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Reactor pressure vessel annealing demonstration evaluation

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ABSTRACT

An independent evaluation has been made of a reactor pressure vessel (RPV) annealing demonstration project (ADP). Successful completion of the ADP has demonstrated that functional requirements for in-place annealing (IPA) of a U.S. RPV can be met using existing equipment and procedures. Leakage air flow between the RPV and the reflective insulation appears to have contributed to the generation of higher-than-anticipated axial temperature gradients in the region of the RPV above the annealing zone.

1. INTRODUCTION

Welding materials used in the construction of some of the older RPVs in the USA contain elements which accelerate the irradiation embrittlement process. Surveillance data for these RPVs have shown that permissible limits for irradiation-embrittlement of the RPV materials may be reached before the end of the licensed operating period for the nuclear plants. In-place annealing (IPA) of these RPVs provides a means of restoring the ductility and fracture toughness of the irradiation-embrittled material and, thereby, extending the service life of the nuclear plants. The U.S. Department of Energy (DOE), in conjunction with industry partners, has sponsored an annealing demonstration project to develop and validate technology for the IPA of U.S. RPVs. Sandia National Laboratories (SNL) managed the ADPs for DOE by providing technical and project review oversight of the projects, including on-site oversight at Marble Hill. The Oak Ridge National Laboratory (ORNL) is performing an independent annealing demonstration evaluation (ADE) of the ADP for the U.S. Nuclear Regulatory Commission (NRC).

The focus of the ADE is on engineering issues associated with IPA of RPVs. These issues arise from, (a) functional requirements which dictate that the RPV temperatures during IPA be much higher than normal operating temperatures, and (b) practical constraints on heating of the reactor system. These constraints cause the temperature differences and gradients generated in the RPV and the heat transport system during IPA to be much higher than they would be during normal operation. Increased RPV temperatures are of concern because of the potential for, (a) overheating the concrete adjacent to the RPV supports, and (b) causing temper embrittlement of the RPV material. Increased temperature differences and temperature gradients are of concern because of their potential to generate high stresses and permanent deformations in the RPV and the adjacent heat transport system piping.

The program plan for the ADE is shown in Fig. 1. Objectives are to develop and validate instrumentation requirements and analysis methods for demonstrating compliance with acceptance criteria for IPA of RPVs. Functional requirements for the ADP are defined in terms of the temperature-time history which must be achieved in all of the highly irradiated RPV beltline material to effectively restore the ductility and fracture toughness properties of that material. Codes, standards, and regulatory guides for in-place annealing of RPVs [1-4] define acceptance criteria which assure that IPA stresses and temperatures in the RPV and the adjacent reactor system structures and components do not exceed values which could compromise the future safe operation of the plant. Thermal analysis, stress analysis, and instrumentation evaluation, are the primary elements of the ADE. Analysis results are used to verify compliance with the acceptance criteria. Comparison of analytical predictions for temperatures, strains, and displacements, with values obtained from the instrumentation during the ADP provides the input required for refinement and validation of the analytical models. The ADE program will provide NRC with validated instrumentation requirements and analysis methods for assessing the acceptability of future licensee applications for in-place annealing of RPVs.

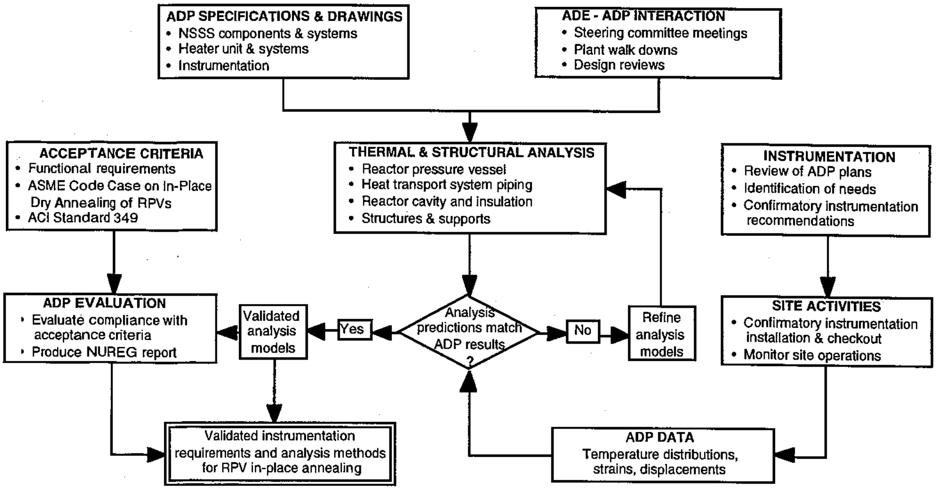

Fig. 1. Program plan for the annealing demonstration evaluation.

