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

August 10, 2015

Mr. Louis P. Cortopassi
Site Vice President and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station
9610 Power Lane, Mail Stop FC-2-4
Blair, NE 68008

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: ADOPT AMERICAN SOCIETY OF MECHANICAL ENGINEERS *BOILER AND PRESSURE VESSEL CODE*, SECTION III, AS AN ALTERNATIVE TO THE CURRENT CODE OF RECORD (TAC NO. MF4160)

Dear Mr. Cortopassi:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 283 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the licensing basis in response to your application dated May 16, 2014, as supplemented by letters dated January 9, March 27, and July 2, 2015.

The amendment revised the Updated Safety Analysis Report to allow pipe stress analysis of non-reactor coolant system safety-related piping to be performed in accordance with the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, Section III, 1980 Edition (no Addenda) as an alternative to the current Code of Record (i.e., United States of America Standards B31.7, 1968 (DRAFT) Edition).

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,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 283 to DPR-40
2. Safety Evaluation

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

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 283
Renewed 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 May 16, 2014, as supplemented by letters dated January 9, March 27, and July 2, 2015, 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.

Enclosure 1

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

    B. Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 283, are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance. In addition, the licensee shall include the revised information in the next Final Safety Analysis Report update submitted to the NRC in accordance with 10 CFR 50.71(e), as described in the licensee's application dated May 16, 2014, and supplemented by letters dated January 9, March 27, and July 2, 2015, and evaluated in the staff's safety evaluation enclosed with this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-40

Date of Issuance: August 10, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 283

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following page of the Renewed Facility Operating License No. DPR-40 with the attached revised page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

License Page

REMOVE	INSERT
-3-	-3-

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is, subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not in excess of 1500 megawatts thermal (rate power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 283 are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan," submitted by letter dated May 19, 2006.

OPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The OPPD CSP was approved by License Amendment No. 266.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 283 TO RENEWED 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 May 16, 2014, as supplemented by letters dated January 9, March 27, and July 2, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14143A370, ML15009A319, ML15086A545, and ML15188A155, respectively), Omaha Public Power District (OPPD) requested changes to the licensing basis for the Fort Calhoun Station, Unit No. 1 (FCS).

The proposed amendment would revise the licensing basis as described in the FCS Updated Safety Analysis Report (USAR) to allow pipe stress analysis of non-reactor coolant system (RCS) safety-related piping to be performed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV) Code, Section III, 1980 Edition (no Addenda) as an alternative to the current Code of Record (i.e., United States of America Standards (USAS) B31.7, 1968 (DRAFT) Edition, hereafter referred to as USAS B31.7). The RCS piping was designed to USAS B31.1 and is not the subject of this application.

The supplemental letters dated January 9, March 27, and July 2, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 8, 2014 (79 FR 38593).

2.0 REGULATORY EVALUATION

The proposed change is limited to application of the ASME BPV Code to pipe stress analysis of non-RCS safety-related piping. Safety-related piping is generally designated Class 1, 2, or 3 piping. USAS (aka ANSI [American National Standards Institute] or ASME due to organizational changes over the years) B31.7, "Nuclear Power Piping," remained applicable until 1971, when it was withdrawn and superseded by ASME BPV Code, Section III. More accurate and improved

methods have been added to the ASME BPV Code over time, so that it is equal to or better than USAS B31.7, from which it evolved. The ASME BPV Code committee recognized the evolution, as demonstrated by ANSI B31 Code Case Interpretation No. 115, "Accept Rules of Section III ASME BPV Code," which states,

> It is the opinion of the committee for B31.7, that piping that has been designed and constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code including addenda and applicable cases may be accepted as complying with the requirements of B31.7, 1969 and applicable addenda for the respective class of construction. The Section III requirements represent the best opinions on these subjects subsequent to the last issue of B31.7.

