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Perry Nuclear Power Plant
Docket No. 50-440

Subject: Status Update on Issues Resulting From Review of a Condition Report on Feedwater Isolation Valves (TAC No. MB1905)

This letter provides an update on the status of actions associated with a Feedwater Isolation Valve Condition Report (CR) Number 01-0853, which was previously described to the Nuclear Regulatory Commission (NRC) in a letter dated August 30, 2001 (PY-CEI/NRR-2590L). The CR discussed issues associated with Amendment No. 105 to the Perry Nuclear Power Plant (PNPP) operating license. Amendment 105, dated March 26, 1999, involved changes to the design and licensing basis for the containment isolation valves of the Feedwater system.

The CR status update is provided in Attachment 1. This letter also responds to three additional questions asked by the NRC reviewer of the August 30, 2001 letter, which were forwarded in an NRC letter dated January 15, 2003. The responses to these three additional items are provided in Attachment 2.

There are no regulatory commitments in this letter or its attachments. If you have questions or require additional information, please contact Mr. Henry L. Hegrat, FENOC Fleet Licensing Supervisor, at (330) 315-6944.

Very truly yours,

Attachments:

1. Status Update on Condition Report 01-0853 Corrective Actions
2. Response to Three Additional NRC Items

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

A001

STATUS UPDATE ON CONDITION REPORT 01-0853 CORRECTIVE ACTIONS

A Nuclear Regulatory Commission (NRC) letter dated January 15, 2003, requested an update on several Corrective Actions for Condition Report (CR) 01-0853, which addressed issues associated with Amendment No. 105 to the Perry Nuclear Power Plant (PNPP) operating license. Amendment 105, dated March 26, 1999, involved changes to the design and licensing basis for the containment isolation valves of the Feedwater system. The results of the CR investigation were previously described to the NRC in a FirstEnergy Nuclear Operating Company (FENOC) letter dated August 30, 2001 (PY-CEI/NRR-2590L).

For ease of reference, the headers for each Corrective Action being updated are copied from the January 15, 2003 NRC letter, followed by the update. The corrective actions being updated were for Issues 2, 3 and 6 of CR 01-0853.

ISSUE 2: FEEDWATER CHECK VALVE TESTING CONFORMANCE WITH 10 CFR 50.55A AND THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS (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 (1N27-F0515) or the gate valve (1N27-F0200) in the bypass line.**

Update: This corrective action is complete. A Preventive Maintenance Task (1N27F0515, File 001) was created in June, 2002. This task re-instituted a periodic disassembly and inspection of check valve 1N27-F0515. The previous disassembly and inspection of this valve was in 1992. The new inspection requirement was added into the scope of Refuel Outage (RFO) 9 (spring 2003), and was completed successfully. The valve was in good condition. The responsible engineer has revised the periodicity of this repetitive task so it will be performed every third refueling outage, or approximately once every six years. Re-institution of this "disassemble and inspect" requirement on check valve 1N27-F0515 satisfies this corrective action.

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 two turbine-driven pumps do not rotate.**

   Update: This corrective action is complete. This Periodic Test Instruction (PTI) entitled "Feedwater System Pump Discharge Check Valve Operability" has been revised to include explicit sign-off steps to better document that the three feed pumps do not rotate when backpressure is applied to the pump discharge check valves. It also was revised to specify that disassembly and inspection will be performed on the turbine/motor-driven feed pump discharge check valves if they do not meet the no-rotation acceptance criteria in the PTI.

Also, the Integrated Operating Instruction (IOI) that controls the plant shutdown evolution was revised to include a step to perform this PTI during the shutdown process, at approximately 40% power. The performance of this PTI during the shutdowns for RFO9 and RFO10 was successful for the discharge check valves on the turbine and the motor-driven feed pumps.

