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W3F1-2005-0062

December 1, 2005

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

Subject: Technical Specification Bases Update to the NRC for the Period
October 1, 2004 through October 31, 2005
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Pursuant to Waterford Steam Electric Station Unit 3 Technical Specification 6.16, Entergy Operations, Inc. (EOI) hereby submits an update of all changes made to Waterford 3 Technical Specification Bases since the last submittal per letter W3F1-2004-0090, dated October 7, 2004. This TS Bases update satisfies the requirement listed in 10 CFR 50.71(e).

There are no commitments associated with this submittal. Should you have any questions or comments concerning this submittal, please contact Ron Williams at (504) 739-6255.

Very truly yours,

RJM/RLW/cbh

Attachment Waterford 3 Technical Specification Bases Revised Pages

A001

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ATTACHMENT 1
To W3F1-2005-0062

Waterford 3 Technical Specification Bases Revised Pages

T.S. Bases Change No.	Implementation Date	Affected TS Bases Pages	Topic of Change
36	11/04/2004	B 3/4 5-1a	Change No. 36 to TS Bases section 3/4.5.1 was implemented by ER-W3-2003-0548-000. TS Bases Section 3/4.5.1 was changed to clarify the verification of boron in the SITs could be accomplished by either sampling or calculation method. When the calculation method is utilized, it will minimize containment entries and as a result eliminate a personnel safety hazard as well as keep radiation dose to ALARA.
37	02/21/2005	B 3/4 7-2d	Change No. 37 to TS Bases section 3/4.7.1.2 was implemented by ER-W3-2004-0548-000. TS Bases 3/4.7.1.2 was changed to delete an inaccurate statement that performance frequency of the 18 month EFAS SR was based on the potential for causing a plant transient. This change allowed SR 4.7.1.2.c to be performed with the plant online.
38	05/10/2005	B 2-3 B 2-4 B 2-6 B 2-7 (new page) B 3/4 1-2 B 3/4 1-3 B 3/4 1-4 B 3/4 1-4a B 3/4 2-4 B 3/4 4-3 B 3/4 4-4e B 3/4 4-5 B 3/4 4-10 B 3/4 5-3 B 3/4 6-2 B 3/4 6-3 B 3/4 7-1 B 3/4 7-2f B 3/4 7-3 B 3/4 7-3c B 3/4 7-3e B 3/4 7-3g B 3/4 7-3h (new page) B 3/4 7-3i (new page) B 3/4 7-3j (new page) B 3/4 7-4 B 3/4 7-4(1) B 3/4 8-1 B 3/4 8-1a	Change No. 38 to TS Bases sections were implemented by ER-W3-2001-1109-016. Change clarified the TS Bases to be consistent with the TS as amended in TS Amendment 199. The amendment increases the maximum steady-state reactor core power level from 3441 megawatts thermal (MWt) to 3716MWt, which is an increase of approximately 8 percent. The increase is considered an extended power uprate (EPU).

39	05/11/2005	B 3/4 4-5 B 3/4 6-1 B 3/4 6-3 B 3/4 6-7 B 3/4 7-3 B 3/4 7-3a B 3/4 7-4a B 3/4 9-4 B 3/4 11-3	Change No. 39 to various TS Bases sections was implemented by ER-W3-2004-0276-003. The TS Bases were changed to be consistent with the TS as amended in TS Amendment 198, use of Alternate Source Term (AST). The changes to the TS Bases included, in part, replacing the reference to 10 CFR 100, Reactor Site Criteria with 10 CFR 50.67, Accident Source Monitoring, replacing the reference to whole body with total effective dose equivalent and replacing the text in sections 3/4 .9.10 and 3/4.9.11, Water Level-Reactor Vessel and Spent Fuel Pool.
40	05/11/2005	I thru XXIII (new pages)	Change No. 40 was implemented by ER-W3-2005-0162-000. This TS Bases change added new Index pages controlled and updated under the Licensee-approved TS Bases Control Program. This change clarified the TS as amended in TS Amendment 200. The amendment consisted of the following changes: (1)deleted the Index from the TS as an NRC-approved document and committed to maintain and the Index through a licensee document control program similar to the NRC approved TS Bases Control Program using Waterford 3 internal controls; (2) revised TS sections 5.3.1 Fuel Assemblies and 5.6.1, Criticality to allow a limited number of lead test assemblies (LTAs), the use of ZIRLOTM fuel cladding, and limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods; (3) revised TS 6.9.1.11.1 to allow the use of the Westinghouse Nuclear Physics code package and the incorporation of the methodology used to support Zirlo™ cladding material; and (4) revised TS 6.9.1.11.1 item 6 to correct the title of an analytical method.
41	06/09/2005	B 3/4 1-1a	Change No. 41 to TS Bases section 3/4.1.1.4 change was implemented by ER-W3-2001-1149-000. The change clarified the TS Bases by deleting the statement "the ECCS analysis remains valid for the peak linear heat rate of Specification 3.2.1" to reflect the new TS as amended in TS Amendment 199.

42	06/13/2005	Index XXIII	Change No. 42 to Index page XXIII was implemented by ER-W3-2005-0260. Condition Report CR-WF3-2005-2404 identified that TS Bases Index page XXIII issued with Amendment 199, included a line entry for Table "B3/4.0-1, Analytical - Indicated Values" that did not exist in the TS Bases. The change deleted this line item to accurately reflect corresponding information contained in the TS as amended by Amendment 199.
43	08/15/2005	Index XVII	Change No. 43 to Index page XVII was implemented by ER-W3-2005-0155. Change clarified the index page to reflect the elimination of requirements to provide occupational radiation exposure reports and monthly operating reports, as amended in TS Amendment 202. The amendment deleted TS 6.9.1.5, occupational radiation exposure reports and TS 6.9.1.6, monthly operating reports.
44	08/18/2005	B 3/4 4-4d	Change No. 44 to TS Bases section 3/4.4.5.1 was implemented by ER-W3-2005-0360. Change clarified the TS Bases to accurately reflect Surveillance Requirement 4.4.5.1.c for a containment fan cooler flow switch as an 18 month Channel Functional Test, as amended in TS Amendment 197. The TS Bases SR section, erroneously describes SR 4.4.5.1.c as an 18 month channel calibration. This change will appropriately relocate reference to SR 4.4.5.1.c from the TS Bases Channel Calibration section to the Channel Functional Test section and include discussion on the 18 month frequency channel function.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 38 REPLACEMENT PAGE(S)
(29 pages)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 38 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 2-3	B 2-3
B 2-4	B 2-4
B 2-6	B 2-6
-------	B 2-7
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
B 3/4 1-4a	B 3/4 1-4a
B 3/4 2-4	B 3/4 2-4
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4e	B 3/4 4-4e
B 3/4 4-5	B 3/4 4-5
B 3/4 4-10	B 3/4 4-10
B 3/4 5-3	B 3/4 5-3
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3	B 3/4 6-3
B 3/4 7-1	B 3/4 7-1
B 3/4 7-2f	B 3/4 7-2f
B 3/4 7-3	B 3/4 7-3
B 3/4 7-3c	B 3/4 7-3c
B 3/4 7-3e	B 3/4 7-3e
B 3/4 7-3g	B 3/4 7-3g
-----------	B 3/4 7-3h
-----------	B 3/4 7-3i
-----------	B 3/4 7-3j
B 3/4 7-4	B 3/4 7-4
B 3/4 7-4(1)	B 3/4 7-4(1)
B 3/4 8-1	B 3/4 8-1
B 3/4 8-1a	B 3/4 8-1a

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Linear Power Level - High

→(DRN 04-1243, Ch. 38)

The Linear Power Level - High trip provides reactor core protection against rapid reactivity excursions. This trip initiates a reactor trip at a linear power level of less than or equal to 108% of RATED THERMAL POWER.

