August 15, 2003

Mr. R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station, FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MB6468)

Dear Mr. Ridenoure:

The Commission has issued the enclosed Amendment No. 221 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1 (FCS). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 8, 2002, as supplemented by letters dated April 10, June 4, July 31, and August 5, 2003.

The October 8, 2002, submittal proposed the following: (1) the use of a pressure temperature limits report (PTLR), (2) change the minimum boltup temperature, (3) revise the low temperature overpressure protection (LTOP) methodology and analysis, (4) perform the LTOP analyses "in-house," (5) change the LTOP enable temperature, (6) modify TS 2.10.1 to exactly specify the reactor coolant system (RCS) temperature at which the reactor can be made critical, and (7) add a TS for a maximum pressure value for the safety injection tanks. The enclosed safety evaluation approves the use of the PTLR at the FCS and associated TS changes. The amendment modified the following TSs to reflect the implementation of the PTLR: relocated TS Figure 2-1 (RCS Pressure - Temperature Limits for Heatup, Cooldown, and In-service Test) into Figure 5-1 of the PTLR, defined the PTLR in Definitions; added references to the PTLR in TS 2.1.1(8); TS 2.1.1(11); TS 2.1.2 and 2.1.2 References; TS 2.1.6(4); TS 2.3(1)(c); TS 2.3(3); TS 2.3 References; TS 2.10.1; Table 3-5, item 23; TS 3.3(1)(c); and added TS 5.9.6. The following TS Bases sections were modified to reflect the implementation of the PTLR: TS 2.1.1; TS 2.1.2; and TS 2.10.1.

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

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 221 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

August 15, 2003

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 221
License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee) dated October 8, 2002, as supplemented by letters dated April 10, June 4, July 31, and August 5, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance. The amendment shall be implemented within 30 days from the date of issuance, including submitting the first Pressure Temperature Limits Report to the NRC Document Control Desk with copies to the Region IV Regional Administrator and Resident Inspector.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Jack Donohew for/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 15, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 221

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

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

REMOVE	INSERT
iii	iii
viii	viii
8	8
2-2a	2-2a
2-2b	2-2b
2-2c	2-2c
2-2d	2-2d
2-3	2-3
2-4	2-4
2-5	---
2-6	---
Figure 2-1	---
2-7	---
2-7a	2-7a
2-15	2-15
2-20	2-20
2-22	2-22
2-23b	2-23b
2-48	2-48
2-49	2-49
3-20f	3-20f
3-21	3-21
5-10a	5-10a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 221 TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated October 8, 2002, as supplemented by letters dated April 10, June 4, July 31, and August 5, 2003, Omaha Public Power District (OPPD) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1 (FCS). OPPD requested to implement a pressure temperature limit report (PTLR) for FCS. OPPD submitted technical specification (TS) changes intended to be consistent with Technical Specification Task Force Traveler 419 (TSTF-419), which modifies the original guidance on PTLR development in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." In addition, OPPD requested to change TS 2.1.2(c), the minimum bolt-up temperature, and TS 2.3(1)c, to add a maximum safety injection tank (SIT) pressure limit. OPPD also requested that the following TS Bases sections be changed to reflect the implementation of the PTLR: TS 2.1.1; TS 2.1.2, and TS 2.10.1.

The supplemental letters dated April 10, June 4, July 31, and August 5, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 24, 2003 (68 *FR* 37579).

2.0 REGULATORY EVALUATION

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the acceptability of a facility's proposed PTLR methodology and initial PTLR based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Appendix H to 10 CFR Part 50; Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"; GL 92-01, Revision 1, "Reactor Vessel Structural Integrity"; GL 92-01, Revision 1, Supplement 1; Standard Review Plan (SRP) Section 5.3.2; and GL 96-03. Appendix G to 10 CFR Part 50 requires that facility pressure-temperature (P-T) limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure

Vessel Code. Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code.

The attributes of the vessel fluence methodology are described in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." RG 1.190 is based on General Design Criteria (GDC) 14, 30 and 31 of Appendix A to 10 CFR Part 50. In this context, GDC-14 relates to an extremely low probability of leakage from the pressure coolant boundary; GDC-30 relates to the design of the reactor coolant boundary; and GDC-31 relates to material embrittlement and the effect of irradiation.

The review requirements for the low temperature overpressure protection (LTOP) transients are described in SRP Section 5.2.2. SRP Section 5.2.2 is based on GDC-15 as it relates to the reactor coolant boundary design margin and GDC-31 as it relates to embrittlement and the effect of irradiation.

GL 96-03 addresses the technical information necessary for a licensee's implementation of a PTLR. GL 96-03 establishes the information which should be included in an acceptable PTLR methodology (which will be used to develop the PTLR), and the information which should be included within the PTLR itself. These information criteria are principally addressed in a table contained in Attachment 1 of GL 96-03 entitled, "Requirements for Methodology and PTLR," and are subdivided into seven technical elements which must be addressed by the licensee.

GL 96-03 also addresses the appropriate modifications to the administrative controls section of a facility's TS which are necessary to implement a PTLR. TSTF-419 provides guidance on an alternative set of TS administrative control section changes which may be made to implement a PTLR.

