Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

April 10, 2003
LIC-03-0052

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

References:
1. Docket No. 50-285
2. Letter from OPPD (D. J. Bannister) to NRC (Document Control Desk) dated October 8, 2002, Fort Calhoun Station Unit No. 1 License Amendment Request, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" (LIC-02-0109)
3. Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenhoure) dated March 19, 2003, Request For Additional Information Related To Ft. Calhoun Station Pressure-Temperature Limit Report Submittal (TAC No. MB6468) (NRC-03-048)

SUBJECT: Response to Request for Additional Information, Pressure-Temperature Limits Report Amendment Request

In support of the license amendment request, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" (Reference 2), the Omaha Public Power District (OPPD) provides the attached response to the Nuclear Regulatory Commission's (NRC's) Request for Additional Information of Reference 3.

The Fort Calhoun Station PTLR will be issued after approval of the requested license amendment and contain the items as committed to in the attachments of this letter. (OPPD Action Request 32338)

I declare under penalty of perjury that the forgoing is true and correct. (Executed on April 10, 2003).

A001

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

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/RLJ/rlj

Attachments:

1. Response to NRC Request for Additional Information, Pressure-Temperature Limits Report
2. Revised Technical Data Book RCS Pressure and Temperature Limits Report

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Winston & Straw

Response to NRC Request for Additional Information
Pressure-Temperature Limits Report (PTLR)

NRC Question 1:

Peak neutron fluence values for each reactor pressure vessel (RPV) beltline material should be included in the PTLR, if separate fluence values are actually used in the licensing basis calculation of adjusted reference temperature (ART) and pressurized thermal shock reference temperatures (RT_{PTS}) values for each material. Alternatively, where such material-specific fluence values are located (e.g., Table x.x in Ref. 8.x) could be clearly documented in the PTLR. This comment goes to the intent of demonstrating that the information on the limiting material has been included in the PTLR.

OPPD Response:

The peak neutron fluence values at the clad/base metal interface, the 1/4 T location and the 3/4 T location for each reactor pressure vessel beltline material is incorporated into Section 1.0 of the revised PTLR (Attachment 2).

NRC Question 2:

In section 2.0, it would be appropriate to note in the PTLR exactly which edition of ASTM Standard E 185 is being used by the licensee in order to comply with Appendix H withdrawal schedule requirements. In addition, a statement regarding testing to ASTM Standard E 185-82 (i.e., the 1982 edition, consistent with the requirements in 10 CFR Part 50, Appendix H) would also appropriate.

OPPD Response:

OPPD has incorporated the required information into Section 2.0 of the revised PTLR (Attachment 2).

NRC Question 3:

ART or RT_{PTS} values for each RPV material should be included in the PTLR. Alternatively, where each material's ART or RT_{PTS} values are located (e.g., Table x.x in Ref. 8.x) could be clearly documented in the PTLR. The staff recognizes the material ART values of interest for the purpose of the development of the Ft. Calhoun pressure-temperature (P-T) limit curves are associated with 40 effective full power years (EFPY) of operation, whereas the RT_{PTS} values associated with the vessel PTS evaluation are associated with end of license conditions per 10 CFR 50.61. Documentation of ART or RT_{PTS} values in the PTLR should also clearly state for what operational time the values are calculated. This comment goes to the intent of demonstrating that the information on the limiting material has been included in the PTLR.

OPPD Response:

The ART values for each reactor pressure vessel material has been incorporated into Section 4.0 of the revised PTLR (Attachment 2). Additionally, OPPD has included statements in Section 4.0 of the revised PTLR (Attachment 2) to clearly identify the operational time associated with the ART and RT_{PTS} values.

NRC Question 4:

In Section 5.0 of the PTLR, the term "relief exemption" is used. "Relief exemption" is a non-standard term. In the context of the section, the word "exemption" would be appropriate.

OPPD Response:

The wording has been corrected to "exemption" in Section 5.0 of the revised PTLR (Attachment 2).

NRC Question 5:

Confirm, with the exception of the information in Section 2.0 of Attachment 1 to LIC-02-0109, that the proposed PTLR P-T limit curves are exactly the same as those currently in the Ft. Calhoun Technical Specifications (TS). The P-T limit curves in TS approved by SE dated 4/22/2002.

OPPD Response:

The P-T limit curve for the proposed PTLR is exactly the same as the one approved by the NRC in the safety evaluation dated 4/22/02 except for those modifications as stated in Section 2.0 of Attachment 1 to LIC-02-0109.