2. PRIOR IPA EXPERIENCE AND ITS RELEVANCE TO RPVs IN U.S. NUCLEAR PLANTS

IPA of RPVs has been performed at a number of commercial nuclear plants in Russia. These plants all have VVER 400 Nuclear Steam Supply Systems (NSSS). Important differences exist between the RPVs of VVER 400 plants and the aging RPVs of concern in the USA. The VVER 400 RPVs were constructed using ring forgings joined by circumferential welds. The U.S. RPVs of concern were constructed from formed plates joined by both longitudinal and circumferential welds. Life-limiting material is generally contained in the RPV welds. The annealing zone must cover all areas of the RPV where irradiation-embrittled weld material is located. In the VVER 400 RPV, only one circumferential weld is located in the highly irradiated belt-line region. The annealing zone in a VVER 400 RPV can, therefore, be confined to a relatively short axial segment of the RPV surrounding the circumferential weld. The upper boundary of this annealing zone is far removed from the RPV nozzles. The annealing zone in a U.S. formed-plate RPV, however, must extend over the full length of the irradiation-embrittled axial welds. The upper boundary of the annealing zone in the U.S. RPVs is, therefore, much closer to the

RPV nozzles than was the case for the VVER 400 RPVs. The effect of IPA on interaction between the RPV and the adjacent components and structures is, therefore, accentuated in the case of the U.S. RPVs.

3. ADP TEAM AND PLANT

The annealing demonstration was performed at the Marble Hill (MH) plant in southern Indiana. This plant was abandoned after being partially completed. The Marble Hill plant is owned by Public Service of Indiana. The American Society of Mechanical Engineers (ASME) Center for Research and Technology Development (CRTD) is the lead contractor for the MH ADP. Westinghouse Electric Corp., Cooperheat Inc., and Parsons Engineering Corp., are the subcontractors to ASME-CRTD for this ADP. The general arrangement of the Marble Hill nuclear plant in the in-place annealing configuration is given in Fig. 2. Transfer of heat from the heater to the RPV is primarily by radiation. Thermocouples (T/Cs) welded to the inner surface of the RPV and to the heater unit provide feedback for control of the hot gas generator unit. Reflective insulation, typical of a production installation, was in place outside most of the RPV. Prototypical insulation was added where the reflective insulation was incomplete. The reactor cavity cooling air flow was restored. Resistance temperature detectors, strain gauges, and displacement transducers, were positioned on the outside of the RPV, and on the heat transport system (HTS) components and the support structures. Items of special interest for this NSSS configuration include, (a) concrete temperatures adjacent to the RPV nozzle-support embedments, (b) axial temperature gradients in the area immediately above the heater, and (c) stresses induced in the RPV and the nozzle safe-ends caused by the combined effect of temperature gradients and NSSS piping loads.

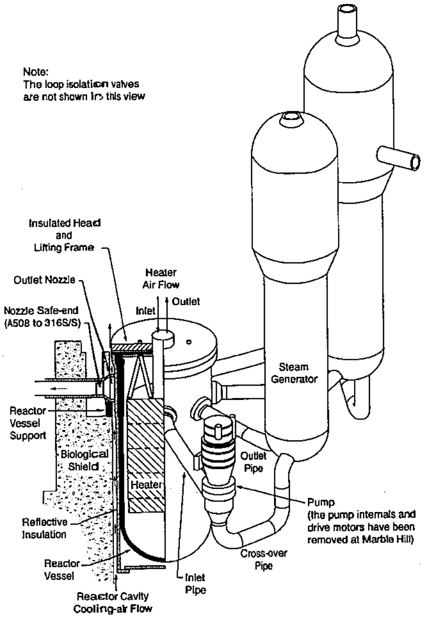

Fig. 2. General arrangement of Marble Hill Nuclear Plant in the in-place annealing configuration.

