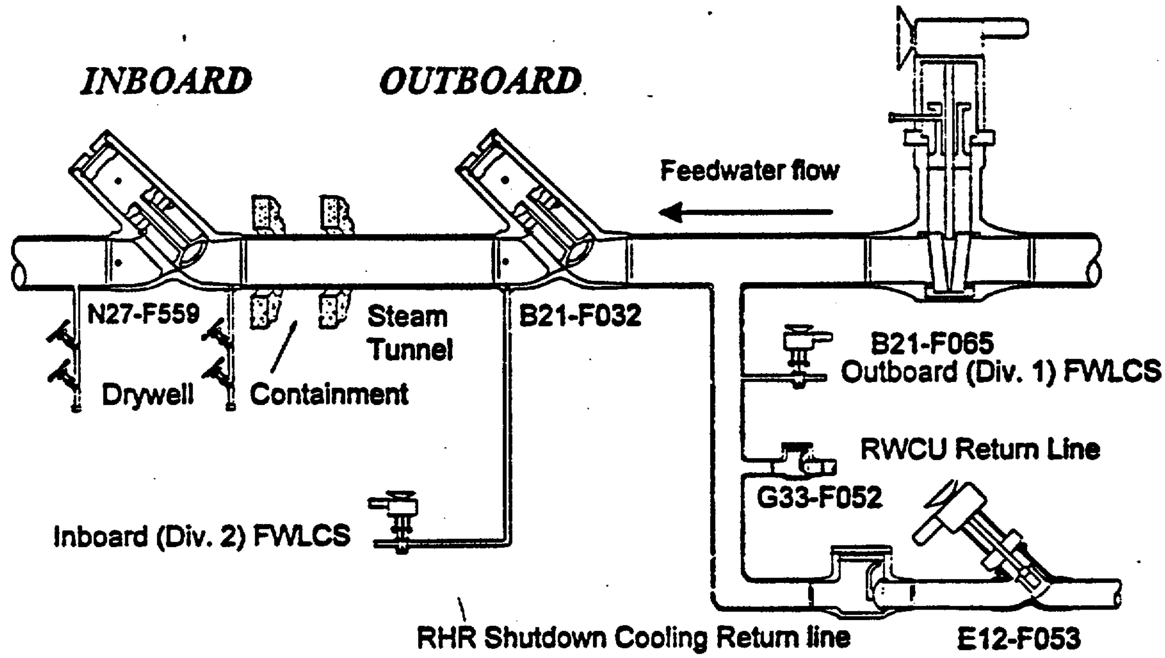

FIGURE 1 - EXISTING FWLCS CONFIGURATION

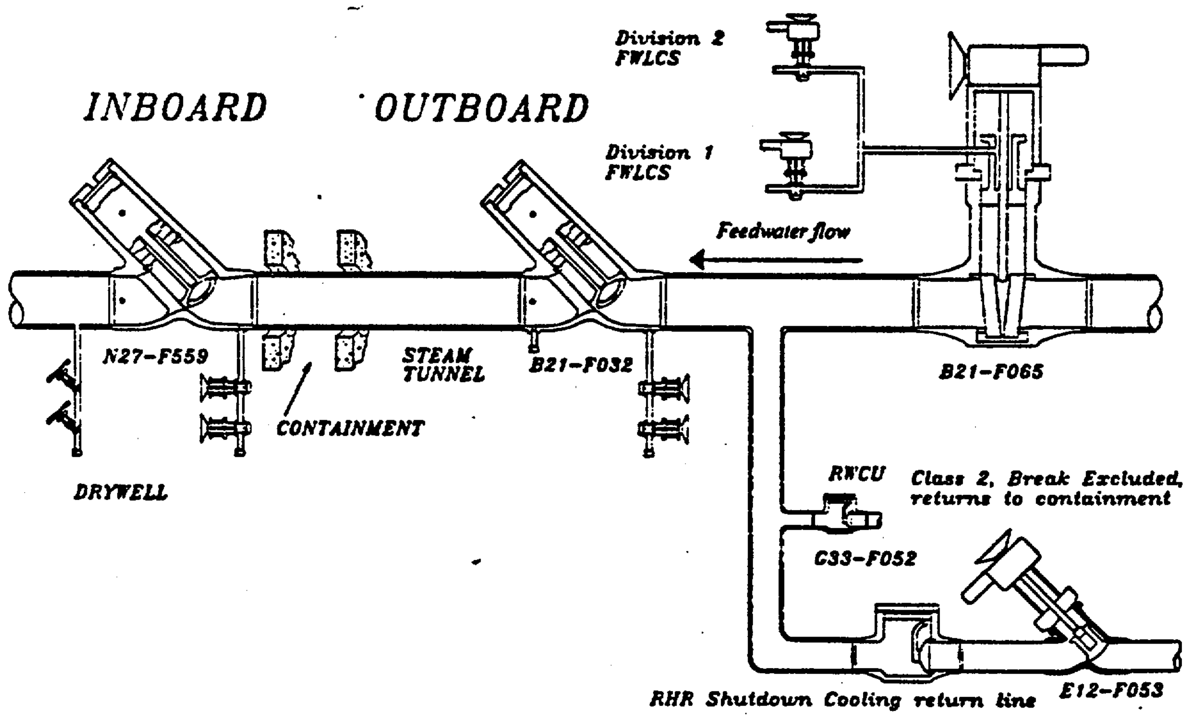

FIGURE 2 - PROPOSED FWLCS CONFIGURATION