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OPPD
Omaha Public Power District

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October 8, 2002
LIC-02-0109

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

References:
1. Docket No. 50-285
2. ASME Boiler and Pressure Vessel Code, 1995 Edition and Addenda through the 1996 Addenda, Section XI, Appendix G
3. NRC GL 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996
4. CE NPSD-683-A, Rev 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001
5. Letter from NRC (S. A. Richards) to CEOG (R. Bernier), dated March 16, 2001, "Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561)
6. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk), dated August 23, 2002, "Fort Calhoun Station Unit No. 1 Technical Specifications with respect to Maximum Safety Injection Tank Cover Gas Pressure" (LIC-02-0087)

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby proposes the following amendment to the Technical Specifications (TSs). OPPD proposes to: use a PTLR, change the minimum boltup temperature, modify the TSs to reflect the revised low temperature overpressure protection (LTOP) methodology and analysis that is submitted for review and approval, perform LTOP analyses "in-house", change the LTOP enable temperature, modify TS 2.10.1 to exactly specify the reactor coolant system (RCS) temperature at which the reactor can be made critical, and add a TS for a maximum pressure value for the safety injection tanks (SITs). The use of a PTLR requires the relocation of TS Figure 2-1 (RCS Pressure - Temperature Limits for Heatup, Cooldown, and In-service Test) into Figure 5-1 of the PTLR. As a result of these changes, the

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following TSs are required to either be modified or added: Define the PTLR in Definitions, 2.1.1(8), 2.1.1(11), Basis Section of 2.1.1, 2.1.2 including the 2.1.2 Basis and Reference Sections, 2.1.6(4), 2.3(1)(c), 2.3(3), 2.3 References, 2.10.1 and 2.10.1 Basis Section, Table 3-5 item 23, 3.3(1)(c), and 5.9.6.

OPPD's relief request to use Westinghouse Electric Company/Combustion Engineering's pressure and temperature limit curve methodology is also being submitted as part of this letter. This relief request is in support of FCS use of a PTLR as required by Reference 5.

Attachment 1 provides the No Significant Hazards Evaluation and the technical bases for these requested changes to the TSs. Attachment 2 contains a marked-up and clean version reflecting the requested TS and Basis changes. Attachment 3 contains the plant-specific PTLR. Attachment 4 contains the non-proprietary version of the applicable References from Attachment 1.

The basis for the use of a PTLR is Reference 3. The plant-specific PTLR is based on Reference 4, as modified per Attachment 1. These modifications were due to Reference 5 and the revised LTOP methodology and analysis that is being submitted for review and approval. The LTOP methodology and analysis use the RELAP5/Mod 3.2 computer code and OPPD proposes to be able to perform LTOP analyses "in-house". Additionally, due to the revised LTOP analysis, OPPD will be able to eliminate the potential for a non-conservative condition allowed by TS 2.1.11 that was not covered by the LTOP analysis of record. Since the revised LTOP analysis uses the same minimum boltup and LTOP enable temperature as Figure 5-1 of the PTLR, and both of these values are in compliance with Reference 2, OPPD proposes to use these lower values. Modifying TS 2.10.1 will exactly specify the RCS temperature at which the reactor can be made critical. Finally, OPPD proposes to add a maximum pressure value for the SITs to the TSs per Reference 6. This SIT parameter meets Criterion 2 of 10 CFR 50.36(c)(2)(ii).

If during the process of NRC review and approval of the proposed amendments, a revision of TS 5.9.6 is necessary to accommodate Reference 5, OPPD will provide an update to TS 5.9.6.

OPPD requests the approval of this License Amendment Request before July 12, 2003 for use during the fall 2003 refueling outage which is planned to start on September 12, 2003. Once approved, the amendment shall be implemented within 60 days. No commitments are made to the NRC in this letter.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on October 8, 2002)

If you have any questions or require additional information, please contact Dr. R. L. Jaworski of my staff at 402-533-6833.

Sincerely,

D. J. Bannister
Manager – Fort Calhoun Station

DJB/rlj/fjj

Attachments
1. Fort Calhoun Station's Evaluation
2. Markup and Clean Versions of Technical Specification Pages
3. Reactor Coolant System Pressure and Temperature Limits Report
4. References for Attachment 1(Non-proprietary Version)

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
John G. Kramer, NRC Senior Resident Inspector
B. E. Casari, Director - Environmental Health Division, State of Nebraska
Winston & Strawn

Fort Calhoun Station's Evaluation
For
Reactor Coolant System Pressure and Temperature Limits Report

1.0 INTRODUCTION

This letter is a request to amend Operating License DPR-40 for the Fort Calhoun Station (FCS) Unit No. 1.

The Omaha Public Power District (OPPD) proposes to: 1) implement the use of a pressure and temperature limits report (PTLR); 2) change the minimum boltup temperature; 3) incorporate the system setpoints into the PTLR and limitations into the Technical Specifications (TSs) for the revised low temperature overpressure protection (LTOP) analysis; 4) use a LTOP methodology based on the RELAP5/Mod 3.2 computer code; 5) perform LTOP analyses "in-house"; 6) reduce the LTOP enable temperature; 7) exactly specify the reactor coolant system (RCS) temperature at which the reactor can be made critical; (8) add a maximum pressure specification for the safety injection tanks (SITs) to the TSs; 9) and use Westinghouse Electric Company/Combustion Engineering's (W/CE's) methodology for developing pressure and temperature (P-T) limit curves.

NRC approval of this amendment request by July 12, 2003 and an implementation period of 60 days is desired by OPPD.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

1) The proposed changes include relocating TS Figure 2-1 (RCS Pressure - Temperature Limits for Heatup, Cooldown, and Inservice Test) into Figure 5-1 of the PTLR. Figure 5-1 of the PTLR is based on Figure 2-1 of Reference 10.1 and approved by Reference 10.13 with the following modifications:
   a) Incorporates a lower minimum boltup temperature from 82°F to 64°F. The P-T limit curve stated in Reference 10.1 was analyzed to an analytical value of 50°F or an indicated temperature of 64°F. Those values below an indicated temperature of 82°F (which corresponded to an approved minimum boltup temperature of 82°F) were not used since the LTOP system at that time was not revised to include this lower temperature. Figure 5-1 of the PTLR uses the same P-T limit values that were used in Figure 2-1 of Reference 10.1 except it incorporates the following additional values from Table 1, Attachment 4a of Reference 10.1 for the heat-up and cooldown curve:
      i) Indicated Temperature of 64°F: 498 psi
      ii) Indicated Temperature of 74°F: 502 psi
   b) The LTOP analysis (Reference 10.2) has been revised and uses an analytical value of 50°F or an indicated temperature of 64°F. Figure 2-1 of Reference 10.1 was analyzed to the same temperatures but only incorporated P-T limit values to an indicated temperature of 82°F. The LTOP analysis ensures that the peak pressures during any LTOP event does not exceed any P-T limit from 50°F or greater. Therefore, since Figure 5-1 of the PTLR incorporates the additional P-T limit values from 50°F and greater, the LTOP system is applicable and bounding. OPPD did not invoke the use of American Society for Mechanical Engineer's

(ASME) Code Case N-514 in its revised LTOP analysis, which allows the P-T limit curve to be exceeded by an additional 10%

c) The LTOP enable temperature is sensitive to the P-T limit curve only due to where pressure instrument uncertainty is applied. Above the LTOP enable temperature, pressure instrument uncertainty is incorporated into the P-T limit curve and below this temperature it is not. Figure 2-1 of Reference 10.1 incorporated a value of 350°F but maintained an LTOP enable temperature of 385°F due to the previous LTOP analysis, as approved by Reference 10.4, was analyzed to that temperature. The LTOP analysis has been revised to use a value of 350°F (Reference 10.2), hence the excessive conservatism stated in Reference 10.1 is not applicable to Figure 5-1 of the PTLR.

2) OPPD requests the approval of the revised LTOP analysis (Reference 10.2) using the LTOP methodology based on the RELAP5/Mod 3.2 computer code as described in Reference 10.3. This methodology is incorporated into the PTLR and would allow OPPD to change LTOP system setpoints without prior NRC approval. In February, 2001 it was discovered by OPPD engineers that the TSs allowed for a condition that wasn't analyzed by the LTOP analysis of record (The LTOP analysis was performed by Combustion Engineering now Westinghouse Electric Company) as approved in Reference 10.4. Therefore a TS interpretation (TSI) on TS 2.1.1(11)(a) and (b) was invoked in March, 2001 to prevent FCS from operating outside design basis. The approval of this revised analysis and methodology will allow OPPD to modify the TSs in accordance with this new analysis and methodology hence the TSI would not be required. Additionally, since OPPD also seeks approval to perform LTOP analyses "in-house" using the RELAP5/Mod 3.2 computer code, OPPD will be able to minimize this type of error from occurring in the future.

3) OPPD requests the approval of the relief exemption in accordance with 10 CFR 50.60(b) allowing the use of W/CE's methodology for performing P-T limit curves. The NRC has reviewed this methodology in References 10.5 and 10.6 and approved it per Reference 10.6.

4) OPPD proposes the use of a plant-specific PTLR (Attachment 3). The development of the PTLR was in accordance with References 10.5 and 10.7. These References stipulate the requirements for information to be incorporated into it. The basis for Attachment 3 is Reference 10.5. Additionally, the NRC safety evaluation (Reference 10.8) that approved Reference 10.5 contains requirements that must be incorporated into a plant-specific PTLR. Use of the PTLR and approval of the LTOP methodology would enable OPPD to modify the P-T limit curves and LTOP system setpoints without prior NRC approval. This in turn would reduce the unnecessary burden on OPPD and NRC resources of processing P-T limit amendment requests. Allowing OPPD to perform LTOP analyses "in-house" would allow the following:

a) Minimizes OPPD dependence upon a vendor produced LTOP system and reduces this unnecessary external cost.

b) Enhances OPPD understanding of LTOP analysis and methodology. In November 2000, OPPD submitted Licensee Event Report 2000-002 (Reference 10.24) dealing with the inadequate control of LTOP analysis assumptions. In February, 2001, OPPD wrote a condition report when it was identified that the TSs allowed for a condition that was outside of the design basis for the LTOP analysis of record. An underlying cause of this issue was the need for better OPPD understanding of the vendor produced LTOP analysis. Therefore, OPPD co-developed the LTOP methodology (Reference 10.3) and provided significant input on the revised LTOP analysis (Reference 10.2), which are being submitted for review and approval. As a result of performing LTOP analysis "in-house", OPPD's understanding of the analysis is greatly enhanced.

The PTLR shall be provided to the NRC upon the issuance for each reactor vessel fluence period or effective full power years (EFPYs) and for any revision or supplement thereto as required by Reference 10.7.

5) Reduce the minimum boltup temperature from an indicated 82°F to an indicated 64°F. The use of this conservative, but lower minimum boltup temperature would provide additional margin during high-risk evolutions such as RCS mid-loop operations by increasing the time to boil. This in turn would give operator's additional time to take corrective action during a loss of shutdown cooling event, hence, increasing the margin to safety for FCS.

Based on Section 2.0, items 1 – 5 above, the following TSs are required to be modified: 2.1.1(8), 2.1.1(11), Basis Section of 2.1.1, 2.1.2 including the Basis and Reference sections, 2.1.6(4), 2.3(3), 2.3 References, and 3.3(1)(c).

Based on Section 2.0, items 1 – 5 above, the following TSs are required to be added: Define the PTLR in Definitions, Table 3-5 item 23, and 5.9.6.

OPPD proposes to modify TS 2.10.1 and its Basis section to exactly specify the RCS temperature at which the reactor can be made critical. Removing the provision that allows the reactor to be made critical during physics tests at less than 10^{-1}% power and ensuring that RCS temperature is greater than 515°F is more restrictive than TS 2.10.1(2). Monitoring of TS 2.10.1(2) is required by Reference 10.11 to ensure that fracture mechanics are considered prior to making the reactor critical.

OPPD proposes to add a maximum pressure value for the SITs to the TSs. This value meets Criterion 2 of 10 CFR 50.36(c)(2)(ii) and needs to be incorporated into the TSs (Reference 10.9). Therefore TS 2.3(1)(c) will be modified to incorporate this value.

In Summary, the proposed amendment relocates TS Figure 2-1 (RCS Pressure - Temperature Limits for Heatup, Cooldown, and Inservice Test). Additionally, it allows

the use of a PTLR, it changes the minimum boltup temperature to 64°F, incorporates LTOP design assumptions and modifies existing LTOP design assumptions into the TSs, reduces the LTOP enable temperature to 350°F, exactly specifies and restricts the RCS temperature at which the reactor can be made critical, incorporates a maximum SIT pressure value, and requests a relief exemption in accordance with 10 CFR 50.60(b) to use W/CE's methodology for performing P-T limit curves. The Basis section for TS 2.1.1, 2.1.2, and 2.10.1 are being updated to reflect these changes as applicable. The following TSs must either be modified or added: Define the PTLR in Definitions, 2.1.1(8), 2.1.1(11), 2.1.2, 2.1.2 References, 2.1.6(4), 2.3(1)(c), 2.3(3), 2.3 References, 2.10.1, Table 3-5 item 23, 3.3(1)(c), and 5.9.6.

(Notes: The PTLR (Attachment 3) is written assuming approval by the NRC. Modifications (if any) that result from NRC review will be incorporated into the PTLR prior to plant implementation. Currently, FCS uses a single Figure that incorporates the Reference 10.11 heatup, cooldown, and in-service hydrostatic test curves. In the future, these curves may be split into more than one Figure, therefore to facilitate this approach a "(s)" is added onto the word Figure (i.e. Figure(s)) in the TS markups. This would preclude an administrative change to the TSs if these curves were split into multiple figures.)

3.0 BACKGROUND

FCS will be conducting a fall 2003 refueling outage. In order to facilitate the use of a PTLR, a lower minimum boltup temperature, a higher operating window between the reactor coolant pump (RCP) net positive suction head curve and power operated relief valve (PORV) setpoint curve based on the revised LTOP analysis and methodology, reduce the LTOP enable temperature, remove the TSI associated with TS 2.1.1(11), exactly specify and restrict the RCS temperature at which the reactor can be made critical, add a maximum SIT pressure to the TSs, and use W/CE's methodology for developing P-T limit curves, OPPD requests the approval of this license amendment request (LAR).

The following items, except for items 7 and 8, are being requested as a part of the plant-specific PTLR LAR. Specifically, this LAR proposes:

1) The use of a PTLR in accordance with References 10.7 and 10.8.
2) The use of a conservative, but lower minimum boltup temperature. This lower boltup temperature will provide additional margin during high-risk evolutions such as RCS mid-loop operations by increasing the time to boil. This in turn provides operator's additional time to take corrective action during a loss of shutdown cooling event, hence, increasing the margin to safety for FCS.
3) The use of the revised LTOP analysis described in Reference 10.2 eliminates a non-conservative condition allowed by TS 2.1.1(11)(a) and (b) that was not analyzed by the LTOP analysis of record. (Note: TS 2.1.1(11) will be modified based on the revised LTOP analysis (Reference 10.2) that is being submitted for review and approval.)

4) The use of the LTOP methodology described in Reference 10.3.
5) The use of RELAP5/Mod 3.2 to perform LTOP analyses "in-house". The verification and validation (V and V) of the computer code is described in Section 5.1.5.
6) The use of a conservative, but lower LTOP enable temperature.
7) An exact specification of the RCS temperature at which the reactor can be made critical.
8) Addition of a TS for the maximum SIT pressure fullfilling the commitment made in Reference 10.9.
9) An exemption request to use W/CE's methodology for developing P-T limits curves.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

The technical areas of this LAR and the reference to their associated FCS regulatory requirements and/or guidance documents are listed in the table below:

AREA	REFERENCE
PTLR	10. 7
LTOP system setpoints and operating limitations	10.10
Minimum boltup temperature	10.12
LTOP enable temperature	10.12
The minimum temperature requirement at which the reactor can be made critical based on fracture mechanics considerations	10.11
The maximum pressure value for SITs	Criterion 2 of 10 CFR 50.36(c)(2)(ii)
Use of W/CE's alternate methodology for performing P-T limit curves	10 CFR 50.60(b)

FCS was licensed for construction prior to May 21, 1971, and at that time committed to the preliminary General Design Criteria (GDC). These preliminary design criteria are contained in the FCS Updated Safety Analysis Report Appendix G.

This activity complies with FCS Design Criterion 9, "Reactor Coolant Pressure Boundary," which is similar to 10 CFR 50 Appendix A GDC 14, "Reactor coolant pressure boundary." FCS Design Criterion 9 states the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

This activity complies with FCS Design Criterion 20, "Protection Systems Redundancy and Independence," which is similar to 10 CFR 50 Appendix A GDC 21 and 22, "Protection system reliability and testability and Protection system independence." FCS Design Criterion 20 states redundancy and independence designed into protection

systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

This activity complies with FCS Design Criterion 33, "Reactor Coolant Pressure Boundary Capability," which is similar to 10 CFR 50 Appendix A GDC 31, "Fracture prevention of reactor coolant pressure boundary." FCS Design Criterion 33 states the reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

This activity complies with FCS Design Criterion 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention," which is similar to 10 CFR 50 Appendix A GDC 31, "Fracture prevention of reactor coolant pressure boundary." FCS Design Criterion 34 states the reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

This activity complies with FCS Design Criterion 35, "Reactor Coolant Pressure Boundary Brittle Fracture," which is similar to 10 CFR 50 Appendix A GDC 31, "Fracture prevention of reactor coolant pressure boundary." FCS Design Criterion 35 states under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

All of these FCS Design Criteria will continue to be satisfied after these proposed changes to the TS are accepted.

