

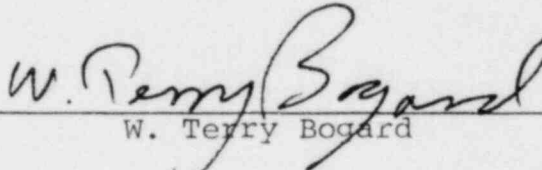
UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In The Matter of)
)
)
COMMONWEALTH EDISON COMPANY) Docket Nos. 50-454 0L
) 50-455 0L
)
(Byron Nuclear Power Station,)
Units 1 & 2))

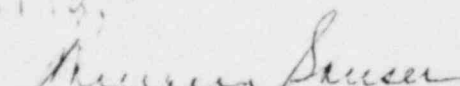
AFFIDAVIT OF W. TERRY BOGARD

The attached questions and answers constitute my testimony in the above-captioned proceeding. The testimony is true and accurate to the best of my knowledge, information and belief.



W. Terry Bogard

Subscribed and sworn to
before me this 3rd day
of June, 1982.



Notary Public

TESTIMONY OF W. TERRY BOGARD

ON DAARE/SAFE CONTENTION 9b

Q.1. Please state your name, employer and current position.

A.1. My name is W. Terry Bogard. I am employed by Westinghouse Nuclear Technology Division as a Senior Engineer at Pittsburgh, Pennsylvania.

Q.2. Briefly state your educational and professional qualifications.

A.2. I have a Bachelor of Science Degree (1973) in Aeronautical and Astronautical Engineering (with Structural Mechanics emphasis) from the University of Illinois, and I have a Master of Science Degree (1977) in Mechanical Engineering from Carnegie-Mellon University.

A considerable part of my education was directed toward methods for mathematical solution and simulation of dynamic structural responses. I joined Westinghouse Electric Corporation in 1974 as an Engineer with responsibility for the evaluation of reactor structures for postulated pipe breaks and seismic events. I have performed such evaluations for several nuclear power plants, including the Byron plant. As a result of addressing the postulated accidents, I have developed modeling methods which are documented in industry technical literature, including advanced modeling

techniques for coupled reactor building and reactor system seismic evaluation, methods for representing the reactor internals fluid structure interaction during pipe breaks, and techniques for evaluating the reactor core for pipe break and seismic events.

Since 1979, I have been associated with dynamic testing and analysis of electrical and mechanical equipment. This includes performance of shaketable tests for seismic qualification of equipment, and data collection for subsequent correlation to mathematical models. I have also been associated with assessment of the probability of pipe ruptures and development of pipe whip restraints to mitigate the effect of pipe breaks.

Q.3. To which contention is this testimony addressed?

A.3. Contention 9b, which reads:

Asymmetric blowdown loads on reactor primary coolant system. This problem may develop from a reactor coolant pipe rupture at the vessel nozzle. The result, after a LOCA incident, could be to place a significant load on the reactor vessel supports, which, in the extreme, could cause their failure. This, in turn, might damage the ECCS lines and/or prevent proper functioning of the control rods. This problem is particularly severe in PWRs. Applicant's response to this problem, a computer model of stresses at FSAR 3.9.1.4.6, is insufficient, and a full scale mechanical test is necessary, especially given the complexity of the reactor vessel geometry.

Q.4. What is the purpose of your testimony?

A.4. To provide information demonstrating that asymmetric blowdown loads are adequately addressed in the design of the Byron Plant. Specifically, I will address the adequacy of analyses which utilize a computer model to conservatively predict asymmetric blowdown responses.

Q.5. Would you briefly describe the term asymmetric blowdown loads as it relates to nuclear power plant design?

A.5. The subject contention relates to an asymmetric loading condition on the reactor vessel and associated support system. This loading condition results from a postulated pipe break at the reactor vessel nozzle. The limiting condition for asymmetric loadings results from a postulated pipe break at the reactor vessel inlet (cold leg) nozzle. The postulated pipe break results in a loss-of-coolant from the reactor coolant system. Because of the energy released through the postulated pipe break, asymmetrical loads are placed on the reactor vessel support system. The reactor vessel is supported by support pads, support shoes and a steel supporting structure that is connected to the building concrete structure at each of four alternate nozzles (See Figure 1). The loads resulting from the postulated pipe break are of short duration. It should be noted

that the occurrence of a pipe break at the vessel inlet nozzle is a highly unlikely event. Designing the reactor coolant system to sustain loadings for such postulated pipe breaks increases the margin of structural integrity.

Q.6. Please describe the loading condition resulting from the postulated pipe break in more detail.

A.6. Forces result from the release of the pressurized primary system coolant, and for guillotine pipe breaks, from the disturbance of the mechanical equilibrium in the piping system prior to the rupture. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront which has low pressure on one side and high pressure on the other; the wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

In the case of a postulated RPV nozzle break, the release of coolant also results in a pressure increase in the region surrounding the postulated break. Pressurization occurs rapidly in the cavity around the reactor vessel. This can exert an asymmetric force on the outside of the vessel.

The loads on the RPV and internals result from the depressurization of the system and from the pressurization of the area around the break. Specifically, the loads may be characterized as: (1) reactor coolant loop mechanical loads, (2) reactor internal hydraulic loads and (3) RPV cavity pressurization loads (only for breaks at the reactor vessel safe end locations).

