Nebraska Public Power District

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NSD920629
July 1, 1992

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

Gentlemen:

Subject: Response to Generic Letter 92-01, Revision 1
Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46

Reference: Letter from J. G. Partlow (NRC) to all Licensees dated March 6, 1992, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01, Revision 1)"

The Nebraska Public Power District (District) hereby provides its response to Generic Letter 92-01, Revision 1 issued March 6, 1992 (Reference). Generic Letter 92-01 Revision 1 supersedes Generic Letter 92-01 which was issued February 28, 1992. Generic Letter 92-01, Revision 1 requests licensees to provide information to demonstrate compliance with 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Reactors for Normal Operation," and 10 CFR 50 Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements." The District has verified that Cooper Nuclear Station (CNS) meets the requirements of these regulations. The District's responses to the Generic Letter 92-01, Revision 1 questions are provided in the attachment.

As requested in Generic Letter 92-01 Revision 1 and pursuant to 10 CFR 50.54(f), this response is submitted under oath.

Please contact me if you have any questions or require any additional information.

Sincerely,

Guy Horn

G. R. Horn
Nuclear Power Group Manager

GRH:MJB/MHM

Attachment

cc: NRC Regional Office
Region IV
Arlington, TX

NRC Resident Inspector
Cooper Nuclear Station

A028
1/1

STATE OF NEBRASKA)
)
PLATTE COUNTY)

G. R. Horn, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this response on behalf of Nebraska Public Power District; and that the statements contained herein are true to the best of his knowledge and belief.

G. R. Horn

Subscribed in my presence and sworn to before me this 1st day of July, 1992.

GENERAL NOTARY-State of Nebraska
CAROLYN M. LANGAN
My Comm. Exp. Aug. 21, 1995

Carolyn M. Langan
NOTARY PUBLIC

RESPONSE TO GENERIC LETTER 92-01 REVISION 1
REACTOR VESSEL STRUCTURAL INTEGRITY

I. INTRODUCTION

The information in this attachment provides the Nebraska Public Power District's (District's) response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity."[1/] Generic Letter 92-01, Revision 1 replaced Generic Letter 92-01, dated February 28, 1992.[2/] Generic Letter 92-01 Revision 1 clarified some of the information pertaining to the review of the Yankee Nuclear Power Station structural integrity, but did not change the information requirements.

Generic Letter 92-01, Revision 1 requests licensees to respond to a number of questions intended to verify that licensees have adequately addressed the requirements contained in 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," 10 CFR 50 Appendix G, "Fracture Toughness Requirements," 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and for Pressurized Water Reactors, 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." The following discussion provides the District's response to Generic Letter 92-01, Revision 1.

II. DISCUSSION

The District's response to the specific information requirements contained in Generic Letter 92-01, Revision 1 is provided below.

Request 1

Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR

1/ Letter from J. G. Partlow (NRC) to all Licensees dated March 6, 1992, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01, Revision 1)."

2/ Letter from J. G. Partlow (NRC) to all Licensees dated February 28, 1992, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)."

Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

District Response

Although Cooper Nuclear Station (CNS) is not listed in Enclosure 2 of Generic Letter 92-01, Revision 1, it is the District's position that its surveillance program for Cooper Nuclear Station is currently in compliance with 10 CFR 50 Appendix H, and has been acknowledged previously by the NRC as being in compliance with Appendix H. 10 CFR 50 Appendix H states:

> "That part of the surveillance program conducted prior to the first withdrawal must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased."

Additionally, Appendix H states:

> "For each capsule, withdrawn after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practical for the configurations of the specimens in the capsule."

The CNS vessel was designed to the Winter 1966 Addenda of the 1965 ASME Code. ASTM E 185-66 was the standard in place at the time the surveillance program was designed. Therefore, the vessel material surveillance specimens were fabricated and located within the CNS vessel in accordance with that guidance, and accordingly, the fabrication and installation of the CNS vessel surveillance specimens comply with the requirements of 10 CFR 50 Appendix H.