5.0 TECHNICAL ANALYSIS

5.1 Design Basis

5.1.1 P-T Limit Curve

The basis for the P-T limit curve shown in the PTLR (Figure 5-1, Attachment 3) is Reference 10.1 and as updated per Sections 5.1.1 through 5.1.3. (Note: All parameters not modified in Sections 5.1.1 through 5.1.3 will be the same per Reference 10.1. The P-T limit curve from Reference 10.1 was approved without any modifications in Reference 10.13.)

The P-T limit curve stated in Reference 10.1 was analyzed to an analytical value of 50°F or an indicated temperature of 64°F. Those values below an indicated temperature of 82°F (which corresponds to a currently approved minimum boltup temperature of 82°F. See Section 5.1.2 for more detail) were not used since the LTOP system at that time was not analyzed to this lower temperature. (Note: See Section 5.1.3 for more detail) Figure 5-1 of the PTLR uses the same values that were used to develop the P-T limit curve from Reference 10.1 except it incorporates the following additional values from Table 1, Attachment 4a of Reference 10.1 for the heat-up and cooldown curve:

1) Indicated Temperature of 64°F: 498 psi
2) Indicated Temperature of 74°F: 502 psi

5.1.1.1 Relief Exemption

OPPD requests a relief exemption to use W/CE's methodology for developing P-T limit curves per the requirement stated in Reference 10.8. This relief exemption specifically requests approval via 10 CFR 50.60(b) and 10 CFR 50.12 to use a proposed alternative to the described requirements in Reference 10.11. The NRC has previously reviewed W/CE's methodology per References 10.5 and 10.6 and approved it per Reference 10.6.

5.1.1.1.1 Justification for Exemption

10 CFR 50.12(a) states that the NRC may grant exemptions from the requirements of the regulations contained in 10 CFR 50 provided that:

(1) the exemption is authorized by law;
(2) the exemption does not present an undue risk to the public health and safety;
(3) the exemption is consistent with the common defense and security; and
(4) special circumstances, as defined by 10 CFR 50.12(a)(2), are present.

(1) The requested exemption is authorized by law.

The NRC is authorized by law to grant this exemption. 10 CFR 50.60(b) states that the use of alternative methods to Reference 10.11 is acceptable when an exemption is granted by the NRC.

(2) The requested exemption does not present an undue risk to the public health and safety.

The proposed exemption request has no impact on the safe operation of the plant. An exemption from the requirements would allow the use of an alternate methodology to calculate the thermal stress intensity factor (K_{IT}). Specifically, this methodology uses a polynomial fit of the temperature profile and superposition using influence coefficients to calculate K_{IT}. As shown in Reference 10.14 Attachment II, the results of the W/CE's methodology are comparable to the results obtained per Reference 10.12. Therefore, this exemption request does not present an undue risk to the public health and safety.

(3) The requested exemption will not endanger the common defense and security.

The common defense and security are not affected by this exemption request.

(4) Special circumstances are present which necessitate the request for an exemption.

10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption unless special circumstances are present. This exemption meets the special circumstances listed in 10 CFR 50.12(a)(2)(ii).

> 10 CFR 50.12(a)(2)(ii) – Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule
>
> The primary purpose of 10 CFR 50.60 is to protect the reactor vessel against non-ductile failure. The use of W/CE's alternated methodology requested by this exemption provides greater operational flexibility while still maintaining reactor vessel integrity. In addition, the use of W/CE's methodology to generate P-T limit curves yields comparable results to the Reference 10.12 methodology. Therefore, the reactor vessel is protected against non-ductile failure and the underlying purpose of the rule is achieved.

Conclusion

OPPD concludes that the use of W/CE's alternate methodology to calculate the thermal stress intensity factor (K_{IT}) provides greater operational flexibility while providing adequate protection of the reactor vessel against non-ductile failure.

5.1.2 Minimum Boltup Temperature

The minimum boltup temperature is 24°F and its basis for the initial RT_{NDT} are stated in Reference 10.1. For any value of minimum boltup temperature from 24°F or greater to be incorporated into the TSs, it must meet two criteria. These criteria are that both the P-T limit curve and the LTOP analysis must be analyzed to that temperature. The P-T limit curve and the revised LTOP system are evaluated to a minimum boltup temperature of 50°F without temperature indication uncertainty or 64°F with temperature indication uncertainty. Even though a lower boltup temperature is justifiable, it is not practicable to OPPD to analyze both the P-T limit curve and the LTOP system to a lower temperature. Therefore a conservative value of 64°F is incorporated into the TSs and Figure 5-1 of the PTLR which includes instrument uncertainty.

5.1.3 LTOP Analysis

The LTOP analysis (Reference 10.2) was revised and used an analytical value of 50°F or an indicated temperature of 64°F. Figure 2-1 of Reference 10.1 was analyzed to the same temperatures but only incorporated values to an indicated temperature of 82°F. The LTOP analysis ensured that the peak pressures do not exceed any P-T limit from 50°F or greater. Therefore, since the P-T limit curve of Figure 5-1 in the PTLR incorporates the additional values from 50°F and greater, the LTOP system is applicable and bounding. (Note: OPPD did not invoke ASME Code Case N-514 and limited LTOP overpressurization events to 100% of the P-T limits of Figure 5-1 in the PTLR.) The LTOP analysis is a complete re-analysis of limiting LTOP events using the RELAP5/Mod 3.2 computer code. (Note: See Section 5.1.5 for details about the V and V of RELAP5/Mod 3.2 for performing LTOP analyses.)

An independent evaluation was performed by Innovative Technology Solutions Corporation (ITSC) on the RELAP5/Mod 3.2 model that was used in Reference 10.2. The details of this evaluation are located in Reference 10.23 but the overall results conclude that the RELAP model seems well suited for performing LTOP transients and is very well documented. Modeling uncertainties appear to have been consistently addressed in a conservative manner.

5.1.3.1 LTOP Enable Temperature

The LTOP enable temperature is sensitive to the P-T limit curve only due to where pressure instrument uncertainty is applied. Above the LTOP enable temperature, pressure instrument uncertainty is incorporated into the P-T limit curve and below this temperature it is not. The P-T limit curve submitted in Reference 10.1 incorporated a value of 350°F but maintained an LTOP enable temperature of 385°F due to the previous LTOP analysis was analyzed to that temperature. The LTOP analysis has been revised to use a value of 350°F (Reference 10.2), hence the excessive conservatism stated in Reference 10.1 is not applicable to Figure 5-1 of the PTLR.

5.1.4 PTLR (Attachment 3)

The development of the PTLR was in accordance with References 10.5 and 10.7. These References stipulate the requirements for information that is required to be incorporated into it. The basis for the plant-specific PTLR is Reference 10.5. The NRC safety evaluation (Reference 10.8) that approved Reference 10.5 contains requirements that must be incorporated into a plant-specific PTLR. Variances between Reference 10.5 and the plant-specific PTLR and inclusion of the requirements stated in Reference 10.8 will be discussed below.

5.1.4.1 Neutron Fluence Values

NRC Requirement:

1) Describe the methodology used to calculate the neutron fluence values for the reactor vessel materials, including a description of whether or not the methodology is consistent with the guidance of Draft Regulatory Guide DG-1053, a description of the computer codes used to calculate the neutron fluence values, and a description of how the computer codes for calculating the neutron fluence values were benchmarked.
   a) Response: Section 1.0 of the PTLR contains the referenced information.
2) Provide the values of neutron fluence used for the adjusted reference temperature (RT_{NDT}) calculations, including the values of neutron fluence for the inner surface (ID), 1/4T and 3/4T locations of the reactor pressure vessel (RPV).
   b) Response: The values required are stated in Section 1.0 of the PTLR.

Variance: None.

5.1.4.2 Reactor Vessel Surveillance Program

NRC Requirement:

3) Either provide the surveillance capsule withdrawal schedule in the proposed PTLR for the amendment or reference in the PTLR by title and number the documents in which the withdrawal schedule is located.

c) Response: Section 2.0 of the PTLR references the location of the surveillance capsule withdrawal schedule by title.

4) Reference the surveillance capsule reports by title and number if the RT_{NDT} values are calculated using RPV surveillance capsule data.

d) Response: The RT_{NDT} values were calculated using the fluence values from Reference 10.27. RPV surveillance data were used to validate the Reference 10.27 fluence results therefore Section 2.0 of Attachment 3 references the surveillance capsule reports.

Variance: None.

5.1.4.3 LTOP System Limits

NRC Requirement:

5) Provide a description of the analytical method used in the energy addition transient analysis.

e) Response: Section 3.0 of the PTLR references the methodology used to perform the energy addition transient analysis (Reference 10.3).

6) Provide a description of the analytical method used in the mass addition transient analysis, if different from that in Section 3.3.5 of the topical report.

f) Response: The mass addition transient analysis is different from that in Section 3.3.5 of the topical report. Section 3.0 of the PTLR references the methodology used to perform the mass addition transient analysis. (Reference 10.3).

7) Provide a description of the method for selection of relief valve setpoints.

g) Response: A description of the method for the selection of relief valve setpoints is located in the methodology referenced in Section 3.0 of the PTLR (Reference 10.3).

8) Provide a justification for use of subcooled water conditions or a steam volume in the pressurizer.

h) Response: The justification for use of subcooled water conditions or a steam volume in the pressurizer is located in the methodology referenced in Section 3.0 of the PTLR (Reference 10.3).

9) Provide a justification for a less conservative method for determination of decay heat contribution if the method used is less conservative than the "most conservative method" described in the topical report.

i) Response: The method used for the determination of decay heat contribution is located in the methodology referenced in Section 3.0 of the PTLR (Reference 10.3).

10) Provide justification for operator action time used in transient mitigation or termination.

j) Response: The LTOP analysis (Reference 10.2) referenced in Section 3.0 of the PTLR does not require any operator action to mitigate or terminate an LTOP event. Furthermore, the LTOP methodology referenced in Section 3.0 of the PTLR (Reference 10.3) provides a provision if operator action would be needed during a future analysis to mitigate or terminate an LTOP event.

k) Response: The correlations used for developing PORV discharge characteristics are per the methodology referenced in Section 3.0 of the PTLR (Reference 10.3).

12) Provide spring relief valve discharge characteristics if different from those described in the topical report or if the peak transient pressure is above the set pressure of the valve plus 10 percent.

l) Response: FCS uses PORVs versus spring relief valves for the mitigation of LTOP events.

13) Provide a description of how the reactor coolant temperature instrumentation uncertainty was accounted for.

m) Response: A complete description of how instrumentation uncertainty is accounted for is described in the methodology referenced in Section 3.0 of the PTLR (Reference 10.3).

14) Provide a justification for the mass and energy addition transient mitigation which credit presence of nitrogen in the pressurizer.

n) Response: The LTOP analysis (Reference 10.2) referenced in Section 3.0 of the PTLR does not credit the use of nitrogen in the pressurizer for LTOP transient mitigation. The LTOP methodology (Reference 10.3) does not contain a provision to analyze the presence of nitrogen in the pressurizer to mitigate an LTOP event. Therefore, FCS would submit a change to the methodology for review and approval prior to implementing an LTOP system that would require the presence of nitrogen in the pressurizer to mitigate the LTOP event.

15) Identify and explain any other deviation from the methodology included in Section 3.0 of the topical report.

o) Response: FCS will not be using Section 3.0 of the topical report to discuss the methodology associated with performing an LTOP analysis. The methodology that will be used is referenced in Section 3.0 of the PTLR (Reference 10.3).

Variance: The FCS methodology for performing LTOP analyses will be in accordance with the methodology referenced in Reference 10.3 versus Section 3.0 of the topical report. The FCS methodology and LTOP analysis that are being submitted for review and approval discusses the system setpoints and additional limitations that are required to ensure that peak pressure does not exceed Figure 5-1 of the PTLR during a limiting LTOP event.

5.1.4.4 Beltline Material Adjusted Reference Temperature (ART)

NRC Requirement:

16) Identify the limiting materials and corresponding RT_{NDT} values for both the quarter-thickness (1/4T) and three-quarter-thickness (3/4T) locations of the RPV shell.

p) Response: Section 4.0 of the PTLR contains the required information.

17) For pressurized water reactor (PWR) design facilities, identify the limiting RT_{PTS} value for RPV as calculated in accordance with the methods and criteria of 10 CFR 50.61.
   q) Response: Section 4.0 of the PTLR contains the required information.

Variance: None.

5.1.4.5 Pressure-Temperature Limits Using Limiting ART in the P-T Curve Calculation

NRC Requirement:

18) Ensure that the ferritic RPV materials that have accumulated neutron fluences in excess of 1.0×10^{17} n/cm^2 (E > 1 Mev) will be assessed according to Section 4.0 of the CE Topical Report CE NPSD-683 Revision 6, regardless of whether the materials are located within the region immediately surrounding the active core.
   r) Response: Section 5.0 of the PTLR states the Reference that contains the required information (Reference 10.15).
19) Identify which method (i.e., K_{IC} or K_{IA}) will be used to calculate the reference intensity factor (K_{IR}) values for the RPV as a function of temperature.
   s) Response: Figure 5-1 of the PTLR is based on TS Figure 2-1 (Reference 10.1) which uses K_{IC} to calculate K_{IR}.
20) Submit an exemption request to use the methods of Code Case N-640 and apply them to the P-T limit calculations. Note that the staff will approve an exemption request to use Code Case N-640 and K_{IC} as the bases for generating the P-T limit curves only if a licensee indicates that it will limit the maximum pressure in the vessel to 100 percent of the pressure satisfying Paragraph G-2215 of the 1996 edition of Appendix G to the Code for establishing LTOP limit setpoints.
   t) Response: An exemption request to use the methods of Code Case N-640 has been approved by the NRC (Reference 10.13). FCS limits the maximum pressure in the vessel to 100 percent of the pressure satisfying Paragraph G-2215 of the 1996 edition of Appendix G to the Code for establishing LTOP limit setpoints.
21) Apply for an exemption against requirements of Section IV.A.2. of Appendix G to Part 50 to apply the CE NSSS methods to their P-T curves.
   u) Response: This licensing letter contains the exemption request.
22) Include in their PTLRs the P-T curves for heatup, cooldown, criticality, and hydrostatic and leak tests of their reactors.
   v) Response: With the exception of the criticality curve, these curves are located on Figure 5-1 of Attachment 3. The NRC approved FCS request to delete the criticality curve is per Reference 10.26. The basis for deleting this curve is TS 2.10.1(1), which states that the reactor shall not be made critical unless average reactor coolant temperature is greater than 515°F. Figure 5-1 of the PTLR is based on Reference 10.1, which did not include the explicit P-T limits when the core is critical. Reference 10.25 describes the process to determine these limits from Attachment 4b of Reference 10.1 to verify they do not exceed 515°F. Using References 10.1 and 10.25, an evaluation was performed and it was determined that the core critical P-T limits are less than

515°F. OPPD will ensure that the core critical P-T limits are less restrictive than the temperature value required by TS 2.10.1(1) any time the P-T limit curves are revised.

Variance: None.

5.1.4.6 Minimum Temperature Requirements in the P-T Curves

NRC Requirement:

23) Demonstrate how the P-T curves for pressure testing conditions and normal operations with the core critical and not-critical will be in compliance with the appropriate minimum temperature requirements as given in Table 1 to Appendix G to Part 50.
    w) Response: Section 6.0 of the PTLR contains the information required.

Variance: None.

5.1.4.7 Application of Surveillance Data to ART Calculations

NRC Requirement:

24) Include in their PTLRs the supplemental surveillance data and calculations of the chemistry factors if surveillance data are used for the calculations of the ARTs.
    x) Response: Section 7.0 of the PTLR contains the referenced information.
25) Provide the evaluation of whether the surveillance data are credible in accordance with the credibility criteria of Regulatory Guide 1.99, Revision 2.
    y) Response: Section 7.0 of the PTLR contains the referenced information.
26) In addition, if licensees seek to use surveillance data from supplemental plant sources, licensees must
    a) Identify the source(s) of the data.
        z) Response: Section 7.0 of the PTLR contains the required information.
    b) Either identify by title and number the safety evaluation report that approved the use of the supplemental data, along with a justification of why the data is applicable; or compare the licensee's (applicant's) data with the data from the supplemental plant(s) for both the radiation environments (i.e., neutron spectrums and irradiation temperatures) and the surveillance test results, and pursuant to Section III.C of Appendix H to Part 50, submit the proposed integrated surveillance program and evaluation of the data to the NRC for review and approval.
        aa) Response: Section 7.0 of the PTLR contains the required information.

