June 7, 2001

Mr. S. K. Gambhir
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 399
Hwy. 75 - North of Fort Calhoun
Fort Calhoun, NE 68023-0399

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT - DELETION OF SECTION 3.D, "LICENSE TERM" (TAC NO. MA9690)

Dear Mr. Gambhir:

The Commission has issued the enclosed Amendment No. 199 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1 (FCS). The amendment consists of a change to the operating license in response to your application dated August 3, 2000, as supplemented by letters dated November 17, 2000, and February 14, 2001.

The amendment deletes Section 3.D, "License Term," from Facility Operating License No. DPR-40. The licensee's analysis resulted in a new limiting beltline material which is the weld fabricated from tandem weld wire heat 12008/13253. The increase in the long-term load factor from 0.77 to 0.85 did not cause the critical weld material to exceed the reference temperature ($RT_{PTS}$) screening criteria of 10 CFR 50.61 (the pressurized thermal shock (PTS) rule). Therefore, the staff has concluded that the FCS reactor vessel is projected to be below the PTS screening criteria of 10 CFR 50.61 at the expiration of its current license (August 9, 2013) as well as the end of the proposed license renewal period (August 9, 2033). The NRC notes that paragraph (B)(v)(2) and footnote 5 of 10 CFR 50.61 requires that the licensees must assess the impact of changes to the FCS PTS evaluation that result from new surveillance data. Specifically, new data from the Mahima Unit 1, Diablo Canyon Unit 1 and Palisades plants must be assessed as it becomes available, since the data from these plants were used in the FCS PTS analysis. Based on the new analysis that demonstrates that the limiting weld is within the current PTS screening criteria of 10 CFR 50.61, the requirements of 10 CFR 50.61 that assure the analysis remains valid, and given that the requirements in Section 3.D are redundant to 10 CFR 50.61 requirements, as 10 CFR 50.61 requires updating this assessment whenever there is a significant change in projected values of $RT_{PTS}$, the staff has concluded that the request to delete license condition 3.D is acceptable. The licensee's analysis assumes that future core loadings will be such as to limit the core neutron leakage to values similar to those for Cycles 15 and 16 and to limit the end of license fluence accumulation to $1.728 \times 10^{19} n/cm^2$ to the limiting welds. Therefore, the design of future cores must satisfy the above limitation and in addition caution must be exercised to preclude misloading any of the peripheral assemblies which would invalidate the loading requirements.

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 & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 199 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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
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Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 199 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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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. 199
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 August 3, 2000, as supplemented by letters dated November 17, 2000, and February 14, 2001, 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 as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

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

Attachment: Changes to the Operating License

Date of Issuance: June 7, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 199

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages of Facility Operating License No. DPR-40 with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

| REMOVE | INSERT |
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| 4a | 5 |

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 to exceed 1500 megawatts thermal (rated power).

B. Technical Specifications

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

C. Security and Safeguards Contingency Plans

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard 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 Physical Security Plan," with revisions submitted through September 30, 1988: "Fort Calhoun Station Guard Training and Qualification Plan," with revisions submitted through August 17, 1979; and Fort Calhoun Station Safeguards Contingency Plan," with revisions submitted through March 20, 1979. If certain security modifications are delayed beyond expectations of the schedule, approved compensatory measures must be implemented during the transition period.

D. Fire Protection Program

Omaha Public Power District shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report for the facility and as approved in the SERs dated February 14, and August 23, 1978, November 17, 1980, April 8, and August 12, 1982, July 3, and November 5, 1985, July 1, 1986, December 20, 1988, November 14, 1990, March 17, 1993, and January 14, 1994, subject to the following provision:

> Omaha Public Power District may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

E. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 181, are hereby incorporated into this license. Omaha Public Power District shall operate the facility in accordance with the Additional Conditions.

4. This amended license is effective as of the date of issuance and shall expire at midnight on August 9, 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosures:
1. Appendix A - Technical Specifications
2. Appendix B - Additional Conditions

Date of Issuance: August 9, 1973

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 199 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 August 3, 2000, as supplemented by letters dated November 17, 2000, and February 14, 2001, Omaha Public Power District (OPPD) requested a change to Facility Operating License No. DPR-40 for the Fort Calhoun Station (FCS), Unit No. 1. The requested change would delete Section 3.D, "License Term," from the FCS operating license.

The November 17, 2000, and February 14, 2001, letters provided clarifying information, 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 published in the *Federal Register* on January 10, 2001 (66 FR 2019).

