Review of Current Literature Related to GSI-15

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REVIEW OF CURRENT LITERATURE RELATED TO GSI-15

ABSTRACT

EG&G Idaho is evaluating the structural integrity of Reactor Pressure Vessel Supports as part of its resolution of NRC Generic Safety Issue (GSI) 15, "Radiation effects on Reactor Vessel Supports". Structural analyses have been performed by the NRC, ORNL, LLNL, Portland General Electric Co., and several consultants. Two methods of analyses were used. This paper contains a short discussion of the analyses and the merits of the two analysis methods.

1 INTRODUCTION

Generic Safety Issue (GSI) 15, "Radiation Effects on Reactor Vessel Supports," was originally classified as a low priority issue pending additional data. Upon evaluation by the U.S. Department of Energy (DOE) of the embrittlement and fitness for continued service of the High Flux Isotope Reactor (HFIR), located at the Oak Ridge National Laboratory (ORNL), it was concluded that significantly more embrittlement of the HFIR pressure vessel had occurred than would be expected based on traditional irradiation damage data. Since reactor pressure vessel (RPV) supports typically operate in the same temperature range as the HFIR pressure vessel walls, 100°F to 200°F, and the cavity flux in the vicinity of the core midplane is similar to that of the HFIR vessel wall, it was suspected that reactor pressure vessel supports may also have more embrittlement than expected. In June 1987 the Advisory Committee on Reactor Safeguards (ACRS) asked the U.S. Nuclear Regulatory Commission (NRC) to reconsider the possibility of more rapid than expected embrittlement of RPV supports when they are exposed to low flux at low temperature. In December 1988, the NRC reclassified GSI-15 to a high priority ranking. As a result of this decision a number of analyses have been performed which are described below.

2 SELECTION OF PLANTS FOR ANALYSIS

All of the operating reactors in the continental United States have been surveyed and categorized according to the design of their supports and their vulnerability to embrittlement effects. The survey considered the susceptibility of the supports to irradiation damage, the type of loading, and the character of loading (in general, brittle fracture can take place only in a tensile stress region). All together 125 plants were surveyed, and the reactor support designs were divided into five categories: skirt, long column, neutron shield tank, short column, and suspension. In addition, the short column category was subdivided into seven subgroups. Two plants, Trojan and Turkey Point Unit 3, were selected for plant-specific analyses. These plants were chosen because cantilever beams support their RPVs, thus generating tensile stresses as the primary loads. Also, the beams are located in the high-flux region. In addition, the Trojan plant has 4-in.-diameter grout holes in the top and bottom beam flanges, as well as in the webs of the lower supporting beams. This intensifies the stresses and aggravates the problem (see Figure 1).

3 REVIEW OF THE LITERATURE

At this writing, there are eight reports addressing structural integrity of reactor pressure vessel supports (RPVS) related to GSI-15. Of the eight reports, one evaluated the RPVS of both Trojan and Turkey Point Unit 3 plants (NUREG/CR-5320). The others considered only the Trojan plant. The reports are listed in Table 1. Only two of the reports (NUREG/CR-5320 and PGE) addressed embrittlement due to radiation. The remaining analyses focused on "consequences" of failure, i. e., whether the reactor coolant loop (RCL) piping will transmit the additional loads resulting from failure of the RPVS to other components of the RCL, or in case the supports fail, can the reactor vessel (RPV) be supported by the piping. One concern is that support failure could displace the RPV enough to preclude control rod insertion. All of the analyses are described in some detail in NUREG/CR-5556 (Lipinski and Garner, 1990).

3.1 Analyses addressing radiation effects

The new low temperature and low flux embrittlement information resulted in a request by the U.S. Nuclear Regulatory Commission (NRC) that ORNL conduct further studies of the consequences of embrittlement of RPVS from neutron irradiation. The ORNL researchers (Cheverton et. al., 1989) surveyed operating reactors in the continental USA and selected two plants for site-specific analysis, Trojan and Turkey Point Unit 3. Four different loading conditions were considered in the analyses which included thermal load, dead weight, operating basis earthquake (OBE), safe shutdown earthquake (SSE), and small-break loss-of-coolant accident (SBLOCA).

