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SOUTHERN COMPANY
Energy to Serve Your World℠

June 5, 2001

Docket Nos.: 50-348
50-364

NEL-01-0099

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Joseph M. Farley Nuclear Plant
Request For Technical Specification Changes
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves Located In The Residual Heat Removal Flow Path

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) proposes to amend the Farley Nuclear Plant (FNP) Technical Specifications (TS), Appendix A to Operating Licenses NPF-2 and NPF-8. This TS amendment clarifies a potential source of confusion introduced into the TS during the conversion to the Improved Technical Specifications (ITS).

As part of the conversion to the ITS, an asterisked note was deleted from the old TS on Table 3.4-1 that listed the Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs). The purpose of this note was to identify the valves to which the following surveillance requirement (SR) frequency applied: "Following valve actuation due to automatic or manual action or flow through the valve for valves identified in Table 3.4-1 by an asterisk." This frequency did not apply to the RCS PIVs in the Residual Heat Removal (RHR) flow path. In the documentation for conversion from the old TS to the ITS, the change was identified as an administrative change because the Mode 4 applicability and note 2 to SR 3.4.14.1 effectively provided the same exception. This proposed TS change clarifies that the above surveillance frequency does not apply to the RCS PIVs in the RHR flow path consistent with the requirements of the old pre-conversion Farley TS.

Enclosure 1 provides a basis for the proposed changes. Enclosure 2 provides the basis for a determination that the proposed changes do not involve significant hazards considerations pursuant to 10 CFR 50.92. Enclosure 3 provides a markup of the proposed changes to the TS. Enclosure 4 provides the clean typed version of proposed changes to the TS. Enclosures 5 and 6 contain markups and clean typed copies of the associated TS Bases changes. The Bases changes are submitted for information only and will be approved in accordance with the Farley Bases Control Program. Enclosure 7 contains copies of the pre-ITS Conversion TS 3/4.4.7.2 for Units 1 and 2.

A001

SNC requests that the NRC review and approve the proposed TS change prior to May 31, 2002.

SNC has reviewed the proposed amendment pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant increase in the amounts of effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed amendment satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment and the human environment is not affected by this amendment.

A copy of the proposed changes has been sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

Mr. D. N. Morey states that he is a vice president of SNC, is authorized to execute this oath on behalf of SNC and that, to the best of his knowledge and belief, the facts set forth in this letter and enclosures are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

D N Morey

Dave Morey

Sworn to and subscribed before me this 5th day of June 2001

Martha Gayle Dow

Notary Public

My Commission Expires: November 1, 2001

WAS/maf: RHR PIV Testing NRC.doc

Enclosures:

1. Basis for the TS Change
2. 10 CFR 50.92 Evaluation
3. Marked-Up Technical Specification Pages
4. Clean Typed Technical Specification Pages
5. Marked-Up Technical Specification Bases Pages
6. Clean Typed Technical Specification Bases Pages
7. Pre-ITS Conversion TS 3/4.4.7.2

cc: Southern Nuclear Operating Company
Mr. L. M. Stinson, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. F. Rinaldi, Licensing Project Manager – Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Enclosure 1

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

Basis for the TS Change

Enclosure 1

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

Basis for the TS Change

Description of Changes:

The proposed technical specification (TS) change modifies Surveillance Requirement (SR) 3.4.14.1 to clarify that the frequency "Following valve actuation due to automatic or manual action or flow through the valve" does not apply to Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs) in the Residual Heat Removal (RHR) System flow path. Appropriate Bases changes are made, along with minor editorial changes, in support of this change to reflect the historical licensing basis for these valves for Farley and to make the Bases internally consistent. The Bases changes are submitted for information only and will be approved in accordance with the Farley Bases Control Program.

Background:

There is a long docketed history related to PIV testing at Farley. A discussion of this history and a list of references which document the review of this issue are contained in the June 5, 1987 Alabama Power Company letter from R. P. McDonald to the NRC entitled "Joseph M. Farley Nuclear Plant - Units 1 and 2, Response to Generic Letter 87-06, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves." The September 22, 1980 Alabama Power Company letter from F. L. Clayton, Jr. responding to NRC letters from R. L. Tedesco dated August 25, 1980 and September 10, 1980 contained the basis for not testing the RHR Suction valves and Low Head Safety Injection (LHSI) Cold Leg valves after seat disturbances due to flow. Section 3.9.4 of Supplement 5 to the Unit 2 Safety Evaluation Report (SER) discusses the resolution of this issue at the time that the full power license was issued. Amendments 50 and 41 for Units 1 and 2, respectively, finalized and standardized the lists for PIV testing in the TS. A copy of the pre-ITS Conversion TS for Units 1 and 2 are included in Enclosure 7.