## 2.0 BACKGROUND

By letter dated November 15, 1999, OPPD submitted final report CEN-636, Revision 0, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the FCS Reactor Vessel Beltline Materials - Basis for Prediction of $RT_{PTS}$ at Expiration of License." In a teleconference on November 22, 1999, the staff discussed with the licensee its concerns regarding the methodology in CEN-636, Revision 0. The primary concern was that the licensee's pressurized thermal shock (PTS) evaluation used surveillance data from other plants that do not have the limiting FCS weld wire heat combination (12008/27204). These concerns were subsequently documented to the licensee in a letter dated November 30, 1999. The limiting FCS weld wire heat is not a part of any United States surveillance program. However, a Japanese plant (Mihama Unit 1) has this limiting FCS weld wire heat combination in its surveillance program. The licensee informed the staff that they would attempt to obtain the Mihama Unit 1 data.

The staff met with the licensee at NRC Headquarters on January 6 and March 13, 2000. The staff requested that the licensee obtain the appropriate quality assurance information for the Mihama Unit 1 data and documented this request in a meeting summary dated February 10, 2000.

During the January and March 2000, meetings, the licensee indicated that FCS plans to submit a license renewal application. The licensee also noted that using the current rate of embrittlement, welds fabricated using tandem weld wire heat 27204/27204 would exceed the PTS screening criteria before the proposed license renewal period ends. The licensee proposed to use surveillance data from Diablo Canyon Unit 1 and from a supplemental capsule in Palisades to calculate reference temperature ($RT_{PTS}$) with a reduced margin, as permitted by 10 CFR 50.61 (the PTS rule) when surveillance data meet the credibility criteria in the rule. The evaluation of the rate of embrittlement for welds fabricated with tandem heat 27204/27204 in support of license renewal is discussed in Section 3.10 of this safety evaluation (SE). The licensee withdrew CEN-636, Revision 0 and submitted CEN-636, Revision 2 by letter dated August 3, 2000. The revised report includes evaluation of the Mihama Unit 1 data. In addition, information in the August 3, 2000, letter resolved fluence issues that the staff had previously identified. The licensee submitted supplementary information regarding other aspects of the PTS evaluation by letters dated November 17, 2000, and February 14, 2001. The November 17, 2000 letter, forwarded a proprietary report which included unirradiated and irradiated baseline Charpy impact data for the Mihama Unit 1 surveillance weld. The February 14, 2001, letter forwarded details on the irradiation temperature adjustment that the licensee used in its evaluation. The differences in the staff's and the licensee's evaluation methodology are described in this SE.

The PTS rule adopted on July 23, 1985, and revised on May 15, 1991, and December 19, 1995, established screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature, $RT_{PTS}$. The screening criteria are 270°F for plates and axial welds and 300°F for circumferential welds. The $RT_{PTS}$ is defined as:

$$RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + M$$

where: (a) $RT_{NDT(U)}$ is the initial reference temperature, (b) $\Delta RT_{PTS}$ is the mean value in the adjustment in reference temperature caused by irradiation, and (c) M is the margin to be added to cover uncertainties in the initial reference temperature, copper and nickel contents, fluence, and calculational procedures.

The initial reference temperature is the measured unirradiated value as defined in the American Society of Mechanical Engineers (ASME) Code, Section III, Paragraph NB-2331. If measured values are unavailable for the heat of material of interest, generic values may be used. The generic values are based on the data for materials of all heats that were made by the same vendor using similar processes. The generic values of initial reference temperature for welds are defined in the PTS rule.

The $\Delta RT_{PTS}$ depends upon the amount of neutron irradiation and the amounts of copper and nickel in the material and is calculated as the product of a fluence factor and a chemistry factor. The fluence factor is calculated from the best estimate neutron fluence at the clad-weld-metal interface on the inside surface of the vessel at the location where the material receives the highest fluence at the end of the period of evaluation. The chemistry factor may be determined using credible surveillance data or from the chemistry factor tables in the PTS rule. The chemistry factors in the tables are dependent upon the best-estimate values of the amount of copper and nickel in the material. The term "best-estimate" is not well defined statistically, but has normally been interpreted as the mean of the measured values.

The PTS rule contains criteria for determining whether surveillance data are credible. The rule also contains the procedure for calculating the vessel weld chemistry factor from the adjusted or measured values of $\Delta RT_{PTS}$. Specifically, the rule states that if there is clear evidence that the copper and nickel content of the surveillance weld differs from that of the vessel weld, the measured values of $\Delta RT_{PTS}$ should be adjusted by multiplying them by the ratio of the chemistry factor of the vessel weld to that of the surveillance weld. The chemistry factor is calculated by multiplying each adjusted or measured value of $\Delta RT_{PTS}$ by its corresponding fluence factor, summing the products, and dividing by the sum of the squares of the fluence factors. The resulting chemistry factor will give the relationship of $\Delta RT_{PTS}$ to fluence that fits the plant's surveillance data in such a way as to minimize the sum of the squares of the errors.