Under the heading DOCUMENTATION OF METHODOLOGY, one Action:

1. **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 No. 105.**

   Update: This corrective action is considered complete. Upon further consideration, it was determined that better descriptions of the testing that became part of the licensing basis as a result of Amendment 105 would be more appropriate in the Updated Safety Analysis Report (USAR). USAR changes to better reflect the testing being performed have been processed, and will be included in the next USAR update.

ISSUE 3: LEAKAGE ACCEPTANCE CRITERIA

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

1. **Change the Feedwater check valve test method and acceptance criteria, to ensure the issues identified in Issue 3 [as described in the letter dated August 30, 2001] do not become concerns in the future.**

   Update: This corrective action is complete. Additional acceptance criteria and a contingent visual inspection have supplemented the testing committed to in Amendment 105.

   Issue 3 was the key issue in the Condition Report, specifically, the possibility that an orifice defect could develop in the Feedwater check valve seats which could result in higher than anticipated leakage during a high pressure transient event. A number of options were examined to determine the best method to detect an orifice defect. The selected option is to supplement the current licensing basis test[1] with a visual inspection of the seating interface of the Feedwater check valves to identify any such orifice defect(s)[2].

   The supplemental visual inspection can be performed using a fiberscope inserted into the body of the valve, through a drain connection on the bottom of the valve. In order to better view the seating surface and verify proper opening and closure of the check valves, a method was developed prior to RFO9 to lift the disc off its seat without having to disassemble the valve. A description of the supplemental visual (fiberscopic) examination of

[1] The licensing basis test is a low-pressure water leak rate test performed per the Inservice Testing Program (ISTP). More detail on how the current licensing basis tests are performed was provided in a letter dated April 14, 1999 (PY-CEI/NRR-2387L). That information is repeated in Attachment 2, in response to the first of the three 'additional' NRC items that were identified in the NRC letter dated January 15, 2003.

[2] This visual inspection is contingent on the results of the water leak rate tests - more detail on this contingency is provided in the following discussions.

the Feedwater check valves, and when it is required to be performed, has been added into the Inservice Testing Program (ISTP) Plan.

This supplemental fiberscopic inspection is performed if a valve's leakage rate during its low-pressure Inservice Test Program leak rate test exceeds a new acceptance criterion, established prior to RFO9. This leak rate criterion is much lower than the 200-gpm licensing basis value approved in Amendment 105. The new value is approximately 1/10th of the current 200-gpm licensing basis value. It was developed to satisfy the intent of ASME OM Part 10, Section 4.2.2.3.b.4, which notes that check valves can be tested at pressures below the service (function) pressure, provided the observed leakage is adjusted to the function pressure differential. This lower leak rate criterion is actually a curve, which can be roughly approximated (for discussion purposes), as 20 gpm at 1.1 P_a. The curve was calculated by engineering as being the low-pressure test result that could be extrapolated upwards to equal 200 gpm during a high-pressure transient.[3]

The supplemental fiberscopic inspection described above will ensure there are no significant valve seat orifice defects that could result in postulated leakage greater than 200 gpm during a high-pressure transient. Therefore, for valves that require a fiberscopic inspection, a successful inspection with no significant defects will continue to validate low-pressure test results of 200 gpm or less (at 1.1 P_a) as acceptable.

The frequency of the low-pressure 200 gpm water leak rate test on the Feedwater check valves (the licensing basis test) is fixed within the ISTP program at once every 2 years, with an extension of 25% permitted, so this test is performed every refueling outage.

During RFO9, the fiberscopic inspection on the four check valves provided an adequate view of the seating surfaces, but difficulty was experienced in lifting the disc on one of the four valves using the newly developed lifting technique. Therefore that check valve (1B21-F032B) was disassembled during RFO9, and the direct inspection of the disassembled valve also found the seat to be acceptable.

During RFO10, the fiberscopic inspections again found the check valve seats to be acceptable. Fiberscopic inspections were only necessary on three of the four valves. One valve (1N27-F559A) did not require inspection, because it passed its water leak rate test with results below the new, lower engineering acceptance limit discussed above.

