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September 15, 2005

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

Subject: Oconee Nuclear Station, Units 1, 2, and 3
Dockets Nos. 50-269,270, and 287
License Amendment Request: Adoption of GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" and TSTF-419-A, Rev. 0, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR"
Technical Specification Change Number 2005-04

Pursuant to 10 CFR 50.90, Duke Energy Corporation (Duke) hereby requests an amendment to its Renewed Facility Operating Licenses DPR-38, DPR-47 and DPR-55 for Oconee Nuclear Station (ONS), Units 1, 2,and 3, respectively. The proposed change will create a Pressure and Temperature Limits Report (PTLR) based on the guidance provided in Generic Letter (GL) 96-03 "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" and TSTF-419-A, Rev. 0, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR." The proposed change relocates the Pressure Temperature Limit (PTL) Curves of Technical Specification (TS) 3.4.3 to the Selected Licensee Commitments (SLC) Manual and adds TS section 5.6.9 to reflect the GL requirements for the relocation.

This change is consistent with GL 96-03 and TSTF 419-A, Rev. 0 that was approved by the NRC on March 21, 2002. The required NRC approved methodology is contained in the Babcock and Wilcox Owners Group Topical Report, BAW-10046A, Rev 2, dated June 1986 and approved by the NRC on April 30, 1986. Amendment Nos. 307, 307, and 307 dated October 1, 1999, extended the PTL curves to

A001

33 EFPY based on the methodology in BAW-10046A, Rev 2, and alternative rules provided by ASME Code Case N-514, N-588 for circumferential flaws, and N-626 (Now designated N-640) for K_{1c} fracture toughness.

Attachment 1 provides the re-typed TS pages. Attachment 2 provides a mark-up of the affected TS pages. Attachment 3 provides the proposed PTLR. The technical justification for the amendment request is included in Attachment 4. Attachments 5 and 6 contain the No Significant Hazards Consideration Evaluation and the Environmental Impact Analysis, respectively.

The proposed change to the TS has been reviewed and approved by the Plant Operations Review Committee and Nuclear Safety Review Board.

No new commitments are being made as a result of this request. NRC approval of this LAR is requested by October 15, 2005 to support Unit 2's shutdown for refueling. An implementation period of 7 days is requested.

The following Utilities have submitted and gained approval for similar submittals: FENOC-Beaver Valley in an SER dated July 15, 2003; Exelon Generation Company-Byron and Braidwood in as SER dated January 23, 1998; Southern Company-Vogtle in an SER dated March 28, 2005; PGE-Diablo Canyon in an SER dated May 13, 2004; and OPPD-Fort Calhoun in an SER dated August 15, 2003.

Implementation of these changes will not result in an undue risk to the health and safety of the public.

The Oconee Updated Final Safety Analysis Report has been reviewed and no changes are necessary to support this LAR.

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the South Carolina Department of Health and Environmental Control for review, and as deemed necessary and appropriate, subsequent consultation with the NRC staff.

If there are any additional questions, please contact Reene' Gambrell at (864) 885-3364.

Very truly yours,

R.A. Jones, Vice President
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Ron Jones, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Corporation, that he is authorized on the part of said company to sign and file with the U.S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, DPR-55; and that all the statements and matters set forth herein are true and correct to the best of his knowledge.

R.A. Jones, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 15th day of Sept., 2005

Notary Public

My Commission Expires:

9-20-2009

ATTACHMENT 1
REVISED TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
1.1-4	1.1-4
1.1-5	1.1-5
3.4.3-1	3.4.3-1
3.4.3-2	3.4.3-2
3.4.3-3	-------
3.4.3-4	-------
3.4.3-5	-------
3.4.3-6	-------
3.4.3-7	-------
3.4.3-8	-------
3.4.3-9	-------
3.4.3-10	-------
3.4.3-11	-------
3.4.3-12	-------
3.4.3-13	-------
3.4.12-1	3.4.12-1
3.4.12-2	3.4.12-2
5.0-38	5.0-38
---------	5.0-39
B 3.4.3-1	B 3.4.3-1
B 3.4.3-2	B 3.4.3-2
B 3.4.3-3	B 3.4.3-3
B 3.4.3-4	B 3.4.3-4
B 3.4.3-5	B 3.4.3-5
B 3.4.3-6	B 3.4.3-6
B 3.4.3-7	B 3.4.3-7
B 3.4.3-8	---------
B 3.4.12-1	B 3.4.12-1
B 3.4.12-8	B 3.4.12-8

1.1 Definitions (continued)

OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.9.
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1\right)$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2568 MWt.

1.1 Definitions

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and

c. There is no change in APSR position.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates and allowable RC pump combinations shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE------------ Required Action A.2 shall be completed whenever this Condition is entered. ------------------------------ Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours 36 hours
C. -----------NOTE------------ Required Action C.2 shall be completed whenever this Condition is entered. ------------------------------ Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limit. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------------NOTE------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS leak and hydrostatic testing. --- Verify RCS pressure, RCS temperature and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

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 the PTLR.

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 in the PTLR.	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)

5.6 Reporting Requirements (continued)

5.6.8 Steam Generator Tube Inspection Report

The steam generator tube inspection report shall comply with the following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification to the NRC shall be reported prior to resumption of plant operation. The written report shall provide the results of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The designation of affected and unaffected areas will be reported to the NRC when they are determined.

5.6.9 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

5.6 Reporting Requirements (continued)

5.6.9 Reactor Coolant System (RCS) PTLR (continued)

1. BAW-10046, Rev. 2, "B & W Owners Group Materials Committee Methods Of Compliance With Fracture Toughness And Operational Requirements of 10 CFR 50, Appendix G."
2. BAW-1543A, Rev. 2, "Integrated Reactor Vessel Material Surveillance Program."
3. BAW-1875, "The B&WOG Cavity Dosimetry Program."
4. BAW-2241P, "Fluence and Uncertainty Methodologies."
5. ASME Code Case N-514, "Low Temperature Overpressure Protection, Section XI, Division 1."
6. ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1."
7. ASME Code Case N-626, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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.

The PTLR contains P/T limit curves for heatup, cooldown, leak and hydrostatic (LH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines 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. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 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. 3).

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. 4).

BASES

BACKGROUND (continued)

Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with ASTM E 185 (Ref. 5), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 3.

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. 6). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

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. 3) 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 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. 7) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. 8).

LCO

The three elements of this LCO are:

a. The limit curves for heatup, cooldown, and LH testing and

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

c. Allowable RC pump combinations.

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.

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.

BASES (continued)

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. 2). 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.

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. 7) 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.

BASES

ACTIONS

A.1 and A.2 (continued)

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.

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.

BASES

ACTIONS

C.1 and C.2 (continued)

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. 7), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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 the PTLR 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. BAW-10046A, Rev. 2, June, 1986.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

BASES

REFERENCES (continued)

4. Regulatory Guide 1.99, Revision 2, May 1988.
5. ASTM E 185-82, July 1982.
6. 10 CFR 50, Appendix H.
7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
8. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for providing such protection. The PTLR provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less ductile at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and may be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or related anticipated transients must be controlled to not violate the PTLR. Exceeding these limits could lead to brittle fracture of the reactor vessel. The PTLR presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity and a limit on coolant addition capability. The pressure relief capacity requires the power operated relief valve (PORV) lift setpoint to be reduced and administrative controls implemented which assure ≥ 10 minutes available for operator action to mitigate an LTOP event. The administrative controls include limits on pressurizer level, limits on RCS pressure when RCS temperature is < 325°F, limits on RCS makeup flow, the number of available pressurizer heater banks, requirements for alarms and restrictions upon use of the High Pressure Nitrogen System.

The LTOP approach to protecting the vessel by limiting coolant addition capability requires controls upon RCS makeup flow, the number of available pressurizer heater banks, and requires deactivating HPI, and isolating the core flood tanks (CFTs).

Should one or more HPI pumps inject on an HPI actuation (HPI-ES) or a CFT discharge to the RCS, the pressurizer level and PORV may not prevent overpressurizing the RCS.

BASES

APPLICABILITY (continued)

temperature of 325°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet the PTLR 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.

The PTLR 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 the PTLR. 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 the PTLR.

ATTACHMENT 2

MARKUP OF TECHNICAL SPECIFICATIONS

1.1 Definitions (continued)

OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1\right)$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2568 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.9.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3

RCS pressure, ~~and~~ RCS temperature, ~~shall be within the limits specified in Figures 3.4.3-1 and 3.4.3-2 for Unit 1; Figures 3.4.3-4 and 3.4.3-5 for Unit 2; and Figures 3.4.3-7~~ and ~~3.4.3-8 for Unit 3.~~ RCS heatup and cooldown rates and allowable RC pump combinations shall be maintained within the limits specified in ~~Tables 3.4.3-1 and 3.4.3-2.~~ the PTLR.

~~NOTES~~

1. ~~For leak tests of the RCS, leak tests of connected systems required by Tech Specification 5.5.3 where RCS allowable combinations of temperature and pressure are controlling the RCS may be pressurized to within the limits of Figure 3.4.3-3 for Unit 1, Figure 3.4.3-6 for Unit 2 and Figure 3.4.3-9 for Unit 3.~~

2. ~~For thermal steady state system hydro tests required by ASME Section XI the RCS may be pressurized to within the limits of Tech Specification 2.1.2 and Figure 3.4.3-3 for Unit 1, Figure 3.4.3-6 for Unit 2 and Figure 3.4.3-9 for Unit 3.~~

3. ~~Not applicable to the pressurizer.~~

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ——NOTE—— Required Action A.2 shall be completed whenever this Condition is entered. ————— Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours

(continued)

