UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 30 1978

Docket Nos: 50-329
50-330

APPLICANT: Consumers Power Company

FACILITY: Midland Plant, Units 1 & 2

SUBJECT: SUMMARY OF MAY 23, 1978 MEETING ON STAFF'S REQUESTS ON ASYMMETRIC LOCA LOADS

On May 23, 1978, the NRC staff met in Bethesda, Maryland with Consumers Power Company (CPCO), Bechtel Associates and the Babcock & Wilcox (B&W) Company. Attendees are listed in Enclosure 1.

CPCO requested the meeting to advise the staff of their program for analyzing asymmetric LOCA loads and to determine what further requests would be made by the staff in this area. CPCO's program activities, schedule and criteria are shown in Enclosure 2, 3 and 4.

The analyses involve the use of the B&W computer code CRAFT 2 which is being reviewed by the staff as Topical Report BAW 10132, "Reactor Coolant System Hydrodynamic Loading During LOCA." The staff's review completion schedule for this report is September 1978. The staff expressed concern that this completion date is not compatible with the schedule for issuance of staff positions (second-round questions) for Midland in August 1978 and that adjustments would be necessary.

CPCO stated that most of the staff's requests to date for plants not yet operating have primarily focused on the vessel support design for transient differential pressures in the annular region between the reactor vessel and the cavity shield wall and across the core barrel. However, letters to PWR licensees from Victor Stello, Jr. dated January 25, 1978 show that other areas in the nuclear steam supply system are of concern to the staff. CPCO requested that these requests be made in a timely manner relative

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to the Midland schedule. The staff confirmed that the potential for damage to other NSSS component supports, fuel assemblies, control rod drives, and ECCS piping attached to the reactor coolant system also required analyses and provided CPCO a draft copy (Enclosure 5) of the staff request being developed to this end.

Darl Hood

Darl Hood, Project Manager
Light Water Reactors Branch No. 4
Division of Project Management

Enclosures:
1. Attendees List
2. Schedule for Plant Design Activities
3. Flow Chart
4. Comparison List
5. Revised Request for Additional Information

cc: See next page

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Consumers Power Company

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ENCLOSURE 1

ATTENDEES

May 23, 1978

NRC

D. Hood
F. Cherny
A. Hafiz
T. Green

Consumers Power Company

R. Bowman
J. Zabritski

Babcock & Wilcox

R. Reyns
C. Mahaney
W. Speight

Bechtel

D. Tooker

ENCLOSURE 2

CONSUMERS POWER CO.
MIDLAND II
DESIGN ANALYSIS
SCHEDULE FOR PLANT DESIGN ACTIVITIES

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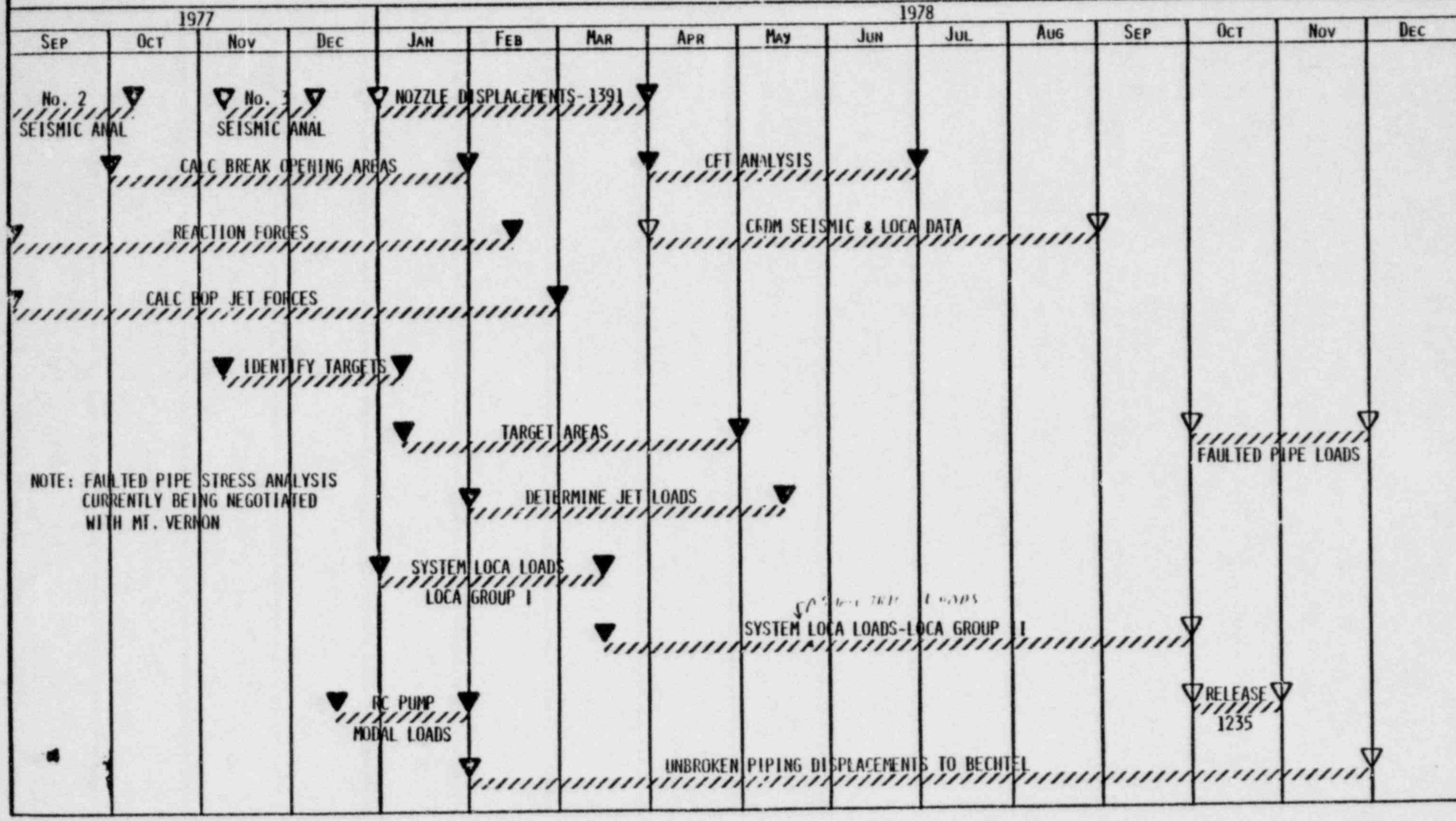

