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Perry Nuclear Power Plant
Docket No. 50-440
Feedwater Penetration Condition Report Investigation Summary

Ladies and Gentlemen:

By letter dated April 6, 2001, the Nuclear Regulatory Commission (NRC) noted that Perry Nuclear Power Plant (PNPP) staff was examining issues raised in a recent internal Condition Report on the Feedwater penetrations. These penetrations were the subject of License Amendment 105, dated March 26, 1999. An immediate investigation of the issues was completed prior to restart from refueling outage 8, in March 2001. This investigation concluded that the Feedwater penetrations were Operable and that plant restart was acceptable.

A more comprehensive investigation has since been completed. The April 6, 2001 NRC letter requested that when this subsequent investigation was complete, a disposition of each issue (and any additional actions that will be taken) be submitted for NRC information. Therefore, the attachment to this letter includes a summary of the investigation into the eight issues that were examined, including a listing of the corrective actions that have been entered into the corrective action process. The majority of the issues raised were determined to be adequately addressed, although it has been determined that an improved/supplemented method of testing and/or inspection is required to assure adequate verification of long term operability.

Although the attachment to this letter describes actions that will be completed per the PNPP corrective action process, those actions are not considered to be regulatory commitments. In addition to the attachment to this letter, the Condition Report investigation is available for review through the corrective action program's computer system, for onsite NRC inspection. Also, if desired, a meeting can be arranged to review this subject. If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

John K. Wood

Attachment

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

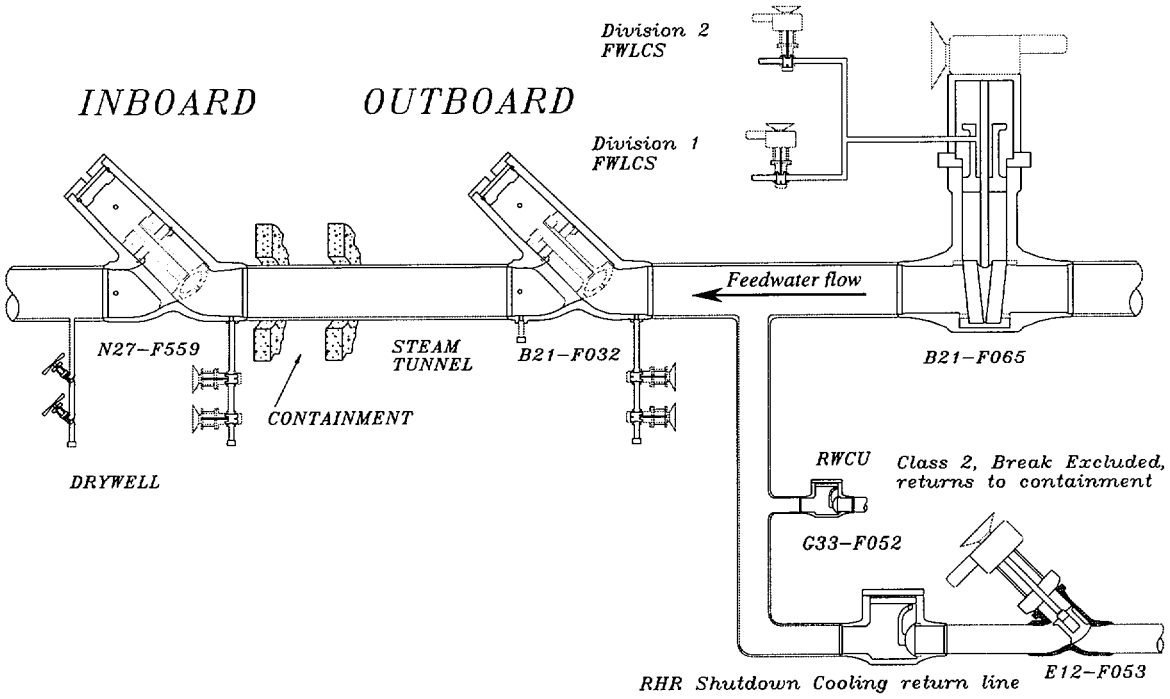
A001

FEEDWATER PENETRATIONS - EXECUTIVE SUMMARY OF THE FINAL INVESTIGATION

1. Introduction

Feedwater penetration Condition Report (CR) 01-0853 documented technical concerns with respect to Design Change Package (DCP) 98-0052 and License Amendment 105. The DCP was the product of a "Feedwater Check Valve Project" team. The project began in 1997; the License Amendment and the majority of the design changes coincided with Refuel Outage 7 (RFO7) in early 1999; and final closure of the DCP occurred during Refuel Outage 8 (RFO8) in early 2001.