The first surveillance capsule was withdrawn during the CNS Reload 9 Cycle 10 refueling outage in 1985 following 6.8 Effective Full Power Years (EFPY) of operation, and tested, to the extent practical, in accordance with ASTM E 185-82. Following testing and analysis of the surveillance specimens, the District submitted a proposed change to the CNS Technical Specifications to revise the CNS pressure-temperature operating limit curves accordingly. In its safety evaluation approving Amendment No. 120 to the CNS operating license[3/] the NRC recommended, to meet ASTM E 182-82 as closely as possible, that the District accelerate the withdrawal schedule of the second surveillance capsule to 12 EFPY, and consider insertion of a fourth capsule into the CNS vessel, possibly with reconstituted specimens from an earlier capsule.

[3/] Letter from W. O. Long (NRC) to G. A Trevors (NPPD) dated April 26, 1988, "Cooper Nuclear Station - Amendment No. 120 To Facility Operation License No. DPR-46 (TAC No. 65793)."

Following various communications, and in support of a license change application to extend the operating license expiration date to 40 years from receipt of the CNS operating license, the District committed to 1) remove the second surveillance capsule during the Reload 14, Cycle 15 refueling outage during 1991 (following approximately 11 EFPY of operation), and 2) reconstitute the specimens from this capsule and re-insert the reconstituted specimens during the Reload 15, Cycle 16 refueling outage. The District also indicated that the withdrawal schedule for the third capsule will be based on the results of testing the second surveillance capsule.[4/]

In its safety evaluation accompanying Amendment No. 143 to the CNS operating license which extended the CNS license expiration date to January 18, 2014,[5/] the NRC acknowledged the District's commitment to reconstitute the surveillance capsule withdrawn during the 1991 refueling outage. The NRC stated further that the reconstitution of the capsule withdrawn in 1991 is equivalent to a fourth capsule and thereby makes the District's surveillance program consistent with the requirements of ASTM E-185-82 and 10 CFR Part 50 Appendix H. The NRC also acknowledged that the withdrawal schedule for the original third capsule and the reconstituted fourth capsule should be based on the results of the analysis of the second capsule.

The District has withdrawn the second surveillance capsule from the CNS reactor vessel; these specimens are currently undergoing testing and analysis, and will be reconstituted and reinserted in the CNS reactor vessel during the Reload 15 Cycle 16 currently scheduled to commence in March 1993.

Therefore, due to changes in ASTM E 185 surveillance specimen fabrication requirements between 1966, and 1973, the District's vessel material surveillance program for CNS does not strictly comply with ASTM E 185-73 or later revisions. However, as discussed above and as acknowledged previously by the NRC, the CNS surveillance program does meet, as closely as possible, the requirements of ASTM E 185-82, and therefore is in compliance with 10 CFR 50 Appendix H. Further details on the CNS surveillance program are provided in the responses to Requests 2.b.4 and 2.b.5.

In addition, the District continues to participate in industry-sponsored efforts to gather additional data on radiation embrittlement. The

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[4/] Letter from G. R. Horn (NPPD) to NRC dated June 7, 1991, "Response to Questions on License Extension to 40 Years from Operating License Issuance."

[5/] Letter from P. W. O'Connor (NRC) to G. R. Horn dated July 5, 1991, "Cooper Nuclear Station - Amendment No. 143 to Facility Operating License No. DPR-46 (TAC No. 74843)."

District is a member of the Boiling Water Reactor Owners' Group (BWROG) Supplemental Surveillance Program (SSP) Committee, and CNS is participating as a host reactor. The objective of the SSP is to develop supplemental surveillance data which will allow the District to better understand the extent of beltline embrittlement with increasing fluence for a variety of plate and weld materials. The first supplemental surveillance capsule was installed in CNS during its 1991 Refueling Outage. The testing being undertaken by the SSP Committee will greatly increase the BWR surveillance data base. Descriptions of the SSP test program and hardware are presented in the Committee's Phase 1 report and Phase 2 Progress report. These reports are being prepared for submittal to the NRC in the near future.

Request 2.a

Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.

