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Waterford 3

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PR

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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
Technical Specification Bases Update to the NRC for the Period January 16, 2002 Through September 30, 2002

Gentlemen:

Pursuant to Waterford Steam Electric Station Unit 3 Technical Specification 6.16, Entergy Operations, Inc. (EOI) hereby submits an update of all changes made to Waterford 3 Technical Specification Bases since the last submittal per letter W3F1-2002-0005, dated January 16, 2002. This TS Bases update is consistent with the update frequency listed in 10 CFR 50.71(e).

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

Very truly yours,

K.J. Peters
Director, Nuclear Safety Assurance

KJP/RLW/cbh

Attachment: Waterford 3 Technical Specification Bases Revised Pages

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

A001

ATTACHMENT 1
TO W3F1-2002-0090

Waterford 3 Technical Specification Bases Revised Pages

T.S. Bases Change No.	Implement Date	Affected TS Bases Pages	Topic of Change
10	1/31/02	B 3/4 9-3	Delete TS Bases section 3/4.9.12, Fuel Handling Building Ventilation concurrently with TS Amendment 176 implementation.
11	2/18/02	B 3/4 4-7 New page B 3/4 4-7a	Change to TS Bases section 3/4.4.8 implemented by ER-W3-2002-0080 concurrently with TS Amendment 177 to reflect relaxation in the allowable cooldown rate in the Reactor Coolant System TS 3.4.8.1, "Pressure/Temperature Limits."
12	3/21/02	B 2-1 B 2-2 New page B 2-2a B 2-5 B 3/4 2-4	Change to TS Bases sections 2.1.1, 2.2.1, and 3/4.2.7 implemented by ER-W3-2002-0110 concurrently with TS Amendment 181 to reflect a Safety Limit change from Peak Fuel Centerline Temperature Heat Rate to a Peak Fuel Centerline Temperature.
13	3/27/02	B 3/4 8-2	Change to TS Bases section 3/4.8.1, 3/4.8.2, and 3/4.8.3 implemented by ER-W3-2002-0105-000 concurrently with TS Amendment 180 to reflect changes in performance of Emergency Diesel Generator Surveillance Requirements (SR 4.8.1.1.2.3.1, 2, 4, 6, 10 and 12) during any mode of plant operation.
14	4/4/02	B 3/4 1-6	Change to TS Bases section 3/4.1.3 implemented by ER-W3-1999-0411-002 concurrently with TS Amendment 182 to delete the requirements associated with part-length control element assemblies (PLCEAs) and reflect the removal of the four element CEAs on the core periphery.

TS BASES CHANGE NO. 10 REPLACEMENT PAGE

(1 page)

Replace the following page of the Waterford 3 Technical Specifications Bases with the attached page. The change is being made to the Bases concurrent with the implementation of TS Amendment 176. The revised section is identified by Change number 10 and contains vertical lines indicating the areas of change.

Remove	Insert
B 3/4 9-3	B 3/4 9-3

BASES

3/4.9.10 and 3/4 9 11 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

TS BASES CHANGE NO. 11 REPLACEMENT PAGES

Replace the following page of the Waterford 3 Technical Specifications Bases with the attached pages. The change is being made to the Bases concurrent with the implementation of TS Amendment 177. The revised section is identified by a vertical line indicating the area for change.

Remove	Insert
B3/4 4-7	B3/4 4-7
	B3/4 4-7a

As used in this specification, the term 'cold leg temperature' is intended to be representative of that entering the reactor vessel beltline. During periods with the reactor coolant pumps in operation, the T_{COLD} temperature indication meets this intent. However, during periods when the reactor coolant pumps are not in service, the T_{COLD} temperature indicator is in a stagnant segment of piping and the indication may not necessarily be indicative of that entering the reactor vessel beltline. During the condition when the reactor coolant pumps are operating, the lowest T_{COLD} of a loop with an operating reactor coolant pump is used to monitor the P-T limits. However, during periods when the shutdown cooling system is in operation and following coastdown of the last RCP, the shutdown cooling temperature is the 'cold leg temperature' used to monitor P-T limits.

