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SUBJECT: Responds to GL 92-01, "Reactor Vessel Structural Integrity," re licensees' reactor vessel mtl surveillance program & fracture toughness.

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Iowa Electric Light and Power Company

JOHN F. FRANZ, JR.
VICE PRESIDENT, NUCLEAR

July 6, 1992
NG-92-2816

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Response to NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity"
File: A-101b, B-11

Dear Dr. Murley:

NRC Generic Letter 92-01 (Rev. 1), "Reactor Vessel Structural Integrity," requests information concerning licensees' reactor vessel material surveillance programs and fracture toughness. We have a material surveillance program for the Duane Arnold Energy Center (DAEC) reactor vessel, the details of which are included in the attachment to this letter. In addition, we are participants on the BWROG Supplemental Surveillance Program Committee which was formed to develop additional information on the effects of radiation on reactor vessel materials. We intend to use information gained through the efforts of that committee as a further aid in assessing the effects of radiation on the DAEC reactor vessel.

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Dr. Thomas E. Murley
July 6, 1992
NG-92-2816
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Should you require any further information, please contact this office.

This letter is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

By John F. Franz, Jr.
John F. Franz, Jr.
Vice President, Nuclear

State of Iowa
(County) of Linn

Signed and sworn to before me on this 2nd day of July, 1992, by John F. Franz, Jr.

Kathleen M. Furman
Notary Public in and for the State of Iowa

September 28, 1992
Commission Expires

JFF/SRC/pjv

Attachment: Response to NRC Generic Letter 92-01 (Rev.1), "Reactor Vessel Structural Integrity"

cc: S. Catron
L. Liu
L. Root
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NRC Resident Office
DCRC

Response to NRC Generic Letter 92-01 (Rev. 1)

"Reactor Vessel Structural Integrity"

Duane Arnold Energy Center

Question 1.

Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

> 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 Part 50, addresses 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).

Response:

ASTM E185 was originally issued in 1961, and was revised in 1966, 1970, 1973, 1979 and 1982. 10 CFR Part 50, Appendix H makes compliance with E185 a requirement. The DAEC reactor vessel was designed to the summer 1967 Addenda of the 1965 ASME Code with Code Cases 1332-4, 1335-2 and 1420. ASTM E185-66 was the standard in place at the time the DAEC Surveillance Program was designed, but the program was also evaluated again E 185-70.

The DAEC Surveillance Program was reviewed by the AEC as part of the FSAR approval. That review (Reference 14) concluded that the program complied with the then proposed regulations 10 CFR Part 50.55a and Appendix H. 10CFR Part 50 Appendix H, paragraph II.B.1 states, "That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of ASTM E185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased." Since the <u>design</u> of the surveillance program is "part of the surveillance program conducted prior to the first capsule withdrawal", the edition of the standard applicable to the design of the DAEC Surveillance Program is E185-66. While the DAEC Surveillance Program does not meet the total number of samples specified by that standard to establish base line or the notch location on HAZ, it does meet the intent of the most significant requirements, as detailed further in the responses to requests 2 and 3 below. Iowa Electric believes that, although the DAEC Surveillance Program was designed prior to the adoption of Appendix H, it meets the purpose of Appendix H to "monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region . . . resulting from exposure of these materials to neutron irradiation and the thermal environment." See Table 1.

The Surveillance Program meets 10 CFR Part 50 Appendix H, per paragraph II.B.1 and was previously approved through the FSAR approval. An exemption request is not considered necessary.

Iowa Electric is a member of the BWROG Supplemental Surveillance Program (SSP) Committee. The objective of that committee is to develop supplemental surveillance data which will allow Iowa Electric to better understand the extent of beltline embrittlement with increasing fluence. 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. This report is being prepared for submittal to the NRC in the near future (Reference 12). DAEC has submitted specimens (ten base metal and ten weld) to be included in the SSP, which will provide more plant specific data. These specimens will be installed in Oyster Creek's vessel.