District Response

The Upper Shelf Energies (USE) of the beltline materials at CNS are not expected to be less than 50 ft-lb by the end of its licensed operating period. The following discussion provides a brief description of the USE evaluation supporting this conclusion.

USE data were taken during fabrication of the plates used in the CNS beltline. Evaluation of these plates according to Regulatory Guide 1.99, Revision 2, shows the USE at 32 Effective Full Power Years (EFPY) to be above 50 ft-lb. For the beltline submerged arc welds, fabrication USE data were not taken; rather, fabrication Charpy tests were performed at 10°F with a 30 ft-lb requirement. However, testing performed on the vessel surveillance weld, which is representative of the beltline welds, demonstrated an unirradiated USE of 112 ft-lb. Evaluation according to Regulatory Guide 1.99, Revision 2 of the beltline welds assuming initial USE of 112 ft-lb results in 32 EFPY USE predictions well above 50 ft-lb.

In addition, the District is a member of the BWROG RPV Fracture Toughness Committee. The objective of that committee is to develop methods to estimate USE in cases where plant beltline materials were tested at only transition temperatures; the methods being developed will greatly increase the BWR fracture toughness data base. A description of the method is presented in the BWROG Fracture Toughness Committee Report which was transmitted to the NRC in June, 1992.[6]

Request 2.b

b. Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

Subpart 2.b.(1)

(1) the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;

District Response

As indicated above, the CNS vessel was fabricated to the Winter 1966 Addenda of the 1965 ASME Code. For the CNS vessel beltline plate materials, Charpy and dropweight tests were performed. The Charpy specimen orientation was longitudinal and the test requirement was to meet 30 ft-lb at the specified temperature. In order to demonstrate fracture toughness equivalent to Appendix G requirements, a General Electric procedure, described in the Cooper Nuclear Station Reactor Pressure Vessel Surveillance Materials Testing and Fracture Toughness Analysis report[7] was used to adjust the 30 ft-lb longitudinal Charpy data to determine the temperature T_{50T} at which an equivalent 50 ft-lb transverse Charpy energy could be expected. The unirradiated RT_{NDT} was then selected as the higher of (T_{50T}-60°F) or the dropweight Nil-Ductility Temperature (NDT).

For the beltline weld materials, only Charpy tests were performed. The specimens were cut transverse to the weld length and the test requirement was 30 ft-lb at 10°F. As with the plate, the GE procedure was used to adjust the 30 ft-lb Charpy data to determine T_{50T} and to account for the absence of dropweight testing data. The unirradiated RT_{NDT} was determined from the procedure as the higher of either (T_{50T}-60°F) or -50°F.

6/ Letter from C. L. Tully (BWROG) to NRC dated June 12, 1992, "BWR Owners' Group Submittal on Upper Shelf Energy Estimation."

7/ Report MDE-103-0986, dated May 1987, "Cooper Nuclear Station Reactor Pressure Vessel Surveillance Materials Testing and Fracture Toughness Analysis," submitted to the NRC by letter from G. A. Trevors (NPPD) to NRC dated July 6, 1987, "Reactor Vessel Material Surveillance Program, NRC Docket No. 50-298/DPR-46."

Charpy data, dropweight test results and estimated RT_{NDT} values for the beltline materials are provided in Appendix A of this attachment.

Subpart 2.b.(2)

(2) the heat treatment received by all beltline and surveillance materials;

District Response

Heat treatment was not explicitly considered in the Appendix G analysis, as there are no requirements or methods provided which relate to heat treatment. However, implicit in the Appendix G analysis is the assumption that the Charpy data used to develop the RT_{NDT} values is representative of the beltline materials, so the heat treatment of Charpy specimens should represent or bound that of the beltline materials.

After the beltline plates were quenched and tempered, specimen samples and plate used in the surveillance program were trimmed from the plates. The specimen samples and surveillance materials received a simulated Post-Weld Heat Treatment (PWHT) at 1150°F±25°F for 40 hours. The beltline material PWHT temperature was the same, but the beltline PWHT time was significantly less. The additional PWHT time for the specimens was intended to cover the possibility of future vessel repairs requiring subsequent PWHT. Since the surveillance specimen PWHT was as long or longer than the beltline PWHT time periods, the surveillance specimens and the corresponding surveillance data are representative of the vessel beltline materials.

