October 1, 1999

Mr. William R. McCollum, ~~Jr.~~
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

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SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: ISSUANCE OF AMENDMENTS - PRESSURE-TEMPERATURE OPERATING CURVES (TAC NOS. MA5459, MA5460, AND MA5461)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 307 , 307 , and 307 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 11, 1999, which was supplemented by letter dated July 13, 1999.

The amendments incorporate revisions to the pressure-temperature limits; the heatup, cooldown, and inservice test limits for the reactor coolant system to a maximum of 33 effective Full Power Years; the low temperature overpressure protection system; and operational requirements for the reactor coolant pumps.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

ORIGINAL SIGNED BY:

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270 and 50-287

Enclosures:
1. Amendment No. 307 to DPR-38
2. Amendment No. 307 to DPR-47
3. Amendment No. 307 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

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

October 1, 1999

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

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4. Safety Evaluation

cc w/encls: See next page

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 307
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Energy Corporation (the licensee) dated May 11, 1999, as supplemented by letter dated July 13, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 307 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 1, 1999

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 307
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Energy Corporation (the licensee) dated May 11, 1999, as supplemented by letter dated July 13, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 307 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: October 1, 1999

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 307
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Energy Corporation (the licensee) dated May 11, 1999, as supplemented by letter dated July 13, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 307 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 1, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 307

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 307

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 307

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert	Remove	Insert
3.4.3-3	3.4.3-3	B 3.4.3-1	B 3.4.3-1
3.4.3-4	3.4.3-4	B 3.4.3-2	B 3.4.3-2
3.4.3-5	3.4.3-5	B 3.4.3-3	B 3.4.3-3
3.4.3-6	3.4.3-6	B 3.4.3-4	B 3.4.3-4
3.4.3-7	3.4.3-7	B 3.4.3-5	B 3.4.3-5
3.4.3-8	3.4.3-8	B 3.4.3-6	B 3.4.3-6
3.4.3-9	3.4.3-9	B 3.4.3-7	B 3.4.3-7
3.4.3-10	3.4.3-10	B 3.4.3-8	-------
3.4.3-11	3.4.3-11	B 3.4.12-4	B 3.4.12-4
3.4.3-12	3.4.3-12	B 3.4.12-5	B 3.4.12-5
3.4.3-13	3.4.3-13	B 3.4.12-6	B 3.4.12-6
3.4.12-1	3.4.12-1	B 3.4.12-7	B 3.4.12-7
3.4.12-2	3.4.12-2	B 3.4.12-8	B 3.4.12-8
		B 3.4.12-9	B 3.4.12-9
		B 3.4.12-10	B 3.4.12-10
		B 3.4.12-11	B 3.4.12-11
		B 3.4.12-12	B 3.4.12-12

Table 3.4.3-1 (page 1 of 1)
Operational Requirements for Unit Heatup

CONSTRAINT	RC TEMPERATURE[a]	MAXIMUM HEATUP RATE	ALLOWED PUMP COMBINATION
RC Temperature[a]	T < 280°F T ≥ 280°F	50°F/hr 100°F/hr	NA NA
RC Pumps	T < 250°F T ≥ 250°F	NA NA	≤ two pumps Any

(a) RC Temperature is cold leg temperature if one or more RC pumps are in operation; otherwise it is the LPI cooler outlet temperature.

Table 3.4.3-2 (page 1 of 1)
Operational Requirements for Unit Cooldown

CONSTRAINT	RC TEMPERATURE[a]	MAXIMUM COOLDOWN RATE[b]	ALLOWED PUMP COMBINATION
RC Temperature[a]	T ≥ 280°F	≤ 50°F in any 1/2 hour period	NA
	150°F ≤ T < 280°F	≤ 25°F in any 1/2 hour period	NA
	T< 150°F	≤ 10°F in any one hour period	NA
	RCS depressurized[c]	≤ 50°F in any one hour period	NA
RC Pumps	T ≥ 250°F	NA	Any
	T < 250°F	NA	≤ two pumps

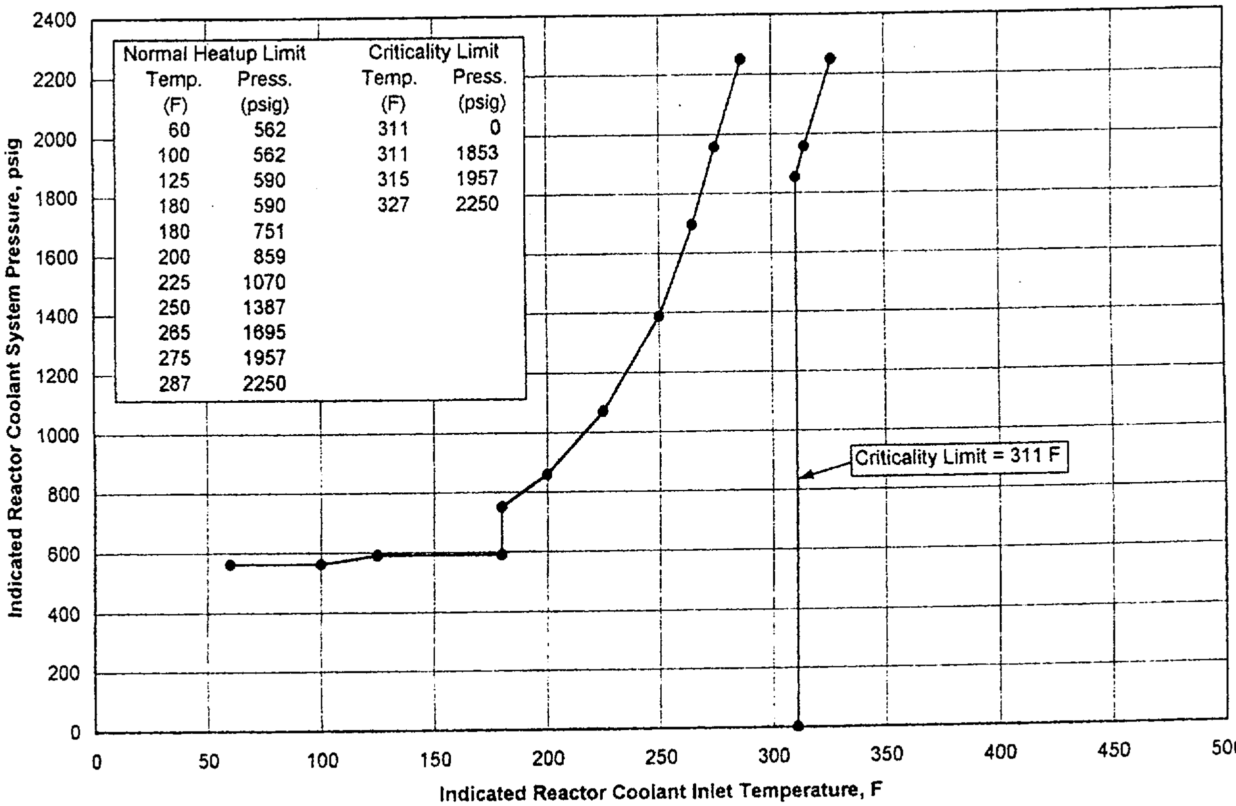
(a) RC Temperature is cold leg temperature if one or more RC pumps are in operation or if on natural circulation cooldown; otherwise it is the LPI cooler outlet temperature.

(b) These rate limits must be applied to the change in temperature indication from cold leg temperature to LPI cooler outlet temperature per Note (a).

(c) When the RCS is depressurized such that all three of the following conditions exist:

a) RCS temperature < 200°F,
b) RCS pressure < 50 psig,
c) All RC Pumps off,

the maximum cooldown rate shall be relaxed to ≤ 50°F in any 1 hour period.

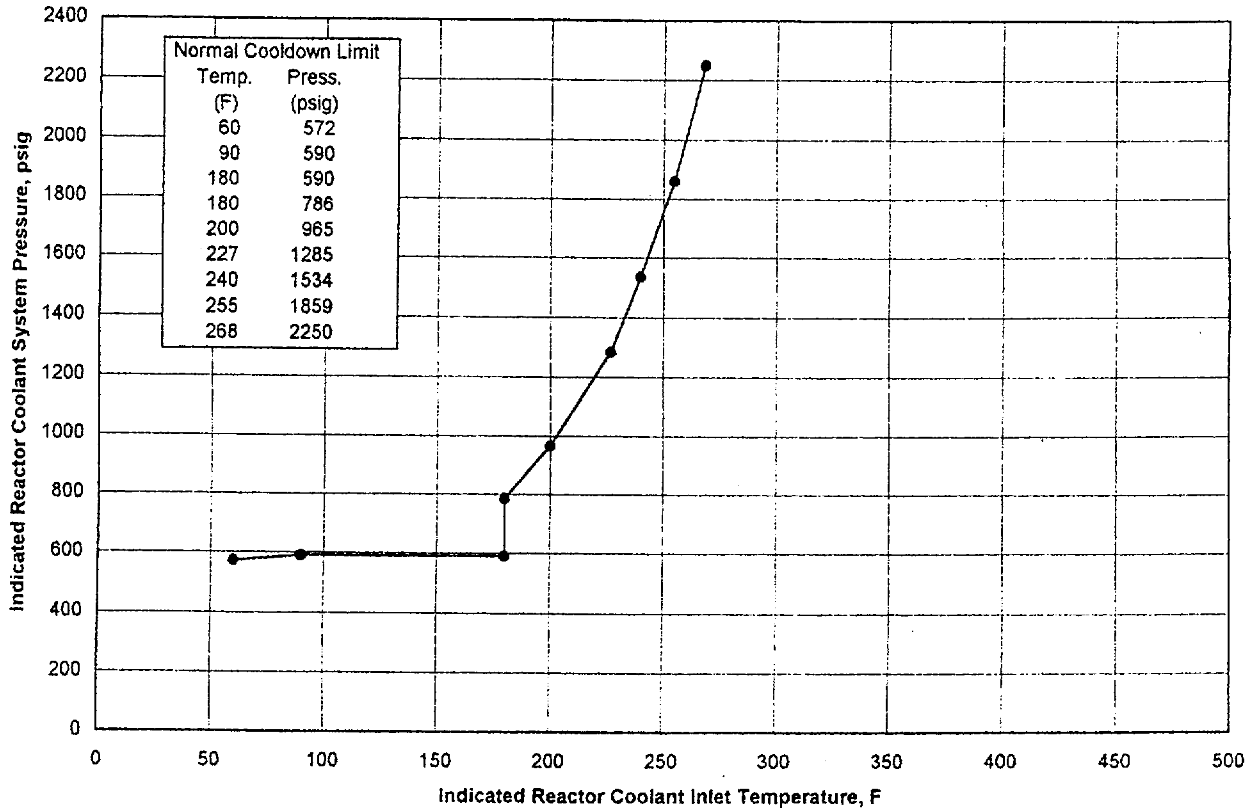


Normal Heatup Limit Temp. (F)	Normal Heatup Limit Press. (psig)	Criticality Limit Temp. (F)	Criticality Limit Press. (psig)
60	562	311	0
100	562	311	1853
125	590	315	1957
180	590	327	2250
180	751		
200	859		
225	1070		
250	1387		
265	1695		
275	1957		
287	2250		