In its letter dated May 16, 2014, the licensee stated, in part, that

> Although ANSI B31 Code Case Interpretation No. 115 references USAS B31.7, 1969, that revision of the Code is virtually the same as the 1968 (DRAFT) Edition, which is the current Code of Record for FCS.
>
> Therefore, it is appropriate to allow the ASME BPV Code as an alternative to the original Code of Record for conducting pipe stress analysis of non-RCS, safety-related piping

In addition, the ASME BPV Code, Section III, 1980 Edition (no Addenda) is an NRC-approved standard for incorporation by reference, as listed in Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(a)(1)(i). Therefore, the NRC staff concludes that the proposed use of ASME BPV Code, Section III, 1980 Edition (no Addenda) as an alternative to the original Code of Record for conducting piping stress analysis of non-RCS safety-related piping is acceptable.

3.0 TECHNICAL EVALUATION

The proposed change is limited to application of ASME BPV Code to pipe stress analysis of non-RCS safety-related piping. The proposed change does not apply to the reactor pressure vessel. As mentioned above, the industry has added more accurate and improved methods to the ASME BPV Code over time. As referenced by the licensee in its application, NUREG/CR-3243, "Comparisons of ASME BPV Code Fatigue Evaluation Methods for Nuclear Class 1 Piping with Class 2 or 3 Piping," May 1983 (http://web.ornl.gov/info/reports/1983/3445605994528.pdf; also ORNL/Sub/82-22252/1) demonstrates that use of the ASME BPV Code is considered equivalent to the USAS B31.7 Code for Class I piping stress and fatigue analysis.

For a discussion of the differences between USAS B31.7 and the ASME BPV Code for Class 2 and 3 piping, the licensee referenced in its application the Electric Power Research Institute

(EPRI) Report 1012078, "Background of SIFs and Stress Indices for Moment Loadings of Piping Components (MRP-184)," dated June 29, 2005, (http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001012078). EPRI 1012078 states, in part, that

> Design and engineering for fatigue are major concerns in piping systems. Stress indices and stress intensification factors (SIFs) are used in the design of piping systems that must meet the requirements of American Society of Mechanical Engineers (ASME) Section III Code. The majority of SIFs and indices were developed years ago when only relatively unsophisticated analysis methods were available. Many of these parameters are very conservative...

In its summary of the EPRI report, the licensee stated in its letter dated May 16, 2014, that

> For Class 2 and 3 piping, the evaluation of fatigue was performed by the use of stress intensification factors (SIF), which are fatigue correlation factors that compare the fatigue life of piping components (for example, tees and elbows) to that of girth butt welds in straight pipe subjected to bending moments. The ASME BPV Code defines the SIF as "the ratio of the bending moment producing fatigue in a given number of cycles in a straight pipe with a girth butt weld to that producing failure in the same number of cycles in the fitting or joint under consideration." The use of SIF constitutes a simplified method to address fatigue caused by thermal cycles that results in an alternating stress. For Class 2 and 3 piping, there are also differences between the USAS B31.7 Code and the ASME BPV Code in how the combined moments are utilized to calculate stresses. In the USAS B31.7 Code, the SIF "i" is applied only to the two bending moments and is not applied to the torsion moment. The resultant bending intensified moments are then divided by the pipe section modulus to determine the longitudinal stress in the pipe. In the ASME BPV Code, the SIF is applied to the unintensified stress calculated from the resultant moment. At the component level, the SIF is applied to all three moment components (i.e., bending and torsion). Additionally, for the ASME BPV Code, a 0.75 factor is applied to the SIF for sustained (i.e., deadweight) and occasional (e.g., seismic and other dynamic accident type) loads. The product of the SIF multiplied by 0.75 cannot be less than 1. The 0.75 factor is not applied to thermal loads in equations 10 or 11.
>
> The ASME BPV Code's removal of some conservatism when applied to sustained loads and occasional loads is appropriate since none of these loads cause a large number of fatigue cycles (i.e. alternating stresses). This approach is consistent with the Code defined application of SIF. In addition, the ASME BPV Code stresses are conservative relative to USAS B31.7 due to the application of the stress intensification factor to all three moment components.

