Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093
Tel 504-739-6685
Fax 504-739-6698
jjarrel@entergy.com

John P. Jarrell
Regulatory Assurance Manager
Waterford 3

W3F1-2017-0079 | 10 CFR 50.71(e)

December 6, 2017

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

Subject: Technical Specification Index and Bases Update to the NRC for the Period June 2, 2016 through November 30, 2017
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Pursuant to Waterford Steam Electric Station Unit 3 (Waterford 3) Technical Specification (TS) 6.16, Entergy Operations, Inc. (EOI) hereby submits an update of all changes made to the Waterford 3 Technical Specification Index and Bases since the last submittal per letter W3F1-2016-0041 (ADAMS Accession No. ML14309A687), dated June 2, 2016. This update satisfies the submittal frequency required by TS 6.16, which indicates that the submittal will be made at a frequency consistent with 10 CFR 50.71(e) and exemptions thereto.

There are no commitments associated with this submittal. Should you have any questions or comments concerning this submittal, please contact the Regulatory Assurance Manager, John Jarrell, at (504) 739-6685.

Sincerely,

JPJ/LLB

Attachments:

1. Waterford 3 Technical Specification Index and Bases Change List

2. Waterford 3 Technical Specification Index and Bases Revised Pages

A001
NRR

cc: Mr. Kriss Kennedy
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
1600 East Lamar Blvd.
Arlington, TX 76011-4125

RidsRgn4MailCenter@nrc.gov

NRC Senior Resident Inspector
Waterford Steam Electric Station Unit 3
P.O. Box 822
Killona, LA 70066-0751

Frances.Ramirez@nrc.gov
Chris.Speer@nrc.gov

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

April.Pulvirenti@nrc.gov

Attachment 1 to

W3F1-2017-0079

Waterford 3 Technical Specification Index and Bases Change List

86	10/20/2016	B 3/4 1-1 B 3/4 1-1a B 3/4 1-3 B 3/4 1-4 B 3/4 1-5 B 3/4 1-5a B 3/4 1-6 B 3/4 2-1a B 3/4 2-2 B 3/4 2-3 B 3/4 2-4 B 3/4 2-5 B 3/4 3-1c B 3/4 3-1d B 3/4 3-1e B 3/4 3-2 B 3/4 3-3 B 3/4 3-3a B 3/4 3-3b B 3/4 3-4 B 3/4 4-1a B 3/4 4-2 B 3/4 4-4c B 3/4 4-4e B 3/4 4-5 B 3/4 4-7a B 3/4 4-10 B 3/4 4-11 B 3/4 5-1a B 3/4 5-1d B 3/4 5-2 B 3/4 5-3 B 3/4 6-1 B 3/4 6-2 B 3/4 6-3 B 3/4 6-4 B 3/4 6-5 B 3/4 6-5a B 3/4 6-6 B 3/4 6-7 B 3/4 7-2e	Change No. 86 to TS Bases sections 3/4 1-1 thru 3/4 11-3 was implemented by Licensing Basis Document Change Request (LBDCR) 16-046 to reflect Surveillance Frequency Control Program

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
86 (Con't)	10/20/2016	B 3/4 7-3 B 3/4 7-3c B 3/4 7-3e B 3/4 7-3i B 3/4 7-3j B 3/4 7-3j(1) B 3/4 7-4(1) B 3/4 7-4a(5) B 3/4 7-4b B 3/4 7-4d B 3/4 7-8 B 3/4 8-1a1 B 3/4 8-1c B 3/4 8-2 B 3/4 8-2a B 3/4 8-3 B 3/4 9-1 B 3/4 9-1a B 3/4 9-3 B 3/4 9-4 B 3/4 9-5 B 3/4 10-1 B 3/4 11-3	Change No. 86 to TS Bases sections 3/4 1-1 thru 3/4 11-3 was implemented by Licensing Basis Document Change Request (LBDCR) 16-046 to reflect Surveillance Frequency Control Program.
87	11/9/2016	B 3/4 8-1a1	Change No. 87 to TS Bases sections 3/4 8-1a1 was implemented by Licensing Basis Document Change Request (LBDCR) 16-018 reworded to remove the stated minimum gallons for the administrative limit. It will also be reworded to allow the administrative limit to address other conditions/effects impacting usable fuel inventory.

Attachment 1 to
W3F1-2017-0079

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
88	7/10/2017	B 3/4 7-4	Change No. 88 to TS Bases B 3/4 7-4 change the Technical Specification (TS) 3.7.4 action 'c' basis to add amplifying information to clarify that both trains of Dry Cooling Tower (DCT) fans (those under the missile shield) are required to be operable when in a tornado watch. The design basis (reference Calc. MN (Q)-9-17, "Tornado Multiple Missile Protection of Cooling Towers") assumes the tornado causes a loss of offsite power and one EDG failure and therefore a failure of one train of the DCT is assumed.
89	10/24/2017	XVI B 3/4 6-6b B 3/4 7-2e B 3/4 7-3c B 3/4 7-3e B 3/4 7-3f B 3/4 7-3g B 3/4 7-3j	Change No. 89 to TS Bases sections was implemented by License Bases Document Change Request (LBDCR) 17-032 removal of 6.5.8 (Inservice Testing Program)

Attachment 2 to

W3F1-2017-0079

Waterford 3 Technical Specification Index and Bases Revised Pages

(There are **75** unnumbered pages following this cover page)

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert.

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with (T_{cold}) at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 200°F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 200°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of SHUTDOWN MARGIN. Accordingly, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN are single CEA withdrawal and startup of an inactive reactor coolant pump.

If the SHUTDOWN MARGIN requirements are not met, boration must be initiated immediately. Boration will continue until the SHUTDOWN MARGIN requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied provided the boration source is sufficient to achieve the SHUTDOWN MARGIN. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid markup tanks or the refueling water storage pool. The Operator should borate with the best source available for the plant conditions.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

Other technical specifications that reference the Specifications on SHUTDOWN MARGIN are: 3/4.1.2, BORATION SYSTEMS, 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, 3/4.9.1, REFUELING OPERATIONS - BORON CONCENTRATION, and 3/4.10.1, SHUTDOWN MARGIN

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

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 X 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 X 10^{-4} delta k/k/°F.

→(DRN 06-814, Ch. 47)

For fuel cycles that meet the applicability requirements of WCAP-16011-P-A, Revision 0, "Startup Test Activity Reduction Program," SR 4.1.1.3.2.a may be met prior to exceeding 5% of RATED THERMAL POWER after each fuel loading by confirmation that the predicted MTC, when adjusted for the measured RCS boron concentration, is within the MTC limits. WCAP-16011-P-A also provides the basis for using only the near 40 EFPD surveillance test result to justify elimination of the near two-thirds of expected core burnup surveillance when applicability requirements are met. Performance of only one measurement at power is justified based on the WCAP-16011-P-A conclusion that ITC startup test data between different operating conditions is poolable.

The applicability requirements in WCAP-16011-P-A ensure core designs are not significantly different than those used to benchmark predictions and require that the measured RCS boron concentration meets specific test criteria. This provides assurance that the MTC obtained from the adjusted predicted MTC is accurate.

For fuel cycles that do not meet the applicability requirements in WCAP-16011-P-A, the verification of MTC required prior to entering MODE 1 after each fuel loading is performed by measurement of the isothermal temperature coefficient.

←(DRN 06-814, Ch. 47)

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

→(DRN 05-896, Ch. 41; 06-790, Ch. 46)

This specification ensures that the reactor will not be made critical with the indicated Reactor Coolant System cold leg temperature less than 533°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; 06-790, Ch. 46)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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 addressed 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)

continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

→(LBDCR 16-046, Ch. 86)

CEA positions and OPERABILITY of the CEA position indicators are required to be verified in accordance with the Surveillance Frequency Control Program with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

The arithmetic average CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. The maximum CEA drop time restriction limits the CEA drop time distribution about the average to that used to support the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the meausured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. The CEA drop time restriction is representative of the design and operating conditions for Cycle 3 and reverification may be required for (1) any fuel management change that significantly affects the core wide axial or radial power profiles, and (2) any mechanical, flow, control, or CEA location changes that would significantly affect the CEA drop time distribution.