ENCLOSURE 3

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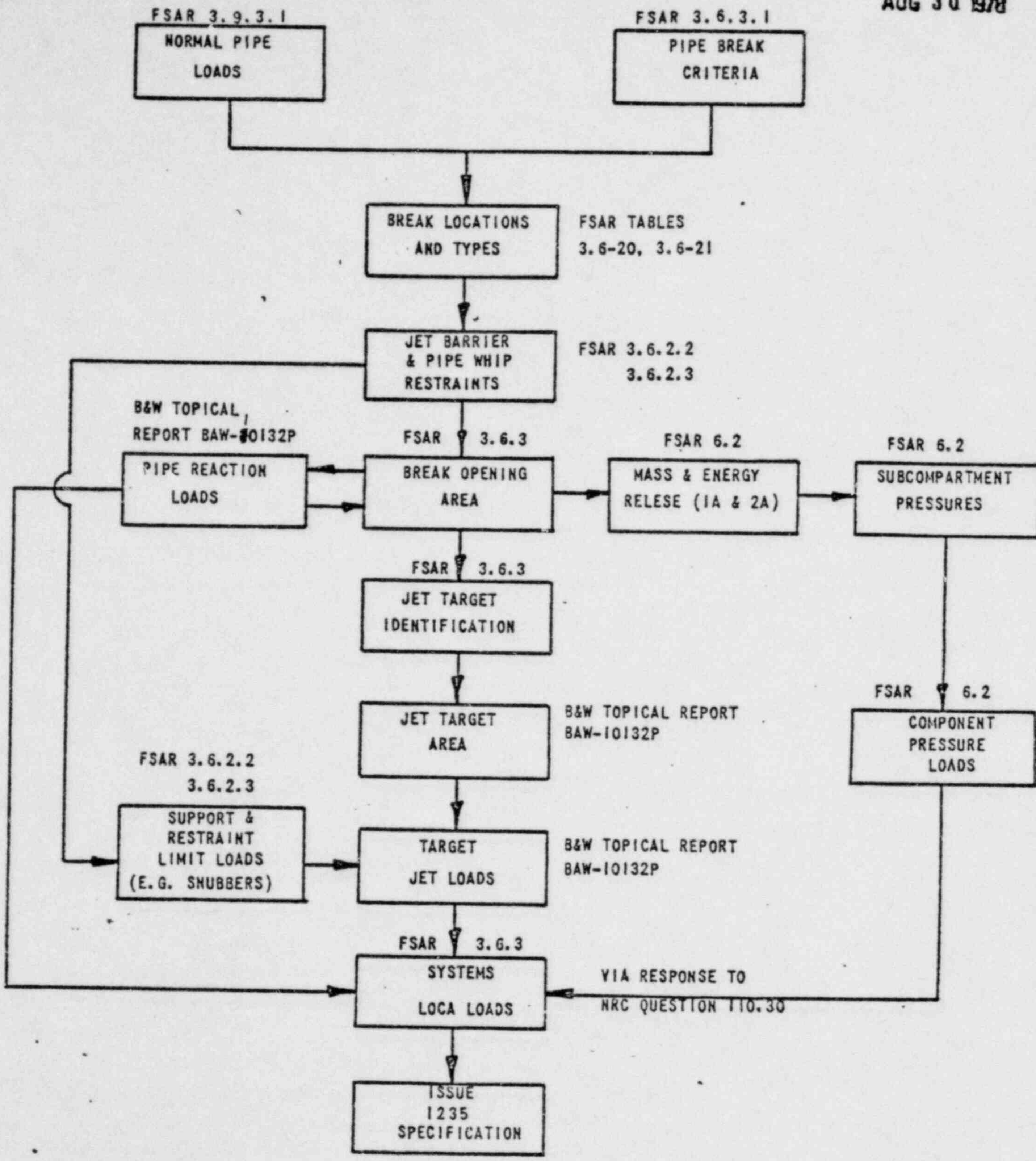