The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60°F per hour or cooldown rate of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3. The limitations on the Reactor Coolant System heatup and cooldown rates are further restricted due to stress limitations in the Reactor Coolant Pump. As part of the LOCA support scheme, the Reactor Coolant Pump has a ring around the suction nozzle of the pump. The support skirt is welded to the ring. Due to this design, the heatup and cooldown rates must be limited to maintain acceptable thermal stresses.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper and nickel content of the material in question, can be predicted using FSAR Table 5.3-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10 CFR Part 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in FSAR Table 5.3-10. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

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

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

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 12 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	Insert
B 2-1	B 2-1
B 2-2	B 2-2
-	B 2-2a
B 2-5	B 2-5
B 3/4 2-4	B 3/4 2-4

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

2.1.1 REACTOR CORE

• **(DRN 02-458)**

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the peak fuel centerline temperature below the melting point.

• **(DRN 02-458)**

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to 1.26 for the CE-1 correlation and is established as a Safety Limit. This value is based on a statistical combination of uncertainties. It includes uncertainties in the CHF correlation, allowances for rod bow and hot channel factors (related to fuel manufacturing variations) and allowances for other hot channel calculative uncertainties.

• **(DRN 02-458)**

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, fuel centerline melting is established as a Safety Limit. The design melting point of new fuel with no burnable poison is 5080°F. The melting point is adjusted downward from this temperature depending on the amount of burnup and amount and type of burnable poison in the

• **(DRN 02-458)**

2.1.1 REACTOR CORE (Continued)

• **(DRN 02-458)**
fuel. The 58°F per 10,000 MWD/MTU adjustment for burnup was accepted by the NRC in Topical Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992. Adjustments for burnable poisons are established based on NRC approved Topical Report CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

A steady state peak linear heat rate of 21 kW/ft has been established as the Limiting Safety System Setting to prevent fuel centerline melting during normal steady state operation. Following design basis anticipated operational occurrences, the transient linear heat rate may exceed 21 kW/ft provided the fuel centerline melt temperature is not exceeded.
• **(DRN 02-458)**

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1974 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. RPS Trip Setpoints values are determined by means of an explicit setpoint calculation analysis.

2.2 1 REACTOR TRIP SETPOINTS (Continued)

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

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.26 and 21.0 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density -High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in the latest applicable revision of CEN-305-P, "Functional Design Requirements for a Core Protection Calculator" and; CEN-304-P, "Functional Design Requirements for a Control Element Assembly Calculator."

Local Power Density - High (Continued)

• **(DRN 02-458)**

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the fuel centerline melt Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

• **(DRN 02-458)**

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1860 psia. At this pressure a DNBR - Low trip will automatically occur. This low pressure trip also provides protection against steam generator tube rupture events. The DNBR is calculated in the CPC utilizing the following information:

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

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

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

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than the fuel design limit such that the decrease

DNBR MARGIN (Continued)

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

3/4.2.5 RCS FLOW RATE

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

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses, with adjustment for instrument accuracy of ±2• F, and that the peak linear heat generation rate and the moderator temperature coefficient effects are validated.

3/4.2.7 AXIAL SHAPE INDEX

• **(DRN 02-458)**

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

• **(DRN 02-458)**

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses. The inputs to CPCs and COLSS are the most limiting. The values are adjusted for an instrument accuracy of ± 25 psi. The sensitive events are SGTR, LOCA, FWLB and loss of condenser vacuum to initial high pressure, and MSLB to initial low pressure.

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

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are consistent with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision I, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Other provisions are derived from Generic Letter 93-05 "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation" 94-01 "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," and NUREG 1432 Standard Technical Specifications Combustion Engineering Plants.

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

- **(DRN 02-0607)**
- **(DRN 02-0607)**

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

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

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

TECHNICAL SPECIFICATION BASES
CHANGE NO. 13 REPLACEMENT PAGE(S)
(1 page)

Replace the following page(s) of the Waterford 3 Technical Specification Bases with the attached page(s). The revised pages are identified by Change Number 13 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 8-2	B 3/4 8-2

TECHNICAL SPECIFICATION BASES
CHANGE NO. 14 REPLACEMENT PAGE(S)
(1 page)

Replace the following page(s) of the Waterford 3 Technical Specification Bases with the attached page(s). The revised page(s) are identified by Change Number 14 and contain the appropriate DRN number and a vertical line indicating the areas of change.

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B 3/4 1-6	B 3/4 1-6

MOVABLE CONTROL ASSEMBLIES (Continued)

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

- **(DRN 02-632)**
- **(DRN 02-632)**