←(DRN 04-1243, Ch. 38)

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER* level of less than or equal to 0.257% of RATED THERMAL POWER* unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER* level is above 10-4% of RATED THERMAL POWER*; this bypass is automatically removed when the THERMAL POWER* level decreases to 10-4% of RATED THERMAL POWER*.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2350 psia which is below the nominal lift setting of 2500 psia for the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1684 psia. This trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

* As measured by the Logaritmic Power Channels.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated concurrently with a safety injection, a containment isolation, and a main steam isolation. The setpoint for this trip is identical to the ESFAS setpoint.

Steam Generator Pressure - Low

→(DRN 04-1243, Ch. 38)

The Steam Generator Pressure - Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 810 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

←(DRN 04-1243, Ch. 38)

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against events involving a mismatch between steam and feedwater flow. These may be due to a steam or feed line pipe break or other increased steam flow or decreased feed flow events. A large feedwater line break event inside containment establishes the trip setpoint. The setpoint ensures that a trip will occur before the steam generator heat sink is lost. The trip setpoint also ensures that the Reactor Coolant System design pressure will not be exceeded prior to the time emergency feedwater can be supplied for decreased heat removal events such as a loss of condenser vacuum or loss of feedwater flow.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

DNBR - Low (Continued)

in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit of 1.26. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	≥ 495°F
b.	RCS Cold Leg Temperature-High	< 580°F
c.	Axial Shape Index-Positive	Not more positive than +0.5
d.	Axial Shape Index-Negative	Not more negative than -0.5
e.	Pressurizer Pressure-Low	≥ 1860 psia
f.	Pressurizer Pressure-High	< 2375 psia
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	≤ 7.00
i.	Quality Margin-Low	> 0

→(DRN 04-1243, Ch. 38)

The CPCs contain several auxiliary trip functions which are credited in the safety analysis. These trips manifest themselves as DNBR trips however they are making the trip determination on parameters other than DNBR.

The CPC Variable Overpower Trip (VOPT) is provided to include a trip on power which is compensated for the decalibrating effects of changes in coolant temperature in the reactor vessel downcormer. Additionally, the trip setpoint is allowed to change with slow changes in plant power. Thus at intermediate steady state powers, the plant is protected by a power trip which is a small distance above steady state power levels. The rate at which the automatic increases and decreases in the setpoint may change are limited and accounted for in the safety analysis.

The CPCs contain a trip which detects asymmetries in cold leg loop temperatures resulting from as asymmetric steam generator transient. The trip occurs if the cold leg asymmetry exceeds 11 °F.

The CPCs contain a trip monitoring margin to saturation conditions in the hot legs. A trip will be generated if margin to saturation is less than 13 °F.

The CPCs contain a direct trip on low RCP speed. The trip will occur if the RCP speed drops below 0.965.

←(DRN 04-1243, Ch. 38)

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Reactor Coolant Flow - Low

→(DRN 03-6, Ch. 20)

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a steam line break event with a loss-of-offsite power. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a nominal setpoint of 19.00 psid. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

←(DRN 03-6, Ch. 20)

3/4.1.2 BORATION SYSTEMS

→(DRN 04-1243, Ch. 38)

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, and (5) an emergency power supply from OPERABLE diesel generators.

←(DRN 04-1243, Ch. 38)

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

→(DRN 04-1243, Ch. 38)

The boration capability is sufficient to provide a SHUTDOWN MARGIN of 5.15% delta k/k with zenon-free conditions, RCS temperature greater than 200°F, and letdown secured. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions assuming the most reactive CEA stuck out of the core and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 plus 10,064 gallons of 2050 ppm borated water from the refueling water storage pool. The higher limit of 83% indicated is specified to be consistent with Specification 3.5.4 in order to meet the ECCS requirements.

←(DRN 04-1243, Ch. 38)

→(DRN 03-375, Ch.19)

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. Temperature changes in the RCS impose reactivity changes by means of the moderator temperature coefficient. Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. This will require a new SDM calculation be performed if the current SDM calculation does not bound the temperature change. Small changes in RCS temperature are unavoidable and so long as the required SDM is maintained during these changes, any positive reactivity additions will be limited to acceptable levels. Introduction of temperature changes must be evaluated to ensure they do not result in a loss of required SDM.

←(DRN 03-375, Ch. 19)

→(DRN 04-1243, Ch. 38)

The boron capability required below 200°F is based upon providing a 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 1,727 gallons of 2050 ppm borated water from the refueling water storage pool or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

←(DRN 04-1243, Ch. 38)

BORATION SYSTEMS (Continued)

→(DRN 04-1243, Ch. 38)

The contained water volume limits include allowance for water not available because of discharge line location, instrument tolerances, and other physical characteristics. The unusable water volume in one Boric Acid Makeup Tank is half the unusable water volume when using two Boric Acid Makeup Tanks. Consequently, Figures 3.1-1 and 3.1-2 are provided for using one or two Boric Acid Makeup Tanks to satisfy the requirements of TS 3.1.2.2 and 3.1.2.8.

The 60 °F minimum Boric Acid Makeup Tank solution indicated temperature limit insures that the boron will not precipitate even at the maximum allowed boron concentration when instrument accuracies are considered. The precipitation temperature at the maximum allowed Boric Acid Makeup Tank boron concentration is 50.2 °F. The 60 °F minimum indicated temperature limit also insures that the minimum Boric Acid Makeup Tank solution temperature assumed in the safety analysis (49 °F) is bounded. The 55 °F Reactor Auxiliary Building temperature prerequisite for monitoring Boric Acid Makeup Tank solution temperature is acceptable due to the increased accuracy of the Reactor Auxiliary Building temperature indications available on the plant monitoring computer.

←(DRN 04-1243, Ch. 38)

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

→(DRN 04-1243, Ch. 38)

←(DRN 04-1243, Ch. 38)

3/4.1.2.9 BORON DILUTION

This specification is provided to prevent a boron dilution event, and to prevent a loss of SHUTDOWN MARGIN should an inadvertent boron dilution event occur. Due to boron concentration requirements for the RWSP and boric acid makeup tanks, the only possible boron dilution that would remain undetected by the operator occurs from the primary makeup water through the CVCS system. Isolating this potential dilution path or the OPERABILITY of the startup channel high neutron flux alarms, which alert the operator with sufficient time available to take corrective action, ensures that no loss of SHUTDOWN MARGIN and unanticipated criticality occur.

The ACTION requirements specified in the event startup channel high neutron flux alarms are inoperable provide an alternate means to detect boron dilution by monitoring the RCS boron concentration to detect any changes. The frequencies specified in the COLR provide the operator sufficient time to recognize a decrease in boron concentration and take appropriate corrective action without loss of SHUTDOWN MARGIN. More frequent checks are required with more charging pumps in operation due to the higher potential boron dilution rate.

BORON DILUTION (Continued)

The surveillance requirements specified provide assurance that the startup channel high neutron flux alarms remain OPERABLE and that required valve and electrical lineups remain in effect.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, or to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN. CEAs that are confirmed to be inoperable due to problems other than addresed by ACTION a. of TS 3.1.3.1 and that are trippable, will not impact SHUTDOWN MARGIN as long as their relative positions satisfy the applicable alignment requirements.

For small misalignments (less than 19 inches) of the CEAs, there is (1) a small effect on the time dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with trippable but small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution. Problems may also cause more than one control rod to be immovable where the control rods continue to be trippable. With trippable but multiple inoperable rods: the alignment limits and restriction on THERMAL POWER in accordance with the provisions of Specification 3.1.3.6 for insertion limits, assures fuel rod integrity during continued operation. These provisions are sufficient to allow 72 hours to restore the inoperable rods to operable status when it is confirmed that the cause of the immovable rods is an electrical problem in the rod control system or an electrical or mechanical

MOVABLE CONTROL ASSEMBLIES (Continued)

problem with the rod stepping mechanism exclusive of the rod holding coil that must function for a reactor trip. In such cases, the control rods will continue to be capable of fulfilling their primary safety function.

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to I9 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to trippable but misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit

DNBR MARGIN (Continued)

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses, and that the DNBR is maintained within the safety limit for Anticipated Operational Occurrences (AOO).