As expected, the testing and inspections performed on the Feedwater check valves during RFO9 and RFO10 showed that no significant defects actually exist on the check valve seats. Previous investigations into Feedwater check valve leakage had concluded that the reason for past low-pressure leak rate test results being greater than 20 gpm at 1.1 P_a is the low back-seating pressure on these check valves during the leak tests. P_a at PNPP is 7.8 psig. The vendor recommends approximately 250-psid backpressure for initial seating, and 50 psid during the test. The visual inspection results in RFO9 and RFO10, which showed no orifice defects, helped to confirm this previous conclusion.

The leak tests combined with the visual inspections of the seats provide the necessary confidence that the check valves have not developed orifice defects that could lead to the concerns identified in Issue 3.

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[3] More information on this "Factor of 10" adjustment factor is provided in Attachment 2 in the response to the first of the three 'additional' NRC concerns that were presented in the NRC letter dated January 15, 2003.

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

1. **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 Motor-Operated Valve (MOV) Program for the Feedwater gate valves. To be consistent with the two hour duration in the Feedwater Line break calculation, consider updating the MOV Program to require that at least a 135 psid (residual heat removal shutdown cooling permissive) closure capability be maintained for these valves in the future.**

   Update: This corrective action is complete. The calculation which determines the differential pressure that the 1B21-F065 MOVs must be able to operate against has been revised to reflect a value (146 psid) greater than the 135 psid Residual Heat Removal (RHR) Shutdown Cooling permissive value. The supporting MOV Program calculation that utilizes this new value to determine the required closing thrust has also been revised. The 1B21-F065 valves' actual torque switch settings in the field have been and continue to be set to provide the capability to close these valves against the 135 psid Shutdown Cooling differential pressure.

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

1. **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.**

   Background: The Feedwater line break analysis in USAR 15.6.6 requires that the total mass of reactor coolant released from the Feedwater break be bounded by the "Main Steam Line Break Outside Containment". At the time the Feedwater design change was being developed, the USAR description of the Main Steam Line Break analysis did not discuss any type of an iodine spike, even though NRC Standard Review Plan Section 15.6.4 guidance indicates that a pre-accident iodine spike should be considered for a BWR Main Steam Line Break. Therefore it was appropriate to ensure the Main Steam Line Break analyses were revised to include an iodine spike. By including a pre-accident iodine spike in the Main Steam Line Break event, the simple mass comparison between the revised Feedwater event and the Main Steam Line Break remains valid and an iodine spike is addressed for both analyses. The second aspect of this issue was to determine if there is a requirement or a need to consider a post-accident iodine spike in the analysis for the Feedwater line break.

   Update: This corrective action is complete. Iodine spiking is now adequately addressed in the licensing basis calculations.

   With respect to consideration of a pre-accident iodine spike, because the Feedwater line break must be compared against the bounding Main Steam Line Break (MSLB), the appropriate resolution to this concern was to incorporate a pre-accident iodine spike into the MSLB analysis documented in the USAR. Since a 5% Power Uprate project was ongoing at the time this issue was identified, this issue was resolved through the calculations performed in support of the Power Uprate effort.

First, it was identified that an iodine spike for a MSLB had in fact already been incorporated into the PNPP licensing basis. This was verified by reviewing the original NRC Safety Evaluation Report (SER) for PNPP, dated May 1982. SER Section 15.3.4 entitled "Steamline Break Accident" describes the dose calculations performed by the NRC for PNPP, and Table 15.1 lists the resulting doses. Case 2 of the NRC MSLB analyses incorporated an iodine spike with a magnitude of 4.0 μCi per gram of dose equivalent I-131, per the procedures outlined in Section 15.6.4 of the NRC Standard Review Plan (SRP), "Radiological Consequences of Main Steam Line Failure Outside Containment". Once it was verified that the iodine spike version of the MSLB was part of the licensing basis, a Power Uprate version of the event was calculated. The calculation confirmed the dose result remained within the acceptance criteria listed in SER Section 15.3.4 and SRP Section 15.6.4. This version of the MSLB dose result which includes the required iodine spike has since been reflected into the USAR in Table 15.6-8, entitled "Steam Line Break Accident (Iodine Concentration in Coolant = 4.0 μCi/gm dose-equivalent I-131) – Radiological Effects".