Both the amendment and the DCP concern hardware modifications as well as changes to the inservice testing of the Feedwater penetrations P121 (A Side) and P414 (B side). The major hardware involved in the scope of this CR includes: the ASME Class 1, 20 inch controlled closure Feedwater check valves, 1N27-F0559A/B (inboard check valves) and 1B21-F0032A/B (outboard check valves), the ASME Class 2 motor-operated gate valves, 1B21-F0065A/B, a 12 inch Residual Heat Removal (RHR) branch line off of each train of the ASME Class 2 Feedwater piping (between 1B21-F0065A/B and 1B21-F0032A/B) and corresponding isolation valves consisting of check valves 1E12-F0050A/B and globe valves 1E12-F0053A/B. Lastly, an ASME Class 2, 6 inch Reactor Water Cleanup line, connects to each train of the RHR line between each Feedwater train and the first isolation valve of the RHR line.



The DCP reconfigured the Feedwater Leakage Control System (FWLCS) from filling the Feedwater piping volume between the 1B21-F0065A/B and the check valves, to directly injecting into the bonnets and between the gate valve discs of 1B21-F0065A/B.

License Amendment 105 obtained Nuclear Regulatory Commission (NRC) review of the proposed design change. The initial submittal was made in early September 1998. Several teleconferences, meetings and supplemental letters augmented the submittal. The supplemental letters were provided in January and March of 1999. The amendment was issued

on March 26, 1999. A post-amendment letter describing testing methods was submitted in mid-April 1999, and NRC issued a clarification letter on the amendment in late April 1999.

List of Issues Addressed in the Condition Report Investigation:

Issue 1: Feedwater Check Valve Testing Conformance With Appendix J
Issue 2: Feedwater Check Valve Testing Conformance With 10 CFR 50.55a And The ASME Code
Issue 3: Leakage Acceptance Criteria
Issue 4: Feedwater Gate Valve Capability To Close During High Pressure Transients/Accidents
Issue 5: Pressure Isolation Valve (PIV) Classifications And High-To-Low Pressure Interface Tests On Valves In The Feedwater Penetration
Issue 6: Closed Systems Outside Containment And Secondary Containment Bypass Leakage
Issue 7: Modification Is Not Single-Failure Proof
Issue 8: NRC Use Of Regulatory Guidance In Approving The License Amendment

2. Summary of Investigation Results

OVERVIEW

NOTE: Detailed analyses of the individual issues are included in the corrective action program's computer system, available for onsite review by NRC inspectors.

The Feedwater penetration DCP was addressing the repeated failure of the Feedwater check valves to pass their water leak tests. Prior to the DCP, these tests were designed to ensure the Feedwater piping volume could be filled post-accident. The Feedwater check valves were disassembled, inspected and the seats were polished or relapped, even though there were no problems evident. The check valves then retested satisfactorily. This was a very labor and dose intensive process, which was repeated each refueling outage after the valves failed their as-found tests.

The Feedwater Penetration DCP/License Amendment 105 therefore transferred the credited post-LOCA isolation of the Feedwater penetration from the filling of the pipe between the Feedwater check valves and the motor-operated gate valves, to the Feedwater gate valves and an RHR branch line with a new containment isolation valve/closed system combination. Thus, the Feedwater check valves no longer have a credited function during low pressure design-basis accidents/transients. For such events, the credited isolation on the Feedwater line itself is provided by the Feedwater gate valves, coupled with the Feedwater Leakage Control System. The remaining function of the Feedwater check valves is for high-pressure events, as described in the Updated Safety Analysis Report (USAR) as, "Should a break occur in a Feedwater line, the control closure check valves prevent significant loss of reactor coolant inventory and provide immediate isolation."

This CR raised issues with the final resolution of this DCP/license amendment. The most significant issues questioned the leak rate testing that is being performed on the check valves.

The concept of performing a leak rate test on the check valves was not the DCP 98-0052 project team's intent, as documented in the first two amendment submittal letters to the NRC. The leak rate test concept developed late in the time-line during final approval of the amendment, only

weeks before the start of RFO7, and it changed even further after the initial leak tests were performed during RFO7.

