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Waterford 3

W3F1-2001-0047
A4.05
PR

June 25, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Initial Technical Specification Bases Update to the NRC
For the Period July 7, 2000 through June 11, 2001

Gentlemen:

Pursuant to Waterford Steam Electric Station Unit 3 Technical Specification 6.16, "Technical Specifications Bases Control Program," Entergy Operations, Inc. (EOI) hereby submits the initial update of all changes to the Technical Specification Bases approved under the Technical Specifications (TS) Bases Control Program. The TS Bases Control Program was implemented on July 7, 2000 following NRC issuance of Amendment 161 on May 9, 2000. This submittal includes the TS Bases changes implemented by the TS Bases Control Program and for continuity purposes, a list of TS Amendments issued with associated TS Bases changes. These TS Amendments were submitted to the NRC for approval prior to TS Bases Control Program implementation.

This initial TS Bases update is consistent with the update frequency listed in 10 CFR 50.71(e); however, EOI intends to provide these updates on a more frequent basis.

A001

June 25, 2001

This letter does not contain any commitments. Should you have any questions or comments concerning this submittal, please contact Ron Williams at (504) 739-6255.

Very truly yours,

Alan J Harris

A.J. Harris
Director
Nuclear Safety Assurance

AJH/RLW/cbh

Attachment Waterford 3 Technical Specification Bases Revised Pages For the Period July 7, 2000 through June 11, 2001

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office

ATTACHMENT1
TO W3F1-2001-0047

Waterford 3 Technical Specification Bases Revised Pages
For the Period July 7, 2000 through June 11, 2001

Table of Changes
(2 pages)

Revised Bases Pages and Replacement Instructions
(21 pages)

**Waterford 3 Technical Specification Bases Revised Pages
For the Period July 7, 2000 through June 11, 2001**

TS Amendment* / TS Bases Change No.	Implement Date	Affected TS Bases Pages	Topic of Change
Amendment 164	7/24/00	B 3/4 5-1b, B 3/4 5-1c[new], B 3/4 5-1d[new], B 3/4 5-2	Revise TS 3.5.2 to extend the allowable outage time (AOT) from 72 hours to seven days for one inoperable LPSI train and impose 72 hour AOTs for other conditions.
Amendment 163	8/10/00	B 3/4 6-3, B 3/4 6-4, B 3/4 6-4a[delete], B 3/4 6-5, B 3/4 6-6[new], B 3/4 6-7[new]	Revise TS 3.6.2.1 to extend the allowable outage time from 72 hours to seven days for one inoperable containment spray system (CSS) train and add a new ACTION statement to provide a shutdown requirement for the inoperability of two CSS trains.
Bases Change No. 1	8/28/00	B 3/4 0-4a	Revise Bases section 4.0.2 to incorporate NUREG-1432 discussions that clarify acceptability of the 25% surveillance interval extension for Surv. Reqm'ts located in the TS ACTION statements.
Amendment 165	8/31/00	B 3/4 6-4, B 3/4 6-5	Revise TS 3.6.2.2 to allow operation with two independent trains of containment cooling, consisting of one cooler per operable train during Modes 1, 2, 3, and 4.
Amendment 166	9/19/00	B 3/4 8-1, B 3/4 8-1a[new]	Revise TS 3.8.1.1 and 4.8.1.1 to extend the Emergency Diesel Generator (EDG) allowed outage time from 72 hours to ten days only when an alternate source of A.C. power for the onsite power system is available.
Amendment 169	10/15/00	B 3/4 9-1	Revise TS 3.9.4 and 4.9.4 to allow the containment equipment hatch, airlocks, and other penetrations to remain open, but capable of being closed during core alterations or movement of irradiated fuel in containment.
Bases Change No. 2	10/18/00	B 3/4 6-2	Clarify the Bases and remove existing inaccurate values for calculated peak containment pressure and containment pressure measurement uncertainty.
Amendment 168	10/27/00	USQ - No changes to TS or Bases pages	USQ Tornado Missile Design Basis
Amendment 167	11/3/00	B 3/4 7-3, B 3/4 7-3a, B 3/4 7-3b	Add new TS 3.7.1.6 and 4.7.1.6 for the Main Feedwater Isolation Valves (MFIV) section modeled after the guidelines of TS 3.7.3 in NUREG-1432, "Standard Technical Specifications-Combustion Engineering Plants."
Bases Change No. 3	1/2/01	B 3/4 7-3, B 3/4 7-3a, B 3/4 7-3b	Revise Bases section 3/4.7.1.5 to clarify that the 4-sec surv. time requirement in the TS pertains to the static closure time limit that will ensure the MSIV will close within the required 7 sec. safety analysis value under max. postulated DBA differential pressure.