With respect to consideration of a post-accident iodine spike, engineering review determined there are no specific requirements or regulatory guidance on the need to include this in an analysis for a Feedwater line break. However, simple evaluations can be performed to show the 4.0 μCi/gm iodine spike analysis discussed above (which is required by the regulatory guidance documents and which has been performed for PNPP) is sufficiently bounding to address a Feedwater line break event.

An iodine spike of 4.0 μCi/gm is bounding for an iodine spike that might occur following a Feedwater line break event. Even assuming the iodine levels during a plant shutdown spike as much as 500 times over the equilibrium values normally maintained at PNPP during plant operation, the resultant iodine spike would only reach 0.133 μCi/gm in the coolant[4]. This 0.133 μCi/gm spike is thirty (30) times below the 4.0 μCi/gm concentration spike assumed in the design basis MSLB event. Another example evaluation uses the Dose Equivalent I-131 values measured just prior to a mid-cycle fuel replacement shutdown in June 2000. Even with the elevated values that resulted in the shutdown, a post-event iodine spike of as much as 500 times would have only reached 0.4 μCi/gm[5]. Such a post-event spike would still be ten (10) times below the 4.0 μCi/gm iodine spike already included in the calculations.

In conclusion, the 4.0 μCi/gm iodine spike postulated in the MSLB calculation (which bounds the Feedwater line break event) meets the Standard Review Plan Section 15.6.4 guidance for a pre-accident iodine spike. Also, although there is no regulatory guidance directing consideration of an iodine spike following a Feedwater line break event, the 4.0 μCi/gm iodine spike can be seen to be sufficiently conservative to bound a post-accident iodine spike for a Feedwater line break. Therefore, the mass-equivalent comparison between the MSLB and the Feedwater line break performed as part of the Feedwater amendment remains valid.

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[4] 500 x 2.66 E-4 μCi/gm = 0.133 μCi/gm, where 2.66 E-4 is the average Dose Equivalent I-131 value for the period from January 2000 to September 2004.

[5] 500 x 8 E-04 μCi/gm = 0.4 μCi/gm, where 8 E-04 μCi/gm is the value the Dose Equivalent I-131 ramped up to just prior to the June 2000 fuel replacement plant shutdown.

ISSUE 6: CLOSED SYSTEMS OUTSIDE CONTAINMENT AND SECONDARY CONTAINMENT BYPASS LEAKAGE

Under the heading SECONDARY CONTAINMENT BYPASS LEAKAGE, two Actions:

1. **Evaluate and revise, if determined necessary, Updated Safety Analysis Report (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.**

   Update: The initial Condition Report investigation had concluded that the commitment made in the Feedwater License Amendment submittals was too conservative, and had recommended that the USAR and plant procedures be revised to incorporate a less conservative position. Upon further consideration, it has been determined that the USAR and procedures will not be changed from the Amendment 105 position. This retains the more conservative position. This corrective action is therefore considered to be closed.

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.**

   Update: This corrective action is complete.

   As noted at the end of the Issue 2 discussion, more descriptive information has been added into the USAR to describe the various tests being performed as a result of the Feedwater License Amendment. Changes to plant procedures, including surveillance instructions and the Plant Data Book Containment Isolation Valve Table, have also been completed to require examination for external leakage on the 1E12-F050 and 1E12-F053 valves.