The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

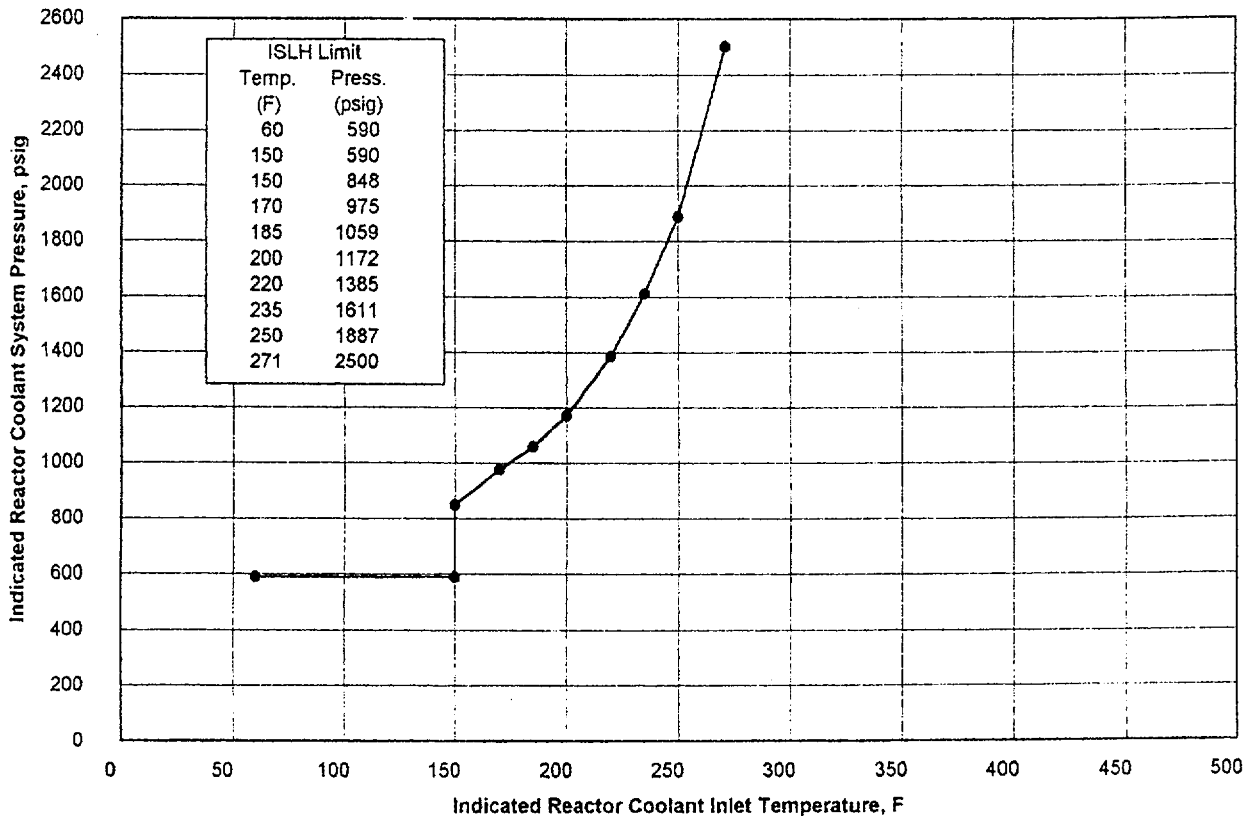
Figure 3.4.3-1 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

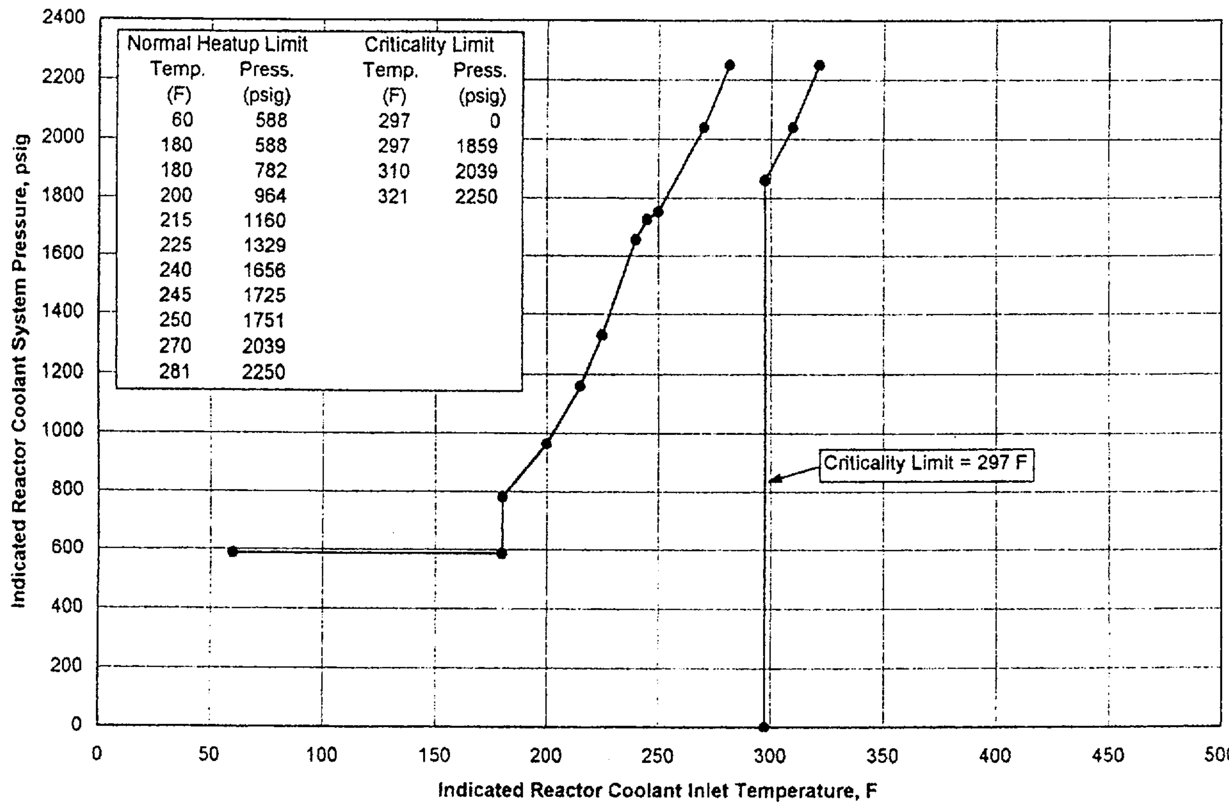
Figure 3.4.3-2 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-3 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1

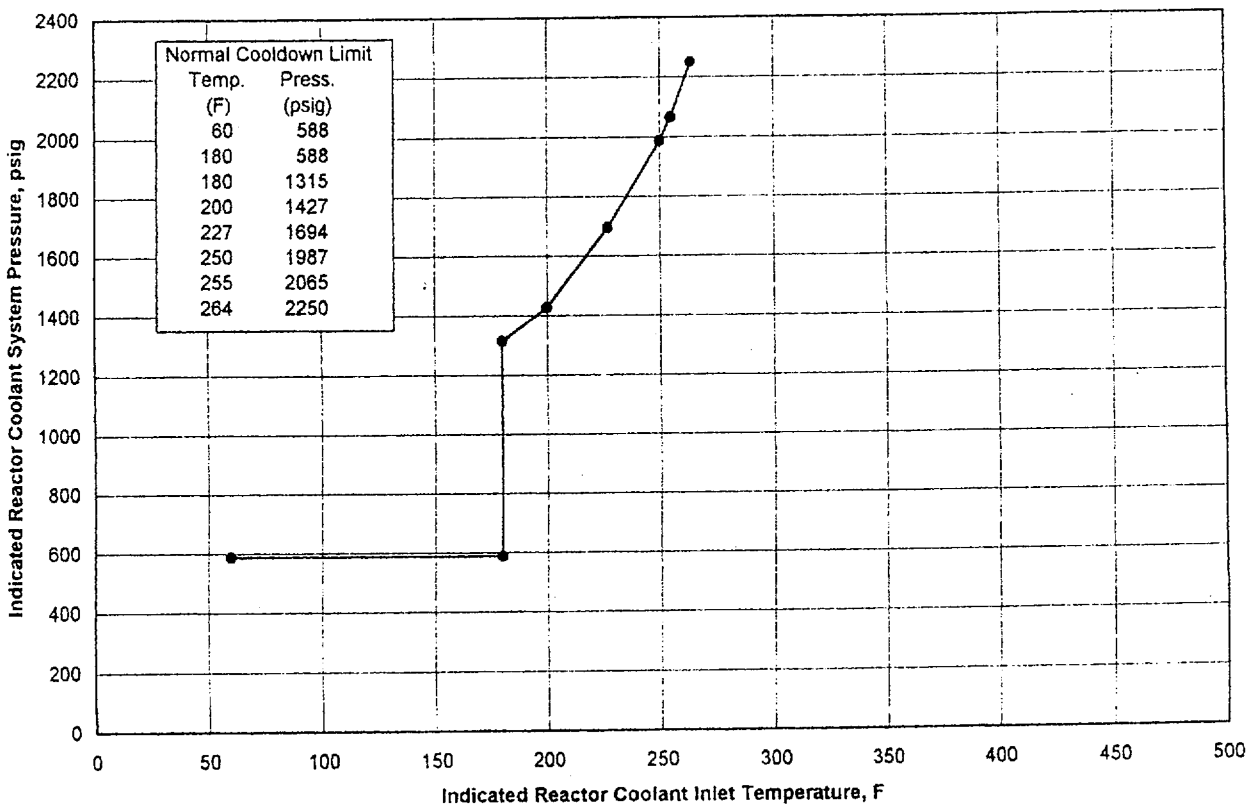


Normal Heatup Limit Temp. (F)	Normal Heatup Limit Press. (psig)	Criticality Limit Temp. (F)	Criticality Limit Press. (psig)
60	588	297	0
180	588	297	1859
180	782	310	2039
200	964	321	2250
215	1160		
225	1329		
240	1656		
245	1725		
250	1751		
270	2039		
281	2250		