Variance: None

5.1.5 RELAP5/Mod 3.2 Verification and Validation

The primary objective of the V and V on RELAP5/Mod 3.2 (i.e. computer code) was to verify its capabilities in performing LTOP licensing analyses. The V and V activities performed on the computer code have been designed to ensure that it is correctly installed and to demonstrate that the program is suitable for performing LTOP and similar

analyses. Correct installation of the computer code is verified by running the standard sample problems supplied with the installation package and assuring that the results are essentially identical to the output supplied. Validation of the program for performing LTOP analyses is demonstrated by the following:

1) Performing the LTOP test cases that were performed by another version of RELAP5/Mod 3.2 and verify that the computer code can reproduce the results.
2) Performing a test case that verifies pressurizer behavior and determining that the results compare favorably to the provided data.
3) Performing a test case of the accident at TMI-2 (where the thermal-hydraulic phenomena occurring during the early stages of the TMI-2 event is similar to an LTOP event) and determining that the results compare favorably to plant data.
4) Taking credit for the numerous assessments that have been performed on the RELAP5/Mod 3.2 computer code.

5.1.5.1 Installation

The computer code was compiled, and optimized for the HP-UX 11.0 systems. The compiler used for this installation is a Fortran90 Version 2.4 compiler. The computer code was received from the NRC via Information Systems Laboratories, Inc. (ISL) (Reference 10.16). The computer code is a modification of RELAP5/Mod 3.2 as stated in the readme.txt file (Reference 10.17) and the changesummary.pdf file (Reference 10.18) that were provided with the computer code. Installation of the computer code was in accordance with Reference 10.17.

5.1.5.2 Method

This Section discusses the general method in which the computer code was tested to determine if it was installed properly and evaluate its capabilities in performing LTOP analyses. Specific details for the standard sample problems and the case runs will be discussed sequentially. The procedure that was used is as follows:

1) Run the standard sample problems and case runs using the computer code.
2) If necessary, perform a strip run on their associated restart-plot files of those parameters needed for comparison to the referenced data.
3) For those case runs in which pertinent parameter data was received, perform comparison graphs (using XMGR5 graphical software) between the referenced data and the data that was generated using the computer code.
4) Compare the associated output or strip files (if necessary) by using the unix command "diff"[1].
5) Evaluate and discuss the results of the "diffs" and comparison graphs as applicable.

5.1.5.3 Standard Sample Problems

[1] The unix command "diff" is a differential file and directory comparator.

The standard sample problems were run sequentially on the computer code. These sample problems were provided by ISL (Reference 10.16) and were performed to verify the different capabilities of the computer code (e.g. restart and strip capabilities, reactor kinetics, pump simulations, etc.) and ensure that the computer code is installed correctly. These standard sample problems are described as follows:

1) ans79.i: Long-term decay heat sample problem that contains minimum hydrodynamics to allow testing of the reactor kinetics and decay heat calculation for long time periods.
2) edhtrk.i: This problem has the hydrodynamic components from the Edward's Pipe Blowdown plus heat structures coupled to the pipe, a heat structure with a simple analytic solution, control components with analytic solutions, reactor kinetics, and a few trips.
3) edhtrkd.i: This problem is the same as edhtrk.i except that on the 120 card, heavy water is used instead of light water.
4) edhtrkn.i: This problem is the same as edhtrk.i except that the nearly implicit advancement is used instead of the semi-implicit advancement.
5) edrst.i: The edhtrk.i problem runs to 0.5 seconds and writes a restart file (edhtrk.r). This problem restarts that problem at 0.2 seconds and runs to 0.5 seconds. Problem results at the end of this and the edhtrk.i problem should have identical simulation results. Hence this problem tests the restart capabilities of the computer code.
6) edstrip.i: This problem runs a strip option problem using the restart-plot file from the edhtrk.i problem. A strip file is written and hence tests the data stripping routines of the computer code.
7) marpzd4.i: This problem simulates the behavior of a gas driven pressurizer. Using boundary conditions, a forced small compression is followed by a large expansion.
8) pump2.i: This input deck consists of two problems. The first problem has two mostly identical loops, each with friction, an orifice, and a pump. Built-in pump data are used. The first loop uses an implied motor. The second loop uses pump motor torque data to represent an induction motor. The pump is initially at rest. The pump accelerates to near synchronous speed and fluid is accelerated, reaching near steady state. The pump is then tripped, resulting in decreasing pump speed and fluid velocities. The second problem is identical to the first except that shaft and generator components are used. Therefore this case tests the computer codes built-in pump data curves and its ability to "stack" multiple RELAP cases.
9) typpwr.i: This problem is a simulation of a four loop PWR undergoing a small break. The loop containing the break is modeled as a single loop but the other three loops are combined into one loop. This problem performs an integrated test of many of the RELAP5 code features.
10) typpwrn.i: This problem is the same as typpwr.i except that the nearly implicit

advancement is used with heat conduction/transfer implicitly coupled to hydrodynamics.

11) gasgap.i: This problem is a variation of typpwr.i that verifies that the code fails when there is more than one gas gap interval.

The strip and output files that were distributed with Reference 10.16, were performed by ISL on a HP-UXB.10.20 workstation using a Fortran90 Version 1.0 compiler. These strip and output files are the reference set to determine the adequacy of the installation of the computer code. The success criteria for these standard sample problems were to perform a "diff" of the associated strip and output files and then discuss the differences.

5.1.5.3.1 Results

An evaluation of the "diff" files showed that the strip and output files generated by the computer code and those strip and output files from Reference 10.16 were essentially identical. The differences include the following:

1) CPU runtime.
2) Date/time stamps.
3) Address locations.
4) Numerical value differences were not significant.
5) The computer code on several instances produced a negative 0.0 value versus a 0.0 value from the reference set. The cause was unable to be determined. Since mathematically there isn't any difference between these two values, it was determined to be insignificant.

None of these differences affect the overall results that were generated using the computer code.

5.1.5.4 LTOP Case Runs

The computer code was used to re-perform the LTOP case runs from Reference 10.2. It then was used to generate strip files to match the strip files from the LTOP analysis that was performed by ENERCON Services, Inc. (Reference 10.2) An LTOP event is considered to be a "controlled" small break loss-of-coolant-accident (SBLOCA) that is actuated by a type of pressurized thermal shock (PTS) event. (Note: See Reference 10.2 for more detail.)

1) PTS Type Event: A PTS event is categorized by a cooldown of the RCS followed by a re-pressurization and failure of the reactor vessel. An LTOP event is similar to this with the exception of reactor vessel failure. Reactor vessel integrity is ensured during an LTOP event due to the design and operating limitations imposed by the LTOP system. This system ensures that the PORVs have sufficient capacity to maintain peak pressure below the applicable Reference 10.11 P-T limit curve and hence prevents reactor vessel failure. Specifically, when the RCS is cooled down, a re-pressurization could occur due to either a mass addition (MA) or heat addition (HA) event. A MA event is possible if a

spurious safety injection actuation signal occurs and hence, one or more high pressure safety injection pumps and/or the charging pumps are started and inject water into the RCS causing the pressurizer to be filled and then pressurizes the RCS. A HA event is possible when the RCS is cooled down less than the SGs due to the shutdown cooling system and a RCP is started. This in turn causes a reverse heat transfer between the RCS and the steam generators, fills the pressurizer and then pressurizes the RCS.

2) "Controlled" SBLOCA: It is assumed that an LTOP event can be considered a "controlled" SBLOCA. A SBLOCA is generally defined to include any loss of integrity or break in the PWR pressure boundary that has a break area of 0.5 ft^2 or less. This range of break areas encompasses all small lines that penetrate the RCS pressure boundary including relief and safety valves, such as the PORVs which lift during an LTOP event to maintain the peak pressure below the applicable Reference 10.11 P-T limit curves.

The Reference 10.2 results were generated using a V and V version of RELAP5/Mod 3.2 that was performed by ENERCON Services, Inc and audited by NUPIC (Reference 10.19). On pages 158-163 of Reference 10.2 is a computer code certification that was performed by ENERCON Services, Inc. It concluded that RELAP5/Mod 3.2 is fully capable to perform LTOP analyses based on the numerous publications that show RELAP5/Mod 3.2 is suitable for use in SBLOCA analyses, pressurizer response analyses, and PTS analyses. These thermal and hydraulic phenomena are basically the elements for an LTOP event. Therefore, these case runs were performed to verify the results using the computer code are essentially identical to those results from Reference 10.2. The success criteria for these LTOP case runs were to perform an evaluation between the associated strip files and discuss the differences.

5.1.5.4.1 Results

An evaluation of the "diff" files demonstrated that the results of the case runs that were performed using the computer code produced essentially identical results when compared to Reference 10.2.

5.1.5.5 Neptunus Pressurizer Case Run

The Neptunus pressurizer is a 1/40th scaled model pressurizer located in the Laboratory of Thermal Power Engineering at the Delft University of Technology in the Netherlands. This case run simulated successive insurges, combined with spray, and outsurges in a pressurizer. This assessment tests the accuracy of the interfacial heat and mass transfer models that come into play at the surface of the moving liquid level and in the steam space above the liquid level. To a lesser extent, it also tests the internal wall heat transfer modeling. Figure 5.1.5.5-1 is a schematic of the Neptunus pressurizer system. (Note: Reference 10.20 is the validation report that compares the prediction between a version of RELAP5/Mod 3.2 used by Idaho National Engineering and Environmental Laboratories (INEEL) and the experimental data.)

When an LTOP event occurs, the RCS is being pressurized due to either a MA or HA event which causes an insurge into the pressurizer. Hence, this case run is being performed to verify the computer codes capability in predicting pressurizer behavior. The success criteria for this case run is to perform comparison graphs using XMGR5 graphical software and evaluate any differences between the data provided by INEEL and the data that was generated using the computer code. Additionally an evaluation was performed between these Figures and their corresponding Figures in Reference 10.20.

5.1.5.5.1 Results

Figures 5.1.5.5-2 through 5.1.5.5-4 display the comparisons between the data provided by INEEL and the data produced by the computer code. Figure 5.1.5.5-2 – 5.1.5.5.4 displays that the computer code effectively predicts the pressure and temperature variations during insurges and outsurges in the pressurizer when compared to test data. Additionally, the computer code predicts the overall trend well, but over-predicts the peak pressures and temperatures that occur during the insurge phases. Evaluation of this test case also requires a comparison to the Neptunus test report results. Reference 10.20 discusses the variances between the results produced by RELAP5/Mod 3.2 used by INEEL and the test data. When Figures 5.1.5.5-2 through 5.1.5.5-4 are compared to Figures 5, 6, and 8 of Reference 10.20, it shows that the computer code produces similar results to the RELAP5/Mod 3.2 computer code used by INEEL in the test report. Therefore, the same conclusions (Section 4.0 Reference 10.20) in the test report are applicable to the computer code.

In summary, it is determined that the computer code effectively predicts the pressure and temperature variations and overall trend during an insurge and outsurge to the pressurizer. Since Figures 5.1.5.5-2 through 5.1.5.5-4 match the corresponding Figures of Reference 10.20, the conclusions stated in Section 4.0 of Reference 10.20 are also applicable.

5.1.5.6 TMI-2 Case Run

With the TMI-2 reactor at power, an error during routine maintenance occurred on the secondary side and feedwater to the SGs was interrupted causing a loss of heat sink to the reactor and the reactor scrammed. The RCS overheated and system pressure increased. When pressure reached the setpoint of the pressure regulating valves they lifted but failed to reseat causing a continuation of the SBLOCA. This initiating scenario is very similar to an LTOP event with the exception of the failed open pressure regulating valves. During an LTOP event, the pressurizer fills and the RCS pressurizes due to either a MA or HA event and could potentially lift the PORVs. Therefore, since this case run essentially models the same thermal and hydraulic phenomena that occurred during the initial stages at TMI-2, it is being run to verify RCS and pressurizer behavior and ensure that the computer code can produce similar results. The success criteria for this case run is to perform comparison graphs using XMGR5 graphical software for RCS pressure and

pressurizer level using the data provided from INEEL and the data that was generated using the computer code and discuss any differences. (Note: These two parameters provided from INEEL are the only ones of relevance to an LTOP event. Therefore they are the only ones being used to verify the computer code capabilities in performing LTOP analyses.) Additionally, an evaluation will be performed between these Figures and those displayed in the Reference 10.21 report.

5.1.5.6.1 Results

Figure 5.1.5.6-1 is a comparison of RCS pressure between the computer code and the data provided from INEEL. Figure 5.1.5.6-2 is a comparison of pressurizer level between the computer code and the data provided from INEEL. An evaluation of these Figures demonstrates that the computer code is capable of predicting the overall trend for RCS pressure and pressurizer level when compared to actual data. A comparison of these Figures to Reference 10.21 demonstrates that the computer code produces similar results to the RELAP5/Mod 3.2 version that INEEL used in Reference 10.21. (Note: Reference 10.21 discusses in detail how the TMI-2 event was modeled and includes discussion of boundary conditions, limitations on the results, etc.)

5.1.5.7 Current Errors

The NRC website for RELAP5 is currently unavailable. When this website returns, it will be the host for any open errors associated with RELAP5. Currently, any new errors are being reported via the NRC Newsletter for RELAP5. A review of recent newsletters and per a telephone conversation with ISL has determined that there aren't any known open errors associated with RELAP5/Mod 3.2. If one occurs in the future, it will be evaluated on a case-by-case basis to determine its impact on any results that the computer code has provided.

5.1.5.8 Conclusions

An evaluation of the results for the standard sample problems and case runs that were performed using the computer code has demonstrated the following:

1) The computer code is installed correctly.
2) When proper inputs are provided, the computer code is capable of predicting pressure and temperature variations due to insurges and outsurges in the pressurizer.
3) When proper inputs are provided, the computer code is capable of predicting RCS pressure and pressurizer level during the TMI-2 event.
4) The computer code is capable of reproducing results from other versions of RELAP5/Mod 3.2.

Additionally, an evaluation was performed by ENERCON Services, Inc on the RELAP5/Mod 3.2 computer code. It concluded that RELAP5/Mod 3.2 is fully capable to perform LTOP analyses based on the numerous publications that show RELAP5/Mod 3.2 is suitable for use in SBLOCA analyses, pressurizer response analyses, and PTS analyses.

These thermal and hydraulic phenomena are basically the elements for an LTOP event. Furthermore, there are no known open errors associated with RELAP5/Mod 3.2. If one occurs in the future, it will be evaluated on a case-by-case basis to determine its impact on the results that the computer code has provided. Based on the results of Section 5.1.5, OPPD has determined that the computer code is acceptable for LTOP analyses.

5.1.6 Training

5.1.6.1 RELAP5/Mod 3.2

OPPD:

1) has been formally trained in RELAP5/Mod 3.2 by INEEL.
2) is an active participant of the NRC RELAP5 support group, recently attending the RELAP5/Mod 3.3 workshop in Potomac, Maryland.
3) prepared the RELAP5/Mod 3.2 V and V to ensure that the computer code is installed correctly at FCS and is acceptable for LTOP analyses. It was reviewed by ITSC for completeness and accuracy.
4) has attended INEEL annual meetings for RELAP5.
5) has recent experience using Framatome, ANP proprietary version called ANF-RELAP. This code is based on RELAP5/Mod 2.
6) has 20 years of experience in performing reload and code based safety analyses. The FCS program consists of: a) formal and on-the-job training, b) mentoring of inexperienced users whenever they prepare a safety analysis, and c) independent review of the safety analysis for accuracy and completeness.

5.1.6.2 LTOP Analyses

OPPD:

1) co-developed the LTOP methodology (Reference 10.3) and the LTOP analysis (Reference 10.2) that is submitted for review and approval. The analysis and methodology were performed in accordance with References 10.8 and 10.10.
2) received formal training on modeling LTOP events using RELAP5/Mod 3.2 by ENERCON Services, Incorporated.
3) has 20 years of experience in performing reload and code based safety analyses. The FCS program consists of: a) formal and on-the-job training, b) mentoring of inexperienced users whenever they prepare a safety analysis, and c) independent review of the safety analysis for accuracy and completeness.

