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Elevated-Temperature Structural Design Evaluation Issues in LMFBR Licensing

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Abstract

This paper describes elevated-temperature structural design issues and concerns identified by NRC licensing review of the Clinch River Breeder Reactor Plant for a construction permit. Major issues concern weldment evaluation, notch weakening, steam generator tubesheet evaluation, and the use of "average" rather than "minimum" material properties in inelastic analysis. All of the issues were resolved but several required CRBRP Project commitment to performance of additional confirmatory programs.

1. Introduction

The Clinch River Breeder Reactor Plant was designed to demonstrate that a liquid metal fast breeder reactor can operate safely and reliably in an electric utility system. The plant is a 350 MWe, three loop design and was to be located in the Tennessee Valley Authority system at a site on the Clinch River near Oak Ridge, Tennessee. With a reactor vessel outlet temperature of 995°F (535°C) it was necessary in the structural design of the plant to take account of the thermal loading conditions, time-dependent response of material, and failure modes unique to elevated-temperature service. Although the CRBRP project has been terminated, the design was essentially completed and the evaluation procedures thoroughly reviewed by the U.S. Nuclear Regulatory Commission relative to issuance of a construction permit.

Elevated-temperature CRBRP systems and components were designed to meet the limits of the ASME Boiler and Pressure Vessel Code, Section III, Case N-47 [1] which applies for ferritic steels at temperatures above 700°F (371°C) and for austenitic stainless steels above 800°F (427°C). Case N-47 includes strain and deformation limits that require evaluation of time-dependent, inelastic response to steady and cyclic loads. The Nuclear Regulatory Commission identified a number of concerns with both the design limits and the evaluation procedures, and identified specific development programs to be completed prior to issuance of a plant operating license. The findings should be of interest to LMFBR design efforts ongoing in other countries and to advanced and low cost plant designs being pursued in the United States.

2. NRC Licensing Concerns

In the Structural Evaluation Report [2] related to construction of the CRBRP, the Nuclear Regulatory Commission described the complicating effects of elevated-temperature

"Systems and components in service at elevated temperatures are subjected to larger temperature variations and differentials than LWR hardware. Moreover, the materials have lower strength at elevated temperatures. The resulting higher thermal strain ranges and increased inelastic strain concentrations tend to accelerate fatigue damage. In addition, the materials are susceptible to creep-rupture damage that results from both applied and residual stresses persisting after transient conditions. Relaxation of such stresses tends to cause ratcheting on subsequent load cycles. The effective microscopic ductility of many of the materials and product forms is reduced by concentration of creep strains in grain boundaries. Consequently, cracking can occur at accumulated strain levels that would cause no problems at temperatures below the creep regime."

Based on a review of the material presented by the CRBRP Project, the Nuclear Regulatory Commission identified concerns in nine areas: (1) Weldment Safety Evaluation, (2) Elevated-Temperature Seismic Effects, (3) Design Analysis Methods, Codes and Standards, (4) Elastic Follow-up in Elevated Temperature Piping, (5) Notch Weakening, (6) Creep-Fatigue Evaluation, (7) Plastic Strain Concentration Factors, (8) Intermediate Heat Transport System Transition Weld, and (9) Steam Generator.

Ultimately, all of these concerns were resolved. However, for four of the issues resolution was based on Project agreement to perform additional confirmatory programs. These issues are identified and discussed here.

3. Weldment Safety Evaluation

Early weldment cracking, particularly in components subjected to repeated thermal transient loadings was identified by NRC as the foremost structural integrity concern. It is well known that when structural failure occurs it is generally at weldments.

The Code Case N-47 approach to weldment integrity is primarily to assure that weldments are at least as strong as the parent metal. Weldment configuration and processes are controlled, and the amount of delta ferrite which may transform to a brittle sigma phase is limited. In addition, reduced strain limits are specified which encourage the placement of weldments in lower stressed regions. Case N-47 specifies the use of parent metal properties to represent weldment behavior in life assessment calculations, so the complex interaction between stress and strain at weldments is not taken into consideration. Although an experimentally based procedure that accounts for reduction in creep rupture strength of weldments has been developed, it has not yet been adopted by the Code.

The NRC position is that, because of the importance of weldment cracking as a failure mode, the designer should have a better understanding of the metallurgical interactions that take place in weldments and their effects on weldment life. Specifically, the NRC is concerned with (1) early crack initiation at the inside wall surface in the heat-affected zone (HAZ) where the weldment is exposed to thermal cycling, (2) the effects of large variations of material properties within the weldment on creep-fatigue and creep-rupture damage, and (3) the effects of time rate, cyclic rate, and hold time on the propagation of long shallow cracks in the HAZ of a weldment. They are also concerned about creep enhancement of crack growth in a cracked weldment; specifically, enhanced creep in the remaining uncracked wall caused by residual stress and thermal cycling, and effects of creep on stability of the remaining uncracked wall ligament. The NRC feels that as a minimum these effects must be considered and quantitatively evaluated in order to determine the safety margins of weldments in elevated-temperature components.