The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

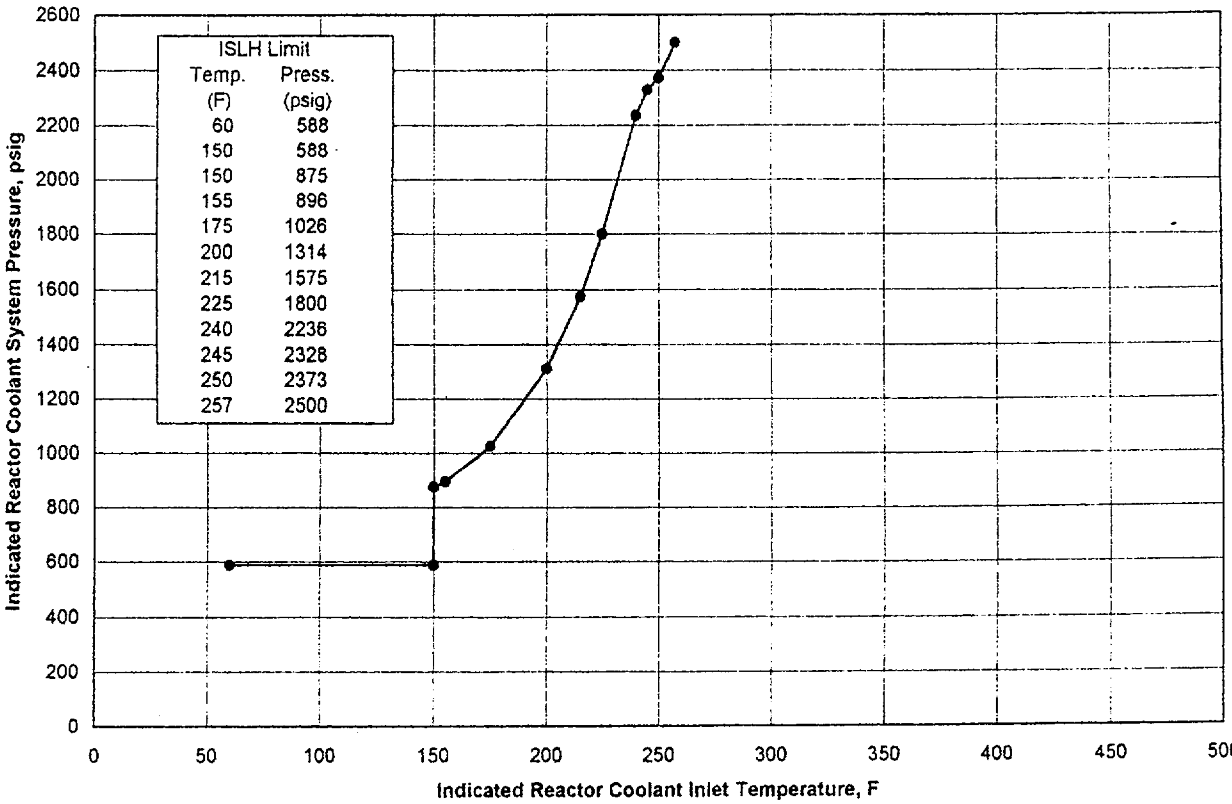
Figure 3.4.3-4 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

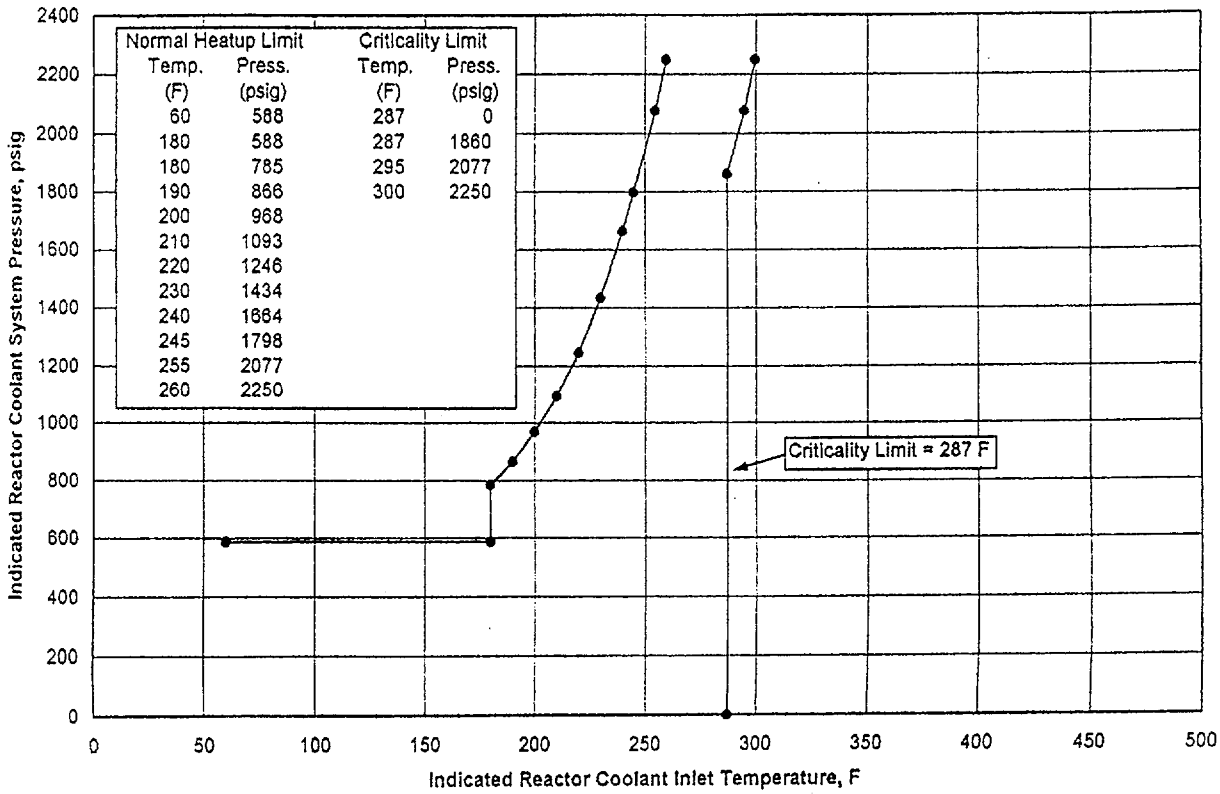
Figure 3.4.3-5 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

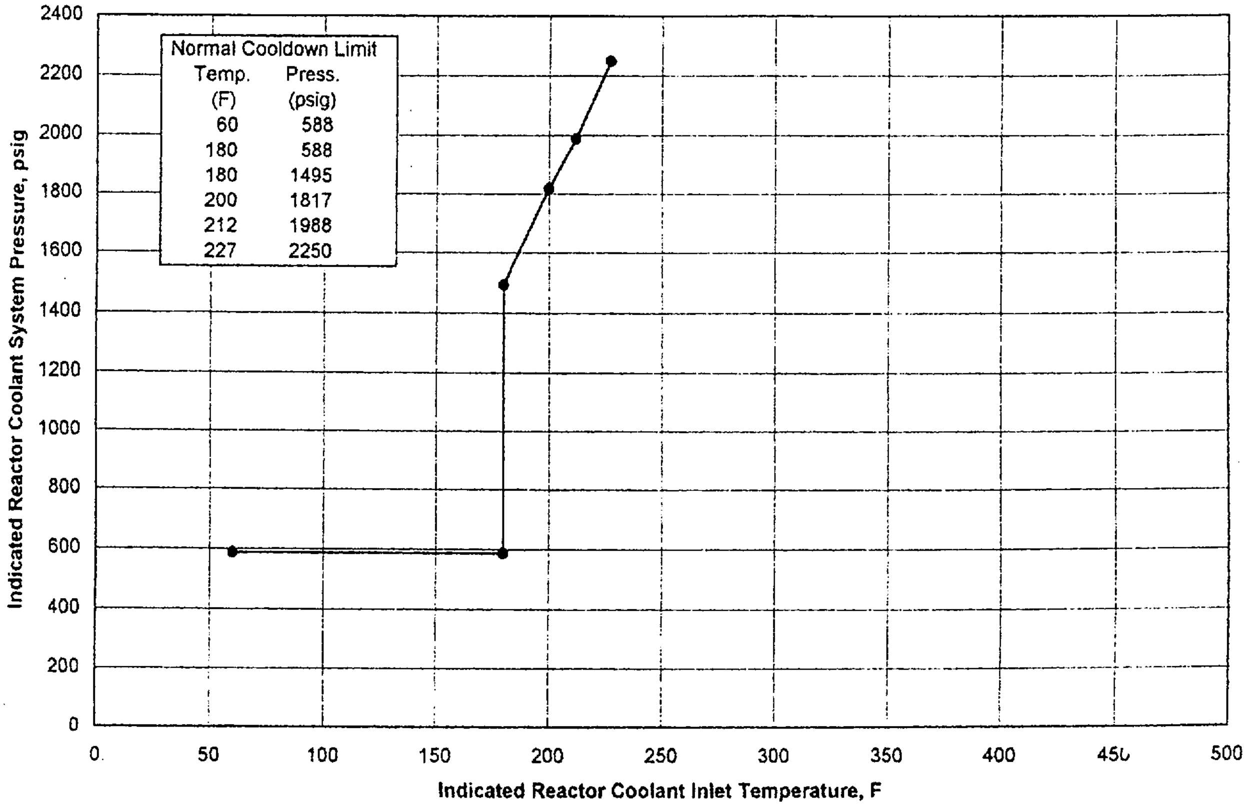
Figure 3.4.3-6 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