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

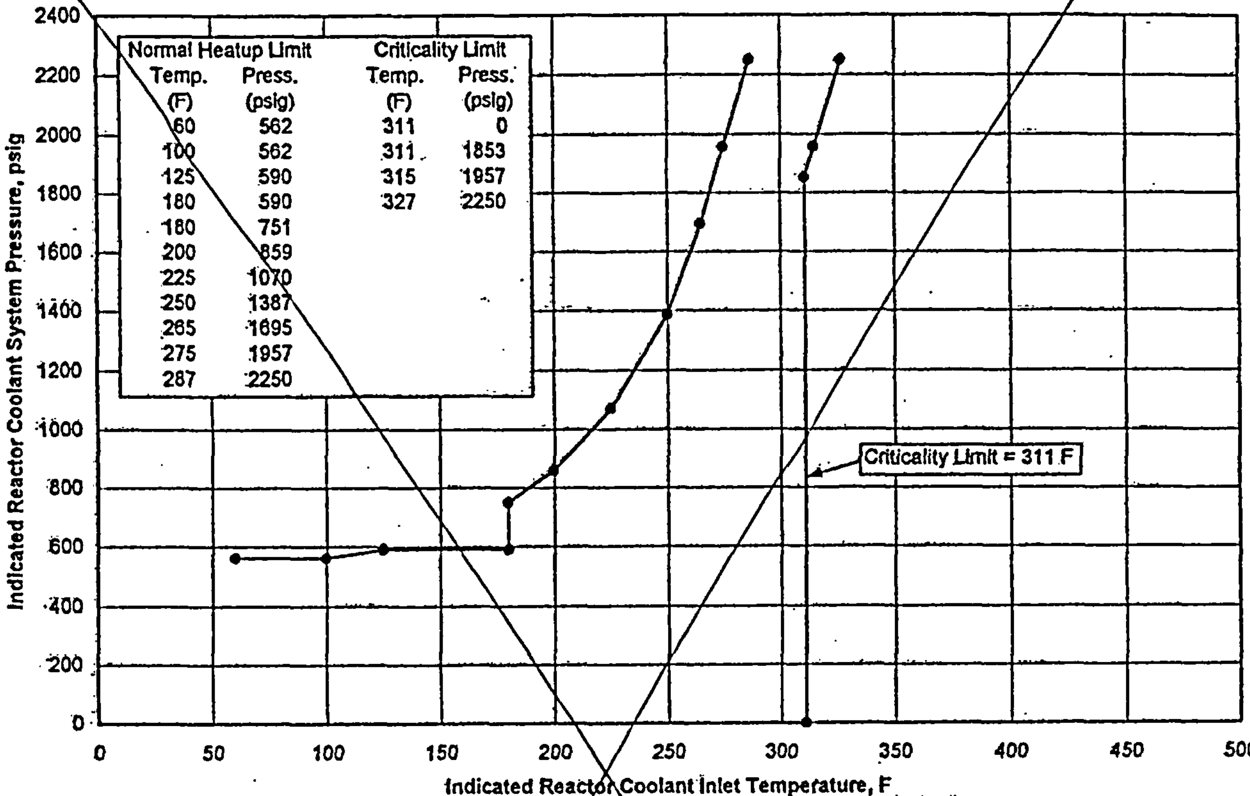
(a) RC Temperature is cold leg temperature if one or more RC pumps are in operation or if on natural circulation cooldown; otherwise it is the LPI cooler outlet temperature.

(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,

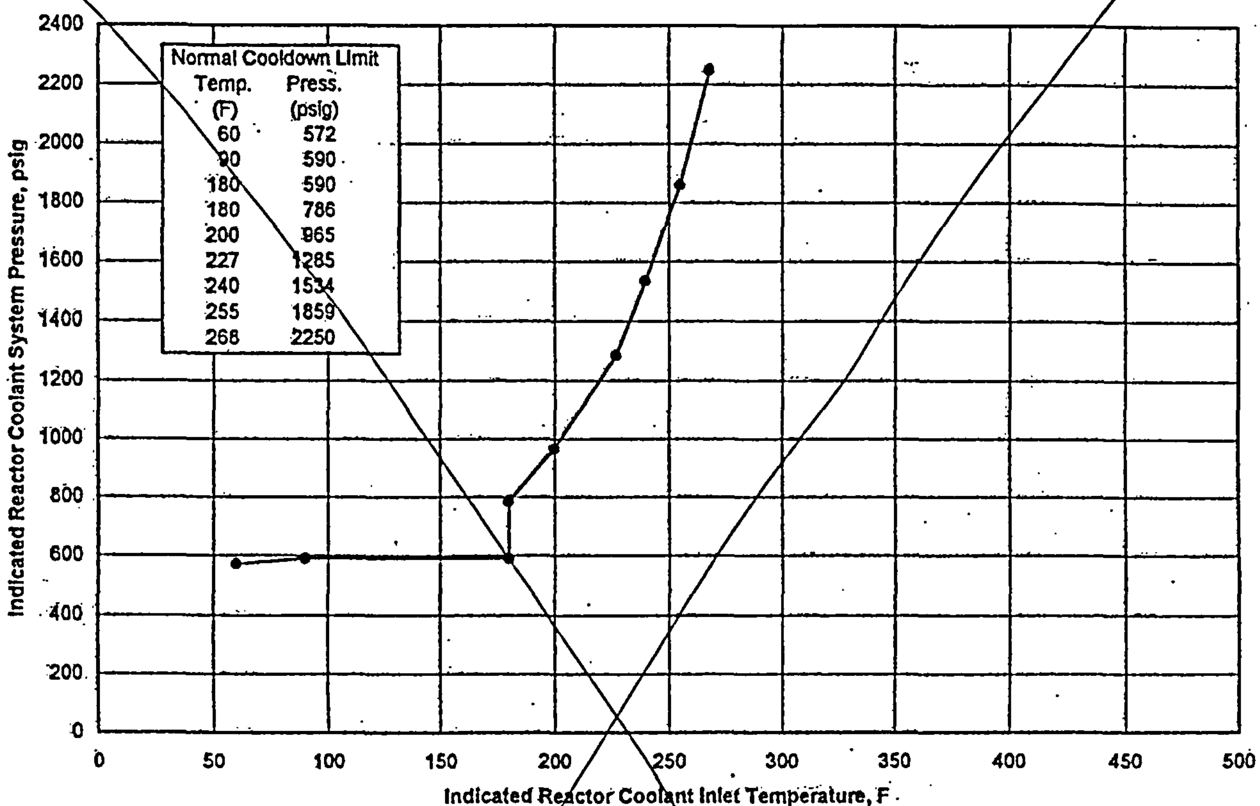
the maximum cooldown rate shall be relaxed to ≤ 50°F in any 1 hour period.



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.

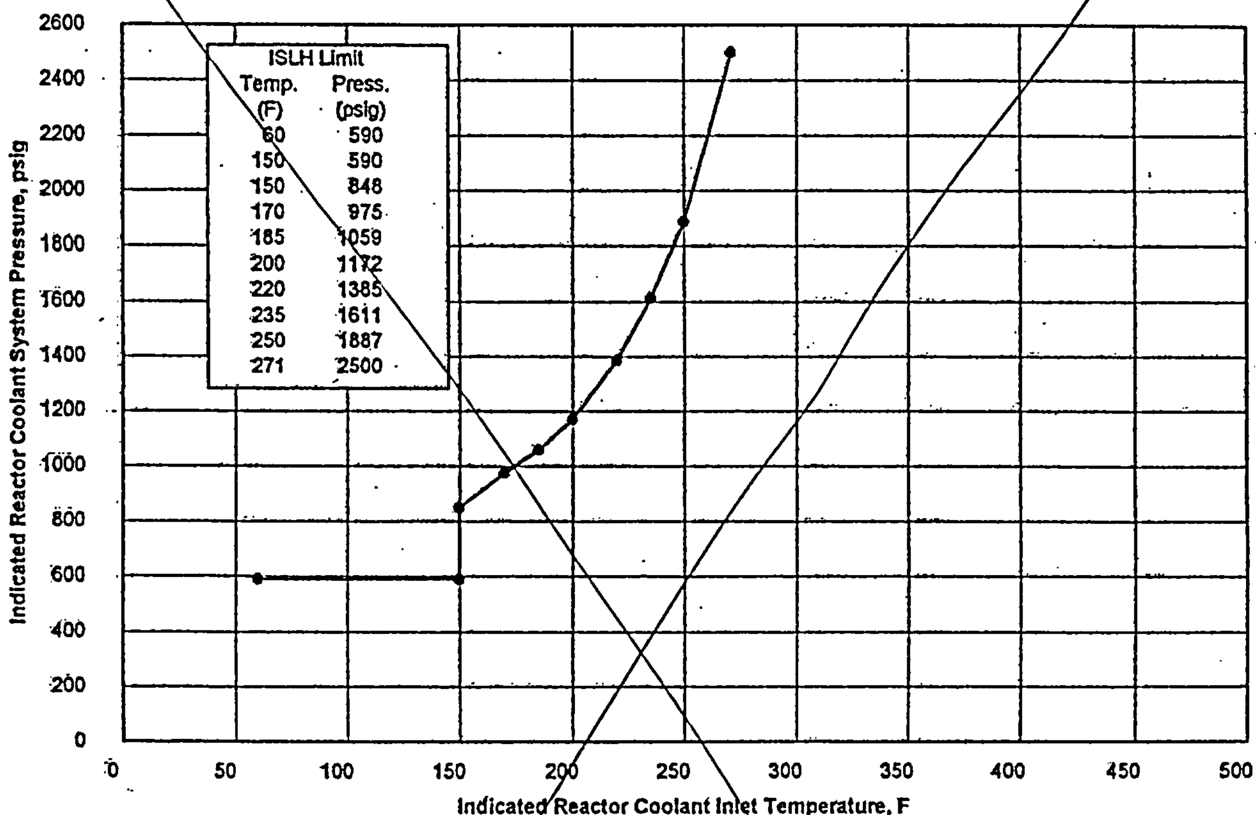
Figure 3.4.3-1 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1



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.

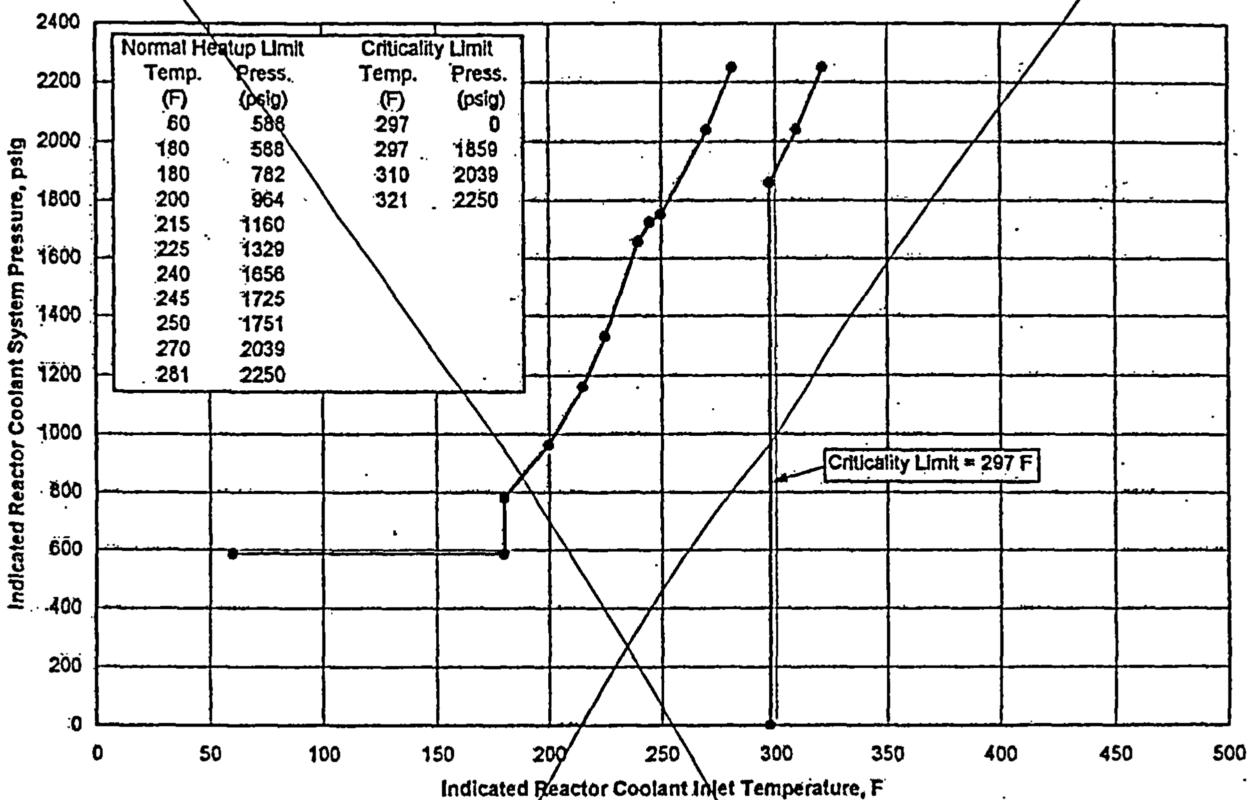
Figure 3.4.3-2 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1



