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Ken Peters
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Waterford 3

W3F1-2003-0015

March 11, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License Amendment Request NPF-38-247
Relocation and Modification of Technical Specifications 4.0.5 and 3/4.4.9 and Extension of Reactor Coolant Pump Flywheel Volumetric Examination Interval

REFERENCES: NRC letter dated May 21, 1997, "Acceptance for Referencing of Topical Report SIR-94-080, Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements"

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment for Waterford Steam Electric Station, Unit 3 (Waterford 3). As part of an Entergy standardization effort, Entergy proposes to revise and relocate Surveillance Requirements 4.0.5 and 4.4.9 to the administrative section of the Technical Specifications under sections 6 5.8 and 6.5.7, respectively. Entergy also proposes to relocate Technical Specification 3.4.9, "Reactor Coolant System Structural Integrity" and its Bases to the Waterford 3 Technical Requirements Manual. The revision and relocation of Surveillance Requirements 4.0.5 and 4.4 9 to the administrative section is consistent with guidance contained in NUREG-1432, Revision 2, *Standard Technical Specification, Combustion Engineering Plants* and will bring this portion of the Waterford 3 Technical Specifications into alignment with the Technical Specifications at the other southern Entergy plants. Relocation of Technical Specification 3.4.9 is also consistent with NUREG-1432 in that it does not meet the 10 CFR 50.36 criteria for being in the Technical Specifications.

With the relocation of Surveillance Requirement 4.4.9 to the administrative section of Technical Specifications, Entergy also proposes to extend the Waterford 3 flywheel volumetric examination interval to ten years. Surveillance Requirement 4.4.9 requires performance of the reactor coolant pump flywheel inspections in accordance with the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, *Reactor Coolant Pump Flywheel Integrity,* Revision 1 (August 1975). Paragraph (1) of Regulatory Position C.4.b requires an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway at approximately three year intervals. Structural Integrity Associates, Inc. prepared Topical Report SIR-94-080-A, Revision 1, *Relaxation of Reactor Coolant Pump Flywheel*

A-001

Inspection Requirements, which provides the bases for the proposed change. The NRC staff reviewed and approved the topical report in the reference listed above.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that these changes involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

The proposed change does not include any new commitments. The NRC has approved similar Technical Specification changes for Arkansas Nuclear One, Unit 2 and Millstone Nuclear Power Station, Unit No. 2.

Entergy requests approval of the proposed amendment by October 1, 2003 to support deferral of the reactor coolant pump flywheel volumetric examinations from the Fall 2003 refueling outage. Once approved, the amendment shall be implemented within 60 days. Although this request is neither exigent nor emergency, your prompt review is requested.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 11, 2003.

Sincerely,

K.J. Peters
Director, Nuclear Safety Assurance
Waterford Steam Electric Station, Unit 3

KJP/DBM/cbh

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specification Bases Pages (For Information Only)

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
American Nuclear Insurers

Attachment 1

W3F1-2003-0015

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

By this letter, Entergy Operations, Inc. (Entergy) is requesting to amend the Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The proposed change will revise and relocate Technical Specification (TS) Surveillance Requirements (SR) 4.0.5 (inservice inspection and testing) and 4.4.9 (reactor coolant pump motor flywheel inspections) to TS 6.5.8 and 6.5.7, respectively to be consistent with NUREG-1432, *Standard Technical Specifications Combustion Engineering Plants*. It should be noted that NUREG-1432 addresses only the inservice testing program, while Waterford 3 SR 4.0.5 currently addresses both the inservice testing program and the inservice inspection program. For consistency with NUREG-1432, the proposed change will eliminate the inservice inspection portion of SR 4.0.5. This proposed change to SR 4.0.5 is an administrative change and no limiting conditions for operation, action statements or equipment specific surveillance requirements are being revised. The new TS paragraph numbers 6.5.7 and 6.5.8 are consistent with the location in NUREG-1432. A place keeper for paragraphs 6.5.1 through 6.5.6 is being added for future changes to the administrative section of TSs.

The proposed change will also relocate TS 3.4.9, Reactor Coolant System Structural Integrity, to the Waterford 3 Technical Requirements Manual (TRM.) This change is consistent with NUREG-1432 in that TS 3.4.9 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for being in TSs.

The proposed change will also revise the Operating License to extend the Waterford 3 reactor coolant pump (RCP) motor flywheel volumetric examinations from three years to ten years. This change is based on Topical Report SIR-94-080, *Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements*, Revision 1.

Marked-up TS pages reflecting the proposed changes are provided in Attachment 2. Marked-up TS Bases pages are provided, for information only, in Attachment 3.

2.0 PROPOSED CHANGE

2.1 Relocation of TS SR 4.0.5

TS SR 4.0.5a will be deleted.

TS SR 4.0.5b will become TS 6.5.8a and be reworded like NUREG-1432, Section 5.5.8a. The proposed TS 6.5.8a will not contain the inservice inspection activities currently contained in TS SR 4.0.5b. Nine month and biennial testing frequencies will be added consistent with NUREG-1432.

TS SR 4.0.5c will become the new TS 6.5.8b and will be reworded like NUREG-1432, Section 5.5.8b. Inservice inspection activities currently included as part of TS SR 4.0.5c will not be contained in the proposed TS 6.5.8b.

TS SR 4.0.5d will be deleted.

New TS 6.5.8c will be added. It will state, "The provisions of SR 4.0.3 are applicable to inservice testing activities, and", which is consistent with NUREG-1432.

TS SR 4.0.5e will become TS 6.5.8d. The wording is currently consistent with NUREG-1432, Section 5.5.8d.

The following TS SRs will be modified to reflect the relocation of TS SR 4.0.5 by replacing "Specification 4.0.5" or "TS 4.0.5" with "the Inservice Testing Program."

SR 4.1.2.3	charging pumps
SR 4.1.2.5	boric acid makeup pumps
SR 4.4.2.1	pressurizer code safety valves shutdown
SR 4.4.2.2	pressurizer code safety valves operating
SR 4.4.8.3.1b	over pressure protection relief valves
SR 4.5.2f	ECCS subsystems pumps
SR 4.6.2.1c	containment spray pumps
SR 4.6.3.3	containment isolation valves
SR 4.6.5	vacuum relief valves
SR 4.7.1.1	main steam line code safety valves
SR 4.7.1.2b	emergency feedwater pumps
SR 4.7.1.5	main steam isolation valves
SR 4.7.1.6a	main feedwater isolation valves

The reference to SR 4.0.5 will be deleted from SR 4.7.8 – snubber inspection program.