5.1.7 Figures

Figure 5.1.5.5-1, NEPTUNUS Pressurizer Facility Diagram

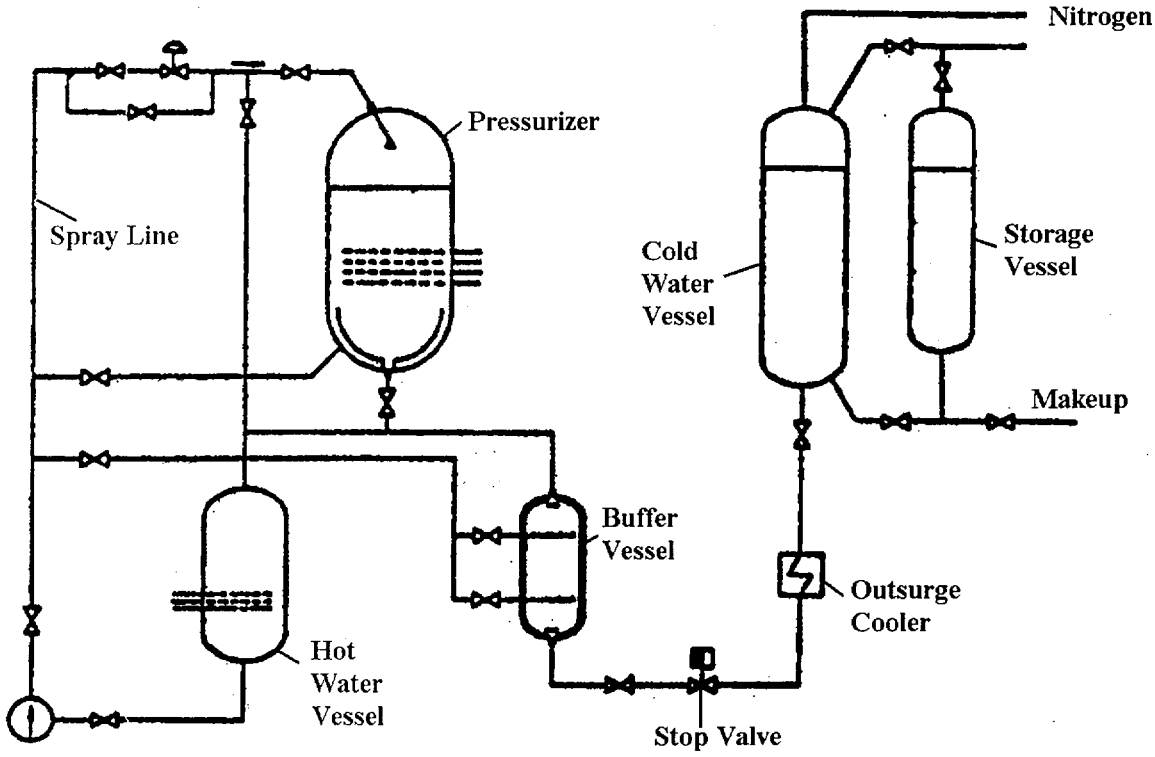


Figure 5.1.5.5−2, Comparison between Calculated and Measured Pressur (Neptunus)

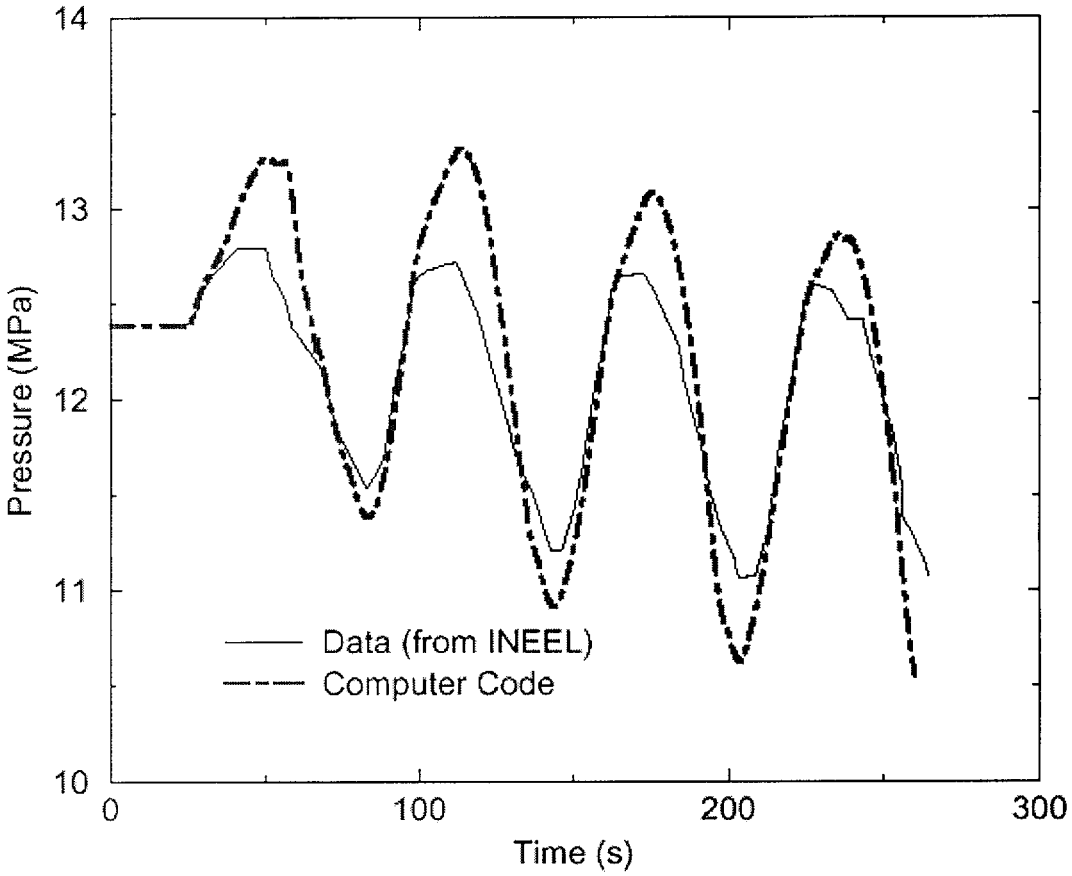


Figure 5.1.5.5-3, Comparison between Calculated and Measured Vapor Temperature (Neptunus)

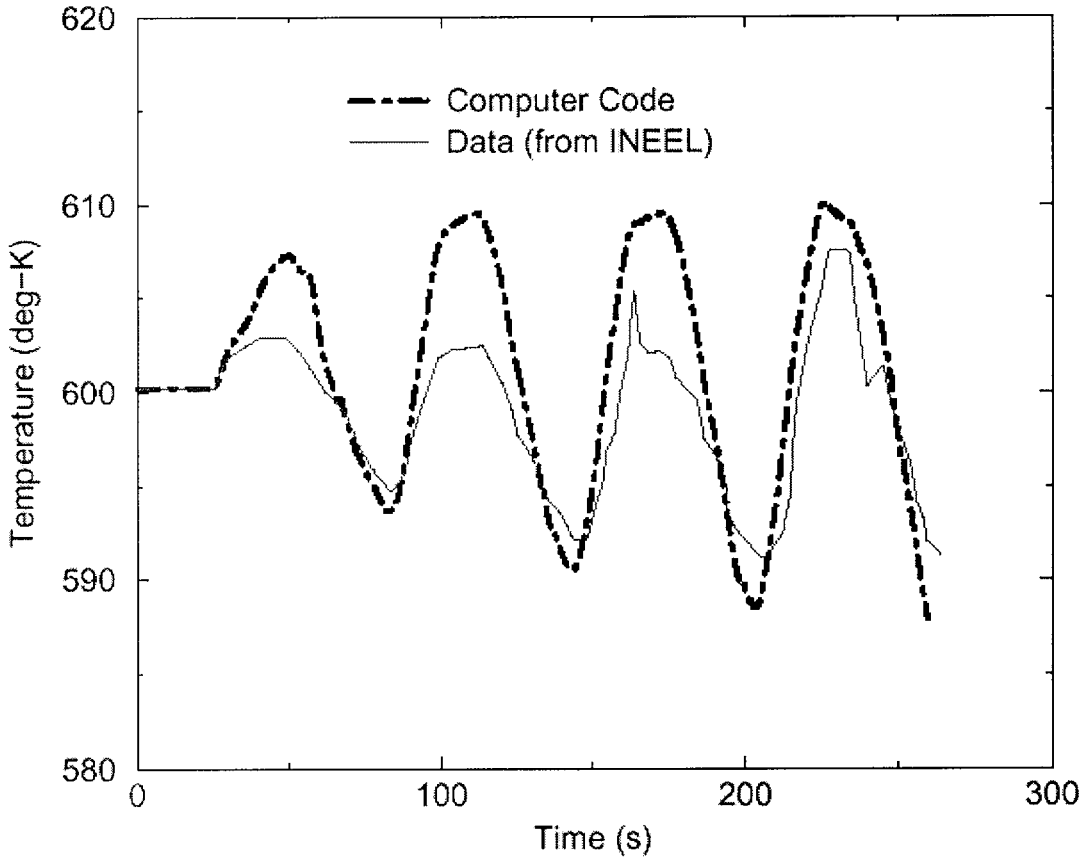


Figure 5.1.5.5-4, Comparison Between Calculated Vapor and Saturation Temperature (Neptunus)

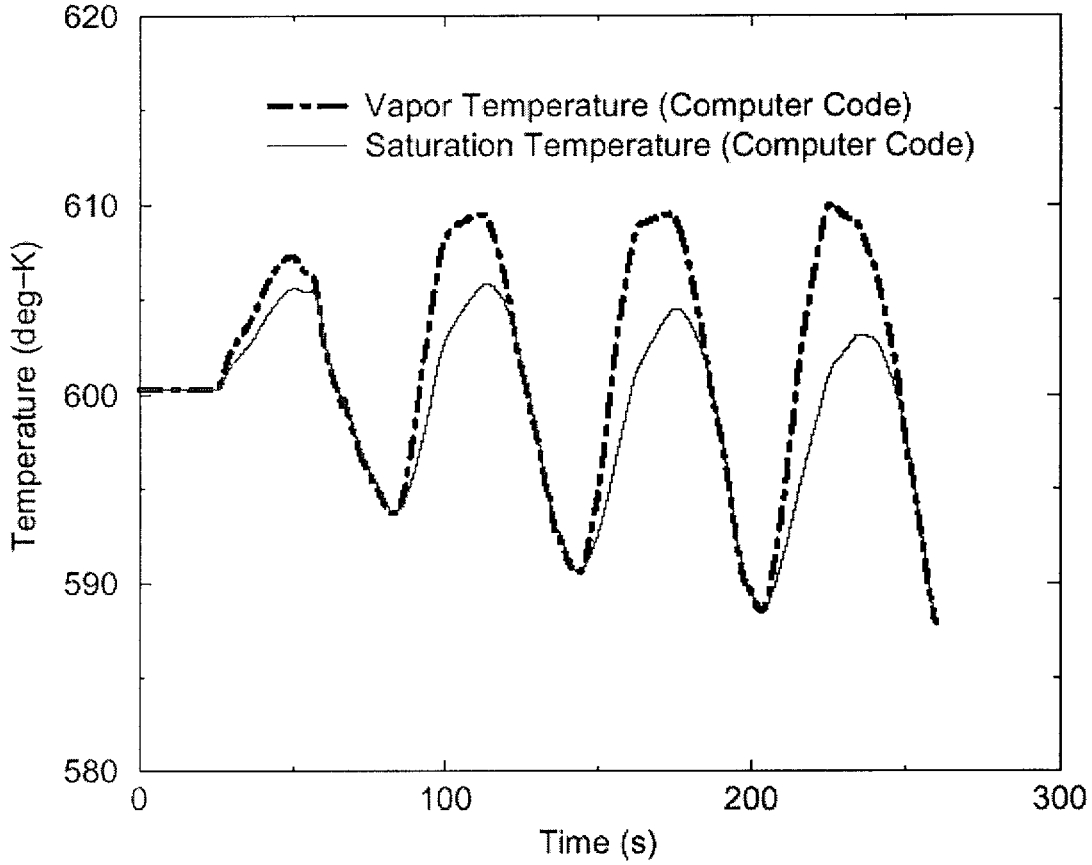


Figure 5.1.5.6-1, Comparison Between Measured and Calculated RCS Pressure (TMI-2)

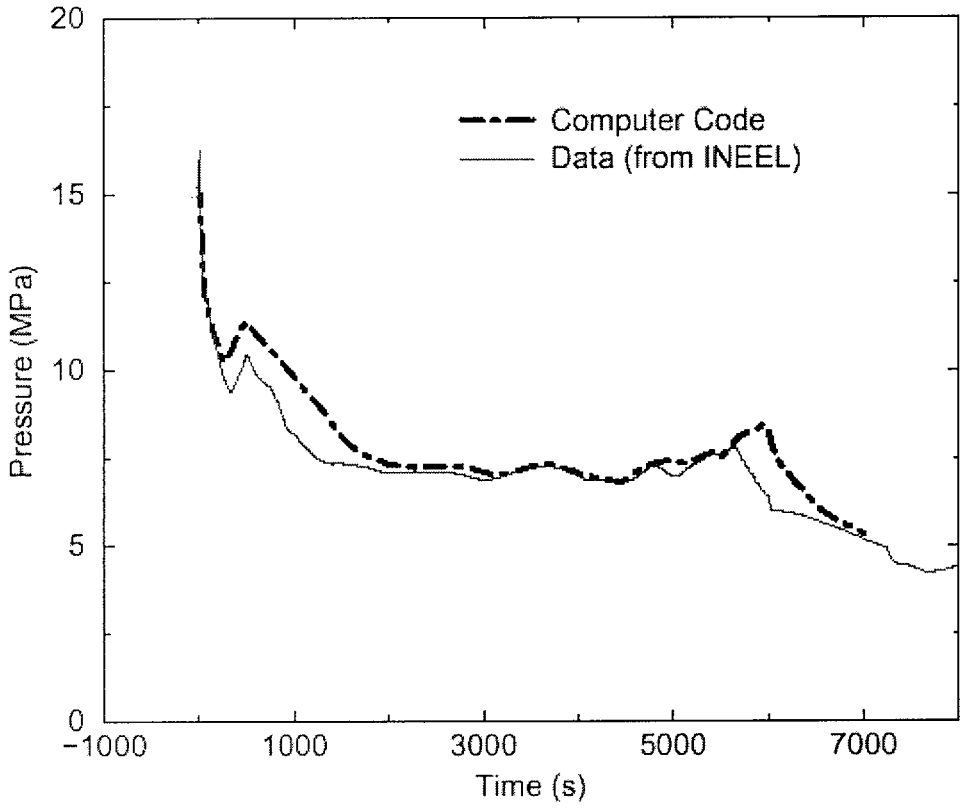
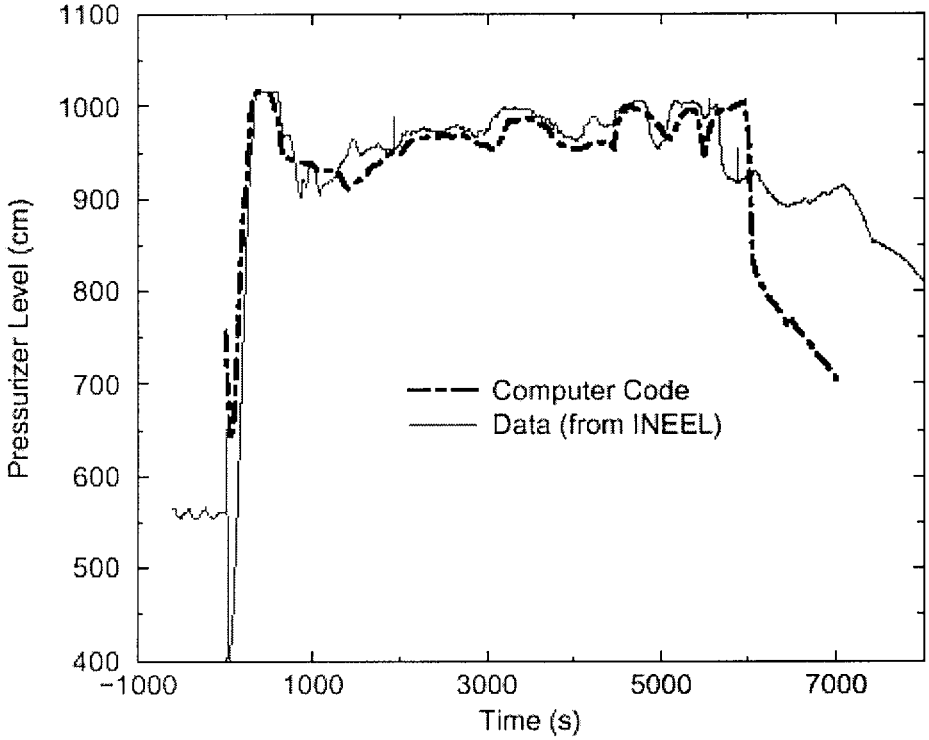


Figure 5.1.5.6-2, Comparison between Measured and Calculated Pressurizer Level (TMI-2)



5.1.8 Maximum Pressure value for SITs

OPPD has committed in Reference 10.9 to add a TS addressing the maximum pressure value for the SITs. This parameter meets Criterion 2 of 10 CFR 50.36(c)(2)(ii). The basis for the 275 psig value chosen is that it is the SIT tank design pressure. This pressure value ensures that excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

5.1.9 Minimum RCS Temperature Reactor Criticality

To exactly specify and restrict the RCS temperature at which the reactor can be made critical TS 2.10.1 and its Basis Section are being modified. Removing the provision that allows the reactor to be made critical during physics tests at less than 10^{-1}% power and requiring that RCS temperature is greater than 515°F is more conservative than TS 2.10.1(2). TS 2.10.1(2) is redundant to Reference 10.11 which requires that fracture mechanics are considered prior to making the reactor critical. Therefore OPPD, per the requirements stated in Reference 10.11, will ensure that the minimum temperature requirement for pressure to exceed 20% of the pre-hydrostatic test pressure to allow core critical operations is less than 515°F whenever the P-T limit curves are revised. (Note: Figure 5-1 of the PTLR is based on Figure 2-1 of Reference 10.1 as approved by Reference 10.13. Per Attachment 4b of Reference 10.1, the core critical temperature is 300°F. This temperature is significantly less than the temperature restriction required by TS 2.10.1(1).)