*

Bases Change No. 4	1/19/01	B 2-6, B 3/4 3-1c, B 3/4 3-1d[delete], B 3/4 7-2, B 3/4 7-2a[new], B 3/4 7-4, **B 3/4 7-4a, B 3/4 7-4b, B 3/4 7-4c**, B 3/4 7-7, B 3/4 7-8 **Note: Bolded Bases pages indicated above have been superceded by the following Amendment 170.**	Per discussion with NRC Project Manager, withdrew request for NRC issuance of TS Bases replacement pages previously evaluated under 10 CFR 50.59, as stated in letters W3F1-99-0093 dated 5/20/99 and W3F1-2000-0023 dated 2/28/00, and administratively added Bases pages via Bases Control Program. (1) Revise Bases pages B 3/4 7-7 and B 3/4 7-8 to clarify the operability requirements for the Essential Services Chilled Water (CHW) System; (2) Revise Bases pages B 3/4 3-1 and B 3/4 3-1a to clarify the operability of reactor trip switchgear with regard to the open or racked out position to allow on-line maintenance (pages not included due to being superceded by Amendment 143, 153 and 154); (3) Revise Bases Section 3/4.3.1 and 3/4.3.2 to correct a pagination editorial oversight with the NRC approval of Amendment 154; (4) Revise Bases Section 2.2.1, Reactor Trip Setpoints, DNBR – Low, item h, Integrated Radial Peaking Factor – High value to reflect the actual limiting value installed in COLSS, CEFAST, and CPC Constants for Cycle 10; and (5) Revise Bases Section 3/4.7.1.3, to show compliance with BTP RSB 5-1; and Section 3/4.7.4 to clarify use of WCT basin water.
Amendment 170	2/22/01	B 3/4 6-7, B 3/4 6-8[new], **B 3/4 7-4a, B 3/4 7-4b, B 3/4 7-4c**, B 3/4 7-4d[new], B 3/4 7-5, B 3/4 9-3	Incorporates the use of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," into TS 4.6.6.1, 4.7.6.1, 4.7.7, 4.9.12, and the associated TS Bases.
Bases Change No. 5	2/22/01	B 3/4 7-4(1)[new]	Maintain TS Bases fidelity by correcting an editorial error made in Bases Changes No.4 to support implementation of TS Amendment 170. Amendment 170 inserts a revised page B 3/4 7-4a that does not contain text overflow from page B 3/4 7-4 implemented in Change No.4.

* TS Bases changes approved under the Bases Control Program and TS Amendments issued with associated TS Bases changes are listed in the table above to provide the sequence of Bases page changes. The Bases pages issued with the TS Amendments will not be included in the attachment to this submittal. Note,

TS BASES CHANGE NO. 1 REPLACEMENT PAGE
(1 page)

Replace the following page of the Waterford 3 Technical Specifications Bases with the attached page. The revised page is identified by Change NO. 1 and contains a vertical line indicating the area of change.

Remove Page	**Insert Page**
B 3/4 0-4a	B 3/4 0-4a

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are withinspecified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

This extension allowed by Specification 4.0.2 is also applicable to Surveillance Requirements required in Technical specification Actions. However, the extension does not apply to the initial performance. The extension only applies to each performance after the initial performance. The initial performance required by the Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single completion time. One reason for not allowing the extension to this completion time is that such an action usually verifies that no loss of function has occurred or accomplishes the function of the inoperable equipment in an alternative manner.

TS BASES CHANGE NO. 2 REPLACEMENT PAGE

(1 page)

Replace the following page of the Waterford 3 Technical Specifications Bases with the attached page. The revised page is identified by Change NO. 2 and contains a vertical line indicating the area of change.

Remove Page	Insert Page
B 3/4 6-2	B 3/4 6-2

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 14.375 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.

3/4.6.1.5 AIR TEMPERATURE

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.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

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

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

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

TS BASES CHANGE NO. 3 REPLACEMENT PAGES

(3 pages)

Replace the following pages of the Waterford 3 Technical Specifications Bases with the attached pages. The revised page B 3/4 7-3 is identified by Change No. 3 and contains a vertical line indicating the area of change. The remaining two pages B 3/4 7-3a and B 3/4 7-3b were changed due to the overflow of text from the previous page.

Remove Page	**Insert Page**
B 3/4 7-3	B 3/4 7-3
B 3/4 7-3a	B 3/4 7-3a
B 3/4 7-3b	B 3/4 7-3b

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm 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.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

The OPERABILITY of the main steam line isolation valves 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 OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses. The Surveillance Requirement to verify isolation in less than or equal to 4 seconds is based on static testing. The static test using 4 seconds demonstrates the ability of the MSIVs to close in less than or equal to the 7 second required closure time under design basis accident conditions.

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.