   In a FENOC letter PY-CEI/NRR-2590L dated August 30, 2001, it was stated that this examination for external leakage on the 1E12-F050 and 1E12-F053 valves would be performed during the high pressure vessel leak test walkdowns (zero external water leakage from mechanical joints during the high pressure vessel leak test). This test was successfully completed on the mechanical joints of these two valves during RFO9 and RFO10.

   It has since been decided that examinations of the 1E12-F050 and 1E12-F053 valves for external air leakage will also be performed, as part of the air seat leakage test that is performed on the 1E12-F053 valves. The external portions of mechanical joints on these two valves, and the mechanical joints on several instruments in these same piping sections (pressure indicators R290 A and B, and pressure transmitters N145 A and B) are being tested for air leakage. Consistent with existing requirements, detected external leakage will be eliminated, or quantified and added to the secondary containment bypass leakage totals. This test was successfully completed on the mechanical joints during RFO10, with no air leakage detected.

   Similar to the discussion in the August 30, 2001 letter, insulation removal is not necessary on the remainder of the piping, because it is "break-excluded" and inspections of the welds in the break-excluded regions are completed through the Inservice Examination Program.

RESPONSE TO THREE ADDITIONAL NRC ITEMS

The NRC letter dated January 15, 2003, in addition to requesting updates on the Condition Report Corrective Actions, requested that three additional items be addressed. This attachment provides the responses to these three items:

NRC Item 1: While the Nuclear Regulatory Commission safety evaluation supporting Amendment No. 105 removed leak testing of the Feedwater check valves from Appendix J and placed it in the Perry Inservice Testing Program (ISTP), it was not done correctly. The Perry ISTP references ASME/ANSI OM Part 10. Section 4.2.2.2 of OM Part 10, Containment Isolation Valves, states that containment isolation valves shall be tested in accordance with Appendix J to 10 CFR Part 50. However, as a result of the staff's actions in Amendment No. 105, the requirements of Appendix J are not applicable to the Feedwater check valves.

Section 4.2.2.2 of OM Part 10 further states that containment isolation valves which "also provide a reactor coolant system pressure isolation function" (i.e., pressure isolation valves) shall be tested in accordance with Paragraph 4.2.2.3. Paragraph 4.2.2.3 contains specific requirements for ISTP valve testing such as test frequency, adjustments from the test pressure to the operating pressure, test methods and corrective actions if the tested valve(s) do not meet their acceptance criteria. Since the Feedwater check valves are containment isolation valves but not pressure isolation valves, the requirements of Paragraph 4.2.2.3 are also not applicable.

In summary, the staff's safety evaluation supporting Amendment No. 105 inadvertently failed to provide any alternatives to the leak testing requirements of either Appendix J or OM Part 10 for the Feedwater check valves. Therefore, please provide your standards for *test frequency, adjustments to the test pressure, test methods,* and *corrective actions* (emphasis added) for the Feedwater check valves. The adjustment from test pressure to system operating pressure described in OM Part 10 Section 4.2.2.3(b)(4) or an equivalent should apply unless otherwise justified.

Response: As noted by the NRC above, the water leak rate testing being performed on the Feedwater check valves per Amendment 105 is unique to these valves. The water leak rate test performed to identify check valve leakage exceeding 200 gpm, as committed to in the Amendment 105 process:

[1] is not a standard Appendix J test;
[2] is not a standard Pressure Isolation Valve test; and
[3] is not a standard ASME Code test;

but when combined with the new, lower leak rate acceptance criterion calculated by engineering and the contingent visual inspections, satisfies the intent of ASME OM Part 10, Section 4.2.2.3.b.4 for adjusting from test pressure to system operating pressure, and is within the licensing basis established in connection with Amendment 105.

NRC Item 1 requested that the following four topics be specifically addressed:

What are the Test Frequency Requirements? – The water leak rate test on the Feedwater check valves is the test that was committed to be performed to comply with License

Amendment 105. The ISTP Test Interval for the water leak rate test on the Feedwater check valves is "2-Year (2Y)" for each of the four Feedwater check valves. This is defined as:

> "Testing every two years (T/S 5.5.6 states at least once every 731 days with an extension of 25% being permitted [per] T/S SR 3.0.2)."