The margin term is intended to account for variability in initial reference temperature and the adjustment in reference temperature caused by irradiation. The value of the margin term is dependent upon whether the initial reference temperature was a measured or generic value and whether the adjustment in reference temperature was determined from credible surveillance data or from the chemistry factor tables in the PTS rule.

## 3.0 EVALUATION

The FCS reactor vessel beltline includes the intermediate shell plates D-4802-1, D-4802-2, and D-4802-3, heats C2585-3, A1768-1, and A1768-2 respectively; lower shell plates D-4812-1, D-4812-2, and D-4812-3, heats C3213-2, C3143-2 and C3143-3 respectively; intermediate to lower shell circumferential weld 9-410, heat 20291, intermediate shell axial welds 2-410 A/C, tandem heat 51989; lower shell axial welds 3-410 A/C, tandem heats 13253, 12008/13253, 27204, and 12008/27204. The material with the greatest amount of embrittlement (limiting material) for the FCS reactor vessel was initially the weld fabricated from tandem heat number 12008/27204. However, with the evaluation of the Mihama Unit 1 data, the new limiting material is the weld fabricated from tandem heat number 12008/13253. The initial limiting weld (weld wire heat 12008/27204) was fabricated by Combustion Engineering (CE) using the automatic submerged arc weld process, copper-coated electrodes, and Linde 1092 or Linde 124 flux.

Surveillance data for the weld fabricated from tandem heat 12008/27204 are not available in the FCS surveillance program or any program in the United States. However, the data is available in the Mihama Unit 1 surveillance program. Mihama Unit 1 is a 320 megawatt electric (MWe) pressurized water reactor operated by Kansai Electric Power Company in Japan. The licensee obtained data from the Mihama Unit 1 surveillance program through a proprietary agreement between Kansai Electric Power Company and OPPD. The Mihama Unit 1 vessel and the surveillance weld were fabricated by CE and designed by Westinghouse (W). The surveillance weld was fabricated using the automatic submerged arc weld process which is the same process that was used to fabricate the FCS lower shell axial welds 3-410 A/C. There are three surveillance capsules that have been tested from Mihama Unit 1.

The Mihama Unit 1 and FCS cold leg inlet temperatures are 552°F and 543°F, respectively. Since the Mihama Unit 1 surveillance capsules were irradiated at a higher cold leg temperature as compared to the FCS cold leg temperature, the Mihama Unit 1 surveillance data requires a temperature correction for use.

3.1 Quality Assurance of Mihama Unit 1 Surveillance Data

Charpy testing of the Mihama Unit 1 surveillance materials was conducted using American Society of Testing and Materials (ASTM) E-23, "Standard Test Methods for Notched Bar Impact Testing of Metallic Materials." Section 10 (Verification of Charpy Machines) describes how the verification is performed.

Kansai Electric notified OPPD that W provided the chemical analysis information. W staff involved in the evaluations confirmed that they used the same chemical analysis testing techniques and standards that they used for the U.S. surveillance programs. Since Kansai Electric used the same standard and chemical analysis as in the U.S., results from their surveillance program are compatible with those from domestic U.S. programs.

3.2 Irradiation Environment

The staff evaluated the applicability of the Mihama Unit 1 surveillance data to the FCS vessel in terms of similarity of the irradiation environments. Kansai Electric reported the neutron flux corresponding to each irradiated and tested capsule from Mihama Unit 1 together with their source reference and a description of the methodology used to calculate the neutron flux. The reported flux is consistent with similarly configured reactor vessels designed by W.

The design and construction of the Mihama Unit 1 surveillance capsules are the same as that for other surveillance capsules that W fabricated. Therefore, it is reasonable to conclude that the gamma heating in the Mihama Unit 1 surveillance capsules is similar to other domestic W capsules.

With regards to neutron spectra, in 1996 the Combustion Engineering Owners Group (CEOG) sponsored a program whose purpose was to determine whether or not CE fabricated reactor vessel materials were equally predictable using Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," for plants designed by both W and CE. The CEOG concluded that there was no definitive difference between the spectra such that one needs only to consider differences in the irradiation temperature and the neutron flux. Based on a rigorous statistical analysis, the staff agreed with the licensee's determination, and concluded that there is no significant difference or bias between the CE fabricated, CE and W designed surveillance data with regards to neutron spectra. Therefore, the neutron spectra in the Mihama Unit 1 surveillance capsules are not expected to adversely affect the application of those surveillance data to the FCS vessel.