Discussion:

The applicability of LCO 3.4.14 is as follows: "MODES 1, 2, and 3, and MODE 4, except valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation." Surveillance Requirement (SR) 3.4.14.1 is currently required to be performed at two frequencies; "18 months, prior to entering MODE 2" and "Following valve actuation due to automatic or manual action or flow through the valve." SR 3.4.14.1 is modified by three notes. Note 2 states the following: "Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation."

As part of the conversion to the Improved Technical Specifications (ITS), an asterisked note was deleted from the old TS on Table 3.4-1 that listed the Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs). The purpose of this note was to identify the valves to which the following SR applied: "Following valve actuation due to automatic or manual action or flow through the valve for valves identified in Table 3.4-1 by an asterisk." RCS PIVs in the Residual Heat Removal

(RHR) flow path were not identified by an asterisk. In the conversion, this asterisked note was deleted and the STS RCS PIV LCO 3.4.14 Mode 4 applicability for valves in the RHR flow path and the STS LCO 3.4.14 note 2 to SR 3.4.14.1 were adopted. In the Discussion of Change (DOC) associated with that change, this was identified as an administrative change because the Mode 4 applicability and note 2 to SR 3.4.14.1 effectively provided the same exception as the asterisked note in the old TS. The NRC safety evaluation also characterizes this change as administrative. Administrative changes are defined in part as "editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions."

The ITS was implemented on March 1, 2000. Recently, some confusion has arisen concerning the relationship between the RCS PIV LCO 3.4.14 Mode 4 applicability for valves in the RHR flow path, note 2 to SR 3.4.14.1, and the frequency "Following valve actuation due to automatic or manual action or flow through the valve" as it relates to RCS PIVs located in the RHR flow path. This TS change clarifies, as discussed in the conversion to the ITS, that the above surveillance requirement does not apply to the RCS PIVs in the RHR flow path.

Summary

During the conversion to the ITS, a potential source of confusion was introduced into the TS. The proposed TS change modifies Surveillance Requirement 3.4.14.1 frequency to clarify that the frequency "Following valve actuation due to automatic or manual action or flow through the valve" does not apply to RCS PIVs in the RHR System flow path consistent with the requirements of the old pre-conversion Farley TS.

Enclosure 2

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

<u>10 CFR 50.92 Evaluation</u>

Enclosure 2

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

10 CFR 50.92 Evaluation

Pursuant to 10 CFR 50.92, SNC has evaluated the proposed amendment and has determined that operation of the facility in accordance with the proposed amendment would not involve a significant hazards consideration. The basis for this determination is as follows:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to Surveillance Requirement (SR) 3.4.14.1 clarifies that the requirement to test Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs) following valve actuation due to automatic or manual action or flow through the valve does not apply to PIVs in the Residual Heat Removal (RHR) flow path. This resolves a source of potential confusion and ensures that the testing requirements are implemented consistent with the historical licensing basis for Farley and the Improved Technical Specification conversion NRC Safety Evaluation Report. The valves will continue to be tested for back leakage every 18 months. The proposed change does not affect the consequences of a previously analyzed accident since the magnitude and duration of analyzed events are not impacted by this change. Thus, the consequences of a previously evaluated accident are unchanged.

   Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change involves no change to the physical plant. It allows for a clarification to the testing requirements to ensure that the historical licensing basis for Farley is maintained. These valves are tested every 18 months to ensure that the back leakage is within acceptable limits. This testing will continue. These changes do not impact the function of the valves.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

   The physical plant is unaffected by this change. The proposed change does not impact accident offsite dose, containment pressure or temperature, emergency core cooling system (ECCS) or reactor protection system (RPS) settings or any other parameter that could affect a margin of safety. The clarification of the testing requirements ensures that future testing is consistent with the historical licensing basis for Farley.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Conclusion

Based on the preceding analysis, SNC has determined that the proposed change to the Technical Specifications will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. SNC therefore concludes that the proposed change meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

Enclosure 3

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

Marked-Up Technical Specification Page

Affected Page

3.4.14-3

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	----------NOTES---------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. ---------- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig. (except for RCS PIVs located in the RHR flow path)	18 months, prior to entering MODE 2 AND Following valve actuation due to automatic or manual action or flow through the valve
SR 3.4.14.2	----------NOTE---------- Not required to be met when the RHR System valves are required open in accordance with SR 3.4.12.3. ---------- Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ 700 psig and ≤ 750 psig.	18 months