The project team's original intent was that the check valves would receive a visual inspection to verify closure in accordance with the Inservice Testing Program (ISTP). This would ensure the valves were capable of closing to their seat. It would also ensure, as noted in the DCP Safety Evaluation, that by maintaining the seating surfaces free of visual defects, the Feedwater check valves would be expected to provide a high integrity seal (Design Specification Leakage Rate) following the high dynamic seating forces associated with a high-pressure event. The project team was aware that leak rate tests performed during refueling outages at low back-pressures were not capable of fully seating the valves. The valve vendor had informed the team that actual leak tests had shown a back-pressure of at least 250 psid was necessary to fully seat this type of check valve and achieve consistent leakage results.

The testing personnel on the project team were aware that when Feedwater is shut down during a plant shutdown, there is very little if any back pressure to seat the valves, and that RHR and Reactor Water Cleanup (RWCU) flow is usually put through the valves, bumping them back off their seats again, with no reclosure differential pressure (dP). A previous 1997 Condition Report had already concluded that the low test pressures were the cause of test failures, not any problems in the valves themselves. The low test pressures were not at all representative of the high backpressures that would exist during a high pressure transient. Thus, the project team's premise was that leak rates experienced during low-pressure tests did not represent the much lower flow rates that could exist after a high-pressure event. The project team investigated possible ways to do a high pressure leak test, but could not ensure the conditions could be established to force the valves fully onto their seat. The reactor pressure vessel (RPV) leak test would provide sufficient pressure, but the project team considered testing during this test to be impractical for a variety of reasons. Consequently, in letters dated September 9, 1998 and January 6, 1999, the project team proposed the ASME Code Category C closure verification visual exam as the method of demonstrating the valves could provide their high pressure function. The goal was to eliminate costly and dose intensive disassembly and leak testing in favor of visual inspections.

In February of 1999, based on discussions with the NRC, the leakage integrity basis shifted from a visual exam to a water leak rate test. The impending scheduled outage start on March 27, 1999 affected personnel availability to evaluate alternative leak rate test methods that might be used in place of the visual exams. Objections were raised that it was too close to the outage to be making even apparently simple test changes (such as incorporating higher acceptance criterions and higher test pressures), and that any testing performed using the old test method might fail a valve that was actually acceptable as-is. Leak rate tests had to be performed on the check valves in order to obtain NRC approval of the license amendment. This was committed to in a letter dated March 4, 1999.

A high leak acceptance criterion was chosen, to ensure that acceptable valves would not fail the low pressure, non-fully-seated test. The 200 gpm per line criteria was taken from a calculation that had been previously performed as a sensitivity study. The purpose of the calculation had not been to establish a test acceptance criterion. Its purpose was to determine what leakage on a mass basis following a Feedwater Line Break event outside containment was equivalent to a Main Steam Line Break mass release.

The license amendment was issued on March 26, 1999. Refuel Outage 7 began on March 27, 1999. In the final days of the approval process, the leakage integrity basis had shifted away from the visual exams to the leak tests, yet the intent remained that the leak test

would provide the long-term assurance of leak tight integrity of the check valves for their high-pressure event isolation function. The concept was that the leak tests were still going to satisfy the ASME Code Category C ISTP "Exercise Closed" requirements. The primary issues from this Condition Report (Issues 2, 3, 4 and 5, with the primary focus being on Issue 3) stem from this mindset that the testing (now a leak rate test rather than a visual exam) would continue to ensure long-term, high leak tight integrity of the checks. (Note: See page 2 of this Executive Summary for the subjects of the various issues.)

OPERABILITY

Although the Investigation identified actions that should be taken to better ensure long-term (40 to 60 year life of the plant) leak tightness of the Feedwater check valves during high pressure events, the conclusion is that the Feedwater penetrations, including the check valves, are currently Operable. The basis for this conclusion is provided below.