Index page XVI will be revised to reflect the changes made to TS section 6.5.

Additionally, the bases section associated with SR 4.0.5 will be deleted and references to SR 4.0.5 in other portions of the bases will be modified appropriately.

2.2 Relocation of TS 3/4.4.9 and Extension of Flywheel Examination Interval

Waterford 3 TS SR 4.4.9 states:

> In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

The proposed change will eliminate this SR and move it to Administrative Controls under Programs as a new section 6.5.7, "Reactor Coolant Pump Flywheel Inspection Program". The wording under this new flywheel inspection program will be revised to state:

> This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. The volumetric examination per Regulatory Position C.4.b.1 will be performed on approximately 10-year intervals.

The reference to SR 4.4.9 in the Limiting Condition for Operation for TS 3.4.9 is being removed and TS 3.4.9 and its associated Bases will be relocated to the TRM.

TS section 6.5 will be re-titled Programs.

A place keeper for paragraphs 6.5.1 through 6.5.6 is being added for future changes to the administrative section of TSs.

Index pages will be revised to reflect the changes made to TS section 6.5 and the deletion of TS 3.4.9.

2.3 Summary

In summary, SRs 4.0.5 and 4.4.9 will be revised and relocated to new sections 6.5.8, "Inservice Testing Program" and 6.5.7, "Reactor Coolant Pump Flywheel Inspection Program," respectively and TS 3.4.9 will be relocated to the TRM. The reactor coolant pump flywheel volumetric examination interval will be extended from three years to ten years. And, references to SR 4.0.5 throughout TSs will be modified to reflect the relocation of SR 4.0.5.

3.0 BACKGROUND

3.1 Relocation of TS SR 4.0.5

This change is desired for three reasons: 1) to eliminate the need to receive specific written relief from the NRC for alternatives to the American Society of Mechanical Engineers (ASME) Code of Record for Waterford 3; 2) to be consistent with other southern Entergy plant TSs; and 3) to be consistent with NUREG-1432.

Included in SR 4.0.5a is the requirement to receive specific written relief from the NRC staff for each proposed relief request. The proposed change, which is consistent with NUREG-1432, eliminates this requirement. With refueling outages becoming shorter in duration as a result of improved planning, the potential delay associated with the requirement to receive a written relief could impact the shorter schedule.

All southern Entergy plant TSs, except for the Waterford 3 TSs, have been amended to include the Inservice Testing Program as a program in the administrative section of TS. This change will align this portion of the Waterford 3 TSs with the other southern Entergy plant TSs.

3.2 Relocation of TS SR 4.4.9 and Flywheel Interval Extension

The reactor coolant pumps (RCP) circulate reactor coolant from the steam generators to the reactor. The pumps are powered by the RCP motor, which contains a flywheel that is attached to the pump shaft with an interference fit. The flywheel serves to increase rotating inertia of the RCP assembly. This increases the pump coast down time and reduces the rate of decay of coolant flow when power to the motor is lost. The coast down flow helps ensure that fuel design limits are not exceeded during an accident. The normal operating speed of the RCP motor is 1200 revolutions per minute (rpm) and the design overspeed is 1500 rpm.

The flywheels are designed to withstand normal operating conditions, anticipated transients, and the design basis loss-of-coolant accidents combined with the safe shutdown earthquake. Thus the following criteria are met:

- The combined primary stresses at normal operating speed do not exceed one-third of

the minimum specified yield strength or one-third of the measured yield strength in the weak-direction of the material.

- The combined primary stresses at design overspeed do not exceed two-thirds of the minimum specified yield strength or two-thirds of the measured yield strength in the weak direction of the material. Design overspeed is defined as 125 percent of normal operating speed.

Further discussion of the RCPs is provided in the Final Safety Analysis Report, Section 5.4.1.

10 CFR 50, Appendix A, General Design Criteria 4 (GDC 4) requires that nuclear power plant structures, systems, and components important to safety be protected against the effects of missiles that might result from equipment failures. The NRC staff has concluded that Regulatory Guide (RG) 1.14 *Reactor Coolant Pump Flywheel Integrity,* describes an acceptable method of implementing GDC 4 with regard to minimizing the potential for failures of the flywheels of RCP motors.

RG 1.14, Revision 1, Regulatory Position C.4.b requires an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway at approximately three year intervals. The RCP flywheels are augmented examinations contained in the Inservice Inspection (ISI) Program and are inspected three times within a 10 year period. These requirements are currently located in Waterford 3 TS SR 4.4.9.

SR 4.4.9 is conducted during the refueling/maintenance outages coinciding with the ISI schedule as required by ISI Plan. These flywheel inspections result in outage time, man-rem exposure and cost which may be minimized by use of a more carefully designed inspection program for the flywheels. Therefore, Entergy is proposing to extend the volumetric examination interval to ten years.

On April 4, 1995, Entergy submitted a TS change request (0CAN049504) for Arkansas Nuclear One Unit 1 (ANO-1) and Unit 2 (ANO-2) which proposed deleting the requirements for augmented inservice inspection of the RCP flywheels. This TS change request was based on information contained in the topical report entitled, *Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements* (SIR-94-080, Revision 1.) The submittal for ANO served as a lead submittal for other Combustion Engineering Owners Group (CEOG) plants participating in the topical report development. Structural Integrity Associates, Inc. (SIA) prepared the topical report to provide a basis for complete elimination of the RCP flywheel inspections for the CEOG plants. While the topical report review was in progress, Entergy requested and received two one-time amendments permitting deferral of upcoming outage based flywheel inspections for ANO-2. The staff subsequently approved the topical report on May 21, 1997 (as SIR-94-080-A) concluding that the flywheel inspection period could be extended from the current three years to ten years and that total elimination of flywheel inspections was not justified.

For the CEOG plants included in Topical Report SIR-94-080-A, Revision 1, the report concluded that:

- Inspections performed to date at these plants have not revealed the presence of any service-induced flaws. A survey of several other plants also revealed that service induced flaws has not been identified.

- Fatigue crack growth is the only degradation mechanism that affects service performance of the RCP flywheel. Analyses were performed that showed that fatigue crack growth is negligibly small even assuming a conservatively sized initial flaw.
- Flaw tolerance evaluations performed using conservative linear elastic fracture mechanics principles revealed that flywheels do not present a safety concern for current plant lives and for life extensions.

3.3 Relocation of TS 3.4.9 to the TRM

The purpose of TS 3.4.9, "Reactor Coolant System Structural Integrity" is to specify the requirements of maintaining the structural integrity of ASME Class 1, 2, and 3 components. This specification ensures the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. The requirements of TS 3.4.9 (i.e., Limiting Condition for Operation and Applicability), and its associated Bases will be relocated to the Waterford 3 TRM.