3.3 License Condition 3.D and Fluence Evaluation

The licensee's letter dated August 3, 2000, requested a license amendment to delete license condition 3.D which requires "...monitoring of the long term load factor to assure that it does not exceed the assumed value of 0.77....and that the $RT_{PTS}$ will not exceed the screening criterion being in place." This license condition is redundant to 10 CFR 50.61 requirements. The evaluation involves the fast neutron fluence to the pressure vessel projected to the end of the current license which expires August 9, 2013. The proposed fluence value for 30 effective full power years (EFPYs) of operation was estimated by W and described in WCAP-15443, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel." The 30 EFPYs are estimated at the end of the current license. The submittal proposes to raise the average load factor to 0.85 (the normal value is 0.80), however, the historical operating record for FCS was about 0.77, thus, the 0.85 load factor (for the remaining plant life) will fall short of 32 EFPYs.

The August 3, 2000, submittal is a modified version of the original submittal dated January 30, 1998. The original submittal was unacceptable because the proposed best estimate value for the vessel fast neutron fluence was adjusted to selected dosimeter measured values. Additional data were submitted by letter dated November 15, 1999. Staff review of that information identified a number of concerns and the applicant withdrew the original application on January 24, 2000.

The methodology in the August 3, 2000, submittal for the estimation of the fluence value complies with the recommendations of Draft RG DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The two-dimensional DORT neutron transport code was used which has been benchmarked to the pool critical assembly (PCA) measurements. The code was also verified to plant-specific measurements. The BUGLE-93 library was used (with 47 energy groups) which is based on the ENDF/B-VI data file. The quadrature approximation was $S_8$ and the scattering expansion approximation was $P_3$. Forward calculations in the (r,θ) and the (r,z) modes were carried out. The source distribution was representative of the average over the first 14 cycles at FCS. The methodology outlined above complies with the provisions of DG-1053, and, therefore, is acceptable.

The resulting numerical value of $1.728 \times 10^{19} n/cm^2$ is applicable to the 60°-azimuthal axial critical welds 3-410 for 30 EFPYs (i.e., the end of the current license in August 2013). This value has not been adjusted and it is lower than the value proposed in the original submittal. The licensee states that this lower value is due to very low leakage loading to be practiced on all future fuel cycles to the end of license. The licensee states that the 3-410 axial welds are the critical elements, therefore, all other vessel plates and welds remain below the 3-410 weld PTS.

The staff reviewed the submitted information regarding the proposed fluence value at FCS and finds that it is acceptable because the methodology, the approximations and the cross sections used in the evaluation satisfy the DG-1053 recommendations. The numerical value was estimated for a load factor of 0.85 which is greater than the load factor of 0.77 used in previous estimates. This load factor is acceptable because: (1) the projected total EFPYs will be lower than 32, and (2) the projected critical element satisfies the screening criteria of 10 CFR 50.61. Therefore, the request to eliminate License Condition 3.D is acceptable.

This approval assumes that future core loadings will be such as to limit the core neutron leakage to values similar to those for Cycles 15 and 16 which will satisfy the requirement of end of license fluence accumulation of $1.728 \times 10^{19} n/cm^2$ to the limiting welds. Therefore, the licensee will design future cores satisfying the above limitations and in addition must exercise the necessary caution to preclude misloading any of the peripheral assemblies which would invalidate the loading requirements.

## 3.4 Initial Reference Temperature for Weld Wire Heat 12008/27204

### 3.4.1 Unirradiated Charpy Data

No unirradiated drop weight test results were reported for the baseline testing of the Mihama Unit 1 surveillance weld material. The baseline report was issued in January 1970, before the requirements for determining $RT_{NDT(U)}$ properties were introduced in the Summer 1972 addenda to Section III of the ASME Code. Paragraph NB-2331 of Section III of the ASME Code requires that drop weight and Charpy data be utilized in determining a $RT_{NDT(U)}$ value. The licensee reported -58°F as the plant specific $RT_{NDT(U)}$ value for weld wire heat 12008/27204; however, this value was obtained from Charpy data only. Since there are no drop weight test results, the staff concluded that the generic mean value for Linde 1092 flux welds is the correct value to use for weld wire heat 12008/27204. The generic $RT_{NDT(U)}$ value is -56°F with a standard deviation of 17°F.