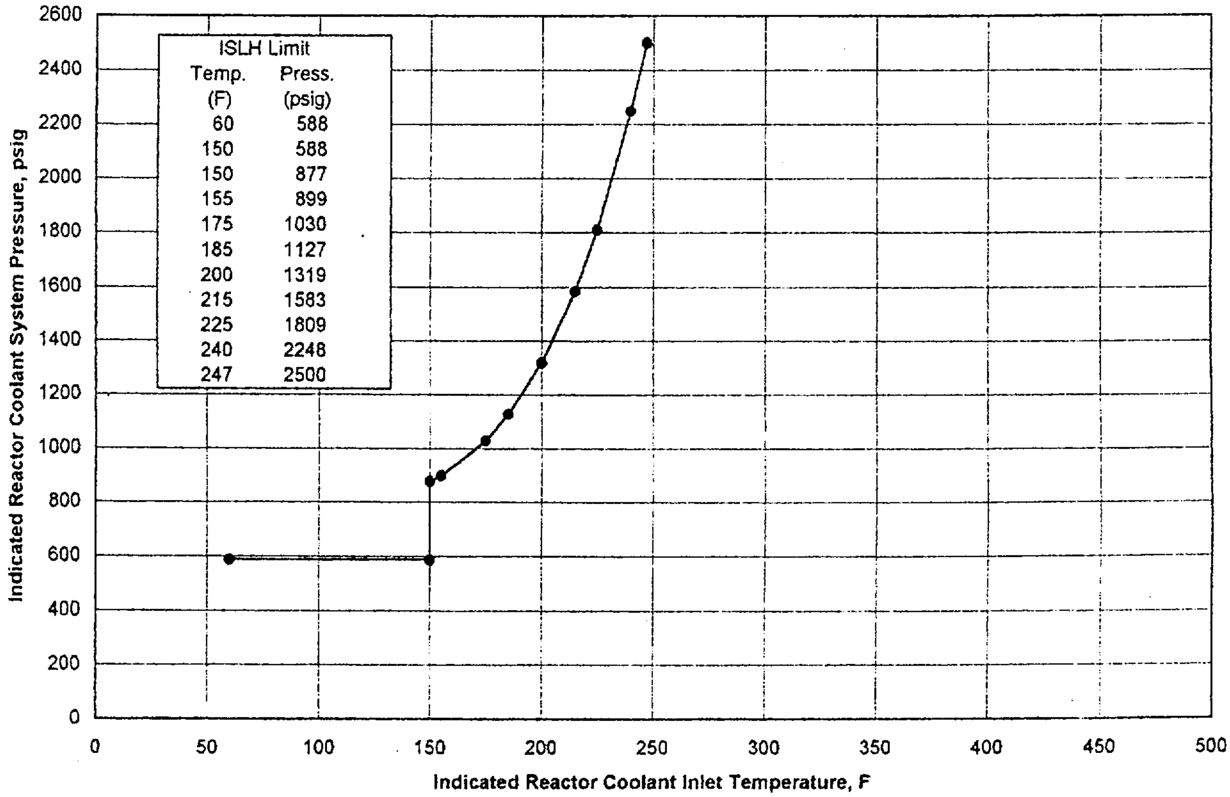
Figure 3.4.3-7 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-8 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-9 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12

An LTOP System shall be OPERABLE with high pressure injection (HPI) deactivated, and the core flood tanks (CFTs) isolated and:

a. An OPERABLE power operated relief valve (PORV) with a lift setpoint of ≤ 535 psig; and

b. Administrative controls implemented that assure ≥ 10 minutes are available for operator action to mitigate an LTOP event.

APPLICABILITY:

MODE 3 when any RCS cold leg temperature is ≤ 325°F,
MODES 4, 5, and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open.

--NOTES--

1. CFT isolation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in Specification 3.4.3.

2. The PORV is not required to be OPERABLE when no HPI pumps are running and RCS pressure < 100 psig.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. HPI activated.	A.1	Initiate action to deactivate HPI.	Immediately
B. A CFT not isolated when CFT pressure is greater than or equal to the maximum RCS pressure for existing temperature allowed by Specification 3.4.3.	B.1	Isolate affected CFT.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1	Be in MODE 4 with RCS temperature > 200°F.	12 hours
	<u>OR</u>		
	C.2	Depressurize affected CFT to < 373 psig.	12 hours
D. PORV inoperable.	D.1	Restore PORV to OPERABLE status.	1 hour
E. Required Action and associated Completion Time of Condition D not met.	E.1	Be in MODE 3 with RCS average temperature > 325°F.	23 hours
	<u>OR</u>		
	E.2	Depressurize RCS to < 100 psig.	35 hours

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1 through 3.4.3-9 contain P/T limit curves for heatup, cooldown, and leak and hydrostatic (LH) testing. Tables 3.4.3-1 and 3.4.3-2 contain data for the maximum rate of change of reactor coolant temperature. The minimum temperature indicated in the P/T limit curves and tables of 60°F is the lowest unirradiated nil ductility reference temperature (RT_{NDT}) of all materials in the reactor vessel. This temperature (60°F) is the minimum allowable reactor pressure vessel temperature if any head closure stud is not fully detensioned.

Figures 3.4.3-1, 3.4.3-2, 3.4.3-4, 3.4.3-5, 3.4.3-7 and 3.4.3-8 define an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 2).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 3).

BASES

BACKGROUND (continued)

Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with ASTM E 185 (Ref. 4), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 2.

The actual shift in the nil ductility reference temperature (RT_{NDT}) of the vessel material will be established periodically by evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 5) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 2.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The calculation to generate the LH testing curve uses different safety factors (per Ref. 2) than the heatup and cooldown curves.

The P/T limit curves and associated temperature rate of change limits are developed in conjunction with stress analyses for large numbers of operating cycles and provide conservative margins to nonductile failure. Although created to provide limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

The criticality limit curve includes the Reference 1 requirement that it be 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for LH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

BASES

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from accident analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Since the P/T limits are not derived from any accident analysis, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 7).

LCO

The three elements of this LCO are:

a. The limit curves for heatup and cooldown,

b. Limits on the rate of change of temperature, and

c. Allowable RC pump combinations.

The LCO is modified by three Notes. Note 1 states that for leak tests of the RCS and leak tests of connected systems where RCS pressure and temperature are controlling, the RCS may be pressurized to the limits of the specified figures. Note 2 states that for thermal steady state hydro tests required by ASME Section XI RCS may be pressurized to the limits Specification 2.1.2 and the specified figures. The limits on the rate of change of reactor coolant temperature RCS P/T Limits are the same ones used for normal heatup and cooldown operations. Note 3 states the RCS P/T limits are not applicable to the pressurizer.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

Table 3.4.3-1 includes temperature rate of change limits with allowable pump combinations for RCS heatup while Table 3.4.3-2 includes temperature rate of change limits with allowable pump combinations for RCS cooldown. The breakpoints between temperature rate of change limits in these two tables are selected to limit reactor vessel thermal gradients to acceptable limits. The breakpoint between allowable pump combinations was selected based on operational requirements and are used to determine the change of RCS pressure associated with the change in number of operating reactor coolant pumps.

BASES

LCO (continued)

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and LH P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

The limits on allowable RC pump combinations controls the pressure differential between the vessel wall and the pressure measurement point and are used as inputs for calculating the heatup, cooldown and LH P/T limit curves. Thus, the LCO for the allowable RC pump combinations restricts the pressure at the vessel wall and ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or LH testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit (SL) 2.1, "SLs," also provide operational restrictions for pressure and temperature and maximum pressure. MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

BASES

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established plant procedures and administrative controls.

ASME Code, Section XI, Appendix E (Ref. 6) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The evaluation must extend to all components of the RCPB.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the unit must be brought to a lower MODE because: (a) the RCS remained in an unacceptable pressure and temperature region for an extended period of increased stress, or (b) a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

BASES

ACTIONS

B.1 and B.2 (continued)

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions B.1 and B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while reducing pressure and temperature conditions, a return to power operation may be considered without completing Required Action B.2.

Pressure and temperature are reduced by bringing the unit to MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished within this time in a controlled manner.

In addition to restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone, per Required Action C.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within limits is required every 30 minutes when RCS pressure or temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Thirty minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or LH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that requires this SR to be performed only during system heatup, cooldown, and LH testing.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. Regulatory Guide 1.99, Revision 2, May 1988.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. 10 CFR 50.36.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The following controls are required during the LTOP MODES to ensure that transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Limiting RCS makeup flow capability;

b. Deactivating HPI-ES;

c. Immobilizing CFT discharge isolation valves in their closed positions; and

d. Limiting the number of available pressurizer heater banks.

The Reference 3 analyses demonstrate the PORV can maintain RCS pressure below limits when both makeup flow capability and the number of available pressurizer heater banks is restricted. Consequently, the administrative controls require makeup flow capability and the number of available pressurizer heater banks to be limited in the LTOP MODES.

Since the PORV cannot protect the reactor vessel for engineered safeguards actuation of one or more HPI pumps, or discharging the CFTs, the LCO also requires the HPI-ES actuation circuits be deactivated and the CFTs isolated. The isolated CFTs must have their discharge valves closed and the valve power breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at 325°F. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 33 effective full power years (EFPYs) of operation for Units 1, 2, and 3.

This LCO will deactivate the HPI-ES actuation when the RCS temperature is ≤ 325°F.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORV is set to open at ≤ 535 psig. The setpoint is derived by modeling the performance of the LTOP system for different LTOP events. The PORV setpoint at or below the derived limit ensures the Reference 1 limits will be met.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The PORV setpoint is re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement induced by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.

The PORV is considered an active component. Therefore, its failure represents the worst case LTOP single active failure.

