Consumers Power Company

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General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

August 8, 1978

Director, Office of Nuclear Reactor Regulation
Att: Mr H R Denton
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKETS 50-317/318, CALVERT CLIFFS 1 & 2;
50-255, PALISADES; 50-285, FORT CALHOUN;
50-336, MILLSTONE 2 - COMBUSTION ENGINEERING
OWNERS' GROUP PROPOSED PLAN TO RESOLVE THE
ASYMMETRIC LOCA LOADS ISSUE

The Combustion Engineering Owners' Group (CEOG), comprised of Baltimore Gas & Electric, Consumers Power Company, Northeast Utilities, Omaha Public Power District and others, have been actively pursuing since late 1975 a solution to NRC concerns relative to "Asymmetric LOCA Loads." Through sponsorship of the CEOG, Science Application, Inc (SAI) was contracted to evaluate the issue. SAI has produced a topical report entitled "An Analysis of the Relative Probability of Pipe Rupture at Various Locations in the Primary Cooling Loop of a PWR Including Effects of a Periodic Inspection." This report was submitted to the NRC in September 1976. The conclusion in the report was that asymmetric LOCA loads are of such low probability that specific analyses are not required. We agree with this conclusion but, in response to your January 25, 1978 letter to all PWR licensees, we have prepared the attached plan for conducting an evaluation of reactor coolant system components and supports subjected to asymmetric pressure loads following postulated pipe ruptures. The plan has been developed by Combustion Engineering for the CEOG for the evaluation of Calvert Cliffs 1 & 2, Palisades, Millstone 2 and Fort Calhoun.

Summary of Evaluation Plan

The proposed plan evaluates the forces on components as a result of postulated pipe ruptures, the distribution of reactions among components supports, and the effect of component motion on component internals and attachments. A comparison of calculated stresses and deformation with acceptance criteria is included in the plan. The acceptance criteria are presented only as bases for evaluating the results of the analyses. As the evaluation progresses, alternate criteria and/or special or further analyses may be indicated.

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The methods and computer codes proposed to be used for the evaluation are either approved or under active review. Also, Combustion Engineering has extensive experience with these methods and codes which have been and are being employed in the evaluation of reactor coolant system components and supports of more recent nuclear plants.

The possibility of utilizing models and/or analytical results developed for a particular plant for the evaluation of components or supports for another plant has also been studied. The geometrical, thermal-hydraulic and physical characteristics of reactor coolant system components and supports for the plants to be evaluated have been compared. As a result of these comparisons, we propose to perform generic evaluation of components, supports and foundations; and, to utilize the results of the generic evaluation to assess the effects of asymmetric loads on specific plants.

The scope of the proposed evaluation is comprehensive. It begins with pipe rupture definition for postulated pipe ruptures at the steam generator and reactor vessel nozzles. The speed and extent of flow area opening following the appearance of the postulated crack will be calculated. Reactor coolant system characteristics, such as stiffness and component supports will be considered in the calculation.

Hydraulic flow inside and outside of the reactor coolant system will be examined. Effects of flow within the system (internal asymmetric pressures) and dispersion of the fluid within the containment (subcompartment pressures) will be computed using methods previously submitted for NRC review. The time histories of loads due to thrust, internal asymmetric pressure and subcompartment pressures will be used in a mathematical representation of the reactor coolant system structure. This representation will consist of shell model lumped mass- and beam-element models which include the effect of gaps and other nonlinear characteristics of components and supports. The resultant support reactions and/or deformations will be compared to acceptance criteria to provide evaluation of supports. The resultant motions will be applied to structural representations of the control element drive mechanisms and ECCS piping to provide stresses and deformations for evaluation. The motions of the reactor vessel and the vessel internal asymmetric pressures will be applied to a structural model of the vessel internals to provide loads on the internals and motions for subsequent analysis of the fuel.

Scheduler Aspects of Evaluation Plan

At a meeting with the staff on June 21, 1978, several key points were made by the staff. We believe these can be summarized by:

1. The analysis should be generic where possible, as if it can be shown that the specific plant is bounded by the generic analysis, a plant specific analysis is not required.