Therefore, this test is performed every refueling outage.

How is the Adjustment from Low IST Test Pressures to the System Operating Pressure addressed? – Issue 3 of Condition Report 01-0853 addressed the fact that the results of the low pressure tests (in the range of 8 to 30 psig) were not being adjusted up to the much higher system operating pressure. This "adjustment" concept is the subject addressed in OM 4.2.2.3(b)(4). Assuming the valve seats do not have orifice defects, the discussion in response to Question 5 of the April 14, 1999 letter holds true, specifically, the high differential pressures across the check valves during a high pressure transient such as a Feedwater line break would result in less leakage than measured during the low pressure test.

To address the intent of ASME OM Part 10, Section 4.2.2.3.b.4, an engineering calculation was completed prior to RFO9, which provides additional guidance on leakage rates past the Feedwater check valves. The calculation determined a new acceptance criterion which can accommodate the adjustment from the low IST test pressure to the higher system operating pressure (the "Factor of 10" adjustment factor discussed in the Condition Report). A curve is provided in the calculation that accounts for this "Factor Of 10" concept. In order to illustrate this concept, a rough approximation (for discussion purposes) of the values provided on the curve would be 20 gpm (20 is 1/10th of 200; where 200 gpm is the licensing limit for leakage past the check valves). If the low pressure leak rate test results for a check valve are less than the curve provided in this calculation, then the leakage for that valve at the system's normal operating pressure would be less than 200 gpm, and no further actions are necessary for that valve. Conversely, If the low pressure leak rate test results for a check valve are greater than the new curve, then it is necessary to verify that the check valve seating surfaces do not have an orifice defect that could produce the postulated 'higher leakage at higher pressures'. As discussed above for the corrective action for Issue 3, the ISTP currently requires this verification to be performed through a supplemental fiberscopic inspection of the check valve internals, including lifting of the check valve disc up off of its seat. If no significant defects are identified, this visual inspection validates that low-pressure test results below 200 gpm at 1.1 P_a are acceptable.

What is the Test Method for the Water Leak Rate Test? – The water leak rate test method was initially developed during RFO7 following issuance of Amendment 105, and was discussed with the NRC staff in a conference call on April 6, 1999. In addition, the methodology for the water leak rate test was previously described to the NRC in FirstEnergy letter PY-CEI/NRR-2387L, dated April 14, 1999. The following indented paragraphs provide the description from the attachment to that April 14, 1999 letter, from the responses to Questions 3 and 4. This test methodology was reviewed by the NRC staff, who noted in a letter dated April 27, 1999, that "The staff has reviewed the information in your letter and concludes that the revised test methodology provides an adequate means to demonstrate check valve leakage."

The test methodology description in the April 14, 1999 letter stated:

> "For the purpose of this description, the upstream side of the check valve is the side toward the feedwater pumps and the downstream side of the check valve is the side toward the reactor pressure vessel. Hydraulic principles indicate that leakage across

a check valve is proportional to the square root of the differential pressure across the valve. As the differential pressure increases, the leakage increases. For this application, since the change in pressures are small, the valve disk was assumed to not move (i.e., any gap between disk and valve seats remained a constant). When water is drained from the upstream side of the check valve, the pressure in the Feedwater pipe upstream of the check valve drops due to the check valve restricting flow. The pressure on the upstream side will drop until the differential pressure across the check valve (the driving force for the leakage) causes the leakage across the check valve to match the drain rate out of the test appendage. Once steady state is achieved, the flow past the check valve seat will match the flow out of the test appendage. Knowing the observed flow rate out of the test appendage and the observed differential pressure across the check valve, the total leakage that would exist across the check valve could be determined by a standard hydraulic calculation.