Administrative Controls Performance

Limiting RCS pressure when RCS temperature is < 325°F provides a minimum margin to the RCS P/T limit. Restricting RCS makeup flow capability, the number of available pressurizer heater banks, pressurizer level, and controls on the use of high pressure nitrogen limit the pressurization rate during an LTOP event. Alarms ensure early operator recognition of the occurrence of an incipient LTOP event. The combination of minimum margin to the limit, limited pressurization rate and OPERABLE alarms ensure ten minutes are available for operator action to mitigate an LTOP event.

RCS Vent Requirements for Testing

With the RCS depressurized, analyses show:

a. For HPI System testing, a vent of ≥ 3.6 square inches is capable of mitigating the transient resulting from HPI-ES actuation testing in which three HPI pumps inject to the RCS through two injection flow paths.

b. For CFT Discharge Testing, a vent of ≥ 201 square inches is capable of mitigating the transient resulting for discharge of both CFTs to the RCS.

The capacity of vents of these minimum sizes is sufficient to limit the RCS pressure to ≤ 400 psig, which is less than the maximum allowable pressure at minimum RCS temperature.

The RCS vent size will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.

These vents are passive and not subject to active failure.

The LTOP System satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36 (Ref.6).

BASES

LCO

The LCO requires an LTOP System OPERABLE with a limited coolant input capability and a pressure relief capability. The LCO requires HPI to be deactivated and the CFTs to be isolated. For pressure relief, it requires the pressurizer coolant at or below a maximum level and the PORV OPERABLE with a lift setting ≤ the LTOP limit.

The PORV is OPERABLE when its block valve is open, its lift setpoint is set at ≤ 535 psig and testing has proven its ability to open at that setpoint, and power is available to the two valves and their control circuits.

An RCS vent path capable of mitigating the most limiting LTOP event (except for HPI-ES actuation or CFT discharge) has a minimum equivalent diameter of 1-3/32 inches, which is equal to the inner throat diameter of the PORV.

Implementation of the following administrative controls assure that ≥ 10 minutes are available for operator action to mitigate an LTOP event:

1. RCS pressure:

   < 375 psig when RCS temperature ≤ 220°F

   < 525 psig when RCS temperature > 220°F and ≤ 325°F

2. Pressurizer level is maintained within the following limits:

   a. RCS pressure is > 100 psig:

      ≤ 220 inches when RCS temperature ≤ 325°F

   b. RCS pressure is ≤ 100 psig:

      ≤ 310 inches when RCS temperature ≤ 220°F.

      ≤ 380 inches while filling or draining the RCS when RCS temperature ≤ 160°F and no HPI pumps are running.

      When the RCS pressure is ≤ 100 psig, pressurizer level is normally maintained ≤ 220 inches except for certain RCS evolutions. The specified pressurizer level limits provide assurance that at least 10 minutes is available for operator action during those evolutions. The temperature limits are based on operational limits for the evolutions and are used in the analyses to determine allowable pressurizer levels.

3. Makeup flow is restricted with the HP-120 (makeup control valve) travel stop set to ≤ 98.0 gpm for all three units.

BASES

LCO (continued)

4. Three audible pressurizer level alarms at ≤ 225 inches, ≤ 260 inches, and ≤ 315 inches from the temperature compensated pressurizer level indication.

5. Two audible RCS pressure alarms at 375 psig and 525 psig.

6. High pressure nitrogen system is administratively controlled to prevent inadvertent pressurization of the RCS.

7. Core Flood Tank(s) are isolated as required by the LCO by closing the appropriate isolation valve(s) (either CF-1 and/or CF-2), tagging open the valve breaker(s), and tagging the valve(s) in the closed position.

8. The HPI safety injection flowpaths must be deactivated.

   a. Deactivating Train A of HPI is accomplished by either:

      1) Shutting and deactivating valve HP-26 by tagging open the valve breaker and tagging the valve handwheel in the closed position, shutting valve HP-410 and tagging the valve switch in the closed position.

      2) Deactivating all HPI pumps aligned to HPI train A and tagging the pump breakers open.

   b. Deactivating Train B of HPI is accomplished by either:

      1) Shutting and deactivating valve HP-27 by tagging open the valve breaker and tagging the valve handwheel in the closed position, shutting valve HP-409 and tagging the valve switch in the closed position.

      2) Deactivating all HPI pumps aligned to HPI train B and tagging the pump breakers open.

9. Pressurizer heater bank 3 or 4 must be deactivated.

Operational parameters identified in TS 3.4.12 and this TS Bases include allowances for instrument uncertainty.

APPLICABILITY

This LCO is applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, and in MODES 4, 5 and 6 when an RCS vent capable of mitigating the most limiting LTOP event is not open. The Applicability

BASES

APPLICABILITY (continued)

temperature of 325°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above 325°F. With the vessel head off, overpressurization is not possible. With an RCS vent capable of mitigating the most limiting LTOP event open, an LTOP event (including HPI-ES actuation or CFT discharge) is incapable of pressurizing the RCS above the RCS P/T limits.

A RCS vent ≥ 3.6 square inches is capable of mitigating a HPI-ES actuation of three pumps through two flow paths to the RCS. A RCS vent ≥ 201 square inches is capable of mitigating a discharge of both CFTs.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3 above 325°F.

The Applicability is modified by two Notes. Note 1 states that CFT isolation is only required when the CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This Note permits the CFT discharge valve surveillance performed only under these pressure and temperature conditions.

Note 2 permits the PORV to be inoperable when no HPI pumps are running and RCS pressure is < 100 psig. PORV operability is not required when RCS pressure is < 100 psig and HPI pumps are not operating since credible LTOP events progress relatively slowly, thus giving the operator ample time to respond.

ACTIONS

A.1

With the HPI activated, immediate actions are required to deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with one or more HPI pump OPERABLE is the event of greatest significance, since these events cause the greatest pressure increase in the shortest time.

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

B.1, C.1, and C.2

An unisolated CFT requires isolation within 1 hour only when the CFT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in LCO 3.4.3.

ACTIONS

B.1, C.1, and C.2 (continued)

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in 12 hours. By placing the unit in MODE 4 with the RCS temperature > 200°F, the CFT pressure of 650 psig cannot exceed the LTOP limits if both tanks are fully injected. Depressurizing the CFTs below the LTOP limit of 373 psig also prevents exceeding the LTOP limits in the same event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering judgement indicating that a limiting LTOP event is not likely in the allowed times.

D.1, E.1, and E.2

With the PORV inoperable, overpressure relieving capability is lost, and restoration of the PORV within 1 hour is required.

If restoration cannot be completed within 1 hour, either Required Action E.1 or Required Action E.2 must be performed. Required Action E.1 requires increasing RCS temperature within 23 hours to exit the Applicability of the specification. With RCS temperature > 325°F, the CFTs are not required to be isolated. Required Action E.2 requires the RCS be depressurized to less than 100 psig within 35 hours. With reactor pressure < 100 psig more time is available for operator action to mitigate an LTOP event.

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in these times.

F.1 and G.1

With Administrative Controls that assure ≥ 10 minutes are available to mitigate the consequences of an event not implemented, the capability for operator action to mitigate an LTOP event may be lost. In this circumstance, compensatory measures must be established to monitor for initiation of an LTOP event. Establishing a dedicated operator within 4 hours to monitor for initiation of an LTOP event is sufficient to compensate for inoperability of makeup flow restrictions, having too many pressurizer heater banks available, inoperability of required alarms, or deviation from pressure, temperature or level limits. Establishing a dedicated operator is not sufficient to compensate for not deactivating HPI or isolating CFTs. If the Required Action and associated Completion Time of Condition F is not met, the RCS must be depressurized and an

BASES

ACTIONS

F.1 and G.1 (continued)

RCS vent path capable of mitigating the most limiting LTOP event must be established within 12 hours. These Completion Times also consider that these activities can be accomplished in these time periods. A limiting LTOP event is not likely in these periods.

H.1 and H.2

With administrative controls which assure ≥ 10 minutes are available to mitigate the consequences of an LTOP event not implemented and the PORV inoperable; or the LTOP System inoperable for any reason other than cited in Condition A through G, the system must be restored to OPERABLE status within one hour. When this is not possible, Required Action H.2 requires the RCS depressurized and vented within 12 hours.

One or more vents may be used. A vent path capable of mitigating the most limiting LTOP event is specified. Because makeup may be required, the vent size accommodates inadvertent full makeup system operation. Such a vent keeps the pressure from full flow of the makeup pump(s) with a wide open makeup control valve within the LCO limit.

The Completion Time is based on operating experience that these activity can be accomplished in this time period and on engineering judgement indicating that a limiting LTOP transient is not likely in this time.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

Verifications must be performed that HPI is deactivated, and the CFTs are isolated. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP System. The Surveillances are required at 12 hour intervals. The 12 hour intervals are shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation within the safety analysis.

SR 3.4.12.3

Verification that the pressurizer level is less than the volume necessary to assure ≥ 10 minutes are available for operator action to mitigate an LTOP event by observing control room or other indications ensures a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.3 (continued)

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when the ends of these conditions are satisfied, as defined in plant procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.12.4

Verification that the PORV block valve is open ensures a flow path to the PORV. This is required at 12 hour intervals.

The interval has been shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation is within the safety analysis.

SR 3.4.12.5

A CHANNEL FUNCTIONAL TEST is required within 12 hours after decreasing RCS temperature to ≤ 325°F and every 31 days thereafter to ensure the setpoint is proper for using the PORV for LTOP. PORV actuation is not needed, as it could depressurize the RCS.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during the time. The 31 day Frequency is based on industry accepted practice and is acceptable by experience with equipment reliability.

SR 3.4.12.6

Verification that administrative controls, other than limits for pressurizer level, that assure ≥ 10 minutes are available for operator action to mitigate the consequences of an LTOP event are implemented is necessary every 12 hours. This verification consists of a combination of administrative checks for alarm availability, verification that pressurizer heater bank 3 or 4 is deactivated, appropriate restrictions on pressurizer level, controls for High Pressure Nitrogen, etc., as well as visual confirmation using available indications that associated physical parameters are within limits.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.6 (continued)

The Frequency is shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.12.7

The performance of a CHANNEL CALIBRATION is required every 18 months. The CHANNEL CALIBRATION for the LTOP setpoint ensures that the PORV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The Frequency considers a typical refueling cycle and industry accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, 5.2.3.7.
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. 10 CFR 50.36.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 307 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 307 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 307 TO FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated May 11, 1999, (Reference 1), as supplemented by letter dated July 13, 1999, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3 Technical Specifications (TS). The requested changes would incorporate revisions to the pressure-temperature (P/T) limits; the heatup, cooldown, and inservice test (IST) limits for the reactor coolant system (RCS) to a maximum of 33 Effective Full Power Years (EFPY); the low temperature overpressure protection (LTOP) system setpoints; and operational requirements for the reactor coolant pumps (RCPs). The supplement dated July 13, 1999, provided clarifying information that did not change the scope of the May 11, 1999, application and the initial proposed no significant hazards consideration determination.