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

→(DRN 04-1243, Ch. 38)

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses, with adjustment for instrument accuracy of ±3°F, and that the peak linear heat generation rate and the moderator temperature coefficient effects are validated. The safety analysis assumes that cold leg temperature is maintained between 533°F and 552°F or indicated temperatures of 536°F and 549°F.

←(DRN 04-1243, Ch. 38)

3/4.2.7 AXIAL SHAPE INDEX

→(DRN 02-458, Ch. 12)

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses, to ensure that the peak fuel centerline temperature and DNBR remain within the safety limits for Anticipated Operational Occurrences (AOO).

←(DRN 02-458, Ch. 12)

3/4.2.8 PRESSURIZER PRESSURE

→(DRN 04-1243, Ch. 38)

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses. The inputs to CPCs and COLSS are the most limiting. The values are adjusted for an instrument uncertainty of ± 35 psi. The safety analysis assumes that pressurizer pressure is maintained between 2090 psia and 2310 psia or indicated pressurizer pressures of 2125 psia and 2275 psia.

←(DRN 04-1243, Ch. 38)

STEAM GENERATORS (Continued)

based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

→(DRN 04-1243, Ch. 38)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 75 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 75 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of the 75 gallon per day limit in Specification 3.4.5.2 will require plant shutdown and an unscheduled inspection, during which the leakage tubes will be located and plugged or repaired.

←(DRN 04-1243, Ch. 38)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit as defined in Surveillance Requirement 4.4.4.4. Defective tubes may be repaired by sleeving in accordance with CENS Report CEN-605-P, "Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves," Revision 00-P, dated December 1992. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Sleeved tubes will be included in the periodic tube inspections for the inservice inspection program.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

→ (DRN 04-1223, Ch. 33)

Monitoring Containment Sump In-Leakage Flow

During automatic operation of the containment sump pumps (after a containment sump pump has operated), the flow calculation performed by the plant monitoring computer based on a level change will no longer be accurate since the level in the sump will be lowering. A 20 minute time period has been conservatively determined based on engineering calculations for this equipment operation. In addition, upon reboot of the plant monitoring computer, a period of 10 minutes is required for the leak rate calculation to become available. It has been determined these time periods (independent or combined) of calculation sump in-leakage flow inaccuracies, the instrumentation remains adequate to detect a leakage rate, or its equivalent, of one gpm in less than one hour; therefore, the containment sump level instrumentation and the corresponding flow calculation is considered to remain operable.

References

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45, Revision 0, dated May 1973.
3. UFSAR, Sections 5.2.5 and 12.3.

← (DRN 04-1223, Ch. 33)

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

→(DRN 04-1243, Ch. 38)

The 75 gallon per day (gpd) per steam generator tube leakage limit ensures that the radiological consequences, including that from tube leakage, will be limited to the 10CFR50.67 limits for offsite dose and within the limits of General Design Criterion 19 for control room dose. For those analyzed events that do not result in faulted steam generators, greater than or equal to 75 gpd primary-to-secondary leakage per steam generator is assumed in the analysis. For those analyzed events that result in a faulted steam generator (e.g., MSLB), 540 gpd primary-to-secondary leakage is assumed through the faulted steam generator while greater than or equal to 75 gpd primary-to-secondary leakage is assumed through the intact steam generator.

←(DRN 04-1243, Ch. 38)

OPERATIONAL LEAKAGE (Continued)

→(DRN 04-1243, Ch. 38)

Steam generator tube cracks having primary-to-secondary leakage less than 150 gpd per steam generator during operation will have an acceptable margin of safety to withstand loads imposed during normal operation and postulated accidents (Reference NEI 97-06). Due to the proximity of the east atmospheric dump valve to the east control room intake, the primary-to-secondary leakage limit required to achieve acceptable radiological consequences, for accidents that rely on reactor coolant system cooldown using the steam generators, is limiting. Therefore, 75 gpd per steam generator is imposed as the primary-to-secondary operational leakage limit.

←(DRN 04-1243, Ch. 38)

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within Steady State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

→(DRN 03-173, Ch. 18)

The Code of Federal Regulations, 10 CFR 100 specifies the maximum dose to the whole body and the thyroid an individual offsite can receive during a design basis accident. The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The specific activity limits ensure that these doses are held within the appropriate 10 CFR 100 requirements (small fraction, well within, or within) during analyzed transients and accidents.

Operation with iodine specific activity levels greater than the LCO limit is permissible for up to 48 hours, provided the activity levels do not exceed 60 uCi/gm. A 48 hour limit was established because of the low probability of an accident occurring during this period. The dose consequences of an accident during this 48 hour period would not exceed the full 10 CFR 100 limits.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

←(DRN 03-173, Ch. 18)

PRESSURE/TEMPERATURE LIMITS (Continued)

→(DRN 04-1241, Ch. 34)

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia (as corrected for elevation). Instrument uncertainty is not included in the Figures 3.4-2 and 3.4-3.

←(DRN 04-1241, Ch. 34)

→(DRN 04-1233, Ch. 35; 04-1243, Ch. 38)

←(DRN 04-1233, Ch. 35; 04-1243, Ch. 38)

→(DRN 04-1241, Ch. 34)

The OPERABILITY of the shutdown cooling system relief valve or an RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 200°F. Each shutdown cooling system relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with injection into a water-solid RCS. The limiting transient includes simultaneous, inadvertent operation of three HPSI pumps, three charging pumps, and all pressurizer backup heaters in operation. Since SIAS starts only two HPSI pumps, a 20% margin is realized.

The restrictions on starting a reactor coolant pump in MODE 4 and with the reactor coolant loops filled in MODE 5, with one or more RCS cold legs less than or equal to 200°F, are provided in Specification 3.4.1.3 and 3.4.1.4 to prevent RCS pressure transients caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures. Maintaining the steam generator less than 100°F above each of the Reactor Coolant System cold leg temperatures (even with the RCS filled solid) or maintaining a large surge volume in the pressurizer ensures that this transient is less severe than the limiting transient considered above.

←(DRN 04-1241, Ch. 34)

3/4.5.4 REFUELING WATER STORAGE POOL (RWSP)

The OPERABILITY of the refueling water storage pool (RWSP) as part of the ECCS also ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWSP minimum volume and boron concentration ensure that (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWSP and the RCS water volumes with all CEAs inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

→(DRN 04-1243, Ch. 38)

The minimum contained borated water volume limit, 83% indicated, includes an allowance for water not usable because of pool discharge line location, other physical characteristics, and instrument uncertainty. The safety analysis assumes an available volume of 383,000 gallons which is bounded by the 83% level indicated.

←(DRN 04-1243, Ch. 38)

The lower limit on contained water volume, the specific boron concentration and the physical size (approximately 600,000 gallons) of the RWSP also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum limit on the RWSP temperature ensures that the assumptions used in the containment pressure analysis under design base accident conditions remain valid and avoids the possibility of containment overpressure. The minimum limit on the RWSP temperature is required to prevent freezing and/or boron precipitation in the RWSP.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.65 psid, (2) the containment peak pressure does not exceed the design pressure of 44 psig during either LOCA or steam line break conditions, and (3) the minimum pressure of the ECCS performance analysis (BTP CSB 61) is satisfied.

The limit of +27 inches water (approximately 1.0 psig) for initial positive containment pressure is consistent with the limiting containment pressure and temperature response analyses inputs and assumptions.

The limit of 14.275 psia for initial negative containment pressure ensures that the minimum containment pressure is consistent with the ECCS performance analysis ensuring core reflood under LOCA conditions, thus ensuring peak cladding temperature and cladding oxidation remain within limits. The 14.275 psia limit also ensures the containment pressure will not exceed the containment design negative pressure differential with respect to the annulus atmosphere in the event of an inadvertent actuation of the containment spray system.

