B&W 177 FA OWNERS GROUP

ASYMMETRIC LOCA LOADS EVALUATIONS PROGRAM,

PHASE 2

Arkansas Power & Light – ANO-1
Duke Power Company – Oconee 1, 2, 3
Florida Power Corporation – Crystal River 3
Metropolitan Edison Company – Three Mile Island 1, 2
Sacramento Municipal Utility District – Rancho Seco
Toledo Edison Company – Davis-Besse 1
Consumers Power Company – Midland 1 and 2

April 10, 1979

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1.0 INTRODUCTION

This report summarizes Phase 2 of the detailed plan prepared by the B&W 177 FA Owners Group in response to the NRC Division of Operating Reactors letter dated January 25, 1978.

Whereas, Phase 1 performed preliminary investigations using estimates and generalities to bracket the evaluation requirements, Phase 2 will go into greater details to determine more specific loads and component/structural evaluations.

The need to proceed with Phase 3 is still unresolved at this time, and will be addressed at a later date after discussions with the NRC.

2.0 EVALUATION BASES

2.1 The components to be evaluated during Phase 2 for the LOCA breaks analyzed include:

a. Reactor Pressure Vessel
b. Fuel Assemblies, Including Grid Structures
c. Control Rod Drive Mechanisms
d. ECCS Piping that is Attached to the Reactor Coolant Piping
e. Reactor Coolant Piping in Close Proximity to the Reactor Vessel
f. Reactor Vessel Supports
g. Reactor Internals
h. Reactor Cavity Wall
i. Core Flood Piping
j. Building Structures Associated With the Above Components

2.2 LOCA analyses will be performed for breaks rendering the worst loadings for the Reactor Vessel Supports and Reactor Internals. For these breaks, all components listed in Paragraph 2.1 will be evaluated to assure that core coolable geometry is maintained mitigating the consequences of a loss of coolant accident.

2.3 Jet impingement effects will be evaluated for the breaks analyzed. This evaluation was not explicitly stated in the NRC letter, but was identified as a requirement in a previous meeting (March 31, 1978) with the NRC.

2.4 As appropriate, the evaluation will consider:

a. Limited displacement break areas where applicable
b. Use of actual time-dependent forcing function
c. Reactor support stiffness
d. Break opening times
e. Break location utilizing stress criteria

2.5 Where possible, generic evaluations of the B&W Owners Group plant components will be performed.

3.0 WORK PLAN (PHASE 2)

3.1 The objective of this task is to define the resultant forces and moments which would act externally on the reactor vessel in the event of a reactor coolant system (RCS) pipe rupture inside of the reactor subcompartment. The CRAFT2 computer code will be used to calculate the transient, asymmetric pressure distributions inside the subcompartment for a spectrum of break cases. Analysis guidelines established in Standard Review Plant (SRP) 6.2.1.2 for subcompartment pressurization calculations and in SRP 6.2.1.3 for mass and energy release calculations will be followed.

The reactor cavity designs will be evaluated based upon the following considerations:

1. Cavity volume between reactor vessel and primary shield wall.
2. Insulation design.
3. Vent areas of primary piping penetrations.
4. Shield plugs or blow-out devices.
5. Flow obstructions.

3.2 Develop a mass and energy release calculation model for a generic 177 lowered-loop RCS operating at a power level of 1.02 X 2772 MWt. Generate mass and energy release data for the initial two (2) seconds of blowdown for hot and cold leg breaks of the following sizes:

a. 2.0A
b. 1.5A
c. 1.0A
d. .6A
e. .3A

3.3 Develop a mass and energy release calculation model for the Davis-Besse 1 plant at a power level of 1.02 X 2772 MWt. Generate mass and energy release data for the initial two (2) seconds of blowdown for hot and cold leg breaks of the following sizes:

a. .5 ft^2
b. 1.0 ft^2
c. .5A
d. 1.0A
e. 2.0A

3.4 The core flood line will be treated as a cold leg break of appropriate size.

3.5 For evaluation of the mass and energy data generated in Paragraph 3.2 develop five (5) reactor cavity models. Calculate the reactor cavity pressurization rates for the spectrum of hot and cold leg breaks identified. Calculate the time-histories of the lateral and vertical forces and moments acting on the reactor vessel out to a blowdown time which is sufficient to define the peak magnitudes of these forces and moments. The development of five (5) reactor cavity models will enable the determination of forces and moments on a plant specific basis.