5.2 Risk Information

The proposed amendment does not involve application or use of risk-informed decisions. The risk to the health and safety of the public as a result of implementing these changes is not impacted and does not reduce the margin to safety.

6.0 REGULATORY ANALYSIS

The proposed amendment to change the P-T limit curve conforms to 10 CFR 50.36(c)(2); Appendices G and H of 10 CFR 50; Generic Letter 88-11; Regulatory Guide 1.99, Revision 2; and Standard Review Plan Section 5.3.2. An exemption request to use W/CE's methodology for performing P-T limit curves is required per Reference 10.8 and is submitted in accordance with 10 CFR 50.60(b).

The technical basis for relocating the P-T limit curve and LTOP system setpoints into a PTLR or similar document is Reference 10.7. The regulatory analysis to use a PTLR or similar document has been accomplished by the NRC in Reference 10.7.

Reducing the minimum boltup temperature is possible due to the P-T limit curves and LTOP analysis are both analyzed to the same temperature. Therefore this is an administrative requirement that is in compliance with Reference 10.12.

Reducing the LTOP enable temperature is possible due to the revised LTOP analysis and Figure 5-1 of the PTLR use this temperature. The basis for the LTOP enable temperature is Attachment 4b of Reference 10.1. Therefore, this is an administrative requirement that is in compliance with Reference 10.12.

The LTOP methodology and revised LTOP analysis are in accordance with Reference 10.10. It ensures that the P-T limits required per Reference 10.11 are not exceeded during limiting LTOP events. Therefore there is no reduction to the margin of safety.

Restricting the RCS temperature at which the reactor can be made critical is more conservative than the minimum temperature requirement for core critical operations based on fracture mechanics considerations required by Reference 10.11 during physics testing. Therefore there is no reduction to the margin of safety.

Adding a maximum pressure to the SITs is required to maintain compliance with Criterion 2 of 10 CFR 50.36(c)(2)(ii). Therefore this is an administrative requirement to control key input parameters to design bases analyses.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Omaha Public Power District (OPPD) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes will not increase the probability or consequence of any accident for the following reasons:

1) The proposed changes relocate the Pressure – Temperature (P-T) limit curves and low temperature over pressure protection (LTOP) system setpoints to the Pressure and Temperature Limits Report (PTLR). Compliance with these curves and limits continues to be required by the Technical Specifications (TSs). Changes to the curves will be controlled by TS 5.9.6, which contains the NRC approved methodologies used in the development of the PTLR. The change to the P-T limit

curve as shown on Figure 5-1 of the PTLR is in compliance with Reference 10.11, Westinghouse Electric Company/Combustion Engineering's (W/CE's) methodology and ASME Code Case N-640 for performing P-T limit curves.

2) Revisions to the LTOP system limits can only be made in accordance with the approved methodologies stated in TS 5.9.6 with any resulting setpoint changes controlled by the 10 CFR 50.59 process. The PTLR in combination with the limitations imposed by the TSs will ensure the integrity of the reactor vessel pressure boundary.
3) The conservative, but lower minimum boltup temperature and LTOP enable temperature are in compliance with Reference 10.12. Since the P-T limit curves and LTOP analysis are analyzed to the same temperatures as these proposed temperature values, there is no reduction to the margin of safety.
4) Restricting the RCS temperature at which the reactor can be made critical is more conservative than the minimum temperature requirements for core critical operations based on fracture mechanics considerations as required by Reference 10.11 during physics testing.
5) Addition of a maximum pressure to the safety injection tanks (SITs) ensures compliance with Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Therefore, the probability or consequence of any accident is not increased.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed revision does not change any equipment required to mitigate the consequences of an accident. The continued use of the same TS administrative controls prevents the possibility of a new or different kind of accident. Since the proposed changes do not involve the addition or modification of equipment nor alter the design of plant systems, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes proposed do not change how design basis accident events are postulated nor do the changes themselves initiate a new kind of accident or failure mode with a unique set of conditions (proposed administrative controls). Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Relocating the P-T limit curves and LTOP system setpoints to the PTLR is in compliance with Reference 10.7. Future updates of the PTLR will be conducted under the 10 CFR 50.59 process utilizing NRC approved methodologies. Updating the P-T limit curve is in accordance with Reference 10.11, W/CE's methodology and

ASME Code Case N-640. Reduction of the minimum boltup temperature and LTOP enable temperature is in compliance with Reference 10.12. Restricting the reactor coolant system (RCS) temperature at which the reactor can be made critical is more conservative than the minimum temperature requirements for core critical operations based on fracture mechanics considerations as required by Reference 10.11 during physics testing. Addition of a maximum pressure to the SITs is in accordance with Criterion 2 of 10 CFR 50.36(c)(2)(ii). Additionally, the LTOP methodology and analysis conforms to Reference 10.10. Therefore, the proposed changes do not involve a significant reduction to the margin of safety.

Based on the above, OPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment is confined to administrative procedures or requirements. The changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1) As demonstrated in Section 7.0, the proposed amendment does not involve a significant hazards consideration.

2) The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. Also, the TS change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite.

3) The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed change does not result in any physical plant changes. No new surveillance requirements are anticipated as a result of these changes that would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENCE

The NRC previously reviewed W/CE's P-T limit curve methodology in References 10.5 and 10.6 and approved it in Reference 10.6.

The NRC has previously reviewed and approved relief exemptions for the use of W/CE's methodology. (Reference 10.6)

The NRC endorses via 10 CFR 50 Appendix G and 10 CFR 50.55a the ASME Boiler and Pressure Vessel Code, 1995 Edition and addenda through the 1996 Addenda Section XI.

The NRC has previously approved an LTOP methodology using RELAP5. (Reference 10.22)

10.0 REFERENCES

10.1 Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), dated December 14, 2001, "Fort Calhoun Station Unit No. 1 License Amendment Request, "Pressure and Temperature (P-T) Limit Curve for 40 Effective Full Power Years (EFPY)" (LIC-01-0114)

10.2 FC06877, Rev. 0, "Low Temperature Overpressure Protection (LTOP) analysis, Revision 1" *****

10.3 FC06876, Rev. 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper" *****

10.4 Letter from NRC (S. Bloom) to OPPD (T. L. Patterson), dated March 23, 1994, "Fort Calhoun Station, Unit No. 1 – Amendment No. 161 to Facility Operating License No. DPR-40 (TAC No. M77351 and M77421)"

10.5 CE NPSD-683-A, Rev 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001

10.6 Letter from NRC (G. F. Wunder) to NYPA (J. Knubel), dated April 10, 1998, "Exemption from the Requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," to Allow for Use of Alternative Methodology for Construction of Pressure Temperature Limit Curves - Indian Point Nuclear Generating Unit No. 3 (TAC No. M99928)"

10.7 NRC GL 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits", January 31, 1996

10.8 Letter from NRC (S. A. Richards) to CEOG (R. Bernier), dated March 16, 2001, "Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561)

10.9 Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk), dated August 23, 2002, "Fort Calhoun Station Unit No. 1 Technical Specifications with respect to Maximum Safety Injection Tank Cover Gas Pressure"

10.10 Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors while Operating at Low Temperatures"

10.11 10 CFR 50 Appendix G, "Fracture Toughness Requirements"

10.12 ASME Boiler and Pressure Vessel Code, 1995 Edition and Addenda through the 1996 Addenda, Section XI, Appendix G

10.13 Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure), dated April 22, 2002, "Fort Calhoun Station, Unit No. 1 – Issuance of Amendment" (TAC No. MB3654)

10.14 Letter from NYPA (J. Knubel) to NRC (Document Control Desk), dated January 28, 1998, "Indian Point 3 Nuclear Power Plant Docket No. 50-286, Proposed Exemption from Requirements of 10 CFR 50.60 to Utilize Alternate Methodology to Determine K_{IT}"

10.15 Letter LTR-CI-02-14, Rev. 00 from WEC (S. T. Byrne) to OPPD (J. Jensen), "Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel", dated February 15, 2002 *****

10.16 Letter from ISL (W. Arcieri) to OPPD (F. J. Jensen), "WCA-09-2002: Transmittal of RELAP5/Mod 3.2d," dated August 2, 2002 *****

10.17 Readme.txt for the installation of RELAP5/Mod 3.2d from Reference 10.16 *****

10.18 Changesummary.pdf that summarizes the modifications to RELAP5/Mod 3.2d from Reference 10.16 *****

10.19 Letter from ENERCON Services, Inc (D. R. Whitson) to OPPD (J. Jensen), dated August 8, 2002, "Quality Assurance for RELAP5/Mod 3.2" *****

10.20 R5-02-01, Validation Report for NEPTUNUS Pressurizer using RELAP5/Mod 3.2, dated April 12, 2002 *****

10.21 INEEL/EXT-02-00589, May 2002, SCDAP/RELAP5-3D Code Manual, Volume 5: Assessment of Modeling of Reactor Core Behavior During Severe Accidents, Appendix A12, "TMI-2 Accident" *****

10.22 Letter from NRC (J. A. Mitchell) to R. E. Ginna (R. C. Mecredy), dated May 23, 1996, "R. E. Ginna – Acceptance for Referencing of Pressure Temperature Limits Report, Revision 1" (TAC No. M94770)

10.23 Letter from ITS Corporation (K. Ross) to OPPD (F. James Jensen), dated September 9, 2002, "ITS Corporation's Cursory Review of OPPD's LTOP Analysis" *****

10.24 Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk), dated December 1, 2000, "Licensee Event Report 2000-002 Revision 0 for the Fort Calhoun Station" (LIC-00-0103)

10.25 Letter from Westinghouse (C. L. Stuart) to OPPD (F. James Jensen), dated September 11, 2002, "Calculation of the Pressure-Temperature Limits and Minimum Temperature Requirements for Core Critical Operation"*****

10.26 Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 74 to Facility Operating License No. DPR-40 Omaha Public Power District Fort Calhoun Station, Unit No.1, dated June 20, 1983

10.27 WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," July 2000 [Contained in Letter LIC-00-0064 from OPPD (W. G. Gates) to NRC (Document Control Desk), dated August 3, 2000]

***** References included in Attachment 4 for NRC use.

Marked-Up and Clean

Technical Specifications
And
Basis Changes

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TECHNICAL SPECIFICATIONS - FIGURES

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DEFINITIONS

$\bar{E}$ - Average Disintegration Energy

$\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

RCS Pressure-Temperature Limits Report (PTLR)

The PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints which provide low temperature overpressure protection (LTOP) to assure the P-T limits are not exceeded during the most limiting LTOP event. The P-T limits and LTOP criteria in the PTLR are applicable through the effective full power years (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information provided in the PTLR.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F. Each steam generator shall be demonstrated operable by performance of the inservice inspection program specified in Section 3.17 prior to exceeding a reactor coolant temperature of 300°F.

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of the pressure and temperature limit Figure(s) shown in the PTLR~~Figure 2-1~~.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) Low Temperature Overpressure Protection (LTOP)

(a) The LTOP enable temperature and RCP operations shall be maintained in accordance with the PTLR.

(b) The unit can not be placed on shutdown cooling until the RCS has cooled to an indicated RCS temperature of less than or equal to 300°F.

(c) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below the LTOP enable temperature stated in the PTLR ~~385°F~~ unless there is a minimum indicated pressurizer steam space of at least 50% by volume.~~at least one of the following conditions is met:~~

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

~~(a) A pressurizer steam space of 53% by volume or greater (50.6% or less actual level) exists, or~~

~~(b) The steam generator secondary side temperature is less than 30°F above that of the reactor coolant system cold leg.~~

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table 2.9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition. Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supply deenergized.

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above 1.18 during all normal operations and anticipated transients.

When Specification 2.1.1(2) is applicable, the reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. Under these conditions, decay heat removal requirements are low enough that a single reactor coolant system (RCS) loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

When Specification 2.1.1(3) is applicable, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling pumps to be OPERABLE.

One of the conditions for which Specification 2.1.1(3) is applicable is when the RCS temperature (T_{cold}) is less than 210°F, fuel is in the reactor and all reactor vessel closure bolts are fully tensioned. As soon as a reactor vessel head closure bolt is loosened, Specification 2.1.1(3) no longer applies, and Specification 2.8 is applicable. Specification 2.8 also requires two shutdown cooling loops to be operable if there is less than 23 feet of water above the top of the core.

The restrictions on availability of the containment spray pumps for shutdown cooling service ensure that the SI/CS pumps' suction header piping is not subjected to an unanalyzed condition in this mode. Analysis has determined that the minimum required RCS vent area is 47 in^2. This requirement may be met by removal of the pressurizer manway which has a cross-sectional area greater than 47 in^2.

When reactor coolant boron concentration is being changed, the process must be uniform throughout the reactor coolant system volume to prevent stratification of reactor coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the reactor coolant is assured if one low pressure safety injection pump or one reactor coolant pump is in operation. The low pressure safety injection pump will circulate the reactor coolant system volume in less than 35 minutes when operated at rated capacity. The pressurizer volume is relatively inactive; therefore, it will tend to have a boron concentration higher than the rest of the reactor coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the reactor coolant system during the addition of boron.[1]

Both steam generators are required to be filled above the low steam generator water level trip set point whenever the temperature of the reactor coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

The bases for the LTOP system requirements are documented in the PTLR.~~The LTOP enable temperature has been established at T_c = 385°F. The pressure transient analyses demonstrate that a single PORV is capable of mitigating overpressure events. Additional uncertainties have been applied to the Pressure-Temperature (P-T) limits to account for the case where a PORV is not available (T_c>385°F) which is the reason for the discontinuity in the P-T Figures. The curves have been conservatively smoothed for operations use.~~

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

The design cyclic transients for the reactor system are given in USAR Section 4.2.2. In addition, the steam generators are designed for additional conditions listed in USAR Section 4.3.4. Flooded and pressurized conditions on the steam side assure minimum tube sheet temperature differential during leak testing. The minimum temperature for pressurizing the steam generator steam side is 70°F; in measuring this temperature, the instrument accuracy must be added to the 70°F; limit to determine the actual measured limit. The measured temperature limit will be 73°F based upon use of an instrument with a maximum inaccuracy of ± 2°F and an additional 1°F safety margin.

~~Formation of a 53% steam space ensures that the resulting pressure increase would not result in any overpressurization should the first reactor coolant pump be started when the steam generator secondary side temperature is greater than that of the RCS cold leg. The steam space requirement is not applicable to the start of a reactor coolant pump if one or more pumps are in operation.~~

~~For the case in which the pressurizer steam space is less than 53%, limitation of the steam generator secondary side/RCS cold leg ΔT to 30°F ensures that a single low setpoint PORV would prevent an overpressurization due to actuation of the first reactor coolant pump. This requirement is not applicable to the start of a reactor coolant pump if one or more pumps are operating.~~

References

(1) USAR Section 4.3.7

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

~~The reactor coolant pressure shall be limited during plant operation in accordance with Figure 2-1 and as follows:~~

a. Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.~~Figure 2-1.~~

b. Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR~~Figures 2-1~~.

c. The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

(a) Immediately initiate action to restore the temperature or pressure to within the limit.

(b) Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

(c) Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR ~~Figure 2-1~~ shall be updated in accordance with the following criteria and procedures:

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

(a) The P-T limit Figure(s) are ~~Figure 2-1 is~~ valid for a fast neutron (E ≥ 1 MeV) fluence value and corresponding ~~of 2.15X10^{19} n/cm^2 which corresponds to 40~~ EFPY as stated in the PTLR.

(b) The limit line on the P-T limit Figure(s) shown in the PTLR ~~figure~~ shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure (E≥1 MeV).

(c) The limit lines in the P-T limit Figure(s) shown in the PTLR ~~Figure 2-1~~ shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 6482°F as it is set by the RT_{NDT} of the reactor vessel flange and is not subjected to a fast neutron flux. The lowest service temperature shall remain at 164°F because components related to this temperature are also not subjected to a fast neutron flux.

(d) ~~The~~ Technical Specification 2.3(3) shall be reviewed and revised as necessary each time the curves on the P-T limit Figure(s) as shown in the PTLR~~of Figure 2-1~~ are revised.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon ~~allowable heatup/cooldown rates and cyclic operation.~~ Cycle dependent information such as the pressure-temperature limit curves, low temperature overpressure protection system limits, neutron fluence, and adjusted reference temperatures are contained in the PTLR, which was developed using the methodologies stated in Technical Specification 5.9.6(b) and in the PTLR[2].