Subpart 2.b.(3)

(3) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;

District Response

The beltline consists of portions of the lower shell and lower-intermediate shell. Each shell is formed from three plates, so the beltline includes portions of six plates, six vertical welds and one girth weld. All beltline plate and weld materials were considered in the Appendix G Evaluation. The requested information is provided in Appendix A to this letter. The District is continuing its search through vessel fabrication records to determine the material chemistries for the beltline weld materials identified in Appendix A as "not available."

Subpart 2.b.(4)

(4) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;

District Response

Appendix G includes, by reference in paragraph III.A, Appendix H, which includes by reference "the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased." For CNS, this is ASTM E 185-66, which required that "test specimens shall be taken from materials used in the irradiated region." ASTM E 185-66 further states that "Samples shall represent one heat of the base metal and one butt weld if a weld occurs in the irradiated region."

The surveillance plate material was trimmed from the beltline plate with heat number C2307-2. The District is continuing its efforts, through the review of the original CNS vessel fabrication documentation to determine the specific wire heat number and flux lot data for the surveillance weld. Notwithstanding this effort, the specification for the surveillance weld required that it be made with the same procedure as the longitudinal beltline weld 1-233, i.e., MIL-E-18193 type B-4 modified wire with flux type Linde 1092, same wire feed rate, heat input, etc. Therefore, the data available on the surveillance weld indicates that it is representative of the beltline welds, as required by ASTM E 185-66. More importantly, the available surveillance weld data provides the District all of the information needed to meet the objective of Appendix H to monitor toughness changes due to irradiation.

The usefulness of the results from surveillance weld testing is augmented by the following:

- Archive surveillance weld material has been tested, providing complete baseline Charpy curve and chemical composition data; this information was provided to the NRC in Report MDE-103-0986, which was transmitted to the NRC by letter dated July 6, 1987.[8]

- Irradiated surveillance weld Charpy specimen test curves can be compared credibly with the baseline Charpy curve. The copper and nickel content are known and the fluence is established from dosimetry in each surveillance capsule. Therefore, all necessary information is available to compare the surveillance weld

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[8] Letter from G. A. Trevors (NPPD) to NRC dated July 6, 1987, "Reactor Vessel Surveillance Program, NRC Docket No. 50-298, DPR-46."

irradiation embrittlement with Regulatory Guide 1.99, Revision 2 predictions.

Subpart 2.b.(5)

(5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material; and

District Response

Chemical composition weight percent data for beltline materials are shown in Appendix A to this letter. In some cases for weld materials, upper bound assumptions of 0.35% copper and 1.0% nickel were used in the absence of actual chemistry data. Beltline material chemistries, or upper bound assumptions, were used with Regulatory Guide 1.99, Revision 2 to determine the limiting beltline material, the adjusted reference temperature versus EFPY for the material, and the predicted plate USE at 32 EFPY.

Chemical composition weight percent data for the surveillance plate and weld are shown in Appendix B to this letter. The chemical composition data were used to compare measured Charpy curve shifts with Regulatory Guide 1.99, Revision 2 predictions. The surveillance weld chemistry was used to estimate weld USE at 32 EFPY.

Subpart 2.b.(6)

(6) the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.

District Response

This does not apply to CNS; see the response to Item 2.b.(3) above.

Request 3

Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

Subpart 3.a

a. How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525°F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.

District Response

Operation with the CNS beltline region below 525°F was not considered in the Appendix G analysis because the steady state operating temperature of the coolant in the beltline region is slightly greater. Based on the temperature in the recirculation suction piping (which draws water directly from the beltline region) the steady state temperature in the beltline is greater than 527°F.

Only during start-up and operation without feedwater heating, which occurs when feedwater heaters are out of service or when the turbine is off line and the reactor steam is routed through the turbine bypass, does the beltline experience coolant less than 525°F when the core is critical. The equivalent full power time of operation in these transient conditions has been estimated to be less than 1%, and the associated temperatures for most of that time are 515°F or higher. The CNS 32 EFPY fluence is 1.5×10^{18} n/cm^2, so the fluence accumulated below 525°F would be approximately 1.5×10^{16} n/cm^2. This combination of low fluence and small deviation from the 525°F level is not expected to significantly affect beltline RT_{NDT} or USE predictions.

