Proposed - For Interim Use and Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

BRANCH TECHNICAL POSITION 6-1

pH FOR EMERGENCY COOLING WATER FOR INTEGRAL PRESSURIZED WATER REACTORS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of component integrity issues related to engineered safety features

Secondary - None

A. BACKGROUND

The emergency cooling water for the mPower™ integral pressurized water weactor (iPWR) is stored in the Refueling Water Storage Tanks (RWSTs) and is periodically circulated through the reactor coolant inventory and purification system (RCIPS) for purposes of demineralization and pH adjustment. pH changes during extended period between RCIPS purification may be monitored by sampling utilizing a sampling mechanism and schedule specified by the applicant and licensee.

To establish the minimum value of pH in postaccident containment flood in iPWRs, the U.S. Nuclear Regulatory Commission staff has reviewed the available information and recommended the criteria listed in the branch technical position below.

The minimum pH value of 7.0 follows from the Westinghouse report conclusion that, in Emergency Core Cooling System (ECCS) solutions adjusted with NaOH to pH 7.0[*] or greater, no cracking should be observed at chloride concentrations up to 1000 parts per million (ppm) during the time of interest. Figure 7 of the Westinghouse report shows that the time for initiation of cracking of sensitized and nonsensitized U-bend specimens of Type 304 austenitic stainless steel in solutions of 7.0 pH having 100 ppm chloride was 7-1/2 months and 10 months, respectively.

The great majority of tests reported in the Oak Ridge report were performed with pH of 4.5, and only two tests were conducted with pH values other than 4.5. Some cracking was observed at pH 7.5 in the sensitized 304 stainless steel U-bend specimens after 2 months exposure to pH 7.5 and chloride concentration of 200 ppm. All of the 316 stainless steel specimens showed no evidence of cracking. Considering the fact that in U-bend specimens the material was sensitized, stressed beyond yield, and plastically deformed, we conclude that the reported test conditions were much more severe than the stress conditions likely to exist in the postaccident emergency coolant systems.

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[*]All pH values are at 25ºC.

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We agree with the Oak Ridge conclusion that absolute freedom from failure of any complex system such as a reactor cavity flood system can never be guaranteed, but, by proper design, fabrication, and control of the corrosive environment, the probability of failure can be significantly reduced. Our recommended minimum pH is somewhat higher than the Oak Ridge recommendation of a minimum of 6.5.

B. BRANCH TECHNICAL POSITION

The criteria for pH level of postaccident emergency coolant water to reduce the probability of stress-corrosion cracking of austenitic stainless steel components, nonsensitized or sensitized, nonstressed or stressed, are as follows:

1. Minimum pH should be 7.0.

2. For the reactor cavity flood water released into the containment sump, the higher the pH in the 7.0 to 9.5 range, the greater the assurance that no stress corrosion cracking will occur.

3. If a pH greater than 7.5 is used, consideration should be given to the hydrogen generation problem from corrosion of aluminum in the containment.

C. EVALUATION FINDINGS

The controls on the pH and chemistry of the reactor ECCS solutions meet the staff positions on postaccident chemistry requirements for iPWR emergency coolant water. It also meets the requirements of General Design Criterion 14 for assuring the low probability of abnormal leakage or failure of the reactor coolant pressure boundary and safety-related structures. We conclude that the proposed pH for emergency coolant water is acceptable.

D. REFERENCES

1. D.D. Whyte and L.F. Picone, Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Environment, WCAP-7798-L, Westinghouse Nuclear Energy Systems, (NES Proprietary Class 2).

2. J.C. Griess and E. E. Creek, Design Considerations of Reactor Containment Spray Systems - Part X, The Stress Corrosion Cracking of Types 304 and 316 Stainless Steel in Boric Acid Solutions, ORNL-TM-2412, Part X, Oak Ridge National Laboratory.