3/4.6.1.5 AIR TEMPERATURE

→(DRN 04-1243, Ch. 38)

The limitation on containment minimum average air temperature ensures that the ECCS is capable of maintaining a peak clad temperature (PCT) less than or equal to 2200°F under LOCA conditions. A lower containment average air temperature results in a lower post accident containment pressure, a lower reflood rate, and therefore a higher PCT. The containment minimum average air temperature limit is only applicable above 70% rated thermal power. At power levels of 70% or below and a containment minimum average air temperature of less than 90°F, ECCS is capable of maintaining the peak clad temperature (PCT) less than or equal to 2200°F under LOCA conditions.

←(DRN 04-1243, Ch. 38)

The limit of 120°F on high average containment temperature is consistent with the limiting containment pressure and temperature response analyses inputs and assumptions. The limits currently adopted by Waterford 3 are 269.3°F during LOCA conditions and 413.5°F during MSLB conditions.

→(DRN 02-1904; 04-1243, Ch. 38)

The 90°F minimum and 120°F maximum indicated values specified in the TS are the values used in the accident analysis.

←(DRN 02-1904; 04-1243, Ch. 38)

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment vessel will withstand the maximum pressure resulting from the design basis LOCA and main steam line break accident. A visual inspection in conjunction with Type A leakage test is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The use of the containment purge valves is restricted to 90 hours per year in accordance with Standard Review Plan 6.2.4 for plants with the Safety Evaluation Report for the Construction License issued prior to July 1, 1975. The purge valves have been modified to limit the opening to approximately 52° to ensure the valves will close during a LOCA or MSLB;and therefore, the SITE BOUNDARY doses are maintained within the guidelines of 10CFR Part 100. The purge valves, as modified, comply with all provisions of BTP CSB 6-4 except for the recommended size of the purge line for systems to be used during plant operation.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 La leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

Operability concerns for purge supply and exhaust isolation valves other than those addressed in Actions "a" and "b" of Specification 3.6.1.7 are addressed under Specification 3.6.3, "Containment Isolation Valves."

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Spray System and the Containment Cooling System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or MSLB for any double-ended break of the largest reactor coolant pipe or main steam line. Under post-accident conditions these systems will maintain the containment pressure below 44 psig and temperatures below 269.3°F during LOCA conditions or 413.5°F during MSLB conditions. The systems also reduce the containment pressure by a factor of 2 from its post-accident peak within 24 hours, resulting in lower containment leakage rates and lower offsite dose rates.

The Containment Spray System (CSS) also provides a mechanism for removing iodine from the containment atmosphere under post-LOCA conditions to maintain doses in accordance with 10 CFR Part 100 limits as described in Section 6.5.2 of the FSAR.

If LCO 3.6.2.1 requirements are not met due to the condition described in ACTION (a), then the inoperable CSS train components must be returned to OPERABLE status within seven (7) days of discovery. This seven (7) day allowed outage time is based on the findings of deterministic and probabilistic analysis, CE NPSD-1045, "Modifications To The Containment Spray System, and Low Pressure Safety Injection System Technical Specifications". Seven (7) days is a reasonable amount of time to perform many corrective and preventative maintenance items on the affected CSS train. CE NPSD-1045 concluded that the overall risk impact of the seven (7) day allowed outage time was either risk-beneficial or risk-neutral.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

→(DRN 04-1243, Ch. 38)

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

←(DRN 04-1243, Ch. 38)

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The MSSVs rated capacity passes the full steam flow at 102% RATED THERMAL POWER (100% + 2% for instrument error) with valves open. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat. All the MSSVs have an orifice size of 28.27 in^2.

→(DRN 04-1243, Ch. 38)

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce THERMAL POWER so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is less than or equal to the maximum allowable power as listed in Table 3.7-2 and the Linear Power Level - High trip setpoint is less than or equal to that listed in Table 3.7-2.

The 4 hour completion time is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 8 hours is allowed to reduce the setpoints. This completion time is based on the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The THERMAL POWER reductions are derived on the following bases: An analysis of a loss of condenser vacuum event initiated at the reduced power levels listed in Table 3.7-2 that shows peak steam generator pressures are maintained below 110% of system design pressure. The maximum allowable power limitations listed in Table 3.7-2 are reduced from the analytical values used in the analysis by at least 2% to account for power measurement uncertainties.

The reactor trip setpoint reductions are determined by adding 8% to the maximum THERMAL POWER limit derived from the analysis of condenser vacuum event. The 8% margin is consistent with margin between the normal Linear Power Level - High trip setpoint and 100% RATED THERMAL POWER. The 8% difference provides sufficient margin to avoid an inadvertent trip. The Linear Power Level - High trip is not credited in the analysis of the loss of condenser vacuum event but is reduced to reinforce the requirement to remain at the reduced power levels for extended periods of time.

←(DRN 04-1243, Ch. 38)

3/4.7.1.3 CONDENSATE STORAGE POOL (Continued)

If natural circulation is required, the combined capacity (CSP and one WCT) is sufficient to maintain the plant at HOT STANDBY for 4 hours, followed by a cooldown to shutdown cooling entry conditions assuming the availability of only onsite or only offsite power, and the worst single failure (loss of a diesel generator or atmospheric dump valve). This requires approximately 303,000 gallons of EFW and complies with BTP RSB 5-1.

→(DRN 04-1243, Ch. 38)

The CSP contained water volume limit (92% indicated in MODES 1, 2, and 3) includes an allowance for water not usable because of vortexing and instrumentation uncertainties. This provides an assurance that a minimum of 170,000 gallons is available for the EFW system and that 3,500 gallons is available for the CCW makeup system. The CSP contained water volume limit (11% indicated in MODE 4) also includes an allowance for water not usable because of vortexing and instrumentation uncertainties. This provides an assurance that minimum of 3,500 gallons is available in the CSP for the CCW makeup system.

The maximum limit on CSP temperature ensures that the assumptions used in design basis accidents with EFW flow remain valid. The minimum limit on CSP temperature ensures that the assumptions used in the MSLB return to power event remain valid.

←(DRN 04-1243, Ch. 38)

3/4.7.1.4 ACTIVITY

→(DRN 04-1243, Ch. 38)

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 540 gallons per day primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

←(DRN 04-1243, Ch. 38)

→DRN 03-1737, Ch. 31)

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE (MSIV)

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break. MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside of, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs), atmospheric dump valves, and emergency feedwater pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal (MSIS) generated by either low steam generator pressure or high containment pressure. The MSIVs fail as is on loss of power to the actuator however; the operators for the MSIV are furnished with redundant hydraulic fluid dump valves powered by diverse power, to ensure that no single electrical failure will prevent valve closure. The MSIVs may also be actuated manually.

A description of the MSIVs is found in Final Safety Analysis Report (FSAR), Section 10.3.

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, as discussed in FSAR, Section 6.2. It is also influenced by the accident analysis of the SLB events presented in FSAR, Section 15.1.3. The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The OPERABILITY of the MSIVs ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

←(DRN 03-1737, Ch. 31)

→(DRN 03-1737, Ch. 31)

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE (MSIV) (Continued)

→(DRN 04-1243, Ch. 38)

SR 4.7.1.5a verifies that the closure time of each MSIV is within its limit when tested pursuant to the Inservice Testing Program. A static test using 4.0 seconds demonstrates the ability of the MSIVs to close in less than or equal to the 8 seconds required closure time under design basis accident conditions. The 8 second required closure time includes a 1 second allowance for instrument response time.

This SR is normally performed during a refueling outage but may be performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Inservice Inspection, Article IWV-3400), requirements during operation in MODES 1 and 2.

←(DRN 04-1243, Ch. 38)

The Frequency for this SR is in accordance with the Inservice Testing Program.

This test may be conducted in MODE 3, with the unit at operating temperature and pressure.

SR 4.7.1.5b verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance may be performed upon returning the plant to operation following a refueling outage. The Frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance. Therefore, this Frequency is acceptable from a reliability standpoint.

←(DRN 03-1737, Ch. 31)

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES

The Main Feedwater Isolation Valves (MFIVs) isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Closure of the MFIVs terminates flow to both steam generators, mitigating the consequences for feedwater line breaks (FWLBs). Closure of the MFIVs effectively terminates the addition of main feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) or FWLBs inside containment, and reducing the cooldown effects for MSLBs.

