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Asymmetric Blowdown Loads on PWR Primary Systems

Resolution of Generic Task Action Plan A-2

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ABSTRACT

The NRC staff, after being informed of newly identified asymmetric loadings resulting from postulated ruptures of primary piping, initiated a generic investigation, Task Action Plan A-2. This investigation was limited to pressurized-water-reactor (PWR) plants because of their higher primary system pressures and potential for larger asymmetric loads. The intent of the investigation was to develop a better understanding of this complicated phenomenon and then to develop acceptable criteria and guidelines for the staff to use for evaluating plant analyses.

The staff has completed its investigation and concludes that an acceptable basis is provided in this report for performing and reviewing plant analyses for asymmetric LOCA (loss-of-coolant accident) loads. Guidelines and criteria were developed for the following areas: evaluation of the loading transients, evaluation of the structural components, and evaluation of the fuel assembly.

The staff reviewed and approved computer programs and modeling techniques submitted by each PWR nuclear-steam-supply-system vendor for development of the subcooled blowdown- and cavity-pressure components of the loading transient. Audit models were developed to evaluate the structural computer programs and modeling techniques. The audit evaluation has been completed and the methods have been approved for the structural-analysis method submitted by Westinghouse for the Indian Point Unit 3 plant. Structural-analysis methods developed by Babcock and Wilcox and Combustion Engineering will be evaluated when their plant analyses are submitted to NRC. Criteria and guidelines are provided to perform a detailed evaluation of the fuel assembly. Acceptance criteria are also provided so deformed fuel-assembly spacer grids may be evaluated.

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ABBREVIATIONS

ACRS Advisory Committee on Reactor Safeguards
B&W Babcock and Wilcox
BWR boiling-water reactor
CE Combustion Engineering
CFR Code of Federal Regulations
CP construction permit
DOR Division of Operating Reactors, NRC
DSS Division of Systems Safety
EB Engineering Branch, DOR
ECCS emergency-core-cooling system
EDO Office of the Executive Director for Operations
EPN equivalent pipe networks
HDR superheated steam reactor
HEM homogeneous equilibrium
ISI inservice inspection
LOCA loss-of-coolant accident
NRC U.S. Nuclear Regulatory Commission
NSSS nuclear-steam-supply system
O.D. outside diameter
OL operating license
PWR pressurized-water reactor
RPV reactor pressure vessel
SAI Science Applications, Inc.
SRP Standard Review Plan
SRSS square root of the sum of squares
SSE safe shutdown earthquake
TAP task action plan
W Westinghouse Electric Coporation

ASYMMETRIC BLOWDOWN LOADS ON PWR PRIMARY SYSTEMS: RESOLUTION OF GENERIC TASK ACTION PLAN A-2

1 INTRODUCTION

The NRC staff was informed on May 7, 1975 of newly identified asymmetric loadings which could act on the reactor's primary system. These asymmetric loadings result from a postulated double-ended rupture of the piping in the primary coolant system. The magnitude of these loads is potentially large enough to damage the supports of the reactor vessel, the reactor internals, and other primary components of the system. Because this newly identified phenomenon was complicated and not well understood at that time, the staff initiated a generic study, Task Action Plan (TAP) A-2, to gain a better understanding of these loads and to develop criteria for an evaluation of the response of the primary systems in pressurized-water reactors (PWRs) to these loads.

The staff investigation, TAP A-2, was limited to primary systems of PWRs. Although similar loads associated with a postulated rupture of piping in primary systems in boiling-water reactors (BWRs) are expected to occur, the overall safety significance is considered to be much less because of the lower operating pressures in primary systems in BWRs. If necessary, a staff program to resolve this issue in BWR plants will be considered, once the evaluation of PWR plants is completed.

This investigation was concerned primarily with the loading phenomenon resulting from a postulated rupture of the piping in a primary loop. Although they act on the primary system, additional loadings, such as thermal and seismic, are not the object of this investigation nor are they included in this report. The results of the asymmetric loss-of-coolant accident (LOCA) evaluation must be combined with other loading responses, as appropriate, by applicable staff-design requirements.

Concurrent with the staff investigation, licensees of PWR plants were notified by letter (see Appendix D) that an evaluation of their primary system for asymmetric LOCA loads would be required.