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

~~The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section XI[2] of the ASME Code including Appendix A and G, Westinghouse Electric Company/Combustion Engineering's P-T (W/CE's) limit curve methodology[11] and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements. This ASME Code assumes that a crack 10-11/16 inches long and 1-25/32 inches deep exists on the inner surface of the vessel. Furthermore, operating limits on pressure and temperature assure that the crack does not grow during heatups and cooldowns.~~

~~The reactor vessel beltline material consists of six plates. The nilductility transition temperature (T_{NDT}) of each plate was established by drop weight tests. Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lbs. absorbed energy and 35 mils lateral expansion for the longitudinal direction. NRC technical position MTEB-5-2 was used to establish a reference temperature for transverse direction (RT_{NDT}) of -12°F.~~

~~The initial RT_{NDT} value for the Fort Calhoun submerged arc vessel weldments was determined to be -56°F with a standard deviation of 17°F. By applying the shift prediction methodology of Regulatory Guide 1.99, Revision 2, a weld material adjusted reference temperature (RT_{NDT}) was established at 10°F based on the mean value plus two standard deviations. The standard deviation was determined by using the root-mean-squares method to combine the margin of 28°F for uncertainty in the shift equation with the margin of 17°F for uncertainty in the initial RT_{NDT} value.~~

~~Similar testing was not performed on all remaining material in the reactor coolant system. However, sufficient impact testing was performed to meet appropriate design code requirements[3] and a conservative RT_{NDT} of 50°F has been established.~~

~~As a result of fast neutron irradiation in the region of the core, there will be an increase in the T_{NDT} with operation. The techniques used to predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Reference 5 with the result that the integrated fast neutron flux ($E \geq 1$ MeV) is 1.73×10^{19} n/cm^2, including tolerance at the inside surface of the critical reactor vessel beltline weld material, over the 40 years design life of the vessel.~~

~~Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calibrated azimuthal neutron flux variation. The maximum integrated fast neutron ($E \geq 1$ MeV) exposure of the reactor vessel at the critical reactor vessel beltline location including tolerance is computed to be 1.73×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at~~

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

~~1500 MWt and 85% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is projected to be 252°F, including margin, using the shift prediction equation of Regulatory Guide 1.99, Revision 2. The actual shift in T_{NDT} will be re-established periodically during the plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4.5-1 of the USAR. To compensate for any increase in the T_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. Analysis of the three removed irradiated reactor vessel surveillance specimens[8,9 and 10], combined with weld chemical composition data and implementation of extreme low radial leakage core loading designs beginning in Cycle 14, indicate that the fluence at the end of 40 Effective Full Power Years (EFPY) at 1500 MWt will be 2.15×10^{19} n/cm^2 on the inside surface of the reactor vessel. This results in a total shift of the RT_{NDT} of 237.76°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location using Regulatory Guide 1.99, Revision 2, and a shift of 187.97°F at the 3/4t location. Operation through fuel Cycle 34 will result in less than 40 EFPY.~~

~~The limit lines in Figures 2-1 are based on Reference 2, Appendix G, W/CE's methodology for P-T limit curve generation[11], and ASME Code Case N-640 as discussed below.~~

~~Reference Stress Intensity Factor~~

~~The reference stress intensity factor (K_{IR}) used in the development of the limit lines in Figure 2-1 is based on ASME Code Case N-640. This Code case allows the use of K_{Ic} (lower bound of static initiation critical stress intensity factor) and is an approved exemption by the NRC in accordance with 10 CFR 50.60(b). K_{Ic} is obtained from a reference fracture toughness curve for reactor pressure vessel low alloy steels as defined in Appendix A to Section XI of the ASME Code and is approximated by the following equation:~~

$$K_{Ic} = 33.20 + 20.734e^{[0.0200(T - RT_{NDT})]}$$

~~where,~~

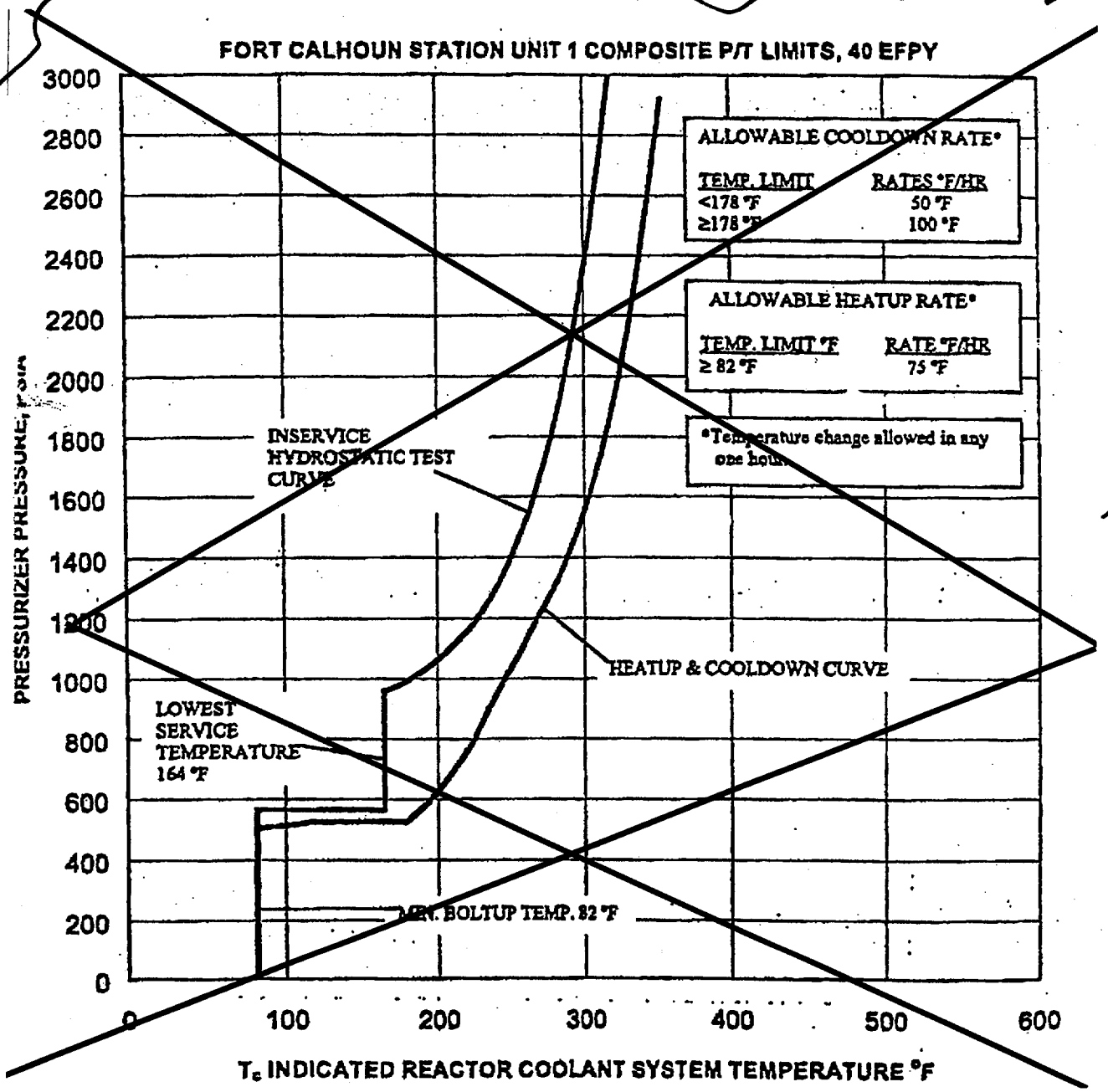
~~K_{Ic} = Crack initiation reference stress intensity factor, $Ksi\sqrt{in}$~~

~~T = temperature at the postulated crack tip, °F~~

~~RT_{NDT} = adjusted reference nil ductility temperature (also called ART) at the postulated crack tip, °F~~

~~For any instant during the postulated heatup or cooldown, K_{Ic} is calculated using the metal temperature at the tip of the flaw, as well as the value of ART at that flaw location.~~

Figure 2-1



~~RCS Pressure-Temperature Limits for Heatup, Cooldown, and Inservice Test~~	~~Omaha Public Power District Fort Calhoun Station-Unit No. 1~~

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Curves (Continued)

~~2) Pressure correction factors that account for the difference in pressure between the reactor vessel beltline and pressurizer pressure instrumentation due to elevation and RCP flow are as follows:~~

~~RCS Temperature < 210°F = 61 psi~~
~~RCS Temperature ≥ 210°F = 67 psi~~

~~3) Below 350°F, pressure instrumentation uncertainty is accounted for in the LTOP system setpoints. Above 350°F, a pressure instrumentation uncertainty of 50 psi is applied to Figure 2-1.~~

~~Lowest Service Temperature = 50°F + 100°F + 14°F = 164°F. As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel beltline was established at 50°F. Reference 2, Section III, NB-2332 requires a lowest service temperature of RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature a pressure of 20 percent of the system hydrostatic test pressure cannot be exceeded. Taking into account pressure correction factors for elevation and flow, this pressure is (.20)(3125) - 61 = 564 psia, where 61 psi is the pressure correction factor.~~

~~Boltup Temperature = 10°F + 14°F = 24°F. A conservative value of 82°F will be used and maintained. At pressure below 564 psia, a minimum vessel temperature must be maintained to comply with the manufacturer's specifications for tensioning the vessel head. This temperature is based on RT_{NDT} methods. This temperature corresponds to the measured 10°F RT_{NDT} of the reactor vessel flange, which is not subject to radiation damage, plus 14°F instrument error.~~

~~The temperature at which the heatup and cooldown rates change in Figure 2-1 reflects the point at which the most limiting heatup and cooldown rates with respect to the inlet temperature (T_c) change.~~

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

References:

(1) USAR, Section 4.2.2

(2) Technical Data Book IX, Fort Calhoun Station Unit No. 1, RCS Pressure-Temperature Limits Report

~~(2) ASME Boiler and Pressure Vessel Code~~

~~(3) USAR, Section 4.2.4~~

~~(4) USAR, Section 3.4.6~~

~~(5) WCAP-15443, Revision 0, Fast Neutron Fluence Evaluation for the Fort Calhoun Unit 1 Reactor Pressure Vessel, July 2000.~~

~~(6) Technical Specification 2.3(3)~~

~~(7) Article IWB-5000, ASME Boiler and Pressure Vessel Code, Section XI~~

~~(8) TR-O-MCM-001, Revision 1, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, August 1980.~~

~~(9) TR-O-MCM-002, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-265, March 1984.~~

~~(10) BAW-2226, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-275, November 1994.~~

~~(11) Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561).~~

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (continued)

2.1.6 Pressurizer and Main Steam Safety Valves

Applicability
Applies to the status of the pressurizer and main steam safety valves.

Objective
To specify minimum requirements pertaining to the pressurizer and main steam safety valves.

Specifications
To provide adequate overpressure protection for the reactor coolant system and steam system, the following safety valve requirements shall be met:

(1) The reactor shall not be made critical unless the two pressurizer safety valves are operable with their lift settings adjusted to ensure valve opening at 2485 psig ±1% and 2530 psig ±1%.[1]

(2) Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one operable safety valve shall be installed on the pressurizer. However, when in at least the cold shutdown condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

(3) At least four of the five Main Steam Safety Valves (MSSVs) associated with each steam generator shall be OPERABLE in MODES 1 and 2. Lift settings shall be at 985 psig +3/-2%, 1000 psig +3/-2%, 1010 psig +3/-2%, 1025 psig +3/-2%, and 1035 psig +3/-2%.[1]

a. With less than four of the five MSSVs associated with each steam generator OPERABLE, be in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within an additional 6 hours.

* (4) Two power-operated relief valves (PORVs) shall be operable during heatups and cooldowns when the RCS temperature is less than 515°F, and in Modes 4 and 5 whenever the head is on the reactor vessel and the RCS is not vented through a 0.94 square inch~~, or larger vent, to prevent violation of the pressure-~~temperature limits designated by the P-T limit Figure(s) shown in the PTLR~~Figure 2-1~~.

a. With one PORV inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, restore the inoperable PORV to operable within 7 days or be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

b. With both PORVs inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

c. With one PORV inoperable in Modes 4 ~~or 5~~, within one ~~hour ensure the~~ pressurizer steam space is greater than 50~~53~~% volume ~~(50.6% or less actual level)~~ and restore the inoperable PORV to operable within 7 days. ~~If adequate steam space cannot be established within one hour, then restore the inoperable~~ PORV to operable within 24 hours. If the PORV cannot be restored in the required time, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core.

Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least the refueling boron concentration at a temperature not less than 50°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig and a maximum of 275 psig with a tank level of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection pump is operable on each associated 4,160 V engineered safety feature bus. *** ****

f. One high-pressure safety injection pump is operable on each associated 4,160 V engineered safety feature bus.

g. Both shutdown heat exchangers are operable. *** ****

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from the motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

*** SEE TSI-95-03
****SEE TSI-95-05

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below ~~350~~385°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.

(4) Trisodium Phosphate (TSP) Dodecahydrate

During operating Modes 1 and 2, the TSP baskets shall contain ≥ 126 ft^3 of active TSP.

a. With the above TSP requirements not within limits, the TSP shall be restored within 72 hours.

b. With Specification 2.3(4)a required action and completion time not met, the plant shall be in hot shutdown within the next 6 hours and cold shutdown within the following 36 hours.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical. The energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable.

2.0 LIMITING CONDITIONS FOR OPERATION
2.3 Emergency Core Cooling System (Continued)

Components in excess of those allowed by Conditions a, b, d, and e may be inoperable provided they are returned to operable status within 1 hour when performing the quarterly recirculation actuation logic channel functional test (Table 3-2 item 20) under administrative controls. This allowance applies only to the remaining portion of Cycle 20 and all of Cycle 21. This prevents violating Technical Specifications or necessitating a unit shutdown due to inability to perform the quarterly recirculation actuation logic channel functional test. These administrative controls consist of stationing three dedicated operators at the Engineered Safeguards Features (ESF) panel controls in the control room. In this way, the following conditions are maintained and actions can be rapidly performed should a valid ESF actuation occur:

- the appropriate Safety Injection Refueling Water Tank (SIRWT) to Safety Injection (SI) and Containment Spray (CS) pumps suction valve control switch is maintained in the OPEN position (spring-return switch),
- the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch is maintained in the OPEN position (spring-return switch),
- the appropriate Recirculation Actuation Signal (RAS) lockout relays and initiating signal can be rapidly reset,
- the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch can be rapidly returned to the AUTO position,
- the appropriate SIRWT to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position, and
- the appropriate Containment Sump to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position.

The appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch and the appropriate SIRWT to SI and CS pumps suction valve control switch are held in the OPEN position during the test to enhance the reliability of the appropriate SI and CS pumps by maintaining the associated valves open.

References

(1) USAR, Section 14.15.1

(2) USAR, Section 6.2.3.1

(3) USAR, Section 14.15.3

(4) USAR, Appendix K

(5) Omaha Public Power District's Submittal, December 1, 1976

~~(6) Technical Specification 2.1.2, Figure 2-1B~~ Deleted

(7) USAR, Section 4.4.3

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core

2.10.1 Minimum Conditions for Criticality

Applicability

Applies to the status of the reactor coolant system during reactor criticality.

Objective

To prevent unanticipated power excursions of an unsafe magnitude.

Specifications

(1) ~~Except during physics tests at less than 10^{-1}% power,~~ The reactor shall not be made critical if the average reactor coolant temperature is below 515°F.

~~(2) In no case shall the reactor be made critical if the reactor coolant temperature is below NDTT +120°F.~~

(2) No more than one CEA at a time in a regulating or non-trippable group shall be exercised or withdrawn until after a steam bubble and normal water level as given in operating procedures are established in the pressurizer.

(3) Reactor coolant boron concentration shall not be reduced below that required for the Hot Shutdown Condition until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of each fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all CEA's withdrawn. However, variations in cycle core loading and the uncertainty of the calculation are such that it is possible that a slightly positive coefficient could exist.

The moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature. It is, therefore, prudent to restrict the operation of the reactor when reactor coolant temperatures are less than 515°F.

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.1 Minimum Conditions for Criticality (Continued)

If the shutdown margin required by the Hot Shutdown Condition is maintained, there is no possibility of an accidental criticality as a result of a change of moderator temperature or a decrease of coolant pressure. Normal water level is established in the pressurizer prior to the withdrawal of CEA or the dilution of boron so as to preclude the possible overpressurization of a solid reactor coolant system.

During physics tests, special operating precautions will be taken. In addition, the strong negative effect of the Doppler coefficient would limit the magnitude of a power excursion resulting from a reduction of moderator density.

~~The requirement that the reactor not be made critical below NDTT +120°F provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained relative to the NDTT of the reactor coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.~~

TABLE 3-5 (Continued)

		Test	Frequency
19.	Refueling Water Level	Verify refueling water level is ≥ 23 ft. above the top of the reactor vessel flange.	Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment.
20.	Spent Fuel Pool Level	Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during REFUELING OPERATIONS in the spent fuel pool.
21.	Containment Penetrations	Verify each required containment penetration is in the required status.	Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment.
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.

