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R. C. DeYoung, Assistant Director
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REQUEST FOR ADDITIONAL INFORMATION FOR THE WPPSS 1 & 4 REACTOR CAVITY PRESSURE RESPONSE (TAR-3217); LOADS ON REACTOR VESSEL SUPPORT STRUCTURE FOR CERTAIN POSTULATED LOCA (TAR-3916)

The Mechanical Engineering Branch (MEB) has reviewed the information submitted by letter dated September 3, 1975 regarding the reactor cavity analysis for WPPSS 1 & 4. We have determined that we will need additional information before we can conclude on the acceptability of the analysis. Enclosed is a list for additional information. Our questions are predicated on the acceptability of the thermal hydraulic Code CRAFT-2 being reviewed by the AB.

We have combined the reviews of two individual TAR's since TAR-3916 is revising break location and configuration criteria, and its net effect could ultimately cause changes within the scope and content of TAR-3217.

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DATE	7/6/76	7/6/76	7/6/76			

MECHANICAL ENGINEERING BRANCH

DIVISION OF SYSTEMS SAFETY

REQUEST FOR ADDITIONAL INFORMATION

WPPSS 1 & 4

1. Dynamic forces caused by time-varying differential pressures across internal structures, core bounce and equivalent thrust at the break form the forcing function for the mathematical model of the reactor internals. Define the locations in the mathematical model where the forcing function would be applied.

2. The model used to obtain pressure differentials across the core is defined in BAW-100092. This model subdivides the core into five control volumes. The pressure differentials from this transient analysis are then used as input to a structural model of the core which defines fuel assembly motion and the resulting forcing function at the core support ledge on the reactor vessel. Provide justification on the adequacy of subdividing the core into five control volumes. Also justify that the forcing function at the core support ledge on the reactor vessel obtained on the basis of this subdivision is conservative.

3. The continuous structures are idealized into a finite number of lumped masses. Indicate the dynamic degrees of freedom associated with each mass location.

4. The two major sources which provide structural rigidity to the reactor vessel are the stiffness of the structural members of the support

structure and the stiffness due to the piping connected to the vessel. Provide an estimate of the contribution to overall stiffness of the vessel support from each of these two sources.

5. Identify the break location which is controlling and produces the highest loading combination for which the reactor pressure vessel supports will be designed. Verify that the loads produced by asymmetric internal and external pressure distribution in the vessel cavity and jet thrust forces have been accounted for.

6. Washington Public Power Supply System have indicated in their letter of April 23, 1976 to the Director of Nuclear Reactor Regulation that break locations and configurations are being revised to conform with the provisions of NRC BTP MEB 3-1 and NRC BTP APCSB 3-1 dated March 1975. The criteria in this position have been accepted for use inside containment and since the postulation of longitudinal breaks at terminal ends of pipes is not a requirement, indicate whether the hot and cold leg pipe sleeves are still necessary. It is our position that sleeving of pipes should not be employed unless there are no other viable design alternatives available. The need for sleeving has not been established. Demonstrate that this is the only feasible design approach.

7. If the pipe sleeves cannot be eliminated provide design details showing the method for maintaining radial clearances from the encapsulated process pipe. The use of encapsulation sleeves is