NRC Question 6:

Consistent with the information noted in Column 2 of the Table in Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," provide a concise list of all documentation (e.g., CEN-683 (Rev. 6)-A, as modified by exemption request for CC N-640, etc.), which, in total, will comprise the Ft. Calhoun "integrated PTLR methodology."

OPPD Response:

The integrated PTLR methodology documents are as follows:

a) 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.
b) 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].
c) 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.
d) CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials," dated July 2000.
e) FC06876, Rev. 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper."
f) FC06877, Rev. 0, "Low Temperature Overpressure Protection (LTOP) analysis, Revision 1."
g) 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.
h) Letter LTR-CI-01-25, Rev. 0 from Westinghouse Electric Company (S. T. Byrne) to OPPD (J. Jensen), "Assessment of Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel," dated December 18, 2001.
i) WCAP-15741, Rev. 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001 [Contained in Letter LIC-01-0107 from OPPD (R. L. Phelps) to NRC (Document Control Desk), dated November 8, 2001].
j) [When the NRC approves Fort Calhoun Stations exemption request for the use of CEs methodology for performing P-T limit curves, this document will also be part of the integrated PTLR methodology.]

NRC Question 7:

The application states that the current "criticality limit" for Ft. Calhoun is 300 °F, and that such a limit is used in lieu of a core critical operation P-T limit curve. However, it appears that an imposed criticality limit of 300 °F may not be adequate to bound the 10 CFR Part 50, Appendix G core critical operation P-T limit curve all the way up to normal operating pressure. Explain how a core criticality limit of 300 °F is adequate to bound operation up to normal operating pressure, or propose a modified limit which would bound the core critical operation P-T limit curve. Further, whatever the criticality limit is, it should be noted on P-T limit figure in PTLR consistent with other minimum temperature requirements (see GL 96-03 Table Items 5 and 6.)

OPPD Response:

Technical specification 2.10.1(1) requires that the reactor can not be made critical unless reactor coolant system temperature is greater than 515 °F which is more conservative than the core critical temperature limit required by 10 CFR 50 Appendix G. OPPD will ensure that when the P-T limit curve must be modified that the new core critical peak temperature limit will be verified to be less than 515 °F, or else the core critical P-T limit curve will be included on the P-T limit curve of the PTLR. Additionally, Section 6.0, item 'e' will be updated as applicable. The criticality temperature has been added to the P-T limit Figure in the revised PTLR (Attachment 2).

NRC Question 8:

Consistent with GL 96-03 Table Items 5 and 6, the minimum hydrotest temperature should be identified on the P-T limit curve in the PTLR.

OPPD Response:

The minimum hydrotest temperature has been identified on the P-T limit curve in the revised PTLR (Attachment 2).

NRC Question 9:

To be consistent with the requirements in GL 96-03 table (item 7), section 7.0 of the PTLR needs to be revised to include data and calculations related to the determination of RPV material chemistry factors (CFs) from any relevant surveillance data. Calculations should include not only those related to determination of CF from applicable surveillance data, but also those calculations related to the evaluation of the credibility of the surveillance data. Alternatively, making PTLR Reference 8.14, report CEN-636, Revision 2, an attachment to the PTLR would be adequate to resolve this item.

OPPD Response:

CEN-636, Revision 2 will be added as an Attachment to the PTLR after the subject PTLR is approved by the NRC. Please refer to page 14 of the revised PTLR (Attachment 2).

NRC Question 10:

For clarity, how instrument uncertainty is treated in the development of the PTLR P-T limit curves should be discussed in the PTLR.

OPPD Response:

Section 5.0 of the revised PTLR (Attachment 2) has been modified to discuss how instrument uncertainty has been incorporated in the PTLR P-T limit curves.

NLIC-03-0052
Attachment 2

REVISED

TECHNICAL DATA BOOK

RCS PRESSURE AND 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):

Table of Contents

List of Figures

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 except as modified by the individual Sections.

This PTLR will be updated prior to exceeding the adjusted reference temperature (ART (RT_{NDT})) utilized to develop Figure 5-1. The PTLR, including any revisions or supplements thereto, shall be provided upon issuance of P-T limit 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, the associated ART values, Figure 5-1, and LTOP system setpoints 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. (Note: The uncertainty associated with the fluence values stated in Reference 8.3 is ± 15.5%.) This report/reference 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 the safety evaluations contained in References 8.4 and 8.5.