A proposed change would extend the current P/T curves (Figures 3.4.3-1 through 3.4.3-9) to 33 EFPY, which is beyond the current Oconee license limit of 26 EFPY. A proposed change to TS 3.4.12, Low Temperature Overpressure Protection, would increase the power operated relief valve (PORV) setpoint from 460 pounds per square inch gage (psig) to 535 psig. The LTOP pressures, temperatures, and setpoints developed in this application are the same for all three units.

Table 3.4.3-1, "Operational Requirements for Unit Heatup," and Table 3.4.3-2, "Operational Requirements for Unit Cooldown," would be revised to allow two RCP operation in a single loop, rather than the present limit of one pump per loop during heatup and cooldown evolutions. Operation of two pumps will reduce the required net positive suction head for each pump, thereby reducing pump impeller wear due to cavitation, which has resulted in excessive impeller wear in the past.

The proposed changes include a new fluence determination based on the topical report BAW-2241P and the use of the American Society of Mechanical Engineers (ASME) Code Cases N-514, N-626, and N-588. An exemption for the application of these Code Cases was processed separately and issued by letter dated July 29, 1999 (Reference 7).

Associated Bases changes were also submitted.

2.0 BACKGROUND

2.1 Pressure -Temperature Limit Curves

The NRC has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P/T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision (Rev.) 1; GL 92-01, Rev. 1, Supplement 1; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P/T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff for inclusion in the staff's Reactor Vessel Information Database (RVID) as the basis for the staff's review of P/T limit curves and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR Part 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P/T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

The licensee's P/T limit curves and LTOP analysis satisfy the requirements of 10 CFR Part 50.60(a) with the additional provisions allowed by the following NRC-approved ASME code cases.

1. ASME Code Case N-626 (Now designated as Code Case N-640):

   Revised P/T limits have been developed using the K_{Ic} fracture toughness curve of ASME Section XI, Appendix A instead of the K_{Ia} curve of Appendix G as authorized and explained in Reference 7.

2. ASME Code Case N-588:

   The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P/T limit curves are the ¼ thickness (¼T) and ¾ thickness (¾T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively. However, if the flaw is postulated to be in a circumferential weld, it is physically unrealistic for the postulated flaw to be 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. It is unlikely that an axial flaw will extend from the circumferential weld into an adjacent plate or forging. In addition, due to the orientation of weld beads in a circumferential weld, the most likely orientation of the flaw is circumferential. Thus, for a circumferential weld, the postulated flaw should have a circumferential orientation. The approval and use of this case is further discussed in Reference 7.

3. Code Case N-514:

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. As part of these requirements, 10 CFR Part 50, Appendix G, requires that P/T limits be established for RPVs during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G, states: "The appropriate requirements on...the pressure-temperature limits and minimum permissible temperature must be met for all conditions." PWR licensees have installed cold overpressure mitigation systems (COMS)/LTOP in order to protect the RCPBs from being operated outside of the boundaries established by the P/T limit curves and to provide pressure relief of the RCPBs during low temperature overpressurization events. The staff has determined that the 110 percent of the Appendix G stress limit provisions of Code Case N-514 and the use of the K_{1c} fracture toughness curve permitted by Code Case N-640 (N-626) may not be applied simultaneously. In this submittal, Code Case N–514 was used for the determination of the LTOP enable temperature rather than the stress level. The LTOP enable temperature has been accepted in a staff position and is thus acceptable for use in conjunction with Code Case N-626 (N-640). This applies to the LTOP determination only and does not affect the determination of the licensee's P/T limit curves discussed in this safety evaluation, but is noted for the sake of completeness, since the licensee's submittal references the code case. The LTOP limits and the approval and use of Code Case N-514 are discussed in detail in Reference 7.

SRP 5.3.2 provides an acceptable method of determining the P/T Limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions and requires a safety factor of 1.5 during hydrostatic testing.

The ASME Code Appendix G methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

2.2 LTOP Changes

The LTOP system is designed to protect the pressure vessel boundary from low temperature over-pressurization by designating P/T limits that satisfy the requirements of the ASME Code, Section XI, Division 1, Code Case N-514. Code case N-514 specifies that "LTOP systems shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 °F, whichever is greater. LTOP systems shall limit the maximum pressure in the vessel to 110 percent of the pressure determined to satisfy Appendix G, paragraph G-2215 of Section XI, Division 1." Code Case N-514 has been approved for many plants including the Oconee Units. Code Case N-588 provides procedures for determining P/T limits derived from postulating a circumferential weld flaw rather than an axial flaw in the computation of the circumferential welds. In addition, a new computational procedure is incorporated. Code Case N-626 provides an alternate method for the computation of the fracture toughness of reactor vessel materials in determining the P/T limits. Code Case N-626 has been approved for use by ASME Section XI on September 1998. (As has been noted above, it has been renumbered and is now referred to as Code Case N-640).

2.3 Regulatory Requirements

For the protection of the RCS boundary, General Design Criteria (GDC) 14 and 31 are applicable. GDC 14 requires that the RCS boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. GDC 31 requires that sufficient margin be provided to assure that the reactor coolant pressure boundary behaves in a non-brittle manner under the stresses of normal operation, maintenance, test, and accident conditions, with a very low probability of rapidly propagating fracture.

Section 50.60 and Section 50.61 requires that licensees demonstrate that the effects of progressive embrittlement by neutron irradiation do not compromise the integrity of the reactor pressure vessel. To this end, two analyses are required: one to determine the P/T limits for normal heatup, criticality, cooldown, and inservice test operations; and another to assess the ability of the reactor vessel to maintain its integrity during an emergency shutdown with cold water injection (i.e., pressurized thermal shock (PTS)). 10 CFR 50.60 invokes Appendices G and H to 10 CFR Part 50, while 10 CFR 50.61 is the PTS rule, which requires a PTS assessment. PTS is not addressed in this evaluation, but has been reviewed by the staff.

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials within the reactor coolant boundary. It requires that the P/T limits for the RCS be at least as conservative as those obtained by the methodology specified in the 1989 edition of Appendix G to Section XI of the ASME Code. Alternatives to Appendix G may be used via an exemption, granted by the NRC. In this submittal, Code Cases N-514, N-626 and N-588 are used. Appendix H to 10 CFR Part 50 requires a reactor vessel materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region. These changes result from exposure of these materials to neutron irradiation and changes of the thermal environment. Material specimens exposed in the surveillance capsules are removed and tested at specified time intervals to monitor changes in the fracture toughness of the material.

2.4 RCP Operating Combination Change

During relatively low temperature RCS operation, the present TS require that no more than one RCP be operated per loop (the Oconee RCS design includes two RCS loops with two RCPs per loop). RCP operation at low pressure with either one pump in one RCS loop, or with one RCP in each RCS loop, results in gradual RCP impeller wear from cavitation. The degraded net positive suction head (NPSH) conditions are caused by the restricted P/T operating envelope at low pressures and temperatures.

3.0 EVALUATION

3.1 Heatup, Criticality, Cooldown and Inservice Test Limits P/T Curves - Licensee Evaluation

Oconee, Unit 1

According to the licensee, the projected 33 EFPY ART values at the 1/4T and 3/4T locations for the beltline regions were calculated by the licensee in accordance with RG 1.99, Rev. 2, and the guidelines presented by the NRC in the November 12, 1997, briefing concerning the review of responses to GL 92-01, Rev. 1, Supplement 1. The RG credibility criteria were applied to determine the appropriate margin, M, term. The licensee calculations determined the ART using RG 1.99, Rev. 2, Regulatory Positions 1.1 and 2.1. The licensee stated that the selected controlling values were those RV locations with the highest ART for 1/4T and 3/4T locations. The ART can be determined using RG 1.99, Rev. 2, Regulatory Position 1.1 by calculations using the Tables 1 and 2 values for CF or using Position 2.1 by calculations using surveillance data for the CF.

According to the tables titled, "Data for Preparation of Pressure-Temperature Limit Curves" in the submittal for Oconee, the licensee determined that the highest ART for the Unit 1 reactor vessel at the ¼T location is the circumferential weld (SA-1229) of the intermediate shell plate to the upper shell plate which was fabricated using weld wire heat 71249. The licensee calculated an ART of 203.1 °F, based on a neutron fluence of 5.22 X 10^{18} n/cm^2. The chemistry factor was 167.6 °F, which was determined using Table 1 (RG 1.99, Rev. 2, Position 1.1). The initial RT_{NDT} for the controlling weld material (SA-1229) was +10 °F. The margin term used in calculating the ART for the limiting weld was 56 °F.

The licensee's P/T data table indicated that for the ¾T location in Unit 1, the controlling ART is the circumferential weld of the intermediate shell plate to the upper shell plate (WF-25) which was fabricated using weld wire heat 299L44. The licensee calculated an ART of 188.0 °F at the ¾T location at 33 EFPY, and the chemistry factor used by the licensee was 223.7 °F, both of which were determined using Position 2.1 of RG 1.99, Rev. 2. The neutron fluence used in the ART calculation was 1.90 X 10^{18} n/cm^2. The licensee's initial RT_{NDT} for the limiting weld was the B&W generic value of -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F, consistent with RVID and RG 1.99, Rev. 2.