The NRC staff reviewed the reference documents and identified that the USAS B31.7 contains SIF indices for only curved pipe or elbows with pressure or moment loading and for certain types of branch connections with internal pressure loading. On the other hand, the ASME BPV Code, Section III, 1980 Edition (no Addenda) provided an improved methodology with detailed

thermal ratchet requirements for design purposes. The stress calculation in the ASME BPV Code, Section III, 1980 Edition (no Addenda), with the SIF applied to all three orthogonal moments, is more conservative than the USAS B31.7 calculation using the SIF applied to only two of the bending moments. Therefore, the NRC staff concludes that the ASME BPV Code, Section III, 1980 Edition (no Addenda) provided a more accurate and improved methodology than USAS B31.7 to address piping stress analysis.

The licensee proposed to revise Table F-1 of USAR Appendix F by adding NOTES (e), (f), and (g). NOTE (e) explains the loading combinations for the piping. The proposed earthquake and fluid transient loading are combined using square-root-of-the-sum-of-the-squares (SRSS) method. The current version of the USAR identifies that earthquake and fluid transient loading are added for Loading Combination 2 of Table F-1. In its letter dated January 9, 2015, in response to an NRC staff request for additional information (RAI) dated September 24, 2014 (ADAMS Accession No. ML14259A365), the licensee stated that the SRSS method for loss-of-coolant accident and seismic loading combination was endorsed by the NRC staff in NUREG-0484, "Methodology for Combining Dynamic Loads and Load Combinations," Revision 1, May 1980 (ADAMS Accession No. ML13260A310). For consistency, the licensee proposed adding the SRSS method of combining the subject loads as an option for the load combinations in NOTE (e) of USAR Table F-1 as well as the existing load combinations in Table F-1 proper. The licensee proposed NOTE (g) to clarify the use of the SRSS method in combining loads in Table F-1 and NOTE (e). The staff confirmed that the SRSS loading combination method was approved by the NRC in NUREG-0484. Therefore, the proposed SRSS combination method for the dynamic loadings is acceptable.

The licensee also proposed to specify that the Service Level C loading combination of NOTE (e) refers to NOTE (d), which states that this load case and limit apply only to the pressurizer relief valve piping and supports. However, the NRC staff noted that the fluid transient loading for Class 2 and 3 piping is not limited to only the pressurizer relief valve. For example, main steam lines have relief valves, and a fast valve open and closure event, such as reactor vessel head vent valve opening, might cause hydrodynamic loading. In its letter dated July 2, 2015, the licensee provided clarification and removed the NOTE (d) from the loading combination. Since the proposed fluid transient loading is not limited to only the pressurizer spray line, and all piping systems will be properly evaluated to address fluid transient loading, the NRC staff concludes that the proposed revision is acceptable.

The NRC staff concluded that the licensee's inclusion of the maximum hypothetical earthquake loading in the emergency condition for piping stress analysis is conservative, when applying Service Level C stress limit for this loading combination. Usually, the maximum earthquake condition is considered in the Service Level D (Faulted) condition and does not have to be considered in Service Level C (Emergency) condition. The licensee stated in its letter dated January 9, 2015, that the wording for the load cases was carried over for consistency to Service Level C of proposed NOTE (e) in USAR Table F-1. Since the proposed change is a more conservative load combination, the NRC staff concludes that it is acceptable.

In its letter dated March 27, 2015, in response to an NRC staff RAI dated February 25, 2015 (ADAMS Accession No. ML15043A061), the licensee clarified that the intention of its application is to apply the ASME BPV Code, Section III, 1980 Edition (no Addenda) to all piping originally

designed to USAS B31.7, which specifically excludes RCS piping because it was designed to USAS B31.1. NOTE (e) clarifies that the RCS piping is excluded. The exclusion of the RCS piping using USAS B31.1 is consistent with current licensing basis, and so the NRC staff concludes that it is acceptable. The use of the ASME BPV Code, Section III, 1980 Edition (no Addenda) is an acceptable alternative for use with all non-RCS piping in accordance with 10 CFR 50.55a(a)(1)(i).

