LEAK-BEFORE-BREAK EVALUATION OF A TYPICAL PRESSURIZED WATER REACTOR PRIMARY LOOP PUMP CASING

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ABSTRACT

In a nuclear plant of Westinghouse design the reactor coolant system components may be forged or cast stainless steel and varies from plant to plant. The reactor coolant pump casings are always cast. Both forged and cast product forms exhibit exceptionally high toughness in the as-build condition; however, the cast material is susceptible to thermal aging. Thermal aging causes significant reduction in fracture toughness, while the tensile strength and yield stress of the materials most often increase.

In this paper an integrity evaluation of a typical Pressurized Water Reactor (PWR) primary coolant loop pump casing is provided. A systematic way of assuring component integrity is accomplished by using the Leak-Before-Break (LBB) methodology. Therefore, an LBB evaluation including the effects of thermal aging degradation is performed as described in this paper. A prerequisite for performing a valid LBB evaluation is an accurate determination of stresses resulting from various loading conditions. The stress results are obtained from a detailed 3-D finite element analysis.

The integrity evaluation has an added benefit. In March 1990, the ASME Section XI Division 1 issued a code case namely Code Case N-481 which provides alternate in-service inspection requirements for cast austenitic stainless steel pump casings. The code case permits a convenient visual examination in lieu of the volumetric examinations, provided the integrity of the pump casings is demonstrated by analytical evaluations. Although the code case emphasizes part through flaws, the LBB evaluation provides strong support for the approach taken (i.e. even if a part through flaw should penetrate the pump casing wall, the integrity of the pump casing against a large failure remains intact).

1.0 INTRODUCTION

General Design Criterion 4 (GDC-4). "Environmental and dynamic effects of design bases," requires the consideration of the dynamic effects of postulated loss-of-coolant accidents, including missiles, pipe whipping, and discharging fluids. In October 1987, the NRC revised GDC-4 via the broad scope final rule, 52FR41288 [1] to allow the use of Leak-Before-Break (LBB) concepts to exclude the dynamic effects of postulated pipe ruptures from the design basis of certain plant structures. Accordingly, all the Westinghouse design nuclear plants in the United States have applied LBB to the primary loop pipings. In addition, LBE has been applied to the auxiliary nuclear piping in many plants. With the successful application of LBB methodology to primary loop and auxiliary nuclear piping, it is reasonable to assume that the same methodology is applicable to pump casings.
In a nuclear plant of Westinghouse design the primary piping may be forged or cast stainless steel and varies from plant to plant. The large primary loop fittings are always cast. Both forged and cast product forms exhibit exceptionally high toughness in the as-built condition; however the cast material is susceptible to thermal aging. Essentially, the fracture toughness may be significantly reduced with time at operating temperatures for the cast material. The toughness degradation in cast austenitic stainless steel has been attributed mainly to the successive precipitation of chromium in the ferrite phase due to the large miscibility gap in the Fe-Cr binary system. During aging at temperature, the ferrite phase gradually develops a cleavage transition behavior somewhat similar to that of ferritic stainless steel.

The thermal aging toughness degradation has only within the last fifteen years been recognized as occurring in cast stainless steels at operating temperatures of nuclear reactors. Useful material test data and acceptance criteria became available only recently. The thermal aging issue has been technically addressed by Westinghouse and the procedure currently used to address thermal aging has been approved by the United States Nuclear Regulatory Commission (USNRC).

The primary loop pump casings of Westinghouse type nuclear plants are cast stainless steel and thus require a consideration of thermal aging. The two cast stainless steel materials of concern are SA351 CF8 and SA351 CF8M, the latter material generally being more susceptible to significant degradation due to thermal aging than the former. An examination of the available material certifications for Westinghouse type PWRs revealed that the pumps of all but three plants were fabricated entirely from SA351 CF8, the material less sensitive to thermal aging. Thus, the typical pump casing LBB evaluation presented here is for SA351 CF8 pump casings.

A prerequisite for performing a valid LBB evaluation is an accurate determination of stresses resulting from various loading conditions. The stress results are obtained from a detailed 3-D finite element analysis.

The objective of this paper is to demonstrate applicability of LBB concept to a typical Westinghouse Pressurized Water Reactor pump casings.

