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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ATYPICAL WELD METAL

Introduction

During 1978, B&W initiated work contracted with the B&W Owners Group on a program for evaluating the material properties of "early vintage" 177-fuel assembly reactor vessel welds. One of the work phases in this program had the objective of characterizing the chemistry of reactor vessel (RV) beltline welds. Extensive chemical analyses of the archive sources of RV welds have been performed as part of this work. Two samples of test weldments made for the Crystal River 3 reactor vessel surveillance program were part of the weld metal archives subjected to chemical analysis. The results of these analyses, performed by the Mt. Vernon Works Quality Assurance Laboratory, indicated that one of these samples had atypical concentrations of nickel and silicone, while the concentrations of the other elements were in the normal range for MnMoNi:Linde 80 submerged-arc RV weldments. The other sample had the nominal chemistry.

The atypical weld was made with weld wire designated by heat number 72105. This heat of weld wire was used in the fabrication of 12 reactor vessels. These vessels and the location of possible atypical welds are listed in Table 1.

Charpy V-notch tests on the atypical weld metal resulted in a higher than normal value of RT_{NDT}, partially because of unusually high scatter. Therefore, we requested that the licensees of the above plants administratively apply revised pressure-temperature operating limits that reflected the possible presence of atypical weld metal. In calculating these limits the atypical weld was assumed to have an unirradiated RT_{NDT} of 120°F and radiation damage is predicted by the upper limit line in Regulatory Guide 1.99. Currently all the affected plants are operating under such revised limits.

Discussion

10 CFR Part 50, Appendix G "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The pressure-temperature limits depend upon the metallurgical properties of the reactor vessel materials. The properties of materials in the vessel beltline region vary over the lifetime of the vessel because of the effects of neutron irradiation. One principle effect of the neutron irradiation is that it causes the vessel material nil-ductility temperature (RT_{NDT}) to increase with time. The pressure-temperature operating limits must be modified periodically to account for this radiation induced increase in RT_{NDT} by increasing the temperature required for a given pressure. The operating limits for a particular operating period are based on the material properties at the end of the operating period. By periodically revising the pressure-temperature limits to account for radiation damage, the stresses and stress intensities in the reactor vessel are maintained within acceptable limits.

The magnitude of the shift in RT_{NDT} is proportional to the neutron fluence that the materials are subjected to. The shift in RT_{NDT} can be predicted from Regulatory Guide 1.99. To check the validity of the predicted shift in RT_{NDT}, a reactor vessel material surveillance program is required. Surveillance specimens are periodically removed from the vessel and tested. The results of these tests are compared to the predicted shifts in RT_{NDT}, and the pressure-temperature operating limits are revised accordingly.

Since the unirradiated RT_{NDT} of the atypical weld metal was determined to be high, and it was assumed to be sensitive to radiation damage, the atypical weld metal would generally be the limiting vessel material. Therefore, all licensees with vessels that might have been fabricated with atypical weld metal were required to revise their pressure-temperature operating limits to reflect the possibility that atypical material was used in their construction.

Evaluation

To resolve the atypical weld issue, B&W has conducted an extensive investigation of records, metallographic examinations, chemical analyses, and fracture mechanics tests on both unirradiated and irradiated atypical weld material. The results of this program are presented in BAW-1556.

Since 1966, 42 heats of submerged-arc weld wire have been purchased for RV and surveillance specimen fabrication at Mt. Vernon, and, except for the discovery of the partial-thickness off-chemistry conditions in the second CR-3 surveillance block, there is no evidence that atypical weld wire reached the shop floor. The results of more than 2000 chemical analyses have been reviewed relative to the 42 wire heats. All, except for the one batch of Crystal River survillance material, have been within the normal ranges. These include through-thickness tests from seven RVs fabricated at Mt. Vernon and tests of wire currently in inventory.

Detailed metallographic examinations were performed on seven fractured Charpy specimens. Both macro- and micro-examination techniques were employed, as well as a fractographic examination with a scanning electron microscope. Relatively little porosity was noted in any of the weld material examined. Examinations revealed columnar grains outlined by proeutectoid ferrite. The orientation of the grains and the unusually high amount of proeutectoid ferrite are believed to be the cause of the high scatter in the Charpy data.