The underlying issues raised by Issue 3 of whether high leakage at high pressure was likely to occur were addressed either during the review and approval of the License Amendment, or during communications with the Nuclear Regulatory Commission (NRC) staff following issuance of the amendment. Issue 3 essentially postulates a scenario of a defect developing in the valves that would prevent them from seating tightly in a high-pressure event. Such a scenario could result in leakage that could lead to undesirable results in piping outside of the containment. This is not considered to be an issue affecting the capability of the Feedwater check valves to currently perform their function. The test methodology being utilized was reviewed with the NRC. The issue of why the low-pressure leak test results being obtained should be conservative with respect to high-pressure event leakage was also discussed in a docketed letter to the NRC, dated April 14, 1999. In addition, as documented in the investigation for a 1997 Condition Report (which evaluated the results of check valve inspections in RFO 3, 4, 5, and 6), the check valves have not had guide rib wear noted on their hard-faced guide ribs, and the seats have only had minimal wear, with no orifice-type defects due to Feedwater system operation. The investigation team's conclusion is that the Feedwater penetrations remain Operable, based on the existing testing and justifications provided for that testing, the NRC review and approval of the existing testing methods, and the judgement that the check valves in the penetration are not likely to have developed defects (orifices) in their seats or extensive guide rib wear during the last several operating cycles. Therefore, the Feedwater check valves and the Feedwater penetrations remain within the licensing basis, and in addition, the valves are judged to currently be capable of performing their design function to seat tightly during high pressure events and are therefore Operable.

The investigation for this Condition Report does conclude that for the future (for long-term, i.e. RFO9+), further evaluations (per the corrective action program) are needed to determine what additional testing, visual examinations, or other means are needed to maintain long-term confidence in the check valves continued operability for high pressure events. This conclusion does not imply that the valves are currently inoperable or will become inoperable during the current operating cycle.

BRIEF SUMMARY OF INDIVIDUAL ISSUES

SUMMARY OF ISSUE 1: "Feedwater Check Valve Testing Conformance With Appendix J". This issue did not identify any issues that were not acceptably addressed in the DCP / license amendment. Since PNPP is an Appendix J Option B plant, an exception was obtained to Regulatory Guide 1.163 "Performance-Based Containment Leak-Test Program". This permits check valve testing as an Inservice Test Program (ISTP) water test, versus the air test that

would otherwise be required by the Appendix J regulatory guidance. There was an ancillary issue discussed on the wording in the NRC Safety Evaluation (SE) for the amendment about the water test's conformance with Appendix J, but this is considered to be resolved by the NRC clarification letter dated April 27, 1999.

No corrective actions were required for this issue.

SUMMARY OF ISSUE 2: "Feedwater Check Valve Testing Conformance With 10 CFR 50.55a And The ASME Code". Throughout the development of the modification, the project team was focused on the Feedwater Check Valves (FWCVs) receiving an ASME Code Category C "Exercise Closed" visual inspection. This visual inspection was to provide confidence that the valves were moving to their closed position, and the valve seats had no visible damage. Thus, when the valves are seated with a high differential pressure very little leakage was expected. However, late in the NRC review process, the focus shifted to a low-pressure seat leakage test per the Inservice Testing Program (ISTP). This introduced two of the Code compliance issues that were investigated for this CR (Issue 2). Also, after RFO7 had already begun, when the standard "collection" seat test method failed, a new alternative test/evaluation method was developed per the corrective action program. This introduced a third Code compliance issue that was investigated.

The 3 Code compliance issues that were considered are summarized as:
- Category A Valves = Appendix J Tests (Code Section 4.2.2.2)
- Factor of 10 Adjustment Factor[1] for leak tests performed at pressures below function pressures (Code Section 4.2.2.3(b)(4))
- Three Code Approved Test Methodologies (Code Section 4.2.2.3(c))

Code personnel and the investigation team reviewed the above issues and concluded that the Code is being complied with, and that Relief Requests are not necessary.

For the first Code issue considered, this conclusion was based on requirements in Option B of Appendix J, and the documentation in the Technical Specifications of the exception to Regulatory Guide 1.163 for the check valve testing. Option B does not contain specific testing criteria, but requires the implementation approach to be "included, by general reference, in the plant technical specifications". This requires that if the licensee chooses to deviate from methods endorsed by NRC in a regulatory guide, that the exceptions should be documented in the technical specifications. This was done for the Feedwater check valve testing exception, so the requirements of Appendix J Option B were met, and Code Section 4.2.2.2 was therefore met.

For the second and third Code issues considered, it was determined that there are other components in the Feedwater system that contribute to performing the Code defined function of "reactor coolant system pressure isolation" to prevent overpressurization of connected lower-pressure piping. This is explained in more detail in Issue 5 below. As a result, alternative testing methods other than those specified in Code Section 4.2.2.3 are acceptable when testing the Feedwater check valves.

Although Relief Requests were not determined necessary, the three corrective actions listed below were generated to continue to ensure that overpressurization of the lower-pressure-rated portions of the Feedwater system does not become an issue in the future.