### 3.4.2 Irradiated Charpy Data

The licensee applied hyperbolic tangent curve fits to the data sets from each of the three Mihama Unit 1 capsules. Comparison of the reported shift values to those determined from the hyperbolic tangent curve fits of the data show good agreement. The maximum difference between the reported values and the estimated shifts from curve fits to the data is 2.4°F. This difference is small and may be attributed to differences in curve fitting procedures, estimate of initial properties, and rounding of values.

The staff concluded that the Charpy shifts determined from comparing the actual unirradiated and irradiated Charpy data are consistent with shift values reported for each of the three Mihama Unit 1 surveillance capsules. Therefore, the Mihama Unit 1 surveillance program is considered to be applicable to the FCS reactor vessel weld fabricated using tandem weld heat 12008/27204.

## 3.5 Best-Estimate Chemical Composition of the FCS Vessel Weld

The licensee's best-estimate values of copper and nickel in the FCS vessel weld fabricated from heat 12008/27204 are 0.219 percent and 0.996 percent, respectively. Linear interpolation of the chemistry factors in Table 1 of the PTS rule indicates that the chemistry factor is 231.1°F for welds with these amounts of copper and nickel. The best estimate values of copper and nickel are mean values of weld deposit data from CE weld deposit analyses.

## 3.6 Best-Estimate Chemical Composition of the Mihama Unit 1 Surveillance Weld

Kansai Electric Company reported copper and nickel values of 0.19 percent and 1.08 percent, respectively, for the Mihama Unit 1 surveillance weld. Linear interpolation of the chemistry factors in Table 1 of the PTS rule indicates that the chemistry factor is 227.2°F for welds with these amounts of copper and nickel. The Kansai values are consistent with the best estimate for tandem weld wire heat 12008/27204.

## 3.7 Evaluation of Surveillance Data

### 3.7.1 Irradiation Temperature Adjustment and Credibility Assessment

The staff's method for adjusting weld surveillance data from one reactor vessel for application to another vessel is to: (1) adjust the data for chemistry differences using the ratio procedure in RG 1.99, Revision 2; (2) evaluate the data for credibility using the credibility criteria in RG 1.99, Revision 2 and the PTS rule (10 CFR 50.61); (3) calculate a chemistry factor from credible surveillance data; and (4) apply a one degree increase/decrease in shift for each one degree difference in irradiation temperature (i.e., the "one degree per degree" method) after the chemistry factor is calculated. RG 1.99, Revision 2 and the PTS rule do not contain a specific methodology for making irradiation temperature adjustments. The "one degree per degree" approach is a conservative method that was derived from Linde 80 weld data and was used in the Yankee Rowe reactor pressure vessel evaluation.

The approach that the licensee used was to adjust the shift measurements for chemistry differences using the ratio procedure, then adjust the irradiation temperature using a new approach. The licensee used the recommended correlation (4-1a) from NUREG/CR-6551, "Improved Embrittlement Correlations for Reactor Pressure Vessel Steels," to compute the predicted shift at both temperatures of interest (the FCS and the Mihama Unit 1 irradiation temperatures). The temperature effect is the difference in the two shifts that is added to or subtracted from the measured shift, whichever is appropriate. NUREG/CR-6551 was prepared for the NRC by E.D. Eason and J.E. Wright from Modeling and Computing Services and G.R. Odette from the University of California. The NUREG/CR-6551 correlation (4-1a) for calculating shift differs from the correlations in RG 1.99, Revision 2 and the PTS rule in format and input variables. Specifically, irradiation temperature, phosphorus content, and irradiation time are input variables in the NUREG correlation. The licensee provided additional information by letter dated February 14, 2001, which included a spreadsheet of all input variables and calculations for the irradiation temperature adjustment.

The licensee made the chemistry and irradiation temperature adjustments prior to the surveillance data credibility evaluation, which differs from the staff's method of adjusting the irradiation temperature until after the credibility evaluation has been completed. The licensee stated that doing the credibility evaluation on the data, adjusted for both chemistry and irradiation temperature, accounts for the time dependence of the presumed temperature effect and properly emphasizes high fluence data in the sum-of-the-squares analysis.

As part of the technical justification for using the NUREG/CR-6551 correlation for the irradiation temperature adjustment, the licensee stated that the correlation provides more rigorous treatment of the data than that afforded by the "one degree per degree" method. The licensee also stated that the correlation offers the benefit of 609 data points for defining the apparent effect of irradiation temperature differences. The database was developed by extracting raw Charpy data from the power reactor embrittlement database (PR-EDB), which is a consolidation of surveillance data that was compiled at Oak Ridge National Laboratory. Some of the data extracted from the PR-EDB required assessment and further processing. The analysis database was reviewed by the American Society for Testing and Materials (ASTM) E10.02.02 task group on embrittlement correlations which resulted in some revisions and additions to the database. The coefficients in the NUREG/CR-6551 correlation were developed from the database and refined by statistical analysis.