The establishment of LSSS and LCOs requires that the expected long and short-term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base loaded, or load maneuvering) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses.

MOVABLE CONTROL ASSEMBLIES (Continued)

The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors. Insertion of Reg. Groups 5 and 6 is permitted to be essentially tip-to-tip within the limits imposed by the

MOVABLE CONTROL ASSEMBLIES (Continued)

Transient Insertion Limit Line. This method of insertion is protected from sequence errors by the Core Protection Calculators.

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

→ (DRN 02-632)

← (DRN 02-632)

The additional uncertainty terms included in the CPC's for transient protection are credited in the limits specified in the COLR since this curve is intended to monitor the LCO only during steady state operation.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→ (DRN 03-656, Ch. 24)

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - Fxy

← (DRN 03-656, Ch. 24)

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic Surveillance Requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provide assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. The LCO requires the maximum azimuthal tilt during normal steady state power operation to be less than or equal to that specified in the COLR. With AZIMUTHAL POWER TILT greater than the limit specified in the COLR, operation is restricted to only those conditions required to identify the cause of the tilt. However, Action item b.2 allows 24 hours to restore the tilt to less than or equal to the limit specified in the COLR following a CEA misalignment event (i.e., CEA drop). A CEA misalignment event causes an asymmetric core power generation and an increase in xenon concentration in the vicinity of the dropped rod. This event may cause the azimuthal tilt to exceed the limit specified in the COLR. The 2 hour action time to reduce core power is not sufficient to recover from the xenon transient. The 24 hour period allows for correction of the misaligned CEA and allows time for the xenon redistribution effects to dampen out due to radioactive decay and absorption. The reduction in xenon concentration (which is aided by operation at full power) will in turn reduce the tilt below the COLR limit.

The 24 hour period is applicable only to a CEA misalignment where the cause of the tilt has been identified. It is based on the time required or the expected xenon transient to dampen out. All other conditions (not due to a CEA misalignment) where the azimuthal tilt exceeds the limit specified in the COLR require action within the specified 2 hours.

The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The Surveillance Requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries in the COLSS and CPCs because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT - Tq (Continued)

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$P_{tilt}/P_{untilt} = 1 + T_q\, g \cos(\theta - \theta_o)$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provides adequate monitoring of the core power distribution and is capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide a 95/95 probability/confidence level that the core power calculated by COLSS, based on the minimum DNBR limit, is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurements, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits specified in the COLR can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors plus those associated with startup test acceptance criteria are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

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.

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

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).

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION (Continued)

When one of the inoperable channels is restored to OPERABLE status, subsequent operation in the applicable MODE(S) may continue in accordance with the provisions of ACTION 19.

Because of the interaction between process measurement circuits and associated functional units as listed in the ACTIONS 19 and 20, placement of an inoperable channel of Steam Generator Level in the bypass or trip condition results in corresponding placements of Steam Generator ΔP (EFAS) instrumentation. Depending on the number of applicable inoperable channels, the provisions of ACTIONS 19 and 20 and the aforesaid scenarios for Steam Generator ΔP (EFAS) would govern.

→(LBDCR 16-046, Ch. 86)

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the frequencies in the Surveillance Frequency Control Program are sufficient to demonstrate this capability. The frequency for the channel functional tests for these systems is controlled by the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(LBDCR 16-046, Ch. 86)

Testing frequency for the Reactor Trip Breakers (RTBs) is controlled by the Surveillance Frequency Control Program. The RTB channel functional test and RPS logic channel functional test are scheduled and performed such that RTBs are verified OPERABLE in accordance with the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

RPS\ESFAS Trip Setpoints values are determined by means of an explicit setpoint calculation analysis. A Total Loop Uncertainty (TLU) is calculated for each RPS/ESFAS instrument channel. The Trip Setpoint is then determined by adding or subtracting the TLU from the Analytical Limit (add TLU for decreasing process value; subtract TLU for increasing process value). The Allowable Value is determined by adding an allowance between the Trip Setpoint and the Analytical Limit to account for RPS/ESFAS cabinet Periodic Test Errors (PTE) which are present during a CHANNEL FUNCTIONAL TEST. PTE combines the RPS/ESFAS cabinet reference accuracy, calibration equipment errors (M&TE), and RPS/ESFAS cabinet bistable Drift. Periodic testing assures that actual setpoints are within their Allowable Values. A channel is inoperable if its actual setpoint is not within its Allowable Value and corrective action must be taken. Operation with a trip set less conservative than its setpoint, but within its specified ALLOWABLE VALUE is acceptable on the basis that the difference between each trip Setpoint and the ALLOWABLE VALUE is equal to or less than the Periodic Test Error allowance assumed for each trip in the safety analyses.

>(EC-26338, Ch. 67)

The Core Protection Calculator, High Logarithmic Power (HLP), and Reactor Coolant System Flow use a single bistable to initiate both the permissive and automatic operating bypass removal functions. A single bistable cannot both energize and de-energize at a single, discrete value due to hysteresis. The CPC automatic bypass removal and permissive for the

<(EC-26338, Ch. 67)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION (Continued)

>(EC-26338, Ch. 67)
HLP trip bypass occur at the bistable setpoint (nominally 10^{-4}% power). However, the HLP automatic bypass removal and permissive for CPC trip bypass occur at the reset value of the bistable. Also note if the bistable setpoint is changed as part of the Special Test Exception 3.10.3, the same dead band transition is applicable.
<(EC-26338, Ch. 67)

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be verified by any series of sequential, overlapping, or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the topical report. Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.
>(EC-26338, Ch. 67)

In the applicable logarithmic power modes, with the Logarithmic Power circuit inoperable or in test, the associated functional units of Local Power Density-High, DNBR-Low, and Reactor Coolant Flow-Low should be placed in the bypassed or tripped condition. With logarithmic power greater than 10^{-4}% bistable setpoint and Local Power Density-High, DNBR-Low, and Reactor Coolant Flow-Low no longer bypassed (either through automatic or manual action), these functional units may be considered OPERABLE.
<(EC-26338, Ch. 67)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

TABLE 3.3-1, Functional Unit 13, Reactor Trip Breakers

The Reactor Trip Breakers Functional Unit in Table 3.3-1 refers to the reactor trip breaker channels. There are four reactor trip breaker channels. Two reactor trip breaker channels with a coincident trip logic of one-out-of-two taken twice (reactor trip breaker channels A or B, and C or D) are required to produce a trip. Each reactor trip breaker channel consists of two reactor trip breakers. For a reactor trip breaker channel to be considered OPERABLE, both of the reactor trip breakers of that reactor trip breaker channel must be capable of performing their safety function (disrupting the flow of power in its respective trip leg). The safety function is satisfied when the reactor trip breaker is capable of automatically opening, or otherwise opened or racked-out.

If a racked-in reactor trip breaker is not capable of automatically opening, the ACTION for an inoperable reactor trip breaker channel shall be entered. The ACTION shall not be exited unless the reactor trip breaker capability to automatically open is restored, or the reactor trip breaker is opened or racked-out.

