NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

July 22, 1977



Mr D K Davis, Acting Chief
Operating Reactors Branch No. 2
Division of Operating Reactors
c/o Distribution Services Branch, DDC, ADM
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Davis:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Reactor Vessel Material Surveillance Program

Your letter dated May 20, 1977 requested information related to the Monticello reactor vessel and its associated fracture toughness surveillance program. We were asked to supply this information within 6 days of receipt of your letter.

Due to the complexity and volume of the material requested in your letter, we asked our reactor supplier to provide you with the required information. The attached report entitled "Reactor Vessel Material Surveillance Program," prepared by General Electric Company, provides their position with respect to each of the concerns of the NRC staff.

A comprehensive fracture toughness surveillance program is in effect at Monticello. This program conforms to ASTM E 185-66 and is described in General Electric Topical Report NEDO-10115, "Mechanical Property Surveillance of General Electric BWR Vessels," July, 1969. Fluence levels ( >1 Mev) based on analysis of the dosimenter removed from the reactor vessel during the first refueling outage are:

1) through 3/31/77 - 1.73 E17 $n/cm^2$ (+63%, -48%) at T/4 based on 4.51 EFPY

2) end of life - 1.23 E18 $n/cm^2$ (+63%, -48%) at T/4 based on 32 EFPY

Please contact us if you require additional information.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/DDM/ak



Attachment

cc: J G Keppler
G Charnoff
MPCA - Attn: J W Ferman

Attachment
NSP letter dated July 22, 1977
L O Mayer, NSP, to D K Davis, USNRC

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

Reference: Letter, D K Davis, NRC to L O Mayer, NSP, dated 5/20/77

The referenced letter has requested the Monticello Nuclear Generating Plant to provide a detailed list of materials relative to the reactor pressure vessel. The staff's concern is that the materials used in reactor vessel fabrication may have a wider variation in sensitivity to radiation damage than originally anticipated. In addition, some reactor vessels incorporate more than one heat of materials, including weld materials in their belt line region, but all of these heats may not be included in the reactor vessel material surveillance program. The purpose of this paper is to show that General Electric's program of reactor vessel surveillance is completely responsive to 10CFR50, Appendix H. Further, it will be shown that the effect on adjusted reference temperature for the most adverse materials in BWR/2 through BWR/4 plants irradiated to the maximum 40-year fluence observed is very small.

General Electric has addressed the problem of obtaining representative surveillance specimens since the beginning of its reactor pressure vessel surveillance program. The material for base metal specimens has been taken from a plate used in the vessel beltline region or from a plate of the same heat of material. The same plate used for base metal specimens is used for production of heat-affected zone specimens, and the weld specimens are produced by the identical weld practice and procedures used in the vessel fabrication. For vessels constructed from plate, the vessel longitudinal welds are represented; while for vessels fabricated from forged rings, the girth welds are represented. When widely varying weld practices such as submerged metal arc and electroslag welding are used jointly in a vessel, both are represented in the surveillance program material. Thus, the surveillance specimens do represent the materials and processing of the vessel beltline region.

The procedures described above were used to select surveillance materials and to prepare specimens for all operating BWR 2 through 4 plants. Examination of this method of selection, even in light of the most recent data, reveals that the reactor pressure vessel surveillance specimens currently in use still provide a reasonable representation of the limiting materials in the reactor vessel beltline region.

The production of the vessel beltline region is generally accomplished by the welding of several plates and, most often, several heats of steel are involved. The vessel surveillance specimens are produced from one of these heats. The possible variation of the other beltline heats, however, is limited by the characteristic range of compositions resulting from the material production practices. Consultation with the domestic heavy-section pressure vessel steel mill, Lukens Steel, concerning process capability and a survey of 10 BWR vessels reveals that the residual element of major importance, copper, lies consistently within the 0.15 to 0.20 weight percent range when special low-copper scrap selection procedures are not invoked on the mill process.

Examination of the predicted effect of residual element composition on the irradiation behavior of pressure vessel steels as provided in Regulatory Guide 1.99 and a preliminary analysis of GE data in the BWR fluence range from 10 operating BWR's representing copper contents in the range .01 to 0.30 weight percent and phosphorous contents in the range .007 to 0.02 weight percent reveals a minimal impact due to the possible variation in base metal composition that could be present in the vessel beltline. Data at the upper end of the copper range (0.30%) was obtained from an atypical source. It represents a foreign plant with a forged ring produced by foreign practice. It does, however, provide additional support for predicting the maximum effect of elevated copper contents.