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

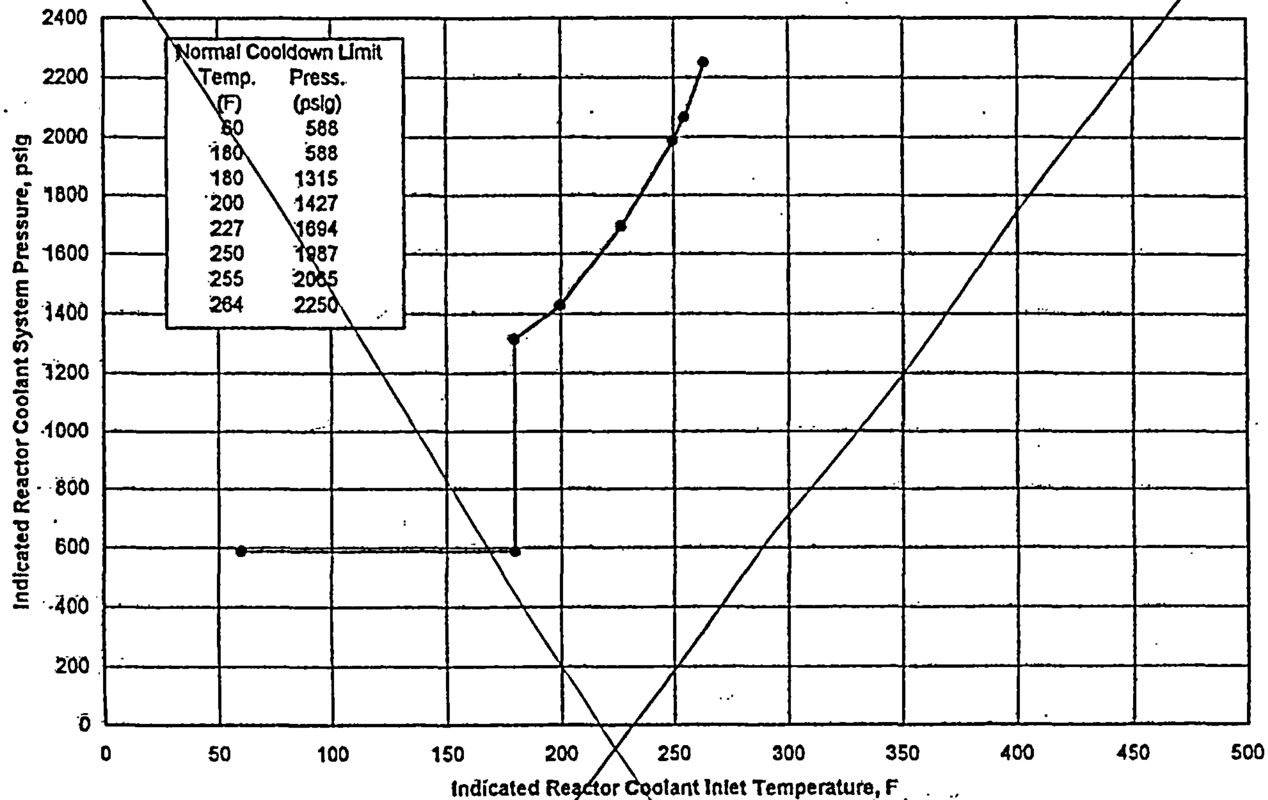


Normal Heatup Limit Temp. (F)	Normal Heatup Limit Press. (psig)	Criticality Limit Temp. (F)	Criticality Limit Press. (psig)
60	588	297	0
180	588	297	1859
180	782	310	2039
200	964	321	2250
215	1160		
225	1329		
240	1656		
245	1725		
250	1751		
270	2039		
281	2250		

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. <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.

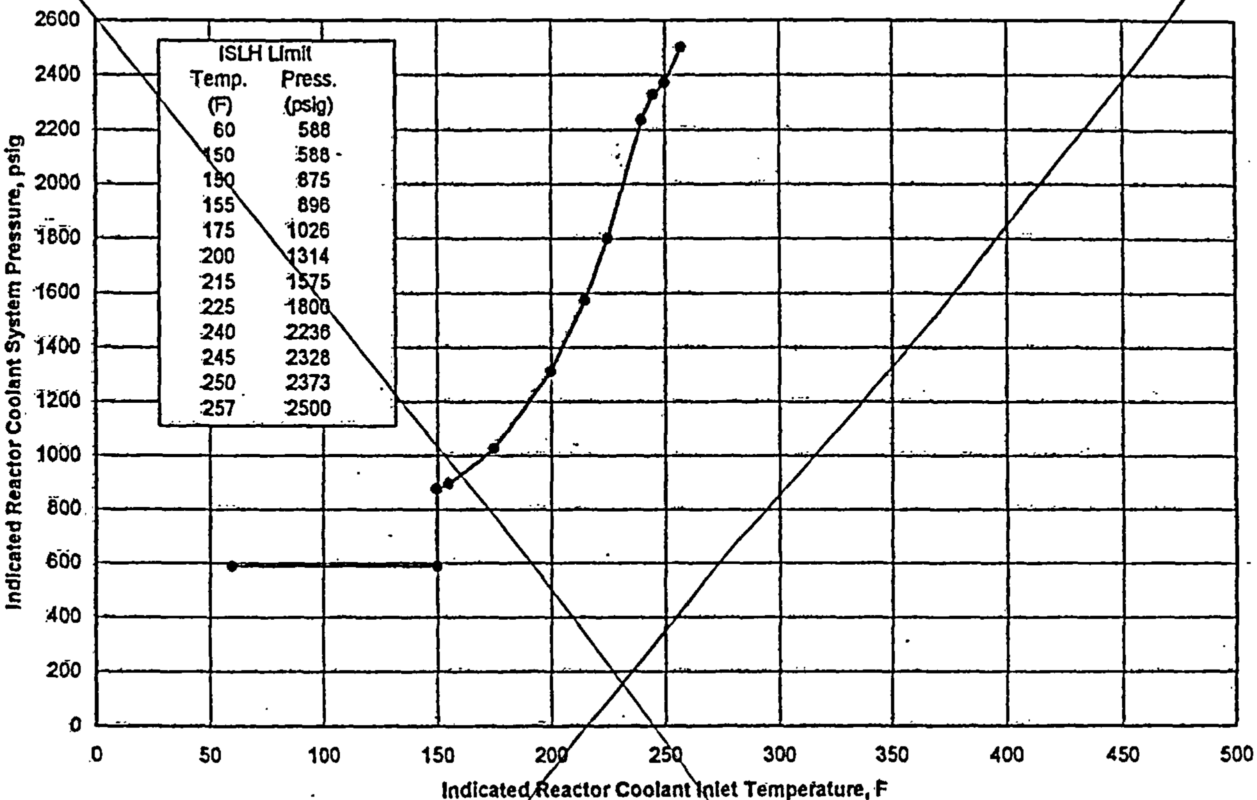
Figure 3.4.3-4 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



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. <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.

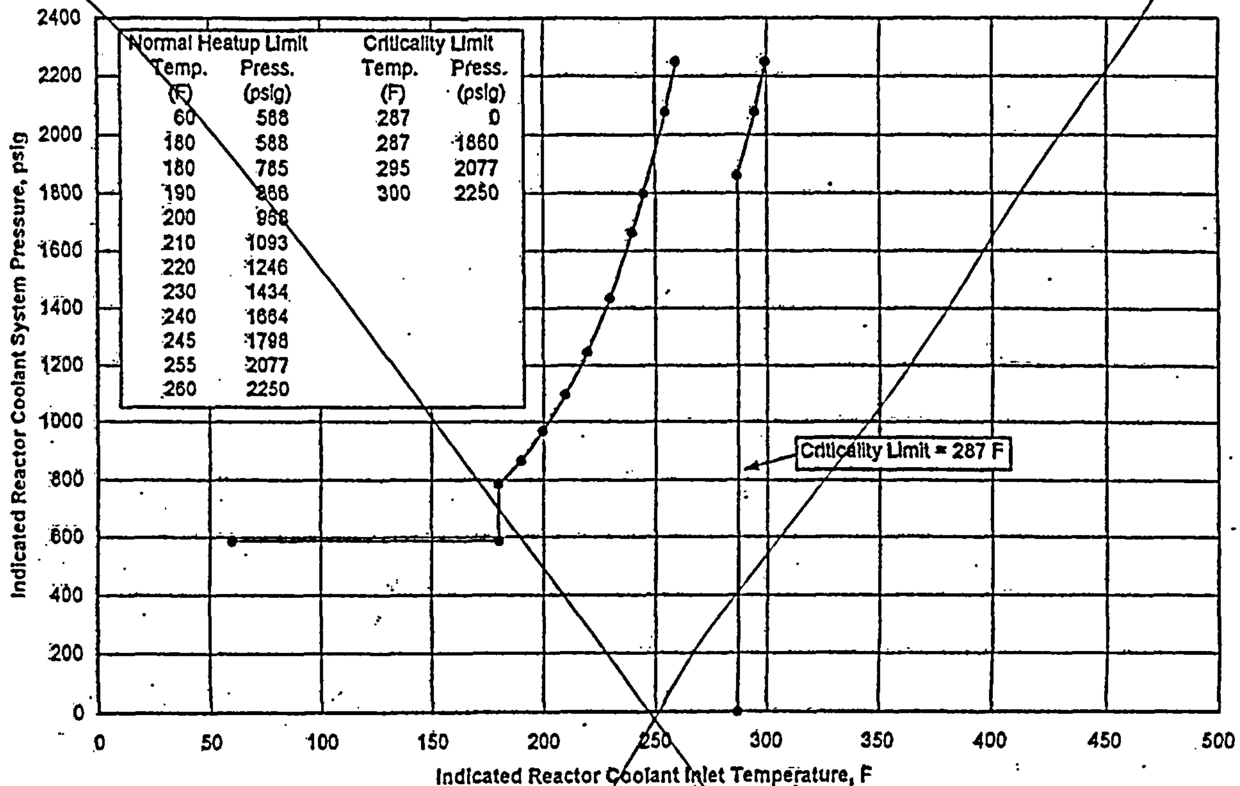
Figure 3.4.3-5 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