>(EC-12084, Ch. 57)

TABLES 3.3-3 and 4.3-2, Functional Unit 6, Loss of Power (LOV)

The Loss of Power Functional Unit 6 in Tables 3.3-3 and 4.3-2 refers to the undervoltage relay channels that detect a loss of bus voltage on the 4kV (A3 & B3) and 480V (A31 & B31) safety buses and a sustained degraded voltage condition on 4kV (A3 & B3) safety buses. The intent of these relays is to ensure that the Emergency Diesel Generator starts on a loss of voltage or a sustained degraded voltage condition. The response time SR in TS 3.3.2 ensures that Bus A3 and B3 undervoltage relays trip and generate a Loss of Voltage (LOV) signal in 2 seconds for initiation of the EDG start. The response time for Bus AB3 and AB31 relays is not as critical as the Bus A3 and B3 undervoltage relays. Bus AB3 and AB31 undervoltage relays [4KVEREL3AB-1A(1B)(1C) and SSDEREL31AB-1A(1B)(1C)] strip bus loads upon an undervoltage condition to preclude any perturbations which might affect the A and B buses and prepare the bus to be energized by an EDG with subsequent loading by the sequencer. Bus AB3 and AB31 undervoltage relays do not provide an EDG start signal. Therefore, TS 3/4.3.2, Tables 3.3-3 and 4.3-2 functional unit 6 requirements, are not applicable to AB3 Bus and AB31 Bus undervoltage relays.

If an AB Bus undervoltage relay becomes inoperable, initiate a condition report and consider operability of the associated EDG based on the AB Bus loads when evaluating the failure.

<(EC-12084, Ch. 57)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION (Continued)

>(DRN 03-871, Ch. 27)

The Steam Generator Blowdown Process Radiation Monitor and the Component Cooling Water Process Radiation Monitors A, B, and A/B are designed to detect leakage into the monitored system from components that may contain radioactive contamination. These process monitors have an alarm function that annunciates when activity levels at or above the alarm setpoints are detected. This alarm provides an opportunity for the operator to isolate the system and/or equipment and perform investigative activities to locate and repair the source of leakage. By design, the sample flow for these monitors is provided by the hydraulic head established in the monitored system during system operation. When flow in the monitored system is terminated, which would occur if the system was being taken out of service for maintenance, the monitor will go into an alarmed condition due to loss of sample flow. If this alarmed condition is due solely to the termination of the flow in the monitored system, and the process monitors were OPERABLE prior to flow termination, then these radiation monitors should be considered OPERABLE. Therefore, the performance of ACTION 28 is not appropriate or required for this condition. During this condition, the monitors are effectively in a standby state and are capable of automatically performing their intended safety function once flow is re-established in the monitored system. The performance of the channel check (and other surveillances, if required) should continue during this condition to maintain compliance with the requirements of this Technical Specification.

<(DRN 03-871, Ch. 27)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

>(EC-38571, Ch. 71)

The fuel handling accident (UFSAR Section 15.7.3.4) analysis assumes protection against load movements with or over irradiated fuel assemblies that could cause fuel assembly damage. Examples of load movements include movement of new fuel assemblies, irradiated fuel assemblies, and the dummy fuel assembly. The load movements do not include the movement over assemblies in a transfer cask using a single-failure-proof handling system. The load movements do not include the movement of the spent fuel machine or refuel machine without loads attached. It also does not include load movements in containment when the reactor vessel head or Upper Guide Structure is still installed. Load movements also exclude suspended loads weighing less than 1000 lbm (e.g. Westinghouse analysis CN-NFPE-09-57 describes no fuel failure for loads weighing less than 1000 lbm based upon the 2000 lbm analysis for drops distributed over two assemblies).

<(EC-38571, Ch. 71)

3/4.3.3.2 INCORE DETECTORS

This section has been deleted.

3/4.3.3.3 SEISMIC INSTRUMENTATION

This section has been deleted.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

This section has been deleted.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by plant specific documents addressing the recommendations of Regulatory Guide 1.97, as required by Supplement 1 to NUREG-0737, "TMI Action Items." Table 3.3.10 includes most of the plant's RG 1.97 Type A and Category 1 variables. The remaining Type A/Category 1 variables are included in their respective specifications. Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).
Category 1 variables are the key variables deemed risk significant because they are needed to: (1) Determine whether other systems important to safety are performing their intended functions; (2) Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and (3) Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

>(DRN 03-656, Ch. 24)

With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, the inoperable channel should be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring accident monitoring instrumentation during this interval. If the 30 day AOT is not met, a Special Report approved by OSRC is required to be submitted to the NRC within the following 14 days. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Actions. This Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Actions are identified before a loss of functional capability condition occurs.

<(DRN 03-656, Ch. 24)

With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; at least one of the inoperable channels should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Continuous operation with less than Minimum Channels OPERABLE requirements is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the accident monitoring instrumentation. Therefore, requiring restoration of one inoperable channel limits the risk that the variable will be in a degraded condition should an accident occur. If the 7 day requirement is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 4 within 12 hours. The completion time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

TS 3/4.3.3.6 applies to the following instrumentation: ESFIPI6750 A, ESFIPR6750 B, ESFIPR6755 A&B, RC ITI0122 HA, RC ITI0112 HB, RC ITI0122 CA, RC ITI0112 CB, RC IPI0102 A,B,C,&D, RC ILI0110 X&Y, SG ILI1113 A,B,C,&D, SG ILI1123 A,B,C,&D, SG ILI1115 A2&B2, SG ILI1125 A2&B2, SI ILI7145 A, SI ILR7145 B, all CET's, all Category 1 Containment Isolation Valve Position Indicators, EFWILI9013 A&B, HJTC's, and ENIIJI0001 C&D.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.3.3.7 CHEMICAL DETECTION SYSTEMS

The chemical detection systems are the chlorine and broad range toxic gas detection systems.

The OPERABILITY of the chemical detection systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chemical release.

The chemical detection systems provide prompt detection of toxic gas releases which could pose an actual threat to safety of the nuclear power plant or significantly hamper site personnel in performance of duties necessary for the safe operation of the plant.

The broad range toxic gas detection system utilizes a Fourier Transform Infrared (FTIR) analysis technique, and therefore, the system is sensitive to a broad range of gases including ammonia. The system is sensitive to normal fluctuations of both atmospheric and chemical composition which affect the Waterford 3 site. The setpoints associated with the system are based on testing and operating experience. Setpoints are set based on control room habitability calculations as described in the FSAR, while providing reliable operation and the optimum detection of toxic gases. The setpoint is therefore subject to change with operating experience such as a result of changes in the Waterford 3 area chemical inventory. The setpoint is established and controlled by procedure.

The LCO and ACTIONs for the broad range gas detection system are annotated such that the system instrument automatic background/reference spectrum check does not constitute system inoperability under the following conditions: (1) both channels are operable and (2) both channels are not performing the check simultaneously. The instrument automatically performs the background/reference spectrum check. During the time that the automatic background/reference spectrum check is taking place (which will be two minutes or less), the channel will not perform the function of isolation of the control room. With both channels OPERABLE, the other system will be available to perform the control room isolation function in

the event of a toxic gas incident. With one channel taken out of service (e.g., for maintenance), when the second channel performs the automatic background/reference spectrum check, both channels will be unable to perform the function of isolating the control room for the short time of the background/reference spectrum check. Qualitative analysis based on a quantitative risk assessment has shown that the impact on operator incapacitation and subsequent core damage risk of the background/reference spectrum check while one monitor is out of service for its 7 day allowed outage time is negligible. Therefore, entry into the ACTION solely due to the automatic background/reference spectrum check is not required.

No specific manual CHANNEL CALIBRATION is required as the system instrument performs this function as the background/reference spectrum check automatically for two minutes or less on a frequency of once every hour to once every four hours. The exact frequency is established based on operating experience with the instrument.

→(LBDCR 16-046, Ch. 86)

A CHANNEL CHECK is performed in accordance with the Surveillance Frequency Control Program to compare channel indications of the same parameter. The performance of the CHANNEL CHECK ensures that a gross failure of the instrument has not occurred. Significant deviations from the expected readings and actual readings could be an indication of a malfunction within the unit. The CHANNEL CHECK will detect gross system failure; thus, it is the key to verifying the instrument continues to operate properly between each CHANNEL FUNCTIONAL TEST.

←(LBDCR 16-046, Ch. 86)

A CHANNEL FUNCTIONAL TEST is performed to ensure the entire channel will perform its required function. This test includes introduction of a standard gas and verification of isolation of the control room. The time of the occurrence of the background/reference spectrum check is set during the CHANNEL FUNCTIONAL TEST such that both channels are not out of service simultaneously.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.3.3.8 This section deleted

3/4.3.3.9 This section deleted

3/4.3.3.10 This section has been deleted.