Oconee, Unit 2

For the Unit 2 reactor vessel, the licensee determined that the most limiting material at the ¼T and ¾T locations is the circumferential weld of the reactor vessel upper shell forging (WF-25) to the lower shell forging. This weld was fabricated using weld wire heat 299L44. The licensee calculated an ART of 248.4 °F at the ¼T location and 189.6 °F at the ¾T location at 33 EFPY. The neutron fluence used in the ART calculation was 5.38 X 10^{18} n/cm^2 at the ¼T location and 1.95 X 10^{18} n/cm^2 at the ¾T location. The chemistry factor used by the licensee was 223.7 °F, which was determined using Position 2.1 of RG 1.99, Rev. 2. The initial RT_{NDT} for the limiting weld used was the B&W generic value of -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F at the ¼T and ¾T locations, as calculated using RG 1.99, Rev. 2, Position 2.1, consistent with the RVID.

Oconee, Unit 3

For the Unit 3 reactor vessel, the licensee determined that the most limiting material at the ¼T and ¾T locations is the circumferential weld of the reactor vessel upper shell forging to the lower shell forging (WF-67). This weld was fabricated using weld wire heat 72442. The licensee calculated an ART of 211.7 °F at the ¼T location and 164.5 °F at the ¾T location at 33 EFPY. The neutron fluence used in the ART calculation was 5.32 X 10^{18} n/cm^2 at the ¼T location and 1.93 X 10^{18} n/cm^2 at the ¾T location at 33 EFPY. The chemistry factor was 180 °F, which was determined using Table 1 of RG 1.99, Rev. 2. The initial RT_{NDT} for the limiting weld was taken as the B&W generic value of -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F at the ¼T and ¾T locations, in accordance with RG 1.99, Rev. 2. All these values are consistent with RVID data.

3.2 Heatup, Criticality, Cooldown and Inservice Test Limits P/T Curves - Staff Evaluation

As stated above, the licensee submitted ART calculations and P/T limit curves, for Oconee, Units 1, 2, and 3, valid for 33 EFPY. The staff independently calculated the ARTs using the staff-reviewed and approved data and calculations found in the publicly available NRC data base, RVID. In addition, the staff independently generated P/T curves for normal operations and inservice hydrostatic testing conditions effective to 33 EFPY for each of the three Oconee units. Although the staff's calculations using the NRC-approved data and methodology differed in some instances from the licensee's, the licensee's curves were found to be conservative with respect to the staff determinations and are, therefore, acceptable. The details of this evaluation are provided below.

The ART is determined using the chemistry values of percent copper and percent nickel for each beltline material of Oconee, Units 1, 2, and 3. RVID contains chemistry values for each beltline material for all light water reactors in the U.S. The licensee's and the vendor's data were verified by the staff before incorporation in the RVID data base. Chemical composition, fluence, and initial RT_{NDT} values in RVID were updated to the data provided for the beltline materials of Oconee, Units 1, 2, and 3, in the letter dated February 2, 1999, from the B&WOG to the NRC that submitted report BAW-2325, Rev. 1, dated January 1999. It should be noted that the staff used the most recent updated chemistry data for the beltline materials in the Oconee, Units 1, 2, and 3, P/T limit evaluations. The staff compared the chemistry data in the licensee's submittal and found that the chemistry data in the licensee's May 11,1999, 33 EFPY, P/T submittal for the beltline materials of Oconee, Units 1, 2, and 3, were the same as those indicated in the BAW-2325, Rev. 1, report. The staff also found that the May 11, 1999, calculations proposed by the licensee in their submittal were at least as conservative as those values derived by the staff.

The NRC-verified data for the chemical compositions, initial RT_{NDT}, and margin values are available in RVID on the NRC INTERNET site (http:/www.nrc.gov) or by request from the NRC. The appropriate methodology and equations are in SRP 3.5.2, RG 1.99, Rev. 2, (also available from NRC). The code cases have the requisite instructions, equations, and curves to perform the calculations and are available from ASME. Therefore, the numerical values used by the staff in its calculations will not be repeated here. The fluences used were those reviewed and verified by the NRC (Reference 8) and also stated above in Section 3.1 of this evaluation.

Oconee, Unit 1, unlike Units 2 and 3, is not fabricated from ring forgings but has two longitudinal welds joining the beltline forgings. Therefore, the staff calculated the P/T curves based on longitudinal flaws in the longitudinal welds after checking to be sure the circumferential weld flaws would not be controlling. It was determined that the licensee's P/T curves were conservative, for both longitudinal (axial) and circumferential weld flaws with respect to the staff's calculations.

With respect to the ART calculations used for weld wire heat 299L44 for Units 1 and 2 (but not Unit 3, which does not contain this heat) the staff has determined that the surveillance data does not meet the staff's credibility criteria and was, therefore, not used in the RVID or in the staff's P/T calculations. Nevertheless, the licensee did use surveillance data and the methodology of RG 1.99, Rev. 2, Position 2.1, in its calculations for Units 1 and 2. The staff's calculated ART values for weld wire heat 299L44 used the CF values in Table 1 of RG 1.99, Rev. 2, Position 1.1 for Units 1 and 2. The licensee's calculated ARTs were as, or more, conservative than the staff's calculations and were, therefore, acceptable. For Unit 3, the licensee and staff used RG 1.99, Rev.1, Position 1.1 (Tables) for its calculations. Since both the staff's and licensee's ARTs were in agreement, the licensee's Unit 3 ART and curves are acceptable.

In summary, the staff evaluated each of the licensee's P/T limit curves for acceptability by performing check calculations using the methodology referenced in the Code (as indicated by SRP 5.2.3) and verified that the licensee's proposed P/T limits satisfy the requirements in Paragraph IV.2.b. of 10 CFR Part 50, Appendix G. The staff independently generated P/T curves for normal operations and hydrostatic test pressures effective to 33 EFPY for each of the three Oconee units. In comparing the curves generated by the staff to those generated by the licensee, the staff determined that the licensee's proposed P/T curves for Units 1, 2, and 3, meet the requirements of Appendix G of Section XI of the ASME Code as modified by the referenced code cases. Therefore, the licensee's curves meet the requirements of 10 CFR 50.60 and Appendix G.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the RPV based on the reference temperature for the flange material. Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange region that is highly stressed by the bolt preload must exceed the adjusted reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the RT_{NDT} of 60 °F for the limiting flange and upper shell materials as stated in RVID and also confirmed by the licensee, the staff has determined that the proposed P/T limits satisfy the requirement for the closure flange region during normal operation and hydrostatic pressure test and leak test for Oconee, Units 1, 2, and 3.

3.3 LTOP Changes

In the Oconee plants, over-pressure mitigation is accomplished using a combination of a pressurizer PORV and steam volume in the pressurizer (by limiting the pressurizer water level) and/or a RCS vent to depressurize the reactor. The system is manually enabled by the operator and uses a single setpoint as the lift pressure for the PORV. The design basis for the Oconee LTOP system considers the Adjusted-RT_{NDT} by estimating the pressure vessel fluence at the end of 33 EFPY. This allows the determination of the material properties, which in turn determines the pressure vs temperature behavior of the material. The maximum pressure is determined from a number of assumed transients, including mass and heat addition. The result of the transient analyses indicate that the mass addition transient is limiting, while the heat addition is self limiting below the P/T limits.

The staff has previously determined that the 110 percent of the Appendix G stress limit provision of Code Case N-514 and the K_{1c} feature of Code Case N-626 cannot be applied simultaneously. However, in the Oconee submittal Code Case N-514 is used for the determination of the enable temperature, rather than the stress level. This use has been accepted by the staff to determine the enable temperature and is, therefore, acceptable for use in conjunction with Code Case N-626.

3.4 Pressure Vessel Fluence

The Oconee units are part of a utility group with Babcock and Wilcox (B&W) designed and fabricated reactor vessels. This group applied for and was granted an exemption from the provisions of Appendix H to 10 CFR Part 50 to use an integrated surveillance program. This program was documented in the topical report BAW-1543A (Reference 3) and, among others, provided for a reactor cavity surveillance program to replace the in-vessel surveillance capsules that were relocated to two host plants (Crystal River and Davis Besse). This program was described in the topical report BAW-1875A (Reference 4) and received staff approval in June 1986. The dosimetry information collected from the integrated surveillance program (along with in-vessel dosimetry) was utilized in the validation of the methodology described in BAW-2241P, which has been reviewed by the staff and approved for use at Oconee. Issuance of the approved version of BAW-2214P is pending.

The projected 33 EFPY Adjusted-RT_{NDT} at the 1/4 T and 3/4T locations for the beltline regions were calculated. The Regulatory Positions 1.1 and 2.1 of Regulatory Guide 1.99, Rev. 2, were observed. From all of the beltline materials, the highest values were selected as the controlling material at each location. The fluence estimates in this submittal (to 33 EFPY) were calculated using the approved methodology in BAW-2241P and Regulatory Guide 1.99, Rev. 2 and, therefore, are acceptable.

3.5 Pressure -Temperature Limits

According to the licensee, the proposed P/T limits were developed using the computer code PTPC-3.3 (Reference 5) as modified for the application of Code Case N-518 for circumferential flaws in welds and by Code Case N-626 for use of the K_{1c} fracture toughness curve. The criteria employed to establish operating pressure and temperature limits are described in the staff approved Topical Report, BAW-10046A (Reference 6). The method used in determining the P/T limits includes the beltline region, the closure head, and the nozzle region for normal heat-up, cool-down, and in-service leak and hydrostatic tests.

Justification for the use of the above code cases and granting of the exemptions were approved by letter dated July 29, 1999 (Reference 7).

The design basis events are the following:

- Erroneous actuation of the high pressure injection (HPI) system. The Oconee TS currently require that both trains of the HPI be deactivated during LTOP. Analysis of this event was not performed because it is not considered credible.

- Erroneous opening of the core flood tank discharge valve. The current TS require that the core flood tanks be deactivated during LTOP conditions. Therefore, analysis of this event was not performed because it is not considered credible.

- Erroneous addition of nitrogen to the pressurizer. The high pressure nitrogen system is administratively controlled. Therefore, analysis of this event was not performed because is not considered credible.

- Makeup control valve failing full open. The maximum makeup flow is limited in this event to ensure that 10 minutes are available for operator action. The analysis distinguishes three regions with respect to pressure and temperature: (1) $T < 220$ °F and $P < 100$ psig, (2) $T < 220$ °F and 100 psig $< P < 375$ psig and (3) 220 °F $< T < 325$ °F. And $P < 525$ psig. In all three regions the PORV is assumed to be inoperable. With appropriate initial conditions for each region, the pressurizer level is determined so as to assure a 10-minute time window for operator mitigative action.