The proposed NOTE (f) states that support analysis will continue to be performed in accordance with the existing licensing basis (i.e., Seventh Edition of American Institute of Steel Construction (AISC)). In its letter dated January 9, 2015, the licensee documented that the applicable piping codes for original design and licensing of FCS were the USAS B31.7, 1968 (DRAFT) Edition and the USAS B31.1, 1967 Edition, as noted in Appendix F of both the final safety analysis report (FSAR) and the USAR. These codes state that supplementary steel shall be designed in accordance with the standards prescribed by the AISC or the equivalent, and the Seventh Edition of AISC was used since it was included in the 1969 AISC specification. In its letter dated January 9, 2015, the licensee also stated that in its letter dated April 21, 1980,[1] OPPD informed the NRC that the Seventh Edition of the AISC Manual of Steel Construction would be used in evaluating supplementary steel of pipe supports. The NRC staff confirmed the licensee's design basis documents; therefore, the proposed NOTE (f) is acceptable.

The licensee also proposed changes in the piping and vessel primary stress limits for the Loading Combinations 2 and 3 of Table F-1. In its letter dated January 9, 2015, in response to an NRC staff RAI dated September 24, 2014, the licensee clarified that:

> The expressions in USAR Appendix F, Table F-1 for the piping and vessel primary stress limits were approved as part of the original operating license for Fort Calhoun Station and remain unchanged from their original form in the FSAR...
>
> In addition, the Load Combination 2 primary bending stress limit equation for vessels was specifically reviewed because it is referenced in the question. The equation shown in USAR Table F-1 is the same as the one provided in the FSAR (Attachment 4 [of the licensee's letter dated January 9, 2015]) and thus is correct. However, as the FSAR was developed before word processing software was in widespread use, the equation as typed in the FSAR and carried through to the USAR could be misapplied. The parentheses are intended to enclose both the P_M and S_D terms as shown in Attachments 2 and 3 [of the licensee's letter dated January 9, 2015]. Similarly, as shown in Attachments 2 and 3, the parentheses in the Load Combination 2 primary bending stress limit equation for piping should enclose both the $\pi/2\ x\ P_M/S_D$ terms to be consistent with the FSAR.
>
> These same formatting discrepancies are also applicable for the Load Combination 3 vessel and piping primary bending stress limits in Table F-1 as

[1] Jones, W.C., Omaha Public Power District, letter to U.S. Nuclear Regulatory Commission, "Completion of the Requirements of Bulletins 79-02 and 79-14," dated April 21, 1980 (LIC-80-0042) (ADAMS Legacy Library No. 8006040024).

well as the same equations in Table 4.2-3. These formatting discrepancies are corrected in Attachments 2 and 3 to reflect the original equation found in the FSAR.

Since the proposed revision formats the primary stress limit equations to reflect the original equations found in the FSAR, the NRC staff has no objections to the changes. Under 10 CFR 50.59, NRC staff approval is not required prior to the formatting change.

The NRC staff reviewed the proposed changes of NOTES (e), (f), and (g) of Table F-1 and determined the changes are acceptable as documented above. The staff reviewed the loading combinations of the piping systems and concluded that the licensee provided reasonable assurance that the piping systems will remain structurally adequate to perform their intended design function. On the basis of the above evaluation, the staff concludes that adopting the ASME BPV Code, Section III, 1980 Edition (no Addenda) as an alternative to the current Code of Record for non-RCS safety-related piping analysis is acceptable.

4.0 STATE CONSULTATION

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 the 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 published in the *Federal Register* on July 8, 2014 (79 FR 38593). 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) there is reasonable assurance that 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.

Principal Contributor: K. Hsu, NRR/DE/EMCB

Date: August 10, 2015

L. Cortopassi
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/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 283 to DPR-40
2. Safety Evaluation

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ADAMS Accession No. ML15209A802 *memo dated

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