The report concludes that brittle fracture is a credible event by the end of Trojan's life span, considering embrittlement of the RPVS accelerated by irradiation. The critical flaw size, which is defined as the size of the smallest flaw that will propagate under a given set of assumed conditions and will result in failure of the support, could be as small as 0.42x2.5 in. (depth times length). These values would be further reduced by 50% if residual stresses are considered. Similar conclusions were reached with respect to the Turkey Point Unit 3 plant.

In their 1988 report, Portland General Electric Company (Trojan Docket 50-344, 1988) (PGE), had a different approach. They assumed values of the expected flaw sizes for the lower beam and for the shear pin connecting the column between the upper and lower beams and using the stresses resulting from the structural analysis of the RPVS, calculated the stress intensity factors K_I. This value was compared with a statistically determined value of the material fracture toughness, K_{Ic}, using a factor of safety of 1.414. The RPVS were screened to determine which sections of the supports are exposed to a flux of at least 10^{17} n/cm^2 and a low-temperature environment. Fracture mechanics evaluations were performed only at the sections not meeting the screening criteria, i.e., the column at the lower pin and the lower horizontal radial beam.

The report concluded that the Trojan RPVS are adequate for both normal and worst-case accident loading conditions with an appropriate margin of safety.

3.2 Analyses not addressing radiation effects

The remaining six analyses postulated failure of one or more of the RPVS and predicted, within limits, the consequences of such an occurrence. These analyses were based on methods widely recognized in the professional community, such as beam on elastic foundation or the finite element method.

Dr. J. Ma* used existing test results from a study of cantilever steel beams embedded in concrete (Merkatis and Mitchell, 1980) and an analytical approach to demonstrate that if one of the Trojan support beams would fail through a fracture in a vertical plane at the centerline of the grout hole, the remaining portion of the beam, which is embedded in concrete, would be capable of supporting the design load. The analysis was performed in conformance with the criteria of the American Concrete Institute Standard ACI-318.

The objective of the analysis by Dr. J. O'Brien** was to estimate the existing margins in the pressurized water reactors (PWR) reactor vessel supports and to evaluate the capability of the coolant loop piping to support the reactor pressure vessel during an earthquake. The calculations were performed, on a generic basis, using plastic analysis of coolant loop piping. The analysis demonstrated that the structural integrity of a reactor vessel and coolant loop will not be jeopardized even if all of the supports fail.

The report prepared by the Brookhaven National Laboratory (Braverman and Miller, 1989) contains an analysis which assumed that both beams in one of the Trojan reactor vessel supports is completely fractured at the centerline of the 4-in. grout hole. The analysis used a system model that represented the reactor vessel supports as four rigid beams. Hinges that permit free rotation about two orthogonal axes were assumed at the end of each of the rigid beams. The analysis was checked by computing the ductility, defined as the ratio of the maximum displacement to the elastic displacement. The ductility ratio was not excessive.

The report by Dr. S. T. Rolfe (Rolfe, 1989) discusses the ORNL report (Cheverton 1989) and makes recommendations that could be helpful in resolving the GSI-15 safety issue. It does not analyze the RPVS from a structural point of view. Instead, it deals with the possible existence of a flaw or crack at the 4-in.-diameter grout hole in the top flanges of the supporting beams. Dr. Rolfe, using the stress analysis performed by Dr. R. W. Furlong, (Furlong, 1989) calculated the critical flaw size as 2.2-in. He questioned the application of fracture mechanics to this situation, stating that such an analysis is based on the assumption that a sharp crack is present. Based on the review of the quality control specifications, the report precludes the possibility that a flaw as big as 2.2-in. (as calculated by Dr. Rolfe) would have gone unnoticed by inspectors at the time of fabrication of the beam.

4 PLANNED RESOLUTION OF GSI-15

In partial fulfillment of its contract with the NRC, EG&G Idaho, Inc., is developing the criteria for evaluation of RPVS. The criteria will consist of a screening phase and the evaluation phase. Two alternatives in the evaluation phase are being proposed, a temperature approach and a fracture mechanics approach. In the temperature approach, the lowest operating temperature (the minimum temperature of the most vulnerable part of the fracture-critical member) must exceed the irradiated nil-ductility-transition-temperature $(NDTT)_{IRR}$ by a certain margin. In the fracture mechanics approach, the stress intensity factor, K_I, must be calculated on the basis of the state of the stresses in the member and compared to the fracture toughness of the material, K_{Ic}. If neither of these conditions can be met a more accurate analysis can be performed, whereby new thermo-hydraulic loads can be developed and elasto-plastic stress analyses performed which should result in lower predicted stresses at the critical sections. Then, the K_I is calculated again and

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*Presentation by Dr. J. Ma to the joint meeting of ACRS and Subcommittees for Materials and Metallurgy/Structural Engineering, March 23, 1989.