4. FUNCTIONAL REQUIREMENTS AND ACCEPTANCE CRITERIA

The target temperature-time history for the annealing zone of the RPV is defined by, (a) a lower-bound temperature-time history required for the annealing operation to be effective, and (b) an upper-bound temperature-time limit set to avoid the adverse effects of temper embrittlement and creep of the RPV material. Prior research has shown substantial recovery of preirradiation mechanical properties when irradiation-embrittled RPV material is annealed for 1 week (168 hrs) at 454°C (850°F) [5,6]. NRC-sponsored programs have shown that no significant temper embrittlement effects will occur in either plate or weld material if the temperature of the RPV material does not exceed 482°C (900°F) [7] for periods of up to 300 hrs. The primary functional requirement set for the IPA demonstration was to maintain a temperature (T) of the inner surface of RPV material in the annealing zone in the range 441°C ≤ T ≤ 468°C (825°F ≤ T ≤ 875°F) for 168 hrs.

Acceptance criteria are defined for both the metallic and concrete portions of the NSSS and the associated support structures. Maximum values of linearized membrane plus bending stress intensities induced in the RPV are limited to the ASME Code $3S_m$ values [1]. An additional local membrane stress intensity limit is applied in the nozzle crotch region of the RPV [1,8]. In the IPA application, the $3S_m$ stress intensity limit serves to assure that any inelastic strains resulting from the IPA operation will be small and localized. Any permanent deformation of the RPV resulting from the IPA will, therefore, be sufficiently small that subsequent operation of the RPV and its associated components will not be compromised. The $3S_m$ stress limit also serves to assure that any IPA-induced ductile-tearing extension of any preexisting flaws in the RPV is reduced to a minimum. Stress intensity limits defined in the original construction code are used for the balance of the NSSS components. Outside of the RPV, the stresses of primary concern are those induced in the heat transport system (HTS) piping by the restraint of free thermal expansion. $3S_m$ is also the acceptance limit for linearized stresses in the piping [9]. The maximum allowable concrete temperature is set at the 177°C (350°F) limit defined for short-period operation in reference 3.

5. THERMAL ANALYSIS

The MH nozzle-supported RPV has an inside diameter of 4.39 m (173 in.) and an approximate outside diameter of 4.83 m (190 in.). The annular gap between the RPV and the reinforced concrete RPV cavity shield wall is nominally 190 mm (7.5 in.). A 76 mm (3 in.) layer of mirror thermal insulation surrounds the RPV. This insulation is located between the RPV and the cavity shield wall. The mirror insulation protects the RPV cavity wall from elevated temperature exposure. At room temperature there is a nominal 19 mm (0.75 in.) air gap between the mirror insulation and the RPV outer surface. It was assumed in the thermal analysis that there was no leakage of air between the cavity air flow and the entrapped air between the mirror insulation and the RPV outer surface. The entrapped air could communicate with the cavity cooling air only at the RPV flange.

The heater was constructed of stainless steel and designed to maintain the inside diameter of the RPV annealing zone at a temperature of 455°C ± 14°C (850°F ± 25°F). The heater heated the entrapped air and the RPV by radiation and convection. The RPV in turn heated the insulation and the air entrapped between the RPV and the insulation. The primary heat sink was the air blown through the RPV cavity between the cavity wall and the RPV insulation. There was forced convective cooling in the cavity and natural convective heating/cooling in the entrapped air spaces.

Air flow in the reactor cavity is the ultimate heat sink. Calculations performed with the cavity air flow set at 50% and 100% of the rated flow showed RPV temperatures to be relatively insensitive to this parameter. Thermal/hydraulic analyses were performed with the cavity air flow set at 50% of the rated blower capacity. With this cavity air flow, the predicted maximum cavity concrete surface temperature did not exceed 52°C (125.3°F) during the steady-state soak period of the annealing cycle. An additional conservative boundary condition assumption used in the analyses was that an adiabatic surface existed at a depth of 30.5 cm (12 in.) into the concrete. Only concrete inboard of that surface was included in the analysis model.

Thermal analyses of the RPV system were performed using the FLUENT [10] computer program. The FLUENT computational fluid dynamics program has comprehensive heat transfer and gas dynamics simulation capabilities. The axisymmetric analysis model used in the FLUENT analysis of temperatures in the RPV and the surrounding structures is shown in Fig. 3. In the nozzle region, the geometry is not symmetric, but it was assumed that the reactor coolant system (RCS) piping and RPV nozzles could be evaluated decoupled from the RPV thermal/hydraulic model. Transient heat transfer analyses were run for the heat-up, steady-state soak, and the cooldown phases of the IPA cycle. Analysis predictions for the axial distribution of temperature on the inner and outer surfaces of the RPV wall during the steady-state soak portion of the IPA cycle are shown in Fig. 4. The plots of Fig. 4 show that predicted temperatures in the annealing zone are in compliance with the IPA functional requirements 441°C ≤ T ≤ 468°C [(825°F ≤ T ≤ 875°F)]. The steep axial temperature gradient predicted for the region above the center line of the nozzles is a primary cause of stresses in the RPV and the piping. The RPV at the MH plant is nozzle supported.