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

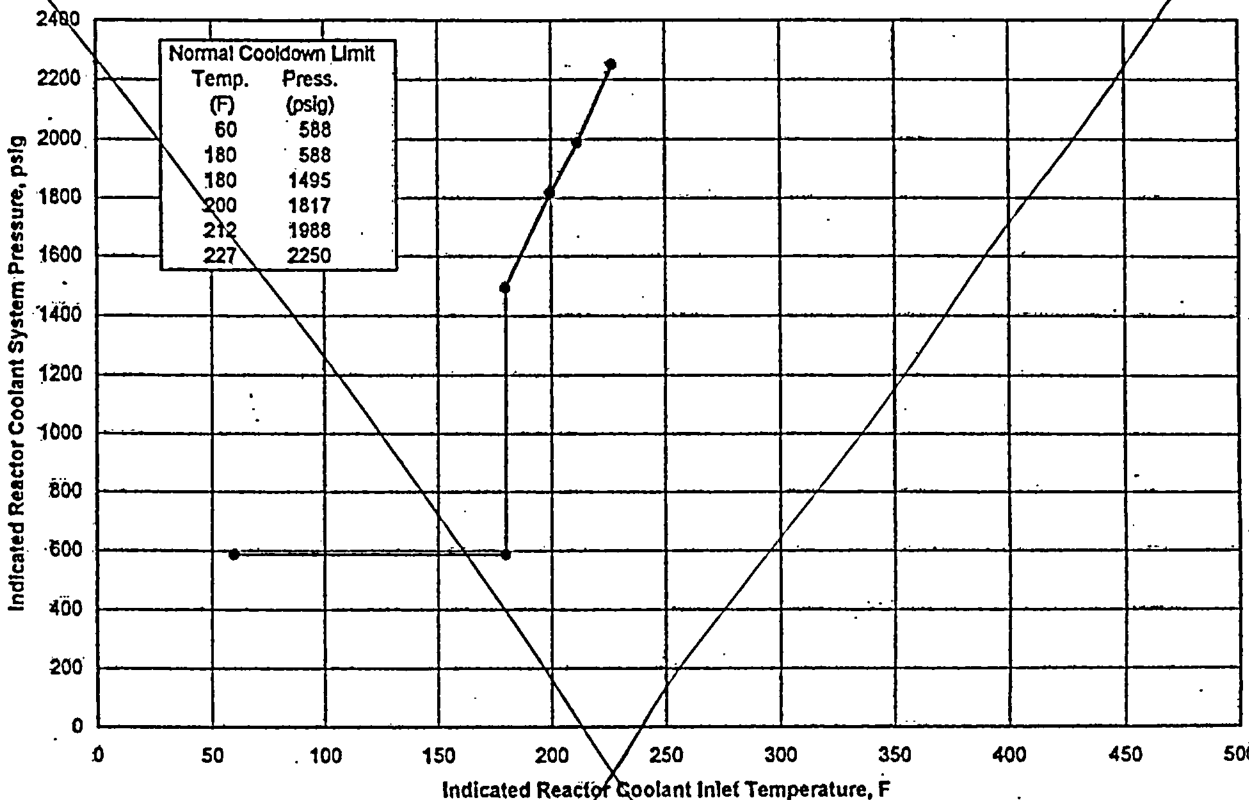


Normal Heatup Limit Temp. (F)	Normal Heatup Limit Press. (psig)	Criticality Limit Temp. (F)	Criticality Limit Press. (psig)
60	588	287	0
180	588	287	1860
180	785	295	2077
190	866	300	2250
200	968		
210	1093		
220	1246		
230	1434		
240	1664		
245	1798		
255	2077		
260	2250		

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. <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.

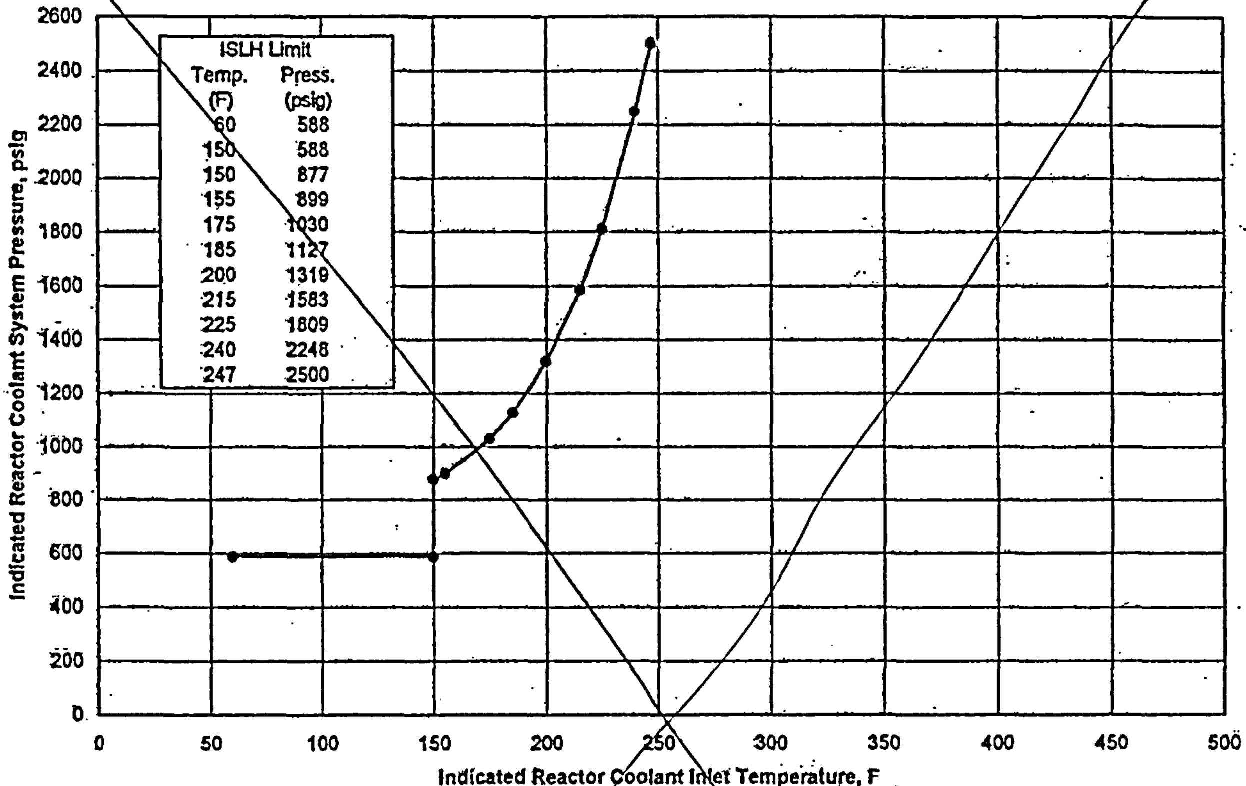
Figure 3.4.3-7 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 3



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. <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 3.4.3-8 (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.~~ the PTLR.

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~~. in the PTLR	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)

5.6 Reporting Requirements (continued)

5.6.8 Steam Generator Tube Inspection Report

The steam generator tube inspection report shall comply with the following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification to the NRC shall be reported prior to resumption of plant operation. The written report shall provide the results of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The designation of affected and unaffected areas will be reported to the NRC when they are determined.

Add to Section 5.0

5.6.9 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

5.6 Reporting Requirements (continued)

5.6.9 Reactor Coolant System (RCS) PTLR (continued)

Add to Sec. 5.0

1. BAW-10046, Rev. 2, "B & W Owners Group Materials Committee Methods Of Compliance With Fracture Toughness And Operational Requirements of 10 CFR 50, Appendix G."

2. BAW-1543A, Rev. 2, "Integrated Reactor Vessel Material Surveillance Program."

3. BAW-1875, "The B&WOG Cavity Dosimetry Program."

4. BAW-2241P, "Fluence and Uncertainty Methodologies."

5. ASME Code Case N-514, "Low Temperature Overpressure Protection, Section XI, Division 1."

6. ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1."

7. ASME Code Case N-626, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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.

The PTLR ~~Figures 3.4.3-1 through 3.4.3-9~~ contains 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 (Ref 1). ~~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. There is no minimum allowable temperature limit for the reactor vessel if all of the studs are fully detensioned.~~

Each P/T limit curve ~~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~~ defines 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~~ 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference ~~1~~ 2 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~~ 3).

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~~ 4).

BASES

BACKGROUND (continued)

Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with ASTM E 185 (Ref. ~~4~~ 5), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference ~~2~~ 3.

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~~ 6). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference ~~2~~ 3.

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~~ 3) 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.

~~As stated in the tables associated with this LCO, reactor coolant (RC) temperature is cold leg temperature if one or more RC pumps are in operation; otherwise, it is the LPI cooler outlet temperature. An analysis examined the effects of initiating flow through a previously idle LPI train (i.e. either placing a train of LPI in operation or swapping from one train to the other) when none of the RC pumps are operating. The analysis assumed the initial temperature of the fluid entering the vessel to be the lowest expected temperature in an idle LPI cooler. As RC fluid is pumped through the system and returns to the reactor vessel, the temperature increases to a "stable" value. The duration of the temperature excursion is dependent on LPI flow and volume of the piping system. This analysis has determined that the brief temperature excursion caused by the fluid initially in the idle LPI train can be accommodated if, at the time the LPI header is put in service, the RCS pressure is less than 295 psig (Instrument Uncertainty Adjusted). This value is less limiting than the~~

BASES

BACKGROUND (continued)

~~LPI initiation pressure limit imposed by procedures to protect the LPI system from overpressure. The brief temperature excursion does not place the reactor vessel outside of the bounds of the stress analyses. (Ref. 8, FTI Doc. 32-5010572-00 Allowable LPI Pressures For LPI Cooler Swap).~~

~~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. 7 ~~6~~) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

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. 8).

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~~

BASES

LCO (continued)

~~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.~~

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.

~~The LPI cooler outlet temperature during the brief period of stabilization does not need to be considered when determining heatup or cooldown rates or RCS P/T conditions when an LPI train is placed in operation with no operating RCPs. The period of stabilization is the time required to fully displace the stagnant fluid in the idle LPI train. The time required for stabilization is a function of LPI flow rate. Operating procedures control both placing a train of LPI in service and swapping trains of LPI to limit the duration of the temperature transient to a value that has been shown to be acceptable. (Ref. 8, FTI Doc. 32-5010572-00 Allowable LPI Pressures For LPI Cooler Swap).~~

BASES

LCO (continued)

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~~ 2). 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.

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

BASES

ACTIONS

A.1 and A.2 (continued)

completed, documented, and approved in accordance with established plant procedures and administrative controls.

ASME Code, Section XI, Appendix E (Ref. ~~8~~ 7) 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.

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

BASES

ACTIONS

B.1 and B.2 (continued)

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~~ 7), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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 the PTLR 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. < IP
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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 (continued)

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

REFERENCES

1. BAW-10046A, Rev 2 June 1986

2 ~~1.~~ 10 CFR 50, Appendix G.