2.0 DESCRIPTION OF THE PRIMARY LOOP PUMP CASINGS

There are essentially eight different models of pumps in Westinghouse type PWRs. They are designated Models 63, 70, 93, 93A, 93A1, 93D, 100A and 100D. Models 93A and 93A1 are most common among Westinghouse type PWRs, making up seventy-five percent of the total on a domestic plant basis. This formed the basis for choosing the Model 93A as a typical representative pump in this evaluation. Typical dimensional details are provided in Figure 1.

3.0 FRACTURE TOUGHNESS OF THE PUMP CASINGS

The material certifications were examined for each pump casing and the parts that were welded together to form the entire component (casing) were identified. The chemistry of each heat of material was used to determine the ferrite content and hence the fracture toughness. The ferrite content was found to be typically in the range of 6 to 10%. The lower bound allowable fracture toughness properties were conservatively established to be $J_{\text{IC}} = 3000 \text{ in-lb/in}^2$, with associated $J_{\text{IC}}$ and $T_{\text{EC}}$, and were used in the flaw stability evaluations.

4.0 NORMAL OPERATING AND FAULTED LOADS FOR THE PUMP CASINGS

For normal operating loads at the nozzles the algebraic sum of the normal forces was taken as the applied normal operating force. The directional moments were combined algebraically to obtain total directional moment loads. The resultant applied moment was found by taking the square root of the sum of the squares of the total directional moments as indicated in Table 1. For
faulted loads, the absolute values were used for all the individual loads. The representative normal operating and faulted loads are shown in Table 1.

5.0 THE FINITE ELEMENT STRESS ANALYSIS OF THE PUMP CASINGS

The finite element model for the 93A pump casing is shown in Figure 2. 3-D isoparametric brick elements (20 nodes) and 3-D isoparametric wedge elements (15 nodes) were used. The model spans 180 degrees accounting for a conservatively assumed symmetry about a plane bisecting the outlet nozzle and pump casing. There are 1210 elements and 6218 individual nodes in the model. Stress analysis results were extracted for various "cuts" perpendicular to the circumference of the pump casing. The stresses were reviewed to identify critical locations for LBB evaluations.

6.0 A SUMMARY OF THE APPLIED LEAK-BEFORE-BREAK METHODOLOGY

The methodology used in this paper for evaluating leak-before-break for the pump casings is consistent with that recommended in NUREG 1061 [2]. The methodology involves:

1) Establishing material properties including fracture toughness values
2) Performing stress analyses of the structure
3) Review of operating history of the structure
4) Selection of locations for postulating flaws
5) Determining a flaw size giving a detectable leak rate
6) Establishing stability of the selected flaw
7) Establishing adequate margins in terms of leak rate detection, flaw size and load
8) Showing that a flaw indication acceptable by inspection remains small throughout service life

7.0 RESULTS

Leak-Before-Break evaluations were performed at six critical locations. Specifically, two-phase flow calculations were performed to determine the flaw sizes giving 10 gpm leak rates — "leakage size flaws." These flaw sizes ranged from 6.6 inches to 10.9 inches in length. J-T analyses were performed by postulating through wall flaw sizes. Postulated flaw sizes ranged from 13.2 inches to 21.8 inches in length (i.e., two times the leakage size flaws). These flaws were subjected to the faulted condition loads. Calculated $J_{applied}$ varied from 53 in-lb/in^2 to 1852 in-lb/in^2 and the tearing modulus varied from 0 to 11.1. These values were lower than the lower bound material toughness, thus, flaw stability was demonstrated.

8.0 DISCUSSION AND CONCLUSIONS

An integrity evaluation of a typical Pressurized Water Reactor (PWR) primary coolant loop pump casing was performed using the leak-before-break methodology. The effects of thermal aging degradation were included in the evaluation. The model 93A pump casing was used in this evaluation to represent a typical PWR pump casing.

In conclusion, leak-before-break analyses were shown to be feasible for the pump casing considered in this paper. The integrity evaluation has an added benefit. In March 1990, the ASME Section XI Division 1 issued a code case, namely Code Case N-481, which permits a convenient visual examination in lieu of the volumetric examinations provided the integrity of the pump casings is demonstrated by analytical evaluations. Although the code case emphasizes part
through flaws, the LBB evaluation provides strong support for the approach taken (i.e. even if a part through flaw should penetrate the pump casing wall, the integrity of the pump casing against a large failure remains intact).

REFERENCES


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Figure 1. Dimensional Sketches of the Model 93A Pump Casing Showing the Weld Seam Location
Figure 2. 3-D Finite Element Model of the Model 93A Pump Casing