"The information obtained from the leak rate testing was analytically scaled down using hydraulic correlations to determine the leak rate that would exist at the check valve if it was subjected to the reference test differential pressure of 1.1Pa. Since no credit is taken for the check valve seating tighter at the higher test differential pressure, the analytical scaling of seat leakage is considered conservative.

"The alternate testing methodology is a more accurate means of determining leak rates. The methodology employs standard engineering hydraulic analyses to evaluate the data. The analyses include factors for test measurement uncertainty. Precision pressure gauges are used to obtain the differential pressure data. Administrative controls are in place to ensure the integrity of the collected data. These actions coupled with conservative manipulation of the raw data provide a high level of assurance that the evaluation conclusions were accurate."

One clarification to the above description is warranted. In the first sentence of the last paragraph, it states that the alternate test currently being utilized is a "more" accurate means of determining the leak rate. The method used prior to RFO7 was a standard ASME collection test. In examining the supporting information/comments on the letter dated April 14, 1999, which was written during RFO7, the input/comments focused on why the alternate method was "accurate" and "conservative", rather than "more" accurate. Points made as to why the new test method is accurate included:

- the exact configuration of the test apparatus, location of test gauges, elevations of gauges and hoses, and the instrumentation to use was specified in the procedure
- engineering personnel were involved with the test performance and data collection to ensure the tests were properly performed
- pre-activity briefings were held to thoroughly familiarize individuals with the test process
- the engineering analysis of the test data used common engineering methodology to evaluate the information gained from the test
- standard hydraulic principles were used to scale the test data to the required minimum test pressure (1.1 P_a)
- the engineering evaluation incorporated instrument and process uncertainty to further provide assurance that the derived value for total check valve leakage was conservative

The NRC letter dated April 27, 1999, makes mention of this "more accurate" wording from the April 14 letter, however, the final NRC conclusion about the adequacy of the testing being performed does not appear to rely on it. Specifically, a paragraph in the April 27 NRC letter which serves to introduce the issue about the revised test methodology states:

> "Your letter of April 14, 1999, described a revised test methodology that has been used for the feedwater check valves. Your letter further states that this revised test methodology provides a more accurate means of determining leak rates and that all feedwater check valves have demonstrated leak rates less than the 200 gpm acceptance criteria."

The next paragraph in the April 27 NRC letter, however, contains the actual conclusions of the NRC staff review, and it states:

> "The staff has reviewed the information in your letter and concludes that the revised test methodology provides an adequate means to demonstrate check valve leakage. In addition, the staff finds the leak rate testing for the feedwater check valves to be acceptable because they are within the acceptance criteria of 200 gpm per feedwater penetration."

The above discussion is provided to describe the test methodology currently being used as the alternate test method for the Feedwater check valves, and its acceptability for that application.

What Corrective Actions are Required? – Per Inservice Test Program requirements, if significant defects are observed during a fiberscopic visual inspection, or if leak rates measured during the low pressure leak tests correspond to values greater than 200 gpm at 1.1 P_a, additional investigation/repair and re-test will be completed.

Overall Summary for NRC Item 1: The water leak rate test for the Feedwater check valves is unique to these valves, but the test methodology and frequency are valid. The test methodology concept has been reviewed and approved by the NRC staff. PNPP reviews of the orifice defect issue (Issue 3) have resulted in development of a technique to satisfy the intent of OM 4.2.2.3(b)(4), even though this section of the OM Code is not directly applicable to these valves.

NRC Item 2: It is not clear that the break exclusion of the reactor water cleanup line was reviewed as part of Amendment No. 105. To qualify as a break exclusion line, the line must satisfy the seven criteria of Standard Review Plan 3.6.2, Branch Technical Position MEB 3-1, B.1.b.1 through 7. Please confirm that these criteria are satisfied.