[1] The "Factor of 10 adjustment factor" concept is explained more fully in Issue 3.

An action was generated to re-institute a test on the check valve or gate valve in the bypass line around the main Feedwater pumps. Another action is for the Periodic Test on the Feedwater pump discharge check valves to be clarified with respect to its scheduling, acceptance criterion and rework requirements. The third action is for the Inservice Testing ASME Section XI Valve Program Basis Document and the Performance Based Leak Testing Program to be updated by the Plant Engineering Section (PES) to reflect the current methodology.

SUMMARY OF ISSUE 3: "Leakage Acceptance Criteria", identifies that the 200 gpm per line (400 total) check valve leak test acceptance criteria could permit check valve leakage that could exceed the project team's assumptions. The project team assumed that the tests at low pressure result in higher leakage than would be seen at high pressure, since the valves would seat with very limited leakage under the dynamic forces associated with high pressure events. However, should an orifice type defect develop in the valve seat, higher leakage may result (this is referred to as the "Factor of 10" issue, i.e., to calculate potential leakage past an orifice defect at high pressure, the low pressure test results are adjusted using an engineering formula (the square root of the ratio of accident pressure to test pressure – in this case, approximately a factor of 10 adjustment). In other words, rather than leaking less at higher pressures, a valve with an orifice type defect could leak more than the low pressure leak test results indicate. Evaluation has shown that if such an orifice defect were to develop, it might not be identified with the current low pressure testing method. Thus the design basis 200 gpm per line leakage value might be exceeded during a high pressure event if such a defect was to develop. This orifice leakage concept raises issues that need to be addressed for the future. Without additional testing, visual inspections, or analysis, the issues of check valve seat orifice leakage and guide rib wear would raise questions of continued long-term operability. Confidence from additional tests or inspections would eliminate all of the issues discussed below.

Postulation of increased leakage related to the "Factor of 10" issue led to the following potential issues. As described in the "Operability" discussion above, these are not concerns because the Feedwater check valves are judged to be capable of performing their design function to seat tightly during high pressure events, but could become concerns in the future if additional tests or inspections are not performed:

- Possible total leakage higher than postulated in the Feedwater Line Break sensitivity study
- Possible impacts on analysis of high-pressure transients involving losses of Feedwater, such as the Feedwater Controller Failure or Loss of Feedwater flow transients
- Possible overpressurization of lower pressure piping upstream in the Feedwater/Condensate Systems, which could change a simple transient into a Feedwater Line Break (resolved in Issues 2 and 5)
- Introduction of a void in the Feedwater piping following a high-pressure event where water level goes below Level 2 (which uncovers the FW nozzles)
- Potential PSA model changes

Therefore Issue 3 resulted in a corrective action to change the Feedwater check valve test method and acceptance criteria to ensure the issues identified above do not become concerns in the future.

Two other corrective actions also resulted. One recommends consideration of a change to the MOV program to maintain currently existing margin in gate valve closure capability to ensure initiation of RHR Shutdown Cooling after 2 hours. As a separate but related issue, a post-scram iodine spike is considered in another USAR event (instrument line break) which also postulates leakage occurring for several hours after an event. The Feedwater line break sensitivity analysis did not include such an iodine spike. Therefore a corrective action is included to perform design

interface review to determine if an iodine spike should be included in the Feedwater line break analysis and to update the licensing basis accordingly.

SUMMARY OF ISSUE 4: "Feedwater Gate Valve Capability To Close During High Pressure Transients/Accidents", is related to the above discussion in Issue 3. The higher postulated leakage during a high pressure event due to the "factor of 10" orifice defect issue (from 200 to 2000 gpm/line) could lead to the conclusion that the 1B21-F0065A/B valves might need to be closed during high pressure events to stop excessive leakage. This issue is addressed by the Issue 3 corrective action for additional testing or visual inspections in future outages to eliminate the orifice leakage concern for the check valves.

Independent of additional testing, the 1B21-F0065A/B valves have significant capability for closure within 1-½ hours after a high-pressure event, at a Tech Spec maximum cooldown rate of 100 degrees per hour. Closure capability could be increased through MOV Test Program changes, such that the gate valves could be closed within approximately an hour. Currently, no credit is taken for a one hour closure of the valves in the licensing basis for high-pressure events (they are only credited to close during the low-pressure LOCA). The System Operating Instruction for RHR (SOI-E12) does direct their closure for initiation of RHR Shutdown Cooling, which would occur within 2 hours at the 100 degree F per hour rate. Plant procedures direct isolation of primary coolant leakage outside containment.