The reactor coolant loop mechanical loads due to the postulated LOCA result from forces applied to the reactor vessel nozzles from the reactor coolant piping. For guillotine pipe breaks, the loop mechanical loads are due to the release of the normal operating forces present in the pipe prior to the postulated rupture. In the RPV LOCA analysis, the applied RCL loads are determined from the normal operating analysis of the reactor coolant system and are the normal operating loads present in the RCS at the postulated break location. The loads result from the pressure, thermal, and deadweight loads induced by normal operation.

Reactor internal hydraulic loads result from depressurization of the reactor vessel. For a postulated RPV inlet break, the depressurization path for waves entering the reactor vessel is through the inlet nozzle which contains the broken pipe and into the region between the core barrel and reactor vessel (see Figure 2). This region is called the downcomer annulus.

The initial waves propagate up, around, and down the downcomer annulus, then up through the region circumferentially enclosed by the core barrel, that is, the fuel region. The region of the downcomer annulus close to the break depressurizes rapidly but, because of restricted flow areas and finite wave speed, the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and RPV. As the depressurization wave propagates around the downcomer annulus and up through the core, the barrel differential pressure decreases, and similarly, the resulting hydraulic forces drop.

Reactor cavity forces arise from the steam and water which are released into the reactor cavity. The reactor cavity is a cylindrical region between the reactor vessel and surrounding concrete (see Figure 3). The reactor cavity is pressurized asymmetrically with higher pressure on the side adjacent to the break. These horizontal differences in pressure across the reactor cavity result in horizontal forces acting on the reactor vessel. A special flow restrictor exists within the Byron vessel cavity to reduce asymmetric cavity pressurization forces acting on the reactor vessel. Small vertical forces acting on the reactor vessel are caused by pressure on all the surfaces.

Q.7. What is the process for accounting for these loads?

A.7. Westinghouse is responsible for the asymmetric load evaluation of the reactor vessel support system. However, reactor cavity pressure loads are developed by Sargent and Lundy. Westinghouse designs the reactor vessel support pads. The reactor vessel steel supporting structures are designed by Sargent and Lundy.

The initial phase of the analysis is performed by Westinghouse. In this phase, the reactor coolant loop mechanical loads and reactor internal hydraulic loads are calculated based on the postulated pipe break. Additionally, the mass and energy of the reactor coolant released from the postulated pipe break are calculated and transmitted to Sargent and Lundy for use in calculation of cavity pressure loads. These loads are then transmitted to Westinghouse.

With the development of the loads on the reactor vessel, a finite element model is used to calculate loads in the support system.

After the development of loads, Westinghouse compares the allowable loads on the support pads with the calculated loads to ensure that the allowable load limits are not exceeded. The calculated loads are also transmitted to Sargent and Lundy for verification of the integrity of the support framing and primary shield wall.

Q.8. Is the finite element model you refer to also referred to in Section 3.9.1.4.6 of the Byron FSAR?

A.8. Yes.

Q.9. Please describe the finite element model.

A.9. In performing this analysis, Westinghouse utilizes a finite element model of the reactor vessel which accurately reflects the behavior of the reactor vessel support systems. Additionally, the evaluation of asymmetric blowdown loads performed by Westinghouse is consistent with NRC NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems". This NUREG provides an acceptable basis for performing analyses of asymmetric blowdown loads. The NRC no longer lists asymmetric blowdown loads as an unresolved generic issue. The NRC has accepted the Westinghouse asymmetric blowdown analysis for the Byron/Braidwood Station.

The general assembly of the reactor pressure vessel is shown in Figure 4. The mathematical model which represents the RPV consists of two separate non-linear models connected at a common node. The one model (WOSTAS) represents the dynamic vertical characteristics of the vessel and its internals, and the other model (DARI-2) represents the translational and rotational characteristics of the vessel. These two models are combined in the DARI-WOSTAS Code to represent

planar motion of the reactor vessel and its internals. The plane of response is the vertical plane contained by the broken nozzle horizontal centerline and the reactor vertical centerline.

The model for horizontal motion (DARI) is shown in Figure 5. Each node has one translational and one rotational degree-of-freedom in the vertical plane which contains the broken nozzle centerline. A combination of beam elements and concentrated masses is used to represent the components (including the vessel, core barrel, neutron panels, fuel assemblies, water mass, and upper support columns). All the elements are assumed to lie along the vessel centerline. These components are connected by pin-pin rigid links, or translational impact springs with dashpots, or rotational springs.

The model for vertical motion (WOSTAS) is shown in Figure 6. Each mass node has one translational degree-of-freedom. All elements lie along a single vertical axis which coincides with the vessel centerline. The structure is represented by concentrated masses, springs, dashpots, gaps, and frictional elements. The model includes: the core barrel, lower support columns, bottom nozzle skeletons, fuel rods, top nozzles, upper support columns, upper support structure, water mass, and reactor vessel. The core barrel and neutron panels

are represented by masses 4, 5, 6, 7 and 8. Node 1 of the horizontal model (DARI) is coupled to node 2 of the vertical model (WOSTAS). This point represents the intersection of the vessel vertical centerline and the nozzle centerline.