3/4.3.3.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6 x 10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety

SAFETY VALVES (Continued)

valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the overpressure protection system provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no
reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during reactor shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized while uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an SIAS test signal the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is used to depressurize the RCS by cooling the pressurizer steam space. The auxiliary pressurizer spray is used during those periods when normal pressurizer spray is not available, such as the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available.

The auxiliary pressurizer spray is used, in conjunction with the of the HPSI pumps, during the recovery from a steam generator tube rupture accident. The auxiliary pressurizer spray is also used during a natural circulation cooldown as a safety related means of RCS depressurization to achieve shutdown cooling system initiation conditions and subsequent COLD SHUTDOWN per the requirements of Branch Technical Position (RSB) 5-1.
→(DRN 06-916, Ch. 48)
←(DRN 06-916, Ch. 48)

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

→ (DRN 04-1223, Ch. 33)

→(DRN 07-203, Ch. 52)
Action c
←(DRN 07-203, Ch. 52)

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
→(DRN 07-203, Ch. 52; LBDCR 16-046, Ch. 86)
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/time rate of change. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(DRN 07-203, Ch. 52; LBDCR 16-046, Ch. 86)

→(DRN 05-1333, Ch. 44)
SR 4.4.5.1.a, - Channel Functional Test
←(DRN 05-1333, Ch. 44)

→(LBDCR 16-046, Ch. 86)
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 in accordance with the Surveillance Frequency Program. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

→(DRN 05-1333, Ch. 44, DRN 07-203, Ch. 52)

←(DRN 07-203, Ch. 52)

SR 4.4.5.1.a, SR 4.4.5.1.b - Channel Calibration
←(DRN 05-1333, Ch. 44)

→(LBDCR 16-046, Ch. 86)
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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(DRN 04-1223, Ch 33, LBDCR 16-046, Ch. 86)

3/4.4.5.2 OPERATIONAL LEAKAGE

> (EC-3173 Ch. 53)

MODE in the Applicability of the associated LCO if any of the following conditions are satisfied: (1) the SR has been performed within the surveillance interval (i.e. it is current) and is known not to be failed or (2) the SR is required to be met, but not performed, in the MODE to be entered and is known not to be failed. The initial surveillance performance will be completed within 12 hours once the plant is at stable operating pressure following the establishment of steady state conditions. Other instruments such as those contained in TS 3/4.4.5.1 can be utilized to determine whether RCS operational leakage limits are being exceeded prior to initial performance.

→(LBDCR 16-046, Ch. 86)

Once the plant establishes steady state operation, 12 hours is allowed for completing the SR. If the SR was not performed within this 12 hour interval, there would then be a failure to perform the SR within the specified interval, and the provisions of 4.0.3 would apply. Should the interval in accordance with the Surveillance Frequency Control Program be exceeded while steady state operation has not been established, this NOTE allows 12 hours after steady state operation has been established to perform the SR. The SR is still considered to be performed within the surveillance interval. Therefore, if the Surveillance was not performed in accordance with the Surveillance Frequency Control Program (plus the extension allowed by 4.0.2) interval, but steady state operation was not established, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of 4.0.4 occurs when changing MODES, even with the surveillance interval in accordance with the Surveillance Frequency Control Program not met, provided operation does not exceed 12 hours with the establishment of steady state operation.

< (EC-3173 Ch. 53)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

> (DRN 04-1243, Ch. 38; 06-916, Ch. 48)

The primary to secondary leakage limit of 75 gallons per day through any one SG is based on the operational leakage performance criterion in NEI 97-06. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The NEI 97-06 limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion (since it is less than 150 gpd through any one SG) in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

< (DRN 04-1243, Ch. 38, 06-916, Ch. 48)

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.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

>(LBDCR 13-003, Ch. 74)

3/4.4.6 DELETED

<(LBDCR 13-003, Ch. 74)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(DRN 03-1807, Ch. 30)

3/4.4.9 STRUCTURAL INTEGRITY This section is deleted.

←(DRN 03-1807, Ch. 30)

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

BASES

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; LBDCR 16-046, Ch. 86)

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.

←(LBDCR 16-046, Ch. 86)

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.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

ECCS SUBSYSTEMS (Continued)

When in mode 3 and with RCS temperature greater than or equal to 500°F two OPERABLE ECCS subsystems are required to ensure sufficient emergency core cooling capability is available to prevent the core from becoming critical during an uncontrolled cooldown (i.e., a steam line break) from greater than 500°F.

With the RCS temperature below 500°F and the RCS pressure below 1750 psia, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0. The requirement to dissolve a representative sample of TSP in a sample of water borated to be representative of post-LOCA sump conditions provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures. A boron concentration of 3011 ppm boron is postulated to be representative of the highest post-LOCA sump boron concentration. Post LOCA sump pH will remain between 7.0 and 8.1 for the maximum (3011 ppm) and minimum (1504 ppm) boron concentrations calculated using the maximum and minimum post-LOCA sump volumes and conservatively assumed maximum and minimum source boron concentrations.

→ (DRN 02-1635, Ch. 16, DRN 03-445, Ch. 26; LBDCR 16-046, Ch. 86)

With the exception of systems in operation, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will prevent water hammer, pump cavitation, and pumping noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SIAS or during SDC. The LPSI system has been evaluated for voids in the discharge piping. The piping system has been qualified for the hydraulic transient. In addition, the reactor has been qualified for an intrusion of a small gas bubble. Therefore, from a design basis standpoint, for injection capacity and prevention of water hammer, pump cavitation, and pumping noncondensible gas the LPSI system will be considered operable and full of water with the existence of voids in the system discharge legs. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

← (DRN 02-1635, Ch. 16, DRN 03-445, Ch. 26, LBDCR 16-046, Ch. 86)

ECCS SUBSYSTEMS (Continued)

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

The requirement to verify the minimum pump differential pressure on recirculation flow ensures that the pump performance curve has not degraded below that used to show that the pump exceeds the design flow condition assumed in the safety analysis and is consistent with the requirements of ASME Section XI.

EMERGENCY CORE COOLING SYSTEMS

BASES

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.

→(DRN 06-188, Ch. 45)

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. A RWSP minimum temperature of 50°F is the analytical limit assumed in the accident analyses. The TS minimum temperature of 55°F is specified to protect this analytical limit.

←(DRN 06-188, Ch. 45)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.6 CONTAINMENT SYSTEMS

BASES

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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.

→(LBDCR 15-029, Ch. 82)

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 (NEI) 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008.

←(LBDCR 15-029, Ch. 82)

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 identified 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.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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

→(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.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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.

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM (con't)

Action (b) addresses the condition in which two CSS trains are inoperable and requires restoration of at least one spray system to OPERABLE status within 1 hour or the plant to be placed in HOT STANDBY in 6 hours and COLD SHUTDOWN within the following 30 hours. (COLD SHUTDOWN is the acceptable end state.)

In MODE 4 when shutdown cooling is placed in operation, the Containment Spray System is realigned in order to allow isolation of the spray headers. This is necessary to avoid a single failure of the spray header isolation valve causing Reactor Coolant System depressurization and inadvertent spraying of the containment. To allow for this realignment, the Containment Spray System may be taken out-of-service when RCS pressure is ≤ 400 psia. At this reduced RCS pressure and the reduced temperature associated with entry into MODE 4, the probability and consequences of a LOCA or MSLB are greatly reduced. The Containment Cooling System is required OPERABLE in MODE 4 and is available to provide depressurization and cooling capability.

The Containment Cooling System consists of two redundant trains and is designed such that a single failure does not degrade the systems' ability to provide the required heat removal capability. A train of Containment Cooling consists of two fans (powered from the same safety bus) and their associated coolers (supplied from the same cooling water loop). An operable train of containment cooling consists of one of the two fans and its associated cooler. One Containment Cooling train, consisting of one fan and its associated cooler, and a Containment Spray train has sufficient capacity to meet post accident heat removal requirements and maintain containment temperatures and pressures below the design values.