A more recent version of the correlation (August 2000) considers about 150 more surveillance datum than the correlation in NUREG/CR-6551 (November 1998). The staff determined the temperature effect using both the earlier and the most recent correlations. The temperature adjustments for the surveillance capsules that were tested from Mihama Unit 1 and the FCS vessel differed by approximately 1 to 2°F (i.e., for one Mihama capsule, the 1998 correlation resulted in a temperature adjustment of 4.3°F, and the 2000 correlation resulted in a 5.4°F temperature adjustment). The staff concluded that the 1998 correlation is acceptable for the purpose of determining irradiation temperature adjustments since the resulting values varied by only a few degrees.

The licensee used what was characterized as "alternate data input" to examine the sensitivity of the temperature adjustment to the data input assumptions. The irradiation times and phosphorus contents were varied, and the correlation in NUREG/CR-6551 was used to determine the temperature effect. The resulting chemistry factor for weld heat 12008/27204 was within 0.1°F (less than 0.05 percent) of the chemistry factor that was calculated in the original analysis. The licensee's and the staff's chemistry factor values are discussed below.

The licensee determined the chemistry factor for the FCS vessel weld using: (a) the Mihama Unit 1 surveillance data, (b) the ratio procedure that is recommended in 10 CFR 50.61 when the chemistry of the surveillance weld is different than the vessel weld, (c) an irradiation temperature adjustment, and (d) the calculational procedures that are recommended in 10 CFR 50.61. The best-estimate chemistry of the FCS vessel weld is 0.219 percent copper and 0.996 percent nickel. The best-estimate chemistry of the Mihama Unit 1 surveillance weld is 0.19 percent copper and 1.08 percent nickel. The ratio of the chemistry factor of the vessel weld to the chemistry factor of the surveillance weld was 1.017. The chemistry factor calculated by the licensee was 206.6°F (using correlation 4-1a from NUREG/CR-6551 to get the irradiation temperature adjustment, then performing credibility analysis prior to calculation of the chemistry factor).

The staff determined the chemistry factor for the FCS vessel weld using its best-estimate chemistry (0.219 percent copper and 0.996 percent nickel) and the surveillance weld from Mihama Unit 1 that was discussed above. The ratio of the chemistry factor of the vessel weld to the chemistry factor of the surveillance weld was 1.017. The chemistry factor calculated by the staff was 209.9°F (using the "one degree per degree" approach, and adjusting for irradiation temperature after the chemistry factor is calculated from data that were evaluated to be credible).

Credibility criterion (C) in Section (c)(2)(i) of 10 CFR 50.61 indicates that the scatter of the measured $\Delta RT_{PTS}$ (shift) values must be less than 17°F for base metal and 28°F for welds. The licensee determined that the scatter for the Mihama Unit 1 surveillance weld data is less than 28°F. Evaluation of this criterion was the basis for the licensee's determination that the Mihama Unit 1 weld surveillance data met the credibility criteria in 10 CFR 50.61. The licensee proposed that the calculated chemistry factor from the surveillance data (206.6°F) be used in determination of $\Delta RT_{PTS}$ and $RT_{PTS}$.

The staff independently evaluated the scatter of the measured $\Delta RT_{PTS}$ values and determined that the weld surveillance data satisfied criterion (C) in Section (c)(2)(i) of 10 CFR 50.61.

Hence, the staff concluded that the surveillance data is credible and can be used to determine the chemistry factor for the vessel weld.

### 3.8 Margin Value

The licensee calculated the margin value in accordance with the methodology in 10 CFR 50.61. The licensee used the generic $RT_{NDT(U)}$ value of -56°F with the associated standard deviation of 17°F. Section 50.61 recommends that the standard deviation for the shift in reference temperature be reduced by half if surveillance data is credible. The licensee used a standard deviation of 14°F for the shift in reference temperature since the Mihama Unit 1 surveillance data was found to be credible. The licensee calculated a margin value of 44°F. This value is acceptable since it was calculated in accordance with the methodology in 10 CFR 50.61.

The staff also used the generic $RT_{NDT(U)}$ value of -56°F with the associated standard deviation of 17°F. In addition, the staff used a standard deviation of 14°F for the shift in reference temperature since the Mihama Unit 1 surveillance data was found to be credible. The staff calculated a margin value of 44°F in accordance with 10 CFR 50.61.