3.6 For evaluation of the mass and energy data generated in Paragraph 3.3, develop a reactor cavity model. Calculate the reactor cavity pressurization rates for the spectrum of hot and cold leg breaks identified. Calculate the time-histories of the lateral and vertical forces and moments acting on the reactor vessel out to a blowdown time which is sufficient to define the peak magnitudes of the forces and moments.

3.7 Calculate the loss-of-coolant-accident (LOCA) loadings on the reactor internals structures of 177 FA plants. The calculations will be performed using procedures documented in BAW Topical Report 10132. A spectrum of break sizes will be considered in the hot and cold leg piping of the reactor coolant system (RCS). Break locations inside the reactor cavity and outside the primary shield wall in the steam generator compartment will be considered.

3.7.1 A generic model will be developed for calculating LOCA loads on 177 FA lowered-loop plants. The modeling criteria established in Topical Report BAW-10132 will be used in the development. Initial reactor fluid conditions which encompass all 177 plants for purposes of LOCA load calculations will be specified in the model.

Design LOCA load calculations will be performed for eight (8) break cases inside the reactor cavity and for two (2) break cases in the steam generator compartment, with a contingency for analyzing up to four (4) additional breaks anywhere in the RCS. The break sizes and the corresponding break opening times will be selected on the basis of the results of the Phase 1 program. The analyses will be conducted out to 0.3 sec of the blowdown.

The following parameters will be recorded as a function of time for each design case calculation:

a. Control volume pressures.
b. Major component ΔPs.
c. Vertical force on the core.
d. Vessel head ΔP.
e. Integrated lateral forces on pressure vessel and core support cylinder.
f. Integrated lateral load on plenum cylinder.
g. Mass and energy release to containment.
h. Jet intensity at break plane.

3.8 The fuel assembly model parameters (mass, spring rate, and damping halves) will be calculated for use in the fuel assembly model. These parameters will be envelope values for the Mark B fuel assembly and will be used as inputs in the development of the Core Bounce Model.

3.9 An existing core bounce model will be modified to reflect the 177 Mark B fuel assembly. The vertical cavity pressure will be applied to the fuel assembly model with a spectra of breaks previously identified, and the resultant load impact at the upper and lower grids will be calculated. These loads will be presented in the form of time-histories and will be used as input into the Reactor Vessel Isolated Model. The core bounce model is non linear in nature due to the springs and gaps. Thus, amplified forces supplied to the linear model include the dynamic impact of the fuel assemblies in the vertical direction.

3.10 The bending and extensional stiffnesses of the reactor vessel internals will be calculated for input into the isolated dynamic model. For the non-redundant structures such as the core barrel, thermal shield, and core support shield the stiffnesses will be calculated using classical methods. For the plenum assembly, the apparent differences of two redundant load paths will be calculated: the plenum cylinder path and the column weldment path. This is accomplished with a three-dimensional finite element model of the plenum assembly. The apparent stiffness of the column weldments will be calculated from their average displacement while the plenum cylinder average displacement at its base will be used to calculate its stiffness. This method accounts for the effects of the plenum cover and upper grid which will not be included in the isolated model.

3.11 Existing 177 fuel assembly reactor vessel internals model (TECO, Davis-Besse 2 and 3) will be modified to reflect the RV skirt support or the TECO Davis-Besse 1 supports. This model will include the reactor vessel internals as beam elements obtained earlier, and the fuel assembly model also obtained earlier. The model will include service support structure, CRDM, cold leg piping, and hot leg piping to the extent feasible. The TECO Davis-Besse 1 model will reflect the internals design for Davis-Besse 1 instead of Davis-Besse 2.