Question 2

Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

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.

Response:

The upper shelf energies (USE) of the beltline materials at DAEC are not predicted to be less than 50 ft-lb by the end of the operating license (see Table 5). Therefore, request 2a does not apply to DAEC. A brief description of the USE evaluation supporting this conclusion follows:

USE data were taken during fabrication on two of four plates used in the DAEC beltline. Evaluation of these plates according to Reg. Guide 1.99, Revision 2, shows the USE at 32 effective full power years (EFPY) (approximately 40 years of operation) to be well above 50 ft-lb. The beltline shielded metal arc welds (SMAW) only had fabrication Charpy tests performed at 10°F, so USE data are not available. However, the surveillance weld, which is representative of the beltline welds, had an unirradiated USE of 101 ft-lb. Evaluation of the beltline welds in accordance with Reg. Guide 1.99, Revision 2, assuming initial USE of 86 ft-lb, results in USE predictions well above 50 ft-lb at 32 EFPY.

Question 2

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;

Request (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 tests;

Response (1)

For the beltline plate material, 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 GE analysis was used to develop updated pressure temperature operating limits for 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. 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 lack of dropweight testing. The unirradiated RT_{NDT} was determined from the procedure as the higher of either (T_{50T}-60°) or -50°F (NDT).

Charpy data, dropweight test results and estimated RT_{NDT} values for the beltline materials are shown in Table 2.

Request (2)

The heat treatment received by all beltline and surveillance materials;

Response (2)

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 are representative of the beltline materials, so heat treatment of the Charpy specimens should represent or bound that of the beltline materials.

After the beltline plates were quenched and tempered, specimen samples 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°, -50°) for 50 hours. The beltline material PWHT temperature was the same as for the surveillance specimens, but the beltline PWHT time was significantly less (typically 10 hours). The additional PWHT time for the specimens was intended to cover the possibility of future vessel repairs requiring subsequent PWHT. Since the specimen PWHT is longer than that of the beltline materials, the Charpy specimens and surveillance program materials provide a bounding representation of the vessel beltline materials.

Request (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;

Response (3)

The beltline consists of portions of the lower shell and lower-intermediate shell. Each shell is formed from two plates, so the beltline includes portions of four plates, four vertical welds and one circumferential weld. All beltline plate and weld materials were considered in the Appendix G evaluation. The material heat numbers are provided in Table

3.

ASME Section III Boiler and Pressure Vessel Code, Appendix G, addenda to and including winter 1985, requires testing beltline materials in accordance with Appendix H. This includes a Surveillance Program Meeting "the requirements of the edition of ASTM E185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased." For DAEC, this is E185-66 which required that "test specimens shall be taken from materials used in the irradiated region." E185-66 further states that "Samples shall represent one heat of the base metal and one butt weld if a weld occurs in the irradiated region."

Request (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;

Response (4)

The surveillance plate material was trimmed from the beltline plate with heat number B-0673-1. Specific wire heat number and flux lot data for the surveillance weld are not available. The specification for the surveillance weld required that it be made with the same procedure as the longitudinal beltline welds.

Since surveillance weld records are not available, Iowa Electric cannot prove that a beltline weld material was used. However, the data which is available on the surveillance weld indicates that it is representative as required by E185-66. More importantly, the available surveillance weld data provide Iowa Electric 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 not dependent on the availability of surveillance weld records, for the following reasons:

- o Archive surveillance weld material has been tested, providing complete baseline Charpy curve and chemical composition data.
- o 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 the necessary information is available to compare the surveillance weld irradiation embrittlement with Regulatory Guide 1.99, Revision 2 predictions.

Request (5)

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

Response (5)

Chemical composition weight percent data for beltline materials are shown in Table 3. This information was used with Regulatory Guide 1.99, Revision 2, to determine the limiting beltline material, the adjusted reference temperature versus EFPY for that material, and the predicted USE at 32 EFPY.