No corrective actions were required for this issue.

SUMMARY OF ISSUE 5: "Pressure Isolation Valve (PIV) Classifications And High-To-Low Interface Tests On Valves In The Feedwater Penetration", is also related to Issue 3. A concern was investigated as to whether high postulated leakage could possibly result in overpressurization failures in the Feedwater/Condensate Systems.

Regulatory guidance on Pressure Isolation Valves (PIVs) focuses exclusively on normally closed valves in standby systems that connect to the reactor coolant pressure boundary (RCPB), and not on the Feedwater checks. Based on discussions with leak rate test consultants that test at various plants, and additional phone contacts, the investigation team is not aware of anyone in the industry that tests their inboard and outboard Feedwater check valves for high pressure leakage (nor do they have a Code Relief Request to exempt them from the Code requirement to do a water test and use the orifice adjustment factor).

The concern as to whether leakage could lead to an overpressurization failure of the lower pressure piping, thereby changing a normal transient (loss of Feedwater) into an accident (Feedwater Line Break outside containment), led the CR investigation team to examine this issue more closely to determine if it has been adequately addressed.

It was determined that this issue has been "asked and answered", and the fact that the PIV definition does not envelope the Feedwater check valves is not a concern. This conclusion is based on the following:

- The industry had already identified the possibility for overpressurization events in Feedwater/Condensate and had proposed how to deal with the issue in INPO SOER 86-03.
- There are components in the Feedwater/Condensate systems other than the Feedwater check valves that serve to prevent overpressurization failures of the piping.
- The piping upstream of the main Feedwater pumps (the side leading back to the Condenser) is protected by the main Feedwater pump discharge check valves, which are tested each cycle to ensure their integrity, in response to the industry recommendations in SOER 86-03.

- The piping which is designed for normal pressures of 500 psig is recognized per piping standards to be fully capable of withstanding occasional loadings of 1100 psig.
- The piping which is designed for normal pressures of 145 psig is adequately protected by large relief valves back to the condenser.

In addition to all the above points, this is not considered to be a concern at the present time since the Feedwater check valves are considered Operable and capable of performing their high-pressure isolation function. For the future, the corrective action from Issue 3 will also provide continued confidence in the Feedwater check valves ability to seal tightly during high pressure events. The Feedwater check valves therefore will also continue to be another barrier to overpressurization of lower pressure Feedwater and Condensate piping. No additional corrective actions beyond those required by Issues 2 and 3 are deemed necessary to address this issue.

With respect to the branch lines leading to RHR Shutdown Cooling, and back to the keepfill pumps through the Feedwater Leakage Control System, there is not a concern over these high-to-low system interfaces, since a test equivalent to a PIV test (5 gpm max) is performed on two isolation valves in each of these lines, even though these valves don't meet the literal definition of PIVs.

No corrective actions were required for this issue.

SUMMARY OF ISSUE 6: "Closed Systems Outside Containment And Secondary Containment Bypass Leakage", proved to be acceptable in all areas except for the need for two USAR updates. One of the issues raised in the CR was whether the RHR system constituted an acceptable closed system outside containment. This was found acceptable and in compliance with regulatory guidance on use of closed systems outside of containment. With respect to secondary containment bypass leakage, the primary issue raised in the Condition Report - i.e., draindown of one of the RHR loops due to a loss of one of the safety related keepfill pumps, resulting in secondary containment air bypass leakage - had already been addressed with the NRC during initial licensing. This issue was therefore considered to be acceptably addressed.

For the RHR branch line, the method in Amendment 105 of adding the leakage past the 1E12-F0053A/B valve seats into the leak test program totals was too conservative. A corrective action will revise the USAR to be more consistent with how other penetrations which lead to closed systems are treated.

Another bypass issue dealt with potential mechanical joint leakage external to the piping on the reactor vessel side of the 1E12-F0053 valves. It was determined that this should have been made a USAR/test program requirement in the amendment. The mechanical joints on the F0050 and F0053 valves were inspected during the high pressure vessel leak test walkdowns in RFO8, per procedural requirements, so this is a program update issue, rather than a field deficiency. A corrective action requires that the USAR/test program for the RHR branch line be updated to be consistent with the requirements that were placed on the RWCU branch line (zero mechanical joint leakage during the high pressure vessel leak test).