The specification which will be relocated to the TRM addresses the passive, pressure boundary function of ASME Code Class 1, 2, and 3 components. The TS, that will be relocated to the TRM, does not fulfill any one or more of the 10 CFR 50.36(c)(2)(ii) criteria on items for which TS must be established. Therefore the TS can be relocated to the TRM.

Relocation of TS 3.4.9 and its associated Bases section to the TRM does not imply any reduction in its importance in specifying the requirements of maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

4.0 TECHNICAL ANALYSIS

4.1 Revision and Relocation of SR 4.0.5

10 CFR 50.55a, *Codes and Standards* governs inservice testing and inspection requirements. SR 4.0.5a and the reference to the inservice inspection activities contained in paragraphs b and c of SR 4.0.5 will be removed in the proposed change since it is redundant to the requirements of 10 CFR 50.55a. Such duplication is unnecessary and results in additional administrative burden to change the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, this portion of the proposed change is considered administrative.

Included in SR 4.0.5a is the requirement to receive specific written relief from the NRC for each proposed relief request. The proposed change, which is consistent with NUREG-1432, eliminates this requirement. The elimination of the need to receive specific written relief is consistent with the wording contained in 10 CFR 50.55a, Paragraph 6(i) of either section (f) or (g), which states:

> "*The Commission may grant relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to*

the burden upon the licensee that could result if the requirements were imposed on the facility."

Due consideration is based on the merits of the basis for an alternative to the ASME Code and not the process for notifying licensees of ASME Code relief acceptability. This determination is not reduced as a result of granting requested Code relief verbally to the licensee. This removes any burden of timing for the NRC staff to complete the formal transmittal paperwork, which is required prior to implementation of a needed ASME alternative under the current TS wording. Therefore, the NRC staff can grant verbal relief to the licensee after the appropriate reviews and approval of the basis of the ASME Code alternative has been completed.

TS 4.0.5d, "Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements." will be deleted. Deletion of this paragraph will not eliminate any of the TS surveillance requirements or inservice inspection and testing activities. The intent of TS 4.0.5d is clearly conveyed by the existing statements for the individual SRs that reference TS 4.0.5. The performance of TS surveillance requirements, in addition to the inservice inspection and testing activities, are tracked and scheduled through the Waterford 3 Work Management Program. Removal of this paragraph is consistent with NUREG-1432.

For consistency with NUREG-1432, TS 5.5.8c, "The provisions of SR 3.0.3 are applicable to inservice testing activities, and" will be added to the proposed TS 6.5.8 with "SR 3.0.3" changed to "SR 4.0.3."

The following SRs currently reference "Specification 4.0.5" or "TS 4.0.5:"

SR 4.1.2.3	charging pumps
SR 4.1.2.5	boric acid makeup pumps
SR 4.4.2.1	pressurizer code safety valves shutdown
SR 4.4.2.2	pressurizer code safety valves operating
SR 4.4.8.3.1b	over pressure protection relief valves
SR 4.5.2f	ECCS subsystems pumps
SR 4.6.2.1	containment spray pumps
SR 4.6.3.3	containment isolation valves
SR 4.6.5	vacuum relief valves
SR 4.7.1.1	main steam line code safety valves
SR 4.7.1.2b	emergency feedwater pumps
SR 4.7.1.5	main steam isolation valves
SR 4.7.1.6b	main feedwater isolation valves

The references to "Specification 4.0.5" or "TS 4.0.5" will be modified to reference "the Inservice Testing Program." This change will not eliminate or modify any of the TS SRs or inservice inspection and testing activities.

SR 4.7.8, which defines the snubber inspection program, references TS 4.0.5. 10 CFR 50.55a as well as the surveillance requirements contained in the TS govern the inservice testing program for snubbers. The deletion of the reference to TS 4.0.5 will not change the snubber inspection program. The snubbers will continue to be inspected based on these requirements.

4.2 Extension of RCP Flywheel Volumetric Examination Intervals

The justification to extend the volumetric examination to ten years is based on the NRC's approval of Topical Report SIR-94-080-A, Revision 1. The NRC staff has granted ANO-2 and other participating Combustion Engineering plant owners the ability to extend the critical inspection of the keyway and bore from three years to ten years.

In the May 21, 1997 cover letter transmitting the safety evaluation for SIR-94-080, the NRC staff stated:

> Licensees for ANO-2, Palisades, Millstone 2, Waterford 3, and St. Lucie 1 & 2 need to verify the reference temperature RT_{NDT} for their reactor coolant pump (RCP) flywheels and demonstrate, with plant-specific test results if possible, that the corresponding fracture toughness (K_{IC}) values are equivalent to those reported in the topical report. ANO-1, which already has a 10-year inspection interval for its flywheels, will not be affected. The topical report indicated that flywheels for Waterford 3 could lose shrink fit at the accident speed. The staff will pursue this issue with Waterford 3 on a plant-specific basis.

In the safety evaluation to SIR-94-080, the NRC staff concluded, "(1) all flywheels meet the proposed non-ductile fracture criteria and will have adequate fracture toughness during their service periods, and (2) all flywheels except those for Waterford 3 satisfy the excessive deformation criterion of RG 1.14." This conclusion was based on the fracture toughness (K_{IC}) values reported in SIR-94-080A for all participating plants.

In the safety evaluation for SIR-94-080, the NRC staff required the applicant referencing the Topical Report to verify the reference temperature RT_{NDT}, and to justify the use of the K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI of the American Society of Mechanical Engineers (ASME) Code for flywheels made of materials other than SA 533 B and SA 508. Additionally, the NRC staff noted that loss of shrink fit for the Waterford 3 flywheel at accident speed would also need to be addressed.

Therefore, to justify extending the Waterford 3 RCP flywheel volumetric examinations from three years to ten years, Entergy must:

- Verify the reference temperature RT_{NDT} for the RCP flywheels and demonstrate that the corresponding fracture toughness (K_{IC}) values are equivalent to those reported in the topical report.
- Justify the use of the K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI of the ASME Code for flywheels made of materials other than SA 533 B and SA 508.
- Address the loss of shrink fit at the accident speed.

4.2.1 Verification of RT_{NDT} and K_{IC}

As stated in the Topical Report the Waterford 3 flywheels are the solid type with an outer diameter of 78 inches and an inner bore diameter of 13.75 inches and a thickness of 8.5 inches. The flywheel assembly is composed of two 4.25 inch plates.

