Enclosure 2

SAFETY EVALUATION
FOR
RV CLOSURE HEAD PENETRATION TUBE
ID WELD OVERLAY REPAIR
A WESTINGHOUSE OWNERS GROUP PROGRAM REPORT
(MUHP-5017)
WCAP-14519

INTRODUCTION

In December 1991, cracks were found in an Alloy 600 vessel head penetration (VHP) in the reactor head at a French plant. Examinations in pressurized water reactors (PWRs) in France, Belgium, Switzerland, Sweden, Spain, and Japan have uncovered additional VHPs with axial cracks present. About 2% of the VHPs examined to date contain short, axial cracks. Close examination of the VHP that leaked at the France plant revealed incipient secondary circumferential cracking of the VHP.

An action plan was implemented by the NRC staff to address primary water stress corrosion cracking (PWSCC) of Alloy 600 VHPs at all U.S. PWRs. As explained more fully below, this action plan included a review of safety assessments by owners groups, the development of VHP mock-ups by the Electric Power Research Institute (EPRI), the qualification of inspectors on the VHP mock-ups by EPRI, the review of proposed generic acceptance criteria from the Nuclear Utility Management and Resource Council (NUMARC) (now the Nuclear Energy Institute, NEI), and VHP inspections. As part of this action plan, the NRC staff met with the Westinghouse Owners Group (WOG) on January 7, 1992, the Combustion Engineering Owners Group (CEOG) on March 25, 1992, and the Babcock & Wilcox Owners Group (B&WOG) on May 12, 1992, to discuss their respective programs for investigating PWSCC of Alloy 600 and to assess the possibility of cracking of VHPs in their respective plants since all of the plants have Alloy 600 VHPs. Subsequently, the staff asked the NEI to coordinate future industry actions because the issue was applicable to all PWRs. Meetings were held with NEI and PWR owners on the issue on August 18 and November 20, 1992, and March 3, 1993. In addition, EPRI is engaging in ongoing research on methods for PWSCC mitigation. EPRI also developed a qualification program to ensure that inspections performed on VHPs are highly reliable in detecting and measuring flaws. The qualification program includes standard, mock-up VHPs containing known flaws that are axial, circumferential, off-axis, and clustered (closely spaced) flaws. The inspector is required to identify the location, orientation, and depth of all of the flaws in the EPRI mock-up VHPs to become a qualified inspector. The NRC has been following this program and has reviewed the qualification results for all of the inspectors that have been qualified by EPRI.

The NRC staff met with the B&WOG, CEOG, and the WOG to discuss the PWSCC of PWR VHPs on several occasions during 1992 and 1993. Each of the owners groups submitted a safety assessment through NEI to the NRC on this issue and the NRC submitted a safety evaluation of the safety assessments to NUMARC on November 16, 1993. After reviewing the industry's safety assessments and examining the overseas inspection findings, the staff concluded in the safety evaluation sent to NEI that VHP cracking is not a significant safety issue at this time. The bases for this conclusion are that if PWSCC occurred at VHPs: 1) the cracks would predominately be axial in orientation; 2) the cracks would result in detectable leakage before catastrophic failure; and 3) the leakage would be detected during visual examinations performed as part of surveillance walkdowns before significant damage would occur to the reactor vessel head. In addition, the staff had concerns related to unnecessary occupational radiation exposures associated with eddy current or other forms of nondestructive examinations if done manually. Field experience in foreign countries has shown that occupational radiation exposures could be significantly reduced if the industry would use remotely controlled or automatic equipment to conduct the inspections. The U.S. nuclear industry has developed such equipment for inspection and possible repairs.

As a follow-up to the safety assessments, NEI submitted proposed generic acceptance criteria for flaws identified during inservice examinations of VHPs to the NRC in July of 1993. The NRC accepted the acceptance criteria for axial flaws above and below the J-groove weld (the weld that holds VHP to the vessel head and is part of the primary pressure boundary), and circumferential flaws below the J-groove weld, but rejected the criteria for circumferential flaws above the J-groove weld. Cracks below the J-groove weld do not violate the reactor vessel pressure boundary even if they are through wall, and axial and circumferential cracks below the J-groove weld were determined to be acceptable by the NRC staff. Axial cracks above the J-groove weld may result in a leak that would be detected by surveillance walkdowns before significant damage could occur. Circumferential cracks above the J-groove weld may increase the potential for ejection of a control rod drive mechanism resulting in a large break loss-of-coolant accident. Furthermore, the stress analyses conducted as part of the owners groups' safety assessments predicted that it would be unlikely that circumferential cracks would form due to the stress distributions in the VHPs. For these reasons, the NRC requested that circumferential crack-like indications above the J-groove weld be reported to the NRC for disposition.