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. The piping volume from the valve to the steam generator must be accounted for in calculating mass and energy releases, and refilled prior to EFW reaching the steam generator following either a MSLB or FWLB.

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

The MFIVs close on receipt of a Main Steam Isolation Signal (MSIS) generated by either low steam generator pressure or high containment pressure. The MFIVs may also be actuated manually from the control room. The MSIS also actuates the Main Steam Isolation Valves (MSIVs), Main Feedwater Regulating Valves (MFRVs) and Startup Feedwater Regulating Valves (SFRVs) to close. The Feedwater Regulating Valve Bypass Valves are normally closed and deactivated during power operation.

In MODES 1, 2, 3, and 4, the MFIVs are required to be OPERABLE, except when they are closed and deactivated or isolated by either a closed manual valve or closed and deactivated automatic valve, in order to limit the amount of available fluid that could be added to the Steam Generator and/or containment in the case of a secondary system pipe break inside containment. When a MFIV is closed and deactivated or isolated by a closed manual valve or closed and deactivated automatic valve, it is already performing its safety function.

In MODES 5 and 6, residual heat removal is through the Shutdown Cooling System and MFW is not required. Therefore, the MFIVs are normally closed.

With one MFIV inoperable, action must be taken to close or isolate the inoperable valve within 72 hours. When the valve is closed or isolated, it is performing the required safety function (e.g., to isolate the main feedwater line) and continued operation in the applicable MODES is allowed.

The 72 hour Completion Time takes into account the back up capability afforded by the OPERABLE MFRVs and the SFRVs, diversity of actuation signals, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable to return the MFIV to OPERABLE status, close the MFIV, or otherwise isolate the affected flow path.

Inoperable MFIVs that cannot be restored to OPERABLE status within 72 hours, but are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day time is reasonable in view of valve status indications available in the control room, and other administrative controls to ensure that these valves are closed or isolated.

If the MFIVs cannot be restored to OPERABLE status, closed, or isolated in the required time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 in the following 30 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 plant systems.

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

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

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

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

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

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

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

TS BASES CHANGE NO. 4 REPLACEMENT PAGES

(10 pages)

Replace the following pages of the Waterford 3 Technical Specifications Bases with the attached pages. The revised pages are identified by Change No. 4 and contain a vertical line indicating the area of change. Pages B 3/4 7-4a, B 3/4 7-4b, and B 3/4 7-4c were changed due to the overflow of text from the previous page.

Remove Page	**Insert Page**
B 2-6	B 2-6
B 3/4 3-1c	B 3/4 3-1c
B 3/4 3-1d	--------------
B 3/4 7-2	B 3/4 7-2
--------------	B 3/4 7-2a
B 3/4 7-4	B 3/4 7-4
B 3/4 7-4a	B 3/4 7-4a
B 3/4 7-4b	B 3/4 7-4b
B 3/4 7-4c	B 3/4 7-4c
B 3/4 7-7	B 3/4 7-7
B 3/4 7-8	B 3/4 7-8

DNBR - Low (Continued)

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

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

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

Steam Generator Level - High

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

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a steam line break event with a loss-of-offsite power. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a nominal setpoint of 23.8 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.

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

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

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

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.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The two electric-driven emergency feedwater pumps combined are capable of delivering a total feedwater flow of 575 gpm at a pressure of 1102 psig to the entrance of the steam generator(s). The steam-driven emergency feedwater pump is capable of delivering a total feedwater flow of 575 gpm at a pressure of 1102 psig to the entrance of the steam generator(s). This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

The surveillance requirement to verify the minimum pump discharge pressure on recirculation flow ensures that the pump performance curve has not degraded below that used to show that the pumps meet the above flow requirements and is consistent with the requirements of ASME Section XI.

3/4.7.1.3 CONDENSATE STORAGE POOL

The OPERABILITY of the condensate storage pool (CSP) with the minimum water volume of 173,500 gallons (170,000 gallons for EFW system usage and 3,500 gallons for CCW makeup system usage), plus makeup from one Wet Cooling Tower (WCT) basin, ensures that sufficient water is available to cool the Reactor Coolant System to shutdown cooling entry conditions following any design basis accident. This makeup water includes the capability to maintain HOT STANDBY for at least an additional 2 hours prior to initiating shutdown cooling.

The combined capacity (CSP and one WCT) provides sufficient cooling for 24 hours until shutdown cooling is initiated in the event the ultimate heat sink sustains tornado damage concurrent with the tornado event.

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

3/4.7.1.3 CONDENSATE STORAGE POOL (Continued)

The CSP contained water volume limit (91% indicated in MODES 1, 2, and 3) includes an allowance for water not usable because of vortexing and instrumentation uncertainties. This provides an assurance that a minimum of 170,000 gallons is available for the EFW system and that 3,500 gallons is available for the CCW makeup system. The CSP contained water volume limit (11% indicated in MODE 4) also includes an allowance for water not usable because of vortexing and instrumentation uncertainties. This provides an assurance that minimum of 3,500 gallons is available in the CSP for the CCW makeup system.