### 3.9 Projected $RT_{PTS}$ Value at Expiration of License

The $RT_{PTS}$ value calculated by the licensee at expiration of license (EOL) for the weld fabricated from weld wire heat 12008/27204 is 226°F. The $RT_{PTS}$ value calculated by the staff for the same weld is 229°F. The staff's value is calculated using (a) the generic value of the initial reference temperature, (b) best-estimate values of copper and nickel for the vessel and surveillance welds, (c) a chemistry factor calculated from surveillance data and adjusted to account for the difference between the best-estimate chemistry of the FCS vessel and Mihama Unit 1 surveillance weld, (d) an EOL neutron fluence of 1.728E19n/$cm^2$, and (e) a margin value of 44°F. The difference between the staff's and the licensee's $RT_{PTS}$ values is due to the licensee’s use of the NUREG/CR-6551 correlation for the irradiation temperature adjustment. The staff concluded that the licensee’s value of $RT_{PTS}$ is acceptable for the weld fabricated from tandem weld wire heat 12008/27204 since the difference is small (3°F) when compared to the staff’s value, and this material is no longer the limiting beltline weld.

Using the Mihama Unit 1 weld surveillance data for the FCS PTS evaluation indicates that the reactor pressure vessel would be below the PTS screening criteria at the expiration of its license. The use of the Mihama Unit 1 data also resulted in a new limiting material for the FCS reactor vessel. The new limiting material is the weld fabricated from weld wire heat 12008/13253. The $RT_{PTS}$ value for this weld is 250°F (calculated from an initial $RT_{NDT(U)}$ value of -56°F with the associated standard deviation of 17°F, a chemistry factor from the tables in 10 CFR 50.61, and a margin value of 65.5°F). Since the chemistry factor for this weld is calculated from the 10 CFR 50.61 tables, the licensee’s and the staff’s values are the same.

### 3.10 Evaluation of Tandem Weld Wire Heat 27204

As mentioned in Section 2.0 of this SE, the licensee indicated that it plans to submit a license renewal application for FCS. The licensee also noted that using the current rate of embrittlement, welds fabricated using tandem weld wire heat 27204/27204 would exceed the PTS screening criteria before the proposed license renewal period ends. The licensee proposed to use surveillance data from Diablo Canyon Unit 1 and from a supplemental capsule in Palisades to calculate $RT_{PTS}$ with a reduced margin to determine the rate of embrittlement for welds fabricated with tandem heat 27204/27204. Although this SE evaluates PTS for the current license, the staff reviewed the results of the 27204/27204 analysis. $RT_{PTS}$ values for this material and all of the beltline materials, were provided for both the end of the current license and the end of the proposed license renewal period.

The Diablo Canyon Unit 1 surveillance welds have Cu and Ni contents of 0.20 percent and 1.00 percent respectively. The Palisades supplemental capsule has Cu and Ni contents of 0.19 percent and 1.07 percent, respectively. The ratio of the chemistry factor of the FCS vessel weld to the chemistry factor of the surveillance weld was 1.022 for the Diablo Canyon Unit 1 data and 0.990 for the Palisades data. The licensee calculated a chemistry factor of 215.5°F based on shifts adjusted for best estimate chemistry and irradiation temperature using the ratio procedure and the NUREG/CR-6551 correlation. The staff calculated a chemistry factor of 210.2°F based on shifts adjusted for best estimate chemistry and irradiation temperature (irradiation temperature adjustment made after chemistry factor calculation). The staff's analysis method of performing the credibility evaluation prior to the irradiation temperature adjustment resulted in the measured minus predicted shift for Diablo Canyon Unit 1 capsule S falling 1°F outside of the standard deviation for welds (28°F). However, this small difference is negative and therefore conservative since the measured shift value was 29°F less than the predicted shift value.

The $RT_{PTS}$ value calculated by the licensee at EOL for the weld fabricated from weld wire heat 27204/27204 is 236°F. The $RT_{PTS}$ value calculated by the staff for the same weld is 230°F. The difference between the staff's and the licensee's $RT_{PTS}$ values is due to the licensee's use of the NUREG/CR-6551 correlation for the irradiation temperature adjustment. The licensee calculated a projected $RT_{PTS}$ value of 255°F for the end of the proposed license renewal period. Therefore, this material would be below the PTS screening criteria for the proposed period of extended operation. According to paragraph (B)(v)(2) and footnote 5 of 10 CFR 50.61 (the PTS rule) the licensee must assess the impact of changes (if any) to the FCS PTS evaluation that result from changes in surveillance data from Diablo Canyon Unit 1 and Palisades.