The values of fast neutron fluence (E > 1 Mev) used in the ART calculations in Section 4.0 are located in Table 1-1 and are applicable for 40 EFPYs. (Note: The fluence associated with 40 EFPYs versus 48 EFPYs was used in the ART calculations for Figure 5-1 to prevent a reduction in the operating window between the P-T limit and the reactor coolant pump net positive suction head curves.) The 1/4 T and the 3/4 T neutron fluence values were calculated as follows:

a) The clad/base metal interface fluence values for the plates and circumferential weld use the peak neutron value listed in Table 6.2-1 of Reference 8.3 for 40 EFPY. This is due to these materials would be exposed to the highest fluence.
b) The clad/base metal interface fluence value used for the limiting axial welds was the value located at the 60° position for 40 EFPY. The axial welds for the 180° position is not limiting due to the fluence at this location is significantly less than at the 60° and 300° locations. The non-limiting 2-410 welds at the 0°, 120°, and 240° positions are located in geometrically symmetric locations as the 3-410 welds at 60°, 180°, and 300 ° positions. 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 the NRC in Reference 8.5.
c) Equation 3 of Reference 8.22 was then used to calculate the 1/4 T and the 3/4 T fluence values as shown in Table 1-1.

(Note: The values in parentheses in Table 1-1 refers to weld wired heat numbers.)

Table 1-1, Neutron Fluence Values for 40 EFPY

Reactor Pressure Vessel Material	1/4 T	3/4 T
D 4802-1	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4802-2	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4802-3	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4812-1	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4812-2	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4812-3	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
2-410	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (12008/13253)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (12008/27204)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (13253)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (27204)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
9-410	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2

2.0 REACTOR VESSEL SURVEILLANCE PROGRAM

The reactor vessel surveillance program is described in Section 2, Reference 8.2. The reactor vessel surveillance withdrawal schedule is located in Reference 8.6, Table 4.5-4. This schedule meets the requirements of ASTM-E-185-82 (Reference 8.25). 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 References 8.15 and 8.16. (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 approximately 250°F at a clad/base metal interface fluence of 1.728 x 10^{19} n/cm^2 at the end of license (August 9, 2013) and 268°F at a clad/base metal interface fluence of 2.43 x 10^{19} n/cm^2 at the end of license extension (August 9, 2033).

The ART values in the beltline region for FCS Unit 1 corresponding to 40 EFPY are listed in Table 4-1. (Note: The limiting ART value for the 1/4 T and 3/4 T (Weld 3-410, Weld Wire Heat 12008/13523) was incorporated into Figure 5-1 (References 8.19 and 8.23).)

Table 4-1, ART Values for Reactor Vessel Materials for 40 EFPY

Reactor Pressure Vessel Material	1/4 T	3/4 T
D 4802-1	131.56	112.27
D 4802-2	120.45	103.55
D 4802-3	120.76	103.60
D 4812-1	132.51	113.03
D 4812-2	111.14	95.89
D 4812-3	111.14	95.89
2-410	106.88	85.64
3-410 (12008/13253)	237.76	187.97
3-410 (12008/27204)	213.98	164.69
3-410 (13253)	196.26	150.84
3-410 (27204)	223.72	172.30
9-410	233.11	188.89

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 for FCS 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 in 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, an 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.

Figure 5-1 was developed in Reference 8.19 and modified per Reference 8.24. Uncertainty was incorporated into Figure 5-1 as follows (Reference 8.19):

a) Above the LTOP enable temperature (350°F), pressure instrument uncertainty is incorporated into the P-T limit curve and below this temperature it is not. (Note: Pressure instrument uncertainty is not applied below the LTOP enable temperature due to it being incorporated into the LTOP system setpoint curve). A pressure instrumentation uncertainty of 50 psi is being used, which bounds the wide and narrow range pressurizer pressure instruments that operators would use to determine RCS pressure.
b) The temperature uncertainty used is 14°F which bounds the instruments that operators would use to determine RCS temperature.

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: 515°F per TS 2.10.1(1). (Note: This TS limit is more conservative than the core critical temperature limit required by Reference 8.20. Whenever the P-T limit curve of Figure 5-1 is modified, it must be verified that the new core critical peak temperature limit is less than 515°F, or else the core critical P-T limit curve must be included on Figure 5-1 and Section 6.0, item 'e' must be updated.)