Chemical composition weight percent data for the surveillance plate and weld materials are shown in Table 4. The chemical composition data were used to compare measured Charpy curve shifts with Regulatory Guide 1.99, Revision 2, predictions.

Request (6)

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

Response (6)

See Table 3.

Question 3

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

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.

Response:

Operation with the DAEC 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 higher. The normal operating temperature in the beltline region is above 530°F. This conclusion is based on the fact that the temperature measured in the recirculation suction pipe is typically at or above 530°F and the recirculation suction pipe draws water from the beltline region.

Only during startup 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 region experience coolant temperature less than 525°F when the core is critical. The time of operation in these transient conditions has been estimated to be about 1%, and the associated temperatures for most of that time are 515°F or higher. The DAEC fluence during that time is estimated to be 3.6×10^{16} n/cm^2. This combination of low fluence and small deviation from the 525°F level is not expected to affect beltline RT_{NDT} or USE predictions significantly.

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.

The BWROG Supplemental Surveillance Program data will be collected periodically over the next 8 years. The capsules will include eutectic temperature monitors with which to determine the approximate maximum irradiation temperature. The data will include results of some PWR materials and Heavy Section Standard Test - 02 standard material. This data should provide additional insight into temperature differences between the BWR and PWR environments.

Question 3

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

Response:

Surveillance results were factored into beltline embrittlment predictions when Regulatory Guide 1.99, Revision 1 was used. However, Revision 2, paragraph C.2, requires credible data from two surveillance capsules before adjustments to the prediction methods are made, and only one capsule from DAEC has been tested. Therefore, the beltline predictions, based on Revision 2 methods, are without consideration of surveillance results.

Question 3

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.

Response:

Measured increases in reference temperature from the first surveillance capsule were within the mean-plus-2σ prediction for the weld plate.

Measured decreases in USE from the first surveillance capsule were within the prediction for the plate, and therefore, this request is not applicable to DAEC.

References

1) ASTM-E-185-66 Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels.

2) ASTM E-185-70 Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels.

3) ASTM E-185-73 Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels.

4) ASTM E-184-79 Standard Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials.

5) ASTM E-208-87 Stand and Test Method for conducting drop-weight test to determine Nil-Ductility Transition Temperatures of ferritic steels.

6) ASTM E-23-87 Method for Notched Bar Impact Testing of Metallic Materials.

7) ASTM E-8-87 Test Method of Tension Testing of Metallic Materials.

8) Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials.

9) "Reactor Vessel Material Surveillance Program Requirements," Appendix H to Part 50 of Title 10 of the Code of Federal Regulations, May 1983.

10) "Fracture Toughness Requirements," Appendix G to Part 50 of Title 10 of the Code of Federal Regulations, May 1983.

11) "Protection against non-ductile Failure," Appendix G to Section III of the ASME Boiler & Pressure Vessel Code Addenda to and including winter 1985.

12) "BWR Owner's Group Supplemental Surveillance Program" Phase II dated January 1992.

13) Iowa Electric Response to NRC Generic Letter 88-11, NG-89-0213, dated January 30, 1989.

(14) Safety Evaluation of the Duane Arnold Energy Center, Docket No. 50-331, dated January 23, 1973, Section 5.2.3.

Table 1

COMPARISON OF UNIRRADIATED AND IRRADIATED (5.9 EFPY)

CHARPY V-NOTCH DATA

Index Temperature

Material	E=30 ft-lb	E=50 ft-lb	MLE=35 mil	Upper Shelf Energy (ft-lb) L/T
Base:				
Unirradiated	-35°F	-22°F	-26°F	164/107
Irradiated	7°F	23°F	16°F	160/104
Difference	42°F	45°F	42°F	4/3(2.5%)
Weld:				
Unirradiated	-33°F	-11°F	-34°F	101/101
Irradiated	-33°F	-11°F	-34°F	101/101
Difference	0°F	0°F	0°F	0/0 (0%)
HAZ:				
Unirradiated	-6°F	34°F	-2°F	112/73
Irradiated	3°F	38°F	40°F	126/82
Difference	9°F	4°F	42°F	-14/-9 (-11.1%)