3.0 EVALUATION

3.1 Licensee Submittal and Evaluation

In the licensee's October 8, 2002, submittal, OPPD provided the following information to be reviewed by the staff:

1. A proposed reference which included all of the documents necessary to define the "PTLR methodology" that would be implemented at FCS;

2. A license amendment request including the TS markup pages showing how the PTLR methodology will be documented in the TS;

3. The proposed FCS PTLR demonstrating the information that OPPD intended to include in the PTLR;

4. An exemption request related to OPPD's intended use of the methodology in Topical Report CE NPSD-683-A, Revision 6 for the determination of flaw stress intensity factors due to thermal stress loadings (K_{It}) [reviewed and granted by the NRC by letter dated July 30, 2003].

The licensee responded on April 10, 2003, to an NRC staff request for information (RAI) and, in their response, provided revisions to items (1) and (3) listed above. Item 1 was subsequently revised by letters dated July 31 and August 5, 2003, to reflect the approval of the exemption request related to the use of the methodology in Topical Report CE NPSD-683-A, Revision 6 for the determination of flaw stress intensity factors due to thermal stress loadings (K_{It}) and the addition of the reference to RELAP5/Mod 3.2.d.

Regarding item (1), OPPD has provided the following list of documents as those comprising its complete PTLR methodology:

a. CE NPSD-683-A, Revision 6, "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.

c. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 199 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station, Unit No. 1, dated June 7, 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, Revision 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper."

f. FC06877, "Low Temperature Overpressure Protection (LTOP) Analysis, Revision 1."

g. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 207 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station, Unit No. 1, dated April 22, 2002.

h. Letter LTR-CI-01-25, Revision 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, Revision 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001.

j. Exemption by the Office of Nuclear Reactor Regulation Related to the Approval for the Use of Combustion Engineering Methodology in CE NPSD-683-A, Revision 6, as Basis for the FCS PTLR, dated July 30, 2003.

k. Letter from Information Systems Laboratory (W. Acieri) to OPPD (J. F. Jensen), "WCA-09-2002: Transmittal of RELAP5/Mod 3.2.d," August 2, 2002.

These documents will be referred to as "Reference (a)" through "Reference (k)" in the remainder of this safety evaluation (SE).

Regarding item (3), a revised version of the proposed FCS "Technical Data Book RCS Pressure and Temperature Limits Report," was provided as Attachment 2 to the licensee's April 10, 2003, letter. This revision of the proposed PTLR was reviewed by the NRC staff against the criteria in GL 96-03.

The licensee concluded in its submittal, as revised by its RAI response, that the information provided was sufficient to address the regulatory requirements for the implementation of a PTLR given in Section 2.0 above.

3.2 NRC Staff's Evaluation of PTLR Methodology and Proposed References (Items 1 and 3)

All components of the RCS are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to 10 CFR Part 50, TS limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation. These limits are defined by P-T limit curves for RCS heatup, cooldown, LTOP, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during RCS heatup and cooldown evolutions, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LTOP system controls RCS pressure at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating Appendix G. The FCS systems for pressure relief at low RCS temperatures (the overpressure protection system, or OPPS) consists of the pressurizer power-operated relief valves (PORVs) and OPPS unit-specific setpoints and enable temperatures. The OPPS is reevaluated each time the P-T limit curves are revised to ensure that it is capable of performing its intended function.

In order to relocate its TS P-T curves and the values of the LTOP system setpoints and enable temperatures to a PTLR, a licensee must obtain NRC review and approval of a methodology for their development. Relocating the P-T curves and the values of the LTOP setpoints and enable temperatures from the TS to a PTLR does not eliminate the regulatory requirement to operate the plant in accordance with the limits specified in Appendix G to 10 CFR Part 50. Once a

PTLR is established, the plant's TS will require and control operation within the limits in the PTLR. Only the figures, values, and parameters associated with the P-T limits and LTOP system limits are to be moved to the PTLR.

A licensee may develop its PTLR methodology for NRC approval in accordance with GL 96-03. This generic letter delineates the staff's recommendations for both the methodology and the PTLR itself, including, but not limited to, the requirements of Appendix G to 10 CFR Part 50, and approved exemptions. There are seven technical elements from the "Requirements for Methodology and PTLR," table in Attachment 1 to GL 96-03 that a licensee must address for approval of a PTLR. OPPD's submittal has addressed these seven technical elements. For each of these technical elements, the staff has reviewed OPPD's submittal to determine whether adequate information had been included within: (1) the PTLR methodology (which will be used to develop the PTLR) and (2) the draft PTLR itself. Each of the seven review topics is addressed below.

3.2.1 Reactor Vessel Material Surveillance Program

For addressing the licensee's reactor vessel material surveillance program, GL 96-03 states that, at a minimum, a licensee's PTLR methodology shall:

> Briefly describe the RPV surveillance program. The licensee should identify by title and number the report containing the RPV surveillance program and surveillance capsule reports.

The NRC staff concluded in its March 16, 2001, SE concerning CE NPSD-683-A, Revision 6 that Reference (a) was adequate to meet the minimum requirements for a licensee's PTLR methodology for this technical element. In addition, OPPD has incorporated supplementary information (Reference (i)) which provides additional details regarding the basis for the FCS RPV surveillance program. Hence, since OPPD will include References (a) and (i) in its PTLR methodology, the NRC staff concludes that this criteria is satisfied.