The staff investigation (TAP A-2) has been completed. Criteria and guidance for conducting an evaluation for asymmetric LOCA loads were developed in this study and are included in this report. In addition, the staff has provided an acceptable basis (provided in this report) for the evaluation and review of the response of PWR primary systems to asymmetric LOCA loads.

The report is organized in the same way as is a typical evaluation or review. Section 2 provides a detailed discussion about the nature of these loads on the primary system, as well as a chronology of staff actions. Section 3 offers a detailed discussion of the analytical development of these loads. In Section 4 a discussion of the structural analysis is found, and Section 5 details the fuel-assembly evaluation.

2 BACKGROUND

On May 7, 1975, NRC was informed by the Virginia Electric and Power Company that in the original design of the support system for reactor vessels for North Anna Units 1 and 2, neither Westinghouse nor Stone and Webster had considered an asymmetric loading on the reactor vessel supports resulting from a postulated rupture of reactor-coolant piping at a specific location (for example, the nozzle of the reactor vessel). In the event of a postulated instantaneous, double-ended, offset rupture at the vessel's nozzle, asymmetric loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel caused by transient differential pressures in the reactor cavity. With the advent of more-sophisticated computer programs and the development of more-detailed analytical models, it became apparent that such differential pressures, although of short duration, could place a significant load on the supports of the reactor vessel and on other components, possibly affecting their integrity. This potential safety concern, first identified during the review of the North Anna facilities, was determined to have generic implications for all PWRs.

Upon closer examination of this situation, it appeared that postulated breaks in a reactor-coolant pipe at nozzles of reactor pressure vessels (RPVs) were not the only area of concern, but that other pipe breaks in the reactor coolant system could cause internal and external transient loads to act upon both the reactor vessel and other components. For the postulated pipe break in the cold leg, asymmetric pressure changes could take place in the annulus between the core barrel and the vessel. Decompression could take place on the side of the RPV annulus nearest the pipe break before the pressure on the opposite side of the RPV changed. This momentary differential pressure across the core barrel could induce lateral loads both on the core barrel and on the reactor vessel. Vertical loads could also be applied to the core internals and to the vessel because of the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, for breaks in RPV nozzles, the annulus between the reactor and biological shield wall could become asymmetrically pressurized, resulting in a differential pressure across the reactor vessel causing additional horizontal and vertical external loads on the reactor vessel. In addition, the reactor vessel could be loaded simultaneously by the effects of strain-energy release and blowdown thrust at the pipe break. For breaks at reactor vessel outlets, the same type of loadings could occur, but the internal loads would be predominantly vertical because of the more-rapid decompression of the upper plenum. Similar asymmetric forces could also be generated by postulated pipe breaks located at the steam generator and reactor-coolant pump. Therefore, the scope of this problem was not limited to the vessel support system itself.

On October 15, 1975, the NRC staff sent out letters surveying all of the operating PWRs (Ref. 1) and determined from the survey responses that these asymmetric loads had not been considered in the design of any PWR primary systems.

In June 1976, the NRC requested (Ref. 2) all operating PWR licensees to evaluate the adequacy of reactor system components and supports at their facilities, with respect to these newly identified loads.

In response to our request, most licensees with Westinghouse plants proposed to augment the inservice inspection (ISI) (Ref. 3) of the weld area between reactor-vessel nozzle and safe-end in lieu of providing an evaluation of postulated piping failures. In September 1976, licensees with Combustion Engineering (CE) plants submitted a probability study (prepared by Science Applications, Inc. (SAI)) (Ref. 4) in support of their conclusion that a pipe break at a particular location (vessel nozzle) was so unlikely to occur that no further analysis was necessary. A similar study was conducted by SAI for Babcock and Wilcox (B&W) plants in September 1977 (Ref. 5). When the Westinghouse and (Ref. 3) CE owners-group reports were received in September 1976, NRC formed a special review task group to evaluate the alternative proposals. In addition, EG&G Idaho, Inc., was contracted to perform an independent review of the SAI probability study submitted for the CE owners group.