Operating each containment cooling train fan unit for 15 minutes and verifying a cooling water flow rate of 625 gpm ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken.

→(LBDCR 13-006, Ch. 76; LBDCR 16-046, Ch. 86)

Verifying the 625 gpm to each cooler in accordance with the Surveillance Frequency Control Program with only one cooler aligned per train at a time provides a reliable representation of cooler operability. Measuring the flow through one cooler at a time provides more accurate characterization of each cooler condition than measuring the flow through two parallel coolers at the same time, the latter of which may mask flow degradation in a single cooler. Performing this portion of the surveillance with only one cooler aligned per train will avoid this potential misrepresentation of cooler condition related to blockage.

←(LBDCR 13-006, Ch. 76, LBDCR 16-046, Ch. 86)

→(LBDCR 16-046, Ch. 86)

The 4.6.2.2.b Surveillance Requirement verifies that each containment cooling fan actuates upon receipt of an actual or simulated SIAS actuation signal in accordance with the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

Verifying a cooling water flow rate of 1200 gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved. The safety analyses assumed a cooling water flow rate of 1100 gpm. The 1200 gpm requirement accounts for measurement instrument uncertainties and potential flow degradation.

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM (Continued)

→(LBDCR 16-046, Ch. 86)
The 4.6.2.2.b flow measurement test shall be done in accordance with the Surveillance Frequency Control Program in a configuration equivalent to the accident lineup to ensure that in an accident situation adequate flow will be provided to the containment fan coolers for them to perform their safety function.

Verifying that each valve actuates to the full open position provides further assurance that the valves will travel to their full open position on a Safety Injection Actuation Signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.
→(DRN 03-666, Ch. 25)
The asterisk "*" footnote associated with the LCO statement allows the opening of closed containment isolation valves on an intermittent basis under administrative controls. The valves within the scope of this footnote include locked or sealed closed containment isolation valves and deactivated automatic containment isolation valves secured in the isolation position. Acceptable administrative controls must include the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.
←(DRN 03-666, Ch. 25)
"Containment Isolation Valves", previously Table 3.6-2, have been incorporated into the Technical Requirements Manual (TRM).

For penetrations with multiple flow paths, only the affected flow path(s) is required to be isolated when a containment isolation valve in that flow path is inoperable. The flow path may be isolated with the inoperable valve in accordance with the Action requirements, provided the leakage rate acceptance criteria, as applicable, is met and controls are in place to ensure the valve is closed. Also, the penetration is required to meed the requirements of GDC-54, and GDC-55 through GDC 57, as applicable, for all the unisolated flow paths.
→(EC-14681, Ch. 58)

The allowed outage time of 72 hours for isolating a penetration associated with a closed system is consistent with Technical Specification Task Force Traveler TSTF-30. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system, which is a line that

←(EC-14681, Ch. 58)

→(EC-14681, Ch. 58)

penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere. The affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

The 72 hour allowed outage time provides the necessary time to perform repairs on a failed containment isolation valve while relying on an intact closed system. This allowed outage time is acceptable considering the reliability of closed systems to act as a penetration boundary. Furthermore, 72 hours is typically provided for the loss of one train of redundancy (similar to inoperability of a containment isolation valve in a closed system penetration) throughout the Technical Specifications.

The Waterford 3 closed system penetrations that would be applicable to this action requirement are Blowdown (Containment Penetrations 5 & 6), the Component Cooling Water for Containment Fan Coolers (Containment Penetrations 15 - 22) and Emergency Feedwater, Main Feedwater (Containment Penetrations 3 & 4), and Main Steam (Containment Penetrations 1 & 2), and Secondary Sampling (Containment Penetrations 52 & 68). The closed systems associated with these penetrations are subject to a containment Type A leak rate test and are designed as safety class 2 and seismic category 1. These systems are systems in accordance with FSAR Section 6.2.4.1.2, The closed systems meet the criteria in SRP 6.2.4.

←(EC-14681, Ch. 58)

→(DRN 03-1541, Ch. 29)

For the Shutdown Cooling System suction line relief valves (SI-406A and SI-406B), TS 3/4.6.3 is only applicable in the close direction. The capability of these valves to lift at the specified setpoint is addressed by TS 3.4.8.3.

←(DRN 03-1541, Ch. 29)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program..

←(LBDCR 16-046, Ch. 86)

→(DRN 04-971, Ch. 32)

←(DRN 04-971, Ch. 32)

3/4.6.5 VACUUM RELIEF VALVES

The vacuum relief valves protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of Containment Spray System. Multiple equipment failures or human errors are necessary to have inadvertent actuation.

The containment pressure vessel contains two 100% vacuum relief lines installed in parallel that protect the containment from excessive external loading. The vacuum relief lines are 24 inch penetrations that connect the shield building annulus to the containment. Each vacuum relief line is isolated by a pneumatically operated butterfly valve in series with a check valve located on the containment side of the penetration.

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.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM (Continued)

Surveillance Requirements (Continued)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

>(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; LBDCR 16-046, Ch. 86)

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(DRN 05-42, Ch. 37, LBDCR 16-046, Ch. 86)

This SR is modified to indicate that 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 cannot be applied to the 24 hour period.

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

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; LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

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.

→(LBDCR 16-046, Ch. 86)

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 in accordance with the Surveillance Frequency Control Program. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

←(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, 05-1650)

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. The design basis correlates a static test utilizing one accumulator to demonstrate the ability of the MFIVs to close in less than or equal to 6 seconds under design basis accident conditions with two accumulators. The static stroke time test that utilizes one accumulator is allowed to exceed the 6 second Surveillance Requirement since both accumulators are credited in the design basis Accidents in order to isolate within the 6 second Surveillance Requirement. The 6 second required closure time includes a 1 second allowance for instrument response time.
← (DRN 05-1650)
→(LBDCR 16-046, Ch. 86)

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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.
←(LBDCR 16-046, Ch. 86)
→(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 Isolation Valves to Technical Specifications and Request for NRC Staff Review of an Unreviewed Safety Question.")

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

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

In MODES 1, 2, 3, and 4, the MFIVs are required to be OPERABLE. Because the MFIVs are required to be OPERABLE in MODES 1, 2, 3, and 4, RTO must be able to decrease SGFP
←(DRN 02-1684, Ch. 15

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3.4.7.1.7 ATMOSPHERIC DUMP VALVES (ADVs) (Continued)

→(LBDCR 16-046, Ch. 86)

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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.

→(LBDCR 16-046, Ch. 86)

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

←(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

→(LBDCR 16-046, Ch. 86)

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.7.4 ULTIMATE HEAT SINK (Continued)

>(LBDCR-12-001, Ch. 73)
with fan requirements have been rounded in the conservative direction and lowered at least one full degree to account for minor inaccuracies. Failure to meet the OPERABILITY requirements of Table 3.7-3 requires entry into the applicable action. Because temperature is 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.
<(LBDCR-12-001, Ch. 73)

>(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)

>(EC-38632, Ch. 72)
Surveillance Requirements

b. This SR demonstrates OPERABILITY of the wet and dry tower fans corresponding to the accident configuration, which for the dry tower fans is in fast speed.
<(EC-38632, Ch. 72)

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

→(EC-15550, Ch. 59)

3/4.7.6.1 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM (CREAFS) (Continued)

Surveillance Requirements

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

g. This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is not greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Action b must be entered. Action b.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 1) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 2). These compensatory measures may also be used as mitigating actions as required by Action b.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 3). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

References

1. Regulatory Guide 1.1.96
2. NEI 99-03, "Control Room Habitability Assessment," June 2001.
3. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability," (ADAMS Accession No. ML040300694).