GL 96-03 also states that, at a minimum, a licensee's PTLR shall:

> Provide the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located. Reference the surveillance capsule reports by title and number if RPV material adjusted reference temperatures (ARTs) are calculated using surveillance data.

The NRC staff reviewed the information provided in the draft FCS PTLR. The draft PTLR references FCS Updated Safety Analysis Report Section 4.5.3 which contains the FCS surveillance capsule withdrawal schedule. Further, OPPD referenced all applicable surveillance capsule reports which provide information relevant to the calculation of FCS RPV material ARTs in Section 8.0 of the draft PTLR. Hence, the NRC staff concludes that this criterion is satisfied.

3.2.2 Calculation of RPV Material ARTs

For addressing the licensee's calculation of RPV material ARTs, GL 96-03 states that, at a minimum, a licensee's PTLR methodology shall:

> Describe the method for calculating the ARTs using NRC RG 1.99, Revision 2.

The NRC staff concluded in its March 16, 2001, SE concerning CE NPSD-683-A, Revision 6 that Reference (a) was adequate to meet the minimum requirements for a licensee's PTLR methodology for this technical element. Hence, since OPPD will include Reference (a) in its PTLR methodology, the staff concludes that this criterion is satisfied.

GL 96-03 also states that, at a minimum, a licensee's PTLR shall:

> Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (i.e., locations 1/4 of the way through the thickness of the ferritic RPV wall from the inside and outside surface) and PWRs [pressurized water reactors] shall identify the RPV's limiting RT_{PTS} value in accordance with 10 CFR 50.61.

The required information was provided in Section 4.0 of the draft PTLR. Hence, the NRC staff concludes that this criterion is satisfied.

3.2.3 Calculation of P-T Limit Curves Based on Limiting Material ART Values

For addressing the licensee's calculation of P-T limit curves based on limiting material ART values, GL 96-03 states that, at a minimum, a licensee's PTLR methodology shall:

> Describe the application of fracture mechanics in constructing P-T limit curves based on Appendix G to Section XI of the ASME Code and NRC Standard Review Plan Section 5.3.2.

The NRC staff concluded in its March 16, 2001, SE concerning Reference (a) (CENPSD-683-A, Revision 6) that Reference (a) was adequate to meet the minimum requirements for the licensee's PTLR methodology for this technical element, with the limitation that licensees who wish to apply the methodology contained in Reference (a) must be granted an exemption in accordance with the requirements of 10 CFR 50.60. The necessity for the licensee's exemption request stems from the fact that while the NRC staff reviewed and approved the methodology of Reference (a), the staff could not conclude that the methodology would always provide results which were at least as conservative as those which would result from the specific requirements found in Appendix G to 10 CFR Part 50.

Consistent with the limitation established in the staff's March 16, 2001, SE and the requirements of 10 CFR 50.60, OPPD requested an exemption to utilize the methodology in Reference (a) as the basis for generating FCS P-T limit curves. The NRC has, in a separate action, granted the exemption requested by OPPD (Reference (j)). Hence, since OPPD will include References (a) and (j) in its PTLR methodology, the NRC staff concludes that this criterion is satisfied.

GL 96-03 also states that, at a minimum, a licensee's PTLR shall:

> Provide the P-T limit curves for heatup, cooldown, criticality, and hydrostatic and leak rate testing.

In the licensee's draft PTLR, Figure 5-1 provided a P-T limit curve for inservice hydrostatic testing and a P-T limit curve applicable to both heatup and cooldown of the RPV with the core

not critical which were developed using the licensee's proposed PTLR methodology. These P-T limit curves, with the exception of specific minimum temperature requirements (discussed in Section 3.2.4 below), were previously reviewed and approved by the NRC in Reference (g) and had been incorporated as Figure 2-1 in the FCS TS. Hence, based on the NRC staff's prior approval of the proposed PTLR P-T limit curves in Reference (g), the NRC staff concludes that this criterion is satisfied.

3.2.4 P-T Limit Curve Minimum Temperature Requirements

For addressing the licensee's incorporation of P-T limit curve minimum temperature requirements as specified by Appendix G to 10 CFR Part 50, GL 96-03 states that, at a minimum, a licensee's PTLR methodology shall:

> Describe how the minimum temperature requirements in Appendix G to 10 FR Part 50 are applied to P-T limit curves.

The NRC staff concluded in its March 16, 2001, SE concerning Reference (a) (CE NPSD-683-A, Revision 6) that Reference (a) was adequate to meet the minimum requirements for licensee's PTLR methodology for this technical element. Hence, since OPPD will include Reference (a) in its PTLR methodology, the NRC staff concludes that this criterion is satisfied.

GL 96-03 also states that, at a minimum, a licensee's PTLR shall:

> Identify minimum temperatures on the P-T limit curves such as minimum boltup temperature and hydrotest temperature.

In the licensee's draft PTLR, minimum temperature requirements for RPV boltup and hydrostatic testing were identified on Figure 5-1. Previously, in the FCS TS, the minimum RPV boltup temperature on the P-T limit curve had been given as 82°F. In its PTLR submittal, the licensee proposed to modify the minimum RPV boltup temperature to 64°F and to extend the P-T limit curves on Figure 5-1 down to 64°F.