The MFIVs isolate the non-safety related feedwater supply from the safety related portion of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of Emergency Feedwater (EFW) to the intact steam generator.

→(DRN 04-1243, Ch. 38)

One MFIV is located on each MFW line, outside, but close to, containment. The MFIVs are located upstream of the EFW injection point so that EFW may be supplied to a steam generator following MFIV closure.

←(DRN 04-1243, Ch. 38)

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.

→(DRN 03-1807, Ch. 30; 04-1243, Ch. 38)

The Surveillance Requirement to verify isolation in less than or equal to 6 seconds is based on the time assumed in the accident and containment analyses. A static test, utilizing the accumulated pressure in one of the two accumulators, using 6.6.seconds demonstrates the ability of the MFIVs to close in less than or equal to 6 seconds under design basis accident conditions. The 6 second required closure time includes a 1 second allowance for instrument response time.

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 the Inservice Testing Program. This frequency is acceptable from a reliability standpoint and is in accordance with the Inservice Testing Program.

→(DRN 03-1807, Ch. 30)

←(DRN 02-1684, Ch. 15; 04-1243, Ch. 38)

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 Ioslation 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, Ch. 15)

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

→(DRN 02-1684, Ch. 15; 03-1807, Ch. 30)

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 the Inservice Testing Program. This frequency is acceptable from a reliability standpoint to detect degradation.

←(DRN 02-1684, ch. 15; 03-1807, Ch. 30)

→(DRN 04-1243, Ch. 38)

3/4.7.1.7 ATMOSPHERIC DUMP VALVES (ADVs)

Two ADVs are provided, one per steam generator. The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ADVs are provided with a pressurized nitrogen gas supply that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The ADVs can also be operated manually once the nitrogen gas supply is depleted.

The ADVs provide a safety grade method for cooling the unit to Shutdown Cooling (SDC) System entry conditions, should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section 10.3. This is done in conjunction with the Emergency Feedwater System providing cooling water from the condensate storage pool (CSP) to meet Branch Technical Position (BTP) RSB 5-1.

The automatic operation of the ADVs to open is assumed in the Small Break LOCA (SBLOCA) analysis at power levels above 70% RATED THERMAL POWER. ADVs are credited for SBLOCA analysis to lower steam generator secondary side pressures, compared to crediting only MSSVs, and thus provide increased cooling of the RCS. This results in a lower calculated peak cladding temperature (PCT) for SBLOCA ECCS analysis.

Analysis has shown that automatic operation of the ADV is not required when the unit is at or below 70% RATED THERMAL POWER for greater than six hours because, based on decay heat load, one high-pressure safety injection train is capable of mitigating the SBLOCA event. At greater than 70% RATED THERMAL POWER, one high-pressure safety injection train and one ADV, in automatic, are capable of mitigating the SBLOCA event. Therefore, the ADVs, in automatic, are required at greater than 70% RATED THERMAL POWER and for six hours after reducing power to less than or equal to 70% RATED THERMAL POWER.

Limiting Condition for Operation

The LCO requires that each ADV be OPERABLE.

The ADV manual controls must be OPERABLE in MODES 1, 2, 3, and 4 to allow operator action needed for decay heat removal and safe shutdown in accordance with BTP RSB 5-1.

←(DRN 04-1243, Ch. 38)

3.4.7.1.7 ATMOSPHERIC DUMP VALVES (ADVs)(Continued)

The LCO is modified by a footnote requiring that ADV automatic actuation controls be OPERABLE (i.e., ADVs in automatic and capable of automatic actuation at less than or equal to 1040 psia (992 psig indicated)) when operating at greater than 70% RATED THERMAL POWER and for six hours after reducing power to less than or equal to 70% RATED THERMAL POWER for mitigation of the SBLOCA.

The ADVs are containment isolation valves and must be capable of manual isolation of the ADV lines in MODE 1, 2, 3, and 4 in order to be considered OPERABLE. Because the OPERABILITY of the ADVs is controlled by this Technical Specification, Technical Specification 3.6.3, "Containment Isolation Valves," does not apply to the ADVs.

ACTIONs

The ACTIONs are modified by a note indicating that the provisions of Specification 3.0.4 are not applicable provided one ADV is OPERABLE. This allows for MODE changes with one ADV inoperable provided the appropriate ACTION is entered upon entry into the applicability MODEs.

ACTIONs (a) and (b) would be applicable only when the automatic actuation channels are required to be OPERABLE per the LCO footnote.

a. This ACTION addresses the condition when one ADV is incapable of automatic actuation. This condition includes:
   - A malfunctioning automatic actuation channel, or
   - When the automatic actuation controls for one ADV have been placed in manual.

   A 72 hour allowed outage time is provided to restore the ADV to an OPERABLE status. The 72 hour allowed outage time takes into account the capability afforded by the remaining OPERABLE ADV and is consistent with the allowed outage time of an inoperable high-pressure safety injection train.

   If the ADV can not be restored to an OPERABLE status within the allowed outage time, the unit must be placed in a status in which the LCO does not apply. To achieve this status, power must be reduced to less than or equal to 70% RATED THERMAL POWER within the next 6 hours. The LCO will no longer apply once the unit has been at less than or equal to 70% RATED THERMAL POWER for greater than six hours.

b. This ACTION addresses the condition when both ADVs are incapable of automatic actuation. This condition includes:
   - Malfunctioning of both ADV's automatic actuation channel,
   - When the automatic actuation controls for both ADVs have been placed in manual, or
   - A combination of the above such that both ADVs are incapable of automatic operation.

←(DRN 04-1243, Ch. 38)

3.4.7.1.7 ATMOSPHERIC DUMP VALVES (Continued)

In this condition, the SBLOCA can not be mitigated by one high-pressure safety injection train alone. Therefore, one of the ADVs must be restored to OPERABLE status within 1 hour or power must be reduced to less than or equal to 70% RATED THERMAL POWER within the next six hours. The LCO will no longer apply once the unit has been at less than or equal to 70% RATED THERMAL POWER for greater than six hours.

c. This ACTION address the condition when one ADV is inoperable for reasons other than those addressed in ACTIONs (a) and (b) above. This condition includes:
   - The inability to operate the ADV manually via the handwheel, or
   - The inability to operate the ADV manually via the controller in the control room, or
   - An inoperable nitrogen accumulator.

   A 72 hour allowed outage time is provided to restore the ADV to an OPERABLE status. The 72 hour allowed outage time takes into account the capability afforded by the remaining OPERABLE ADV, a nonsafety grade backup in the Steam Bypass System and MSSVs, the closed system inside containment, and the backup isolation capability of the block valve.

   If the ADV can not be restored to an OPERABLE status within the allowed outage time, the unit must be placed in a status in which the LCO does not apply. To achieve this status, the unit must be placed in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The following conditions are not addressed by the ACTION statements:
- The automatic actuation channel for one ADV is inoperable and the other ADV is inoperable for other reasons.
- Both ADVs are inoperable for reasons other than the automatic actuation channels.

For these conditions, Specification 3.0.3 is entered.

Surveillance Requirements

a. To mitigate the SBLOCA event, the ADVs must automatically open at a pressure of less than or equal to 1040 psia (992 psig indicated). This CHANNEL CHECK provides assurance that the behavior of the steam line pressure input to the automatic actuation channel is reasonable for the existing plant conditions. This steam line pressure input is available on the plant monitoring computer or from appropriate maintenance and test equipment. This Surveillance Requirement (SR) need not be performed when the ADV automatic actuation channels are not required to be OPERABLE per the LCO footnote.