- Pressurizer heaters erroneously energized. The acceptance criterion is that 10 minutes be available for operator action before the pressure reaches 535 psig. The pressurizer PORV is assumed to be inoperable. Steam or a nitrogen bubble is assumed in the pressurizer. The pressurizer level is 80 inches, which becomes 100 inches assuming a 20-inch measurement uncertainty.

- Loss of decay heat removal system. Three cases are analyzed. The first case assumes a rapid cooldown and end-of-cycle decay heat, followed by failure of the decay heat removal system. The second case assumes that a pressurizer cooldown is in progress with pressure at or below 100 psig, pressurizer level at or below of 310 inches, and a HPI pump in operation. With these initial conditions, loss of the heat decay system is assumed to assure that 10 minutes are available for operator mitigative action. Finally, the third case evaluates a scenario where RCS fill/drain activities are under way with pressure at or below

100 psig pressurizer level at or below 380 inches. The analysis aims to verify that 10 minutes is available for mitigative operator action.

- RCP start induced transient. Two types of RCP-induced transients are evaluated. The first is filling of the once-through steam generator (OTSG) on the secondary side with hot water, followed by starting of the RCPs. The second transient is the restart of an RCP during heatup following a period of stagnant conditions. The results of the first transient indicate that the peak pressure is 505 psig, which is below the allowable reactor vessel pressure. For the second transient, the initial P = 450 psig and T = 275 degrees F are assumed. The resulting peak pressure is 600 psig, but the limiting pressure (for the assumed temperature) is 1050 psig.

The results of the above analysis indicate that there is a minimum of 10 minutes for operator action or that the maximum pressure does not reach the allowable limits.

Because the licensee is using Code Case N-626 in conjunction with ASME XI Appendix G, the P/T limits are based on 100 percent of the steady state (Appendix G) limits. The enable temperature is the greater of RT_{NDT} +50 °F or 200 °F. Unit 2 is the most limiting (has the highest enable temperature) and bounds Units 1 and 3. The most limiting Adjusted-RT_{NDT} is 248.4 °F. With the additional allowance of 50 °F plus 11.6 °F for instrument error plus 15 °F for margin, the LTOP enable temperature becomes 325 °F. (The corresponding values are 279.7 °F for Unit 1 and 288.3 °F for Unit 3). The 15 °F margin was conservatively added because the above limits were based on 100 percent of the Appendix G limits; therefore, there is no allowance for thermal gradient through the vessel thickness (although none is required). Because a steam or a nitrogen bubble is in the pressurizer whenever the vessel is pressurized, an LTOP transient is slowly changing temperatures and pressures, justifying the 100 percent limit.

Differential pressure corrections were applied to the P/T limits to account for the pressure differential between the analyzed regions and the system pressure sensor locations in the reactor vessel. These corrections are based on the RCP constrains as follows:

- Coolant Temperature T < 250 °F two pumps in one loop or one pump on both loops

- Coolant Temperature T > 250 °F two pumps on both loops.

The RCS pump operational constraints are considered in conjunction with the following linear heatup and cooldown rates:

<u>Heatup</u>

Between 60 °F and 280 °F, at 50 °F/hr

Between 280 °F and 570 °F, at 100 °F/hr

Cooldown

Between 570 °F and 280 °F at 50 °F steps with 30 minute hold periods or equivalent.

Between 280 °F and 150 °F at 25 °F steps with 30 minute hold periods or equivalent.

At 240 °F: Starting of the decay heat removal system is modeled as a step change from 240 °F to 207 °F and held for one minute at 207 °F, followed by a step increase to 227 °F. It is assumed that two RCPs in one loop are operating.

Between 150 °F and 60 °F at 10 °F steps with 60 minute hold periods or equivalent.

The maximum allowable pressure is taken to be the lowest of the calculated allowable pressures under transient and steady-state conditions. The collection (loci) of these points form the P/T limits. Pressure and temperature instrument errors are added in the operating procedures. Instrument uncertainty is calculated based on a licensee directive that complies with the intent of (the Instrument Society of America) ISA-67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The LTOP operating limits included in TS 3.4.12 and TS Bases 3.4.12 include an allowance for instrument error.

The pressurizer PORV (single) setpoint is set a 535 psig. This maximum allowable pressure bounds all other pressures over the LTOP temperature range. It includes a 20 psig difference from the 555 psig calculated for the lowest temperature of 60 °F. (This difference accounts for 12.9 psig instrument error and 7.1 psig margin.)

The PORV is a fast opening valve with approximately 0.2 second stroke time (after a 2.1 second time delay) and the transient pressure overshoot is negligible during LTOP events. Therefore, no special allowance is provided.

3.6 Administrative Controls

In the proceeding analysis some LTOP initiators were assumed as not credible; for others, a 10 minute time interval is available for the operator to take the appropriate mitigative action. To ensure that these assumptions are correct the following administrative controls were proposed by the licensee:

A. RCS Pressure:

P < 375 psig when T < 220 °F
P < 525 psig when 220 °F < T < 325 °F

B. Pressurizer Level:

P > 100 psig and T < 325 °F, then L < 220 inches
P < 100 psig and T < 220 °F, then L < 310 inches
P < 100 psig and T < 160 °F, then L < 380 inches while filling or draining the RCS and the HPI pumps running.

C. Make-up flow is restricted with the make-up control valve (HP-120) to < 98 gallons per minute (gpm) for all three units.

D. The high pressure nitrogen system is administratively controlled to prevent inadvertent pressurization of the RCS.

E. Three audible pressurizer level alarms are set at 225, 260, and 315 inches.

F. Two audible RCS pressure alarms are set at 375 and 525 psig.

G. The core flood tanks must be deactivated.

H. The HPI safety injection flowpaths must be deactivated.

I. The pressurizer heater banks 3 and 4 must be deactivated.

The above limitations correspond to the analysis assumptions.

3.7 RCP Operating Combination Change

The limits on allowable operating RCP combinations control the pressure differential between the reactor vessel wall and the pressure measurement point and are used as inputs for calculating the heatup, cooldown, and the leak rate and hydro test limit curves. For example, with one RCP operating in a loop, the pressure differential between the low range pressure transmitter tap and the actual pressure at the vessel beltline is approximately 20 psi. With two RCPs in the same loop operating this differential pressure is approximately 50 psi. The differential pressure created by the operation of two RCPs must be accounted for in the development of the P/T limit curves so the upper pressure limit is not allowed to be exceeded.

Limits on the number of allowable operating RCP(s) at low temperature became necessary as the pressure limits at low temperatures decreased. The pressure limits deceased due to both the effects of ongoing neutron exposure to reactor vessel materials and the conservative methodology then needed to assure that the P/T limit curves provided adequate protection from reactor vessel brittle fracture. As the pressure limits decreased, the higher pressure differential of the two operating RCPs in a loop resulted in an ever shrinking operating P/T window. The number of operating RCPs is currently limited to increase the size of the operating P/T window.

As a result of the change in the P/T limits described in this submittal and safety evaluation, the licensee has proposed increased pressure limits at low temperatures. The licensee has determined that this increase in the pressure limit restores sufficient pressure margin to accommodate operation of two RCPs in a loop or one RCP in each loop at low temperatures as shown in the proposed changes to TS Tables 3.4.3-1 and 3.4.3-2. The staff has reviewed this information and found it to be acceptable.

4.0 SUMMARY

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality, for Oconee, Units 1, 2, and 3, satisfy the requirements in Appendix G to Section XI of the ASME Code and Appendix G of 10 CFR Part 50 for 33 EFPY. The proposed P/T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Oconee, Units 1, 2, and 3, TS as proposed by the licensee.

The staff has reviewed the proposed LTOP changes to TS 3.4.3 and the associated bases and determined that the proposed revisions satisfy the Appendix G (to 10 CFR Part 50) requirements as modified by the ASME Code Cases N-514, N-588 and N-626, for which exemption requests have been approved. The proposed modification extends the period of the LTOP applicability to 33 EFPY. These changes are acceptable.

The LTOP enable temperature and the P/T curves, are the same for all three units and are based on the Unit 2 circumferential weld WE-25, which is the critical weld for each of the units. We find that the estimation of the LTOP enable temperature, the PORV actuation pressure, the P/T curves, and the associated pressurizer level were performed in a manner consistent with the approved methodologies. Therefore, the results are acceptable, and we conclude that the proposed modification of TS 3.4.3 and the associated bases are acceptable.

Based on the acceptability of these changes, the staff has found the changes to the number of operating RCPs per loop to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 32289). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from W. R. McCollum, Jr., Duke Energy Corporation to USNRC "Proposed Revision to Technical Specifications Pressure-Temperature Operating Curves Technical Specification Change No. 99-02," dated May 11, 1999.

2. BAW-2241P, "Fluence and Uncertainties Methodologies," B&W Owners Group, dated May 14, 1997.

3. BAW-1543A, Rev. 2, "Integrated Reactor Vessel Material Surveillance Program," A.L. Lowe, Jr., et al, B&W Nuclear Division, May 1985.

4. BAW-1875A, "The B&WOG Cavity Dosimetry Program," S. Q. King, B&W Nuclear Division, August 1985.

5. FTI Document 32-1171775-05, "Verification of PTPC & User's Manual," by J. W. Moore the III, March, 1994.

6. BAW-10046A, Rev. 2, "Methods of Copliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," by H.W. Behnke, et. al. B&W Nuclear Technologies, Lynchburg, VA, June, 1976.

7. Letter from D. E. LaBarge, NRC, to W. R. McCollum, Jr., Duke Energy Corporation, "Specific Exemptions to Section 50.12, Part 50 of Title 10, Code of Federal Regulations Concerning ASME Code Cases N-588 and N-626," dated July 29,1999.

8. Memorandum to D. E. LaBarge, Project Manager, NRC, from Lambros Lois, Sr. Nuclear Engineer, Reactor Systems Branch, NRC, Subject: Oconee Units 1, 2, 3 License Amendment to Change the Heatup, Cooldown and LTOP Limits - Code Case N-514, "Low Temperature Overpressure Protection," Section XI of the ASME Boiler and Pressure Vessel Code, dated July 1, 1999.

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