A Pressure Tube Rupture Analysis of a Graphite Moderated Reactor Core

S. N. Dermitzakis, R. Allahabadi, T. P. Desmond
Impell Corporation, Walnut Creek, CA USA

1. INTRODUCTION

Design and construction of the Department of Energy's N Reactor located in Richland, Washington was begun in the late 1950s and completed in the early 1960s. Recently, the reactor core's structural integrity has been under review and is considered by some to be a possible safety concern. The safety concern stems from the degradation of the graphite moderator due to the effects of long-term irradiation. Also, as a result of the recent Chernobyl accident in the Soviet Union, two additional concerns have been raised. First, the N Reactor's graphite-moderated reactor is similar to the Chernobyl reactor, which was severely damaged by an accident in 1986 (Ref. 1). Second, like Chernobyl, it does not have a containment structure similar to those of commercial reactors in the U.S. Structural evaluations are presented here that examine whether the N Reactor is safe to operate if a single pressure tube rupture hypothetically occurs (Ref. 2).

The core consists of a graphite stack about 33 feet by 39 feet by 33 feet high, which is enclosed by steel-lined concrete walls approximately 3-1/2 feet thick (Fig. 1). The graphite core serves to moderate fission neutrons to thermal neutrons and serves to support Zircaloy tubes which penetrate the core. One thousand 3-3/4 inch Zircaloy tubes perforate the core in the north-south direction and contain pressurized water and nuclear fuel. An additional 640 3/4 inch Zircaloy coolant tubes perforate the core in the east-west direction and circulate water. The graphite stack is comprised of 89 layers of graphite blocks which are assembled in a 'lincoln-log' lattice to permit venting of steam in the event of a pressure tube rupture (Fig. 2).

Since construction in the early 1960s, some of the graphite blocks may be ineffective due to general distortions of the graphite moderator and due to cracking of the keys which interlock the blocks. For extended operation beyond the original design life, even more keys may become ineffective. Additionally, the long-term irradiation embrittles the Zircaloy tubes which may contribute to a tube rupture. The principal concern was that energy released from one pressure tube rupture could induce a failure in an adjacent tube and thereby result in a cascading failure of numerous pressure tubes. The objective was to evaluate whether the reactor core would maintain structural integrity and the safety systems would remain functionally operable during the postulated pressure tube rupture. To assess the safety of the reactor core and its ability to preclude a cascading failure of Zircaloy pressure tubes, a single pressure tube rupture was postulated and the loads from that rupture were applied to adjacent tubes. If the adjacent tubes retained pressure integrity and precluded a cascading failure of pressure tubes, the reactor core was considered to meet its safety objectives. Thermal-hydraulic analyses were performed to develop loads due to the postulated tube rupture; then, structural response analyses were performed to evaluate the response of adjacent tubes subjected to the thermal-hydraulic loads.

Two rupture types were considered: A circumferential break which is referred to as a double-ended guillotine break (DEGB) and a straight break along the logitudinal axis of the tube which is referred to here as a longitudinal split break (LSB).

2. THERMAL-HYDRAULIC LOADS AND STRUCTURAL MODELS

Thermal-Hydraulic Loads. When a pressure tube breaks, a steam/water mixture travels through the core causing pressure and momentum loads on the graphite blocks. For the LSB, jet impingement loads are also induced on the adjacent moderator blocks. Three locations were selected at mid-height of the core for developing the thermal-hydraulic loads. The break types were: a DEGB at the center; a DEGB at the edge of the fueled zone; and a LSB of the center tube. The computer program RELAP/MOD1 (Ref. 3) which simulates the flow of steam-water-gas mixtures in complex networks was used. Further details of the thermal-hydraulic analyses can be found in Reference 2.

