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ARKANSAS POWER & LIGHT COMPANY

POST OFFICE BOX 551 LITTLE ROCK, ARKANSAS 72203 (501) 371-4000

May 16, 1978

1-058-12

Director of Nuclear Reactor Regulation
Attn: Mr. Victor Stello, Director
Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Arkansas Nuclear One-Unit 1
Docket No. 50-313
License No. DPR-51
Asymmetric LOCA Loads
(File: 1510)

Gentlemen:

In response to the request of your letter dated January 25, 1978, to provide you with a detailed schedule for our evaluation of asymmetric LOCA Loads on Arkansas Nuclear One-Unit 1, please find attached a report entitled, "B&W 177FA Owner's Group Asymmetric LOCA Loads Evaluation Program". The proposed schedule is consistent with your intentions to resolve this issue within two years.

As can be determine from the report, we plan to evaluate the asymmetric loads issue in three distinct phases, with the execution of phases 2 and 3 being dependent upon the results of phase 1. Only the details of the phase 1 evaluation have been included in the attached report. Should we find it necessary to proceed to phases 2 and 3, we will transmit detailed schedules and implementation plans to you in advance of executing these phases.

Please note that we are engaged in the requested evaluation as a participant of the B&W 177 Fuel Assembly Owner's Group and, where permissible, plan to take advantage of generic analyses. When doing so, we will provide justification for the generic grouping.

Very truly yours,

Daniel H. Williams

Daniel H. Williams
Manager, Licensing

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DHW/DM/ew

Attachment

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RESPONSE TO NRC LTR DTD 01/25/78 FOR DETAILED SCHEDULE FOR NRC EVALUATIONN OF ASYMMETRIC LOCA LOADS ON SUBJECT FACILITY...FORWARDING REPT ENTITLED: "B&W 177 FA OWNERS GROUP ASYMMETRIC LOCA LOADS EVALUATIONS PROGRAM", DTD 04/10/78.

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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

April 10, 1978

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CONTENTS

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 the NRC letter will be addressed for the LOCA breaks evaluated. This includes:

a. Reactor Pressure Vessel
b. Fuel Assemblies, Including Grid Structures
c. Control Rod Drives
d. ECCS Piping that is Attached to the Primary Coolant Piping
e. Primary Coolant Piping
f. Reactor Vessel, Steam Generator and Pump Supports
g. Reactor Internals
h. Biological Shield Wall and Neutron Shield Tank (where applicable)
i. Steam Generator Compartment Wall

2.2 LOCA analysis 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) maintaining core coolable geometry and (2) mitigating 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)

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

2.6 Where justifiable, 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 short duration (7 month) preliminary assessment. The specific plant drawings will be reviewed to assess the similarity of the various plants to assure that asymmetric pressures will be similar for all 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 be 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 primary shield wall). The location of the break outside the primary shield wall will be determined with acceptable break location criteria and from these, design cases will be chosen based on parametric studies performed by B&W on their 205FA plants and a results comparison for these plants under evaluation.

3.1.2 The peak magnitudes of the major LOCA load components acting on the reactor internals will be estimated as a function of breaksize. 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 Estimates for the magnitude of peak lateral force which acts externally on the reactor vessel due to asymmetric pressures within the reactor cavity will be made. These estimates will be extrapolations made from existing 177 cavity pressure data to include a consideration of break size.

3.1.4 Using the estimated, asymmetric cavity and internals pressures determined in paragraphs 2.1.3 and 3.1.3, a comparison between the applied loadings and the load carrying capability of the Reactor Internals and the Reactor Vessel support for each plant will be made. Based on this comparison, additional analysis and hardware modifications will be recommended.

3.0 WORK PLAN (PHASES) (continued)

3.2 Phase 2 analysis will be initiated if results of Phase 1 indicates 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 Ph[illegible] this phase will focus on the Reactor Vessel and structu[illegible]mponents 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 NRC.

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

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 NRC.

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. CRAFT2

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:

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

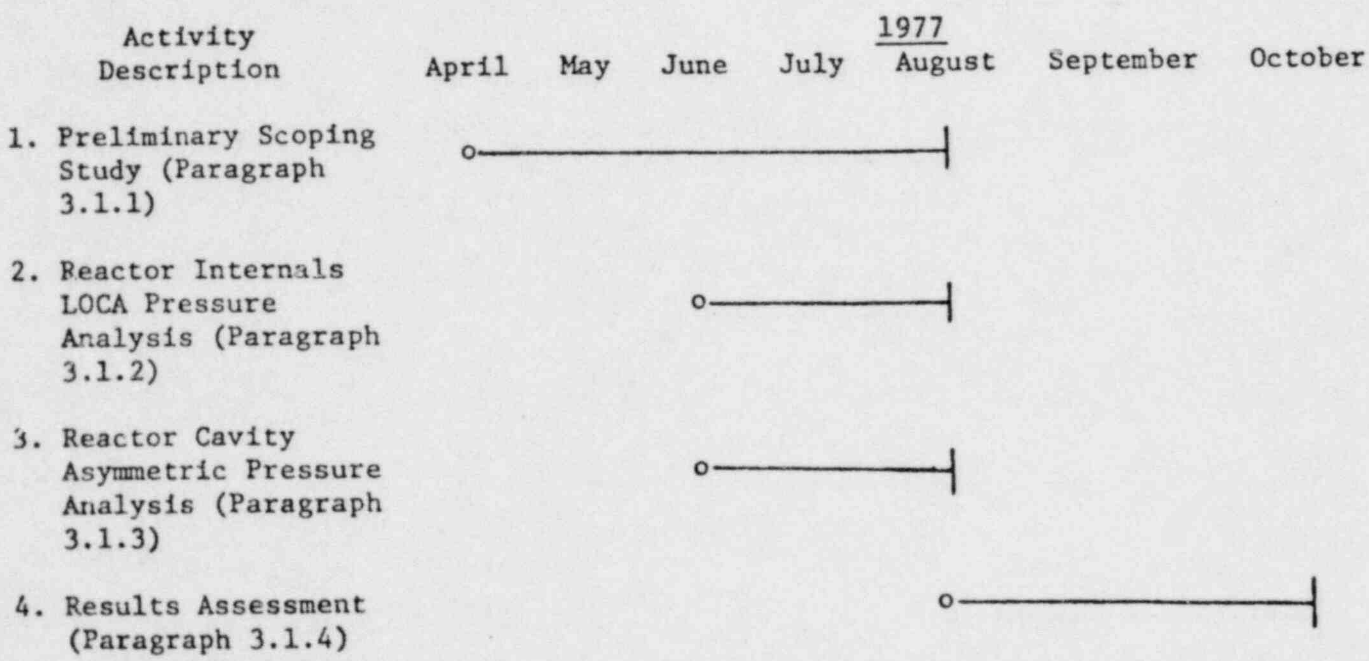
c. BAW-10132 - Analytical Methods Description - Reactor Coolant System Hydrodynamic Loadings During a Loss-of-Coolant Accident