←(DRN 04-1243, Ch. 38)

3.4.7.1.7 ATMOSPHERIC DUMP VALVES (ADVs) (Continued)

b. To mitigate the SBLOCA event, the ADVs must automatically open at a pressure of less than or equal to 1040 psia (992 psig indicated). This Surveillance Requirement (SR) ensures that the ADV controllers are in automatic and set at an appropriate setpoint that is bounded by the SBLOCA safety analysis. The setpoint must be verified using the plant monitoring computer or appropriate maintenance and test equipment. This SR need not be performed when the ADV automatic actuation channels are not required to be OPERABLE per the LCO footnote.

c. To perform a controlled cooldown of the reactor coolant system, the ADVs must be able to be opened and throttled through their full range. Additionally, the ADV must be capable of being closed to fulfill its secondary function of containment isolation. This SR ensures the ADVs are tested through a full control cycle. The test interval is in accordance with the Inservice Testing Program.

d. The SR to calibrate the ADV automatic actuation channels ensures that the system will generate an actuation signal at 1040 psia (992 psig indicated) as assumed for the SBLOCA. The calibration should include the plant monitoring computer points used to set the setpoint.

e. The SR for actuation testing ensures that the ADV will automatically open on a high steam pressure signal, with a response time of less than or equal to 60 seconds, as assumed for the SBLOCA. Credit may be taken for an actual or simulated actuation signal.

←(DRN 04-1243, Ch. 38)

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.

3/4.7.4 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level, temperature, and number of fans ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The UHS consists of two dry cooling towers (DCTs), two wet cooling towers (WCTs), and water stored in WCT basins. Each of two 100 percent capacity loops employs a dry and wet cooling tower.

Each DCT consists of five separate cells. Cooling air for each cell is provided by 3 fans, for a total of 15 per DCT. The cooling coils on three cells of each DCT (i.e. 60%) are protected from tornado missiles by grating located above the coils and capable of withstanding tornado missile impact. With a Tornado Watch in effect and the number of fans OPERABLE within the missile protected area of a DCT less than that required by Table 3.7-3, ACTION c requires the restoration of inoperable fans within 1 hour or plant shutdown as specified. This ACTION is based on FSAR analysis (subsection 9.2.5.3.3) that assumes the worst case single failure as, 1 emergency diesel generator coincident with a loss of offsite power. This failure occurs subsequent to a tornado strike and 60% cooling capacity of a DCT is assumed available.

→(DRN 04-1243, Ch. 38)

Each WCT has a basin which is capable of storing sufficient water to bring the plant to safe shutdown under all design basis accident conditions. Item a of LCO 3/4.7.4 requires a minimum water level in each WCT basin of 97% (-9.86 ft MSL). When the WCT basin water level is maintained at -9.86 ft MSL, each basin has a minimum capacity of 174,000 gallons. This minimum WCT basin capacity contains enough volume to account for water evaporation and drift losses expected during a LOCA. Additional volume is needed from the second WCT basin to handle the non-essential load of fuel pool cooling during the LOCA. (The WCTs can be manually interconnected through a Seismic Category I line.) The WCT basin is also credited as a source of Emergency Feedwater (EFW). The WCT minimum capacity bounds the amount of EFW required from the WCT basin for all design basis accidents. Each WCT consists of two cells, each cell is serviced by 4 induced draft fans, for a total of 8 per WCT. There is a concrete partition between the cells that prevents air recirculation between the fans of each cell. Covers are required on fans declared out-of-service to prevent air recirculation between fans within a cell.

←(DRN 04-1243, Ch. 38)

Table 3.7-3 specifies increased or decreased fan OPERABILITY requirements based on outside air temperature and humidity. The table provides the cooling tower fan OPERABILITY requirements that may vary with outside ambient conditions. Fan OPERABILITY requirements are specified for each controlling parameter (i.e., dry bulb temperatures for DCT fans and wet bulb temperatures for WCT fans). The calculated temperature values (EC-M95-009) associated

3/4.7.4 ULTIMATE HEAT SINK (Continued)

with DCT and WCT fan requirements have been rounded in the conservative direction and lowered at least one full degree to account for minor inaccuracies. Failure to met the OPERABILITY requirements of Table 3.7-3 requires entry into the applicable action. Because temperature and humidity are subject to change during the day, ACTION d requires periodic temperature readings to verify compliance with Table 3.7-3 when any cooling tower fan is inoperable.

→(DRN 04-1243, Ch. 38)

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to essential equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

←(DRN 04-1243, Ch. 38)

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C SOURCES, and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion l7 of Appendix A to 10 CFR Part 50.

→(DRN 04-1243, Ch. 38)

The Limiting Condition for Operation (LCO) ensures that each diesel generator storage tank contains fuel oil of a sufficient volume to operate each diesel generator for a period of 7 days. The LCO limit is 39,300 gallons useable corresponds to a level of 96.41% in the fuel oil storage tank. This useable volume is sufficient to operate the diesel generator for 7 days based on the time-dependent loads of the diesel generator following a loss of offsite power and a design bases accident and includes the capacity to power the engineered safety features in conformance with Regulatory Guile 1.137 October 1979. To account for instrument uncertainty at least 97.86% indicated level is maintained in the fuel oil storage tank to assure that 39,300 usable gallons are available. The minimum onsite stored fuel oil is sufficient to operate the diesel generator for a period longer than the time to replenish the onsite supply from the outside sources discussed in FSAR 9.5.4.2.

An additional provision is included in the LCO which allow the diesel generators to remain operable when their 7 day fuel oil supply is not available provided that at least a 6 day supply of fuel oil is available. This provision is acceptable on the basis that replacement fuel oil is onsite within the first 48 hours after falling below the 7 day supply. The LCO limit of 37,000 gallons useable corresponds to a level of 90.76% in the fuel oil storage tank. This useable volume is sufficient to operate the diesel generator for 5 days based on the full continuous load (4400kW) of the diesel generator and is sufficient to operate the diesel generator for greater than 6 days based on the time dependent loads of the diesel generator following a loss of offsite power and a design basis accident. To account for instrument uncertainty at least 92.21% indicated level is maintained in the fuel oil storage tank to assure that 37,000 usable gallons are available.

←(DRN 04-1243, Ch. 38)

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. When one diesel generator is inoperable to perform either preplanned maintenance (both preventive and corrective) or unplanned corrective maintenance work, the allowed-outage-time (AOT) can be extended from 72 hours to 10 days, if a temporary emergency diesel generator (TEDG) is verified available and aligned for backup operation to the permanent plant EDG removed from service. The TEDG will be available prior to removing the permanent plant EDG from service for the extended preplanned maintenance work or prior to exceeding the 72-hour AOT for the extended unplanned corrective maintenance work. A Configuration Risk Management Program (CRMP) is implemented to assess risk of this activity when applying this ACTION. The TEDG availability is verified by: (1) starting the TEDG and verifying proper operation; (2) verifying 24 hour onsite fuel supply; and (3) ensuring the TEDG

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

is aligned to supply power through a 4.16 kV non-safety bus to the 4.16kV safety bus. A status check for TEDG availability will also be performed at least once every 72 hours following the initial TEDG availability verification. The status check shall consists of: (1) verifying the TEDG equipment is mechanically and electrically ready for manual operation; (2) verifying 24 hour onsite fuel supply; and (3) ensuring the TEDG is aligned to supply power through a 4.16 kV non-safety bus to the 4.16 kV safety bus. If the TEDG becomes unavailable during the 10 day AOT and cannot be restored to available status, the EDG AOT reverts back to 72-hours. The 72 hours begins with the discovery of the TEDG unavailability, not to exceed a total of 10 days from the time the EDG originally became inoperable. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

→(DRN 03-375, Ch. 19)

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that(1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. With the minimum AC and DC power sources and associated distribution systems inoperable the ACTION requires the immediate suspension of various activities including operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SHUTDOWN MARGIN or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling concentration. This may result in an overall reduction in boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including increases when operating with a positive moderator temperature coefficient, must also be evaluated to ensure they do not result in a loss of required SHUTDOWN MARGIN. Suspension of these activities does not preclude completion of actions to establish a safe conservative condition.