Structural Models. These thermal-hydraulic loads were applied to structural models of other pressure tubes to assess the potential for initiating a cascading failure of pressure tubes. To be conservative, the locations of the tubes for the structural response analysis were selected based on the potential for the loads to induce large flexural strains in the tubes. These locations corresponded to the worst-case structural conditions in the core. The structural models were chosen to represent the top central region of the core, where frictional resistance was small, excessive key breakage was suspected and gaps between adjacent pressure tubes had developed as a result of irradiation-induced distortions of the graphite blocks. The locations of the tubes for the structural evaluations were not therefore the same as those for the thermal-hydraulic load development.

The structural models were representative of the state of the core described in surveillance reports of the N Reactor (Refs. 4 through 7). The core was assumed to be in an "extended operation" state. Due to the localized nature of the pressure tube rupture loads, only local regions of the core are affected. Therefore, local structural models were analyzed. The following assumptions were made for the structural models:

(a) Pressure tubes were modeled with linear elastic beam elements. Inertial resistance was modeled by lumped masses at the nodes of the beam elements. No viscous damping was specified for the pressure tubes. Safety evaluation criteria for the N Reactor (Ref. 8) specify a maximum pressure tube strain of 1%; beyond 1% strain, the tube is considered as failed.

(b) Direct bearing of pressure tubes on filler block keys was considered. This was modeled by nonlinear truss elements. The ultimate strength of a pair of filler block keys was taken as 11 kips at an ultimate displacement of 0.04 inch. These values represented the average of the test results reported in Reference 9.

(c) Frictional resistance of the top and bottom faces of pressure tube blocks was considered. This resistance was modeled by elastic-plastic nonlinear truss elements.

(d) Gaps between adjacent pressure tubes were modeled by gap elements. A gap element provided no resistance until the gap was closed, and had high stiffness, once the gap closed.

Local structural models were developed for postulated DEGB and LSB of pressure tube (PT) 3361 or 3358, located at the uppermost tube layer of the reactor core. A structural model for rupture of PT 3361 is shown in Figure 3. This model included two adjacent pressure tubes 3359 and 3360 on the east side of pressure tube 3361, as surveillance reports indicated maximum gaps in the vicinity of PT 3359. A 1-1/5 inch gap was modeled between pressure tubes 3359 and 3360, and 1/2 inch gap between pressure tubes 3359 and 3358 in the central part of the core. Outside this central region, the gap was reduced linearly to zero inch at the edges.

3. STRUCTURAL RESPONSE

Physical gaps are present between pressure tube blocks due to general distortions of graphite moderator and due to cracking of the keys which interlock the blocks. Gap closures result in nonlinear behavior; thus, a nonlinear solution was used to account for gap closures as well as graphite block key breakage. The structural transient dynamic analysis for pressure tube rupture response was performed using Impell Corporation's version of the program ADINA.

DEGB : Transient thermal-hydraulic loads were generated for double-ended guillotine break at (a) the inlet edge of the fueled zone (cold leg), (b) the center of the fueled zone, and (c) the outlet edge of the fueled zone (hot leg). The DEGB at the cold leg results in a more critical load than the DEGB at the center of the core and at the hot leg of the core. For all DEGB analyses, strains induced in adjacent pressure tubes were small ($\leq$0.10%). Therefore, a DEGB of a pressure tube anywhere in the core would not affect the integrity of adjacent pressure tubes.

LSB : Thermal hydraulic loads were generated for a longitudinal split rupture at PT 3361 or PT 3358. The split was assumed to initiate at the cold leg edge of the fueled zone and propagate at a speed of 1000 ft/sec towards the hot leg edge of the fueled zone. Results are presented for a LSB at PT 3361, since it was found to be more critical than LSB at PT 3358. Since, the gaps on the east side of PT 3361 are larger than the west side, the LSB at PT 3361 is assumed to occur in such a way as to generate loads on PT 3360 and PT 3359. However, in contrast with DEGB, LSB results in loads throughout the length of the adjacent tubes, though with increasing time delay for the hot leg edge of the core. These loads have an initial peak and decay rapidly to a steady state value that is significantly lower than the peak.