The effective requirement for minimum boltup temperature in Appendix G to 10 CFR Part 50 is that it must be equivalent to the highest nil-ductility reference temperature (RT_{NDT}) of any material in the RPV flange region that is highly stressed by the bolt preload (i.e., stud tensioning). For the FCS RPV, the highest RT_{NDT} value for a highly stressed RPV flange material is 60°F. The licensee then added 4°F of margin to this value to account for uncertainty in RPV flange material temperature measurement to arrive at the proposed boltup temperature of 64°F.

The NRC staff reviewed the information provided by the licensee in its PTLR submittal and information previously submitted and entered into the NRC staff's Reactor Vessel Integrity Database. The NRC staff confirmed the limiting RT_{NDT} value cited for the FCS RPV flange materials. Based on this information, the NRC staff concludes that the use of a 64°F boltup temperature for the FCS RPV is consistent with the requirements given in Appendix G to 10 CFR Part 50 and is therefore acceptable.

OPPD also identified the minimum hydrostatic test temperature (300°F) on Figure 5-1 of the proposed PTLR. The minimum hydrostatic test temperature may be defined as the lowest temperature at which a facility's RPV may be taken to normal operating pressure and an acceptable hydrostatic/leak test performed. Based on the NRC staff's evaluation of the extended Figure 5-1 in the proposed PTLR, the NRC staff concludes that a minimum hydrostatic test temperature of 300 °F is consistent with the aforementioned definition and is therefore acceptable.

Finally, in lieu of the incorporation of a P-T limit curve for core critical operation, OPPD invoked a "minimum temperature for criticality" of 515°F on the proposed PTLR Figure 5-1. To meet the requirements of Appendix G to 10 CFR Part 50, a licensee should, in general, establish a core critical P-T limit curve which is consistently 40°F more conservative than the most limiting core non-critical heatup or cooldown curve. At normal operating pressure, the limiting point on the proposed FCS P-T limit curve for core non-critical heatup and cooldown is approximately 340°F. The corresponding point on an acceptable P-T limit curve for core critical operation would be approximately 380°F. The licensee's choice of a minimum temperature for criticality of 515°F is, therefore, conservative with respect to the requirements for an acceptable P-T limit curve for core critical operation up to and beyond normal operating pressure. Hence, the NRC staff concludes that the minimum temperature for criticality imposed by OPPD on Figure 5-1 is conservative with respect to the requirements in Appendix G to 10 CFR Part 50 and is therefore an acceptable substitute for a core critical operation P-T limit curve.

Based on the discussion above, the NRC staff concludes that this criterion has been satisfied.

3.2.5 Evaluation and Use of RPV Surveillance Data

For addressing the licensee's evaluation and use of RPV surveillance data, GL 96-03 states that, at a minimum, a licensee's PTLR methodology shall:

> Describe how the data from multiple surveillance capsules are used in the ART calculation. Describe the procedure used if measured value of transition temperature shift from the surveillance capsules exceed predicted values. If data from other facilities is being used, identify the facilities which are providing data and identify by title and number the NRC SE which approved the use of the data for the facility.

For the evaluation of the FCS RPV, OPPD makes extensive use of RPV weld surveillance data from other facilities (Diablo Canyon Unit 1, Palisades, and Mihama Unit 1 [Japan]) as well as plate surveillance data from FCS's own RPV surveillance program. The procedures used by OPPD to evaluate RPV surveillance data are discussed in detail in Reference (d) (CEN-636, Revision 2) and are based on application of the guidance in RG 1.99, Revision 2. The NRC staff has previously reviewed Reference (d) and approved OPPD's methodology and application of RPV surveillance data by SE in Reference (c), subject to the condition stipulated in Reference (c) that as additional RPV surveillance data becomes available it shall be incorporated into OPPD's evaluation. Hence, based on the NRC staff's prior approval of OPPD's treatment of RPV surveillance data and OPPD's incorporation of References (c) and (d) into the proposed FCS PTLR methodology, the NRC staff concludes that this criterion is satisfied.

GL 96-03 also states that, at a minimum, a licensee's PTLR shall:

> Provide supplemental data and calculations of the chemistry factor in the PTLR if the RPV surveillance data are used to establish RPV material ART values. The PTLR shall also include an evaluation of RPV surveillance data to determine if they meet the credibility criteria in RG 1.99, Revision 2 and the results of this evaluation.

In order to address the above, OPPD included report CEN-636, Revision 2 as Attachment 1 to the proposed FCS PTLR which was submitted in the licensee's April 10, 2003, RAI response. The NRC staff has reviewed the information in CEN-636, Revision 2 and finds that it meets the recommendation stated above for what information must be included in a PTLR. Hence, the NRC staff concludes that this criterion has been satisfied.

3.2.6 Neutron Fluence Values

GL 96-03 states that a licensee's PTLR shall:

> Describe how the neutron fluence is calculated (reference new RG 1.190 when it is issued.) Describe transport calculation methods including computer codes and formulas used to calculate the neutron fluence. Provide the values of neutron fluences that are used in the ART calculation.

The RPV fluence was calculated before March 2001, i.e., the time when RG 1.190 was issued. However, the fluence methodology adhered to the guidance in the Draft Guide (DG) 1053. The technical content of DG 1053 is the same as that of RG 1.190, and therefore the methodology is acceptable for referencing in the proposed PTLR. Reference (b) provides a description of the methodology and computer codes used to calculate the neutron fluence values. In Reference (c) the staff determined that Reference (b) is consistent with the guidance of DG 1053. Hence, since OPPD will include References (b) and (c) in its PTLR methodology, the NRC staff concludes that this criterion has been satisfied.