The Nuclear Regulatory Commission and the CRBRP Project developed jointly a confirmatory program to address these issues. The program includes both experimental and theoretical investigations but does not require component testing. The overall objectives are to confirm structural adequacy of the CRBRP design with regard to weldment integrity, and to quantify safety margins of weldments in service at elevated temperatures.

The basic elements of the confirmatory program are summarized here:

- Evaluate potential for premature crack initiation at weldments due to thermal fatigue, residual stresses, and damage caused by welding.
- Confirm adequacy of creep-rupture and creep-fatigue damage evaluation procedures at weldment.
- Assess growth behavior of cracks in the heat affected zone of weldments.
- Evaluate consequences of enhanced creep in uncracked ligaments.
- Assess stability of uncracked ligaments for creep conditions.
- Define effects of long-term elevated-temperature service on crack initiation.
- Evaluate effects of loading sequence on creep-fatigue behavior.

4. <u>Notch Weakening</u>

Cracking at notches and local structural discontinuities is another area of major concern to the NRC. The geometrical configurations lead to local stress concentration and the potential for inelastic strain concentration that may exhaust material ductility. Notches, small radius fillets, and localized structural discontinuities are regions observed in practice, besides weldments, where cracks tend to initiate.

The standard design approach is to avoid use of sharp geometrical discontinuities, and to place structural transitions in low stress regions.

There are no special rules in Code Case N-47 that apply to notches. They must be considered in application of Appendix T, Section T-1300, Deformation and Strain Limits for Structural Integrity; and in Section T-1400, Creep-Fatigue Evaluation. Both of these sections have two sets of requirements, one for elastic analysis and one for inelastic analysis. When the elastic limits cannot be met, it is necessary to do inelastic analysis or redesign the component. Most regions of the CRBRP with significant structural discontinuities were modeled inelastically. In fact, most of the inelastic analysis performed for the CRBRP was to assure compliance with Case N-47, Appendix T rules at structural discontinuities.

The major concern of the NRC is that the design limits for fatigue and creep rupture are based on tests of smooth sided specimens that do not include possible effects of stress gradients in notches. They are also concerned about loss of ductility under long term loadings due to prior cyclic and monotonic straining. The Nuclear Regulatory Commission concerns are described as follows in the Safety Evaluation Report:

"The basic allowable stress limits of the Code are based on unnotched creep specimen test data. Stress raisers influence the creep behavior of the entire wall in two basic ways. They introduce a constraint against inelastic flow by inhibiting slip line development. This is manifested in a reduction in the average stress intensity in the net section (a notch strengthening effect). Stress raisers also introduce a site where creep-rupture damage could cause early crack initiation and more rapid crack propagation (a notch weakening effect). Although the combined effect is notch strengthening in most cases, an evaluation is needed to determine what geometric, loading, and material parameters could cause significant notch weakening, particularly for long-term loading at elevated temperatures. Loading conditions such as transverse shear do not introduce any notch strengthening and have contributed to weldment cracking at structural discontinuities."

A procedure was developed and proposed to the ASME Code cognizant group in 1980 which to a degree addresses these concerns. The proposed optional rules include the effects of stress state on creep rupture strength using a simplified analysis procedure. Elastic analysis is used to calculate average effective stress on the notched cross section and a maximum stress triaxiality factor. These are related to the conventionally determined time-dependent strength of the material based on creep rupture test results on a variety of notched geometries. The procedure includes the effect of stress distribution in the notch as desired by the NRC but does not include the effects of cyclic and bending loads, nor of long-term ductility exhaustion.

A confirmatory program was developed jointly between the NRC and the CRBRP Project to address these concerns. The basic elements of the program are summarized as follows:

- Extend elastic constraint damage evaluation method to include cyclic and bending loads.
- Implement effects of material ductility in damage evaluation procedure.
- Apply the extended method to "worst case" geometric notches in CRBRP components.
- Compare effects of tensile stress vs. stress intensity on creep rupture.
- Develop cyclic creep strain concentration factors for notches in creep fatigue and perform trial applications.

5. Design Analysis Methods, Codes and Standards

Issues in this category were identified by NRC review of the applicable design criteria for elevated-temperature service. These include the ASME Boiler and Pressure Vessel Code Case 1592 and its current version Case N-47; Reg. Guide 1.87, "Guidance for Construction of Class 1 Components in Elevated Temperature Reactors"; and NE Standards F9-4T, "Requirements," and F9-5T, "Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components."

Concerns were expressed in four areas: (1) impact of new technology developments on safety, (2) verification of computer programs for inelastic design analysis, (3) use of alternative strain limits in NE Standard F9-5T, and (4) suitability of using material average properties for design inelastic analysis. The first three issues were resolved readily by Project commitments to: (1) keep abreast of new design technology and assess CRBRP safety implications, (2) provide formal verification of computer programs, and

(3) not use the alternate strain limits based on elastic analysis in design justification.