The reactor pressure vessel is restrained by four reactor vessel supports (under every other reactor vessel nozzle) and by the attached piping. The support provides restraint in both tangential directions and in the downward vertical direction. During upward motion, the reactor vessel is restrained by the attached reactor coolant piping. Radial motion is not restrained, in order to allow for thermal growth. To model the RPV restraints, the coupled node in DARI-WOSTAS models has a 3×3 stiffness matrix between the node and the ground. This represents the reactor coolant loop stiffness in the three degrees-of-freedom of the vessel model. Also attached to this node are linear horizontal springs which describe the tangential resistance of the vessel supports, and four individual non-linear vertical stiffness elements which provide downward restraint only. The RPV supports, as represented in Figure 5 and 6 are not indicative of the complexity of the support model used in the analysis. The individual supports are modeled at the actual support pad locations. They accurately represent the independent non-linear behavior of each support.

The RPV support stiffness values, which were the input to the DARI-WOSTAS model, reflect the local flexibilities of the support load path. These flexibilities include the effects of the vessel/nozzle juncture, of nozzle deformation as a beam and shell, of RPV support shoe and RPV support structure.

The DARI-WOSTAS computer code first formulates a set of equilibrium equations for the structural model and then integrates the equations directly. Time-history nodal information obtained from the computer run includes the reactor vessel displacements and reactor support loads.