3/4.7.4 ULTIMATE HEAT SINK

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

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

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

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). The bases for this elevation is WCT water evaporation and drift loss calculations, which concluded that during a LOCA 164,389 gallons (218,155 gallons with the non-essential load of spent fuel cooling) would be consumed from one WCT basin. When the WCT basin water level is maintained at -9.86 ft MSL, each basin has a minimum capacity of 174,000 gallons. The WCT basin is also credited as a source of Emergency Feedwater (EFW). However, the above LOCA water usage bounds the amount of EFW required from the WCT basin for all design basis accident conditions. Each WCT consists of two cells, each cell is serviced by 4 induced draft fans, for a total of 8 per WCT. There is a concrete partition between the cells that prevents air recirculation between the fans of each cell. Covers are required on fans declared out-of-service to prevent air recirculation between fans within a cell.

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

3/4.7.4 ULTIMATE HEAT SINK (Continued)

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

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related 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.

3/4.7.5 FLOOD PROTECTION

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

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

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

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

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

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

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.

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

3/4.7.6.5 CONTROL ROOM ISOLATION AND PRESSURIZATION

This specification provides the operability requirements for the control room envelope isolation and pressurization boundaries. The Limiting Condition for Operation (LCO) specifies specific ACTION STATEMENTS for inoperable components of the control room ventilation systems, separate from the S-8 and AH-12 units. The operability of the remaining parts of the system affect the ability of the control room envelope to pressurize.

ACTION STATEMENTS a and b focus on maintaining isolation characteristics. The valves in the flow path referred to in ACTION a are HVC-102 & HVC-101. The Outside Air Intake (OAI) "series isolation valves" of ACTION b and c are as follows:

NORTH OAI - HVC-202B & HVC-201A
HVC-202A & HVC-201B

CONTROL ROOM AIR TEMPERATURE (Continued)

SOUTH OAI - HVC-204B & HVC-203A
HVC-204A & HVC-203B

ACTION STATEMENT c preserves the operator action (i.e., manually initiated filtered pressurization) that maintains the control room envelope at a position pressure during a radiological emergency. As indicated above each OAI series isolation valve is powered by the opposite train. With more than one OAI flow path inoperable a single failure (i.e., train A or B) could prohibit the ability to maintain the control envelope at a positive pressure. Therefore, in the specified condition, ACTION c requires an additional flow path to be returned to service within 7 days.

ACTION STATEMENT d.2.a is intended to address an intentional breach in the control room pressurization boundary as necessary to support maintenance or modification. A breach of this nature shall be limited in size and governed under administrative controls. The size restrictions as stated in the ACTION are such that should a toxic event occur control room integrity can be immediately restored as described below. ACTION STATEMENT d.2.b is intended to restore pressurization ability as soon as possible for unintended breaches in the envelope. The 48 hours to locate an unidentified breach is based on an evaluation that considered troubleshooting tasks that would be performed as necessary should the integrity of the Control Room Envelope pressure boundary fall into question. Estimated times associated with each task were based on sound engineering judgement. The ACTION statements also recognize the MODE-independent nature of the toxic chemical threat and provides for operator protection in the event of a toxic chemical release concurrent with a breach in the control room envelope. In addition, provisions have been added to the specification that, in the event of a toxic chemical event that threatens control room habitability while in the ACTION statements, "immediate steps" will be initiated to place the plant in a safe condition. In this context, the phrase "immediate steps" is taken to mean that the operators should immediately take reasonable action to restore a known breach in the envelope to an air-tight condition. Amplifying instructions are provided in Waterford 3 Administrative procedures, which impose special controls for work that will breach the control room envelope.

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

3/4.7.7 CONTROLLED VENTILATION AREA SYSTEM

The OPERABILITY of the controlled ventilation area system ensures that radioactive materials leaking from the penetration area or the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alphaemitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage frombyproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

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 ≤ 42°F and flow rate of ≥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

3/4.7.12 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.

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 either low chilled water inlet or 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 deadbands for both switches allow 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.

The 31 day Surveillance Requirement (SR) to verify the chilled water outlet temperature 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.

TS BASES CHANGE NO. 5 INSERTION PAGE

(1 page)

Insert the following page into the Waterford 3 Technical Specifications Bases concurrently with the implementation of TS Amendment 170. The revised page is identified by Change No. 5. Insertion of this TS Bases page is required to resolve an editorial conflict between recently implemented TS Bases Change No. 4 and approved TS Amendment 170.

Remove Page	**Insert Page**
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3/4.7.4 ULTIMATE HEAT SINK (Continued)

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