Table 2

MATERIAL PROPERTIES OF VESSEL COMPONENTS

Estimated Identification	Heat/Lot No.	Charpy Test Temp. (°F)	Energy (ft-lb)	Dropweight Temp. (°F)	RT_{NDT} (°F)
Lower Plates:					
(Shell 1) 1-18	C6439-2	40	36,48,43	40	40
1-19	B0402-1	40	83,85,72	40	40
Lower-Intermed. Plates:					
(Shell 2) 1-20	B0436-2	40	57,54,62	-30	10
1-21	B0673-1	40	99,104,121	-30	10
Longitudinal Welds	Ht. 432Z0471 Lot B003A27A	10	100,102,106	Not Available	-50
Girth Weld	Ht. 07L669 Lot K004A27A	10	50,50,54	Not Available	-50

Table 3

Chemical Composition of Beltline Plates
and Weld Filler Material
(weight in percent)

Identification	Heat/Lot No.	C	Mn	P	S	Si	Ni	Mo	Cu
Lower Plates Shell 1 Thickness: 5.03 inches	C6439-2	0.21	1.25	0.012	0.012	0.18	0.51	0.48	0.09
	B0402-1	0.20	1.35	0.012	0.015	0.16	0.47	0.45	0.13
Lower-Intermediate Plates Shell 2 Thickness: 4.47 inches	B0436-2	0.19	1.33	0.008	0.01	0.18	0.64	0.50	0.15
	B0673-1	0.20	1.37	0.011	0.014	0.18	0.61	0.55	0.15
Girth Weld	09L853 L017A27A	0.036	1.07	0.014	0.017	0.44	0.88	0.55	0.03
Girth Weld	07L669 K004A27A	0.05	1.24	0.014	0.016	0.48	1.02	0.54	0.03
Girth Weld	CTY538 A027A27A	0.066	1.06	0.020	0.018	0.46	0.83	0.49	0.03
Longitudinal Weld	432Z4521 B020A27A	0.06	1.20	0.018	0.017	0.42	0.98	0.54	0.01
Longitudinal Weld	432Z0471 B003A27A	0.077	0.92	0.017	0.019	0.33	0.91	0.52	0.03

Table 4

PLASMA EMMISSION SPECTROMETRY CHEMICAL ANALYSIS OF RPV
SURVEILLANCE PLATE AND WELD MATERIALS

Metal Element	Base Metal Tensile ETJ	Base Metal Tensile ETK	Weld Metal Tensile EU3	Weld Metal Tensile EU6
Mn	1.4	1.3	1.3	1.2
P	0.006	0.006	0.011	0.010
Cu	0.15	0.15	0.02	0.02
Si [a]	0.07	0.06	0.32	0.33
Ni	0.70	0.69	1.00	0.90
Mo	0.63	0.62	0.49	0.49
Cr	0.14	0.14	0.04	0.03
Co	0.014	0.013	0.013	0.012

Specimens are extra Charpy and Tensile specimens fabricated for the surveillance program from plate 1-21 (Ht. No. B0673-1).

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[a] Si results may be low, because of precipitation during dissolution heating.

Table 5

END-OF-LIFE UPPER SHELF ENERGY

Material	USE (ft-lb) $f = 0$ longitudinal	USE (ft-lb) $f = 4.9.10^{17}$* longitudinal	USE (ft-lb) for $f = 3.6x10^{18}$** (long./trans)
Plate 1-21	164 [a]	144	134/87
Plate 1-20	134 [a]	118	108/70
Weld	101 [a]	92	86

EOL Values of USE
* f value for DAEC
** f value from Reg. Guide 1.99 Rev. 2

a) These are measured values, all others are calculated.