ENCLOSURE 4

COMPARISON OF OPERATING PLANT AND MIDLAND ASYMMETRICAL LOADING CRITERIA

NRC/DOR Letter Requirements	Corresponding Midland Requirements from NRC Questions
A. PROBABILITY STUDY	
Not completely without merit to allow continued operation and may be an alternate to hardware modifications.	Unacceptable and requires detailed analysis requested in Question 022.2
B. POSTULATED BREAK LOCATIONS	
1. RV Safe Ends	1. Reactor vessel hot and cold leg nozzle to piping terminal ends.
2 2. Pump Discharge Nozzle	2. RCP suction and discharge nozzles to piping terminal ends.
3. Pump Discharge Nozzle	3. Steam generator inlet and outlet nozzle to piping terminal ends.
3. Crossover Leg	4. Pressurizer surge and spray lines.
	5. Core flood tank lines.
	6. Other high energy lines.
C. LOADING COMBINATIONS	
Decision not made.	LOCA + SSE
D. INFORMATION TO BE PROVIDED	
Specified on pages 4, 5 and 6 of Enclosure 2 of NRC/DOR letter to PWR Licensees.	Same as NRC/DOR letter (see Question 022.2) except NRC/DOR letter nodalization study requirements more detailed. Also certain technical requirements of Appendix 110-1.
E. ANALYSIS GUIDELINES	
1. Items 4 through 8 (pg 2) of Enclosures 2 of NRC/DOR and items 1 and 2 (pg 3) of same enclosure	None specified.
2. Components requiring evaluation. Enclosure 2, page 1, items a through i.	Not specified.

Note: Detailed analysis techniques, computer codes, and topical reports to be used discussed with NRC/DOR on March 31, 1978.

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ENCLOSURE 5

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REVISED REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. For the purpose of this request for additional information the reactor system components that require reassessment shall include:

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 and Pump Compartment Wall

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the reactor cavity structure.

1. Provide arrangement drawings of the reactor vessel, the steam generator and pump support systems to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural mechanical and thermal hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

   For PWR
   a) Reactor Vessel hot and cold leg nozzle to piping terminal ends.
   b) Pump suction and discharge nozzles to piping terminal ends.
   c) Steam generator inlet and outlet nozzles to piping terminal ends.[1/]

[1/] Postulated steam line breaks may control the design of certain steam generator supports and, therefore, must also be considered in support design.

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For BWR

a) Steam line nozzles to piping terminal ends.
b) Feedwater nozzle to piping terminal ends.
c) Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

Provide an assessment of the effects of asymmetric pressure differentials[2/] on these systems/components in combination with all external loadings including safe shutdown earthquake loads, asymmetric cavity pressurization for both the reactor vessel, steam generator, and reactor coolant pump which might result from the required postulate. This assessment may utilize the following mechanistic effects as applicable:

a. limited displacement break areas
b. fluid-structure interaction
c. actual time-dependent forcing function
d. reactor support siffness
e. break opening times.

4. If the results of the assessment required by 3 above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits provide an evaluation of the following:

   a. Inelastic behavior (including strain hardening) of the material used in the system design and the effect on the load transmitted to the backup structures to which these systems are attached.