This review effort resulted in a substantial number of questions which have been provided to representatives of each group. Based on the nature of these questions, the NRC staff did not accept those reports as resolving the asymmetric LOCA-load generic issue. Based on NRC review, the staff concluded that a sufficient data base did not exist within the nuclear industry to provide satisfactory answers. Several long-term experimental programs would be required to provide much of this information. Although the probability study submitted in September 1977 by SAI (Ref. 5) for certain B&W owners did respond to some of these questions, NRC staff determined that the more fundamental questions still remained unanswered.

A second and equally important reason for not accepting probability/ISI approaches as a solution concerned our need and industry's need to obtain a better understanding of the problem. We considered it essential that an understanding of the important breaks and associated consequences be known before applying any remedy--be it pipe restraints, probability, ISI, or some combination of these measures.

Upon reaching these conclusions, NRC staff initiated an inhouse investigation of this phenomenon (TAP A-2) (see Appendix A) and notified each PWR licensee (January 25, 1978) (see Appendix D) of NRC's requirement to have an evaluation of their facilities for these loads.

3 DEVELOPMENT OF LOADING FUNCTIONS

The first consideration in conducting an evaluation of the primary system for LOCA loads is to define and develop the loading transients associated with the postulated pipe rupture. The loading transients depend on the postulated break conditions, the postulated break locations, and on the operating conditions and geometry of the system. Typical loads acting on the reactor pressure vessel and internals are illustrated in Figures 1 and 2. A summary of the loading components follows.

(1) Subcooled Blowdown Loads--Dynamic hydraulic forces occur throughout the primary system due to the rapid subcooled depressurization of the system. These forces occur in the piping and component internals throughout the system.

(2) Cavity-Pressure Loads--A rupture of the primary-loop piping at a component nozzle can pressurize the cavity between the vessel (reactor-pressure vessel, steam generator, or reactor-coolant pump) and the cavity wall. Lateral and vertical forces are exerted on the component from this cavity pressurization.

(3) Jet Impingement and Jet-Thrust Loads--Jet impingement and thrust loads also act on the primary system at the break location.

(4) Strain-Energy-Release Loads--The pipe rupture is postulated to occur during normal operating conditions when the strain energy has built up in the system. Some of this strain energy is released when the rupture occurs.

    Typically, the strain-energy-release loads are determined by applying the steady-state, normally operating, hydraulic loads and the forces that result from the weight of the structure in a static analysis of the intact system. For this static analysis, snubbers are assumed to be inactive. The steady-state displacements from this static analysis are then used as initial conditions for the dynamic analysis where the broken leg is disconnected from the vessel and allowed to move, and the snubbers are active.

All of the loading conditions discussed must be included in the evaluation of the primary system. These loads provide the information needed to evaluate the dynamic response of the supports of the reactor-pressure vessel, the reactor-core structures, and the primary-coolant loop and its loop components. A detailed discussion of the subcooled blowdown (hydraulic) and cavity loads is provided in Sections 3.1 and 3.2.