←(DRN 03-375, Ch. 19)

<u>TECHNICAL SPECIFICATION BASES</u>
<u>CHANGE NO. 39 REPLACEMENT PAGE(S)</u>
(9 pages)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 39 and contains the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 4-5	B 3/4 4-5
B 3/4 6-1	B 3/4 6-1
B 3/4 6-3	B 3/4 6-3
B 3/4 6-7	B 3/4 6-7
B 3/4 7-3	B 3/4 7-3
B 3/4 7-3a	B 3/4 7-3a
B 3/4 7-4a	B 3/4 7-4a
B 3/4 9-4	B 3/4 9-4
B 3/4 11-3	B 3/4 11-3

OPERATIONAL LEAKAGE (Continued)

→(DRN 04-1243, Ch. 38)

Steam generator tube cracks having primary-to-secondary leakage less than 150 gpd per steam generator during operation will have an acceptable margin of safety to withstand loads imposed during normal operation and postulated accidents (Reference NEI 97-06). Due to the proximity of the east atmospheric dump valve to the east control room intake, the primary-to-secondary leakage limit required to achieve acceptable radiological consequences, for accidents that rely on reactor coolant system cooldown using the steam generators, is limiting. Therefore, 75 gpd per steam generator is imposed as the primary-to-secondary operational leakage limit.

←(DRN 04-1243, Ch. 38)

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within Steady State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

→(DRN 03-173, Ch. 18; 05-131, Ch. 39)

The Code of Federal Regulations, 10 CFR 50.67 specifies the maximum total effective dose equivalent an individual offsite can receive during a design basis accident. The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The specific activity limits ensure that these doses are held within the appropriate 10 CFR 50.67 requirements (small fraction, well within, or within) during analyzed transients and accidents.

←(DRN 05-131, Ch. 39)

Operation with iodine specific activity levels greater than the LCO limit is permissible for up to 48 hours, provided the activity levels do not exceed 60 uCi/gm. A 48 hour limit was established because of the low probability of an accident occurring during this period. The dose consequences of an accident during this 48 hour period would not exceed the full 10 CFR 50.67 limits.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

←(DRN 03-173, Ch. 18)

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

→(DRN 05-131, Ch. 39)

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR 50.67 during accident conditions.

←(DRN 05-131, Ch. 39)

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during the performance of the periodic Type A tests to account for possible degradation of the containment leakage barriers between leakage tests. Also, the summation of penetration leakages measured during Type B and C testing is limited to 0.6 L_a. At all other times between required leakage rate tests, overall containment leakage is limited to L_a. The maximum allowable containment leakage rate, L_a, is 0.5 % by weight of the containment air per 24 hours at the design basis accident pressure, P_a, of 44 psig.

The surveillance requirements for measuring leakage rates are consistent with the requirements of 10 CFR 50, Appendix J, Option B, and leakage rate testing is performed in accordance with the guidelines contained in Regulatory Guide 1.163, "Performanace-Based Containment Leak-Test Program". Leakage rate testing is conducted periodically as specified in the Containment Leakage Rate Testing Program.

The periodic performance of Type A, B and C tests verifies that the containment leakage rate does not exceed the levels assumed in the safety analyses.

Secondary containment bypass leakage paths previously indentified in Table 3.6-1 are now identified in the Technical Requirements Manual.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

→(DRN 05-131, Ch. 39)

The use of the containment purge valves is restricted to 90 hours per year in accordance with Standard Review Plan 6.2.4 for plants with the Safety Evaluation Report for the Construction License issued prior to July 1, 1975. The purge valves have been modified to limit the opening to approximately 52° to ensure the valves will close during a LOCA or MSLB;and therefore, the SITE BOUNDARY doses are maintained within the guidelines of 10CFR 50.67. The purge valves, as modified, comply with all provisions of BTP CSB 6-4 except for the recommended size of the purge line for systems to be used during plant operation.

←(DRN 05-131, Ch. 39)

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 La leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

Operability concerns for purge supply and exhaust isolation valves other than those addressed in Actions "a" and "b" of Specification 3.6.1.7 are addressed under Specification 3.6.3, "Containment Isolation Valves."

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Spray System and the Containment Cooling System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or MSLB for any double-ended break of the largest reactor coolant pipe or main steam line. Under post-accident conditions these systems will maintain the containment pressure below 44 psig and temperatures below 269.3°F during LOCA conditions or 413.5°F during MSLB conditions. The systems also reduce the containment pressure by a factor of 2 from its post-accident peak within 24 hours, resulting in lower containment leakage rates and lower offsite dose rates.

→(DRN 05-131, Ch. 39)

The Containment Spray System (CSS) also provides a mechanism for removing iodine from the containment atmosphere under post-LOCA conditions to maintain doses in accordance with 10 CFR 50.67 limits as described in Section 6.5.2 of the FSAR.

←(DRN 05-131, Ch. 39)

If LCO 3.6.2.1 requirements are not met due to the condition described in ACTION (a), then the inoperable CSS train components must be returned to OPERABLE status within seven (7) days of discovery. This seven (7) day allowed outage time is based on the findings of deterministic and probabilistic analysis, CE NPSD-1045, "Modifications To The Containment Spray System, and Low Pressure Safety Injection System Technical Specifications". Seven (7) days is a reasonable amount of time to perform many corrective and preventative maintenance items on the affected CSS train. CE NPSD-1045 concluded that the overall risk impact of the seven (7) day allowed outage time was either risk-beneficial or risk-neutral.

AMENDMENT NO. ~~163~~,
CHANGE NO. ~~38~~, 39

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SHIELD BUILDING VENTILATION SYSTEM

→(DRN 05-131, Ch. 39)

The OPERABILITY of the shield building ventilation systems ensures that containment vessel leakage occurring during design basis accidents into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 50.67

←(DRN 05-131, Ch. 39)

Acceptable removal efficiency is shown by a methyl iodide penetration of less than 0.5% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 70%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analyses and is consistent with Regulatory Guide 1.52 and ASTM D3803-1989.

3/4.6.6.2 SHIELD BUILDING INTEGRITY

→(DRN 05-131, Ch. 39)

SHIELD BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the shield building ventilation system, will limit the site boundary radiation doses to within the limits of 10 CFR 50.67 during accident conditions.

←(DRN 05-131, Ch. 39)

3/4.7.1.4 ACTIVITY

→(DRN 04-1243, Ch. 38; 05-131, Ch. 39)

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR 50.67 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 540 gallons per day primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

←(DRN 04-1243, Ch. 38; 05-131, Ch. 39)

→DRN 03-1737, Ch. 31)

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE (MSIV)

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break. MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside of, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs), atmospheric dump valves, and emergency feedwater pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal (MSIS) generated by either low steam generator pressure or high containment pressure. The MSIVs fail as is on loss of power to the actuator however; the operators for the MSIV are furnished with redundant hydraulic fluid dump valves powered by diverse power, to ensure that no single electrical failure will prevent valve closure. The MSIVs may also be actuated manually.

A description of the MSIVs is found in Final Safety Analysis Report (FSAR), Section 10.3.

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, as discussed in FSAR, Section 6.2. It is also influenced by the accident analysis of the SLB events presented in FSAR, Section 15.1.3. The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The OPERABILITY of the MSIVs ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

←(DRN 03-1737, Ch. 31)

→(DRN 03-1737, Ch. 31)

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE (MSIV) (Continued)

This Limiting Condition for Operation (LCO) requires that the MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

→(DRN 05-131, Ch. 39)

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.67 limits or the NRC staff approved licensing basis.

←(DRN 05-131, Ch. 39)

The MSIVs must be OPERABLE in MODE 1 and in MODES 2, 3 and 4 except when all MSIVs are closed and deactivated. In these MODES there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

MODE 1 ACTION

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The 8 hour Allowed Outage Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

The 8 hour Allowed Outage Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the ACTION does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and the MODE 2, 3, and 4 ACTION would be entered. The Allowed Outage Time is reasonable, based on operating experience, to reach MODE 2 and close the MSIVs in an orderly manner and without challenging unit systems.