Since surveillance specimens are exposed to the same temperature conditions as the beltline materials, temperature effects, if any, will be reflected in the surveillance results. When the surveillance results are factored into the Appendix G analysis per Regulatory Guide 1.99, Revision 2, temperature effects, if any, will be accounted for inherently.

In addition, the BWROG Supplemental Surveillance Program will collect data periodically over the next 8 years. The capsules will include eutectic temperature monitors with which to determine the appropriate maximum irradiation temperature. The data will include results of some PWR materials and HSST-02 standard material. These data should provide additional insight into temperature effect differences between the BWR and PWR environments.

Subpart 3.b

b. How their surveillance results on the predicted amount of embrittlement were considered.

District Response

Surveillance results were factored into beltline embrittlement predictions following receipt of results of testing the first surveillance capsule. At this time, Regulatory Guide 1.99, Revision 1 was the current guidance. As a result, the District submitted Proposed Change No. 48 to the CNS

Technical Specifications[9], which provided revised pressure-temperature limitation curves based on the methods provided in Regulatory Guide 1.99, Revision 1, and adjusted conservatively for the results of the surveillance testing. In response to a request for additional information, the District submitted to the NRC additional details to support the proposed change.[10] The NRC evaluated the proposed change against Regulatory Guide 1.99, Revision 2, which was approved at that time, although awaiting publication as a final guide. As a result, the NRC issued Amendment No. 120 to the CNS Operating License,[11] concluding in its safety evaluation:

> "Since the ART [Adjusted Reference Temperature] values used by the licensee to calculate the proposed Pressure-Temperature limits are greater than the values predicted using the formula in R.G. 1.99, Rev. 2, the proposed Pressure-Temperature limits will meet R.G. 1.99, Rev. 2."

Additionally, the safety evaluation concluded:

> "To confirm that the Pressure-Temperature limits proposed by the licensee will meet the safety margins of Appendix G, 10 CFR Part 50 for the proposed operating periods, the staff has used the method of calculating Pressure-Temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed Pressure Temperature limits. The staff's calculation includes the licensee's ART values. Our calculations confirm that the proposed Pressure-Temperature limits meet the safety margins of Appendix G, 10 CFR Part 50 for the operating periods identified on the curves."

Therefore, using Regulatory Guide 1.99, Revision 1 guidance, adjusted conservatively for measured reference temperature shift, the District

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[9] Letter from L. G. Kuncl (NPPD) to NRC dated October 28, 1987, "Proposed Change No. 48 to the Cooper Nuclear Station Technical Specifications, NRC Docket No. 50-298, DPR-46"

[10] Letter from G. A. Trevors to NRC dated February 22, 1988, "Supplemental Submittal; Proposed Change No. 48 to the Cooper Nuclear Station Technical Specifications, NRC Docket No. 50-298, DPR-46"

[11] Letter from W. O. Long (NRC) to G. A. Trevors (NPPD) dated April 26, 1988, "Cooper Nuclear Station - Amendment No. 120 to Facility Operation License No. DPR-46 (TAC No. 65793)."

demonstrated that the CNS operating limits enveloped the constraints provided by 10 CFR Appendix G.

Regulatory Guide 1.99, Revision 2, paragraph C.2, requires credible data from two surveillance capsules before adjustments to the predictions methods are made. Only one capsule has been tested; the second surveillance capsule is currently undergoing testing and analysis. Therefore, although the current beltline predictions are based on Regulatory Guide 1.99, Revision 1 methods adjusted conservatively for measured reference temperature shift, these predictions have been shown to be conservative with respect to Regulatory Guide 1.99, Revision 2 methods. The District will reevaluate these predictions when the results of testing the second surveillance capsule become available.

Subpart 3.c

c. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.