The LSB at PT 3361 produced significantly higher displacements and strains when compared to the DEGB. The maximum strains for PT 3360 and PT 3359 were 0.23% and 0.35% . The loads caused multiple key breakage. Figure 4 shows a "snapshot" of the displacement shape for PT 3359 at the time of maximum strain. Figure 5 shows the displacement time history adjacent to the maximum strain location in PT 3359. Initial zero displacement indicates the time taken by the split to propagate to that location. From this analysis, it was inferred that maximum strain for PT 3360 is a function of unkeyed span, (due to breakage of keys), and the maximum relative displacement between adjacent points on the tube. For PT 3359, the maximum strain is a function of neighboring gap sizes and impact between PT 3360 and PT 3359.

LSB Parametric Analyses. Parametric studies were performed for LSB to assess the sensitivity of the response to changes in modeling assumptions. Each parametric study varied only one parameter and are described below.

Pressure Tube Stiffness Study. The pressure tube stiffness was modified to account for pressure tube block stiffness. This increased the stiffness of the pressure tubes significantly and resulted in smaller strains than the base-case analyses.

Graphite Key Ultimate Displacement Study. To account for variations in experimental results for ultimate displacement, one additional analysis was performed with an ultimate displacement for the keys of 0.02 inch. The analysis resulted in an overall response that has lower tube strains compared to the base-case analyses.

Graphite Key Ultimate Strength Study. Experiments (Ref. 9) indicate a range for ultimate strength of 6 kips to 19.3 kips for a pair of keys. Base-case analysis indicated that large local strains may occur, if only a few keys fail. Therefore, it was necessary to consider various arrangements of weak and strong keys to determine a critical arrangement. The critical arrangement resulted in a maximum strain of 0.43% for PT 3360.

Crack Propagation Speed Study. Upper and lower bound crack speeds for the Zircaloy tubes were determined to be 3340 ft/sec and 1000 ft/sec, respectively (Ref. 2). Based on these results, it was concluded that base-case assumptions considered worst-case conditions for the analysis of pressure tubes.

4. CONCLUSIONS

Analyses showed that the maximum strain induced in the pressure tubes was 0.43%. The maximum strain occured for the LSB load case. Based on the 1% strain failure criteria, analyses showed that no cascading pressure tube failure can occur. It is important to note that recent experimental investigations (Ref. 10) have shown that the total elongation of a fractured irradiated tube is 16.6% in tension and 9% in compression. To ensure that changes in the reactor core elements due to long-term thermal and irradiation effects do not invalidate the modeling assumptions of the analyses, a continuing surveillance program is necessary. Items which would affect the analysis results include changes in graphite properties, changes in Zircaloy tube properties and additional distortions of the graphite lattice.

5. REFERENCES

1. "Chernobyl Unit 4 Pressure Tube Rupture Evaluation", Impell Report No. 01-2520-1542, Revision 0, prepared for Westinghouse Hanford Company, Impell Corporation, Walnut Creek, California, September 1986.
2. "Pressure Tube Rupture Evaluation - Supplement 1", Impell Report No. 01-2520-1605, Revision 0, prepared for Westinghouse Hanford Company, Impell Corporation, Walnut Creek, California, December 1987.
3. Ransom, V.H., et. al. "RELAP/MOD1 Code Manual, NUREG/CR-1826", prepared for the U.S. NRC, EG&G Idaho, Idaho Falls, Idaho, March 1982.
4. Alexander, W.K., "Status of the N Reactor Moderator Distortion Problem", UNI-1743, UNC Nuclear Industries, Richland, Washington, May 1981.
5. Ostic, J.K., "N Plant System Surveillance Report for 1984", UNI-391I, UNC Nuclear Industries, Richland, Washington, March 1985.
6. Newby, F.M., Marshal, R.P., "Graphite Stack Inspection and Status Report, Calendar Year 1982", UNI-2109, UNC Nuclear Industries, Richland, Washington, May 1983.
7. Alexander, W.K., Newby, F.M., "Status of the N Reactor Graphite Stack as of November 1, 1978", UNI-1188, UNC Nuclear Industries, Richland, Washington, December 1978.
8. Gonzalez, L.M., "N Reactor Safety Evaluation Criteria", UNI-3171, UNC Nuclear Industries, Richland, Washington, July, 1985.
9. Morgan, W.C., Scott, K.V., "Strength Measurements on TSX Graphite", UNI-3152, UNC Nuclear Industries, Richland, Washington, March 1985.
10. Gardner, H.R., "Interim Report: Bend Testing of N Reactor Pressure Tubes",Westinghouse Hanford Company, Richland, Washington, October 1987.