The material used to manufacture the flywheels is pressure vessel quality ASTM-A-543, Grade B, Class 1 steel plate. This is a quenched and tempered alloy steel with fracture toughness properties similar to SA-533-B Class 1. A comparison of the specific properties of A-543-B and SA-533-B are tabulated below. Test results show that the steel used on the flywheels has very good fracture toughness properties. These properties are equal to or better than the SA-533-B material.

Properties	A-543-B, Class 1	SA-533-B, Class 1
Tensile Strength	105 to 115 ksi	80 to 100 ksi
Yield Strength	85 ksi	50 ksi
Elongation	14 %	18 %
Alloy Content	1.5% Cr, 3% Ni, 0.5% Mo	0.5% Ni, 0.5% Mo
Heat Treatment	1650°F, Q&T at 1100°F	1650°F, Q&T at 1100°F

Manufacturer certified material test reports (CMTRs) show that all the plates used in the construction of the Waterford 3 flywheels had nil ductility temperatures of -160 °F or -170 °F as exhibited by drop weight tests. In accordance with ASME Section III, NB-2330, the nil ductility RT_{NDT} is determined by the following method:

- Obtain nil ductility temperature (NDT) by dropweight testing per ASTM E 208.
- Perform Charpy Impact test at NDT + 60 °F. If values are greater than 50 ft. lbs., then the reference temperature, RT_{NDT} is equal to NDT.
- If charpy values are not greater than 50 ft. lbs., determine temperature (T_{50}) when they are greater than 50 ft. lbs. Then $RT_{NDT} = T_{50} - 60$ °F.

Material Heat No. And Slab	NDT (°F) Determined By Drop Weight Tests	T_{50}: Temperature (°F) at which Charpy Impact Test exceed 50 ft.lbs	RT_{NDT} (°F)
B0638 Slab 2A	-170	-100	-160
B0043 Slab 2B	-170	-150	-170
B0043 Slab 2C	-160	+20	- 40
D2853 Slab 4A	-170	-100	-160
B0638 Slab 4A	-160	-100	-160

Heat B0043 Slab 2C did not obtain the 50 ft. lb. limit until +20 °F; therefore the RT_{NDT} is –40 °F (i.e., $T_{50} - 60$.) Since this is the highest RT_{NDT}, it is conservatively used for all the flywheels in this evaluation.

In accordance with ASME Section XI, Appendix A, 1992 Addenda, fracture toughness may be calculated using equation:

$$K_{IC} = 33.2 + 20.734\, exp[0.02\,(T - RT_{NDT})]$$

K_{IC} is based on the lower bound of static initiation critical K_I (stress intensity factor) values.

Utilizing an RT_{NDT} of – 40 °F, determined per NB-2330 methods based on actual test data, the fracture toughness is 120 ksi √inch at 32 °F. The fracture toughness at the normal RCP motor operating temperature of 100 °F is 374 ksi √inch. The fracture toughness of the flywheels at the conservatively low temperature of 32 °F and the normal operating temperature of 100 °F both exceed the 100 ksi √inch utilized in the evaluation documented in SIR-94-080.

4.2.2 Justification for the Use of K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI of the ASME Code for A-543 Flywheels

The material used to manufacture the Waterford 3 flywheels is pressure vessel quality ASTM-A-543, Grade B, Class 1 steel plate. As noted above, test results show that the steel used on the Waterford 3 flywheels has very good fracture toughness properties and that these properties are equal to or better than the SA-533-B material. Charpy impact tests on the actual material were performed at a wide range of temperatures, from -320 °F to + 212 °F, providing further assurance that A-543 behaves in a similar manner as SA-533 in the ductile-to-brittle transition zone.

General Electric Report 34A180952, *Flywheel Integrity Report, Flywheels on RCP Motors,* Revision 2, issued on December 10, 1979 reported a K_{IC} value of 100 ksi √inch at approximately +5 °F and 140 ksi √inch at 32 °F for the A-543 flywheels at Waterford 3, utilizing fracture mechanics calculations based on critical crack length. When K_{IC} values are calculated using an RT_{NDT} of -40 °F and the K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI of the ASME Code, K_{IC} values of 84 ksi √inch and 120 ksi √inch are obtained at approximately +5 °F at 32 °F, respectively. The General Electric Report further reported that the starting temperature of an RCP motor would likely be greater than 65 °F while the SIR-94-080 reports the normal operating temperature to be approximately 100 °F. Therefore, K_{IC} values for 5, 32, 65, and 100 °F are listed in the table below.

Temperature (°F)	**K_{IC} (ksi √inch) for A-543 as Reported in General Electric Report 34A180952,**	**K_{IC} (ksi √inch) as calculated using K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI (RT_{NDT} = - 40 °F)**
5	100	84
32	140	120
65	NA	202
100	NA	374

The K_{IC} values at 5 °F and 32 °F calculated using the K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI are approximately 85 percent of those values reported in the General Electric report for A-543. This indicates that that the K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI provides conservative values, when used for A-543, which is consistent with its better fracture toughness qualities when compared to A-533. In addition, the K_{IC} values at the temperatures at which the RCP motor will be operated (i.e. above 65 °F) are over twice the value (i.e., 100 ksi √inch) utilized in SIR-94-080. Therefore it is acceptable to utilize the K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI for A-543.

4.2.3 Stress Analysis/Shrink Fit Evaluation

Topical Report SIR-94-080 states that the Waterford 3 RCP flywheel may lose its shrink fit at accident speeds. The Topical Report assumes an initial shrink fit of 0.0052" (radial) and calculates a centrifugal displacement of 0.00584" (radial) at accident speeds. Therefore, the Topical Report concluded that no shrink fit remains (i.e. 0.0052" – 0.00584" = -0.00064".) The shrink fit assumption of 0.0052 (from a typical RCP shaft) was used because the actual Waterford 3 flywheel dimensions were not known at the time.

Per a vendor report, the dimension of the Waterford 3 shaft outside diameter is 13.764" - .0005" and the flywheel bore is 13.750" + .001", -0. Using worst case (loosest fit), the diameter of the smallest shaft is 13.7635" and the largest bore is 13.751". Therefore the minimum interference fit is 13.7635" – 13.751" = 0.0125" (diameter) or 0.00625" (radial.) This shrink fit is greater (tighter) than that assumed in the Topical Report and therefore the shrink fit of the Waterford 3 flywheel will not be lost at design overspeed (i.e., 0.00625" – 0.00584" = 0.00041".)