Enclosure 4

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

Clean Typed Technical Specification Page

Affected Page

3.4.14-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	----------NOTES---------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 4. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. ---------- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	18 months, prior to entering MODE 2 <u>AND</u> Following valve actuation due to automatic or manual action or flow through the valve (except for RCS PIVs located in the RHR flow path)
SR 3.4.14.2	----------NOTE---------- Not required to be met when the RHR System valves are required open in accordance with SR 3.4.12.3. ---------- Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ 700 psig and ≤ 750 psig.	18 months

Enclosure 5

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

Marked-Up Technical Specification Bases Pages

Affected Pages

B 3.4.14-3
B 3.4.14-6

3 or 5 gpm depending on the valve.

BASES

LCO (continued)

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of ~~5 gpm~~. The previous NRC Standard criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, on all PIVs listed in the TRM. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In order to satisfy ALARA requirements, leakage may be measured indirectly (as from performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with leakage criteria.

except for RCS PIVs located in the RHR flow path (Q1/2E11V001A and B, Q1/2E11V016A and B, Q1/2E11V021A, B, C and Q1/2E11V042A and B).

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed after the valve has been reseated. ~~An exception is that this surveillance is not required to be performed on the RHR System valves due to flow being passed from the normal shutdown cooling mode of operation.~~ The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested when RHR is secured and stable unit conditions and the necessary differential pressures are established.

INSERT 1

(continued)

INSERT 1

Leak rate testing is performed manually, with test personnel in the vicinity of the system connections in containment during setup and testing. Should the check valve that was being tested rupture or pressure in the system cause a rupture of the test equipment, there would be a concern for the safety of the personnel in the area. In addition, testing with RCS temperature above 212 °F would result in any leakage past the RHR valves flashing into steam making accurate measurement of the leakage rate impossible. Therefore, testing of the RHR System PIVs should normally be performed in Mode 5, as the test results are meaningful and plant conditions in Mode 5 minimize the potential impact on personnel safety.

Enclosure 6

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

Clean Typed Technical Specification Bases Pages

Affected Pages

B 3.4.14-3
B 3.4.14-6
B 3.4.14-7
B 3.4.14-8

BASES

LCO (continued)

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 3 or 5 gpm depending on the valve. The previous NRC Standard criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, on all PIVs listed in the TRM. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In order to satisfy ALARA requirements, leakage may be measured indirectly (as from performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with leakage criteria.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating except for RCS PIVs located in the RHR flow path (Q1/2E11V001A and B, Q1/2E11V016A and B, Q1/2E11V021A, B, C and Q1/2E11V042A and B). PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed after the valve has been reseated.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested when RHR is secured and stable unit conditions and the necessary differential pressures are established. Leak rate testing is performed manually, with test personnel in the vicinity of the system connections in containment during setup and testing. Should the check valve that was being tested rupture or pressure in the system cause a rupture of the test equipment, there would be a concern for the safety of the personnel in the area. In addition, testing with RCS temperature above 212 °F would result in any leakage past the RHR valves flashing into steam making accurate measurement of the leakage rate impossible. Therefore, testing of the RHR System PIVs should normally be performed in Mode 5, as the test results are meaningful and plant conditions in Mode 5 minimize the potential impact on personnel safety.

SR 3.4.14.2

Verifying that the RHR autoclosure interlock is OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of 600 psig. The autoclosure interlock isolates the RHR System from the RCS when the interlock setpoint is reached. The setpoint ensures the RHR design pressure will not be exceeded. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

The SR is modified by a Note that provides an exception to the requirement to perform this surveillance when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.3.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.3

Verifying that the RHR open permissive interlock is OPERABLE ensures that the RCS will not pressurize the RHR system beyond design of 600 psig. The open permissive interlock prevents opening the RHR System suction valves from the RCS when the RCS pressure is above the setpoint. The setpoint upper value ensures the RHR System design pressure will not be exceeded at the RHR pump discharge and was chosen taking into account instrument uncertainty and calibration tolerances. This value also provides assurance that the RHR System suction relief valves setpoint will not be exceeded.

The minimum value of the setpoint range is chosen based upon operational considerations (differential pressure) for the RCP seals and thus does not have a safety-related function. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

The SR is modified by a Note that provides an exception to the requirement to perform this surveillance when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.3.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. Technical Requirement Manual (TRM).
7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. 10 CFR 50.55a(g).