For all operating BWR 2 through 4 vessels, with one exception, the predicted end of 40-year life fluence at the vessel wall 1/4T location is below $2 \times 10^{18}$ nvt ($>1$ MeV). For this fluence range, an estimated end of life variance of approximately 15°F in transition temperature shift would be indicated for a copper composition range of 0.15 to 0.20 weight percent copper. This variance represents the expected deviation in predicted transition temperature shift due to compositional differences. That is, at the end of life fluence, the predicted shift in transition temperature could vary by 15°F depending on the composition of the heat of plate material in question. Thus, even with the maximum predicted variability of copper content for the beltline plate material, a minimal variation in predicted transition temperature shift is expected.

For the one plant with a predicted 1/4T fluence value of $3 \times 10^{18}$ nvt ($>1$ MeV) at the end of life, the effect of the maximum expected variation of copper content would be approximately a 30°F variation in predicted transition temperature shift. This variation, while larger than that expected for all other operating BWR/2 through 4 plants is not prohibitively large, particularly since it represents the worst case of surveillance specimens with 0.15% Cu while other heats in the beltline contain 0.20% copper.

Similarly, the variability of weld metal properties within the beltline region does not present a major obstacle to their effective representation by the current surveillance specimens. Typically, the range of residual element compositions present in weld metal falls within several major bands determined by weld process, electrode coating, and flux type. This variability inherent to process characteristic is already taken into account by the fact that the identical weld process and procedures used in vessel manufacture are used to produce the surveillance weld specimens. The copper content range resulting strictly from heat to heat variations of filler metal composition within a given process, however, would still require the surveillance specimens to adequately represent a limited range of weld metal composition which could be present in the vessel beltline region when more than one heat of filler metal was used for fabrication of this region.

A survey of weld practices used in 10 BWR pressure vessels has characterized the ranges of copper contents expected for the weld metal in the vessel beltline. Once again, when compared in the fluence region of the BWR based on the predictions of Regulatory Guide 1.99 and a preliminary analysis of extensive GE data, the copper variations within a given process contribute only a minimal estimated variance in the predicted transition temperature shift. For standard submerged metal arc and electroslag welds a range of 0.15 to 0.20 weight percent copper resulting in approximately a 15°F variation in transition temperature shift is expected at the end of a 40-year vessel life. For shielded metal arc welds a copper content less than 0.15 weight percent and an estimated end of 40-year life variation of 5° to 10° F in predicted transition temperature shift is expected, and for the extreme case of submerged metal arc welds made with copper coated electrodes used for circumferential welds in 6 BWR's a range of 0.2 to 0.4 weight percent copper resulting in a projected end of life variability of approximately 25°F in transition temperature shift would be expected. The analysis of the effect of elevated copper in these welds produced with copper-coated electrodes is based on a maximum predicted fluence of $6 \times 10^{17}$ nvt (>1 MeV) at the 1/4T wall location for the six plants affected. Once again, the one operating BWR plant with a predicted end of life fluence of $3 \times 10^{18}$ at the 1/4T location would show a slightly larger effect than the other plants. For this plant, the normal copper content range for submerged metal arc and electroslag welds would result in approximately a 30°F variation in transition shift response, while the copper content variation in the shielded metal arc welds would cause approximately a 20°F variation in the predicted transition temperature shift. No welds were made from copper coated electrodes for this reactor pressure vessel.

Based on the preceding discussion, the selection of materials for the reactor pressure vessel surveillance programs in BWR 2, 3 and 4's does reasonably represent the materials in the beltline region of the vessel. The steps taken by General Electric to assure adequate representation of the welds process and all subsequent material processing steps seen by the vessel materials limits the only possible variation between surveillance specimens and vessel material to the heat to heat variability of base metal and weld metal. The net, end of 40-year life effect of these possible variations, is projected to be only a 10° to 25°F variability in the predicted transition temperature shift for the BWR fluence range.

Although it is still important to know the residual element composition of the vessel steel and surveillance specimens for complete analysis of surveillance test results, this information can easily be obtained by chemical analysis of archive material and analysis of specimens at the time of testing. General Electric believes that the steps taken during the production of BWR pressure vessel surveillance specimens adequately assure reasonable representation of the vessel material and that any variations in irradiation behavior between the surveillance materials and additional heats of vessel materials would be minimal in the BWR fluence range.

NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER: 50-263

FILE NUMBER:

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TO: Mr. D. K. Davis

FROM: Northern States Power Company, Minneapolis, Minnesota, L. O. Mayer

DATE OF DOCUMENT: 7/22/77

DATE RECEIVED: 7/25/77

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ENCLOSURE

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM.....

PLANT NAME: Monticello (1-P) (3-P)

ACKNOWLEDGED

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RJL 7/26/77

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