The motor vendor performed an initial stress analysis on the Waterford 3 RCP flywheels. The stress analysis concluded the following: a) "the combined primary stresses at the design overspeed of 1500 RPM due to centrifugal force and the interference fit between the flywheel and the shaft do not exceed 1/3 of the flywheel material specified minimum yield strength, and b) the flywheel to shaft interference fit is sufficient to prevent separating from the shaft at design overspeed of the motor." Calculations showed that even at a speed of 1600 RPM (133% of design) the interference fit would not be lost.

4.2.4 Conclusion:

Based on the actual material types, resulting RT_{NDT} and fracture toughness values, and shrink fit, the Waterford 3 flywheels are bounded by the analysis conclusions of Topical Report SIR-94-080-A and the NRC's associated safety evaluation. In accordance with the requirements contained in the Safety Evaluation for SIR-94-080:

- The reference temperature RT_{NDT} has been verified to be less than or equal to –40 °F based on actual test data.
- The corresponding fracture toughness (K_{IC}) is 120 ksi √inch at 32 °F which is greater than the 100 ksi √inch utilized in SIR-94-080.
- The K_{IC} versus ($T-RT_{NDT}$) curve in Appendix A of Section XI of the ASME Code provides conservative results for A-543 material.
- The shrink fit is maintained at design overspeed.

In addition, none of the volumetric examinations conducted at Waterford 3, in accordance with RG 1.14, have identified any flaws in the flywheels.

4.3 Revision and Relocation of SR 4.4.9

The proposed change to revise and relocate the reactor coolant pump flywheel examination (SR 4.4.9) from the surveillance requirements to a new program section is based on the approach and wording contained in NUREG-1432, Revision 2. The wording proposed by Entergy is consistent with the wording in NUREG-1432, Section 5.5.7, which generically specifies:

"This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975."

An additional sentence is being added to the program requirements, which states:

"The volumetric examination per Position C.4.b.1 will be performed on a once per 10-year basis."

This specific change is consistent with Topical Report SIR-94-080-A, Revision 1 and the examination requirements of RG 1.14. Item 1 of Position C.4.b specifically requires that the RCP flywheels be volumetrically examined about every three years. Therefore, this sentence is being proposed which will supercede the three-year requirement.

The reference to SR 4.4.9 in the Limiting Condition for Operation for TS 3.4.9 is being removed since the only surveillance requirement associated with TS 3.4.9 was regarding the RCP flywheel inspection. The proposed change does not impact the remainder of the Limiting Condition for Operation to ensure that the structural integrity of ASME Code Class 1, 2, and 3 components is maintained.

4.4 Relocation of TS 3.4.9 to the TRM

The purpose of TS 3.4.9, "Reactor Coolant System Structural Integrity" is to specify the requirements for maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. This specification ensures the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These requirements serve a preventative purpose rather than a mitigative purpose. This specification (i.e., Limiting Condition for Operation, Applicability, and Actions) will be relocated to the TRM. The portion of this specification which will be relocated to the TRM addresses the passive, pressure boundary function of ASME Code Class 1, 2, and 3 components.

The criteria set forth in the final Commission Policy Statement, which was used to develop the Standard TSs (STS), have been codified in NRC Regulation 10 CFR 50.36(c)(2)(ii). If an item satisfies one or more of these criterion, then 10 CFR 50.36(c)(2)(ii) requires that a TS limiting condition for operation be established for that item.

The following discussion will show that TS 3.4.9 does not fulfill any of the 10 CFR 50.36(c)(2)(ii) criteria on items for which TSs must be established:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The portion of this specification which is being relocated to the facility TRM is not applicable to installed instrumentation which is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. This specification does not meet Criterion 1.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The portion of this specification which is being relocated to the facility TRM is not applicable to a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. While this TS imposes an operating restriction regarding pressure boundary integrity, it is not monitored or controlled during plant operation. The assumed integrity of Class 1, 2, and 3 components is assured by means of periodic inspections. Therefore, this specification does not meet Criterion 2.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

In accordance with Criterion 3, ASME Code Class 1, 2, and 3 components, which are part of the primary success path and function to mitigate design basis accidents or transients that either assume the failure of or present a challenge to the integrity/operability of these components, are required to be included in individual specifications that cover those components. These mitigative functional requirements are not being relocated from the TS. However, as stated above, the portion of this specification which is being relocated to the facility TRM ensures that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant and serves no mitigative purpose. Therefore, this specification does not satisfy Criterion 3.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

In accordance with Criterion 4, ASME Code Class 1, 2, and 3 components, which operating experience or probabilistic risk assessment has shown to be significant to public health and safety are required to be included in individual specifications that cover those components. The specifications for these components are not being relocated from the TS. The requirements to maintain structural integrity of ASME Code Class 1, 2, and 3 components covered by TS 3.4.9, which are being relocated to the TRM, will not change and will be implemented in the same manner as currently implemented (i.e., in accordance with 10 CFR 50.55(a)). The location of these requirements have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. The location of the requirements contained in this TS do not affect the risk review/unavailability monitoring of applicable SSCs. This specification does not meet Criterion 4.

This TS does not fulfill any of the 10 CFR 50.36(c)(2)(ii) criteria on items for which TSs must be established. Therefore, this TS can be relocated to the TRM.

Relocation of TS 3.4.9 and the associated Bases section to the TRM does not imply any reduction in its importance in specifying the requirements of maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. Changes to the TRM are controlled in accordance with 10 CFR 50.59. Therefore, the proposed changes will have no adverse effect on plant safety.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50, Appendix A, GDC 4 continues to be met because the technical analysis, provided above, shows that RCP flywheel integrity will be assured after extending the RCP flywheel volumetric examination, required by Regulatory Guide (RG) 1.14 "*Reactor Coolant Pump Flywheel Integrity,*" from three to ten years.

10 CFR 50.36 continues to be met in that the TS (3.3.9, Reactor Coolant System Structural Integrity) being relocated to the Waterford 3 Technical Requirements Manual does not meet the criterion as specified in 10 CFR 50.36 for being included into the Technical Specifications.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the SAR.

5.2 No Significant Hazards Consideration

Technical Specification (TS) Surveillance Requirements (SR) 4.0.5 (inservice inspection and testing) and 4.4.9 (reactor coolant pump motor flywheel inspections) will be revised and relocated to new sections 6.5.8, "Inservice Testing Program" and 6.5.7, "Reactor Coolant Pump Flywheel Inspection Program," respectively and TS 3.4.9, "Reactor Coolant System Structural Integrity," will be relocated to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Requirements Manual (TRM). The reactor coolant pump flywheel volumetric examination interval will be extended from three years to ten years based on Topical Report SIR-94-080-A, *Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements*, Revision 1. The TS index will be revised to reflect the changes and references to SR 4.0.5 throughout TSs will be revised to reflect the relocation of SR 4.0.5.

