B&W 177FA OWNERS GROUP

ASYMMETRIC LOCA LOADS EVALUATIONS PROGRAM

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

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1.0 INTRODUCTION

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

The plan described herein is separated into three phases. Each phase is described to the level of detail possible at this time. The phasing is intended to allow progression toward a completed assessment by providing for intermediate evaluations as the program proceeds.

This plan is based upon the understandings achieved in a meeting between the B&W Owners Group and NRC/DOR on March 31, 1978.

2.0 EVALUATION BASES

2.1 All components listed in Enclosure 2 of Reference (b) will be addressed for the LOCA breaks evaluated. These include:

a. Reactor Vessel
b. Fuel Assemblies, Including Grid Structures
c. Control Rod Drives
d. ECCS Piping attached to the Primary Coolant Piping
e. Reactor Coolant System Piping
f. Reactor Vessel, Steam Generator and Pump Supports
g. Reactor Internals
h. Reactor Cavity Shield Wall and Neutron Shield Tank
i. Steam Generator Sub-compartment Wall

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 (1) maintenance of a coolable core geometry and (2) mitigation of the consequences of an accident.

2.3 Jet impingement effects will be evaluated for breaks analyzed. This evaluation was not explicitly stated in the NRC letter, but was identified as a requirement in the March 31, 1978, meeting mentioned in paragraph 1.0.

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 If results of the evaluation indicate loads leading to inelastic action or displacements exceeding previous design limits, then inelastic behavior (including strain hardening) of the material analyzed and the

2.0 EVALUATION BASES (continued)

effect on the load transmitted to the backup structures to which the component is attached will be included.

2.6 Where applicable, a generic review of the B&W Owners Group plants will be used. The categorization timing and extent will be discussed later in this report.

3.0 WORK PLAN (PHASES)

3.1 Phase 1 will be a seven month preliminary assessment. The specific plant drawings will be reviewed to assess whether asymmetric pressures can be applied to similar plants in each category.

3.1.1 A preliminary scoping study of each plant's restraint design will be performed. The results of this study will provide estimated maximum pipe break opening areas for each of four breaks (upper cold leg and hot leg guillotine at the Reactor Vessel nozzle and upper cold leg and hot leg guillotine outside the reactor cavity shield wall). The location of the break outside the reactor cavity shield wall will be determined with acceptable break location criteria. Design cases will then be selected based on parametric studies performed by B&W on their 205FA plants as compared to the 177FA plants.

3.1.2 The peak magnitudes of the major LOCA load components acting on the reactor internals will be estimated as a function of break size. Sensitivity study results which are available for B&W 205FA plants will be used to develop scaling factors for estimating loads on the 177FA plants. The particular loads which will be considered are (1) total lateral force on the core support cylinder; (2) total vertical force on the reactor vessel due to head differential pressure; and (3) vertical force on the core. These loads will be estimated for the four breaks described in paragraph 3.1.1.

3.1.3 The magnitude of the peak lateral force which acts externally on the reactor vessel due to asymmetric pressures within the reactor cavity will be estimated. These estimates will be extrapolations made from existing 177 cavity pressure data to include a consideration of break size.

3.1.4 The applied loadings and the load carrying capability of the Reactor Internals and the Reactor Vessel support for each plant will be compared using the estimated asymmetric cavity and internals pressures determined in paragraphs 2.1.3 and 3.1.3. Based on this comparison, additional analyses and/or hardware modifications will be recommended.

3.0 WORK PLAN (PHASES) (continued)

3.2 Phase 2 analysis will be initiated if results of Phase 1 indicate a need for more detailed review and/or a need for more detailed review and/or a need to review some of the plants on a specific case basis. The extent of analysis cannot be specified until the results of Phase 1 are known.

During this phase, one, or a combination, of the following three action paths will be pursued:

a. Detailed Analyses
b. Hardware Modifications
c. Licensing Actions

As in Phase 1, this phase will focus on the Reactor Vessel and structures/components in close proximity.

If the results of Phase 1 are acceptable, conclusive and defendable, this phase will not be executed. If it is required to progress on to this phase, an additional detailed plan with schedules will be submitted to the Commission.