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 2500 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 (Attachment 1), 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 surveillance data from 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 2003 or 2004. 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 2003 or 2004.

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 (27204) 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," dated 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 [This document is located in the Attachment 1.].

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."

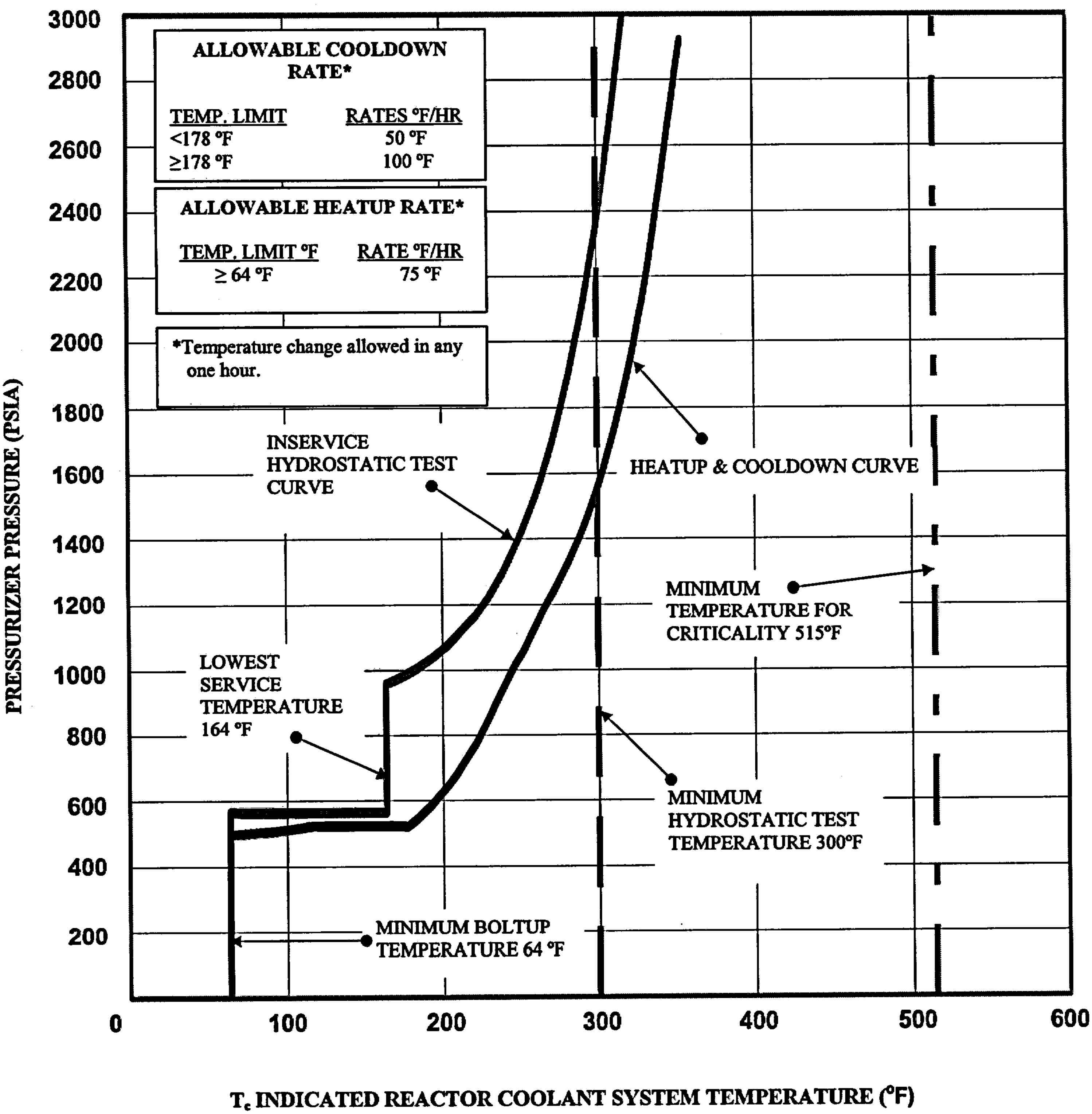
8.24 EA-FC-02-025, Rev. 0, "Development of the RCS PTLR."

8.25 WCAP-15741, Rev. 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001 [Contained in Letter LIC-01-0107 from OPPD (R. L. Phelps) to NRC (Document Control Desk), dated November 8, 2001].

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



Attachment 1:

CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials," dated July 2000