3 ~~2.~~ ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

4 ~~3.~~ Regulatory Guide 1.99, Revision 2, May 1988.

5 ~~4.~~ ASTM E 185-82, July 1982.

6 ~~5.~~ 10 CFR 50, Appendix H.

7 ~~6.~~ ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

8 ~~7.~~ 10 CFR 50.36.

~~8. FTI Doc. 32-5010572-00, Allowable LPI Pressures For LPI Cooler Swap.~~

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for providing such protection. ~~LCO 3.4.3,~~ The PTLR ~~"RCS Pressure and Temperature (P/T) Limits,"~~ provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less ductile at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and may be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or related anticipated transients must be controlled to not violate ~~LCO 3.4.3.~~ the PTLR Exceeding these limits could lead to brittle fracture of the reactor vessel. ~~LCO 3.4.3~~ presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity and a limit on coolant addition capability. The pressure relief capacity requires the power operated relief valve (PORV) lift setpoint to be reduced and administrative controls implemented which assure ≥ 10 minutes available for operator action to mitigate an LTOP event. The administrative controls include limits on pressurizer level, limits on RCS pressure when RCS temperature is < 325°F, limits on RCS makeup flow, the number of available pressurizer heater banks, requirements for alarms and restrictions upon use of the High Pressure Nitrogen System.

The LTOP approach to protecting the vessel by limiting coolant addition capability requires controls upon RCS makeup flow, the number of available pressurizer heater banks, and requires deactivating HPI, and isolating the core flood tanks (CFTs).

Should one or more HPI pumps inject on an HPI actuation (HPI-ES) or a CFT discharge to the RCS, the pressurizer level and PORV may not prevent overpressurizing the RCS.

BASES

APPLICABILITY (continued)

temperature of 325°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet PTLR ~~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.

The PTLR ~~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~~ the PTLR. 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.~~ the PTLR.

ATTACHMENT 3

PROPOSED PRESSURE TEMPERATURE LIMIT REPORT

16.5 REACTOR COOLANT SYSTEM (RCS)

16.5.4. Pressure and Temperature Limits Report (PTLR)

COMMITMENT

RCS pressure and temperature shall be within the limits specified in Figures 16.5.4-1 and 16.5.4-2 for Unit 1; Figures 16.5.4-4 and 16.5.4-5 for Unit 2; and Figures 16.5.4-7 and 16.5.4-8 for Unit 3. RCS heatup and cooldown rates and allowable RC pump combinations shall be maintained within the limits specified in Tables 16.5.4-1 and 16.5.4-2.

---------------------------------NOTES---------------------------------

1. For leak tests of the RCS, leak tests of connected systems required by Technical Specification 5.5.3 where RCS allowable combinations of temperature and pressure are controlling the RCS may be pressurized to within the limits of Figure 16.5.4-3 for Unit 1, Figure 16.5.4-6 for Unit 2 and Figure 16.5.4-9 for Unit 3.

2. For thermal steady state system hydro tests required by ASME Section XI the RCS may be pressurized to within the limits of Technical Specification 2.1.2 and Figure 16.5.4-3 for Unit 1, Figure 16.5.4-6 for Unit 2 and Figure 16.5.4-9 for Unit 3.

3. Not applicable to the pressurizer.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Commitment not met.	A.1 Enter applicable TS Condition for P/T Limits not met.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	N/A	N/A

BASES

1.0 Neutron Fluences

The calculational methodology for predicting the fluence using the cavity dosimetry was validated in the benchmark phase of the cavity dosimetry program. The benchmark consisted of both surveillance capsule and cavity dosimetry comparisons of calculations to dosimetry measurements. The results of these benchmarks are documented in FTI Topical Report BAW-2241P. This report was approved by the Staff in February, 1999. The results demonstrate the conclusions of the Oconee Units 1, 2, and 3 fluence data used to develop the PTL are sufficient for safety and licensing evaluations of reactor vessel embrittlement. The fluences of the predominate limiting materials and locations are included in the table below. The complete set of Fluence Values are provided in the FTI Topical Report.

FLUENCES AT 33 EFPY

PLANT	Location of Limiting Beltline Weld	Fluence, n/cm^2
ONS-1	1/4t Limiting Weld Location (SA-1229)	0.522×10^{19}
	¾ t Limiting Weld Location (WF-25)	0.190×10^{19}
ONS-2	1/4t Limiting Weld Location (WF-25)	0.538×10^{19}
	¾ t Limiting Weld Location (WF-25)	0.195×10^{19}
ONS-3	1/4t Limiting Weld Location (WF-67)	0.532×10^{19}
	¾ t Limiting Weld Location (WF-67)	0.193×10^{19}

2.0 Reactor Vessel Material Surveillance

In 1976, several utilities with reactor vessels designed by Babcock & Wilcox (B&W), including Duke Power, requested exemptions from the 10 CFR 50, Appendix H requirement for an in-vessel material surveillance program. The Staff reviewed and evaluated each licensee's request for an exemption and the plan for an integrated surveillance program. The staff then granted the requested exemption.

A revised 10 CFR 50, Appendix H became effective in July 1983. Section II. C of the revised Appendix H allows an integrated surveillance program provided it was approved by the Director, Office of Nuclear Reactor Regulation. The revised Appendix H provided criteria that were to be used in the evaluation of the surveillance program. In a letter dated March 14, 1984, the B&W Owners Group submitted an updated integrated surveillance program for Staff review and approval. The program was documented in BAW-1543A, Revision 2, Integrated Reactor Vessel Surveillance Program. In the Safety Evaluation Report (SER) for BAW-1543A, the Staff concluded the topical report meets the evaluation criteria of Section II. C of Appendix H. The following was concluded and contained in the SER: In-cavity dosimetry testing should continue in order to reduce uncertainties in neutron fluence for vessels that do not contain in-vessel dosimetry. If these test results provide an effective method of monitoring vessel neutron fluence, the in-cavity dosimetry should be incorporated in plants.

In a letter dated September 16, 1985, the B&W Owners Group requested an evaluation of a "Cavity Dosimetry Program" under development for use in B&W plants. This program is described in Topical Report BAW-1875. This report was approved by the Staff in June 1986. It was noted in BAW-1875 that the material surveillance program will have provided all the required empirical information for the fluence-toughness relationship by the mid-1990's. During this time, a portion of the surveillance capsules would have been removed. The cavity dosimetry program will then continue to provide vessel irradiation data beyond the end of the integrated surveillance capsule dosimetry program in an accurate and convenient manner.

3.0 Low Temperature Overpressure Protection System Limits (LTOP).

The actual temperature at which the pressure in the P/T limit curves falls below the pressurizer safety valve setpoint increases as vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP system will be re-evaluated to ensure that its functional requirements can still be met with the PORV and pressurizer level/administrative controls method.

4.0 Adjusted Reference Temperature (ART) for each Reactor Beltline Material

The limit curves are based on the predicted value of the ART of the limiting reactor vessel materials at the end of 33 EFPY. The ART's are calculated by adding a radiation-induces delta RTndt to the initial RTndt plus a margin term, using Regulatory Guide 1.99 Revision 2 to predict the radiation-induced delta RTndt values as a function of the material's copper and nickel content and neutron fluence. The values below summarize the predicted reactor vessel fluence values at the ¼ t and ¾ t vessel wall locations of the predominate limiting weld materials for the Oconee Units. A value of 60 °F is assumed for the RTndt of the closure head region and the outlet nozzle steel forging, in accordance with BAW-10046A.

ART's at 33EFPY

PLANT	Location of Limiting Beltline Weld	Fluence, n/cm^2	ART, °F
ONS-1	1/4t Limiting Weld Location (SA-1229)	0.522×10^{19}	203.1
	¾ t Limiting Weld Location (WF-25)	0.190×10^{19}	188.0
ONS-2	1/4t Limiting Weld Location (WF-25)	0.538×10^{19}	248.4
	¾ t Limiting Weld Location (WF-25)	0.195×10^{19}	189.6
ONS-3	1/4t Limiting Weld Location (WF-67)	0.532×10^{19}	211.7
	¾ t Limiting Weld Location (WF-67)	0.193×10^{19}	164.5

5.0 P/T Curves for heatup, cooldown, criticality, and hydrostatic and leak tests.

The P/T curves are contained in figures 16.5.4-1 through 16.5.4-9.

6.0 Minimum Temperatures

The minimum boltup temperature for the Reactor Vessel Flange shall be $\geq 60^{\circ}F$. Boltup is a condition in which the reactor vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

The requirements for minimum hydrotest temperature are contained in the RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations curves.

7.0 Chemistry 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.

Table 16.5.4-1 (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 16.5.4-2 (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

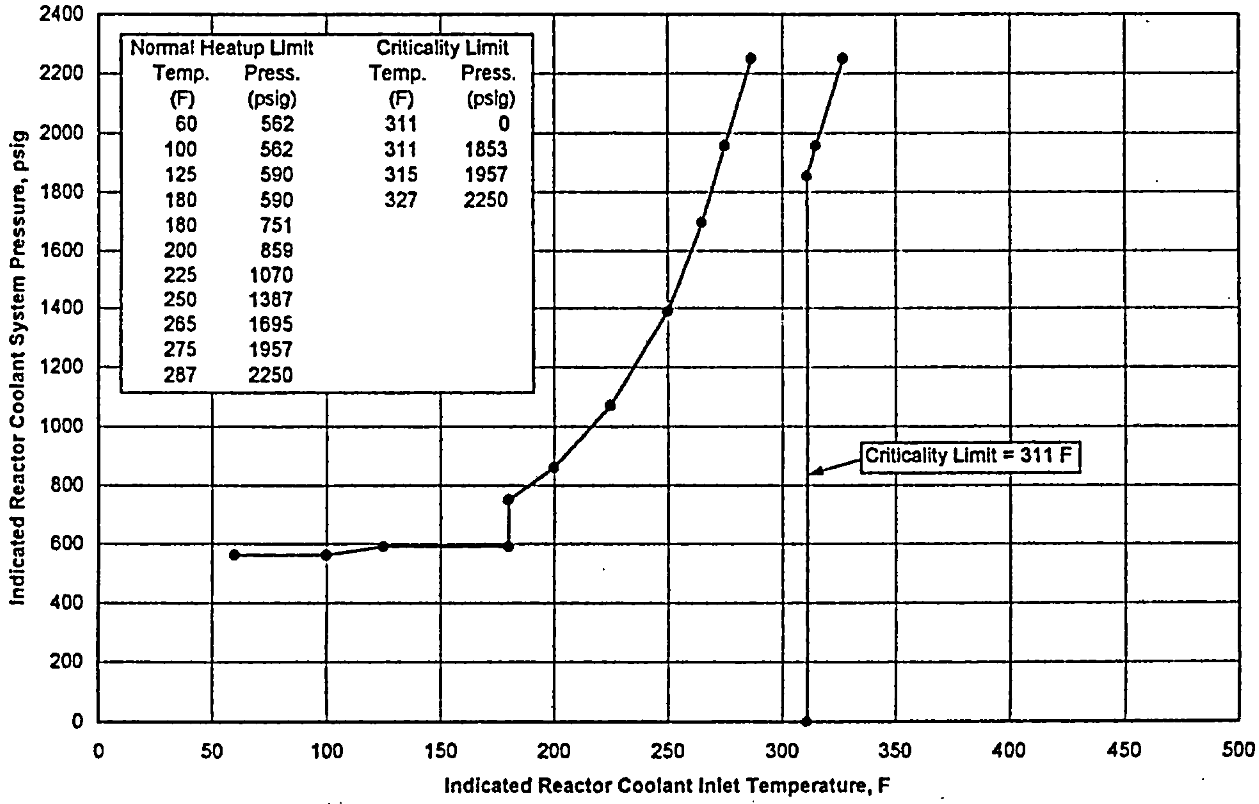
(a) RC Temperature is cold leg temperature if one or more RC pumps are in operation or if on natural circulation cooldown; otherwise it is the LPI cooler outlet temperature.