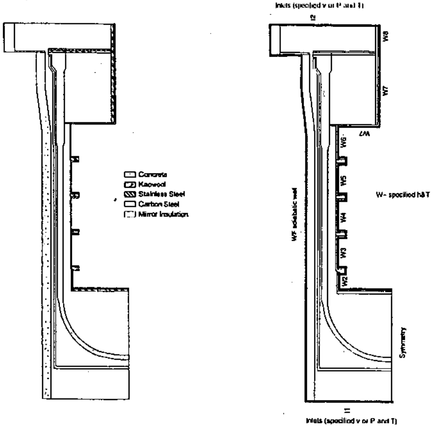

Fig. 3. Axisymmetric model used in the FLUENT analysis of temperatures in the MH RPV and the surrounding structures.

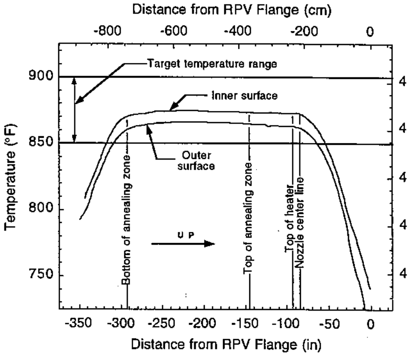

Fig. 4. Predicted axial temperature distributions in the MH RPV during the steady-state soak portion of the IPA cycle.

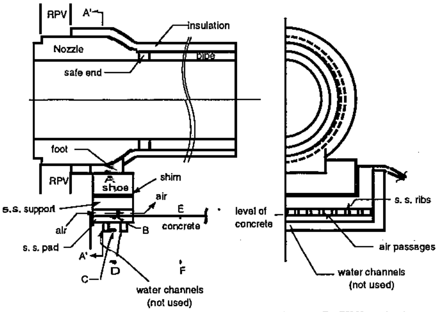

Fig. 5. Geometry of the MH RPV support.

Figure 5 shows a cross-section of an outlet nozzle and the support components. A foot, which is an integral part of the nozzle forging, sits on a carbon steel shoe - which in turn - sits on a stainless steel weldment that is bolted into the concrete shield wall. Shims between the shoe and support are employed to level the vessel. The stainless steel support has provision for passive air cooling. The RPV inlet and outlet nozzles were plugged with a device which acted as a convection and thermal barrier.

Thermal analyses of the RPV nozzles and the support structures were performed using the HEATING computer program. HEATING is a general purpose conduction heat-transfer program for steady state and/or transient analysis of cylindrical and spherical structures [11]. Three-dimensional HEATING models, approximating the RPV nozzle geometry, were developed using r-theta-z cylindrical coordinates with the z-axis being in the direction of the pipe centerline. Thermal analyses were performed for the steady-state IPA condition. Boundary conditions for the nozzle/support analysis were taken from the FLUENT analysis of the RPV. The sink temperature for this analysis was set as the temperature of the cooling air between the RPV insulation and the cavity wall at the level of the nozzles. Since the air cooling channels are essentially passive devices, an attempt was made to bracket the actual flow conditions in and around the support structure. Analyses were performed with, (a) an air velocity of 1 m/s (3.28 ft/s) in the air passages, and (b) no air flow in the passages.

Maximum temperatures in the concrete immediately below the support structure (point C in Fig. 5) were predicted as 120°C (248°F) and 187°C (368°F) for cases (a) and (b) respectively. The predicted case (b) temperature is marginally higher than the concrete temperature limit of 177°C (350°F) set in the design criteria for short-term operation. The substantial decrease in this temperature produced by a modest flow velocity in the cooling channels, however, suggests that the maximum concrete temperature generated during an IPA would probably be acceptable.