Numerous chemical analyses were performed on the atypical weldment. The bulk of these analyses were obtained using a Jarrel-Ash emission spectrometer. The concentrations of 10 elements were measured by this technique. X-ray flourescense analysis was used to measure the concentrations of nickel, molybdenum, and copper in irradiated Charpy specimens. Results show that the copper content was high, averaging between 0.4 to 0.5%. The chemistry of atypical material is compared to typical material in Table 2.

Charpy V-notch tests were performed on both unirradiated and irradiated material. The irradiated specimens were irradiated in the Crystal River 3 reactor vessel. Dynamic and static fracture toughness tests were conducted on one inch thick compact tension specimens at room temperature. Although the dropweight NDT is -20°F, the results of the Charpy tests show that 50 ft-lbs of energy is absorbed at 150°F, therefore the unirradiated value of RT_{NDT} is 90°F. Using RT_{NDT} equal to 90°F, the toughness properties obtained from the fracture mechanics tests, K_{Ic} (static) and K_{Id} (dynamic), are conservative (lie above) the K_{Ic} curve in ASME Code, Section XI and the K_{IR} curve in ASME Code, Section III respectively. Using an RT_{NDT} of -20°F (the dropweight NDT), the fracture mechanics data fall within the scatter of data on normal material used to obtain the K_{Ic} and K_{IR} curves. This indicates that the RT_{NDT} value of 90°F is conservative.

The effect of irradiation on the mechanical properties of atypical material have been evaluated, using the test results on Crystal River 3 surveillance specimens. These specimens were subjected to a fluence of 1.1×10^{18} n/cm^2. This fluence produced an increase in RT_{NDT} of 35°F.

From our review we conclude that the probability that atypical weld metal was used in fabricating the subject vessels is very low. However, we feel that in calculating pressure-temperature operating limits for these vessels, the properties of atypical material should be considered. As discussed above, we have determined that an initial value of RT_{NDT} of 90°F is a very conservative value. The increase in RT_{NDT} due to irradiation should be based on the measured value of 35°F at a fluence of 1.1×10^{18} n/cm^2 and the damage prediction slopes in Regulatory Guide 1.99.

We also find that the administratively applied pressure-temperature operating limits may be removed from these 12 plants. The operating limits in the Technical Specifications for Browns Ferry 1 are presently being reviewed and will include limits based on the atypical weld metal. The Technical Specifications for Rancho Seco have operating limits based on the atypical material that are more restrictive than they would be if based on the criteria developed from this review. Midland 1 is being reviewed for an Operating License and its Technical Specifications have not been finalized to date. The pressure-temperature limits for Three Mile Island 2 should be revised to reflect the possible use of atypical material in this vessel. This poses no immediate problem since this plant is not currently operational. The Technical Specifications of the other nine subject plants contain pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50 based on both typical and atypical weld metal properties.

The staff will continue to monitor the effects of radiation on the properties of the atypical weld material. Six capsules containing the atypical weld metal are in the Crystal River 3 surveillance program. One of these capsules has already been removed and tested. Also, there is enough atypical material in storage at B&W to fabricate fracture toughness specimens up to 1.0T compact fracture toughness specimen size.

TABLE 1. LOCATION OF POSSIBLE ATYPICAL WELDS

PLANT	LOCATION OF WELD
B&W	
Oconee 3	Center Circ. Beltline
TMI 1	Upper Circ. Beltline
	Lower Circ. Beltline
TMI 2	Dutchman to Lowerhead
ANO 1	Head to Flange and Nozzle to Shell
Midland 1	Center Circ. Beltline
CR-3	Center Circ. Beltline
Rancho Seco	Vertical Seam Beltline
WESTINGHOUSE	
Zion 1	Inter to Lower Circ. Beltline
Zion 2	Vertical Seam Beltline (o and 180^{o})
Turkey Pt. 4	Nozzle Shell to Interm. Circ.
GE	
Br. Ferry 1	Shell to Flange and Longitudinal Weld in Beltline
Quad Cities 2	Closure Head to Flange

TABLE 2. ATYPICAL WELD CHEMISTRY

	C	Mn	P	S	Si	Cr	Ni	Mo
CR-3 Weld	.08	1.65	.021	.013	1.0	.07	.10	.45
Mn-Mo-Ni (Typical)	.08	1.6	.018	.015	.5	.07	.60	.40