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to relocate SR 4.0.5 to the administrative section of the TSs, including modifications to the wording to make it consistent with NUREG-1432, will not reduce the current testing and inspection requirements. The performance of a code inservice test is not an accident initiator. Verbally issuing relief to the ASME Code by the NRC staff in lieu of written relief does not reduce assurance of the health and safety of

the public since the NRC staff still reviews the basis for the relief on its technical merit and the NRC staff still obtains management approval prior to granting the relief.

Inspections of the reactor coolant pump (RCP) flywheels are conducted to detect a flaw in the flywheel prior to it becoming a missile that could damage other portions of the facility. The fracture mechanics analyses conducted as part of NRC approved Topical Report SIR-94-080-A, Rev. 1, shows that a conservatively sized pre-existing crack will not grow to a flaw size necessary to create flywheel missiles within the current or extended life of the facility thus the flywheel will remain intact and perform its function to reduce the rate of decay of coolant flow during a postulated loss of power to the RCP motor. This analysis conservatively assumes minimum material properties, maximum flywheel speed, location of the flaw in the highest stress area, and a number of startup and shutdown cycles higher than expected. Since a conservative flaw in the RCP flywheels will not grow to the allowable flaw size under large break LOCA conditions over the life of the plant, reducing the inspection frequency of the flywheels will not significantly increase the probability or consequences of an accident previously evaluated.

The change to move the surveillance requirements for the RCP flywheels to the programs section of the technical specifications is administrative and has no impact on probability or consequences of an accident.

The change to move TS 3.4.9 to the Waterford 3 TRM will have no adverse effect on plant operation or the availability or operation of any accident mitigation equipment. Changes to the TRM are controlled in accordance with 10 CFR 50.59. Therefore, moving TS 3.4.9 to the Waterford 3 TRM will not adversely impact an accident initiator and can not cause an accident.

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

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These changes do not introduce any new failure modes.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The testing and inspection requirements contained in TS 4.0.5 are governed by 10 CFR 50.55a, *Codes and Standards.* The 10 CFR requirements to perform the ASME

code testing and inspections will not be reduced by the proposed change. The inspections and tests will continue to be performed as they are currently. The proposed change has no impact on plant equipment operation.

The fracture mechanics analysis conducted in support of extending the RCP flywheel volumetric examination interval from three years to ten years shows that significant conservatism has been used for calculating the allowable flaw size, critical flaw size, and crack growth rate in the RCP flywheels. These include minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area, and a number of startup/shutdown cycles eight times greater than expected. Since a postulated flaw in a Waterford 3 flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under loss of coolant accident conditions over the life of the plant, reducing the examination requirements for the detection of such cracks over the life of the plant will not involve a significant reduction in the margin of safety. The proposed change has no impact on plant equipment operation.

The change to move the surveillance requirements for the RCP flywheels to the programs section of the technical specifications is administrative and has no impact on plant operation.

Relocation of TS 3.4.9 to the TRM does not imply any reduction in its importance in ensuring that the structural integrity and operational readiness of ASME Code Class 1, 2, and 3 components will be maintained at an acceptable level throughout the life of the plant. The proposed change has no impact on plant operation.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The relocation of TS SR 4.0.5 and 4.4.9 to the administrative section of TS and the relocation of TS 3.4.9 to the TRM is consistent with NUREG-1432, *Standard Technical Specifications Combustion Engineering Plants.*

The relocation of TS SR 4.0.5 to the administrative section of TS was recently approved for Arkansas Nuclear One, Unit 2 (ANO-2) in Amendment 233 on September 4, 2001

The relocation of TS SR 4.4.10.1 (equivalent to 4.4.9 for Waterford 3) to the administrative section of TS and the extension of the RCP flywheel volumetric examinations from three years to ten years was recently approved for ANO-2 in Amendment 241 on April 11, 2002.

The relocation of TS 3/4.4.10 (equivalent to 3/4.4.9 for Waterford 3) and the extension of the RCP flywheel volumetric examinations from three years to ten years was recently approved for Millstone Nuclear Power Station , Unit No. 2 in Amendment 264 on February 1, 2002.

Attachment 2

W3F1-2003-0015

Proposed Technical Specification Changes (mark-up)

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed twenty-five percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2, shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational modes as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by ~~Specification 4.0.5~~ the Inservice Testing Program.

REACTIVITY CONTROL SYSTEMS

BORIC ACID MAKEUP PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid makeup pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a., suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by ~~Specification 4.0.5.~~ the Inservice Testing Program.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 3%.*

APPLICABILITY: MODE 4.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes (except cooldown in shutdown cooling) and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2.1 Verify each required pressurizer code safety valve is OPERABLE in accordance with ~~Specification 4.0.5.~~ the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 3%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 Verify each required pressurizer code safety valve is OPERABLE in accordance with ~~Specification 4.0.5~~ the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 For each SDC System suction line relief valve:

a. verify in the control room at least once per 12 hours that each valve in the suction path between the RCS and the SDC relief valve is open.

b. verify each SDC relief valve is OPERABLE in accordance with ~~TS 4.0.5.~~ the Inservice Testing Program

4.4.8.3.2 With the RCS vented per ACTIONS a, b, or c, the RCS vent(s) and all valves in the vent path shall be verified to be open at least once per 12 hours*.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

REACTOR COOLANT SYSTEM

3/4.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 70°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F, except during hydrostatic testing of components that are nonisolable from the Reactor Coolant System, then restore the structural integrity prior to increasing the Reactor Coolant System temperature more than 30°F above the minimum temperature required by NDT considerations.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

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EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. A visual inspection of the safety injection system sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 380 cubic feet of granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 13.07 ± 0.03 grams of TSP from a TSP storage basket is submerged, without agitation, in 4 ± 0.1 liters of 120 ± 10°F water borated to 3011 ± 30 ppm, the pH of the mixed solution is raised to greater than or equal to 7 within 3 hours.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a. High pressure safety injection pump.

b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the low pressure safety injection pumps stop, the safety injection system sump isolation valves open.

f. By verifying that each of the following pumps required to be OPERABLE performs as indicated on recirculation flow when tested pursuant to ~~Specification~~ the Inservice Testing Program: 4.0.5

1. High pressure safety injection pump differential pressure greater than or equal to 1429 psid.

2. Low pressure safety injection pump differential pressure greater than or equal to 168 psid.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWSP on a containment spray actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal. Each spray system flow path from the safety injection system sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, 3 and 4*.