6. STRUCTURAL ANALYSIS

A 3-D finite-element model of the MH RPV was generated for use with the ABAQUS computer program [12] using a modified version of the ORNOZL automatic mesh generator program [13]. Advantage was taken of the approximate 90° symmetry of the RPV to limit the scope of the model to a 1/4-segment of the RPV extending from the upper flange down to the bottom of the lower head. The 3-D RPV model is shown in Fig. 6. The 1/4-segment model includes the cold- and hot- leg nozzles from a single HTS loop. A support foot is attached to the bottom of the cold-leg nozzle. Material properties for A533B plate material and A508 Cl 2 forging material were used for the RPV shell and the nozzle forgings, respectively. The nozzle models include representation of the safe ends, which are fabricated from Type 316 stainless steel (S/S). Attached to the nozzle safe ends are models of segments of S/S HTS piping. Restraint loads from the HTS piping model were applied to the pipe segments, remote from the junction between the S/S safe end and the ferritic nozzle forgings. This feature of the model made possible an accurate evaluation of the superposition of stresses from nozzle- forging/safe-end differential expansion and stresses from the HTS piping loads. A 3-D finite-element model of the HTS was generated using the quadratic elbow element employed in the ABAQUS program. The model uses elbow elements to represent the hot- and cold-leg, and the crossover piping between the steam generator and coolant pump. Displacements of the steam generator and pump interfaces were restrained, but all other displacements were permitted. Anchor motions were applied at the piping interface with the RPV nozzle safe ends.

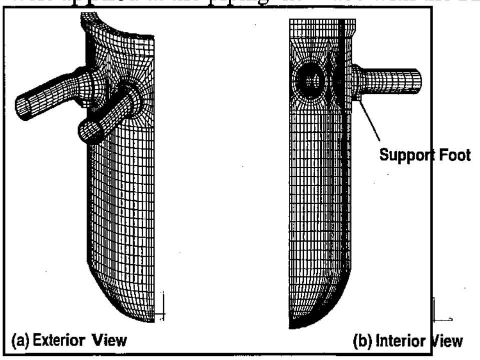

Fig. 6. 3-D finite-element model of MH RPV

Structural analyses of the MH RPV and HTS piping were performed in three steps. In Step 1, temperature distributions from the FLUENT thermal analyses were applied to the 3-D RPV model and anchor motions at the nozzle safe ends were calculated. These anchor motions were then applied to the HTS piping model and the distribution of loads and moments within the HTS piping was determined. In the final step, loads and moments were taken from the pipe model at locations corresponding with the ends of the pipe segments in the RPV model. These loads and moments were then applied to the RPV model, together with temperatures from the FLUENT thermal analysis. Stresses and deflections, for comparison with the acceptance criteria, were taken from the final analysis with combined thermal mechanical loading.

Reserve factor (RF) is defined as the ratio of the allowable stress intensity to the predicted stress intensity. The minimum acceptable RF value is 1.0. A minimum RF of 1.68 was calculated for the junction between the nozzle forging and the stainless steel safe-end. It is evident from the magnitude of this RF that sufficient margin exists to accommodate significant departures of the actual IPA boundary conditions from the boundary conditions assumed in the analysis. The location of the minimum RF in the RPV safe-end weld is shown in Fig. 7.

7. INSTRUMENTATION

Objectives defined for the instrumentation installed as part of the MH ADP were as follows:

1. Measure sufficient temperatures to verify the heat transfer analytical model.
2. Measure displacements at selected locations to confirm the structural model.
3. Measure temperatures at selected concrete and nearby structure locations to determine maximum values.
4. Measure temperatures at selected equipment locations to determine maximum values.
5. Measure strains when temperature or displacement data are not sufficient to confirm the structural model, or to address model uncertainties at critical locations.

Confirmatory instrumentation consisting of temperature, strain, and displacement measurement devices was also installed at the Marble Hill plant as part of the ADE Program. The instrumentation provided a comprehensive capability for monitoring the thermal and structural response of the RPV, the HTS components, and the concrete structures, to the IPA thermal cycle. A dimensional survey of sealing surfaces and keyways in the RPV, before and after the IPA, provided information on the dimensional stability of the RPV.

8. OBSERVATIONS OF THE ANNEALING DEMONSTRATION

The only significant departures from predictions observed in the ADP were, (a) higher-than-predicted axial temperature gradients in the region of the RPV above the annealing zone, and (b) circumferential variations in the temperature of the cavity cooling-air flowing through the annular gap between the RPV flange and the reactor cavity wall. Temperatures in the region of the RPV above the annealing zone were significantly lower than had been predicted. Plots of the predicted and observed temperature distributions in this region are given in Fig. 8.