3.2.7 Evaluation of LTOP Implementation

3.2.7.1 LTOP Heat Injection Transient

A heat addition transient could occur by the spurious activation of a reactor coolant pump (RCP). When the RCS is cooled and the temperature decreases to less than the steam generator (SG) temperature, reverse heat transfer takes place, the pressurizer goes solid and the primary RCS pressure rises. A review of the initial conditions showed that a conservative set of initial conditions was chosen to produce a bounding analysis (October 8, 2002, letter, Attachment 4, Table 3). For the heat injection transient, it is assumed that the letdown cooling is not available; a maximum decay heat level is assumed; pressurizer heater control fails with the heater in the on-position; and an RCP is on. A conservative boundary assumption is made that the SG temperature is at 314°F and stays at this value, while the reactor vessel, hot leg and cold leg cool down. The limiting heat injection case is mitigated by the existence of a minimal size bubble in the pressurizer. The steam bubble maintains pressure, and allows the RCS temperature to equilibrate with the SG temperature, thus terminating the transient. The

necessary minimum bubble size in the pressurizer is specified in the TSs. In addition, the use of 20 percent additional decay heat is considered a conservative assumption in the calculations, maximizing the heat addition. A total of four heat injection transients were analyzed; three with an assumed 314°F SG temperature and one at the more conservative value of 340°F.

The NRC staff reviewed the analyses regarding pressurization due to heat addition and agreed that these analyses represent a bounding analysis of the expected RCS response. The full complement of assumptions forms a conservative set of inputs to perform a bounding analysis. The results of the heat injection transients were compared to the P-T limits and showed that there is a large margin to the P-T limits, demonstrating that the proposed LTOP function and the enabling point will protect the P-T limits. The NRC staff's review concluded that the transient analyses meets the acceptance criteria in SRP Section 5.2.2 and therefore are acceptable.

3.2.7.2 LTOP, Mass Injection Transient

Conservative assumptions were also made in the representation of the initial plant conditions, i.e., the most limiting values allowed by the TSs. Only one power operated relief valve (PORV) was credited in the analyses. No credit was taken for letdown flow, RCS volume expansion and vessel metal expansion. The pressurizer was assumed water solid and the pressurizer heater was assumed failed in the on position. Decay heat was maximized by assuming the fastest cooldown rate assuring the shortest cooldown time and the highest decay power level. The mass injection flow rate is bounded by the combination of the available centrifugal charging pumps (CCPs) and/or high pressure safety injection pumps (HPSIs). In particular, the safety injection pump flow is increased by 10 percent over the corresponding pump curve design value. The number of CCPs and HPSI pumps which are operable is defined in the TSs as a function of temperature.

Nine mass injection cases were analyzed. The injection water temperature was assumed at 250°F, while the mass flow rate (water density) was that corresponding to 32°F. This assumption bounds any injection temperatures and is conservative. All three RCPs were assumed to be running. With the reactor at zero power, the primary side SG inventory and the RCS coolant temperatures are at equilibrium. The SG tubes were modeled as being fully insulated so that heat was not lost, which adds to the conservatism.

The amendment application is proposing 350°F for the LTOP enabling temperature setpoint and a calculated pressure versus temperature function (in the form of a graph). If the pressurizer pressure exceeds the pressure corresponding to the 350°F in the cold leg, the PORVs open to relieve RCS pressure.

The results of the nine mass injection transients were compared to the P-T limits and showed that there is a large margin to the P-T limits, demonstrating that the proposed LTOP function and the enabling point will protect the P-T limits. The NRC staff's review concluded that the transient analyses meet the acceptance criteria in SRP Section 5.2.2, and therefore are acceptable.

3.2.7.3 RELAP Code Evaluation

The NRC staff has reviewed the thermal-hydraulic analyses documented by the OPPD in the RCS PTLR supporting the LTOP methodology for the FCS nuclear steam supply system. The thermal-hydraulic analyses were documented in Attachment 4 of the October 8, 2002, submittal. The review of the thermal-hydraulic calculations contained in Attachment 4 resulted in an RAI which was transmitted to OPPD by letter dated May 21, 2003. OPPD responded to the RAI by letter dated June 4, 2003. The NRC staff's review of the RAI, along with independent calculations, demonstrated the conservatism inherent in the OPPD methodology for computing RCS pressure response during low temperature and pressure operation. This methodology will be used by OPPD to ensure that the peak pressures during any LTOP event at FCS will not exceed any P-T limit from 50°F or greater. The NRC staff's review is provided below.