ACTION:

a. With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

b. With two containment spray systems inoperable, restore at least one spray system to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the water level in the containment spray header riser is > 149.5 feet MSL elevation.

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is correctly positioned to take suction from the RWSP.

c. By verifying, that on recirculation flow, each pump develops a total head of greater than or equal to 219 psid when tested pursuant to ~~Specification 4.0.5.~~ the Inservice Testing Program.

*With Reactor Coolant System Pressure > 400 psia.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.

b. Verifying that on a containment Radiation-High test signal, each containment purge valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to ~~Specification 4.0.5.~~ the Inservice Testing Program.

CONTAINMENT SYSTEMS

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5 Two vacuum relief lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one vacuum relief line inoperable, restore the vacuum relief line to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5 No additional Surveillance Requirements other than those required by ~~Specification 4.0.5.~~ the Inservice Testing Program.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Linear Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify each required main steam line code safety valve lift setpoint per Table 3.7-1 in accordance with ~~Specification 4.0.5.~~ the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.1.2 The emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each manual, power-operated, and automatic valve in each water flow path and in both steam supply flow paths to the turbine-driven EFW pump steam turbine, that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 92 days on a STAGGERED TEST BASIS by testing the EFW pumps pursuant to Specification 4.0.5. This surveillance requirement is not required to be performed for the turbine-driven EFW pump until 24 hours after exceeding 750 psig in the steam generators.

c. At least once per 18 months by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an actual or simulated actuation signal.

NOTE: This surveillance requirement is not required to be performed for the turbine-driven EFW pump until 24 hours after exceeding 750 psig in the steam generators.

2. Verifying that each EFW pump starts automatically upon receipt of an actual or simulated actuation signal.

d. Prior to entering MODE 2, whenever the plant has been in MODE 4, 5, 6 or defueled, for 30 days or longer, or whenever feedwater line cleaning through the emergency feedwater line has been performed, by verifying flow from the condensate storage pool through both parallel flow legs to each steam generator.

the Inservice Testing Program.

PLANT SYSTEMS

MAIN FEEDWATER ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each Main Feedwater Isolation Valve (MFIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Note: Separate Condition entry is allowed for each valve.

With one or more MFIV inoperable, close and deactivate, or isolate the inoperable valve within 72 hours and verify inoperable valve closed and deactivated or isolated once every 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each main feedwater isolation valve shall be demonstrated OPERABLE:

a. By verifying isolation ≤ 5.0 seconds when tested pursuant to ~~Specification 4.0.5.~~ the Inservice Testing Program.

b. By verifying actuation to the isolation position on an actual or simulated actuation signal at least once per 18 months.

PLANT SYSTEMS

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All hydraulic and mechanical snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those OPERATIONAL MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVELLIANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program ~~and the requirements of Specification 4.0.5~~.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-2 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment 73 .

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 except that:

a. The Radiation Protection Superintendent shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

b. Personnel in the Health Physics, Chemistry and Radwaste Departments shall meet or exceed the minimum qualifications of ANSI N18.1-1971.

c. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of 10 CFR Part 55.

d. Personnel in the Nuclear Quality Assurance Department, and other staff personnel who perform inspection, examination, and testing functions, shall meet or exceed the minimum qualifications of Regulatory Guide 1.58, Rev. 1, September 1980. (Endorses ANSI N45.2.6-1978).

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager-Nuclear and shall meet or exceed the requirements and recommendations of Section 5.2 of ANSI 3.1-1978 and 10 CFR Part 55.

6.5 ~~Not Used~~ PROGRAMS

6.5.1 through 6.5.6 will be used later.

Insert 6.5.7 here

Insert 6.5.8 here

Insert 6.5.7

6.5.7 REACTOR COOLANT PUMP FLYWHEEL INSPECTION PROGRAM

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. The volumetric examination per Regulatory Position C.4.b.1 will be performed on approximately 10-year intervals.

Insert 6.5.8

6.5.8 INSERVICE TESTING PROGRAM

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities,

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

Attachment 3

W3F1-2003-0015

Changes to Technical Specification Bases Pages

(For Information Only)

BASES

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Specification 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to Specification 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of Surveillance Requirements to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed Surveillance Requirements. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the Surveillance Requirements.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of a Surveillance Requirement on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of a Surveillance Requirement on another channel in the same trip system.

Specification 4.0.1 through ~~4.0.5~~ 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, the facility operation will be within safety limits, and that the limiting condition of operation will be met."

BASES

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillance must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Condition for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

CONTAINMENT SYSTEMS

BASES

3/4.6.5 VACUUM RELIEF VALVES (Continued)

With one of the required vacuum relief lines inoperable, the inoperable line must be restored to OPERABLE status within 72 hours. The specified time period is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA.

If the vacuum relief line cannot be restored to OPERABLE status within the required Allowed Outage Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours. The Allowed Outage Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

The SR references the ~~TS 4.0.5~~ Inservice Testing Program, which establishes the requirement that inservice testing of the ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. Therefore, SR Frequency is governed by the Inservice Testing Program.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM (Continued)

Surveillance Requirements

a. Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. This Surveillance Requirement (SR) does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

b. The SR to verify pump OPERABILITY pursuant to ~~Specification 4.0.5~~ the Inservice Testing Program ensures that the requirements of ASME Code Section XI are met and provides reasonable assurance that the pumps are capable of satisfying the design basis accident flow requirements. Because it is undesirable to introduce cold EFW into the steam generators while they are operating, testing is typically performed on recirculation flow. Such in-service tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance.

This SR is modified to indicate the SR should be deferred until suitable test conditions have been established. This deferral is required because there is an insufficient steam pressure to perform post maintenance activities which may need to be completed prior to performing the required turbine-driven pump SR. This deferral allows the unit to transition from MODE 4 to MODE 3 prior to the performance of the SR and provides a 24 hour period once a steam generator pressure of 750 psig is reached to complete the required post maintenance activities and SR. If this SR is not completed within the 24 hour period or fails, then the appropriate ACTION must be entered. The twenty-five percent grace period allowed by TS 4.0.2 can not be applied to the 24 hour period.

c. The SR for actuation testing ensures that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates EFAS and/or MSIS signals, by demonstrating that each automatic valve in the flow path actuates to its correct position and that the EFW pumps will start on an actual or simulated actuation signal. This Surveillance covers the automatic flow control valves, automatic isolation valves, and steam admission valves but is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month frequency is acceptable, based on the design reliability and operating experience of the equipment.