(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,

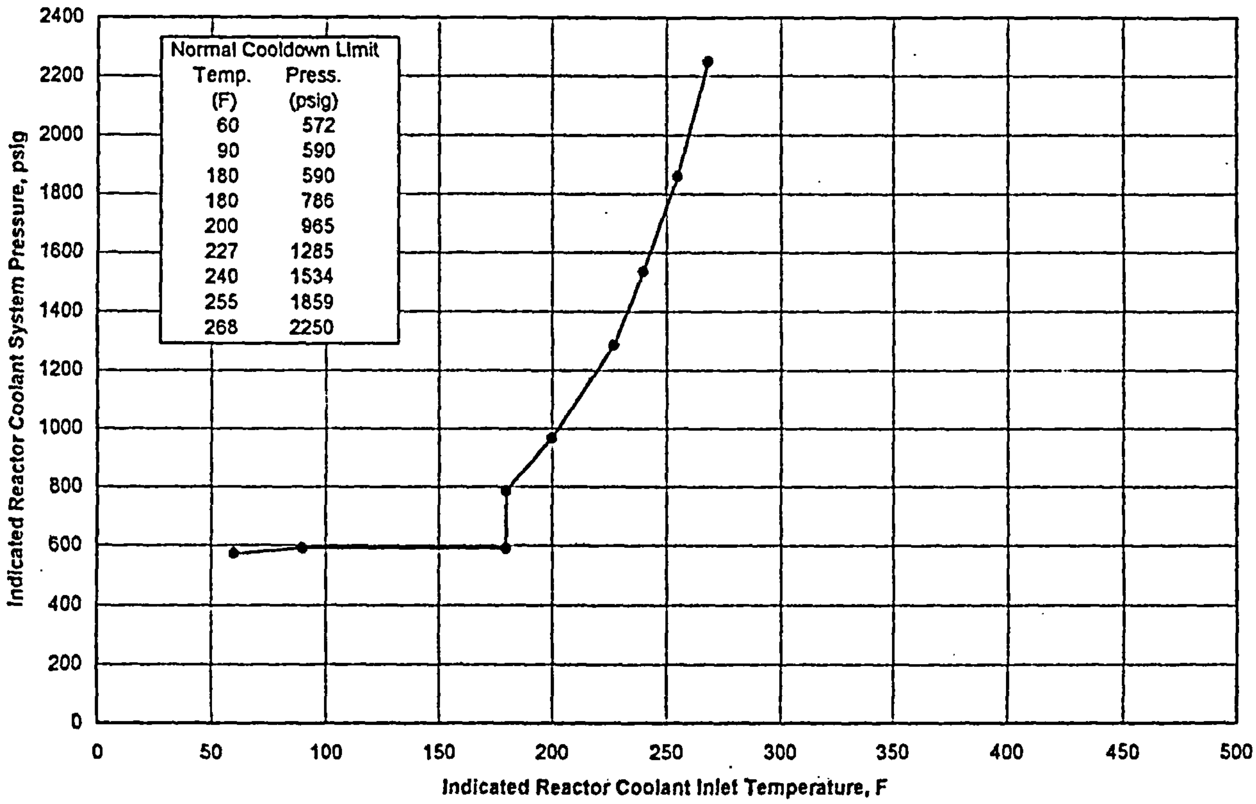
the maximum cooldown rate shall be relaxed to ≤ 50°F in any 1 hour period.



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.

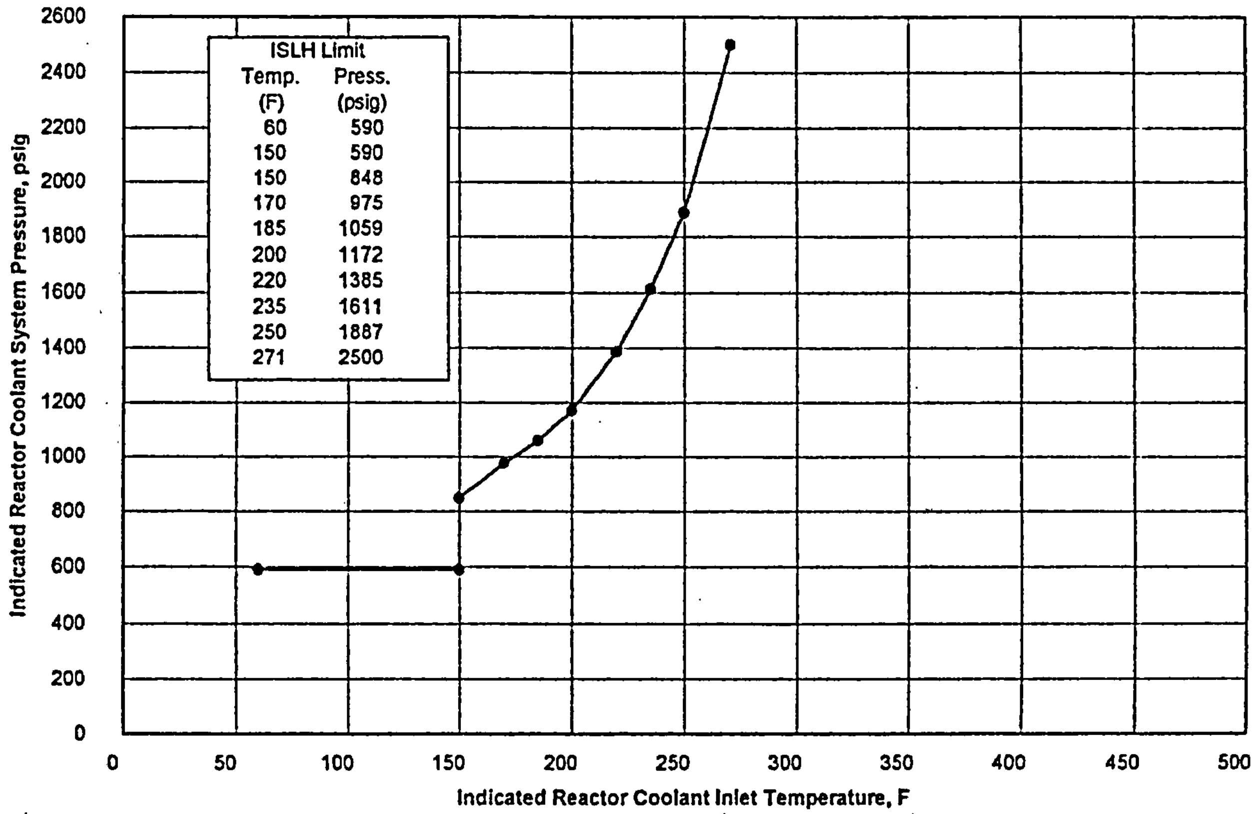
Figure 16.5.4-1 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1



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.

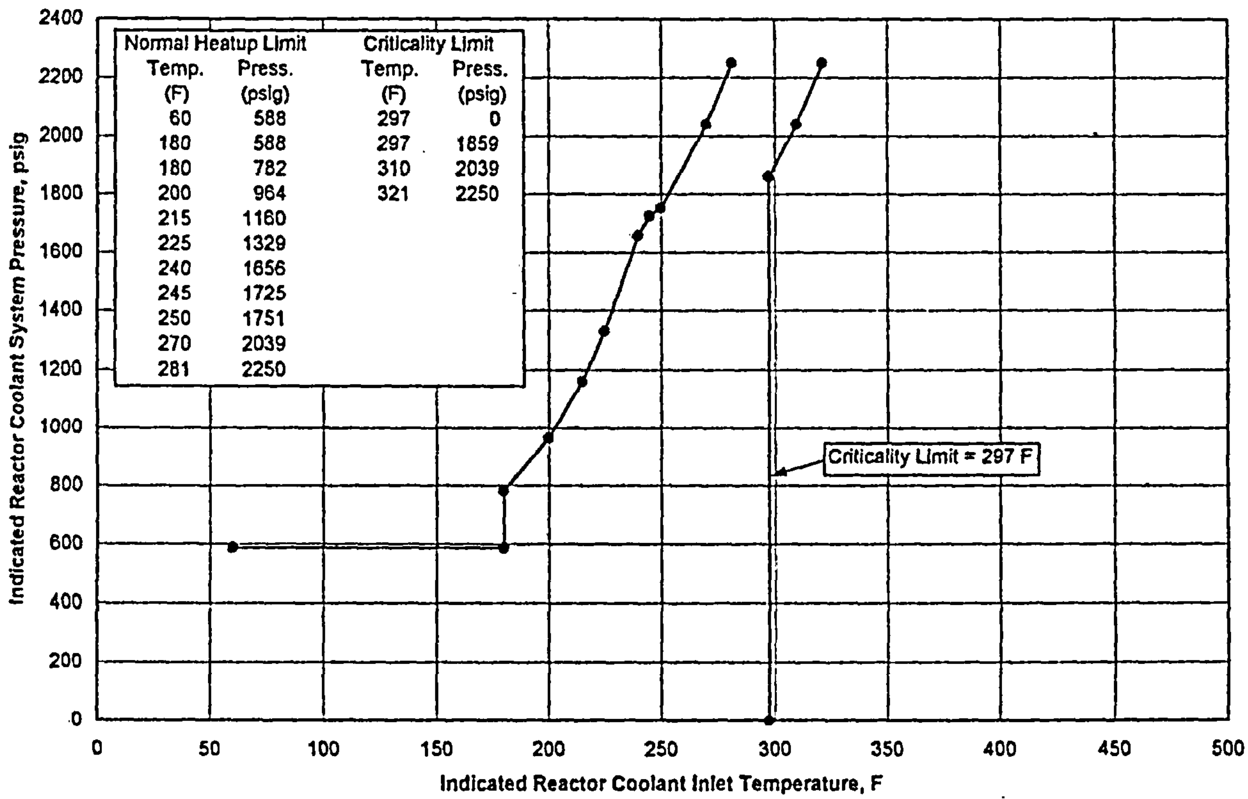
Figure 16.5.4-2 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1



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.