The NRC staff's review of the thermal-hydraulic LTOP methodology is founded on the RELAP5 analysis of the RCS pressure response bounding the range of conditions expected during an LTOP event. Of particular importance is the OPPD modeling of the pressurizer during an LTOP event. The pressurizer model is considered very conservative since the OPPD model does not include the effects of wall heat transfer. Since the peak pressure during an in-surge event is controlled by the transfer of heat between the steam region and the pressurizer walls, omission of the wall heat transfer from the RELAP5 pressurizer model will produce a more rapid pressurization, as well as extremely high peak pressures that bound all in-surge transient data. NRC staff independent in-surge calculations with RELAP5, using the Massachusetts Institute of Technology (MIT) over-pressurization data (Test ST4), showed that this assumption is very conservative. The NRC staff's calculation predicted a significant pressure increase that exceeded the peak pressure by over a factor of two prior to the time of peak pressure in the data. With this approach, the OPPD methodology is considered conservative and bounds the expected plant pressure response during an in-surge transient. Furthermore, this conservative modeling approach precludes the need for sensitivity studies on time-step and nodalization, as well as additional comparisons to the MIT pressurization test data using the OPPD RELAP5 model. As a consequence, responses to all RAI questions pertaining to the system pressure response and RELAP5 modeling were not required. For example, RELAP5 predicts excessive condensation in components containing steam above a liquid region such as the pressurizer during in-surges. RELAP5 also failed to predict the peak pressure for many of the MIT pressurization tests for this reason. Most importantly because the adiabatic RELAP5 pressurized model overwhelms this RELAP5 limitation, it eliminates the need for additional test comparisons as well as sensitivity studies to demonstrate the adequacy of the model as was requested in the RAIs.

In addition, conference calls were held with OPPD to clarify the RAI and OPPD's responses. The conference calls addressed the NRC staff's additional questions on the RAI response. Again, because of the adiabatic pressurizer modeling approach, the key issues regarding peak pressure predictions with this model and the need for additional calculations are eliminated. This conservative modeling approach eliminated the need for responses to several questions from the RAI and precluded the need for further clarification on portions of the RAI where questions remained.

Furthermore, staff independent calculations using the Henry-Fauske critical flow model showed that should the PORV open, the relief capacity of a single PORV with liquid expulsion exceeded the maximum injection rate of 132 gpm when RCS pressure exceeded the shutoff head of the HPSI pumps. When the RCS pressure was below the HPSI shut off head, the PORV limits the RCS pressure to the bounds set by the LTOP P-T limit curve given in Attachment 4 of the October 8, 2002, letter. This includes conditions down to and including an RCS pressure of 392.1 psia and an RCS temperature of 64°F with two high pressure injection pumps operating. Calculations were performed over the range of pressures from 1742.2 psia and 350°F down to and including 392.1 psia and 64°F. As such, the staff agrees that should the valve lift, a single PORV will maintain RCS pressure within the limits of the proposed P-T limit curve in Attachment 4.

The NRC staff disagrees with the approach used by OPPD to provide validation and bench-marking of the RELAP5 code used in the methodology. Specifically, the citing of analyses by others outside OPPD, such as the SCDAP/RELAP5 analyses and MIT pressurization test data comparisons performed by the Idaho National Engineering and Environmental Laboratory, are not considered valid bench-marking citations pertinent to the OPPD plant-specific model. Because the adiabatic model employed by OPPD is considered bounding, the need for detailed in-house bench-marking and the additional sensitivity studies by OPPD are eliminated. OPPD and its consultants should strive to perform their own calculations and benchmarks with their specific version of the code to which the plant model is eventually applied. In this manner, OPPD will be able to maintain and demonstrate its ability and expertise in exercising its particular methodology for the events and conditions of interest. OPPD, in this manner would maintain complete control and knowledge of all of the model inputs and code modifications that produce the benchmarks that are needed to justify their particular plant model and RELAP5 code version. As such, by letter dated August 5, 2003, OPPD added to TS 5.9.6, reference (k) which lists RELAP/Mod3.2.d as the code of record for the LTOP analysis.

The NRC staff performed independent calculations of the MIT pressurization tests to demonstrate that the OPPD model is very conservative, by severely overpredicting the peak pressure in the tests and producing a much earlier and rapid pressurization. The NRC staff also performed independent calculations of the liquid flow rates through a single PORV over the range of pressures and temperatures covered by the P-T limit curve that also demonstrated that a single PORV can maintain RCS pressure below the P-T limit curve for FCS. The staff further agrees that the pressurization analyses due to mass addition and heat addition represent a bounding simulation of the expected plant response. The assumptions and initial conditions assumed in the analyses should assure these analyses bound the plant response during an LTOP event. Based on the NRC staff's review of the RAI and the independent staff calculations and the limitations stated above, the methodology documented in OPPD's October 8, 2002, letter, is found to be acceptable.

3.2.8 PTLR Methodology Conclusions

Based on the NRC staff's review of the information provided in OPPD's October 8, 2002, submittal, as amended by OPPD's April 10, June 4, July 31, and August 5, 2003, RAI responses, the staff concludes that:

(a) OPPD has defined an acceptable PTLR methodology which is consistent with the regulatory requirements given in Section 2.0 of this SE. This acceptable methodology is documented in References (a) through (k) in Section 3.0 of this SE.

(b) OPPD provided as an attachment to its April 10, 2003, RAI response, a proposed PTLR which contains information consistent with NRC regulatory requirements and is acceptable for incorporation into the FCS licensing basis.