BASES

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)

The TS is annotated with a 3.0.4 exemption, allowing entry into the applicable MODES to be made with an inoperable MFIV closed or isolated as required by the ACTIONS. The ACTIONS allow separate condition entry for each valve by using "With one or more MFIV...". This prevents immediate entry into TS 3.0.3 if both MFIVs are declared inoperable.

The Surveillance Requirement to verify isolation in less than or equal to 5 seconds is based on the time assumed in the accident and containment analyses. The static test demonstrates the ability of the MFIVs to close in less than or equal to 5 seconds under design basis accident conditions. The MFIVs should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power and would create added cyclic stresses. The Surveillance to verify each MFIV can close on an actual or simulated actuation signal is normally performed when the plant is returning to operation following a refueling outage. Verification of valve closure on an actuation signal is not required until entry into Mode 3 consistent with TS 3.3.2. The 18 month frequency is based on the refueling cycle. Verification of closure time is performed per TS 4.0.5. This frequency is acceptable from a reliability standpoint and is in accordance with the Inservice Testing Program.
the Inservice Testing Program.

• (DRN 02-1684)

Credited Non-Safety Related Support Systems for MFIV Operability

Reactor Trip Override (RTO) and the Auxiliary Feedwater (AFW) Pump High Discharge Pressure Trip (HDPT) are credited for rapid closure of the Main Feedwater Isolation Valves (MFIVs) during main steam and feedwater line breaks. Crediting of these non-safety features was submitted to the NRC as a USQ and approved. (Reference letter dated September 5, 2000 from the NRC to Charles M. Dugger, "Waterford 3 Steam Electric Station, Unit 3 - Issuance of Amendment RE: Addition of Main Feedwater Isolation Valves to Technical Specifications and Request for NRC Staff Review of an Unreviewed Safety Question.")

The feature of RTO that is credited for MFIV closure is the rapid SGFP speed reduction upon reactor trip initiation. This feature reduces the differential pressure across the valve disc at closure, thus allowing rapid valve closure. Therefore, the RTO feature must be able to decrease SGFP speed to minimum on a reactor trip during SGFP operation for OPERABILITY of the MFIVs.

The AFW Pump HDPT reduces the differential pressure across the valve disc at closure during AFW Pump operation. Therefore, this feature must be functional during AFW Pump operation for OPERABILITY of the MFIVs. When the AFW pump is not running, this trip is not required.

In MODES 1, 2, 3, and 4, the MFIVs are required to be OPERABLE. Because the MFIVs are required to be OPERABLE in MODES 1, 2, 3, and 4, RTO must be able to decrease SGFP

• (DRN 02-1684)

BASES

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)

• (DRN 02-1684)
speed to minimum on a reactor trip and the AFW Pump HDPT must be functional, to support closure of the valve. If RTO is unable to decrease running SGFP(s) speed to minimum on a reactor trip with the SGFPs running, both MFIVs must be declared INOPERABLE, and Technical Specification 3.7.1.6 must be entered. If the AFW Pump HDPT is non-functional with the AFW pump running, the AFW pump should be secured immediately or both MFIVs must be declared INOPERABLE, and Technical Specification 3.7.1.6 must be entered.

RTO and AFW Pump HDPT Test Requirements

The RTO and AFW pump high pressure trip are subjected to a testing program similar to comparable safety related instrumentation to provide assurance of the reliability of these non-safety related functions credited to support the MFIV safety related closure function.

The testing requirements for the RTO credited function should demonstrate the ability of RTO to reduce SGFP speed upon an actual or simulated actuation signal. The test requirements do not require timing the response because in the limiting FWLB scenario, RTO is required for compliance with a 5 second Technical Specification closure; however, the containment analyses allow longer closure times during this event. Even if RTO were to fail, the MFIV would eventually close as the pressure across the valve equalizes to the available actuator thrust, the nitrogen pressure equalizes, and finally as the SGFP speed reduces due to a loss of steam after the MSIV closes. The expected maximum closure time would be less than one minute due to SGFP speed decrease. This phenomenon would act to close the valve within the appropriate time to preserve the safety function. The RTO feature should not be tested at power since it increases the risk of a feedwater transient with the plant generating power, but should normally be performed when the plant is returning to operation following a refueling outage. The testing criteria shall verify functionality of the RTO system, with SGFP pump response, by verifying that the feedwater control system sends the control signal corresponding to minimum speed to the pump upon an actual or simulated RTO signal at least once per 18 months. The functionality of the RTO system shall be verified through the performance of Instrumentation & Controls functional test procedure, "Functional Test of Reactor Trip Override, High Level Override, and Level Channel Deviation FWCS." The 18 month frequency is based on the refueling cycle, similar to testing performed per TS 4.0.5. This frequency is acceptable from a reliability standpoint.
the Inservice Testing Program.

The testing requirements for the AFW Pump HDPT should demonstrate the ability of the pump to trip upon receiving an actual or simulated high pressure signal. The AFW Pump HPDT feature can be tested at power since the AFW pump is not required during normal operations, however, the test is normally performed when the plant is returning to operation following a refueling outage. The testing criteria shall verify functionality of the AFW Pump HDPT by (1) verifying pump trip on an actual or simulated actuation signal at least once per 18 months and (2) verifying that the delay time of Relay AFWEREL 1419-3, the most time critical element of

• (DRN 02-1684)

BASES

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)

• (DRN 02-1684)
the trip circuitry, is less than the setpoint specified in the Component Database plus the specified tolerance at least once per 18 months. The AFW pump trip shall be verified through the performance of Operations surveillance test procedure, "AFW High Discharge Pressure Trip Test." The relay delay time shall be verified through the performance of an Electrical Maintenance task document for relay AFWEREL 1419. The 18 month frequency is based on the refueling cycle, similar to testing performed per ~~TS 4.0.5~~ the Inservice Testing Program. This frequency is acceptable from a reliability standpoint to detect degradation.
• (DRN 02-1684)

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator secondary pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitation to 115°F and 210 psig is based on a steam generator RTNDT of 40°F and is sufficient to prevent brittle fracture. Below this temperature of 115°F the system pressure must be limited to a maximum of 20% of the secondary hydrostatic test pressure of 1375 psia (corrected for instrument error). Should steam generator temperature drop below 115°F an engineering evaluation of the effects of the overpressurization is required. However, to reduce the potential for brittle failure the steam generator temperature may be increased to a limit of 200°F while performing the evaluation. The limitations on the primary side of the steam generator are bounded by the restrictions on the reactor coolant system in Specification 3.4.8.1.

3/4.7.3 COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS

The OPERABILITY of the component cooling water system and its corresponding auxiliary component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the safety analyses.