3/4.7.6.2 [NOT USED]

←(EC-15550, Ch. 59)

PLANT SYSTEMS

BASES

3/4.7.6.3 and 3/4.7.6.4 CONTROL ROOM AIR TEMPERATURE

Maintaining the control room air temperature less than or equal to 80°F ensures that (1) the ambient air temperature does not exceed the allowable air temperature for continuous duty rating for the equipment and instrumentation in the control room, and (2) the control room will remain habitable for operations personnel during plant operation.

The Air Conditioning System is designed to cool the outlet air to approximately 55°F. Then, non-safety-related near-room heaters add enough heat to the air stream to keep the rooms between 70 and 75°F. Although 70 to 75°F is the normal control band, it would be too restrictive as an LCO. Control Room equipment was specified for a more general temperature range to 45 to 120°F. A provision for the CPC microcomputers, which might be more sensitive to heat, is not required here. Since maximum outside air make-up flow in the normal ventilation mode comprises less than ten percent of the air flow from an AH-12 unit, outside air temperature has little affect on the AH-12s cooling coil heat load. Therefore, the ability of an AH-12 unit to maintain control room temperature in the normal mode gives adequate assurance of its capability for emergency situations.
>(EC-38571, Ch. 71)

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

The fuel handling accident (UFSAR Section 15.7.3.4) analysis assumes protection against load movements with or over irradiated fuel assemblies that could cause fuel assembly damage. Examples of load movements include movement of new fuel assemblies, irradiated fuel assemblies, and the dummy fuel assembly. The load movements do not include the movement over assemblies in a transfer cask using a single-failure-proof handling system. The load movements do not include the movement of the spent fuel machine or refuel machine without loads attached. It also does not include load movements in containment when the reactor vessel head or Upper Guide Structure is still installed. Load movements also exclude suspended loads weighing less than 1000 lbm (e.g. Westinghouse analysis CN-NFPE-09-57 describes no fuel failure for loads weighing less than 1000 lbm based upon the 2000 lbm analysis for drops distributed over two assemblies).
<(EC-38571, Ch. 71)

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

>(EC-15550, Ch. 59)
3/4.7.6.5 [NOT USED]

<(EC-15550, Ch. 59)

3/4.7.7 CONTROLLED VENTILATION AREA SYSTEM (Continued)

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.

→(LBDCR 16-046, Ch. 86)

Operation of the system with the heaters on for at least 10 hours continuous in accordance with the Surveillance Frequency Control Program 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.

The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.7.12 ESSENTIAL SERVICES CHILLED WATER SYSTEM (Continued)

system to operate in such a manner that ≤ 42°F and/or ≥ 500 gpm may not be directly met, yet CHW System Operability is maintained. During normal operation, when there is insufficient heat load, the following conditions may apply, but the CHW System is still OPERABLE.

(1) The chilled water operational flow control valves for Control Room Ventilation Unit AH-12 and Switchgear Ventilation Units AH-25 and AH-30, control the flow rate through the cooling coils based on discharge air temperature. If there is insufficient load, the flow control valves may be at a minimum, thus, reducing the total chilled water train flow rate to <500 gpm.

2) The CHW System chillers are equipped with a Hot Gas Bypass Valve which opens when chilled water inlet temperature is reduced significantly. This indicates the available heat load on the operating chiller is reduced to a point it will begin to auto recycle if the valve is not opened. This valve diverts a portion of hot compressor discharge gas directly to the bottom of the evaporator instead of sending it to the condenser. This diversion artificially increases the evaporators refrigerant pressure and temperature which in turns increases the chilled water outlet temperature. The increased chilled water outlet temperature eventually increases the chilled water inlet temperature which then closes the Hot Gas Bypass Valve. This operation allows the chiller to stay running at minimum heat loads, down to approximately 10% rated capacity, but allows the chilled water outlet temperature to cycle. Due to this cycling, the peak chilled water outlet temperature may be >42°F. During DBA conditions, air handling unit cooling coil heat loads would be increased which results in the Hot Gas Bypass Valve going to the closed position.

→(DRN 03-1046, Ch. 28)

3) If the Hot Gas Bypass Valve does not open (i.e., is not operational), as described in Item 2, the chiller will auto recycle based on low chilled water outlet temperature. The chiller will automatically secure at a preset low temperature, then automatically restart when the chilled water temperature increases past the reset deadband of the switch. The reset deadband for the switch allows the chilled water outlet temperature to be >42°F. As chiller loading is increased (as would occur during a DBA) the chiller will load sufficiently to reduce chilled water outlet temperature ≤42°F.

←(DRN 03-1046, Ch. 28)
→(LBDCR 16-046, Ch. 86)

The Surveillance Requirement (SR) to verify the chilled water outlet temperature in accordance with the Surveillance Frequency Control Program is ≤42°F at a flow rate of ≥500 gpm ensures the assumptions of the DBA are preserved. This SR will be performed with sufficient heat load to ensure the Hot Gas Bypass Valve is closed and the chiller is not auto recycling on low load. This may require shifting loads from one chilled train to one being tested. This requirement is reflective of an actual post DBA condition, and ensures the chiller will control the chilled water outlet temperature within limits when sufficient heat load is applied.

The Surveillance Frequencies are based on operating experience, equipment reliability, and plant risk and are controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

The NRC evaluation section in Safety Evaluation of Amendment No. 157, for the EDG FOST not having 10% margin in fuel oil inventory, credited acceptability of the design based upon Waterford 3 having EDG Fuel Oil Storage and Transfer Systems cross connecting capabilities. With the ability to cross-tie the two EDG Fuel Oil Storage and Transfer Systems, one EDG will be able to operate continuously for a period of well over 7 days.

Per Safety Evaluation in Amendment 180, TS SR 4.8.1.1.2e verifies that each fuel oil transfer pump transfers fuel to its associated diesel oil feed tank by taking suction from the opposite train FOST via the installed cross connect. This test is performed by aligning the "A" fuel oil transfer pump suction to the "B" FOST, or the "B" fuel oil transfer pump suction to the "A" FOST. Only one train is tested at a time, and that train is considered inoperable during the test. The train that is being tested is considered inoperable. The test alignment requires the normal fuel transfer suction valve to be closed and two cross-connect valves to the opposite train to be opened. When an increase in volume is observed in the associated train's diesel oil feed tank, the fuel oil transfer pump is secured and valves realigned.
←(LBDCR 13-017, Ch. 80)

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

→(EC-10752, Ch. 56)
LCO 3.8.1.3

ACTION a
→(EC-15945, Ch. 61)
This ACTION ensures that each diesel generator fuel oil storage tank (FOST) contains fuel oil of a sufficient volume to operate each diesel generator for a period of 7 days. An administrative limit of greater than 40,033 gallons assures at least 39,300 usable gallons are stored in the tank accounting for volumetric shrink and instrumentation uncertainty. 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
←(EC-10725, Ch. 56; EC-15945, Ch. 61)

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

→(EC-10752, Ch. 56)

SR 4.8.1.3.1

→(LBDCR 16-046, Ch. 86)

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each EDG's operation for 7 days at full load. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

SR 4.8.1.3.2

SR 4.8.1.3.2 provides a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from the tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. The tests are to be conducted prior to adding the new fuel to the storage tanks, but in no case is the time between receipt of the new fuel and conducting the tests to exceed 31 days. The tests, limits and applicable ASTM Standards are as follows:

→(EC-15945, Ch. 61)

a. Sample the new fuel oil in accordance with ASTM D4057-06.

←(EC-15945, Ch. 61)

b. Verify in accordance with the tests specified in ASTM D975-7b that the sample has a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point ≥125°F,

c. Verify in accordance with ASTM D1298 or ASTM D4052 that the sample has an absolute specific gravity of 60/60°F of ≥0.85 and ≤0.885 or an API gravity at 60°F of ≥28.4° and ≤35° and

d. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-04 or water and sediment content within limits when tested in accordance with ASTM D2709-96.

→(EC-15945, Ch. 61)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO since the fuel oil is not added to the storage tanks.

←(EC-15945, Ch. 61)

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-7b are met for Grade 2-D

←(EC-10725, Ch. 56)

A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

→(EC 47119, Ch 79)

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are consistent with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," Revision 4, March 2007, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Other provisions are derived from Generic Letter 93-05 "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation" 94-01 "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," and NUREG 1432 Standard Technical Specifications Combustion Engineering Plants.

←(EC 47119, Ch 79)

The minimum voltage and frequency stated in the Surveillance Requirement are those necessary to ensure the diesel generator can accept the Design Basis Accident loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential to establishing diesel generator OPERABILITY, but a time constraint is not imposed. This is because a typical diesel generator will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the Surveillance Requirement. In lieu of a time constraint in the Surveillance Requirement, the actual time to reach steady state operation is monitored and trended. This is to ensure there is no voltage regulator or governor degradation which could cause a diesel generator to become inoperable. The 10 seconds in the Surveillance Requirement is met when the diesel generator first reaches the specified voltage and frequency, at which time the output breaker would close if an automatic actuation had occurred.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(DRN 02-0607)

←(DRN 02-0607)

The maximum voltage limit in Surveillance test 4.8.1.1.2.e.2 was increased to 5023 volts in response to NRC Information Notice 91-13; Inadequate Testing of Emergency Diesel Generators. A maximum voltage limit is provided to ensure that components electrically connected to the diesel generator are not damaged as a result of the momentary voltage excursion experienced during this test.

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests

A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

The Onsite Power System includes three 4.16 kV ESF buses (3A3-S, 3B3-S, and 3AB3-2). Power for safety related loads is normally supplied by the non-safety related 4.16 kV buses (3A2 and 3B2) of the Offsite Power System. Should offsite power from either of these be lost, the Onsite Power System will receive power automatically from the appropriate diesel generator. Non-safety related loads will be automatically disconnected from the safety Onsite Power System. Each ESF bus (3A3-S or 3B3-S) is redundant to the other; each can supply sufficient power to its safety related loads to enable safe shutdown, or to mitigate the consequences of a design basis accident. The third bus, 3AB3-S, may be connected to either 3A3-S or 3B3-S, but never to both. Therefore 3AB3-S is not considered as a third, separate source of ESF power. The three ESF buses and their loads are tested as specified in Surveillance Requirements 4.8.1.1.2.e.3 and 4.8.1.1.2.e.5.

→(LBDCR 16-046, Ch. 86)

Surveillance requirement 4.8.1.1.2.e.1 requires the verification in accordance with the Surveillance Frequency Control Program of the diesel generator's ability to reject a load of greater than or equal to 498 Kw while specific voltage and frequency constraints are maintained. The intent of this Surveillance requirement is to require the diesel generator to reject the largest single load. The largest single load on the diesel generator is the Essential Chiller which requires 430 Kw under tornado/missile conditions. The difference between the specified 498 Kw load in the Surveillance requirement and the 430 Kw required by the actual largest single load is a margin of conservatism. A method of rejecting a load greater than or equal to 498 Kw utilizing the wet and dry cooling tower fans has been developed and will satisfy the Surveillance requirement.

←(LBDCR 16-046, Ch. 86)

The loading range for the diesel generators (4000-4400 Kw) as specified in surveillance requirements is equal to approximately 90 to 100 percent of its continuous rating. This provides for a range to conduct testing without inadvertently overloading of the diesel generators. Inadvertent overloading creates unnecessary wear and mechanical stress that may adversely affect the reliability and longevity of the diesel generators.

A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

→(LBDCR 16-046, Ch. 86)
The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.
←(LBDCR 16-046, Ch. 86)

The OPERABILITY of the motor-operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide l.l06, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

"Containment Penetration Conductor Overcurrent Protection Devices" and "Motor-Operated Valves Thermal Overload Protection and/or Bypass Devices", previously Tables 3.8-1 and 3.8-2, have been incorporated into Technical Requirements Manual (TRM).

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The K_{eff} value specified in the COLR includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value specified in the COLR also includes a conservative uncertainty allowance of 50 ppm boron.

>(DRN 03-375, Ch. 19)

If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action. Suspension of CORE ALTERATIONS or positive reactivity additions shall not preclude moving a component to a safe position.

<(DRN 03-375, Ch. 19)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

>(EC-38571, Ch. 71)

The fuel handling accident (UFSAR Section 15.7.3.4) analysis assumes protection against load movements with or over irradiated fuel assemblies that could cause fuel assembly damage. Examples of load movements include movement of new fuel assemblies, irradiated fuel assemblies, and the dummy fuel assembly. The load movements do not include the movement over assemblies in a transfer cask using a single-failure-proof handling system.

3/4.9 REFUELING OPERATIONS

3/4.9.3 DECAY TIME (continued)

The load movements do not include the movement of the spent fuel machine or refuel machine without loads attached. It also does not include load movements in containment when the reactor vesselhead or Upper Guide Structure is still installed. Load movements also exclude suspended loads weighing less than 1000 lbm (e.g. Westinghouse analysis CN-NFPE-09-57 describes no fuel failure for loads weighing less than 1000 lbm based upon the 2000 lbm analysis for drops distributed over two assemblies).
<(EC-38571, Ch. 71)

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

→(DRN 03-178, Ch. 21; EC-28875, Ch. 69)

instrumentation channels (Note that Technical Specifications 3/4.3.3, Radiation Monitoring is also applicable). The containment purge lines are automatically closed upon a containment purge isolation signal (CPIS) if the fuel handling accident releases activity above prescribed levels. Closure of at least one of the containment purge isolation valves is sufficient to provide closure of the penetration.

Administrative controls shall ensure that appropriate personnel are aware that when the equipment door, both personnel airlock doors, and/or containment penetrations are open, a specific individual(s) is designated and available to close the equipment door, an airlock door and the penetrations as part of a required evacuation of containment, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and the equipment door be capable of being quickly removed.

←(DRN 03-178, Ch. 21; EC-28875, Ch. 69)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(LBDCR 13-003, Ch. 74)

3/4.9.5 DELETED

←(LBDCR 13-003, Ch. 74)

→(LBDCR 15-024, Ch. 81)

3/4.9.6 DELETED

→(EC-17724, Ch. 62)

←(EC-17724, Ch. 62, LBDCR 15-024, Ch. 81)

→(LBDCR 15-024, Ch. 81)

3/4.9.7 DELETED

→(EC-32267, Ch. 70; EC-38571, Ch. 71)

←(EC-32267, Ch. 70; EC-38571, Ch. 71, LBDCR 15-024, Ch. 81)

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

>(DRN 03-375, Ch. 19)

The requirement that at least one shutdown cooling train be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum 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 which would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operations.

<(DRN 03-375, Ch. 19)

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the top of the fuel seated in the reactor pressure vessel ensures that a single failure of the operating shutdown cooling train will not result in a complete loss of decay heat removal capability. When there is no irradiated fuel in the reactor pressure vessel, this is not a consideration and only one shutdown cooling train is required to be OPERABLE. With the reactor vessel head removed and 23 feet of water above the top of the fuel seated in the reactor pressure vessel, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

>(DRN 03-233, Ch. 22; EC-28875, Am. 69)

<(DRN 03-233, Ch. 22; EC-28875, Am. 69)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

>(EC-18742, Ch. 65)

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL STORAGE

TS 5.6, "FUEL STORAGE," reflects the results of the criticality analysis, crediting soluble boron and allowing more flexibility in storing the more reactive Next Generation Fuel (NGF) assemblies in the spent fuel storage racks. The Waterford 3 SFP criticality analysis used a

<(EC-18742, Ch. 65)

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL STORAGE (Continued)

design acceptance criteria of effective (neutron) multiplication factor (k_{eff}) no greater than 0.995, if flooded with unborated water, and k_{eff} no greater than 0.945, if flooded with borated water. This provides an additional 0.005 Δk_{eff} analytical margin to the regulatory requirement. This approach provides sufficient margin to offset minor non-conservatisms to provide reasonable assurance that the regulatory requirements are met. Each storage configuration has a geometric arrangement which must be maintained so that the SFP criticality analysis remains valid.

