UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
License No. NPF-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Entergy Operations, Inc. (the licensee) dated June 22, and December 9, 1994, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-38 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 113, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 5, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 113

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
2-3	2-3
3/4 3-20	3/4 3-20
B 2-2	B 2-2
B 2-3	B 2-3
B 3/4 3-1	B 3/4 3-1
-	B 3/4 3-1a
B 3/4 3-2	B 3/4 3-2

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
Four Reactor Coolant Pumps Operating	≤ 108% of RATED THERMAL POWER	≤ 108.76% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.257% of RATED THERMAL POWER	≤ 0.280% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2350 psia	≤ 2359 psia
5. Pressurizer Pressure - Low	≥ 1684 psia (2)	≥ 1649.7 psia (2)
6. Containment Pressure - High	≤ 17.1 psia	≤ 17.4 psia
7. Steam Generator Pressure - Low	≥ 764 psia (3)	≥ 749.9 psia (3)
8. Steam Generator Level - Low	≥ 27.4% (4)	≥ 26.48% (4)
9. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
10. DNBR - Low	≥ 1.26 (5)	≥ 1.26 (5)
11. Steam Generator Level - High	≤ 87.7% (4)	≤ 88.62% (4)
12. Reactor Protection System Logic	Not Applicable	Not Applicable
13. Reactor Trip Breakers	Not Applicable	Not Applicable
14. Core Protection Calculators	Not Applicable	Not Applicable
15. CEA Calculators	Not Applicable	Not Applicable
16. Reactor Coolant Flow - Low	≥ 19.00 psid (7)	≥ 18.47 psid (7)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER.

(6) Note 6 has been deleted.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.3 psia
c. Pressurizer Pressure - Low	$\geq$ 1684 psia$^{(1)}$	$\geq$ 1644 psia$^{(1)}$
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	$\leq$ 17.7 psia	$\leq$ 18.0 psia
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.3 psia
c. Pressurizer Pressure - Low	$\geq$ 1684 psia$^{(1)}$	$\geq$ 1644 psia$^{(1)}$
d. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ 764 psia$^{(2)}$	$\geq$ 748 psia$^{(2)}$
c. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.3 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. SAFETY INJECTION SYSTEM SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Pool - Low	10.0% (57,967 gallons)	9.08% (52,634 gallons)
c. Automatic Actuation Logic	Not Applicable	Not Applicable
6. LOSS OF POWER		
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3245 volts	≥ 3245 volts
b. 480 V Emergency Bus Undervoltage	≥ 372 volts	≥ 354 volts
c. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3875 volts	≥ 3860 volts
7. EMERGENCY FEEDWATER (EFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (1&2) Level - Low	≥ 27.4%(3)(4)	≥ 26.7%(3)(4)
c. Steam Generator ΔP - High (SG-1 > SG-2)	≤ 123 psid	≤ 134 psid
d. Steam Generator ΔP - High (SG-2 > SG-1)	≤ 123 psid	≤ 134 psid
e. Steam Generator (1&2) Pressure - Low	≥ 764 psia(2)	≥ 748 psia(2)
f. Automatic Actuation Logic	Not Applicable	Not Applicable
g. Control Valve Logic (Wide Range SG Level - Low)	≥ 36.3%(3)(5)	≥ 35.3%(3)(5)

2.1.1 REACTOR CORE

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 dynamically adjusted peak linear heat rate of the fuel at or less than 21.0 kW/ft which will not cause fuel centerline melting in any fuel rod.

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.

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, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted.

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. 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."

Manual Reactor Trip

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

Linear Power Level - High

The Linear Power Level - High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of less than or equal to 108% of RATED THERMAL POWER.

Logarithmic Power Level - High

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

Pressurizer Pressure - High

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

Pressurizer Pressure - Low

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

BASES

Containment Pressure - High

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

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level - Low

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

Local Power Density - High

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

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

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

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

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the Reactor Protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The redundancy design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs will use DNBR and LPD penalty factors to restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded, a reactor trip will occur.

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The quarterly frequency for the channel functional tests for these systems comes from the analyses presented in topical report CEN-327: RPS/ESFAS Extended Test Interval Evaluation, as supplemented.

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

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

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

Response time may be 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 either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION
3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

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

3/4.3.3.2 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.