Q.10. Are the model and analysis process representative of typical engineering evaluations?

A.10. Yes. This type of finite element model and dynamic analysis is typically used in state-of-the-art evaluations for various engineering design applications. This type of computerized analysis is used for the design of everything from building structures, automobiles to aircraft as well as nuclear power plant components.

Q.11. Is it necessary to perform full scale mechanical tests to assure that the loads and stress have been adequately analyzed?

A.11. No. The asymmetric blowdown load phenomenon has been studied in detail over the past several years by the NRC, their consultants and industry. As a result of these studies, a thorough understanding of the subject phenomenon has been developed. In conjunction with this effort, sophisticated analytical methods like those I have described have been developed to conservatively represent the asymmetric blowdown loads in terms of their physical interaction with the reactor vessel support system. These methods have been verified and checked with alternate calculational techniques. The understanding of this phenomenon and the development of computer models provide assurance that asymmetric blowdown loads are conservatively accounted for in the design of the Byron Station.

Based on the development efforts mentioned, it has not been necessary to consider a test to evaluate asymmetric blowdown loads in order to conservatively account for this phenomenon.

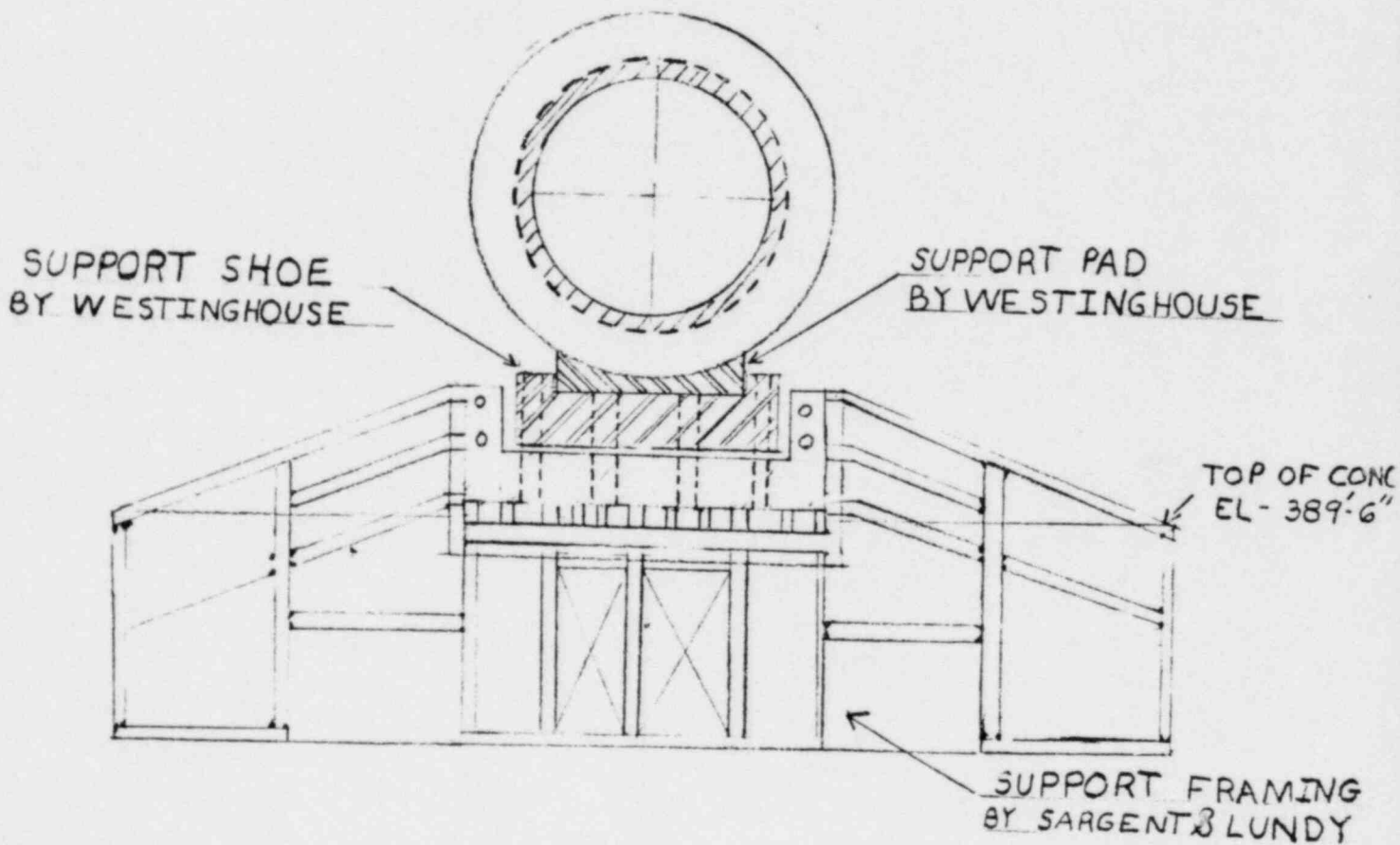
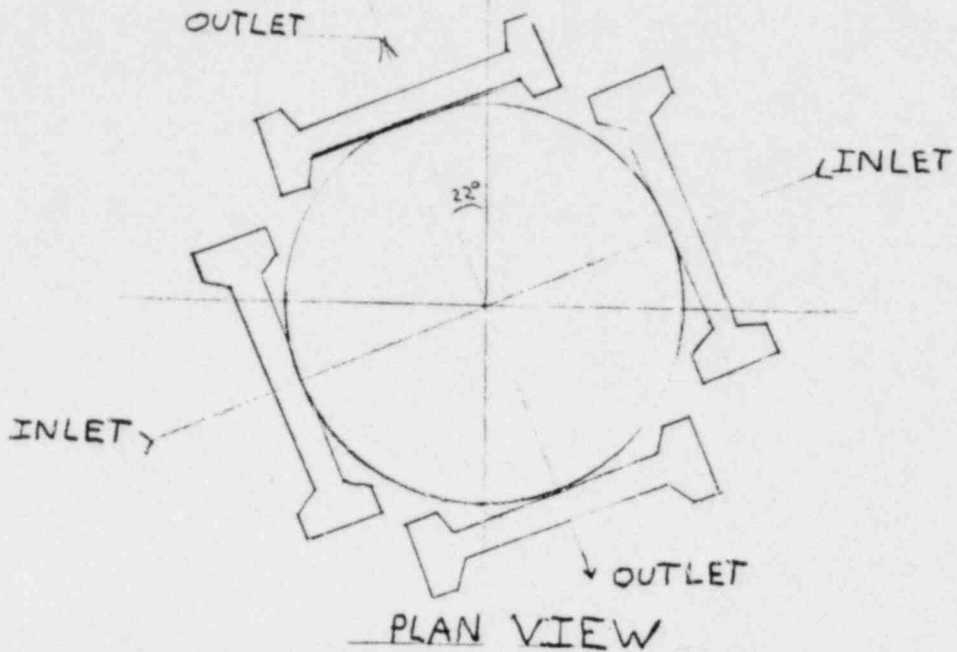
Q.12. Please summarize your testimony.