5. For all analysis performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Provide an estimate of the total amount of permanent deformation sustained by the fuel spacer grids. Include a description of the impact testing that was performed in support of your estimate. Address the effects of operating temperatures, secondary impacts, and irradiated material properties (strength and ductility) on the amount of predicted deformation. Demonstrate that the fuel will remain coolable for all predicted geometries.

7. Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.

8. Demonstrate functionability of any essential piping where service level B limits are exceded.

[2/] Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

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In order to review the methods employed to compute the asymmetrical pressure differences across the core support barrel during subcooled portion of the blowdown analysis, the following information is requested:

1. A complete description of the hydraulic code(s) used including the development of the equations being solved, the assumptions and simplifications used to solve the equations, the limitations resulting from these assumptions and simplifications and the numerical methods used to solve the final set of equations. Provide comparisons with experimental data, covering a wide range of scales, to demonstrate the applicability of the code and of the modeling procedures to the subcooled blowdown portion of the transient. In addition, discuss application of the code to the multi-dimensional aspects of the reactor geometry.

   If an approved vendor code is used to obtain the asymmetric pressure difference across the core support barrel, state the name and version of the code used and the date of the NRC acceptance of the code.

2. If the assessment of the asymmetric pressure difference across the core support barrel is made without the use of a hydraulic blowdown code, present the methodology used to evaluate the asymmetric loads and provide justification that this assessment provides a conservative estimate of the effects of the postulated LOCA.

A compartment multi-node, space-time pressure response analysis is necessary to determine the external forces and moments on components. Analyses should be performed to determine the pressure transient resulting from postulated hot leg and cold leg reactor coolant system pipe ruptures within the reactor cavity and any pipe penetrations. If applicable, similar analyses should be performed for steam generator and reactor coolant pump compartments that may be subject to pressurization where significant component support loads may result. This information can be provided to encompass a group of similarly designed plants (generic approach) or a purely plant specific (custom plant) evaluation can be developed. In either case, the proposed method of evaluation and principal assumptions to be used in the analysis should be provided for review in advance of the final load assessment.

For generic evaluations, perform a survey of the plants to be included and identify the principle parameters which may vary from plant to plant. For instance, this should include blowdown rate and geometrical variations in principle dimensions, volumes, vent areas, and vent locations. A typical or lead plant should be selected to perform sensitivity and envelope calculations. These analyses should include:

(1) nodal model development for the configuration representing the most restrictive geometry; i.e., requiring the greatest nodalization:

(2) the most restrictive configuration regarding vent areas and obstructions to flow should be analyzed; and

(3) sensitivity to code data input should be evaluated: e.g., loss

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coefficients, inertia terms, vent areas, nodal volumes, and any other input data where there may be variations from plant to plant or uncertainty for the given plant.

These studies should be directed at evaluating the maximum lateral and vertical force and moment time functions, recognizing that models may be different for lateral as opposed to vertical load definitions.

The following is the type of information needed for both generic and custom plant evaluations. Although this request was primarily developed for reactor cavity analyses it may be applied to other component subcompartments by general application.

(1) Provide and justify the pipe break type, area, and location for each analysis.

(2) For each compartment, provide a table of blowdown mass flow rate and energy release rate as a function of time for the break which results in the maximum structural load and for the break which was used for the component supports evaluation.

(3) Provide a schematic drawing showing the compartment nodalization for the determination of maximum structural loads, and for the component supports evaluation. Provide sufficiently detailed plan and section drawings for several views, including principal dimensions, showing the arrangement of the compartment structure, major components, piping, and other major obstructions and vent areas to permit verification of the subcompartment nodalization and vent locations.

(4) Provide a tabulation of the nodal net-free volumes and interconnecting flow path areas. For each flow path, provide an L/A (ft^{-1}) ratio, where L is the average distance the fluid flows in that flow path and A is the effective cross sectional area. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.

(5) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variation circumferentially, axially and radially within the compartment. The nodal model development studies should show that a spatially convergent differential pressure distribution has been obtained for the selected evaluation model.

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Describe and justify the nodalization sensitivity study performed for the major component supports evaluated, if different from the structural analysis model, where transient forces and moments acting on the components are of concern. Where component loads are of primary interest, show the effect of noding variations on the transient forces and moments. Use this information to justify the nodal model selected for use in the component supports evaluation.

If the pressurization of subvolumes located in regions away from the break location is of concern for plant safety, show that the selection of parameters which affect the calculations have been conservatively evaluated. This is particularly true for pressurization of the volume beneath the reactor vessel. In this case, a model which predicts the highest pressurization below the vessel should be selected for the evaluation.