3.3 Whereas Phase 2 concentrates on the Reactor Vessel area, Phase 3 will focus on the Steam Generator and R.C. Pump areas, Phase 3 analysis will be initiated only if the results of Phase 1 indicate a need for a more detailed review.

Here again, there exists the possibility of three courses of action, as outlined in paragraph 3.2, and until the specific needs are identified from Phase 1 efforts, the details of this phase cannot be identified. If it is required to execute this phase, an additional detailed plan with schedules will be submitted to the Commission.

4.0 COMPUTER CODES

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

a. ANSYS
b. ADINA
c. ST3DS
d. LUMS
e. STARS
f. C RAFT2
g. RELAP4

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:

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5.0 APPLICABLE B&W TOPICAL REPORTS (continued)

a. BAW-10131 - Reactor Coolant System Structural Analysis

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

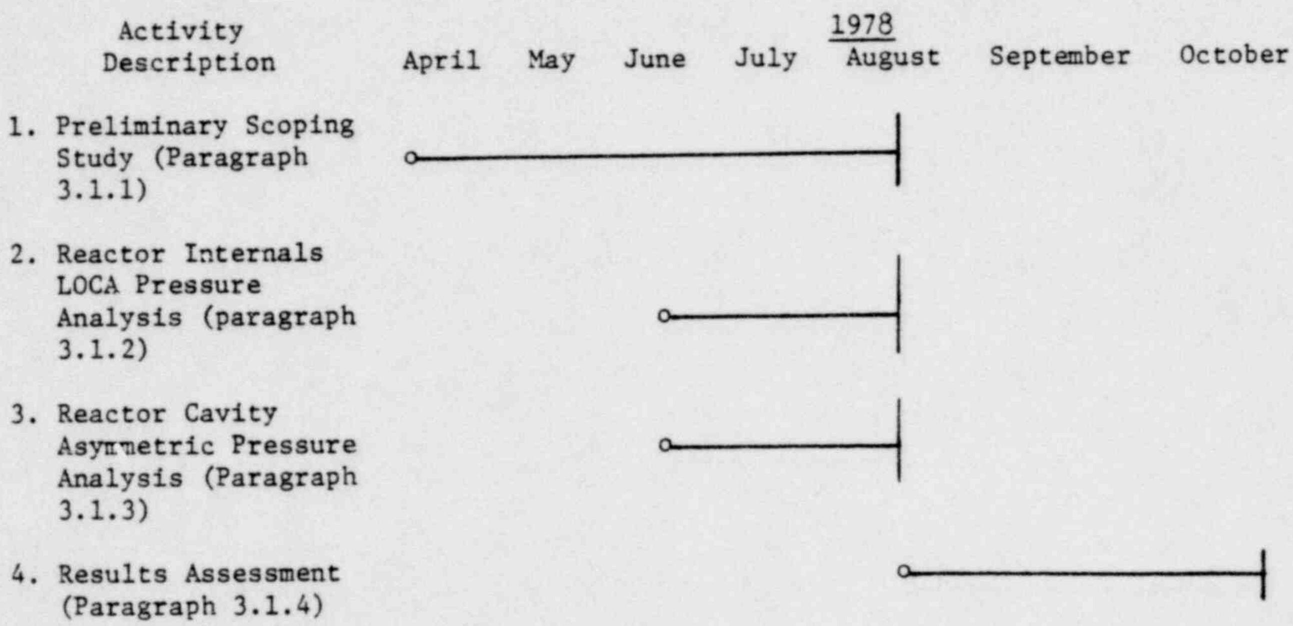
c. BAW-10132 - Analytical Methods Description - Reactor Coolant System Hydrodynamic Loadings During a Loss-of-Coolant Accident