3.12 Dynamic LOCA analysis (linear elastic) will be performed on the model generated above. This analysis will include the following as input forcing functions:

a. Horizontal delta pressures integrated over the wetted surfaces of the internals and the inside of the vessel shell to describe the horizontal forcing functions on the vessel and internals.

b. Vertical delta pressures integrated over the RV heads to describe the vertical force on the vessel.

c. Vertical core bounce forcing functions are applied at the plenum cover ledge and include all vertical delta pressure integrations across the internals and core, and all the vertical dynamics of the internals.

d. Asymmetric cavity pressures are integrated over the outside surface of the vessel and applied to the vessel.

NOTE: 1) The "thrust force" is included in (a) above the area of the broken pipe is excluded from the integration and the area of the unbroken pipe is included.

2) This task will utilize the results of a hydrodynamic mass coupling developed separately.

3.13 Non-linear pipe whip analysis model will be constructed representing the non-linear material properties and existing gaps in each of the plants. These models will reflect the as-designed status and the as-built gaps that can be obtained from Phase 1. Calculations will be performed to determine the break discharge area for each of the breaks identified in the spectrum of the breaks outlined earlier. An iterative approach will be required to obtain the final non-linear pipe break area.

3.14 To obtain the reaction forces of the fluid on the primary coolant boundary, the actual break area must be represented. The reaction forces are a function of area changes and direction changes of the fluid along its flow path due to a leak path from the system boundary. The model for flow volumes will require the definition of the break area (leak area) for a time-history calculation of the forcing functions.

The solution for the forcing functions will require iteration to obtain a solution based on consistent conditions (break area and dynamic response). The results will be obtained in the form of area versus time.

3.15 Fuel assembly deformation limits will be established based upon allowable grid deformations as determined by ECCS requirements. These established requirements will be confirmed by analysis to assure that peak cladding temperatures do not exceed those allowed by 10CFR50.46.

Additional analyses may be necessary if the actual deformations exceed the established values.

3.16 A core evaluation model will be developed to simulate the fuel assembly interaction during dynamic excitation. The model will consider gaps that exist between inner assemblies and between outer assemblies and the baffle wall. Available experimental test information such as spacer grid dynamic properties, damping, and fuel assembly frequencies will be used as input to the core evaluation model.

This model in conjunction with the coolable geometry criteria will be used in evaluating the fuels coolable geometry.

3.17 Using the loading generated (as outlined earlier), the components identified in Paragraph 2.1 will be evaluated. In addition, integrity of the cavity walls will be evaluated when subjected to the effects of asymmetric pressures.

Due to the very limited time available for component evaluations, complete structural analyses for these components will not be performed and stress reports will not be prepared.

However, the components will be evaluated using applied loads and the resultant stresses compared to material capabilities in critical areas of the structures. Based upon the results, conclusions will be drawn with respect to the structural integrity of the affected components, structures, and the coolable geometry of the fuels and core.

4.0 COMPUTER CODES

In the performance of the analyses, several different computer codes will be used. The following list identifies the major codes to be employed:

a. ANSYS
b. ADINA
c. ST3DS
d. LUMS
e. STARS
f. CRAFT2
g. SUPERPIPE
h. GDSGAP
i. PWHIP
j. STALUM
k. FESAP
l. HYDROE
m. NASTRAN

5.0 APPLICABLE B&W TOPICAL REPORTS

Techniques described in topical reports submitted to the NRC by the B&W Company will be used in the evaluation. These topical reports are:

a. BAW-10131 — Reactor Coolant System Structural Analysis

b. BAW-10127 — LOCA Pipe Break Criteria for the Design of Babcock & Wilcox Nuclear Steam Systems

c. BAW-10132 — Analytical Methods Description — Reactor Coolant System Hydrodynamic Loadings During a Loss-of-Coolant Accident

d. BAW-10133 — Mark C Fuel Assembly — LOCA-Seismic Analyses

e. BAW-10060 — Reactor Internals Design/Analysis for Normal, Upset and Faulted Conditions

f. BAW-10104 — B&W's ECCS Evaluation Model

6.0 PHASE 2 SCHEDULE

No detailed schedules are included with this report because of the complex interactions required for the analyses described herein. However, at this time, it is projected the conclusion reports will be available by April, 1980.