Three licensees volunteered to conduct VHP inspections during 1994 as part of the NEI program. The eddy current inspection conducted by the Wisconsin Electric Power Company vendor (Westinghouse) at the Point Beach Nuclear Generating Station in April 1994 uncovered no crack-like indications in any of the 49 VHPs. The eddy current inspection by the Duke Power Company vendor (Babcock & Wilcox) at the Oconee Nuclear Generating Station in October and November 1994, revealed 20 crack-like indications in one penetration. Ultrasonic testing (UT) could not quantify the depth of these indications because they were shallow. (UT cannot accurately size defects that are less than one mil deep (0.03 mm).) These indications may be associated with the original fabrication and may not grow. Even if they do grow, the analysis conducted on the indications by the licensee indicates that they will not grow such that they exceed the acceptance criteria before the next outage. During the next outage, the affected VHP will be reexamined and analyzed to see if the indications will exceed the acceptance criteria before the next outage. This cycle of reexaminations will continue until no growth occurs for two cycles, or until the indications are projected to exceed the acceptance criteria before the next inspection cycle. In the latter case, the VHP will be repaired or replaced. An examination of the VHPs by the Indiana & Michigan Electric Company vendor (Westinghouse) at D.C. Cook revealed three clustered crack-like indications in one penetration. The indications were 46 mm, 16 mm, and 6-8 mm in length and the deepest flaw was 6.8 mm deep. The tip of the 46-mm flaw was just below the J-groove weld. The acceptance criteria permits a through-wall, axial crack of any length below the J-groove weld since such a crack does not violate the primary pressure boundary and do not present a safety concern. An analysis by the D.C. Cook licensee indicates that these flaws will not grow to exceed the acceptance criteria before the next outage when a reinspection will occur. During the next outage, the affected VHP will be reexamined and analyzed to see if the indications will exceed the acceptance criteria before the next outage. This cycle of reexaminations will continue until no growth occurs for two cycles, or until the indications are projected to exceed the acceptance criteria before the next inspection cycle. In the latter case, the VHP will be repaired or replaced. These results are consistent with the owners groups' analyses, the NRC staff safety evaluations, and PWSCC found in the CRDMs in European reactors. The results observed during the these three VHP inspections do not pose a threat to safe plant operation.

Based on the owners groups' safety assessments, the staff concluded that a leak in a VHP would be detected before significant damage could occur to the VHP or the reactor vessel. This would result in the deposition of boric acid crystals on the vessel head and surrounding area that would be detected during surveillance walkdowns.

ALTERNATIVE VHP REPAIR TECHNIQUE

Westinghouse submitted a Topical Report, "RV Closure Head Penetration Tube ID Weld Overlay Repair" to the NRC Staff in November, 1995. Virginia Electric and Power Company, North Anna Power Station Unit 1, submitted a request on November 22, 1995, to the NRC staff requesting approval to use the Topical Report method as an alternative repair technique for reactor vessel head penetrations.

The objective of the program described in the Westinghouse Topical Report was to provide a weld design package that could be applied to repair inside diameter (ID) initiated PWSCC in VHPs. The weld design package was designed to repair partial through-wall and full through-wall cracks. Westinghouse investigated 1) excavation geometries and depths related to flaw geometry, 2) limitations of the weld repair as a function of crack length, 3) definition of welding process parameters, 4) establishment of allowable weld filler metals, 5) establishment of weld surface finish, 6) establishment of surface profile of the penetration ID, 7) use of post weld surface treatments, and 8) post weld inspection requirements.

Westinghouse completed a generic safety evaluation for the weld repair package. Westinghouse demonstrated the weld repair package using a full size penetration mock-up that provided engineering data to enable evaluation of effects of the weld overlay process on penetration residual stress and deformation of the penetration. The extent of weld shrinkage and impacts on the shrink fit between the VHP and the reactor vessel head were evaluated in this evaluation.

Preparation of Penetration Tube Samples

Grooves were machined into 10 inch long penetration tube samples that were representative of Westinghouse VHPs using electric discharge machining (EDM). Calculations by Westinghouse for a typical 4-loop plant indicate that approximately 0.29 inches of wall thickness is needed to meet design pressure requirements. Grooves were machined to various depths with taper at each end of the grooves. Both axial and circumferential grooves were prepared. Axial grooves were 3 or 5 inches long. Circumferential grooves were 45, 90, and 360°.

Fabrication of Reactor Vessel Closure Head/Penetration Tube Mock-up

Full scale reactor vessel closure head and penetration tube mock-ups were prepared that represented the most limiting peripheral location in a 4-loop plant. This location has previously been established as the penetration having the highest residual stress of any of the penetrations. The ovality that occurred as a result of performing the mock-up attachment welding was measured and compared to the ovality measured from actual plant ovality measurements. The ovality in the mock-up was comparable to the observed ovality in actual plants.

Weld Overlay Process Specification

Automated pulsed gas tungsten arc welding (GTAW) was selected for this program. The equipment can weld inside diameters as small as 2 inches. A demonstration weld overlay was completed on a 2 inch diameter nickel base alloy using Inconel filler metal. The GTAW machine uses 0.030 inch diameter filler wire, and the Inconel Alloy filler wire of choice was not available for the demonstration. A spool of 0.035 inch diameter filler wire was located and was reduced to 0.030 inch diameter wire. The weld parameters were adjusted during the welding of the first pipe assembly for the new weld wire. After the first two assemblies were completed, tensile and bend specimens were machined from each assembly in accordance with ASME Section XI requirements. All of the bend specimens were free of cracks except the root bend specimen from the first assembly. This cracking was attributed to the use of improper weld parameters on the first assembly. The shielding gas was adjusted in an attempt to improve the finished surface finish. The use of a modified shielding gas did not change the surface and the original shielding gas used for subsequent test tube samples.

