Correlation of Pressurized Water Reactor Vessel Material Properties Variation with Neutron Fluence by Surveillance Program

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The surveillance capsules included in the reactor vessel surveillance program of Nuclear Power Plants in Taiwan had been removed from the reactor and shipped to the hot laboratory in the Institute of Nuclear Energy Research (INER) for examination after being irradiated for scheduled periods. The objective of the surveillance program is to correlate the mechanical properties of surveillance specimens with accumulated neutron fluence the specimens experienced during irradiation, and then to evaluate the adequacy of fracture toughness of reactor vessel materials for continued operation. Based on ASTM Standard E185-82, Charpy V-notch impact testing and tensile testing were the main technologies used to measure the mechanical properties of the surveillance test materials. There were also neutron radiometric monitors, including iron wire, nickel wire, copper wire, uranium-238 in U_3O_8 powder and neptunium-237 in NpO_2 powder, encapsulated in the capsule for evaluating the neutron fluence by a combination of activities measurement of neutron radiometric monitors and computer code analysis according to Regulatory Guide 1.190. By means of Regulatory Guide 1.99, Revision 2, the results of mechanical properties measurement and neutron fluence analysis were then employed to evaluate the extent of radiation embrittlement of reactor vessel material specimens. The pressure-temperature limit curves for reactor operation and the PTS (Pressurized Thermal Shock) analysis can also be deduced from the correlation, evidencing that the reactor pressure vessel will be kept being bounded by the requirements of 10 CFR APPENDIX G during its life time.

Hot Laboratory Examinations

***Mechanical Testing*.** The post-irradiation mechanical testing, including Charpy V-notch and tensile tests, was performed in accordance with 10 CFR 50, Appendix H and ASTM Standard E185-82. Three kinds of surveillance test materials were tested: weld, heat affected zone and base materials. As per the specimen orientation and location specified in ASTM E185-82, tension and Charpy specimens were machined from the quarter-thickness (1/4T) locations of the representing surveillance test material plate.

- *Charpy V-notch Impact Test.* The Charpy V-notch impact tests were conducted in accordance with ASTM E23-07a with a Tinius Olsen Impact Tester. The maximum impact capacity of the Tinius Olsen Tester set up in the lead cell of INER is 359 Joules. This tester is operated remotely and the specimens can be loaded in automatically, making it possible to implement an impact test within five seconds upon the specimen being removed from the thermostatic sample room. Gas is used as the media for cooling or heating specimens during the test. Test specimens were kept at the test temperature for 40 minutes before testing in accordance with ASTM E23-07a. In accordance with ASTM E2298, the hammer tip is instrumented with a strain gauge system, which yields dynamic impact energy and additional characteristic values in addition to the force-time curve.

- *Tensile Test***.** According to ASTM E8/E8M-09 and E21-09, the tensile tests were conducted. The tensile tester can exert a maximum axial tensile force (load) of 100kN (22000 lb). The tester is also set up in the lead cell, operated remotely. Besides, the tester is equipped with an electric resistance heater. All of the tensile tests were performed under a constant cross head speed of 0.01cm/min(0.005in/min).

Neutron Radiation and Dosimetry Analysis**.** Neutron fluence rate calculation was performed and interactively compared in two parts: neutron transport analysis and passive radiometric monitor measurement. Computational codes of ANISN and DORT were used to calculate the discrete ordinates transport of neutron (Rhoades & Childs, 1998). Iron, nickel and copper wire together with oxide powder of uranium-238 and neptunium-237 were used as the dosimeter to evaluate the representative neutron flux for neutron energy > 1 MeV. SAND-II (McElroy et al., 1967) offered a method to perform least squares analysis technique for the neutron spectrum unfolding, leading to a neutron fluence rate spectrum closer to reality.

- *ANISN and DORT codes*. In the neutron fluence rate analysis within the reactor geometry, both 1-dimention and 2-dimention discrete ordinates transport computer codes played an important role in the calculations. In the 1-dimention analysis, ANISN supported the calculation of the neutron fluence rate in a cylindrical geometry. On the other hand, DORT was used to construct a 2-dimensional model, including R-Θ geometry and R-Z geometry, calculating the neutron fluence rate distribution for different azimuthal angle and height.
- *SAND-II code*. SAND-II provides an adjustment method by means of combining the results of neutron transport calculations with neutron dosimetry measurements in order to obtain an optimal estimate for neutron spectrum with assigned deviation. All the quantities input into SAND-II are simultaneously and iteratively adjusted to give an output spectrum with the minimum weighted least squares error.

References

McElroy, W. N., Berg, S., Crockett, T., & Hawkins, R. G. (1967). A computer-automated iterative method for neutron flux spectra determination by foil activation. AFWL-TR-67-41, 1.

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