Enclosure 7

Joseph M. Farley Nuclear Plant
Reactor Coolant System Pressure Isolation Valve Surveillance Testing
For Pressure Isolation Valves located in the Residual Heat Removal Flow Path
Technical Specification Changes

Pre-ITS Conversion TS 3/4.4.7.2

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 420 gallons per day total primary-to-secondary leakage through all steam generators and 140 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2235 ± 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5 except that in lieu of any leakage testing required by Specification 4.0.5, each valve should be demonstrated OPERABLE by verifying leakage to be within the allowable leakage criteria of 0.5 gpm per inch of nominal valve size with an upper limit of the maximum allowable leakage in Table 3.4-1; and the measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%:#

a. Every refueling outage during startup.

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve affecting the seating capability of the valve.

c. Following valve actuation due to automatic or manual action or flow through the valve for valves identified in Table 3.4-1 by an asterisk.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

To satisfy ALARA requirements, leakage may be measured indirectly (as from performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

	VALVE NUMBER	DESCRIPTION	MAXIMUM ALLOWABLE LEAKAGE**
	Q1E11V001A	12" GATE	5.000 GPM
	Q1E11V001B	12" GATE	5.000 GPM
	Q1E11V016A	12" GATE	5.000 GPM
	Q1E11V016B	12" GATE	5.000 GPM
	Q1E11V021A	6" CHECK	3.000 GPM
	Q1E11V021B	6" CHECK	3.000 GPM
	Q1E11V021C	6" CHECK	3.000 GPM
*	Q1E21V032A	12" CHECK	5.000 GPM
*	Q1E21V032B	12" CHECK	5.000 GPM
*	Q1E21V032C	12" CHECK	5.000 GPM
*	Q1E21V037A	12" CHECK	5.000 GPM
*	Q1E21V037B	12" CHECK	5.000 GPM
*	Q1E21V037C	12" CHECK	5.000 GPM
	Q1E11V042A	10" CHECK	5.000 GPM
	Q1E11V042B	10" CHECK	5.000 GPM
*	Q1E21V076A	6" CHECK	3.000 GPM
*	Q1E21V076B	6" CHECK	3.000 GPM
*	Q1E21V077A	6" CHECK	3.000 GPM
*	Q1E21V077B	6" CHECK	3.000 GPM
	Q1E21V077C	6" CHECK	3.000 GPM

* Indicates the requirements of Section 4.4.7.2.2. Item (c) are applicable.

** The measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. Primary-to-secondary leakage through all steam generators shall be limited to 450 gallons per day and 150 gallons per day through any one steam generator.

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2235 ± 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5 except that in lieu of any leakage testing required by Specification 4.0.5, each valve should be demonstrated OPERABLE by verifying leakage to be within the allowable leakage criteria of 0.5 gpm per inch of nominal valve size with an upper limit of the maximum allowable leakage in Table 3.4-1; and the measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%:#

a. Every refueling outage during startup.

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve affecting the seating capability of the valve.

c. Following valve actuation due to automatic or manual action or flow through the valve for valves identified in Table 3.4-1 by an asterisk.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

To satisfy ALARA requirements, leakage may be measured indirectly (as from performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	DESCRIPTION	MAXIMUM ALLOWABLE LEAKAGE**
Q2E11V001A	12" GATE	5.000 GPM
Q2E11V001B	12" GATE	5.000 GPM
Q2E11V016A	12" GATE	5.000 GPM
Q2E11V016B	12" GATE	5.000 GPM
Q2E11V021A	6" CHECK	3.000 GPM
Q2E11V021B	6" CHECK	3.000 GPM
Q2E11V021C	6" CHECK	3.000 GPM
* Q2E21V032A	12" CHECK	5.000 GPM
* Q2E21V032B	12" CHECK	5.000 GPM
* Q2E21V032C	12" CHECK	5.000 GPM
* Q2E21V037A	12" CHECK	5.000 GPM
* Q2E21V037B	12" CHECK	5.000 GPM
* Q2E21V037C	12" CHECK	5.000 GPM
Q2E11V042A	10" CHECK	5.000 GPM
Q2E11V042B	10" CHECK	5.000 GPM
* Q2E21V076A	6" CHECK	3.000 GPM
* Q2E21V076B	6" CHECK	3.000 GPM
* Q2E21V077A	6" CHECK	3.000 GPM
* Q2E21V077B	6" CHECK	3.000 GPM
Q2E21V077C	6" CHECK	3.000 GPM

* Indicates the requirements of Section 4.4.7.2.2 Item (c) are applicable.

** The measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%.