A question was raised as to whether inspections of the high pressure portion of RWCU piping with insulation installed is adequate for the "zero leakage boundary". This piping is break-excluded and only mechanical joints were committed to be inspected during high pressure vessel leak test walkdowns, which doesn't require insulation removal.

Also, a question was raised about how to treat leakage from these zero leakage boundary joints if the leakage developed after the plant is already at power. This would be addressed the same way that the plant currently addresses leakage from any outboard containment isolation valve mechanical joint. The leak would be repaired on-line if the valve were in an accessible area. The Leak Detection System will provide adequate assessment of leakage at power in inaccessible areas, and leaks in the external boundary of the Feedwater piping would likely result in a plant shutdown for repairs due to high radiation in those areas when the plant is at power.

SUMMARY OF ISSUE 7: "Modification Is Not Single-Failure Proof", was addressed by the license amendment. Although this is true, it was demonstrated that the net effect of the design change was an improvement in plant safety. The original FWLCS system was not truly redundant, even though it appeared to be. Success always depended on the closure of the motor-operated gate valves on the outboard subsystem. Human error potential is the most likely failure mechanism for initiation of FWLCS, far out-weighing mechanical failure. The FWLCS DCP reduced the original failure estimate from 27 percent to 4 or 5 percent. The concept of adding an additional gate valve was considered, but this only reduced the failure rate to a 3 - 4 percent failure estimate. Therefore addition of another valve to reduce mechanical failure susceptibility was not considered to be cost-justified. The NRC acknowledged this assessment. Also, addition of another gate valve would not have addressed the high pressure event leakage, since the gate valves are not the valves that provide the immediate closure capability on the penetration while the vessel is still pressurized.

No corrective actions were required for this issue.

SUMMARY OF ISSUE 8: "NRC Use Of Regulatory Guidance In Approving The License Amendment", discusses conformance to the General Design Criteria (GDC) and the Standard Review Plan (SRP) Sections 6.2.3, 6.2.4, and 6.2.6. GDC 55 addresses lines connected to the reactor coolant pressure boundary which penetrate containment. GDC 56 addresses lines connected to containment atmosphere which penetrate containment. Both were acceptably addressed in the license amendment.

Differences with Standard Review Plan (SRP) Sections 6.2.3, 6.2.4, and 6.2.6 were likewise found acceptable.

No corrective actions were required for this issue.

3. Causes Identified

The majority of the eight issues identified were determined to be Cause N/A (Not Applicable), based on the investigations performed.

Issue 3 was considered the focal point of the Condition Report. Issue 3, along with Issue 2, were determined to have a basic cause of "Design Modification Development - Inadequate Or Incomplete Design Aspects". Specifically, the potential impact of a leakage-based criteria on the design and license basis was not fully realized at the time it was instituted. This potential impact is attributed to inadequate review of the governing documents when changes were made to the project just prior to RFO7. Time pressure and unavailability of personnel due to the beginning of the refueling outage contributed to the inadequate reviews. A number of issues raised in this Condition Report were exacerbated by time pressure caused by the need to implement License Amendment 105 at the beginning of the refueling outage. This license

amendment was submitted to the NRC only six and a half months prior to the beginning of RFO7. Comment resolution of issues raised during NRC review caused the amendment to be issued only one day before the start of the outage. At this point, Engineering personnel were already committed to outage assignments. A longer lead-time for NRC review may have allowed many of these issues to be examined in greater breadth and depth. (Two completed corrective actions are identified below which address these issues).

Issue 6 had the same basic cause as Issues 2 and 3 above, namely "Design Modification Development - Inadequate Or Incomplete Design Aspects". Two corrective actions are assigned in Section 4 below to correct the USAR sections and clarify supporting procedures that address the RHR branch line leakage.

4. Corrective Actions/Enhancements

The completed Corrective Actions are:

First, the site has implemented use of the Project Review Committee (PRC) for review of License Amendment Requests before their development begins. This is a group of managers from across the site who determine project priorities, funding, and scheduling. The PRC ensures that the necessary funding, personnel resources, and scheduling are planned before undertaking a project.