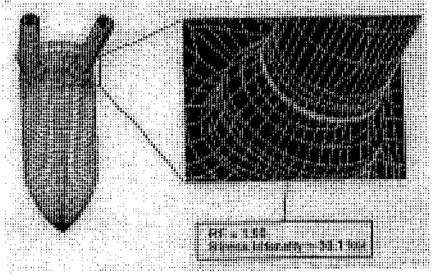

Fig. 7. Stresses at all points within the Marble Hill model are below the $3 \bullet S_m$ limit defined by the ASME Code Case (RF = 1.68 in the hot-leg nozzle weld for steady-state conditions).

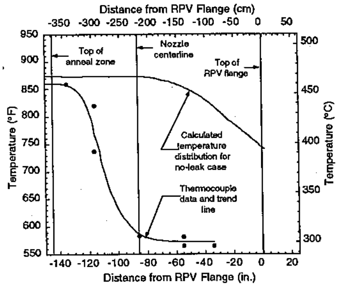

Fig. 8. Measured temperature gradients in the MH RPV in the region immediately above the annealing zone were significantly higher than predicted.

Temperature gradients in the region immediately above the annealing zone are significantly higher than the predicted gradients used in the stress and deformation analysis of the RPV. The impact of this observation on stresses and deformations in the RPV and the HTS piping has yet to be evaluated.

The observation of circumferential variations in the temperature of the air flowing from the reactor cavity indicates that some of this air may have leaked into the annulus between the RPV and the mirror insulation. Leakage air flow in this annulus would act to cool regions of the RPV above the heater. This cooling could account for some of the observed discrepancy between predicted and measured temperatures in this region.

9. INTERIM CONCLUSIONS

Successful completion of the ADP at the abandoned Marble Hill nuclear plant has demonstrated that all of the functional requirements for IPA of a U.S. RPV can be met using existing equipment and procedures. The ADP and ADE instrumentation worked well and provided adequate feedback of data for both control of the heater unit and tracking of the response of the RPV and adjacent components and structures to the IPA annealing cycle. Temperature distributions predicted by the MH ADE analysis model of the RPV were close to those observed during the IPA, except in the region immediately above the annealing zone. Leakage air flow between the RPV and the reflective insulation appears to have contributed to the generation of higher-than-anticipated axial temperature gradients in the region above the annealing zone. Stresses calculated for the Marble Hill RPV, using the original predictions for IPA temperature distributions, were significantly less than the allowable values. Measured temperatures in the concrete, adjacent to the RPV supports, were lower than the predicted values and were acceptable.

REFERENCES

1. ASME Code Case N-557, Dry Annealing of a Nuclear Reactor Vessel, March 1996.
2. USNRC Regulatory Guide 1.162, Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels, February 1996.
3. ANSI/ACI 349-90, Code Requirements for Nuclear Safety Related Concrete Structures, Appendix A - Thermal Considerations, ACI Manual of Concrete Practice, Part 4, page 349-79, 1995.
4. USNRC Regulatory Guide 1.142, Safety Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments), Rev. 1, October 1981.
5. T. R. Mager, Feasibility of and Methodology for Thermal Annealing an Embrittled Reactor Vessel, EPRI NP-2712, Vol. 2, Electric Power Research Institute, Palo Alto, California, November 1982.
6. T. R. Mager et al., Thermal Annealing of an Embrittled Reactor Vessel, Feasibility and Methodology, EPRI NP-6113-SD, Electric Power Research Institute, Palo Alto, California, January 1989.
7. E. D. Eason et al., Models for Embrittlement Recovery Due to Annealing of Reactor Pressure Vessel Steels, NUREG/CR-6327, February 1995.
8. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components, Subparagraph NB-3227.5, Nozzle Piping Transition, 1995.
9. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components, Subarticle NB-3600, Piping Design, Subparagraph NB-3653.1, Satisfaction of Primary Plus Secondary Stress Intensity Range, 1995.
10. FLUENT Version 4.4.1, Computational Fluid Dynamics Code, Fluent Inc., New Hampshire, 1996.
11. W. Childs, Oak Ridge National Laboratory, HEATING 7.2 Users Manual, ORNL/TM-12262, February 1993.
12. ABAQUS Theory Manual, Version 5.4, Hibbit, Karlson, and Sorensen, Inc., Providence, Rhode Island, 1995.
13. J. A. Keeney-Walker and B. R. Bass, "ORNOZL: A Finite-Element Mesh Generator for Nozzle-Cylinder Intersections Containing Inner-Corner Cracks," Oak Ridge National Laboratory, NUREG/CR-5872 (ORNL/TM-11049) September 1992.