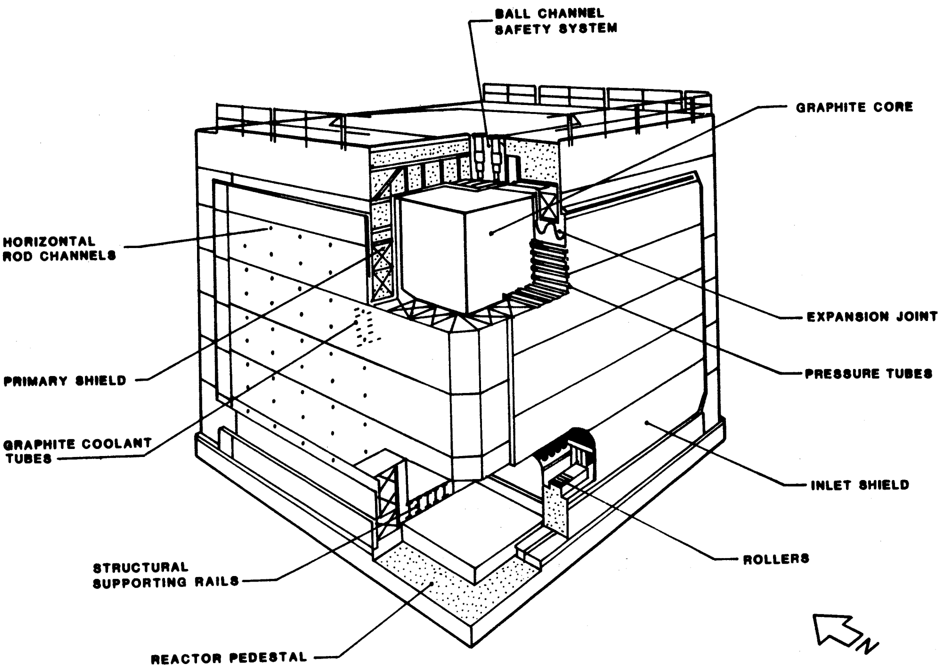

FIGURE 1 - SCHEMATIC REPRESENTATION OF THE N-REACTOR CORE

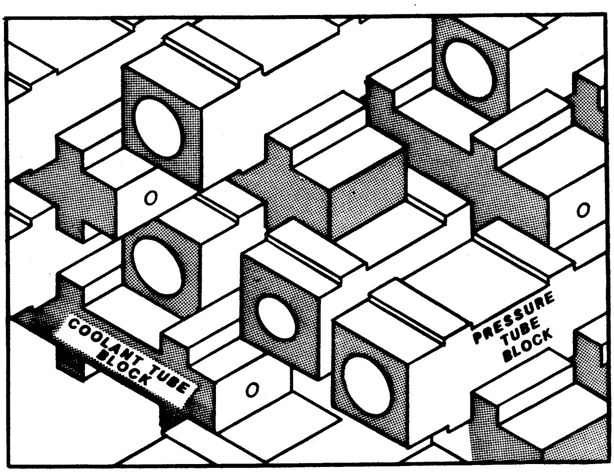

FIGURE 2 - GRAPHITE CORE DETAILS

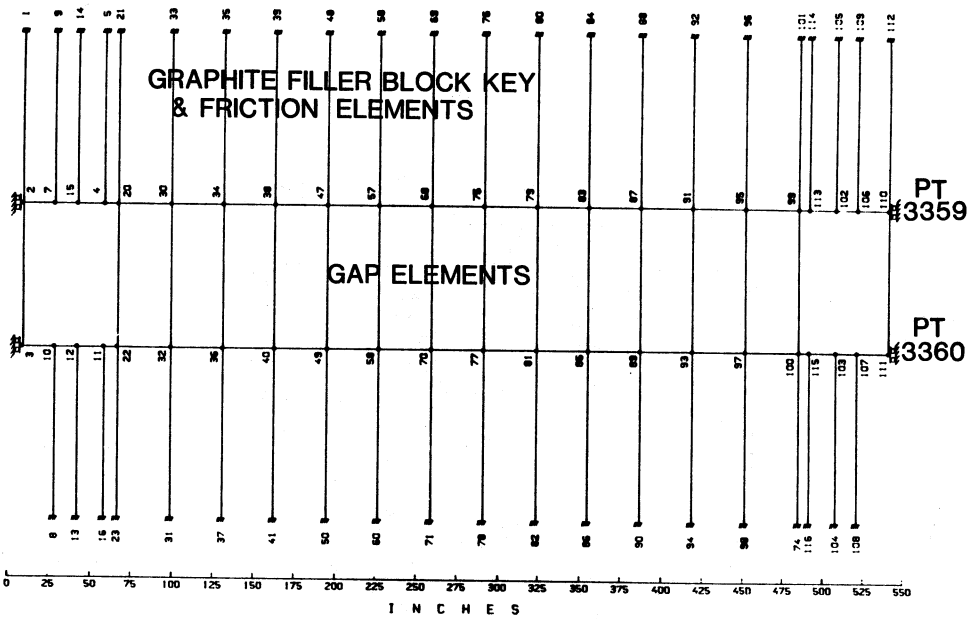