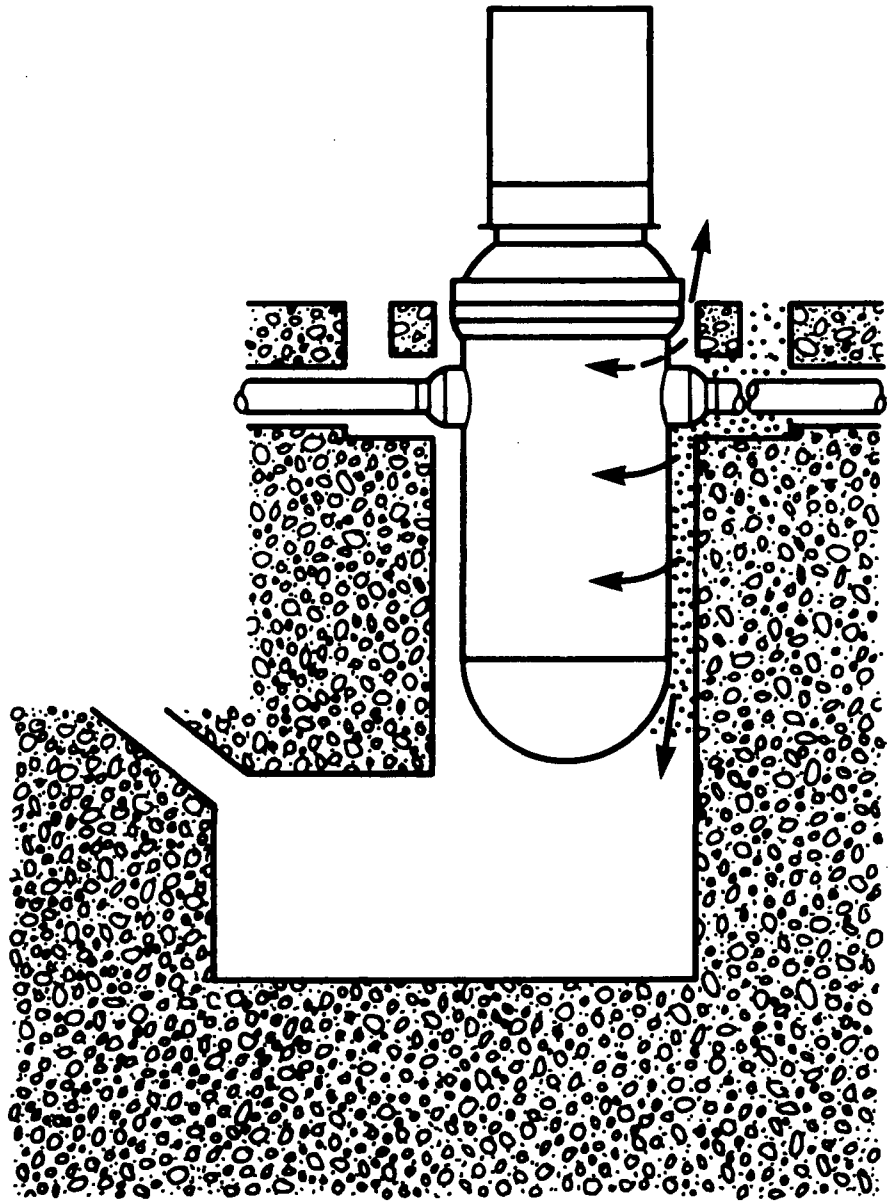

Figure 1. Pressurization of Reactor Annulus.