The issue concerning material property representation for inelastic analysis was more difficult to resolve and resulted in a confirmatory program to be completed by the CRBRP Project prior to application for an Operating License. The Project design approach was based on an interpretation that, except for buckling limits, the effects of material variations from "average" are covered by Code design margins. Therefore, it is appropriate to use average (or representative) plasticity and thermal creep properties in inelastic analysis. This interpretation was established and recently reaffirmed by a strong consensus of Code committee members.

The NRC concern is that creep-rupture damage calculated using average properties may be too low when compared with the considerable strain and cyclic hardening that occurs during fabrication and operation, and that the fatigue damage and accumulated strains may be too low if the actual yield strength is below the average value used in design analysis.

The confirmatory program identified to resolve these concerns requires an evaluation of the significance of material property variations where inelastic analysis is used to evaluate elevated-temperature components containing radioactive sodium. The following requirements were imposed:

- Minimum yield strength and minimum creep strength (80% of the average isochronous curves) properties shall be used to evaluate fatigue damage, as represented by the use fraction, and the accumulated inelastic strains.
- The fatigue damage fractions and the creep-rupture damage, represented by the time fraction, are to be reported to the NRC for both minimum and average material properties using the method of Case N-47.
- The creep portions of the total accumulated inelastic strains (membrane, bending, and peak) are to be reported using the method of Case N-47.
- Structural adequacy of the components shall be demonstrated using these calculated values of damage and inelastic strain.
- Minimum and average properties shall be considered in performing the other confirmatory programs on Weldment Safety Evaluation, Notch Weakening, and the Steam Generator.

6. Steam Generator

The major NRC concern relative to the Steam Generator is assurance of adequacy of the tubesheet for intended life. Tubesheets are complex three-dimensional structures that are difficult to analyze. Section III of the ASME Code provides a simplified method of analysis based on the equivalent solid plate concept. However, this method is not applicable for the CRBRP steam generator tubesheet where the loading is dominated by large thermal gradients, and deformations are inelastic.

During thermal transients the temperature of tubesheet ligaments closely follows the temperature of fluid in the tubes. Thermal response of the unperforated rim is significantly delayed so that thermal gradients occur in the vicinity of the boundary between the perforated and unperforated regions. Stresses induced by these thermal gradients, and the bulk temperature difference between the perforated and unperforated regions are highly localized in the outer row of ligaments. Therefore, it is necessary to model this local region in great detail in order to evaluate accurately fatigue and creep-rupture damage.

The CRBRP Project plan was to use detailed inelastic finite element analysis of sectors of the tubesheet in conjunction with the strain and creep-fatigue limits for inelastic analysis in Case N-47. The Nuclear Regulatory Commission has concerns with this approach because of difficulties in modeling ligaments and the complex thermal-structural interaction with the rim and the tubes. Their approach essentially is to extend the Section III design procedure based on the equivalent solid plate concept to include the effects of thermal gradients, plasticity, and creep. The specific confirmatory program, that the Project agreed to carry out, is summarized as follows:

- Develop effective properties of perforated region and extend existing Appendix A-8000 Code methods for calculating the linearized membrane, shear, and in-plane bending stresses in the ligaments using the equivalent solid plate stresses.
- Develop methods of evaluating local cyclic plastic and creep strain concentration effects for use in the fatigue evaluation.
- Evaluate elastic follow-up in outermost ligaments as a basis for classification of stresses as "primary".
- Develop ratcheting and creep-rupture damage evaluation methods for outermost ligaments based on equivalent solid plate stresses.
- Perform detailed tube-to-tubesheet joint analysis for tubes in high radial thermal transient region at periphery of the perforated region.

7. Other Issues

The NRC concerns relating to seismic interaction effects, elastic follow-up, creep-fatigue evaluation, plastic strain concentration factors, and transition welds focused primarily on areas where guidance provided by design codes and standards is inadequate. These issues were resolved by Project demonstration that the evaluation procedures used were appropriate for CRBRP specific design conditions. A discussion of the concerns and their resolutions is included in a paper submitted for publication to the International Journal of Nuclear Engineering and Design.

8. Commentary

In a general sense, the NRC review of the CRBRP confirms adequacy of the high-temperature structural design methodology that has been developed over the last 20 years, largely under sponsorship of the U.S. Department of Energy. The design criteria and basic approach to design evaluation have been accepted, and no major inadequacies were discovered. The review identified and resolved a number of issues relative to Code interpretation, and it identified areas where more detailed evaluation techniques would be useful. The required confirmatory programs would both improve design assurance of the CRBRP, and simplify design and evaluation of future plants.

References

[1] Case N-47-21, Class 1 Components in Elevated Temperature Service, Section III, Division 1, Cases of the ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, December 1981.

[2] Safety Evaluation Report related to the construction of the Clinch River Breeder Reactor Plant, NUREG-0968, Vol. 1, Main Report, U.S. Nuclear Regulatory Commission, March 1983.