NOTE: It has been our experience that for the reactor cavity, three regions should be considered (i.e., nodalized) when developing a total model. These are:

(1) the volume around or in the vicinity of the break location out to a radius approximated by the adjacent nozzles, and including protions of the penetration volume for some plants;

(2) the volume or region covering the upper reactor cavity, primarily the RPV nozzles other than the break nozzle; and

(3) the region encompassing the lower reactor cavity and other portions of the reactor cavity not included in Items (1) and (2)

(6) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification that vent areas will not be partially or completely plugged by displaced objects. Discuss how insulation for piping and components was considered in determining volumes and vent areas.

(7) Graphically show the pressure (psia) and differential pressure (psi) response as functions of time for a representative number of nodes to indicate the spatial pressure response. Discuss the basis for establishing the differential pressure on structures and components.

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(8) For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed, particularly for the reactor cavity as discussed above.

(9) Provide the peak and transient loading on the major components used to establish the adequacy of the support design. This should include the load forcing functions (e.g., $f_x(t)$, $f_y(t)$, $f_z(t)$) and transient moments (e.g., $M_x(t)$, $M_y(t)$, $M_z(t)$) as resolved about a specific, identified coordinate system. The centerline of the break nozzle is recommended as the X coordinate and the center line of the vessel as the Z axis. Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

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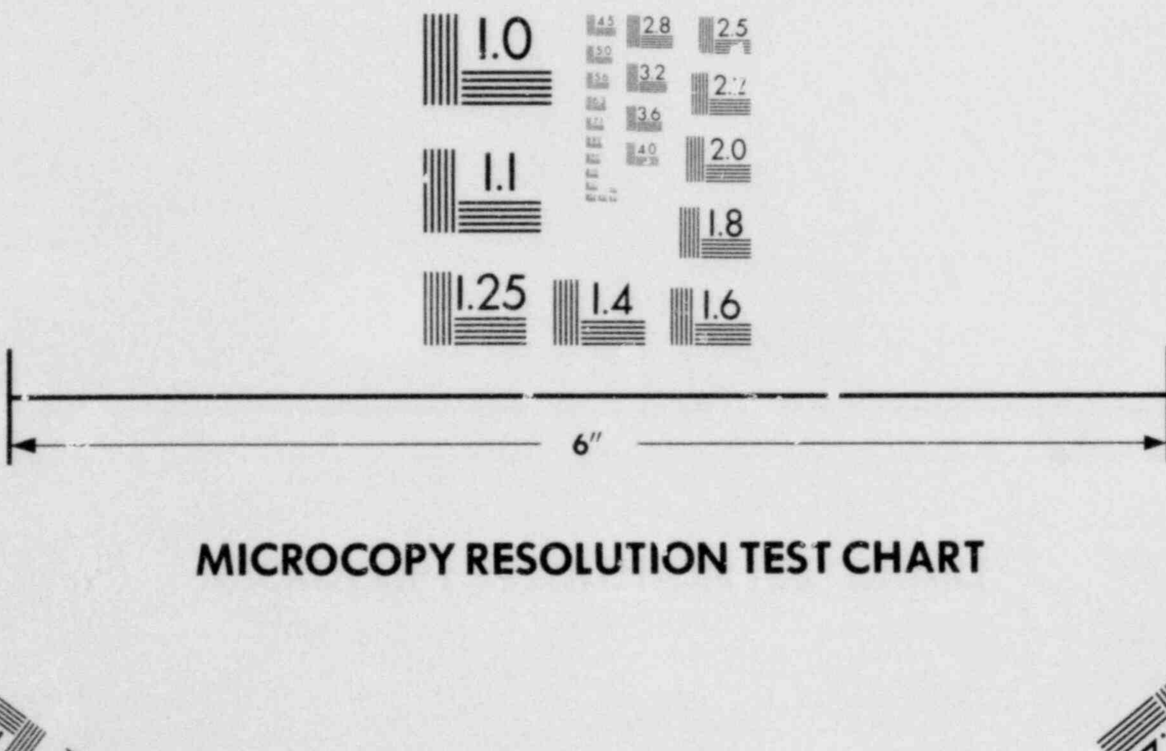
MEETING SUMMARY

AUG 30 1978

Docket File
NRC PDR
Local PDR
TIC
NRR Reading
LWR #4 File
E. Case
R. Boyd
~~R. DeYoung~~
D. Vassallo
J. Stolz
R. Baer
O. Parr
S. Varga
L. Crocker
D. Crutchfield
F. Williams
R. Mattson
R. DEYOUNG
~~D. Muller~~
Project Manager: D. HOOD
Attorney, ELD
M. Service
IE (3)
ACRS (16)
L. Dreher
S. Rubenstein
NRC Participants:

IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART