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Waterford 3

W3F1-2013-0040

July 9, 2013

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 May 1, 2012 through June 30, 2013
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 W3F1-2012-0040 (ADAMS Accession No. ML12153A052), dated May 30, 2012. 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 acting Licensing Manager, Bryan Pellegrin, at 504.739.6203.

Sincerely,

BJP/RJP

Attachments:

1. Waterford 3 Technical Specification Index and Bases Change List

2. Waterford 3 Technical Specification Index and Bases Revised Pages

cc:

Mr. Arthur T. Howell, III 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	Marlone.Davis@nrc.gov
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Attachment 1 to

W3F1-2013-0040

Waterford 3 Technical Specification Index and Bases Change List

Waterford 3 Technical Specification (TS) Index and Bases Change List

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
71	10/4/12	B 3/4 3-2 B 3/4 3-3 B 3/4 7-4a(3) B 3/4 7-4a(4) B 3/4 7-4b B 3/4 8-1a B 3/4 8-1b B 3/4 9-1 B 3/4 9-2 B 3/4 9-3 B 3/4 9-4 B 3/4 9-5	Change No. 71 to TS Bases sections 3/4.7.6.1; 3/4.7.6.3 and 3/4.7.6.4; 3/4.8.1, 3/4.8.2, and 3/4.8.3; 3/4.9.3, 3/4.9.4, and 3/4.9.7 was implemented by Engineering Change 38571 as a result of NRC approved License Amendment 235 (TAC No. ME6049). The amendment changed multiple Technical Specifications due to revising the Fuel Handling Accident analysis. The license amendment provided new applicability and/or action language addressing load movements over irradiated fuel assemblies in containment and in the fuel storage pool.

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
72	10/4/12	B 3/4 7-4 B 3/4 7-4(1)	Change No. 72 to TS Bases section 3/4.7.4 was implemented by Engineering Change 38632 to clarify the Dry Cooling Tower (DCT) fan operability requirements and the associated TS Surveillance Requirement. The change added information indicating that DCT fan operability is maintained by operating in fast or auto mode, and that the TS Surveillance Requirement demonstrates DCT fan operability in a manner corresponding to the accident configuration, which for dry cooling tower fans is in fast speed. This clarifies that, with DCT fans in slow speed, the DCT fans are inoperable. The TS Bases change improved the licensing basis documentation, and did not change the existing design requirements or assumptions. Thus, it was determined that there were no adverse consequences, and that NRC approval was not required for the change per 10CFR50.59.

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
73	12/4/12	B 3/4 7-4(1)	Change No. 73 to TS Bases section 3/4.7.4 was implemented by Licensing Basis Document Change Request (LBDCR)12-001 as a result of License Amendment 237 (TAC No. ME7342). Amendment 237 revised TS 3/4.7.4 Table 3.7-3, "Ultimate Heat Sink Minimum Fan Requirements per Train," which specifies the minimum dry cooling tower (DCT) and wet cooling tower (WCT) fan requirements for given ambient temperature conditions. This approval modified the WCT fan requirements by requiring that all eight WCT fans be operable for the WCT to be considered operable, regardless of ambient temperature. This change was needed because the previous WCT fan requirement limits were based on an inappropriate analysis methodology. The bases document was revised to incorporate the conditions as approved by amendment 237. Also, a typographical error was corrected in the Bases by changing a word from “met” to “meet.”

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
74	2/14/13	Index pages: VI, VIII, IX, XII, and XIII B 3/4 4-5 B 3/4 7-4a B 3/4 7-7 B 3/4 9-3	Change No. 74 to the TS Index and TS Bases sections 3/4.4.6, 3/4.7.5, and 3/4.7.9 was implemented by LBDCR 13-003 as a result of NRC approved License Amendment 238 (TAC No. ME7614). Amendment 238 relocated TS 3.4.6, "Chemistry," TS 3.7.5, "Flood Protection," TS 3.7.9, "Sealed Source Contamination," and TS 3.9.5, "Communications," to the Waterford 3 Technical Requirements Manual. The TS Bases text for each of the TS's was also moved to the respective TRM bases. The affected TS Bases and Index pages provided in this submittal were revised to reflect removal of the information from the TS.

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
75	3/14/13	B 2-6	Change No. 75 to TS Bases section 2.2.1, Reactor Trip Setpoints, DNBR-Low was implemented by Engineering Change 22385. The speed of each RCP motor is measured to provide a basis for calculation of reactor coolant flow through each pump. If the speed of a RCP begins to coast down, the CPC System will calculate a reduced DBNR and will initiate a DNBR – Low trip when less than the setpoint. This is not a direct trip as discussed in the TS Bases. The CPC System auxiliary trips include a direct trip for less than two Reactor Coolant Pumps running. The EC reflects that clarification was needed to this section to more accurately describe the input that Reactor Coolant Pump Speed has on the Core Protection Calculator auxiliary (direct) trip. It was determined that this change did not affect the design function of the CPC System, and that NRC approval was not required for the TS Bases change per 10CFR50.59. Specifically, TS Bases Section 2.2.1 was revised to change the description of the CPC auxiliary trip from "The CPCs contain a direct trip on low RCP speed. This trip will occur if the RCP speed drops below 0.965." to “The CPCs contain an auxiliary trip on low RCP speed. The trip will occur if there are less than 2 RCPs running.”

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
75	3/14/13	B 2-7	Change No. 75 also included a change to TS Bases section 2.2.1, Reactor Trip Setpoints, Reactor Coolant Flow – Low, which was implemented by Engineering Change 8894. This EC corrected information relative to the protection provided by the Reactor Coolant Flow – Low trip by removing the statement "and a steam line break event with a loss-of-offsite power." The design function of the low reactor coolant flow trip setpoint is to provide protection against a reactor coolant pump sheared shaft event. The steam line break calculations do not credit the reactor coolant low flow trip setpoint for steam line breaks. Additionally, the Waterford 3 UFSAR specifies the trips available to mitigate the effects of a steam line break. These include low steam generator pressure, CPC high variable over power, low steam generator level, high linear power, low RCS pressure, high containment pressure, and an array of CPC trips. The Waterford 3 UFSAR does not credit the reactor coolant low flow reactor trip function for a steam line break. It was determined that this change did not affect the design function of the low reactor coolant flow trip setpoint, and that NRC approval was not required for the change per 10CFR50.59.

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
76	5/7/13	Index page XVI	Change No. 76 to the TS Index was implemented by LBDCR 13-005 as a result of NRC approved License Amendment 240 (TAC Nos. 7774 and 7786). The NRC approved changes to staff qualifications, which resulted in deletion of the contents in Administrative Controls TS 6.4, Training. The TS Index page is updated to reflect that TS 6.4 is not used.
76	5/7/13	B 3/4 6-4	Change No. 76 also included a change to TS Bases section 3/4.6.2.1 and 3/4.6.2.2, which was implemented by LBDCR 13-006. The change is associated with the alignment of Component Cooling Water flow to each Containment Cooler Fan during TS Surveillance testing. Specific clarification was made to reflect that the TS surveillance test configuration is with one fan running in each train. This change does not reduce or render less capable any Waterford 3 System, Structure, or Component, nor is the proposed change in conflict with other license basis requirements. It was determined that NRC approval was not required for the change per 10CFR50.59.

Attachment 2 to

W3F1-2013-0040

Waterford 3 Technical Specification Index and Bases Revised Pages

(There are thirty-four unnumbered pages following this cover page)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 71 REPLACEMENT PAGE(S)
(12 pages)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 71 and contains the appropriate EC number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 7-4a(3)	B 3/4 7-4a(3)
B 3/4 7-4a(4)	B 3/4 7-4a(4)
B 3/4 7-4b	B 3/4 7-4b
B 3/4 8-1a	B 3/4 8-1a
B 3/4 8-1b	B 3/4 8-1b
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3
B 3/4 9-4	B 3/4 9-4
B 3/4 9-5	B 3/4 9-5

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 shiftly 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)

>(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.

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.

Continuous operation with less than the Minimum Channels OPERABLE requirements is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the accident

>(EC-15550, Ch. 59)

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

>(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)

Actions

ACTION STATEMENT a addresses the condition of one CREAFS train inoperable for reasons other than an inoperable CRE boundary. Action must be taken to restore OPERABLE status within 7 days. In this condition, the remaining OPERABLE CREAFS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREAFS train could result in loss of CREAFS function. The 7 day completion time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

ACTION STATEMENTS b.1, b.2, and b.3 address the condition of an inoperable control room envelope boundary. If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body, 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour completion time is

<(EC-15550, Ch. 59)

>(EC-15550, Ch. 59)

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

reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day completion time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day completion time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

ACTION STATEMENT c requires that, in MODE 1, 2, 3, or 4, if the inoperable CREAFS or the CRE boundary cannot be restored to OPERABLE status within the required completion time, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

ACTION STATEMENT d.1 requires that, in MODE 5 or 6, or during movement of irradiated fuel assemblies, if required Action a cannot be completed within the required completion time, the OPERABLE CREAFS train must be immediately placed in the emergency radiation protection mode of operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

ACTION STATEMENT d.2 is an alternative to Action d.1 and requires immediate suspension of activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.

ACTION STATEMENT e requires that, in MODES 5 or 6, or during movement of irradiated fuel assemblies, with one or more CREAFS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

ACTION STATEMENT f addresses the condition of both CREAFS trains being inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary. The CREAFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LC0 3.0.3 must be entered immediately.

ACTION STATEMENT g requires that, in MODES 5 or 6, or during movement of irradiated fuel assemblies, with both CREAFS trains inoperable action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

<(EC-15550, Ch. 59)

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)

>(EC-15550, Ch. 59)

3/4.7.6.5 [NOT USED]

<(EC-15550, Ch. 59)

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

>(DRN 03-375, Ch. 19)

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

<(DRN 03-375, Ch. 19)

>(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)

>(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-10725, Ch. 56; EC-15945, Ch. 61)

Guide 1.137 October 1979. The minimum onsite stored fuel oil is sufficient to operate the diesel generator for a period longer than the time to replenish the onsite supply from the outside sources discussed in FSAR 9.5.4.2.

An additional provision is included in the ACTION which allows the diesel generators to remain operable when their 7 day fuel oil supply is not available provided that at least a 6 day supply of fuel oil is available. This provision is acceptable on the basis that replacement fuel oil is onsite within the first 48 hours after falling below the 7 day supply. An administrative limit of greater than 37,696 gallons assures at least 37,000 usable gallons are stored in the tank,

<(EC-10725, Ch. 56; EC-15945, Ch. 61)

LCO 3.8.1.3 (Continued)

ACTION a (Continued)

accounting for volumetric shrink and instrumentation uncertainty. This useable volume is sufficient to operate the diesel generator for 5 days based on the full continuous load (4400kW) of the diesel generator and is sufficient to operable the diesel generator for greater than 6 days based on the time dependent loads of the diesel generator following a loss of offsite power and a design basis accident.

ACTION b

>(EC-15945, Ch. 61)

This ACTION is entered as a result of a failure to meet the acceptance criterion of particulate limits. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between surveillance frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7-day Completion Time allows for further evaluation, re-sampling, and re-analysis of the DG fuel oil.

<(EC-15945, Ch. 61)

ACTION c

With the new fuel oil properties defined in the Bases for SR 4.8.1.1.2.c not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a diesel generator start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the diesel generator would still be capable of performing its intended function.

ACTION d

>(EC-15945, Ch. 61)

This ACTION is entered as a result of the failure to meet any of the other ACTIONS.

<(EC-10725, Ch. 56; EC-15945, Ch. 61)

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)

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.

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. 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)

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

>(DRN 03-178, Ch. 21)

In MODES 1, 2, 3 and 4 the escape of radioactivity to the environment is minimized by maintaining containment OPERABLE as described in LCO 3.6.1, “Primary Containment”.

In Mode 6 (REFUELING), the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as “containment closure” rather than “containment OPERABILITY.” Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

>(EC-38571, Ch.71)

The action to suspend all operations involving load movements with or over irradiated fuel assemblies shall not preclude completion of movement to a safe conservative position. During CORE ALTERATIONS or load movements with or over irradiated fuel within the containment, the escape of radioactivity to the environment is minimized when the LCO requirements are met. 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)

>(EC-28875, Ch. 69)

Containment penetrations, the personnel airlock doors, and/or the equipment door may be open under administrative control during CORE ALTERATIONS or movement of irradiated fuel in the containment provided a minimum of one closure method (manual or automatic valve, blind flange, or equivalent) in each penetration, one door in each airlock, and the equipment door are capable of being closed in the event of a fuel handling accident. For closure, the equipment door will be held in place by a minimum of four symmetrically-placed bolts. Containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being isolated on at least one side. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during CORE ALTERATIONS or movement of irradiated fuel.

The containment purge and exhaust isolation system must also be OPERABLE during CORE ALTERATIONS or movement of irradiated fuel when open to the outside atmosphere or must be closed under LCO 3.9.4.c.1. An OPERABLE containment purge and exhaust isolation system consists of a containment purge valve capable of isolating on an actual or simulated actuation signal from containment purge isolation from each of the required radiation monitoring

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

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)

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE
>(EC-17724, Ch. 62)
The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of CEAs and fuel assemblies, (2) each hoist has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. The Technical Specification Actions ‘a.’ and ‘b.’ statements allow the movement of a fuel assembly or CEA to safe condition using administrative controls in the event of a refueling machine failure.
<(EC-17724, Ch. 62)

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING
>(EC-32267, Ch. 70; 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). Movements of loads using a single failure proof handling system, consisting of a crane that has been upgraded to meeting the single-failure-proof criteria of NUREG 0554 and NUREG 0612, and lifting devices that meet the requirements of ANSI N14.6 or ASME B30.9, do not require the assumption of a dropped load, and activity releases assumed in the safety analysis are not affected.
<(EC-32267, Ch. 70)

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.

>(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)

>(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.

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 on the 7-day frequency of 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 on the 7-day frequency of 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.

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)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 72 REPLACEMENT PAGE(S)
(2 pages)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 72 and contains the appropriate EC number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 7-4	B 3/4 7-4
B 3/4 7-4(1)	B 3/4 7-4(1)

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)

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.

<(EC-38632, Ch. 72)

>(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

3/4.7.4 ULTIMATE HEAT SINK (Continued)

with fan requirements have been rounded in the conservative direction and lowered at least one full degree to account for minor inaccuracies. Failure to met the OPERABILITY requirements of Table 3.7-3 requires entry into the applicable action. Because temperature 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.

>(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)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 73 REPLACEMENT PAGE(S)
(1 page)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 73 and contains the appropriate EC/LBDCR number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 7-4(1)	B 3/4 7-4(1)

3/4.7.4 ULTIMATE HEAT SINK (Continued)

>(LBDR 12-0001, Ch. 73)

with fan requirements have been rounded in the conservative direction and lowered at least one full degree. 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.

<(LBDR 12-0001, 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)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 74 REPLACEMENT PAGE(S)
(9 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 74 and contain the appropriate EC/LBDCR number and a vertical line indicating the areas of change.

Remove	**Insert**
VI	VI
VIII	VIII
IX	IX
XII	XII
XIII	XIII
B 3/4 4-5	B 3/4 4-5
B 3/4 7-4a	B 3/4 7-4a
B 3/4 7-7	B 3/4 7-7
B 3/4 9-3	B 3/4 9-3

>(DRN 05-747, Ch. 40)

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<(DRN 05-747, Ch. 40)

>(DRN 05-747, Ch. 40)

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<(DRN 05-747, Ch. 40)

>(DRN 05-747, Ch. 40)

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<(DRN 05-747, Ch. 40)

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BASES

SECTION | PAGE

3/4.4 REACTOR COOLANT SYSTEM (Continued)

<(DRN 05-747, Ch. 40)

>(DRN 05-747, Ch. 40)

INDEX

BASES

<(DRN 05-747, Ch. 40)

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 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 13-003, Ch. 74)

3/4.7.5 DELETED

<(LBDCR 13-003, Ch. 74)

>(DRN 03-656, Ch. 24)

3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM

<(DRN 03-656, Ch. 24)

>(EC-15550, Ch. 59)

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

Background

The CREAFS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The CREAFS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREAFS train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines, and provides 100% back-up in case of failure of the main HEPA filter bank.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREAFS is an emergency standby system. Upon receipt of the actuating signal(s), the emergency filtration units start and filter a portion of the recirculated supply air to the control room. Operators can take manual actions to align the north or south outside air paths to pressurize the CRE. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. The heater is important to the effectiveness of the charcoal adsorbers.

<(EC-15550, Ch. 59)

>(LBDCR 13-003, Ch. 74)

3/4.7.9 DELETED

<(LBDCR 13-003, Ch. 74)

3/4.7.10 This section deleted.

3/4.7.11 This section deleted.

3/4.7.12 ESSENTIAL SERVICES CHILLED WATER SYSTEM

The Essential Services Chilled Water (CHW) System provides a heat sink for the removal of process and operating heat from selected safety related air handling systems during normal operation and Design Basis Accidents (DBAs). These air handling systems cool spaces containing equipment required for safety related operations. The CHW System is a closed loop system consisting of three 100 percent capacity subsystems, each consisting of one chiller; one chilled water pump; one chilled water expansion tank; instrumentation and controls; and piping and valves. Two subsystems are required to be OPERABLE to provide redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single failure.

The design basis of the CHW System is to remove the post accident heat load from ESF spaces following a DBA coincident with a loss of offsite power. During a DBA, each train is required to provide chilled water to the air handling systems at the design temperature of $\leq$ 42°F and flow rate of $\geq$500 gpm.

During normal operations, the CHW System may be unloaded (low heat within the cooling space, typically found during the winter months) in which air handling unit cooling coil heat loads are at a minimum. Therefore, during normal operation, it is acceptable for the CHW

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 13-003, Ch. 74)

3/4.9.5 DELETED |

<(LBDCR 13-003, Ch. 74)

3/4.9.6 REFUELING MACHINE
>(EC-17724, Ch. 62)
The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of CEAs and fuel assemblies, (2) each hoist has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. The Technical Specification Actions ‘a.’ and ‘b.’ statements allow the movement of a fuel assembly or CEA to safe condition using administrative controls in the event of a refueling machine failure.
<(EC-17724, Ch. 62)

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING
>(EC-32267, Ch. 70; 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). Movements of loads using a single failure proof handling system, consisting of a crane that has been upgraded to meeting the single-failure-proof criteria of NUREG 0554 and NUREG 0612, and lifting devices that meet the requirements of ANSI N14.6 or ASME B30.9, do not require the assumption of a dropped load, and activity releases assumed in the safety analysis are not affected.
<(EC-32267, Ch. 70; EC-38571, Ch. 71)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 75 REPLACEMENT PAGE(S)
(2 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 75 and contain the appropriate EC/LBDCR number and a vertical line indicating the areas of change.

Remove	**Insert**
B 2-6	B 2-6
B 2-7	B 2-7

DNBR - Low (Continued)
>(EC-18510, Ch. 64)
in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit of 1.24. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.
<(EC-18510, Ch. 64)

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

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

>(DRN 04-1243, Ch. 38)
The CPCs contain several auxiliary trip functions which are credited in the safety analysis. These trips manifest themselves as DNBR trips however they are making the trip determination on parameters other than DNBR.

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

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

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

>(EC-22385, Ch. 75)
The CPCs contain an auxiliary trip on low RCP speed. The trip will occur if there are less than 2 RCPs running.
<(DRN 04-1243, Ch. 38; EC-22385, Ch. 75)

>(EC-22790, Ch. 66)

<(EC-22790, Ch. 66)

Reactor Coolant Flow - Low

>(DRN 03-6, Ch. 20; EC-8894, Ch. 75)

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

<(DRN 03-6, Ch. 20; EC-8894, Ch. 75)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 76 REPLACEMENT PAGE(S)
(2 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 76 and contain the appropriate EC/LBDCR number and a vertical line indicating the areas of change.

Remove	**Insert**
XVI	XVI
B 3/4 6-4	B 3/4 6-4

>(DRN 05-747, Ch. 40)

INDEX

ADMINISTRATIVE CONTROLS

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 $\leq$ 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)

Verifying the 625 gpm to each cooler once per 31 days 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)

The 18 month Surveillance Requirement verifies that each containment cooling fan actuates upon receipt of an actual or simulated SIAS actuation signal. The 18 month frequency is based on engineering judgment and has been shown to be acceptable through operating experience.

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. Also considered in