2. The staff wants all analysis completed within two years (from January 25, 1978).

3. The staff will not require that analysis be started prior to receiving the staff's approval of techniques.

Should plant specific analysis be required to demonstrate the adequacy of one or more plants, the schedule for these plants may be extended.

Further, as shown on Figure 1, we believe that utilizing this generic approach will allow completion in 19 months. We anticipate starting on or about August 15, 1978 and expect to be completed in early 1980.

Since it is your stated intention to approve the calculational techniques, your assistance will be required to maintain our schedule. You will note that identified on the schedule are specific tasks that can be started without your approval. We will start these. Other tasks are noted as requiring your approval For these tasks, the date we need approval is also shown.

We believe that the evaluation plan and references contained therein provide all the necessary information for your review and approval of specific tasks as noted on the schedule. Please note that your early approval of CE FLASH 4B is required. We suggest that you may want to approve CE FLASH 4B only for the asymmetric loads analysis for operating plants.

Appendix B of the plan provides the justification for not combining the effects of pipe rupture and seismic loads in the evaluation of components and supports of operating plants for the effects of asymmetric pressure loads.

It is the intention of the CEOG to have periodic progress meetings during the course of the analysis. These meetings could be initiated at either NRC's or our request.

Further, we request that the NRC staff provide an early indication as to the acceptability of the evaluation plan to resolve the "Asymmetric LOCA Load" issue.