3.0 SURVEILLANCE REQUIREMENTS

3.3 Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance

Applicability

Applies to in-service surveillance of primary system components and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Objective

To ensure the integrity of the reactor coolant system and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Specifications

(1) Surveillance of the ASME Code Class 1, 2 and 3 systems, except the steam generator tubes inspection, should be covered by ASME XI Boiler & Pressure Vessel Code.

a. In-service inspection of ASME Code Class 1, Class 2, and Class 3 components, including applicable supports, and in-service testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a (g)(6)(i).

b. Surveillance of the reactor coolant pump flywheels shall be performed as indicated in Table 3-6.

c. A surveillance program to monitor radiation-induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained in accordance with 10 CFR Part 50 Appendix H.[1] Examinations results shall be used to update the PTLR.

(2) Surveillance of Reactor Coolant System Pressure Isolation Valves

a. Periodic leakage testing* on each valve listed in Table 2-9 shall be accomplished prior to entering the power operation mode every time the plant is placed in the cold shutdown

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

5.0 ADMINISTRATIVE CONTROLS

5.9.6 Reactor Coolant System (RCS) Pressure - Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Technical Specifications 2.1.1 and 2.1.2.

b. The analytical methods used in the PTLR shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CE NPSD-683-A, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," approved version as specified in the PTLR

2. WCAP-15443, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," approved version as specified in the PTLR

3. CEN-636, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials," approved version as specified in the PTLR

4. FC06876, "Performance of Low Temperature Overpressure Protection System Analysis Using RELAP5: Methodology Paper" approved version as specified in the PTLR

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period (i.e., the number of EFPY used in the P-T limit/LTOP analysis) and for any revision or supplement thereto.

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TECHNICAL SPECIFICATIONS - FIGURES

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DEFINITIONS

$\bar{E}$ - Average Disintegration Energy

$\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

RCS Pressure-Temperature Limits Report (PTLR)

The PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints which provide low temperature overpressure protection (LTOP) to assure the P-T limits are not exceeded during the most limiting LTOP event. The P-T limits and LTOP criteria in the PTLR are applicable through the effective full power years (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information provided in the PTLR.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F. Each steam generator shall be demonstrated operable by performance of the inservice inspection program specified in Section 3.17 prior to exceeding a reactor coolant temperature of 300°F.

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of the pressure and temperature limit Figure(s) shown in the PTLR.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) Low Temperature Overpressure Protection (LTOP)

(a) The LTOP enable temperature and RCP operations shall be maintained in accordance with the PTLR.

(b) The unit can not be placed on shutdown cooling until the RCS has cooled to an indicated RCS temperature of less than or equal to 300°F.

(c) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below the LTOP enable temperature stated in the PTLR unless there is a minimum indicated pressurizer steam space of at least 50% by volume.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table 2.9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition. Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supply deenergized.

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above 1.18 during all normal operations and anticipated transients.

When Specification 2.1.1(2) is applicable, the reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. Under these conditions, decay heat removal requirements are low enough that a single reactor coolant system (RCS) loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

When Specification 2.1.1(3) is applicable, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling pumps to be OPERABLE.

One of the conditions for which Specification 2.1.1(3) is applicable is when the RCS temperature (T_{cold}) is less than 210°F, fuel is in the reactor and all reactor vessel closure bolts are fully tensioned. As soon as a reactor vessel head closure bolt is loosened, Specification 2.1.1(3) no longer applies, and Specification 2.8 is applicable. Specification 2.8 also requires two shutdown cooling loops to be operable if there is less than 23 feet of water above the top of the core.

The restrictions on availability of the containment spray pumps for shutdown cooling service ensure that the SI/CS pumps' suction header piping is not subjected to an unanalyzed condition in this mode. Analysis has determined that the minimum required RCS vent area is 47 in^2. This requirement may be met by removal of the pressurizer manway which has a cross-sectional area greater than 47 in^2.

When reactor coolant boron concentration is being changed, the process must be uniform throughout the reactor coolant system volume to prevent stratification of reactor coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the reactor coolant is assured if one low pressure safety injection pump or one reactor coolant pump is in operation. The low pressure safety injection pump will circulate the reactor coolant system volume in less than 35 minutes when operated at rated capacity. The pressurizer volume is relatively inactive; therefore, it will tend to have a boron concentration higher than the rest of the reactor coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the reactor coolant system during the addition of boron.[1]

Both steam generators are required to be filled above the low steam generator water level trip set point whenever the temperature of the reactor coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

The bases for the LTOP system requirements are documented in the PTLR. I

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

The design cyclic transients for the reactor system are given in USAR Section 4.2.2. In addition, the steam generators are designed for additional conditions listed in USAR Section 4.3.4. Flooded and pressurized conditions on the steam side assure minimum tube sheet temperature differential during leak testing. The minimum temperature for pressurizing the steam generator steam side is 70°F; in measuring this temperature, the instrument accuracy must be added to the 70°F; limit to determine the actual measured limit. The measured temperature limit will be 73°F based upon use of an instrument with a maximum inaccuracy of ± 2°F and an additional 1°F safety margin.

References

(1) USAR Section 4.3.7

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

a. Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

b. Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

(a) Immediately initiate action to restore the temperature or pressure to within the limit.

(b) Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

(c) Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR shall be updated in accordance with the following criteria and procedures:

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

(a) The P-T limit Figure(s) are valid for a fast neutron ($E \geq 1$ MeV) fluence value and corresponding EFPY as stated in the PTLR.

(b) The limit line on the P-T limit Figure(s) shown in the PTLR shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV).

(c) The limit lines in the P-T limit Figure(s) shown in the PTLR shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 64°F as it is set by the RT_{NDT} of the reactor vessel flange and is not subjected to a fast neutron flux. The lowest service temperature shall remain at 164°F because components related to this temperature are also not subjected to a fast neutron flux.

(d) Technical Specification 2.3(3) shall be reviewed and revised as necessary each time the curves on the P-T limit Figure(s) as shown in the PTLR are revised.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon allowable heatup/cooldown rates and cyclic operation. Cycle dependent information such as the pressure-temperature limit curves, low temperature overpressure protection system limits, neutron fluence, and adjusted reference temperatures are contained in the PTLR, which was developed using the methodologies stated in Technical Specification 5.9.6(b) and in the PTLR[2].

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

References:

(1) USAR, Section 4.2.2

(2) Technical Data Book IX, Fort Calhoun Station Unit No. 1, RCS Pressure-Temperature Limits Report

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (continued)

2.1.6 Pressurizer and Main Steam Safety Valves

Applicability
Applies to the status of the pressurizer and main steam safety valves.

Objective
To specify minimum requirements pertaining to the pressurizer and main steam safety valves.

Specifications
To provide adequate overpressure protection for the reactor coolant system and steam system, the following safety valve requirements shall be met:

(1) The reactor shall not be made critical unless the two pressurizer safety valves are operable with their lift settings adjusted to ensure valve opening at 2485 psig ±1% and 2530 psig ±1%.[1]

(2) Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one operable safety valve shall be installed on the pressurizer. However, when in at least the cold shutdown condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

(3) At least four of the five Main Steam Safety Valves (MSSVs) associated with each steam generator shall be OPERABLE in MODES 1 and 2. Lift settings shall be at 985 psig +3/-2%, 1000 psig +3/-2%, 1010 psig +3/-2%, 1025 psig +3/-2%, and 1035 psig +3/-2%.[1]

a. With less than four of the five MSSVs associated with each steam generator OPERABLE, be in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within an additional 6 hours.

(4) Two power-operated relief valves (PORVs) shall be operable during heatups and cooldowns when the RCS temperature is less than 515°F, and in Modes 4 and 5 whenever the head is on the reactor vessel and the RCS is not vented through a 0.94 square inch or larger vent, to prevent violation of the pressure-temperature limits designated by the P-T limit Figure(s) shown in the PTLR.

a. With one PORV inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, restore the inoperable PORV to operable within 7 days or be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

b. With both PORVs inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

c. With one PORV inoperable in Modes 4 or 5, within one hour ensure the pressurizer steam space is greater than 50% volume and restore the inoperable PORV to operable within 7 days. If adequate steam space cannot be established within one hour, then restore the inoperable PORV to operable within 24 hours. If the PORV cannot be restored in the required time, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core.

Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least the refueling boron concentration at a temperature not less than 50°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig and a maximum of 275 psig with a tank level of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection pump is operable on each associated 4,160 V engineered safety feature bus.

f. One high-pressure safety injection pump is operable on each associated 4,160 V engineered safety feature bus.

g. Both shutdown heat exchangers are operable.

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from the motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below 350°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.

(4) Trisodium Phosphate (TSP) Dodecahydrate

During operating Modes 1 and 2, the TSP baskets shall contain $\geq$ 126 ft^3 of active TSP.

a. With the above TSP requirements not within limits, the TSP shall be restored within 72 hours.

b. With Specification 2.3(4)a required action and completion time not met, the plant shall be in hot shutdown within the next 6 hours and cold shutdown within the following 36 hours.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical. The energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable.

2.0 LIMITING CONDITIONS FOR OPERATION
2.3 Emergency Core Cooling System (Continued)

Components in excess of those allowed by Conditions a, b, d, and e may be inoperable provided they are returned to operable status within 1 hour when performing the quarterly recirculation actuation logic channel functional test (Table 3-2 item 20) under administrative controls. This allowance applies only to the remaining portion of Cycle 20 and all of Cycle 21. This prevents violating Technical Specifications or necessitating a unit shutdown due to inability to perform the quarterly recirculation actuation logic channel functional test. These administrative controls consist of stationing three dedicated operators at the Engineered Safeguards Features (ESF) panel controls in the control room. In this way, the following conditions are maintained and actions can be rapidly performed should a valid ESF actuation occur:

- the appropriate Safety Injection Refueling Water Tank (SIRWT) to Safety Injection (SI) and Containment Spray (CS) pumps suction valve control switch is maintained in the OPEN position (spring-return switch),
- the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch is maintained in the OPEN position (spring-return switch),
- the appropriate Recirculation Actuation Signal (RAS) lockout relays and initiating signal can be rapidly reset,
- the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch can be rapidly returned to the AUTO position,
- the appropriate SIRWT to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position, and
- the appropriate Containment Sump to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position.

The appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch and the appropriate SIRWT to SI and CS pumps suction valve control switch are held in the OPEN position during the test to enhance the reliability of the appropriate SI and CS pumps by maintaining the associated valves open.

References

(1) USAR, Section 14.15.1

(2) USAR, Section 6.2.3.1

(3) USAR, Section 14.15.3

(4) USAR, Appendix K

(5) Omaha Public Power District's Submittal, December 1, 1976

(6) Deleted

(7) USAR, Section 4.4.3

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core

2.10.1 Minimum Conditions for Criticality

Applicability

Applies to the status of the reactor coolant system during reactor criticality.

Objective

To prevent unanticipated power excursions of an unsafe magnitude.

Specifications

(1) The reactor shall not be made critical if the average reactor coolant temperature is below 515°F.

(2) No more than one CEA at a time in a regulating or non-trippable group shall be exercised or withdrawn until after a steam bubble and normal water level as given in operating procedures are established in the pressurizer.

(3) Reactor coolant boron concentration shall not be reduced below that required for the Hot Shutdown Condition until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of each fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all CEA's withdrawn. However, variations in cycle core loading and the uncertainty of the calculation are such that it is possible that a slightly positive coefficient could exist.

The moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature. It is, therefore, prudent to restrict the operation of the reactor when reactor coolant temperatures are less than 515°F.

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.1 Minimum Conditions for Criticality (Continued)

If the shutdown margin required by the Hot Shutdown Condition is maintained, there is no possibility of an accidental criticality as a result of a change of moderator temperature or a decrease of coolant pressure. Normal water level is established in the pressurizer prior to the withdrawal of CEA or the dilution of boron so as to preclude the possible overpressurization of a solid reactor coolant system.

During physics tests, special operating precautions will be taken. In addition, the strong negative effect of the Doppler coefficient would limit the magnitude of a power excursion resulting from a reduction of moderator density.

TABLE 3-5 (Continued)

		Test	Frequency
19.	Refueling Water Level	Verify refueling water level is ≥ 23 ft. above the top of the reactor vessel flange.	Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment.
20.	Spent Fuel Pool Level	Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during REFUELING OPERATIONS in the spent fuel pool.
21.	Containment Penetrations	Verify each required containment penetration is in the required status.	Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment.
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.

3.0 SURVEILLANCE REQUIREMENTS

3.3 Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance

Applicability

Applies to in-service surveillance of primary system components and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Objective

To ensure the integrity of the reactor coolant system and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Specifications

(1) Surveillance of the ASME Code Class 1, 2 and 3 systems, except the steam generator tubes inspection, should be covered by ASME XI Boiler & Pressure Vessel Code.

a. In-service inspection of ASME Code Class 1, Class 2, and Class 3 components, including applicable supports, and in-service testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a (g)(6)(i).

b. Surveillance of the reactor coolant pump flywheels shall be performed as indicated in Table 3-6.

c. A surveillance program to monitor radiation-induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained in accordance with 10 CFR Part 50 Appendix H.[1] Examinations results shall be used to update the PTLR. I

(2) Surveillance of Reactor Coolant System Pressure Isolation Valves

a. Periodic leakage testing* on each valve listed in Table 2-9 shall be accomplished prior to entering the power operation mode every time the plant is placed in the cold shutdown

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

5.0 ADMINISTRATIVE CONTROLS

5.9.6 Reactor Coolant System (RCS) Pressure - Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Technical Specifications 2.1.1 and 2.1.2.

b. The analytical methods used in the PTLR shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CE NPSD-683-A, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," approved version as specified in the PTLR

2. WCAP-15443, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," approved version as specified in the PTLR

3. CEN-636, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials," approved version as specified in the PTLR

4. FC06876, "Performance of Low Temperature Overpressure Protection System Analysis Using RELAP5: Methodology Paper" approved version as specified in the PTLR

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period (i.e., the number of EFPY used in the P-T limit/LTOP analysis) and for any revision or supplement thereto.

Fort Calhoun Station

Plant Specific

Pressure Temperature Limits Report

Fort Calhoun Station

Unit No. 1

TDB-IX

TECHNICAL DATA BOOK

Title: RCS PRESSURE AND TEMPERATURE LIMITS REPORT

FC-68 Number:

Reason for Change:

Requestor: Fredrick Jensen

Preparer: Fredrick Jensen

Correction (a):

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INTRODUCTION

The purpose of this Technical Data Book (TDB) section is to provide Fort Calhoun Station (FCS) with an administrative document that defines updating the pressure and temperature (P-T) limit curves and low temperature overpressure protection (LTOP) setpoints and delineates Nuclear Regulatory Commission (NRC) review requirements as defined in the Technical Specifications (TSs) Definitions section.

This Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) for FCS Unit No. 1 contains P-T limits corresponding to 40 Effective Full Power Years (EFPY) of operation. In addition, this report references the LTOP methodology and current analysis that contains the system limits and operating restrictions that protect the P-T limits from being exceeded during limiting LTOP events. Reference 8.1 allows the relocation of the P-T limit curves and LTOP system limits from the plants TSs and relocates them into a PTLR. Reference 8.2 is the topical PTLR that forms the basis for this document.

This PTLR will be updated prior to exceeding the adjusted reference temperature (RT_{NDT}) utilized to develop Figure 5-1. The PTLR, including any revisions or supplements thereto, shall be provided upon issuance of new heatup and cooldown curves to the NRC Document Control Desk with copies to the Regional Administrator and Senior Resident Inspector.

In addition, anytime it becomes necessary to change the methodology and/or any TSs that were used to develop data generated for this report, a license amendment will also be prepared describing the new methodology and/or TS change and will be submitted for NRC review and approval prior to implementation in this report.

(Note: FCS is currently licensed to operate until August 9, 2013. The values for fluence and the associated adjusted reference temperature (RT_{NDT}) values are valid beyond this date to 40 EFPY.)

1.0 NEUTRON FLUENCE VALUES

The most recent reactor vessel beltline neutron fluence has been calculated for the critical locations in Reference 8.3. It contains the following:

a) A description of the methodology used to perform the neutron fluence calculation.

b) A description of the computer codes used to calculate the neutron fluence values.

c) A description of how the computer codes for calculating the neutron fluence values were benchmarked.

The methodology stated in Reference 8.3 is consistent with the guidance of Draft Regulatory Guide DG-1053 (now Regulatory Guide RG 1.190) as stated by the NRC staff in References 8.4 and 8.5. The peak value of the fast neutron fluence (E > 1 Mev) at the clad metal to reactor vessel base metal interface for the limiting FCS 3-410 welds located at 60°/300° for 40 EFPY is 2.15×10^{19} neutrons per square centimeter (n/cm^2) with an associated uncertainty of $\pm$15.5%. The 3-410 weld for the 180° position is not limiting due to the fluence at this location is significantly less than at the 60° and 300° locations. In Cycle 14, extreme low radial leakage fuel management was implemented to reduce the reactor vessel fast neutron flux. This management scheme and the incorporation of surveillance data from other nuclear power plants per Reference 8.14 ensures that FCS has the potential to operate to August 9, 2033 without exceeding the 10 CFR 50.61 pressurized thermal shock (PTS) screening criteria as approved by Reference 8.5. Table 1-1 states the peak neutron fluence values (E > 1 Mev) for the 1/4T and 3/4T locations for 40 EFPY. (Reference 8.23) These values were determined using Equation 3 located in Reference 8.22. These neutron fluence values were used in the adjusted reference temperature (ART) calculations as stated in Section 4.0 of this report.