A.12. The above information demonstrates that Asymmetric blowdown loads are adequately addressed in the design of the Byron plant. The analysis techniques used to evaluate asymmetric blowdown loads conservatively

account for this phenomenon and assure that the plant is adequately designed to withstand such loads for the unlikely occurrence of a pipe break.

FIGURE 1

BYRON PROJECT REACTOR VESSEL SUPPORT SYSTEM



ELEVATION VIEW
TYPICAL 4 PLACES

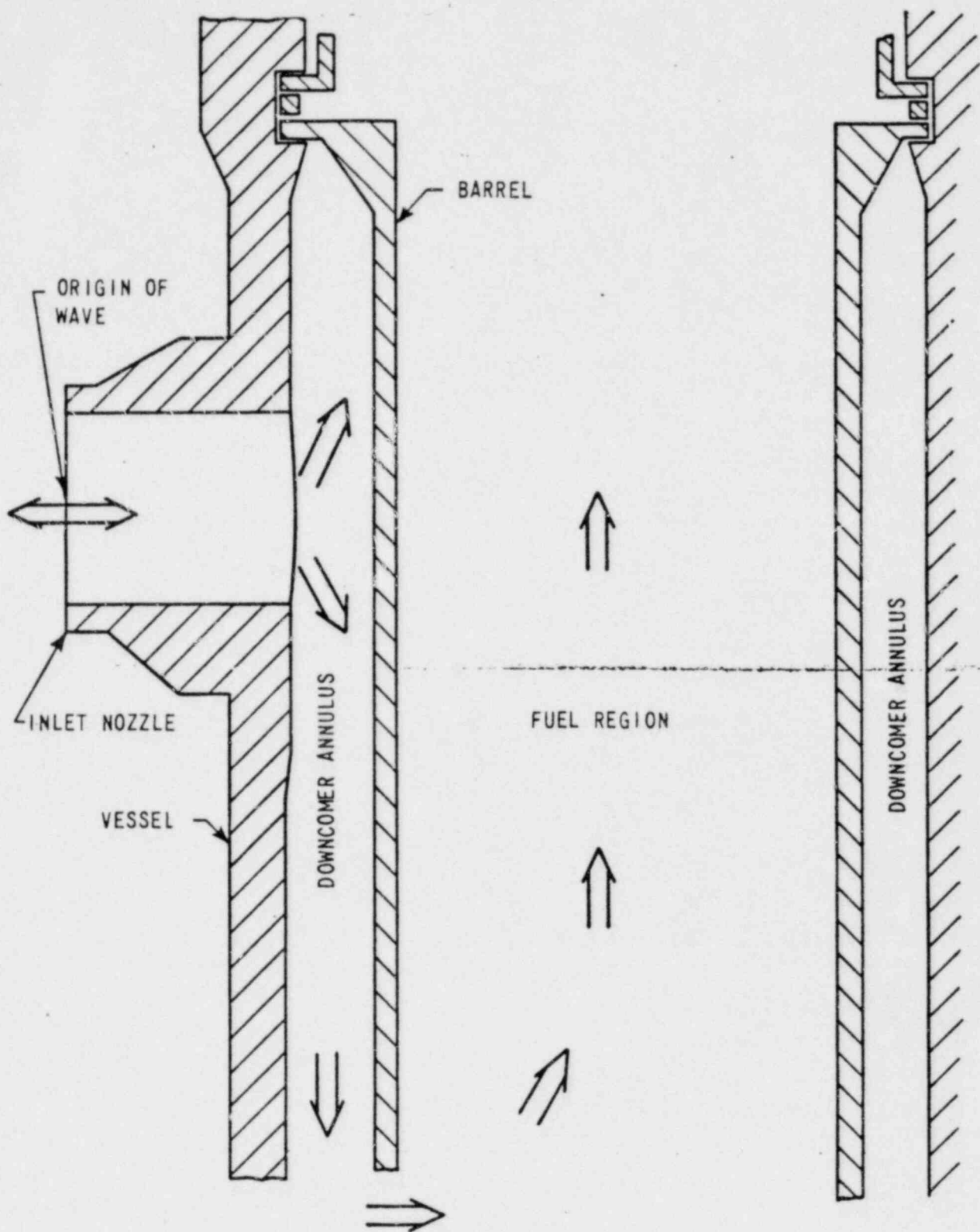
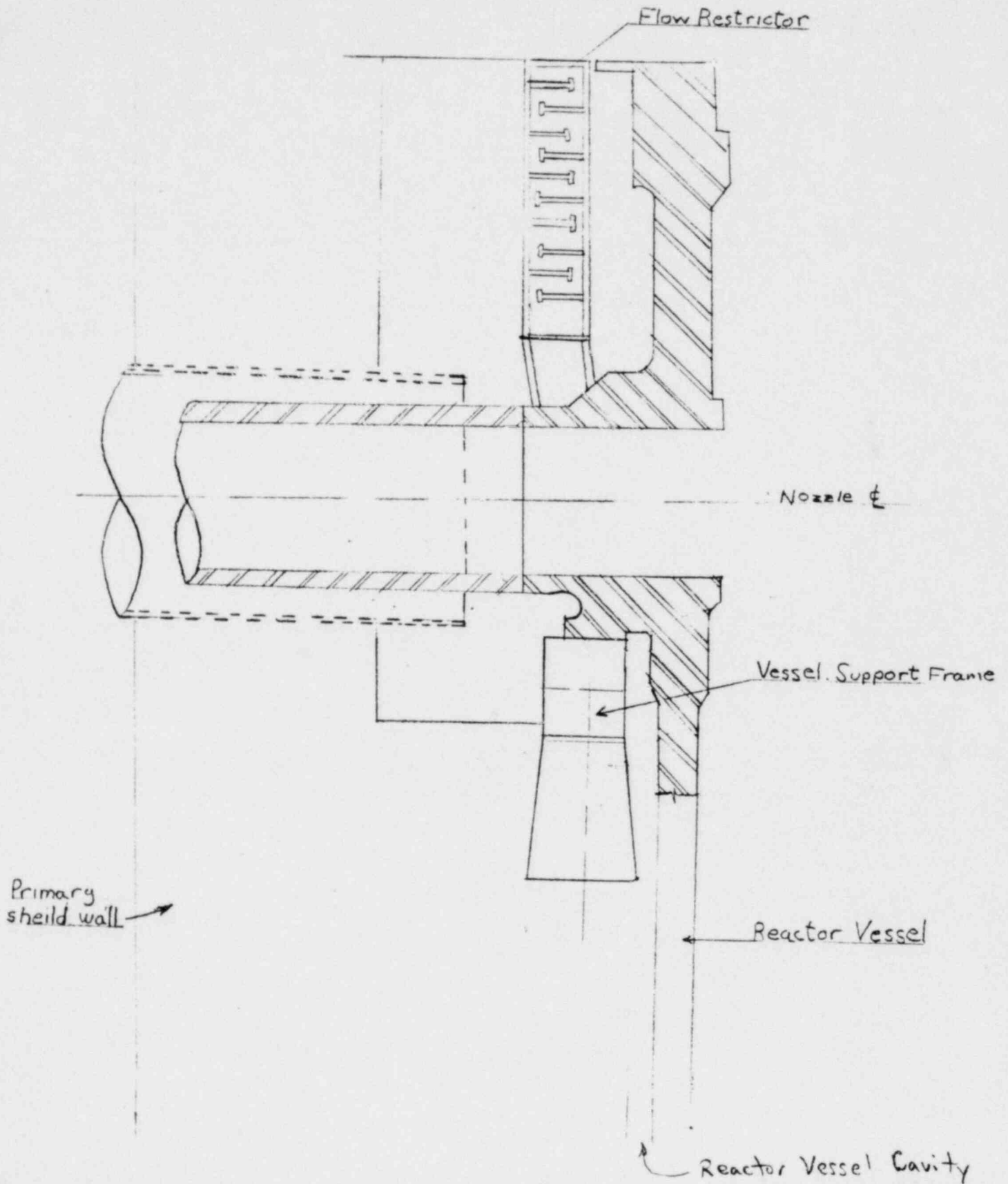


Figure 2 Wave Path for Depressurization Waves Entering RPV Inlet Nozzle (Cold Leg)

Figure 3.

BYRON PROJECT REACTOR VESSEL CAVITY



TYPICAL SECTION AT VESSEL SUPPORT

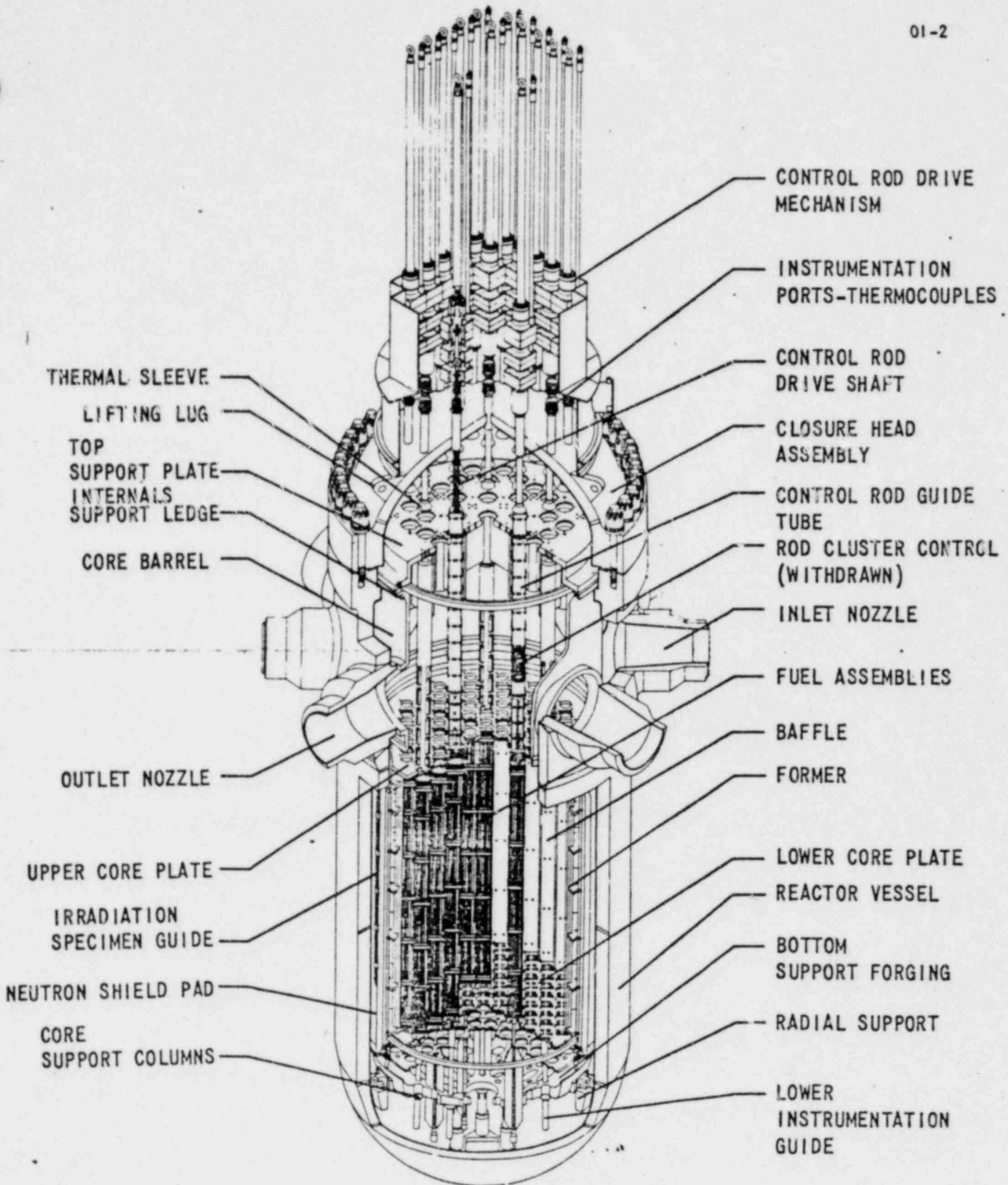


Figure 4 Reactor Vessel General Assembly

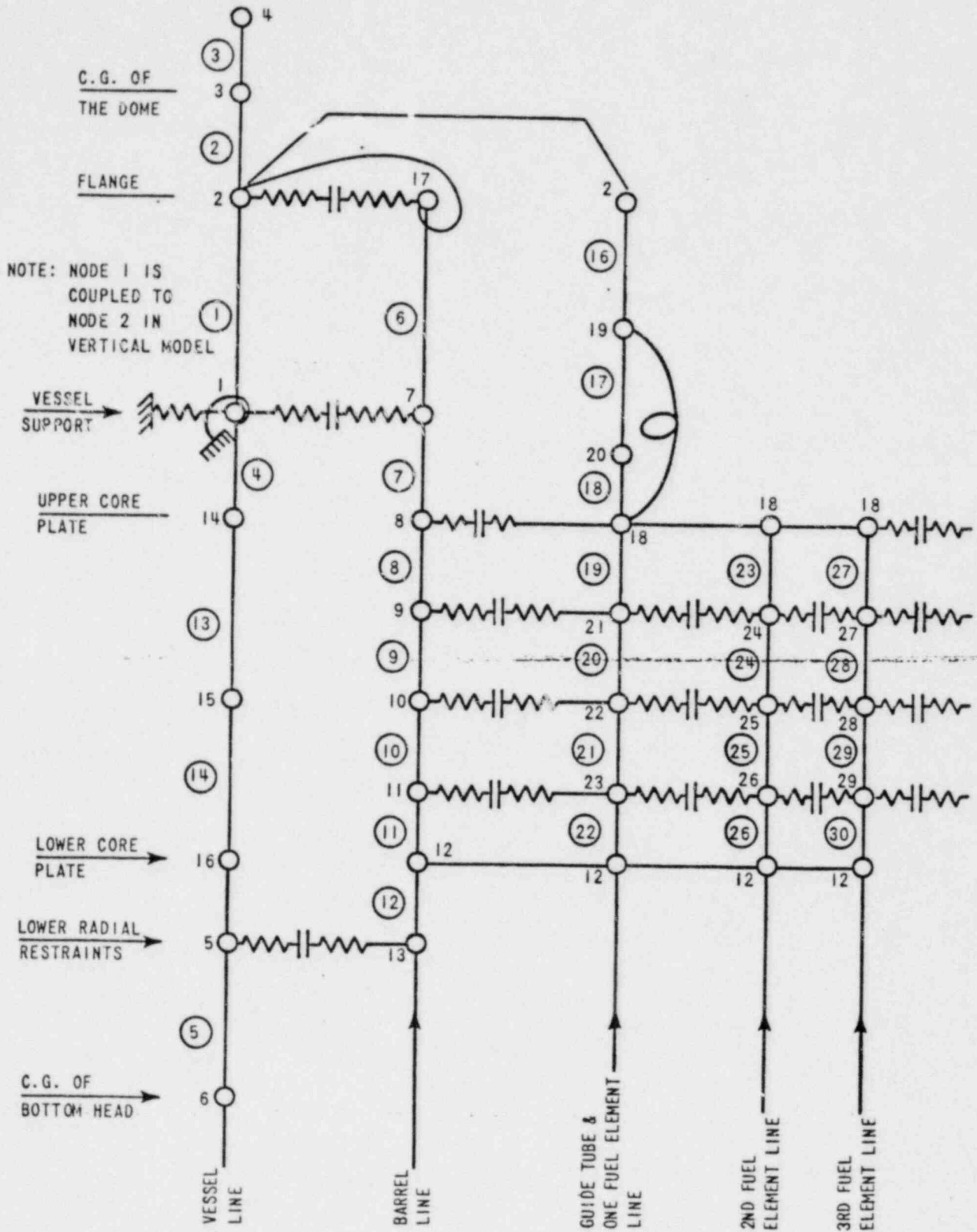


Figure 5 Reactor Internals Model for DARI 2 Variables

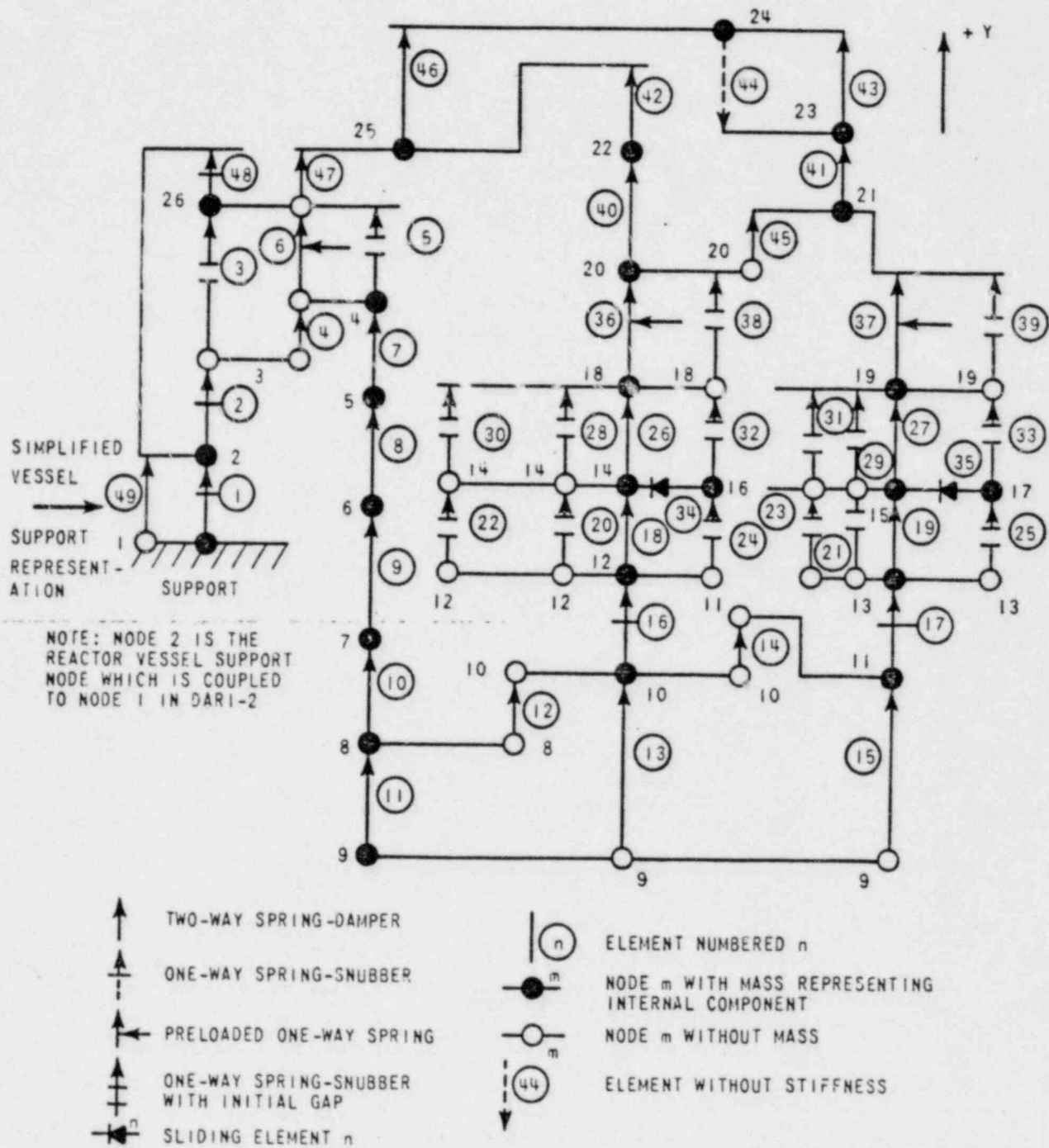


Figure 6 Reactor Internals Mathematical Model for WOSTAS Variables

DAARE/SAFE CONTENTION 9(d)

CONTENTION 9(d)

Intervenors contend that there are many unresolved safety problems with clear health and safety implications and which are demonstrably applicable to the Byron Station design, but are not dealt with adequately in the FSAR. These issues include but are not limited to:

- d. Fracture toughness of steam generators and reactor coolant pump supports. The steel used as steam generator and reactor coolant pump support materials may be subject to cracks in the material near a weld under lower-than-normal temperature conditions. For this reason, under certain circumstances, auxiliary electric heating should, according to NRC generic problem analyses, be provided to keep the temperatures of these structural elements high enough to avoid brittle fracture. The problem may become severe under a LOCA condition. Auxiliary heating is not provided for in the Byron design, as indicated at FSAR 5.2.3.3 or 3.9.3.4.

MATERIAL FACTS AS TO WHICH THERE IS NO
GENUINE ISSUE TO BE HEARD

1. Fracture toughness is the ability of materials to absorb energy despite the presence of flaws in the material. Flaws in the materials which make up the reactor coolant pipes and steam generator supports are expected to be present as a result of the steel manufacturing and fabrication processes and welding.