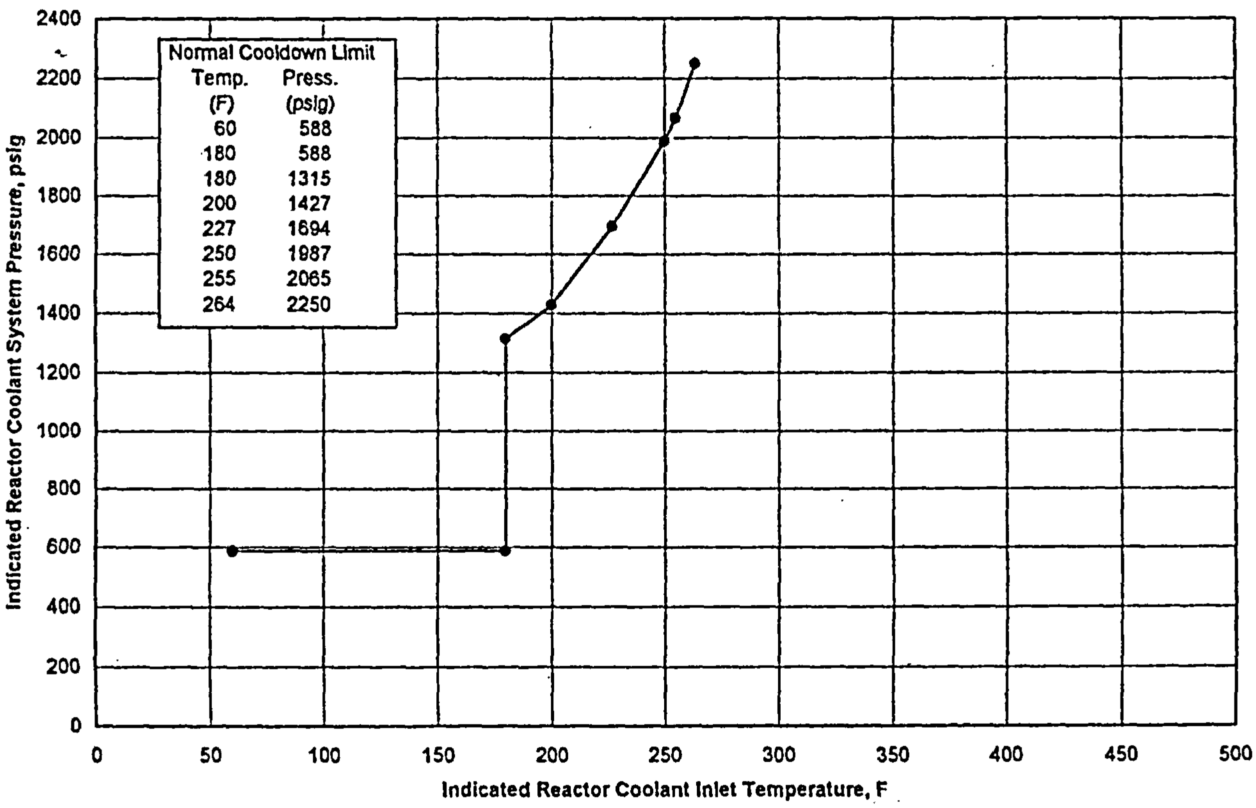
Figure 16.5.4-3 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 1



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.

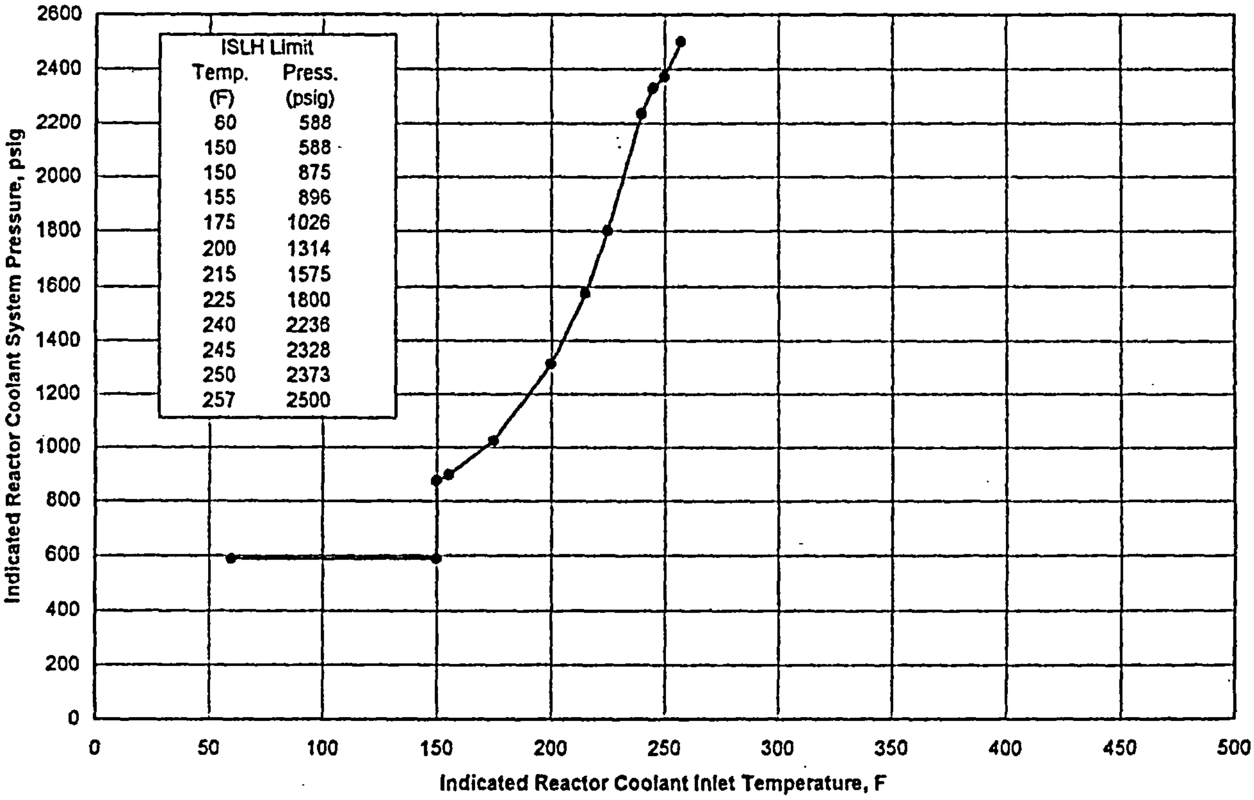
Figure 16.5.4-4 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



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.

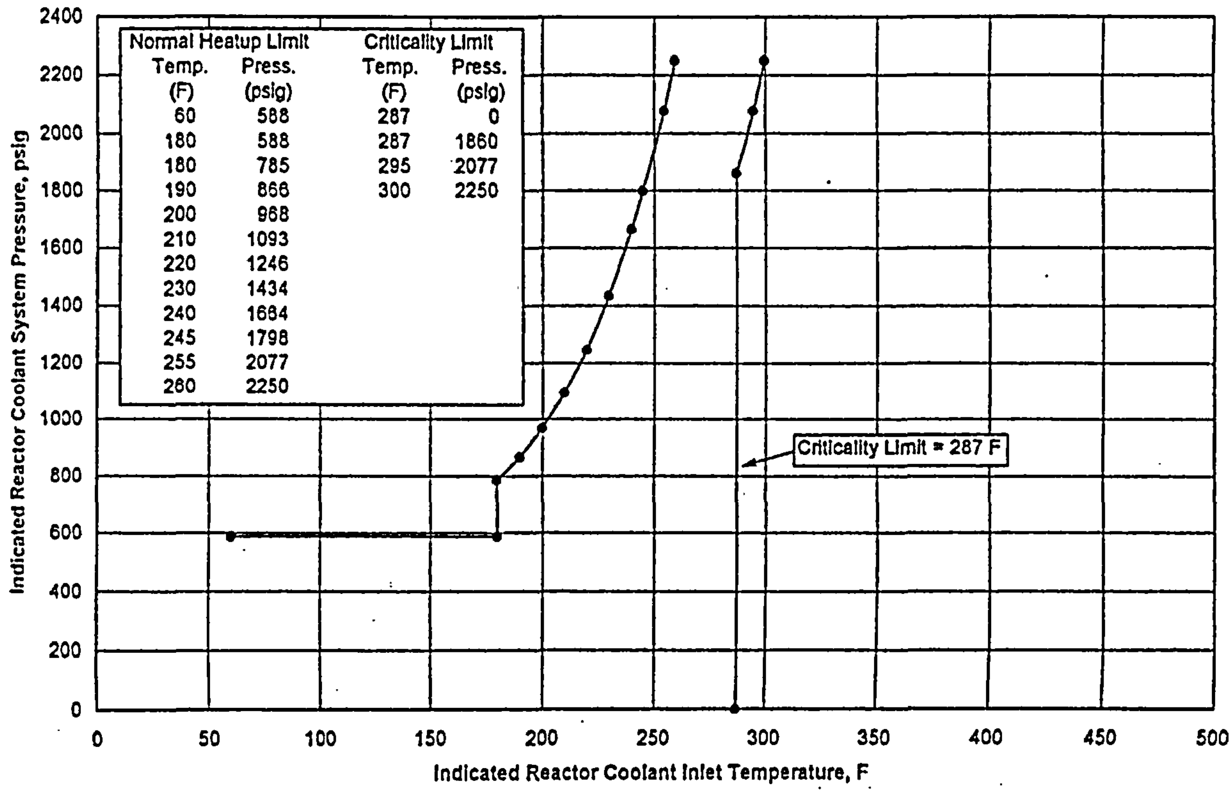
Figure 16.5.4-5 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



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.