→(LBDCR 16-046, Ch. 86)

The spent fuel pool (SFP) criticality analysis credits 524 parts per million (ppm) of soluble boron to maintain k_{eff} less than 0.95 in the SFP during normal conditions, and 870 ppm under the worst-case accident conditions. The analysis determined that a misloading event in the spent fuel checkerboard loading pattern would have the largest reactivity increase, requiring 870 ppm of soluble boron to meet the regulation. The boron dilution analysis identified a number of assorted sources for slow addition of unborated water to the SFP that could possibly continue undetected for an extended period of time. The maximum flow from any of these sources was determined to be 2 gpm, and dilution of the SFP from 1900 ppm to 870 ppm soluble boron would take approximately 72 days. Slow dilution by undetected sources is adequately addressed by sampling the SFP in accordance with SR 4.9.12. Higher flow-rate dilution scenarios would be identified through various alarms and building walkdowns, and could be addressed by sampling the SFP in accordance with SR 4.9.12. Higher flow-rate dilution scenarios would be identified through various alarms and building walkdowns, and could be addressed within a sufficient time to preclude dilution of the SFP to 870 ppm soluble room. Adequate safety is maintained in the case of a high flow-rate dilution of the SFP in accordance with 10 CFR 50.68(b)(4) because k_{eff} must remain below 1.0 (subcritical), even if the SFP were flooded with unborated water.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program

←(LBDCR 16-046, Ch. 86)

Three qualified storage configurations are allowed for Region 2 Fuel Storage locations, based on burnup versus enrichment restrictions: 1) uniform loading of assemblies, 2) checkerboard loading of high and low reactivity assemblies, and 3) checkerboard loading of fresh assemblies and empty cells. The storage configurations may be interspersed with each other throughout the SFP, provided that the geometric interface requirements are met. Checkerboard loading is not required for Region I Fuel Storage locations.

<(EC-18742, Ch. 65)

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.10.2 MTC, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth,(2) determine core characteristics and (3) calibrate the reactor protection system.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 NATURAL CIRCULATION TESTING

This special test exception permits all reactor coolant pumps to be secured during natural circulation testing and operator training for periods in excess of the 1 hour allowed by Specification 3.4.1.2.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

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)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

The NRC evaluation section in Safety Evaluation of Amendment No. 157, for the EDG FOST not having 10% margin in fuel oil inventory, credited acceptability of the design based upon Waterford 3 having EDG Fuel Oil Storage and Transfer Systems cross connecting capabilities. With the ability to cross-tie the two EDG Fuel Oil Storage and Transfer Systems, one EDG will be able to operate continuously for a period of well over 7 days.

Per Safety Evaluation in Amendment 180, TS SR 4.8.1.1.2e verifies that each fuel oil transfer pump transfers fuel to its associated diesel oil feed tank by taking suction from the opposite train FOST via the installed cross connect. This test is performed by aligning the "A" fuel oil transfer pump suction to the "B" FOST, or the "B" fuel oil transfer pump suction to the "A" FOST. Only one train is tested at a time, and that train is considered inoperable during the test. The train that is being tested is considered inoperable. The test alignment requires the normal fuel transfer suction valve to be closed and two cross-connect valves to the opposite train to be opened. When an increase in volume is observed in the associated train's diesel oil feed tank, the fuel oil transfer pump is secured and valves realigned.
←(LBDCR 13-017, Ch. 80)

→((LBDCR 16-046, Ch. 86)
The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.
← (LBDCR 16-046, Ch. 86)

→(EC-10752, Ch. 56)
LCO 3.8.1.3

ACTION a
→(EC-15945, Ch. 61) (LBDCR 16-018, Ch. 87)

This ACTION ensures that each diesel generator fuel oil storage tank (FOST) contains fuel oil of a sufficient volume to operate each diesel generator for a period of 7 days. An administrative limit is used to assure at least 39,300 usable gallons are stored in the tank when accounting for volumetric shrink, instrumentation uncertainty, and other effects that impact usable fuel volume. 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
←(EC-10725, Ch. 56; EC-15945, Ch. 61) (LBDCR 16-018, Ch. 87)

PLANT SYSTEMS

BASES

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.

→(EC-38632, Ch. 72; LBDCR 17-025, Ch. 88)

Each DCT consists of five separate cells. Cooling air for each cell is provided by 3 fans, for a total of 15 per DCT. Dry cooling tower fan operability is maintained by operating in fast or auto mode. 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. Both trains of DCT fans (those within the missile protected area) are required to be operable when complying with ACTION c. This is to ensure that one train of DCT remains available following the postulated loss of offsite power and one EDG failure as assumed in the design basis.

←(EC-38632, Ch. 72; LBDCR 17-025, Ch. 88)

→(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.

←(DRN 04-1243, Ch. 38)

Table 3.7-3 specifies increased or decreased fan OPERABILITY requirements based on outside air temperature. 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. The calculated temperature values (EC-M95-009) associated

→(DRN 05-747, Ch. 40)

INDEX

ADMINISTRATIVE CONTROLS

3/4.6.5 VACUUM RELIEF VALVES (Continued)

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

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

→ (DRN 03-1807, Ch. 30, LBDCR 17-032, Ch. 89)

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

← (DRN 03-1807, Ch. 30; LBDCR 17-032, Ch. 89)

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM (Continued)

Surveillance Requirements (Continued)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(DRN 03-1807, Ch. 30, LBDCR 17-032, Ch. 89)

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, LBDCR 17-032, Ch. 89)

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, LBDCR 16-046, Ch. 86)

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(DRN 05-42, Ch. 37, LBDCR 16-046, Ch. 86)

This SR is modified to indicate that 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 cannot be applied to the 24 hour period.

→ (DRN 03-1737, Ch. 31)

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

→ (DRN 04-1243, Ch. 38; LBDCR 17-032, Ch. 89)

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.

←(LBDCR 17-032, Ch. 89)

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)

→(LBDCR 17-032, Ch. 89)

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM.

←(LBDCR 17-032, Ch. 89)

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

→(LBDCR 16-046, Ch. 86)

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 in accordance with the Surveillance Frequency Control Program. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

← (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, 05-1650)

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. The design basis correlates a static test utilizing one accumulator to demonstrate the ability of the MFIVs to close in less than or equal to 6 seconds under design basis accident conditions with two accumulators. The static stroke time test that utilizes one accumulator is allowed to exceed the 6 second Surveillance Requirement since both accumulators are credited in the design basis Accidents in order to isolate within the 6 second Surveillance Requirement. The 6 second required closure time includes a 1 second allowance for instrument response time.
←(DRN 05-1650)
→(LBDCR 16-046, Ch. 86; LBDCR 17-032, Ch. 89)

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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.
←(LBDCR 16-046, Ch. 86, LBDCR 17-032, Ch. 89)
→ (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 Isolation Valves to Technical Specifications and Request for NRC Staff Review of an Unreviewed Safety Question.")

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

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

In MODES 1, 2, 3, and 4, the MFIVs are required to be OPERABLE. Because the MFIVs are required to be OPERABLE in MODES 1, 2, 3, and 4, RTO must be able to decrease SGFP
←(DRN 02-1684, Ch. 15

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

→ (DRN 02-1684, Ch. 15)

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

RTO and AFW Pump HDPT Test Requirements

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

→ (DRN 03-1807, Ch. 30; LBDCR 17-032, Ch. 89)

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

← (DRN 03-1807, Ch. 30, LBDCR 17-032, Ch. 89)

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

← (DRN 02-1684, Ch. 15)

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)
→(DRN 02-1684, Ch. 15; 03-1807, Ch. 30; LBDCR 17-032, Ch. 89)
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; LBDCR 17-032, Ch. 89)
→(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)

→(LBDCR 16-046, Ch. 86)

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)
→ (LBDCR 17-032, Ch. 89)

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.

←(LBDCR 17-032, Ch. 89)
→(LBDCR 16-046, Ch. 86)

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

←(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.