The new limiting weld for the current license (weld heat 12008/13253) is also projected to be limiting for the proposed license renewal period. Further assessment of the other reactor pressure vessel (RPV) welds in the FCS RPV will need to be addressed as part of the license renewal application review.

Based on its review, the staff has concluded the following:

(1) Results from the Kansai Electric surveillance program are compatible with those from domestic U.S. programs, since Kansai Electric used the same ASTM standard and chemical analysis as in the U.S.

(2) The Mihama Unit 1 weld surveillance data met the credibility criteria in 10 CFR 50.61. The weld data was determined to be acceptable for use in the FCS PTS evaluation by comparison of the irradiation environments.

(3) Specifically, since the Mihama Unit 1 weld surveillance data met the credibility criteria of 10 CFR 50.61, the data was used to determine the chemistry factor for the limiting FCS vessel weld. The analysis of the Mihama Unit 1 data resulted in a new limiting material for the FCS reactor vessel. The new limiting material is the weld fabricated from weld wire heat 12008/13253. The $RT_{PTS}$ value for the new limiting weld is 250°F.

(4) The licensee's and staff's calculated values of $RT_{PTS}$ for FCS at expiration of license are below the 270°F screening criterion specified in 10 CFR 50.61 for axial welds.

(5) Since the conclusions in (3) and (4) are dependent upon the available chemistry and surveillance data, they are subject to change when new data becomes available. It should also be noted that OPPD must track and assess any changes in the Mihama Unit 1 (weld heat 12008/27204), Diablo Canyon Unit 1, and Palisades (weld heat 27204/27204) data that would effect the FCS PTS evaluation. According to paragraph (B)(v)(2) and footnote 5 of 10 CFR 50.61 (the PTS rule), OPPD must assess the impact of changes (if any) to the FCS PTS evaluation that result from changes in surveillance data from Mihama Unit 1, Diablo Canyon Unit 1 and Palisades.

(6) This SE evaluates the PTS for the current license, however, the staff reviewed the PTS evaluation for tandem weld heat 27204/27204 which was projected to the end of the proposed license renewal period. The projected $RT_{PTS}$ value for the end of the proposed license renewal period is 255°F. Therefore, this material is no longer limiting for the proposed period of extended operation. Further assessment of the other RPV welds in the FCS RPV evaluation for the proposed period of extended operation will need to be addressed as part of the license renewal application review.

(7) The staff reviewed the submitted information regarding the proposed fluence value at FCS and finds that it is acceptable because the methodology, the approximations and the cross sections used in the evaluation satisfy the DG-1053 recommendations. The numerical value was estimated for a load factor of 0.85 which is greater than the load factor of 0.77 used in previous estimates. This load factor is acceptable because: (1) the projected total EFPYs will be lower than 32, and (2) the projected critical element satisfies the screening criteria of 10 CFR 50.61.

The amendment will delete Section 3.D, "License Term," from the FCS operating license. The licensee's analysis resulted in a new limiting beltline material which is the weld fabricated from tandem weld wire heat 12008/13253. The increase in the long term load factor from 0.77 to 0.85 did not cause the critical weld material to exceed the reference temperature ($RT_{PTS}$) screening criteria of 10 CFR 50.61 (the PTS rule). Therefore, the staff has concluded that the FCS reactor vessel is projected to be below the PTS screening criteria of 10 CFR 50.61 at the expiration of its current license (August 9, 2013) as well as the end of the proposed license renewal period (August 9, 2033). The staff notes that paragraph (B)(v)(2) and footnote 5 of 10 CFR 50.61 requires that the licensee must assess the impact of changes to the FCS PTS evaluation that result from new surveillance data. Specifically, new data from the Mahima Unit 1, Diablo Canyon Unit 1 and Palisades plants must be assessed as it becomes available, since the data from these plants was used in the FCS PTS analysis. Based on the new analysis that demonstrates that the limiting weld is within the current PTS screening criteria of 10 CFR 50.61, the requirements of 10 CFR 50.61 that assure the analysis remains valid, and given that the requirements in Section 3.D are redundant to 10 CFR 50.61 requirements, as 10 CFR 50.61 requires updating this assessment whenever there is a significant change in projected values of $RT_{PTS}$, the staff has concluded that the request to delete license condition 3.D is acceptable. The licensee's analysis assumes that future core loadings will be such as to limit the core neutron leakage to values similar to those for Cycles 15 and 16 to limit the end of license fluence accumulation to $1.728 \times 10^{19} n/cm^2$ to the limiting welds. Therefore, the design of future cores must satisfy the above limitation and in addition caution must be exercised to preclude misloading any of the peripheral assemblies which could invalidate the loading requirements.

## 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 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 (66 FR 2019). 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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