District Response

Measured increases in reference temperature and measured decreases in USE based on the first surveillance capsule are provided in Appendix C to this letter. Measured increases in reference temperature from the first surveillance capsule were within the mean-plus-2σ Regulatory Guide 1.99, Revision 2 prediction for the weld, but not for the plate. As discussed in the response to 3.b above, the current CNS pressure-temperature operating limit curves were developed based on the Regulatory Guide 1.99, Revision 1 methods, adjusted for the increase in reference temperature measured from the surveillance test specimens, resulting in predicted reference temperature shifts greater than that predicted using Regulatory Guide 1.99, Revision 2 methods.

Measured decreases in USE from the first surveillance capsule were within the prediction for the plate, but not for the weld. Since only one surveillance result is available, the effect of the measured decreases on the beltline predictions has not been considered, per paragraph C.2 of Regulatory Guide 1.99, Revision 2.

I. CONCLUSION

It is the District's position that it is in full compliance with 10 CFR 50.60 and 10 CFR 50 Appendices G and H. However, as the licensee of Cooper Nuclear Station, the District is committed to assuring that the plant is always operated safely within the bounds of its design. Accordingly, the District will continue its multi-pronged approach to obtain additional radiation embrittlement data. The District is continuing its search of vessel fabrication records to locate additional CNS vessel beltline weld information. In addition, the District is working with the BWROG on two separate programs relating to embrittlement issues. The District is actively involved with the BWROG Fracture Toughness Committee which has developed a topical report on USE estimation methods for plants where only transition data is available. The District is also a member of the BWROG Supplemental Surveillance Committee of which CNS is a host reactor. This effort will provide additional CNS and BWR-specific embrittlement data through the testing of additional surveillance materials. The District will proceed with these efforts to assure that CNS continues to operate safely.

APPENDIX A

BELTLINE MATERIAL CHEMISTRY AND RT_{NDT} INFORMATION

CHEMICAL COMPOSITION OF RPV BELTLINE MATERIALS

Identification	Heat/Lot No.	Composition by Weight Percent							
		C	Mn	P	S	Si	Ni	Mo	Cu
Lower Plates:									
G-2803-1	C2274-1	0.20	1.31	0.010	0.018	0.21	0.68	0.45	0.20
G-2803-2	C2307-1	0.23	1.30	0.009	0.015	0.21	0.73	0.45	0.21
G-2803-3	C2274-2	0.23	1.32	0.009	0.017	0.22	0.68	0.46	0.20
Lower-Intermediate Plates:									
G-2801-7	C2407-1	0.22	1.35	0.010	0.016	0.24	0.65	0.46	0.13
G-2802-1	C2331-2	0.21	1.35	0.010	0.017	0.22	0.58	0.48	0.17
G-2802-2	C2307-2	0.23	1.25	0.010	0.014	0.20	0.73	0.47	0.21
Lower Longitudinal Welds:									
2-233 A,B,C	Heat 12420, Flux 1092, Lot 3724	not available							
	Heat 12420, Flux 1092, Lot 3708	not available							
Lower-Intermediate Longitudinal Welds:									
1-233 A,B,C Tandem Weld	Heat 27204 with [a] Heat 12008, Flux 1092, Lot 3724	-	1.16	0.013	0.007	0.21	0.97	0.46	0.19
Lower to Lower-Int. Girth Weld:									
1-240	Heat 21935, Flux 1092, Lot 3869	0.17		0.016	0.010				0.20

[a] Mihama, Unit 1 Surveillance Test Report, Kobe Technical Institute, Sept. 1973.