Response: The Reactor Water Cleanup (RWCU) line is a break-excluded line, satisfying the PNPP design basis criteria for break exclusion, which are based on BTP MEB 3-1,1975. Because the PNPP criteria are based on the 1975 version of MEB 3-1, which contains six criteria rather than seven, the NRC Safety Evaluation Report (SER) for PNPP, dated May 1982, included a discussion of the seventh criterion, which had been added into the 1981 version of MEB 3-1. The seventh criterion addresses augmented inservice inspection of pipe welds. Section 3.6.2 of the SER noted:

> "the applicant has committed to perform a 100% ultrasonic examination on all welds.... The staff finds that the applicant's commitment meets the staff criteria for the augmented inservice inspection program."

Therefore, the seventh criterion was also determined to be satisfied.

Note that a USAR change to Section 5.2.4.9, which has recently been made "effective" and will be included in the next USAR update submittal, has identified that future inservice inspections are planned to be performed using the NRC-approved Risk-Informed ISI Methodology (TAC NO. MB1344, June 27, 2002).

The break-exclusion on the RWCU line was addressed as part of Amendment 105. The RWCU break-exclusion was noted in the letter dated January 6, 1999 (PY-CEI/NRR-2352L), on the Figure on page 5 of 26 of Attachment 1. This was also stated on page 15 of 26 of Attachment 1. This same confirmation was added into the USAR, as shown on pages 18, 19, 45, and 46 of Attachment 4 of the January 6, 1999 letter. The NRC staff acknowledged this break-exclusion as part of the reviews for Amendment 105. In the NRC Safety Evaluation for the amendment, dated March 26, 1999, in Section 3.0 "Evaluation", on page 7, it is noted that an RWCU branch line exists, which is analyzed for break exclusion.

NRC Item 3: Isolation of the Feedwater penetrations is not single-failure proof and relies on the closure of the Feedwater gate valves. The design changes to the Feedwater gate valves in Amendment No. 105 improve their reliability to close. However, if these valves cannot be closed by remote manual means, procedures should be in place for local closure of these valves. It is recommended that such procedures be available.

Response: The Feedwater System Operating Instruction (SOI) has been revised to add provisions regarding manual operator actions.

First, steps were added to allow for closure of Feedwater gate valves B21-F065 A & B from the local motor control center. The new steps use jumpers to bypass the remote-manual switch in the Control Room. These new steps could be implemented to close a gate valve from the local motor control center if Control Room remote-manual closure is unsuccessful.

Second, information was added directing the Emergency Plan Response Organization (ERO) to consider manual action to close a valve using its handwheel, in the unlikely event a motor-operator on one of these gate valves is not able to fully close the valve. The decision-making process about how to respond when any piece of equipment does not work as expected following an accident is one of the functions of the ERO and the Severe Accident Management advisors. Manual actions such as fuse replacement or valve closure using a hand-wheel are "skill-of-the-craft" tasks for personnel in the emergency response field teams. Sending personnel to close a valve using the handwheel would be carefully weighed by the emergency response teams, since the B21-F065 gate valves are located in the plant's Steam Tunnel. Several factors would be considered in the ERO decision: the temperature and radiological environment in that area of the Steam Tunnel at the time; how far closed the motor-operator got the valve; and the difficulty of closing these large, normally motor-operated gate valves using their handwheel.

It is unlikely the ERO will be faced with making this decision. As noted by the NRC above, design and procedural changes associated with these Feedwater gate valves due to Amendment 105 have improved their reliability to close, including supplying power from two separate safety-related electrical divisions. Therefore, should power be lost from Division 1, power can be aligned to these valves from Division 3 using this same SOI. Therefore the valves are not susceptible to the single failure of an electrical division, or of a diesel-generator. The risk-informed review of the Feedwater penetration configuration performed to support Amendment 105 [see letter dated January 6, 1999 (PY-CEI/NRR-2352L)] evaluated the single failure possibility and found it to be acceptably low.