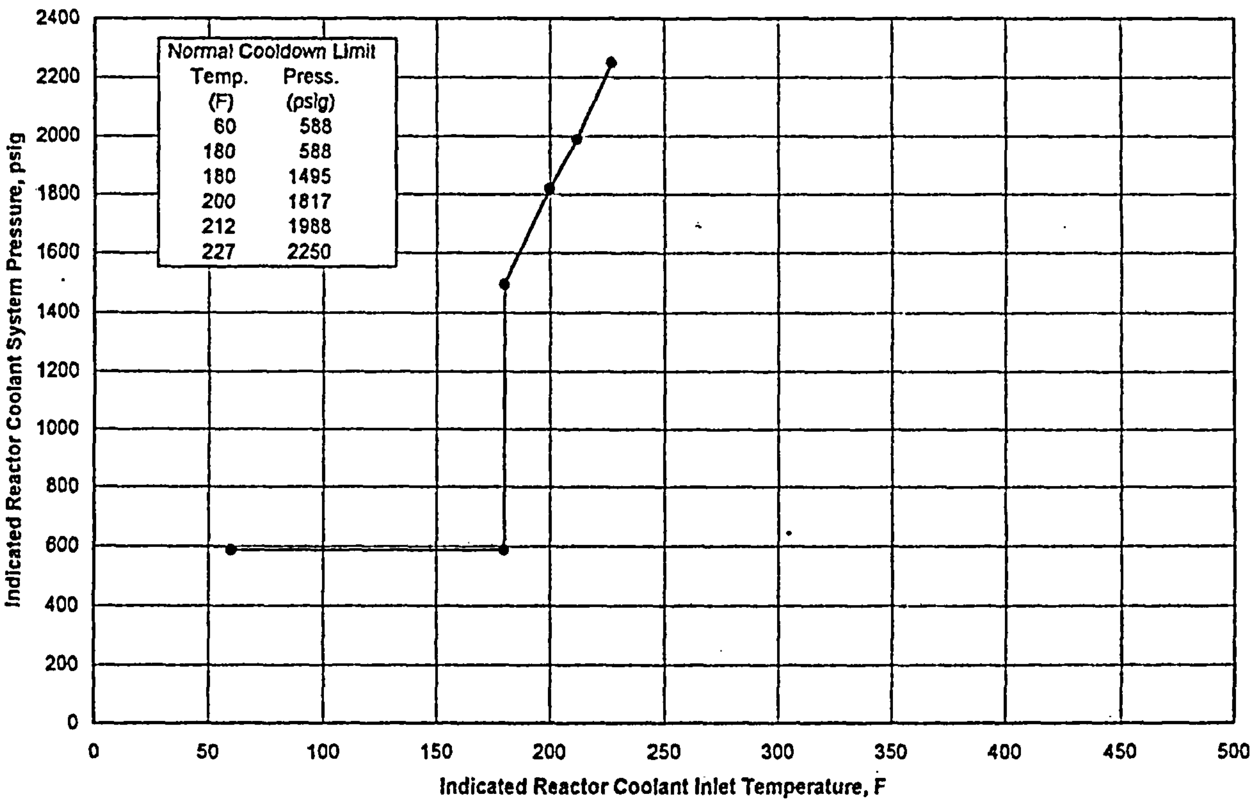
Figure 16.5.4-6 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 2



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.

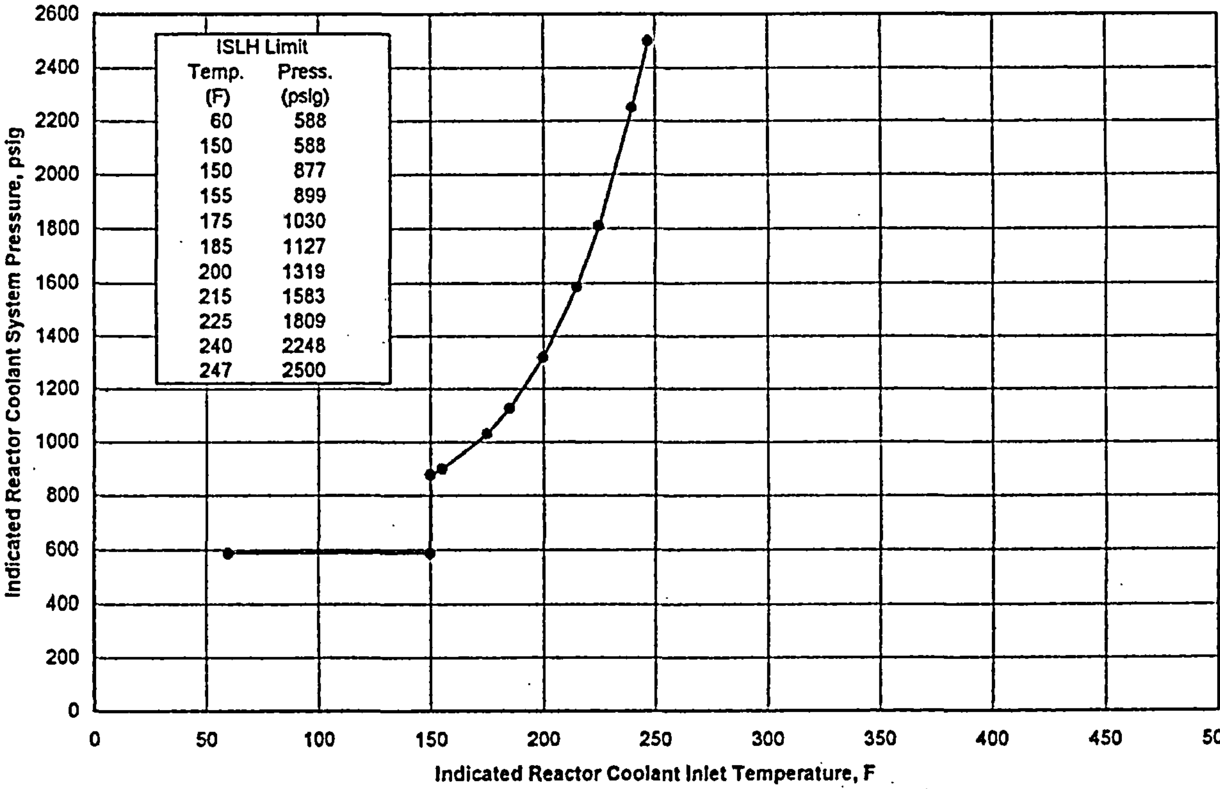
Figure 16.5.4-7 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 3



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 16.5.4-8 (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 16.5.4-9 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 33 EFPY - Oconee Nuclear Station Unit 3

ATTACHMENT 4

TECHNICAL JUSTIFICATION

TECHNICAL JUSTIFICATION

Background and Description of Proposed Change

The pressure-temperature limits for the reactor coolant pressure boundary (RCPB) for Oconee are established in accordance with the requirements of 10 CFR Part 50, Appendix G. The methods and criteria employed to establish operating pressure and temperature limits (PTL) are described in topical report BAW-10046A and the alternative rules provided by ASME Code Case N-588 for circumferential flaws in welds and Code Case N-626 for K_{1c} fracture toughness. These limits are provided to prevent non-ductile failure during any normal operating condition, including anticipated operation occurrences and system hydrostatic tests. The loading conditions of interest include the following: Normal operation, including heatup and cooldown; inservice leak and hydrostatic test (ISLH); and reactor core operation. The major components of the RCPB have been analyzed in accordance with 10 CFR Part 50, Appendix G. The closure head region, the reactor vessel (RV) outlet nozzle, and the nozzle belt and beltline regions have been identified as the only regions of the RV (and consequently the RCPB) that require pressure-temperature limits. The limit curves are based on the predicted value of the adjusted reference temperature (ART) of the limiting RV materials at the end of 33 EFPY. These values are derived in accordance with Regulatory Guide 1.99, Revision 2. The NRC approved a revision to the PTL curves in Amendment Nos. 307, 307, and 307 dated October 1, 1999. The SER discusses the methodology in detail and can be referenced for further details.

This proposed amendment will relocate the results of the above amendment, i.e. revised curves for Pressure Temperature Limits currently contained in TS 3.4.3 to a Pressure Temperature Limits Report (PTLR). The results will be located in the Selected Licensee Commitments (SLC) manual. Similar to the current TS format, the new SLC will contain individual PT curves for each of the Oconee units.

NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996, allows licensees to relocate the pressure temperature limit (PTL) curves from their plant technical specifications (TS) to a PTL report or a similar document. The methodology used to determine the PTL parameters must comply with the specific requirements of Appendices G and H to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR), be documented in an NRC approved topical report or in a plant-specific submittal and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment.

This change also incorporates a minor NRC approved generic change to the Improved Standard TS for BWOG, NUREG-1430, Rev. 3; specifically, TS Task Force (TSTF) change TSTF-419-A, Rev. 0, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR."

The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is added to TS 1.1 as follows:

> The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.9.

TS 3.4.3, RCS Pressure and Temperature Limits is being revised to remove the PTL parameters and curves from the TS. They will be included by reference in the LCO.

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) is being revised to replace references to 'LCO 3.4.3' with PTLR.

TSB 3.4.3 and 3.4.12 will be revised to reflect the above.

Requirements will be added to TS Section 5.0, specifically 5.6.9, Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to identify the NRC approved methodology.

Justification for the Proposed Change

The definition PTLR identifies the specifications in which the pressure and temperature limits are addressed. Specification 5.6.9 identifies the acceptable TS reference and methodologies used to determine RCS pressure and temperature limits. The proposed changes eliminate duplication.

The revision to TS 3.4.3 and its associated Bases is justified by what the referenced amendments, TSTF and GL are allowing licensees to do relative to PTL curves. The curves will be incorporated by reference.

The addition to TS 5.6, specifically 5.6.9, allows the Topical Reports to be identified so that licensees can use current Topical Reports to support limits in the PTLR without having to submit an amendment to facility operating licenses every time the Topical Report is revised. The PTLR will provide specific information identifying the particular approved Topical Reports used to determine the PT limits. This still provides the assurance that only the approved versions of the referenced Topical Reports will be used for the determination of the PT limits since the complete citation will be provided in the PTLR. This proposed change is consistent with TSTF-363, "Revise Topical Report references in ITS 5.6.5, COLR," which was approved by the NRC on April 13, 2000.

The requirement to operate within the limits in the PTLR is specified in and controlled by the TS. Only the figures, values, and parameters associated with the PTL are relocated to the PTLR. The methodology for their development must be reviewed and approved by the NRC. The proposed changes do not change the requirements associated with the review and approval of the methodology or the requirement to operate within the limits specified in the PTLR.

The following plants have submitted and gained approval for similar submittals: Beaver Valley in an safety evaluation report (SER) dated July 15, 2003; Byron and Braidwood in an SER dated January 23, 1998; Vogtle in an SER dated March 28, 2005; Diablo Canyon in an SER dated May 13, 2004; and Fort Calhoun in an SER dated August 15, 2003.

ATTACHMENT 5

NO SIGNIFICANT HAZARDS CONSIDERATION

NO SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10 CFR 50.91, Duke has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) <u>Involve a significant increase in the probability or consequences of an accident previously evaluated:</u>

The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the PTL that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports will require review in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, or components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Therefore, it is concluded that the proposed LAR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) <u>Create the possibility of a new or different kind of</u>

accident from any kind of accident previously evaluated:

The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the PTL that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Involve a significant reduction in the margin of Safety

The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the PTL that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As such, the proposed change does not involve a significant reduction in a margin of safety.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment.

ATTACHMENT 6

ENVIRONMENTAL IMPACT ANALYSIS

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10 CFR 51.22(b), an evaluation of the LAR has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations. The LAR does not involve:

1) A significant hazards consideration.

   This conclusion is supported by the determination of no significant hazards.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   This LAR does not make any physical changes to the plant, nor does it necessitate a change in parameters governing plant operation. Therefore, this LAR will not change the types or amounts of any effluent that may be released offsite.

3) A significant increase in individual or cumulative occupational radiation exposure.

   This LAR does not involve significant changes in parameters governing plant operation, or methods of operation. Therefore, this LAR will not increase the individual or cumulative occupational radiation exposure.

In summary, this LAR meets the criteria set forth in 10 CFR 51.22(c)(9) of the regulations for categorical exclusion from an environment impact statement.