(Affidavit of Richard J. Netzel, at p. 2.)
2. At lower temperatures, the fracture toughness of materials decreases; i.e. the material is more prone to brittle failure. (Affidavit of Richard J. Netzel, at p. 3.)

3. In designing the steam generator and reactor coolant pump supports the fracture toughness properties of the materials which comprise the support are taken into account. Charpy impact tests, which measure the fracture toughness of materials, were performed on samples of the actual materials which make up the supports. (Affidavit of Richard J. Netzel, at p. 5).
4. The sampling procedure was performed in accordance with ASME code specifications. It assures that sufficient and representative samples are tested to demonstrate that the actual materials used in constructing the supports have fracture toughness properties. (Affidavit of Richard J. Netzel, at p. 5.)
5. The materials tested were tested at a 10°F temperature. This test temperature is 30°F below the assumed lowest service temperature of the metals which make up the supports, and 55°F below the minimum operating temperature in the containment structure, where the supports are located. These test temperatures assure that the materials used in the supports have more than adequate fracture toughness at temperatures below the minimum temperatures to which they will be exposed during operation of the facility. (Affidavit of Richard J. Netzel, pp. 6 and 7.)
6. Since the temperature in the containment structure at the time of a postulated loss of coolant accident will be greater than or, at least, equal to the minimum

operating temperature, fracture toughness of the support materials is assured during LOCA conditions. (Affidavit of Richard J. Netzel, p. 6.)

7. Auxiliary heating systems for the supports are not necessary to assure fracture toughness of the steam generator and reactor coolant pump supports since the support materials were chosen and qualified to assure more than adequate fracture toughness at temperatures below the minimum operating temperature for the Byron containments. (Affidavit of Richard J. Netzel, p. 7.)

DISCUSSION

The Affidavit of Richard Netzel amply demonstrates that in designing the reactor coolant pump and steam generator supports the fracture toughness properties of the materials which make up the supports were considered. Fracture toughness of a material is dependent, in large measure, upon the toughness of the metal. To assure that the supports would not be subject to brittle failure, representative samples of the support materials were tested at temperatures far below the temperatures to which these materials will be exposed during operation of the plant. Thus, it is not necessary to provide auxiliary heating systems for the supports to assure fracture toughness of the materials. No factual issue has been raised by DAARE/SAFE Contention 9d which controverts the facts established in the Affidavit of Mr. Richard Netzel, and accordingly, Edison is entitled to a favorable decision on the Contention as a matter of law.