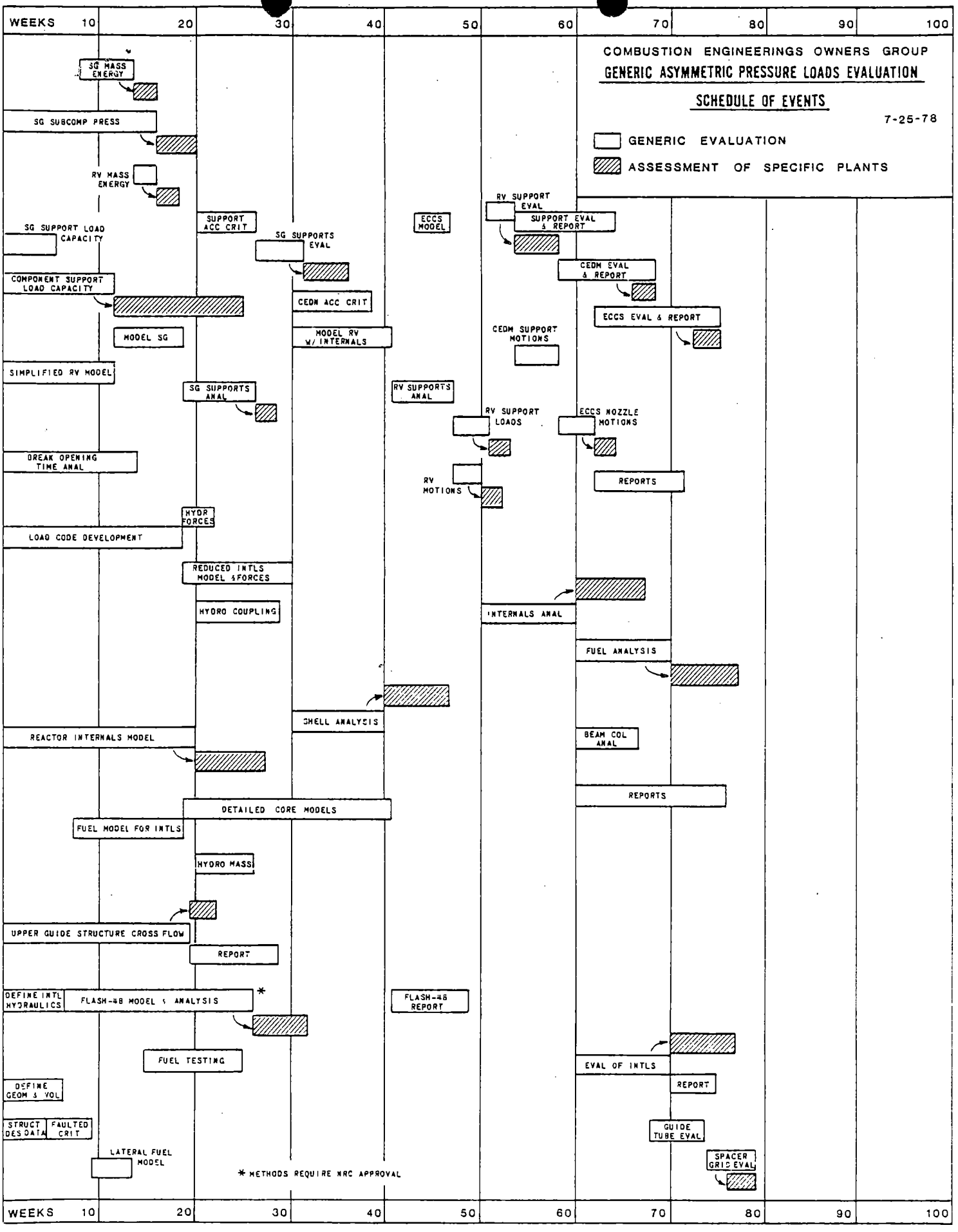
David P Hoffman

David P Hoffman
Assistant Nuclear Licensing Administrator

CC: SBHosford, Engineering Branch, Division of Operating Reactors
DLZiemann, Chief, Operating Reactors, Branch No 2, Division of Operating Reactors (Palisades)
RWReid, Chief, Operating Reactors, Branch No 4, Division of Operating Reactors (Calvert Cliffs 1 & 2, Millstone 2, Fort Calhoun)

See rpts

FIGURE 1


COMBUSTION ENGINEERINGS OWNERS GROUP
GENERIC ASYMMETRIC PRESSURE LOADS EVALUATION
SCHEDULE OF EVENTS
7-25-78
GENERIC EVALUATION
ASSESSMENT OF SPECIFIC PLANTS
WEEKS 10 20 30 40 50 60 70 80 90 100
SG MASS ENERGY
SG SUBCOMP PRESS
RV MASS ENERGY
SG SUPPORT LOAD CAPACITY
SUPPORT ACC CRIT
SG SUPPORTS EVAL
ECCS MODEL
RV SUPPORT EVAL
SUPPORT EVAL & REPORT
COMPONENT SUPPORT LOAD CAPACITY
CEDM EVAL & REPORT
CEDM ACC CRIT
ECCS EVAL & REPORT
MODEL SG
MODEL RV W/ INTERNALS
CEDM SUPPORT MOTIONS
SIMPLIFIED RV MODEL
SG SUPPORTS ANAL
RV SUPPORTS ANAL
RV SUPPORT LOADS
ECCS NOZZLE MOTIONS
BREAK OPENING TIME ANAL
REPORTS
RV MOTIONS
HYDR FORCES
LOAD CODE DEVELOPMENT
REDUCED INTLS MODEL & FORCES
HYDRO COUPLING
INTERNALS ANAL
FUEL ANALYSIS
SHELL ANALYSIS
REACTOR INTERNALS MODEL
BEAM COL ANAL
REPORTS
DETAILED CORE MODELS
FUEL MODEL FOR INTLS
HYDRO MASS
UPPER GUIDE STRUCTURE CROSS FLOW
REPORT
DEFINE INTL HYDRAULICS
FLASH-48 MODEL & ANALYSIS *
FLASH-48 REPORT
FUEL TESTING
EVAL OF INTLS
REPORT
DEFINE GEOM & VOL
STRUCT DES DATA
FAULTED CRIT
GUIDE TUBE EVAL
LATERAL FUEL MODEL
SPACER GRID EVAL
* METHODS REQUIRE NRC APPROVAL
WEEKS 10 20 30 40 50 60 70 80 90 100