**Presentation by Dr. J. O'Brien to the joint meeting of ACRS and Subcommittees for Materials and Metallurgy/Structural Engineering, March 23, 1989.

compared to the K_{Ic}. If the criteria are not satisfied, EG&G recommends that RPVS be modified to preclude possibility of a brittle fracture.

CONCLUSIONS

Two distinct types of analyses have been reviewed. One postulates failure of one or more of the RPVS and predicts, within limits, the consequences of such occurrence. The other examines the stresses and embrittlement due to irradiation and attempts to predict the possibility of a brittle fracture. It is difficult to decide which type of an analysis is more realistic and accurate. The complexity of the problem requires considerable engineering judgement regarding effectiveness of the liner, the possibility of the concrete support under the steel beams being crushed, and possibility of shear failure of concrete above the beams. It follows from the above that the present state of the art does not offer any one reliable methodology by which RPVS can be accurately evaluated.

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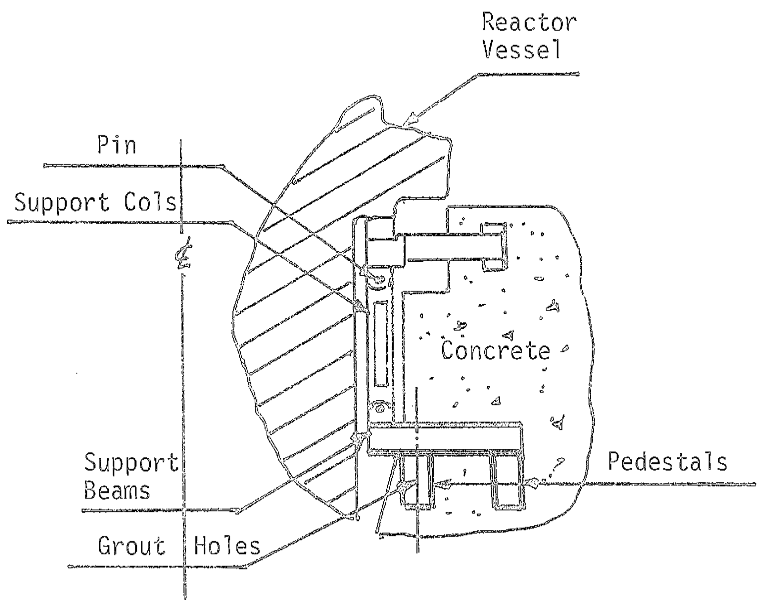

Fig. 1. Radial section through Trojan reactor vessel supports

TABLE 1. ANALYSES AND REPORTS RELATED TO GSI-15

REPORT	PURPOSE	ASSUMPTIONS MADE	CONCLUSIONS
NUREG/CR 5320 (ORNL)	Determine Crit. Flaw	Fluence-Rate Causes Embrittlement	Brittle Fracture is Credible Event
PGE 1988	Investigate RPVS	Flaw Sizes For Pin & Beams	RPVS Meet Safety Factor of 1.414
J. MA (ACRS)	Struct. Eval. of RPVS	Beams Break At Grout Hole	Remaining Portion of Beams Support RPVS
J. O'Brien (ACRS)	Investigate Failure of RPVS	All Supports Fail	RCL Piping Can Support RPV
S. T. Rolfe 1989	Struct. RPVS Evaluation	3v5NDTT=75^{o}F	Brittle Fracture is Questionable
BNL 1989	Invest. Capacity of 3 supports	One Support Fails	No Catastrophic Consequences
NUREG/CR 5644 (LLNL)	Investigate Failure of RPVS	All Supports Fail	RCL Piping Can Support RPV
R. W. Furlong 1989	Struct. RPVS Evaluation	Redistribution of RPV Forces	No Embrittlement Sensitivity