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TOLEDO EDISON COMPANY
ASYMMETRIC LOCA LOADS EVALUATIONS PROGRAM
FOR DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO. 1

CONTENTS

1.0 INTRODUCTION

This report summarizes the detailed plan prepared by the Toledo Edison Company 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

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

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 effect of preliminary asymmetric pressures.

3.1.1 A preliminary scoping study of the 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 205 FA plants and a results comparison for the Davis-Besse 1 plant.

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 205 FA and Davis-Besse 2 and 3 plants will be used to develop scaling factors for estimating loads on Davis-Besse 1. 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 Davis-Besse 2 and 3 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 Davis-Besse 1 will be made. Based on this comparison, additional analysis and hardware modifications will be recommended.

3.2 Phase 2 analysis will be initiated if results of Phase 1 indicates a need for more detailed review. 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 . .alyses
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 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 RC 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 wi h 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:

a. BAW-10131 – Reactor Coolant System Structural Analysis

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

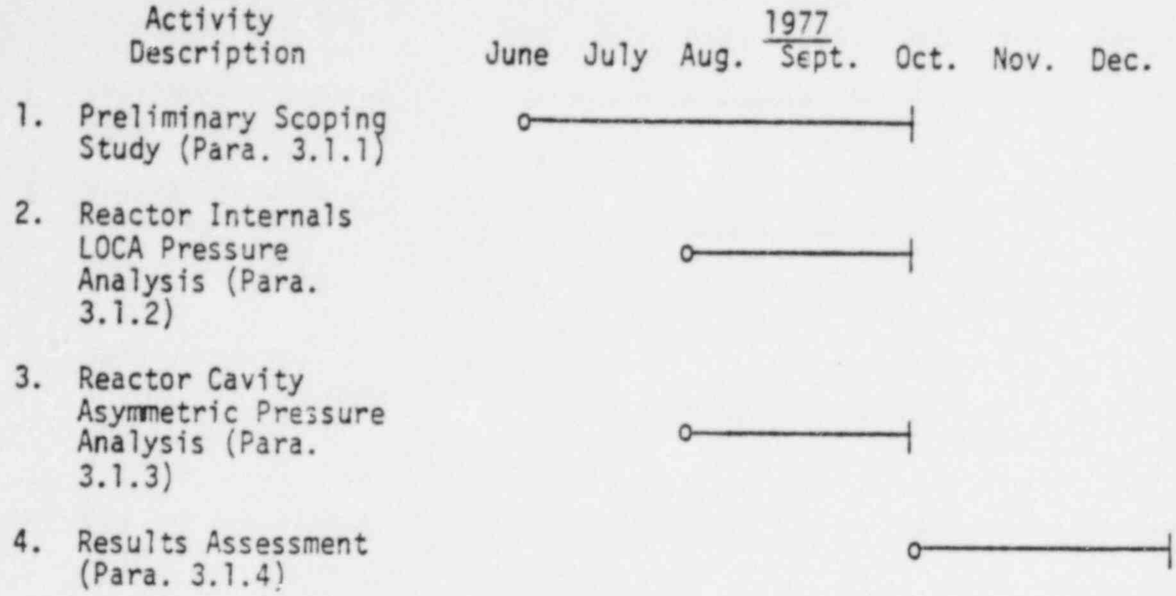
c. BAW-10132 – Analytical Methods Description – Reactor Coolant System Hydrodynamic Loadings During a Loss-of-Coolant Accident

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

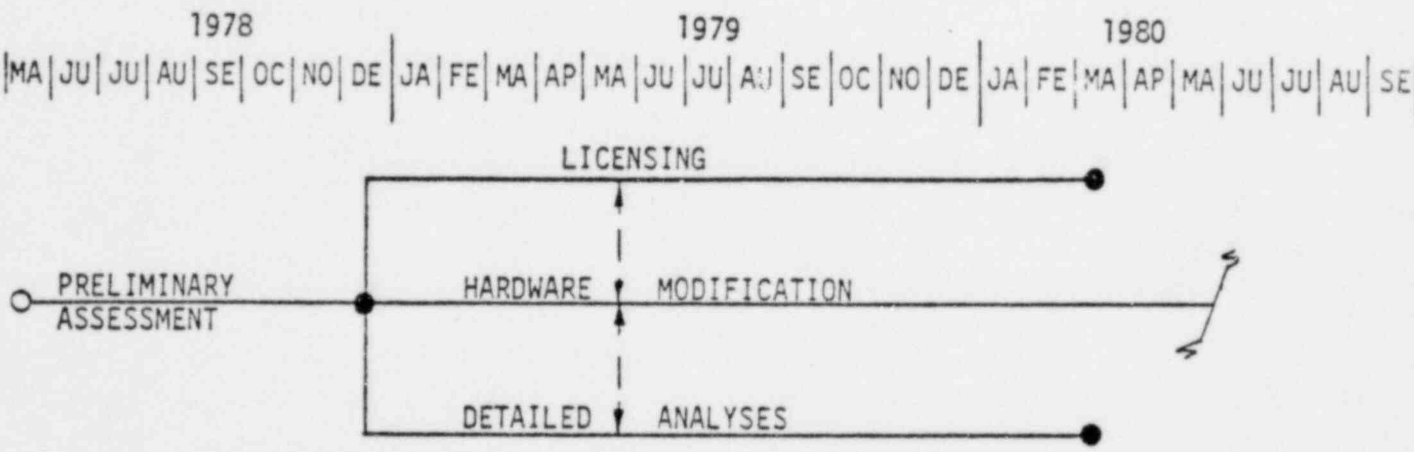
e. BAW-10060 – Reactor Internals Design/Analysis for Normal, Upset and Faulted Conditions.

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:



6.3 As shown in Paragraph 6.2, all analysis can probably be completed within approximately 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.