MODE 2, 3, and 4 ACTION

Since the MSIVs are required to be OPERABLE in MODES 2, 3 and 4, inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

←(DRN 03-1737, Ch. 31)

3/4.7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 27.0 ft Mean Sea Level is based on the maximum elevation at which the levee provides protection, the nuclear plant island structure provides protection to safety-related equipment up to elevation +30 ft Mean Sea Level.

→(DRN 03-656, Ch. 24)

3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM

←(DRN 03-656, Ch. 24)

3/4.7.6.1 and 3/4.7.6.2 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM

During an emergency, both S-8 units are started to provide filtration and adsorption of outside air and control room envelope recirculated air (reference: FSAR 6.4.3.3). Dosages received after a full power design basis LOCA were calculated to be orders of magnitude higher than other accidents involving radiation releases to the environment (reference: FSAR Tables 15.6-18, 15.7-2, 15.7-4, 15.7-5, 15.7-7).

Acceptable removal efficiency is shown by a methyl iodide penetration of less than 0.5% when tests are performed in accordance with ASTM D3803-1989, "Standard test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 70%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

→(DRN 05-131, Ch. 39)

The OPERABILITY of this system and control room design provisions are based on limiting the radiation exposure to personnel occupying the control room to 5 rem total effective dose equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50 and 10 CFR 50.67.

←(DRN 05-131, Ch. 39)

The ACTION to suspend all operations involving movement of irradiated fuel assemblies shall not preclude completion of movement to a safe conservative position.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analysis and is consistent with Regulatory Guide 1.52 and ASTM D3803-1989.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM (Continued)

→(DRN 03-233, Ch. 22)

The containment purge valve isolation system consists of the containment purge isolation valves (CAP-103, CAP-104, CAP-203 and CAP-204), the containment purge and exhaust isolation radiation monitors (one required per train as specified in TS 3/4.3.3), the containment purge isolation signal logic and manual isolation logic.

The ACTION statement to close each of the containment purge penetrations may be met by closing at least one valve per penetration (reference Technical Specification 3/4.9.4 and its Basis).

←(DRN 03-233, Ch. 22)

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

→(DRN 05-131, Ch. 39)

The restrictions on minimum water level ensure that sufficient water depth is available such that the iodine released as a result of a rupture of an irradiated fuel assembly is reduced by a factor of at least 200. Gap fractions are assumed in accordance with Regulatory Guide 1.183 guidance. The minimum water depth is consistent with assumptions of the safety analysis.

←(DRN 05-131, Ch. 39)

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 This section is deleted.

3/4.11.2.2 This section is deleted.

3/4.11.2.3 This section is deleted.

3/4.11.2.4 This section is deleted.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the guidelines of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

→(DRN 05-131, Ch. 39)

Note that this event has been deleted from the NRC Standard Review Plan (NUREG-0800). New Acceptance criteria were not prescribed using the Alternative Source Term dose methodology (10 CFR 50.67), therefore this specification will continue to use the dose acceptance criteria of 10 CFR 100.

←(DRN 05-131, Ch. 39)

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CHANGE NO. 40 PAGES
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Remove	**Insert**
B 3/4 1-1a	B 3/4 1-1a

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The Surveillance Requirements consisting of beginning of cycle measurements, plant parameter monitoring, and end of cycle MTC predictions ensures that the MTC remains within acceptable values. The confirmation that the measured values are within a tolerance of $\pm 0.16 \times 10^{-4}$ delta k/k/°F from the corresponding design values prior to 5% power and 40 EFPD provides assurances that the MTC will be maintained within acceptable values throughout each fuel cycle. CE NPSD 911 and CE NPSD 911 Amendment 1, "Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specifications End of Cycle Negative MTC Limit", provide the analysis that established the design margin of $\pm 0.16 \times 10^{-4}$ delta k/k/°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

→(DRN 05-896, Ch. 41)

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 520°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

←(DRN 05-896, Ch. 41)

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Remove	Insert
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Because of the short duration of the allowed outage time, the contingency Actions of a, b, or c do not have to be completed while the requirements of Action e are being followed. If one of the monitors are restored to OPERABLE status, Action e may be exited and the requirements of Action a, b, or c, whichever is applicable must be complied with.

Action f

If all required monitors are inoperable, no automatic means of monitoring leakage are available and immediate plant shutdown is required. ACTION must be initiated within 1 hour to be in MODE 3 within the next 6 hours and MODE 5 in the following 30 hours. These times are consistent with TS 3.0.3.

Surveillance Requirements

SR 4.4.5.1.a, 4.4.5.1.b - Channel Check

SR 4.4.5.1.a requires the performance of a CHANNEL CHECK of the required containment atmosphere particulate radioactivity monitor. SR 4.4.5.1.b requires the performance of a CHANNEL CHECK on the required containment sump level monitor. The CHANNEL CHECK is not required to be performed on the containment sump flow monitor (weir). The check gives reasonable confidence the channel is operating properly. The frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

→(DRN 05-1333, Ch. 44)

SR 4.4.5.1.a, SR 4.4.5.1.c - Channel Functional Test

←(DRN 05-1333, Ch. 44)

SR 4.4.5.1.a requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere particulate radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. A successful test of the required contacts of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

→(DRN 05-1333, Ch. 44)

SR 4.4.5.1.c requires the performance of a CHANNEL FUNCTIONAL TEST of the containment fan cooler condensate flow switches. The test verifies the alarm function of the channel for the instruments located inside containment by providing less than or equal to a 1 gallon per minute water flow to activate the instrument. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this frequency is acceptable.

SR 4.4.5.1.a, SR 4.4.5.1.b - Channel Calibration

←(DRN 05-1333, Ch. 44)

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this frequency is acceptable.

←(DRN 04-1223, Ch. 33)

TECHNICAL SPECIFICATION BASES
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(1 page)

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B 3/4 5-1a	B 3/4 5-1a

3/4.5.1 SAFETY INJECTION TANKS (Continued)

The TS allow operation below 1750 psia with three SITs at reduced pressure and increased volume or four SITs at reduced SIT pressure and volume. CE NPSD-994 does not address operation with less than 3 SITs. Therefore, since CE NPSD-994 is not applicable at less than 1750 psia, a separate 1 hour ACTION consistent with the Waterford 3 licensing basis is provided. The limits for operation with a safety injection tank inoperable for any reason except boron concentration or inability to verify water level and pressure minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If one of the required SITs cannot be restored within one hour, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required. If more than two SITs are inoperable, then entry into 3.0.3 is required.

→(DRN 04-1559, Ch. 36)

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. The 31 day frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.
Verifying boron concentration of the affected SIT within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water is from the Refueling Water Storage Pool (RWSP), as long as the water contained in the RWSP is within the SIT boron concentration requirements. This is consistent with the recommendations of NUREG-1366. Likewise, movement of water between SITs is within the confines of the tank system (not from an external makeup source) and is within the SIT boron concentration requirements for tank OPERABILITY, thus sampling is not required for these level changes.

The boron concentration in the SITs can be verified by either sampling or calculation. The sampling method requires a containment entry to obtain the SIT samples. The calculation method utilizes the initial and fill boron concentration and the initial, final, and fill volume of the SITs. The fill volume is the amount of delta-volume from the initial to the final volume. The fill boron concentration is the boron concentration from the source of the inleakage. If the source of the inleakage is unknown the RCS boron concentration will be used. The RCS boron concentration is the most limiting boron concentration that can leak into the SITs.

←(DRN 04-1559, Ch. 36)

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

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CHANGE NO. 37 REPLACEMENT PAGE(S)
(1 page)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 37 and contains the appropriate DRN number and a vertical line indicating the areas of change.

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B 3/4 7-2d	B 3/4 7-2d

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.

→(DRN 03-1807, Ch. 30)

b. The SR to verify pump OPERABILITY pursuant to 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.

←(DRN 03-1807, Ch. 30)

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.

→(DRN 05-42, Ch. 37)

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 acceptable, based on the design reliability and operating experience of the equipment.

←(DRN 05-42, Ch. 37)