FIGURE 3 - STRUCTURAL MODEL OF PRESSURE TUBES 3359 AND 3360

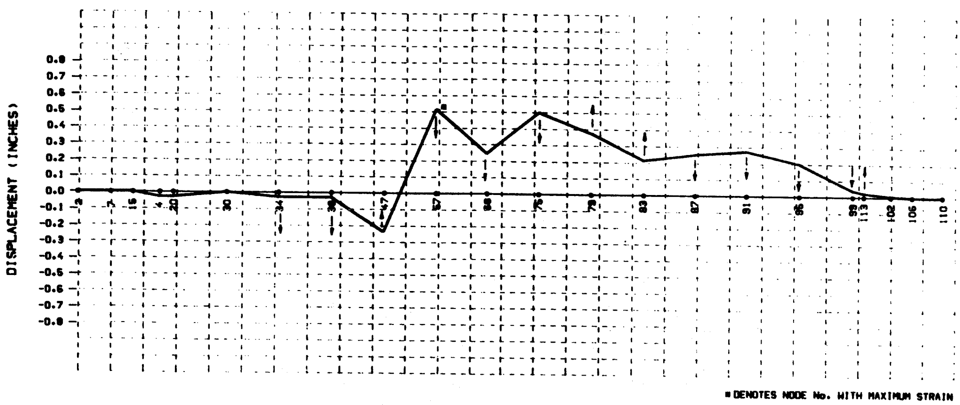

FIGURE 4 - DISPLACEMENT "SNAPSHOT" OF PT 3359 AT TIME OF MAXIMUM STRAIN

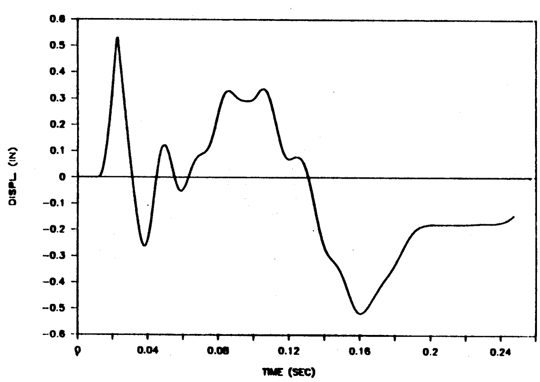

FIGURE 5 - DISPLACEMENT TIME HISTORY OF PT 3359