3.3 TS Changes (Item 2)

The PTLR changes affect the definitions, limiting conditions for operation (LCOs), and administrative controls sections of the TS. Specifically, OPPD has modified its TS by adding:

- The definition of a named formal report (PTLR or a similar document) that would contain the explanations, figures, values, and parameters (currently contained in TS) derived in accordance with an NRC-approved methodology and consistent with all of the design assumptions and stress limits for cyclic operation. (Definitions Section)

- References to the PTLR that require maintaining the P-T limits within the limits specified in the PTLR, in place of the existing P-T limits explanations, figures, values, and parameters. (Affected LCOs)

- A reporting requirement to submit the PTLR to the NRC, when it is issued, for each reactor vessel fluence period. The PTLR administrative controls specification must reference the documents from the NRC that approved the supporting P-T methodology. (Administrative Controls Section)

3.3.1 TS Definition Change

The definitions section of the TS is modified to include a definition of the PTLR to which the figures, values, and parameters for P-T and OPPS limits will be relocated. These figures, values, and parameters are established in accordance with an NRC-approved methodology that maintains the P-T acceptance limits and the P-T limits of the safety analysis. As noted in the definition, plant operation within these limits is addressed by individual specifications. The TS definition section added a definition for "RCS Pressure -Temperature Limits Report (PTLR)" as the following:

> 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 (PORV) 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 methodology are used as the bases for the information provided in the PTLR.

The staff has concluded that this definition is consistent with the improved standard technical specifications and TSTF-419 and, therefore is acceptable.

3.3.2 Reference Changes

The following TSs are revised to replace the numerical values of the P-T and OPPS limits with a reference to the PTLR that provides these values:

TS 2.1 Reactor Coolant System

TS 2.1.1	Operable Components: Delete (5), delete Figure 2-1 (RCS Pressure-Temperature Limits for Heatup and Cooldown, and Inservice Test).
TS 2.1.1(8)	Delete Figure 2-1.
TS2.1.1(11)(a)	Add "The LTOP enable temperature and reactor coolant pump (RCP) operations shall be maintained in accordance with the PTLR."
TS 2.1.1(11)(b)	Add "The unit cannot be placed on shutdown cooling until the RCS has cooled to an indicated temperature of less than or equal to 300°F."
TS 2.1.1(11)(c)	Delete "at least one of the following conditions is met."
TS 2.1.1	Bases also modified.
TS 2.1.2	Heatup and Cooldown Rate: Deletion of referencing Figure 2-1 and revised TS and Basis.
TS 2.1.2 (c)	Add "The boltup temperature limit line shall remain at 64°F. The lowest service temperature shall remain at 164°F."
TS 2.1.6	Pressurizer and Main Steam Valves: Modified to delete Figure 2-1 and indicate pressurizer steam space is greater than 50% volume.

TS 2.3 Emergency Core Cooling System

TS2.3.(1)c	Add safety injection tanks (SITs) TS: "pressurized to at least 240 psig and a maximum of 275 psig with a tank level of at least 116.2 inches and a maximum level of 128.1 inches."
TS 2.3.(3)	Modified to: RCS cold leg temperature is below 350°F.

TS 2.10	Reactor Core	
TS 2.10.1		Minimum conditions for criticality: Modified TS and Basis.
TS 2.10.1(1)		Modified as: "The reactor shall not be made critical if the average reactor coolant temperature is below 515°F."
TS Table 3-5		Add item 23, P-T limit curve.
TS 3.0	Surveillance Requirements	
TS 3.3		Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance.
TS 3.3 (1)c		Add "Examinations results shall be used to update the PTLR."

The NRC staff has reviewed these changes and agrees that they are administrative in nature except for TS 2.1.2(c) and TS 2.3.(1)c which are reviewed separately in Sections 3.5 and 3.6 of this SE. These changes reflect the implementation of the PTLR and therefore are acceptable.

3.3.3 Addition of TS 5.9.6

In Section 5.0, Administrative Controls, TS 5.9.6, "Reactor Coolant System (RCS) Pressure-Temperature Limits Report PTLR)," is added. TS 5.9.6 requires the licensee to submit the PTLR, upon issuance, to the NRC Document Control Desk with copies to the regional administrator and resident inspector. The report provides the explanations, figures, values, and parameters of the P-T and OPPS limits for the applicable effective fluence period. Furthermore, this specification requires the figures, values, and parameters to be (a) established using the FCS plant-specific methodology approved by the NRC for this purpose in this SE, and (b) consistent with all applicable acceptance limits and the limits of the FCS safety analyses. Finally, this specification requires the licensee to document in the PTLR all changes in the values of these limits each effective fluence period and submit to the NRC the revised PTLR upon its issuance.

In addition, TS 5.9.6(b) lists the documents that are necessary to define the FCS's PTLR methodology. These documents are listed in Section 3.1 of this SE as references (a) through (k). The NRC staff agrees that proposed TS 5.9.6(b) accurately defines the FCS's PTLR methodology as required by TSTF-419, and is therefore acceptable.

The NRC staff has concluded that this TS is consistent with the approved TSTF-419, to allow referencing the PTLR and topical reports by number and title to allow licensees to use current topical reports to support limits in the PTLR without having to submit an amendment to the facility operating license every time the topical report is revised.