Table 1-1, Peak Neutron Fluence Values

Location	Fluence Value
Clad/Base Metal Interface	2.15×10^{19} n/cm^2
1/4T	1.4021×10^{19} n/cm^2
3/4T	0.59629×10^{19} n/cm^2

2.0 REACTOR VESSEL SURVEILLANCE PROGRAM

The reactor vessel surveillance program is described in Section 2, Reference 8.2. Thereactor vessel surveillance withdrawal schedule is located in Reference 8.6, Table 4.5-4. The baseline report describing the pre-irradiation evaluation of the FCS reactor surveillance materials are presented in Reference 8.7. The reports describing the post-irradiation evaluation of the FCS surveillance capsules are contained in References 8.8 - 8.10. Each removed capsule has been evaluated in accordance with the testing requirements of the version of ASTM-E-185 in effect at the time of capsule removal.

3.0 LTOP SYSTEM LIMITS

The LTOP system setpoints have been developed by making a comparison between the peak transient pressure for each limiting LTOP event and the P-T limit curve of Figure 5-1 to ensure that the P-T limit curve is not exceeded.

These system setpoints and additional limitations for LTOP have been established based on NRC-accepted methodology and are described in Reference 8.15. (Note: The methodology described in Section 3.0 of Reference 8.2 was not used for the determination of the LTOP system setpoints.)

The LTOP analysis which contains the current system setpoints and operating restrictions to ensure the P-T limit curve is not exceeded during a limiting LTOP event is located in Reference 8.16. The applicable operating restrictions stated in Reference 8.16 will be maintained in the TSs. Reference 8.21 contains the methodology for incorporating the Reference 8.16 setpoints into the LTOP system actuation circuitry. These conservative values will then be used for incorporation into TDB Figures.

The LTOP enable temperature is 350°F. (Reference 8.24)

4.0 BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURE

The calculation of the ART for the reactor vessel beltline region has been performed using the NRC-accepted methodologies as described in Section 4.0, Reference 8.2. Application of surveillance data was used to refine the chemistry factor and the margin term in Reference 8.14. (See Section 7.0) The limiting weld for FCS is the 3-410 weld located at the 60°/300° position using weld wire heat 12008/13253. The RT_{PTS} value for the limiting weld is 250°F at the end of license (August 9, 2013) and 268°F at the end of license extension (August 9, 2033).

The limiting ART values in the beltline region for the FCS Unit 1 corresponding to 40 EFPY are as follows: (Reference 8.23)

Location	ART	Material
Clad/base metal interface	261.56°F	Weld 3-410 at 60° and 300°
1/4T	237.76°F	Weld 3-410 at 60° and 300°
3/4T	187.97°F	Weld 3-410 at 60° and 300°

5.0 PRESSURE-TEMPERATURE LIMITS USING LIMITING ART IN THE P-T CURVE CALCULATION

The analytical methods used to develop the beltline RCS P-T limits are based on NRC reviewed methodologies as discussed in Section 5.0 of Reference 8.2. The NRC approved the use of ASME Code Case N-640 that allows the use of K_{IC} to calculate the reference stress intensity factor K_{IR} values for the reactor pressure vessel as a function of temperature per Reference 8.17. The limit for the maximum pressure in the vessel is 100 percent of the pressure satisfying Paragraph G-2215 of the 1996 Edition of Appendix G to the ASME Code for establishing LTOP limit setpoints. Additionally, a relief exemption was granted by the NRC to apply CE NSSS methods for determining P-T limit curves.

The ferritic reactor pressure vessel materials that have accumulated neutron fluences in excess of 1.0 x 10^{17} n/cm^2 (E > 1 Mev) regardless of whether the materials are located within the region immediately surrounding the active core have been evaluated (Reference 8.18). This evaluation concluded that the limiting material remained the lower shell axial welds, 3-410 A/C.

6.0 MINIMUM TEMPERATURE REQUIREMENTS IN THE P-T CURVES

The minimum temperature requirements specified in Reference 8.20 are applied to the P-T limit curves using the NRC-reviewed methodologies as described in Section 6.0 of Reference 8.2.

The minimum temperature values applied to the P-T limit curves for FCS Unit 1 corresponding to 40 EFPY are (Note: These limits were calculated in Reference 8.19 and incorporates instrument uncertainty):

a. Minimum Boltup Temperature: 64°F.

b. Minimum Hydrostatic Temperature test limits: See Figure 5-1. (Note: The in-service hydrostatic test curve is developed in the same manner as the heatup and cooldown curves with the exception that a safety factor of 1.5 is used in lieu of 2.)

c. Lowest Service Temperature: 164°F.

d. Flange Limit:

   1) Normal Operation: 144°F.

   2) Hydrostatic and Leak Testing: 114°F.

e. Core Critical Temperature Limit: 300°F. (Note: This minimum temperature value that must be attained prior to reactor criticality is based on fracture mechanics

considerations and is significantly less than the 515°F temperature value that is required by TS 2.10.1(1).)

In the development of P-T limits for CE NSSS's, the intent is to utilize the more conservative of either the lowest service temperature or the other minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of the pre-service hydrostatic test pressure (PHTP). The "minimum pressure criteria" specified in Reference 8.20 serves as a regulatory breakpoint in the development of P-T limits and is defined as 20% of PHTP. For CE NSSS plants, the PHTP is defined as 1.25 times the design pressure (Note: Design pressure = 2500 psia). The function of minimum pressure in the development of P-T limits is to provide a transition between the various temperature only based P-T limits, such as minimum bolt up and the lowest service temperature of flange limits.

For FCS Unit 1, the minimum pressure is calculated as follows:

Minimum Pressure = (1.25 x design pressure) x 0.20

= 1.25 x 3125 psia x 0.20

= 625 psia

Therefore, when the pressure correction factors (Reference 8.19) are applied to 625 psia, the minimum pressure(s) are as follows:

Actual RCS Temperature < 210°F = 564 psi

Actual RCS Temperature $\geq$ 210°F = 558 psi

The pressure of 564 psi is the most significant value due to the RCS can not exceed this pressure until RCS temperature is greater than the lowest service temperature value stated in Section 6.0 item 'c' above. The lowest service temperature is the limiting minimum temperature value and is incorporated into Figure 5-1. The heatup and cooldown limit curve is more conservative than the minimum pressure value in the temperature range specified, but the in-service hydrostatic test curve is limited by the regulatory requirement (Reference 8.20).

7.0 APPLICATION OF SURVEILLANCE DATA TO ART CALCULATIONS

Post-irradiation surveillance capsule test results for FCS Unit 1 are given in References 8.8 - 8.10. Additional reports containing surveillance capsule data from other nuclear power plants are located in References 8.11 - 8.13. These additional surveillance reports, along with others that are contained in Reference 8.14, were deemed credible and approved for use in the FCS surveillance program as stated by the NRC staff in Reference 8.5. Additionally, Reference 8.5 requires the following:

a) Future core loadings are limited to the core neutron leakage to values similar to those for Cycles 15 and 16 which will satisfy the requirement of end of license (August 9, 2013) fluence accumulation of 1.728×10^{19} neutrons/cm^2 to the limiting welds.

b) Caution is exercised to preclude misloading any of the peripheral assemblies which would invalidate the loading requirements.

c) New data from the Mihama Unit 1, Diablo Canyon Unit 1 and Palisades plants is assessed by the FCS staff as it becomes available, since the data from these plants were used in the FCS PTS analysis.

The use of these "Sister" reactor vessels (as stated in Section 7.0 item 'c' above) is required to ensure that FCS does not exceed PTS screening criteria during its extended lifetime (August 9, 2033).

A review of the surveillance programs of Mihama Unit 1 (12008/27204), Diablo Canyon Unit 1 (27204), Palisades Supplemental Capsules (27204), and the FCS W-275S Capsule (27204 and 12008/13253) concluded further data should be available for use in the FCS reactor vessel surveillance program as follows: (Note: The values in parentheses correspond to weld wire heat numbers.)

a) Mihama Unit 1 (Weld Wire Heat 12008/27204)

The data from Capsules 1-3 were used in Reference 8.14. The removal schedule for the remaining Mihama Unit 1 capsules are:

1) Capsule 4 was scheduled for removal in 2001; results are expected in 2002.
2) Capsule 5 is scheduled for removal in 2010; results are expected in 2011.
3) Capsule 6 is currently considered in standby with no scheduled removal date.

b) Palisades (Weld Wire Heat 27204/27204)

The removal schedule for the Palisades capsules are:

1) Capsule SA-60-1 was pulled and evaluation data are found in internal report ATI-99-006-002 (8/4/99). The capsule report should be submitted to the NRC in 2002. The data was used in Reference 8.14.
2) Capsule SA-240-1 was pulled and has been evaluated by Framatome. The capsule report should be submitted to the NRC in 2002.

c) Diablo Canyon Unit 1 (Weld Wire Heat 27204)

The removal schedule for the Diablo Canyon Unit 1 capsules and the status of the results that are reported to the NRC are:

1) Capsule DC1-S data are contained in Reference 8.11 and was used in Reference 8.14.
2) Capsule DC1-Y data are contained in Reference 8.12 and was used in Reference 8.14.
3) Capsule DC1-V is scheduled for removal in 2002 with an expected May 2003 report submittal to the NRC. This is the last of the three original capsules containing 27204 weld material.
4) Capsule DC1-C (supplemental) is scheduled for removal in 2004 with an expected report submittal to the NRC in 2005. This supplemental capsule was fabricated using reconstituted Charpy specimens from Capsule DC1-Y.
5) Additional supplemental capsules (A, B, and D) from the FCS 1-410 B nozzle dropout, were installed in Cycle 5. They are currently considered to be in standby with no scheduled removal date.

8.0 REFERENCES

8.1 NRC GL 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996

8.2 CE NPSD-683-A, Rev 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001

8.3 WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," July 2000 [Contained in Letter LIC-00-0064 from OPPD (W. G. Gates) to NRC (Document Control Desk), dated August 3, 2000]

8.4 Safety Evaluation by the Office of NRR Related to Amendment Number 197 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated March 27, 2001

8.5 Safety Evaluation by the Office of NRR Related to Amendment Number 199 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated June 6, 2001

8.6 USAR Section 4.5.3, Revision 3, dated May 29, 2002

8.7 TR-O-MCD-001, "Evaluation of Baseline Specimens Reactor Vessel Materials Irradiation Surveillance Program," March 22, 1977

8.8 TR-O-MCM-001, Revision 1, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-225," dated August 28, 1980 [Contained in Letter LIC-81-0011 from OPPD (W.C. Jones) to NRC (H.R. Denton), dated January 23, 1981]

8.9 TR-O-MCM-002, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-265," dated March 7, 1984 [Contained in Letter LIC-84-124 from OPPD (W.C. Jones) to NRC (D.G. Eisenhut), dated April 25, 1984]

8.10 BAW-2226, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-275," dated July 21, 1994 [Contained in Letter LIC-94-0250 from OPPD (T.L. Patterson) to NRC (Document Control Desk), dated December 9, 1994]

8.11 WCAP-11567, "Analysis of Capsule S from the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," December 1987

8.12 WCAP-13750, "Analysis of Capsule Y from the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," July 1993

8.13 WCAP-13440, "Supplemental Reactor Vessel Radiation Surveillance Program for the Pacific Gas and Electric Company Diablo Canyon Unit No. 1," December 1992

8.14 CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials," dated July 2000 [Contained in Letter LIC-00-0064 from OPPD (W. G. Gates) to NRC (Document Control Desk), dated August 3, 2000]

8.15 FC06876, Rev. 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper"

8.16 FC06877, Rev. 0, "Low Temperature Overpressure Protection (LTOP) analysis, Revision 1"

8.17 Safety Evaluation by the Office of NRR Related to Amendment Number 207 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated April 22, 2002

8.18 Letter LTR-CI-01-25, Rev. 0 from WEC (S. T. Byrne) to OPPD (J. Jensen), "Assessment of Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel," dated December 18, 2001

8.19 EA-FC-01-022, Rev. 0, "Pressure and Temperature Limit Curve for 40 EFPY"

8.20 10 CFR 50 Appendix G, "Fracture Toughness Requirements"

8.21 FC06863, Rev. 1, "LTOP Setpoint Instrument Loop Uncertainty and LTOP Trip Curve Development"

8.22 Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials"

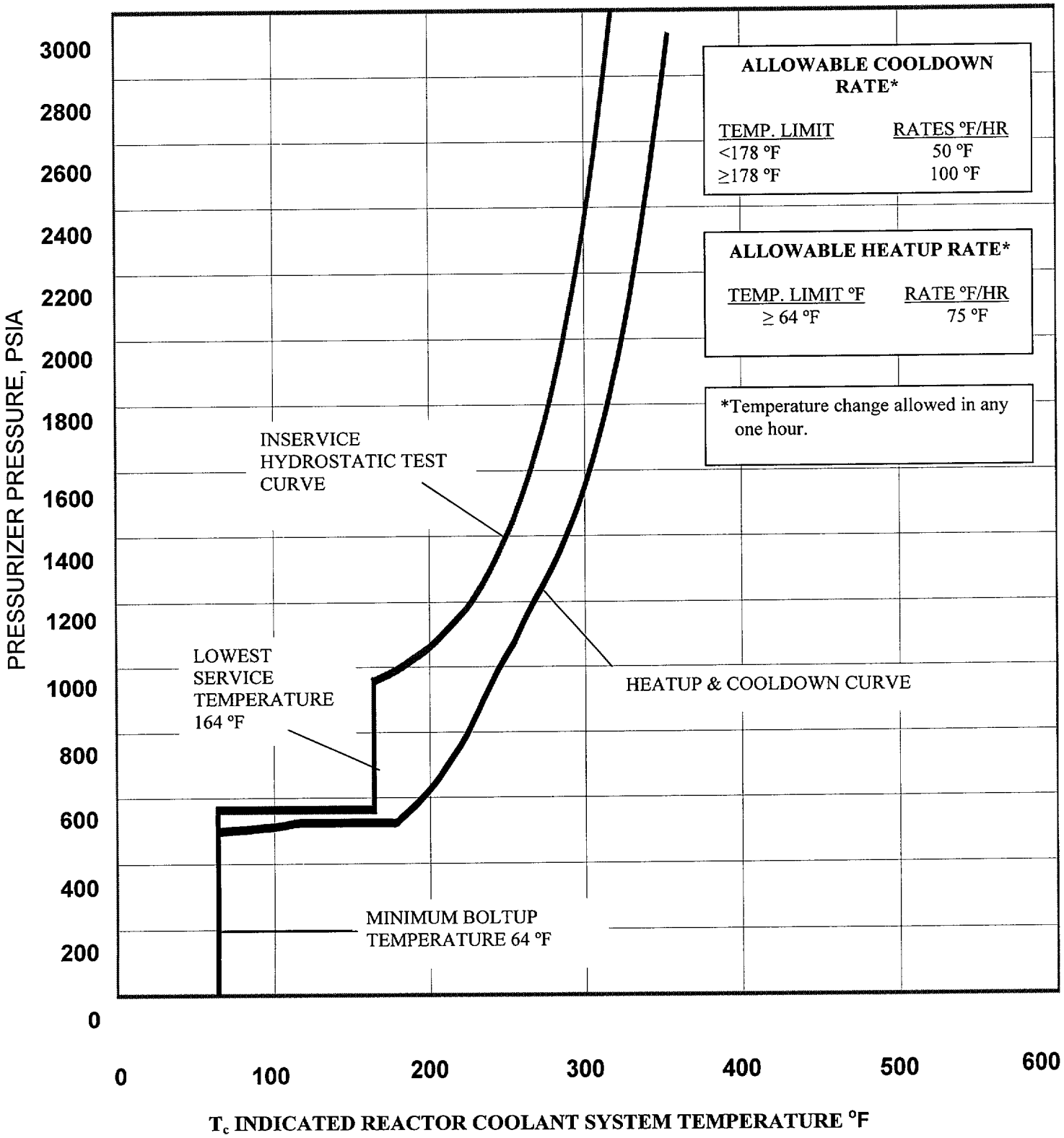
8.23 FC06799, Rev. 0, "40 EFPY Pressure and Temperature Limit Curve Inputs"

8.24 EA-FC-02-025, Rev. 0, "Development of the RCS PTLR"

Figure 5-1

FORT CALHOUN STATION UNIT 1 COMPOSITE P/T LIMITS, 40 EFPY

RCS PRESSURE-TEMPERATURE LIMITS FOR HEATUP, COOLDOWN, AND INSERVICE HYDROSTATIC TEST



Non-Proprietary

References from Attachment 1

Reference 10.2

FC06877, Rev. 0

"Low Temperature Overpressure Protection (LTOP) analysis, Revision 1"