Welding of Penetration Tube Samples

Eight penetration tube samples were prepared that contained axial EDM grooves and 45, 90 or 360 degree circumferential EDM grooves. One sample contained an axial EDM notch and a 90 degree circumferential notch and one sample contained one 360 degree circumferential notch and two axial notches. The NRC staff finds that the selection of samples is a reasonable representation of the conditions that may require repair.

Evaluation of Welded Penetration Tube Samples

The overall weld length, circumferential extent and depth were varied. After welding, the OD dimensions were measured to determine the amount of deformation that resulted from welding. The measurements indicated that the shrinkage was small. The NRC staff finds that the amount of shrinkage measured appears representative of what may be expected in the conduct of an actual repair.

Residual Stress Measurement on Reactor Vessel Head/Penetration Tube Mock-up

Residual stresses were measured using the hole drilling method in the as-fabricated condition, after excavating the penetration tube, and after completion of local repair welding. The major difference between the hole drilling method and the finite element method is that the hole drilling method measures the average stress over the depth of the hole while the finite element method measures the surface stress.

When the results of the stress measurements is compared to the stress results obtained using finite element analysis, the stress distributions had the same general shape, but the measured stresses were smaller than the calculated stresses. Westinghouse explained the difference in stress values based on the premise that the measured stresses are average stresses and the calculated stresses are surface. There was better agreement between the measured and calculated hoop stresses than for the corresponding axial stresses. Westinghouse made several observations during the preparation of the mock-ups. The machining of the grooves generally appeared to reduce the stresses. The repair areas have relatively high residual stresses, but these residual stresses are comparable with the calculated surface stresses. Tensile stresses adjacent to the weld are relatively high but dissipate quickly in areas away from the weld. The NRC staff believes that it would be prudent to estimate the life of the weld repair as is required as a part of the repair plan in Section XI of the ASME Code. This is not required for the short term, but, rather should be a long term action by Westinghouse. The staff considers that accelerated corrosion testing needs to be performed as a part of the life determination.

Post Weld Surface Treatment

Post weld surface treatments were attempted to improve the surface finish, restore acceptable geometry, or reduce residual stresses in the weld metal and adjacent base material. These initial attempts were unsuscessful. The improvement of surface finish and restoration of acceptable geometry are to be controlled by the welding process and by inspection. Shot peening was examined as the post weld surface treatment to reduce the residual stresses. Westinghouse concluded that shot peening could be used to reduce the residual stresses in the weld and in the adjacent heat affected zone. The NRC staff agrees that shot peening may be beneficial, however, the NRC staff believes it would be prudent to estimate the life of a shot peened sample as a long term action.

ASME Code Approach to Weld Repair

Section XI of the ASME Code establishes the guidelines for repair welding. A Section XI repair would require removal of the crack prior to repair welding. If the flaw is not completely removed, the repair is not in accordance with the requirements of Section XI and, therefore, would not meet the requirements of 10 CFR 50.55a. To satisfy NRC requirements, the licensee must submit an alternative repair method to the NRC for approval. The WOG is considering submitting a code case to ASME covering this weld overlay repair method. If the ASME code case is approved and the NRC endorses the code case in Regulatory Guide 1.147, then no submittal to the NRC staff would be required to use this repair method.

Post weld inspection of pressure boundary repair welds is covered in ASME Code Section XI, Subsection IWA-4500. A baseline volumetric inspection is made of the repair weld for future reference.

Staff Evaluation of the Embedded Flaw Weld Repair

The staff finds the proposed weld repair to be an acceptable alternative even though the flaw may not be completely removed. The embedded flaw repair will provide sufficient wall thickness to the VHP such that leakage will not occur through the VHP wall during normal operation. Code Case N-504-1, Alternative Rules Repair of Class 1, 2, and 3 Austenitic Stainless Steel," has been approved by the NRC staff for the use of weld overlays over embedded flaws in stainless steel piping. The proposed weld repair follows the guidelines in Code Case N-504-1 including the excavation of the flaw, the weld overlay, and the post weld repair non-destructive evaluation.

Conclusions

Westinghouse has concluded that an acceptable weld overlay process has been developed and qualified to Section XI of the ASME Code. The NRC staff concludes that the weld parameters and processes have been qualified to Section XI. The NRC staff finds that the Westinghouse overlay process provides an acceptable alternative to the ASME Code in this Topical Report since it provides sufficient wall thickness such that leakage will not occur through the VHP wall during normal operation. Licensees may reference the Topical Report and this safety evaluation when requesting permission to use the alternative to the Code.

The NRC staff considers that Westinghouse needs to estimate the lifetime of the weld repairs, with and without post weld surface treatment.