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

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

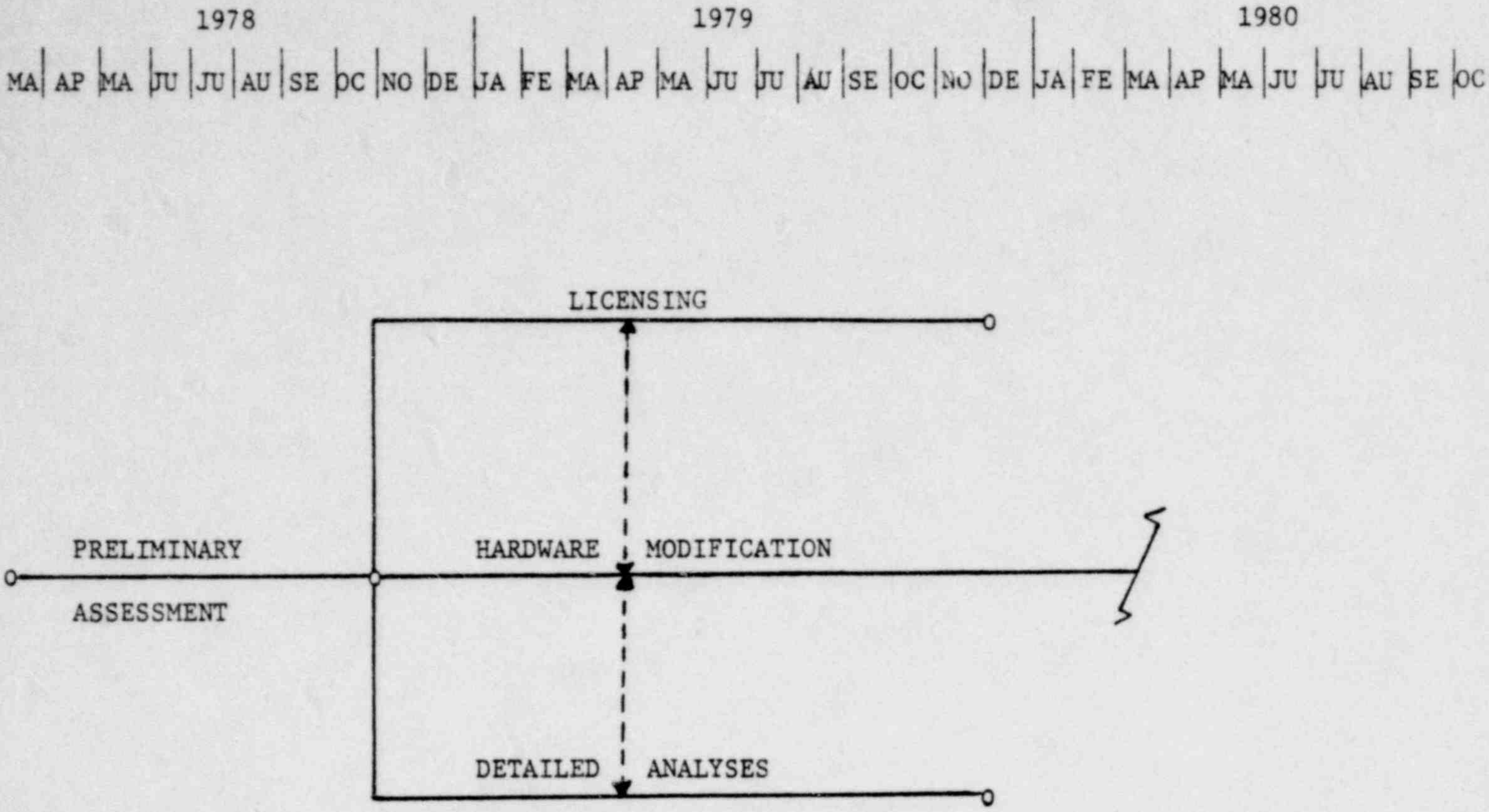
6.0 PLAN SCHEDULES

6.1 Phase 1 schedule is as follows:



6.2 Phase 2 and 3 schedules cannot be considered firm until specific detail needs are known. However, the overall program schedule is as follows:

6.2 (continued)



6.3 As shown in paragraph 6.2, all analysis can probably be completed within the two year time frame discussed in the NRC letter. However, if hardware modifications are required, full implementation would exceed the two year time frame allowing for material procurement, fabrication, scheduled shutdowns and erection. The NRC will be kept advised of firm dates as they are determined.

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