RESULTS OF FABRICATION TEST PROGRAM FOR SELECTED RPV LOCATIONS

Location	Ident. Number	Heat Number	Tensile: Yield (ksi)	Tensile: UTS (ksi)	Tensile: Total Elong (%)	Tensile: Area Reduc. (%)	Test Temp. (°F)	Charpy Energy (ft-lb)	NDT (°F)	$T_{50T}-60$ (°F)	RT_{NDT} (°F)
Beltline:											
Lower Shell Plates	G-2803-1	C2274-1	63.6	86.5	29.5	69.4	10	50,40,33	-10	14	14
	G-2803-2	C2307-1	69.6	91.3	27.5	68.4	10	40,70,50	-10	0	0
	G-2803-3	C2274-2	68.8	90.6	26.5	68.3	10	58,44,49	-10	-8	-8
Lower-Intermediate Shell Plates	G-2801-7	C2407-1	71.2	91.2	25.5	70.9	10	50,68,63	-10	-20	-10
	G-2802-1	C2331-2	72.1	92.9	28.0	68.1	40	58,70,59	-40	10	10
	G-2802-2	C2307-2	71.4	91.9	28.3	67.4	10	89,73,72	-40	-20	-20
Longitudinal Weld	2-233	Ht. 12420 Lot 3724	65.3	81.9	31.0	69.3	10	64,69,56	n/a	-50	-50
Girth Weld	1-240	Ht. 21935 Lot 3869	70.5	86.8	28.0	69.8	10	62,59,60	n/a	-50	-50
Non-Beltline:											
Upper Shell Plate	G-2801-4	C2327-1	70.6	92.6	26.3	68.6	10	45,40,33	-20	14	14
Vessel Flange	G-2809	AWC-67	73.7	95.8	24.0	69.0	10	62,76,65	10	-20	10
Head Flange	G-2810	AXD-68	70.0	91.8	26.5	71.5	10	96,89,74	20	-20	20
Top Head Torus	G-2812	C2660-2	68.2	87.9	29.8	69.8	10	52,67,79	-10	-20	-10
Bottom Head Torus	G-2806-1	C2137-3	65.7	85.7	27.3	69.0	10	39,26,36	-10	28	28
Closure Bolts	G-2863	37385	153.4	167.5	14.3	46.1	10	38,36,35	n/a	LST = 70°F	

APPENDIX B

SURVEILLANCE MATERIAL CHEMISTRY INFORMATION

CHEMICAL COMPOSITION OF SURVEILLANCE MATERIALS

Identification	Composition by Weight Percent							
	C	Mn	P	S	Si	Ni	Mo	Cu
Plate:								
Heat C2307-2	0.23	1.25	0.010	0.014	0.20	0.73	0.47	0.21
Irrad. Specimen J64	-[a]	1.23	0.007	-	-	0.77	0.51	0.22
Irrad. Specimen J6L	-	1.30	0.006	-	-	0.78	0.50	0.22
Weld:								
Unirradiated Weld	0.15	1.37	0.012	0.010	0.24	0.73	0.51	0.22
Irrad. Specimen J74	-	1.26	0.012	-	-	0.75	0.55	0.23
Irrad. Specimen J7D	-	1.37	0.012	-	-	0.74	0.52	0.22

[a] A dash (-) mark denotes an element that was not evaluated.

APPENDIX C

COMPARISON OF SURVEILLANCE RESULTS WITH
REG. GUIDE 1.99, REVISION 2 PREDICTIONS

COMPARISON OF SURVEILLANCE RESULTS WITH
REG. GUIDE 1.99, REVISION 2 PREDICTIONS
(Fluence = 2.3×10^{17} n/cm^2)

REFERENCE TEMPERATURE INCREASE

Plate:

%Cu = 0.22 — Predicted ΔRT_{NDT} = 32.7°F
%Ni = 0.78 — Predicted $\Delta RT_{NDT}+2\sigma$ = 65.4°F

Measured Shift = 74°F

Weld:

%Cu = 0.23 — Predicted ΔRT_{NDT} = 36.7°F
%Ni = 0.75 — Predicted $\Delta RT_{NDT}+2\sigma$ = 73.4°F

Measured Shift = 55°F

USE DECREASE

Plate:

Unirradiated USE = 129 ft-lb
Irradiated USE = 112 ft-lb
Measured Decrease in USE = 17 ft-lb (13%)

Predicted Decrease in USE = 13%

Weld:

Unirradiated USE = 112 ft-lb
Irradiated USE = 85 ft-lb
Measured Decrease in USE = 27 ft-lb (24%)

Predicted Decrease in USE = 16%