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

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

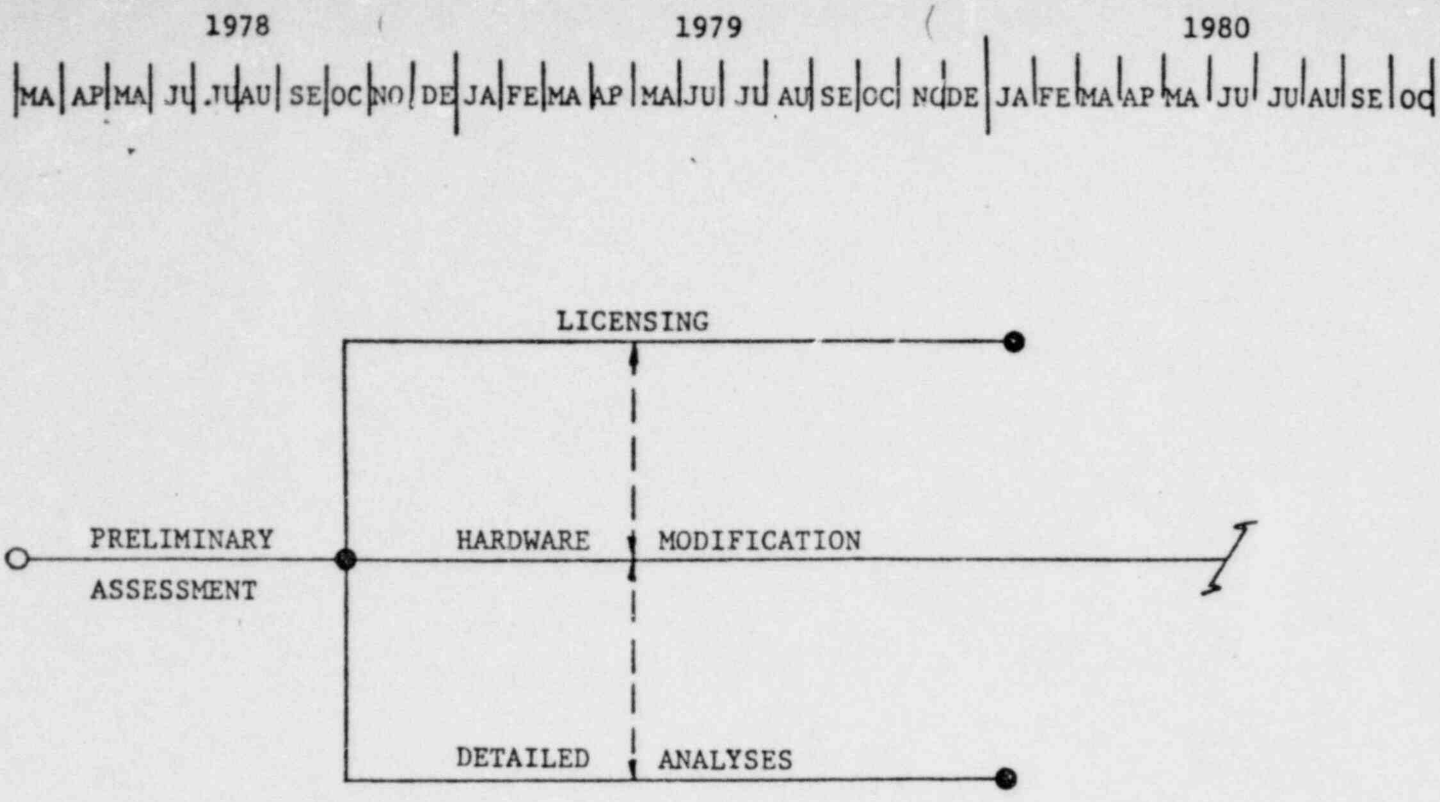
6.0 PLAN SCHEDULES

6.1 Phase 1 schedule is as follows:



6.2 Phases 2 and 3 schedules cannot be firmed up 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 fixes are required, full implementation of all fixes 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.