3.4 Appendix G Exemption (Item 4)

The October 8, 2002, application included an exemption request to utilize the methodology contained in Topical Report CE NPSD-683-A, Revision 6 as the basis for the development of the FCS PTLR. The necessity for the licensee's exemption request stems from the fact that when the NRC staff reviewed and approved the methodology of CE NPSD-683-A, Revision 6, the NRC staff could not conclude that the methodology would always provide results which were at least as conservative as those which would result from the specific requirements found in Appendix G of 10 CFR Part 50. Hence, based on the requirements in 10 CFR 50.60 and the conditions in the NRC staff's SE, dated March 16, 2001, which approved CE NPSD-683-A, Revision 6, an exemption is required for the licensee to adopt CE NPSD-683-A, Revision 6 as part of the overall methodology for the FCS PTLR. The NRC staff approved this exemption request by letter dated July 30, 2003. TS 5.9.6 has been revised to include this as one of the references documenting the complete PTLR methodology.

The staff has reviewed the four items as requested in the licensee's October 8, 2002, submittal. Based on this review, the NRC staff concludes that the licensee has proposed, consistent with GL 96-03, an acceptable means of maintaining the detailed values of the current P-T limit curves and OPPS limits, and making changes to these limits, as needed, in the future. Further, the requirements of Appendix G to 10 CFR Part 50 and the existing TS will continue to limit plant operation in accordance with the PTLR values of the P-T limit curves and OPPS limits on the TS required parameters. These values will be established using an NRC-approved methodology. Therefore, relocating the values of the P-T limits and OPPS limits to the PTLR will not impact plant safety and is, therefore, acceptable.

The information discussed above relating to the P-T limits and OPPS limits is not itself required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. The previously listed associated LCOs, which do satisfy one or more of the four criteria in 10 CFR 50.36(c)(2)(ii), will remain in the TS. These LCOs, consistent with Appendix G P-T requirements, will continue to require operating the plant in accordance with the PTLR P-T limits and OPPS limits. These limits will be maintained and revised using the NRC-approved methodology, as required by TS 5.9.6, or NRC prior approval of a license amendment to revise P-T limits and methodology must be obtained. Accordingly, the NRC staff concludes that the detailed values of the current P-T limit curves and OPPS limits may be removed from the TS and maintained in the PTLR. Therefore, the proposed PTLR and associated TS changes are acceptable. Further, when changes are made to the limits contained in the PTLR, in accordance with the NRC-approved methodology, the licensee will update the PTLR and submit the PTLR to the NRC upon its issuance as required by proposed TS 5.9.6(c).

3.5 Minimum Boltup Temperature

OPPD proposed to modify the minimum RPV boltup temperature from 84°F to 64°F and to extend the P-T limit curves on Figure 5-1 down to 64°F. The effective requirement for minimum boltup temperature in Appendix G to 10 CFR Part 50 is that it must be equivalent to the highest nil-ductility reference temperature (RT_{NDT}) of any material in the RPV flange region that is highly stressed by the bolt preload (i.e., stud tensioning). For the FCS RPV, the highest RT_{NDT} value for a highly stressed RPV flange material is 60°F. The licensee then conservatively added 4°F

of margin to this value to account for uncertainty in RPV flange material temperature measurement to arrive at the proposed boltup temperature of 64°F.

The NRC staff reviewed the information provided by the licensee in its PTLR submittal and information previously submitted and entered into the NRC staff's Reactor Vessel Integrity Database. The NRC staff confirmed the limiting RT_{NDT} value cited for the FCS RPV flange materials. Based on this information, the NRC staff concludes that the use of a 64°F boltup temperature for the FCS RPV is consistent with the requirements given in Appendix G to 10 CFR Part 50 and is therefore acceptable.

3.6 Maximum SIT Cover Gas Pressure

As result of conversations with the NRC staff and OPPD's fuel vendor (Framatome ANP), OPPD has proposed to add a maximum SIT pressure limit in TS. Framatome ANP informed OPPD that a maximum value for the SIT nitrogen gas pressure is an input to the large break loss-of-coolant accident (LOCA) analysis for FCS. The large break LOCA analysis can use either an average from the measured SIT gas pressure data or the average between the minimum and maximum TS SIT gas pressure values. OPPD used the average value from the measured SIT gas pressure data and therefore, by letter dated August 23, 2002, OPPD informed the staff that the maximum SIT gas pressure meets criterion 2 of 10 CFR 50.36(c)(2)(ii) for control of a key input parameter to the design basis analyses.

The design pressure of the SITs is 275 psig at 200°F. SIT pressure is administratively controlled by plant operating procedures and is verified by plant procedures once per shift. The high pressure alarm is set at 265 psig. The SIT tanks are protected by relief valves nominally set at 275 psig. The relief valves are tested periodically per TS 3.3(1)a and the acceptance range for the as-found condition is+3%/-3%.

OPPD has proposed to add a new and more restrictive SIT pressure limit to TS2.3.(1)c. The new TS limit is consistent with the current plant design and the design basis accident analyses. The staff agrees that this limit meets the criterion in 10 CR 50.36 and should be added to the TS.

3.7 Changes to the Bases Section

The Bases sections of TS 2.1.1, TS 2.1.2, and TS 2.10.1 have been revised to reflect the proposed TS changes. TS 5.20, "Technical Specification (TS) Bases Control Program," assures the continuing accuracy and adequacy of the Bases. Therefore, the Bases changes have had the appropriate administrative controls and reviews performed to assure the accuracy and adequacy of the change. The NRC staff has reviewed theses Bases changes and has no objections to them.

4.0 <u>STATE CONSULTATION</u>

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 37579). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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