Second, the pre-refuel milestone for submitting license amendment requests to the NRC was increased from 6.5 months to 9.5 months prior to the outage start. This allowed more time for the NRC to review submittals and resolve comments with the licensee. This proved effective for RFO8 submittals.
The future Corrective Actions are:

Broad scope issues raised by this CR are being addressed by a corrective action which requires:

> "The potential impact of a leakage-based criteria on the design and license basis was not fully realized at the time it was instituted. This potential impact is attributed to inadequate review of the governing documents when changes were made to the project just prior to RFO7. Lessons learned for these issues are to be addressed in Engineering Support Personnel (ESP) training for licensing and engineering personnel. The entire Project Team was not reconvened to formally evaluate the change in leakage criteria. Coordinate with training unit to have a "lessons learned" session added to ESP training."

Issue-specific future corrective actions are listed below.

ISSUE 1: FEEDWATER CHECK VALVE TESTING CONFORMANCE WITH APPENDIX J

- No Corrective Actions.

ISSUE 2: FEEDWATER CHECK VALVE TESTING CONFORMANCE WITH 10 CFR 50.55a AND THE ASME CODE

- Under the heading of FACTOR OF 10 ADJUSTMENT FACTOR FOR LEAK TESTS PERFORMED AT PRESSURES BELOW FUNCTION PRESSURE, two Actions:

1. Some form of testing of the parallel (bypass) piping path around the main Feedwater pumps needs to be re-instituted; either a "disassemble and inspect" on check valve 1N27-F0515, or some other check to ensure long term leak integrity of the check or the gate valve (1N27-F0200) in the bypass line.

2. Appropriate program changes should be made to ensure that rework is required on the turbine/motor-driven feed pump discharge check valves if the PTI acceptance criteria of "no rotation" is not met. Consideration should be given to specifying performance of the PTI during the plant shutdown process, to permit rework to be performed if needed. Also, explicit sign-off steps should be added into the PTI to provide better documentation that the 2 turbine-driven pumps do not rotate.

- Under the heading DOCUMENTATION OF METHODOLOGY, one Action:

  The Inservice Testing ASME Section XI Valve Program Basis Document and the Performance Based Leak Testing Program need to be updated by PES to reflect the current methodology. These would be the proper documents to identify the basis for changes related to License Amendment 105.

ISSUE 3: LEAKAGE ACCEPTANCE CRITERIA

- Under the heading IMPACT OF TEST PRESSURE AND POSTULATION OF AN ORIFICE, one Action:

  Change the Feedwater check valve test method and acceptance criteria, to ensure the issues identified in Issue 3 do not become concerns in the future.

- Under the heading SENSITIVITY CALCULATION - TWO HOUR DURATION, one Action:

  Although the corrective action for Issue 3 should ensure that the valves will not leak excessively in the future during a Feedwater Line Break, consideration should be given to revising the MOV Program for the Feedwater gate valves. To be consistent with the 2 hour duration in the Feedwater Line Break calculation, consider updating the MOV Program to require that at least a 135 psid (RHR Shutdown Cooling permissive) closure capability be maintained for these valves in the future.

- Under the heading SENSITIVITY CALCULATION - IODINE SPIKE, one Action:

  Perform design interface review to determine if an iodine spike should be included in the Feedwater line break analysis and update the licensing basis accordingly.

ISSUE 4: FEEDWATER GATE VALVE CAPABILITY TO CLOSE DURING HIGH PRESSURE TRANSIENTS/ACCIDENTS

- No Corrective Actions

ISSUE 5: PRESSURE ISOLATION VALVE (PIV) CLASSIFICATIONS AND HIGH-TO-LOW INTERFACE TESTS ON VALVES IN THE FEEDWATER PENETRATION

- No Corrective Actions.

ISSUE 6:

- Under the heading CLOSED SYSTEM OUTSIDE CONTAINMENT, no Corrective Actions.

- Under the heading SECONDARY CONTAINMENT BYPASS LEAKAGE, two Actions:

  1. Evaluate and revise, if determined necessary, USAR Table 6.2-40, Note 25 and supporting procedures to reflect that only one 1E12-F0053 valve should be included in the 0.6 La total, and then only if the Division 1 or 2 grouping has the largest leakage.

  2. Revise appropriate USAR leak rate testing Tables, the Plant Data Book Containment Isolation Valve Table, and supporting procedures to reflect the stem/bonnet exams on the 1E12-F0050 and 1E12-F0053 valves.

ISSUE 7: MODIFICATION IS NOT SINGLE FAILURE PROOF

- No Corrective Actions

ISSUE 8: NRC USE OF REGULATORY GUIDANCE IN APPROVING THE LICENSE AMENDMENT

- No Corrective Actions.