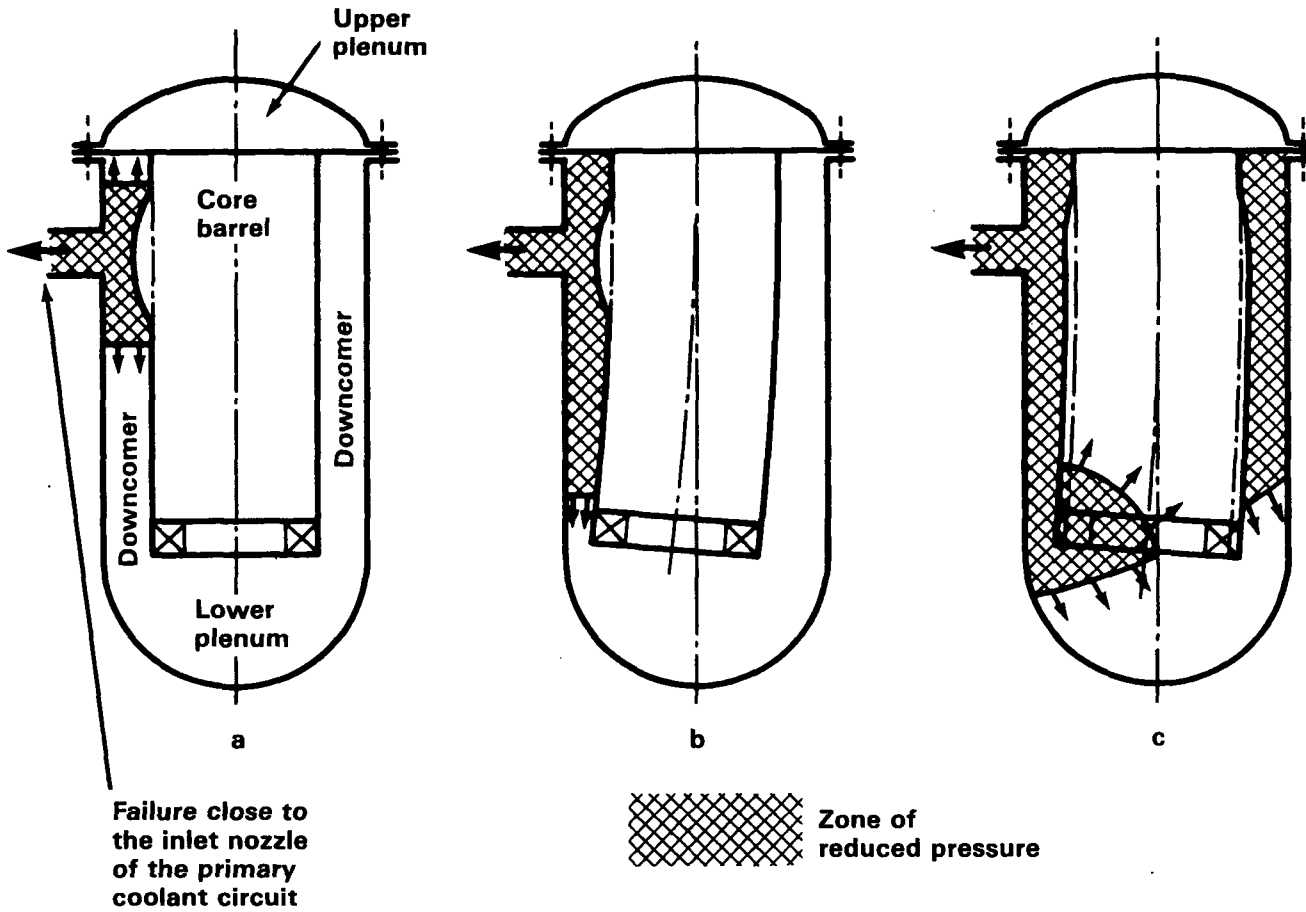

Figure 2. Example of Asymmetrical Internals Load.

3.1 Internal Subcooled Blowdown Loads

This section describes the procedures currently used by NRC to evaluate the methods utilized in a submittal for the calculation of the hydrodynamic loadings on a reactor-coolant system undergoing a postulated LOCA.

Development of these loads involves several steps:

(1) The first step in the development of these loads is the determination of the transient pressure and velocity distribution throughout the system, using an appropriate thermal-hydraulic, computer program.

(2) The second step involves the conversion of these transient pressure and velocity data into equivalent transient forces throughout the primary system. The transient forces developed in these two steps are then included, along with the remaining LOCA loads, in a time-history structural analysis of the primary system.

The staff has reviewed and approved Westinghouse (Ref. 6), Babcock and Wilcox (Ref. 7), and Combustion Engineering (Ref. 8) topical reports for the development of these hydrodynamic loads. The procedures and acceptance criteria used by the staff for those reviews are discussed in Sections 3.1.1 through 3.1.4.

3.1.1 General Review Procedures

NRC staff review covers the analytical methods used by the licensee for the evaluation of the PWR primary-coolant system subjected to a LOCA with emphasis on the subcooled, transition, and early saturation periods of the decompression transient. It is during this timespan, on the order of 0.05 to 0.20 seconds, that the induced hydraulic loads are greatest. The analytical methods used must account for the acoustic wave phenomenon associated with the subcooled blowdown, the transmission, and the reflection of pressure waves throughout the system.

Modeling of the vessel's downcomer region is of major importance to this review. The application of current, one-dimensional computer programs to the calculation of the multidimensional pressure field of the downcomer annulus is given significant attention. It is this pressure field which has the greatest influence on the lateral loads on the reactor vessel, core-support barrel, and fuel assemblies.

Figure 3 provides a summary of the effects of modeling on the predicted hydraulic loads. A Westinghouse three-loop plant was analyzed using the NRC staff program WHAM/MOD-007. Two models were used; one was similar to the Westinghouse model, and the other was developed by the staff for coarse- and fine-noding comparisons, respectively. Also shown in Figure 3 is the effect on the calculated results of increasing the break-opening time.

In principle, the effects of a sudden decompression of a PWR primary system can be determined by a straightforward calculation considering the LOCA pressure wave and its resulting interaction with internal structures of the

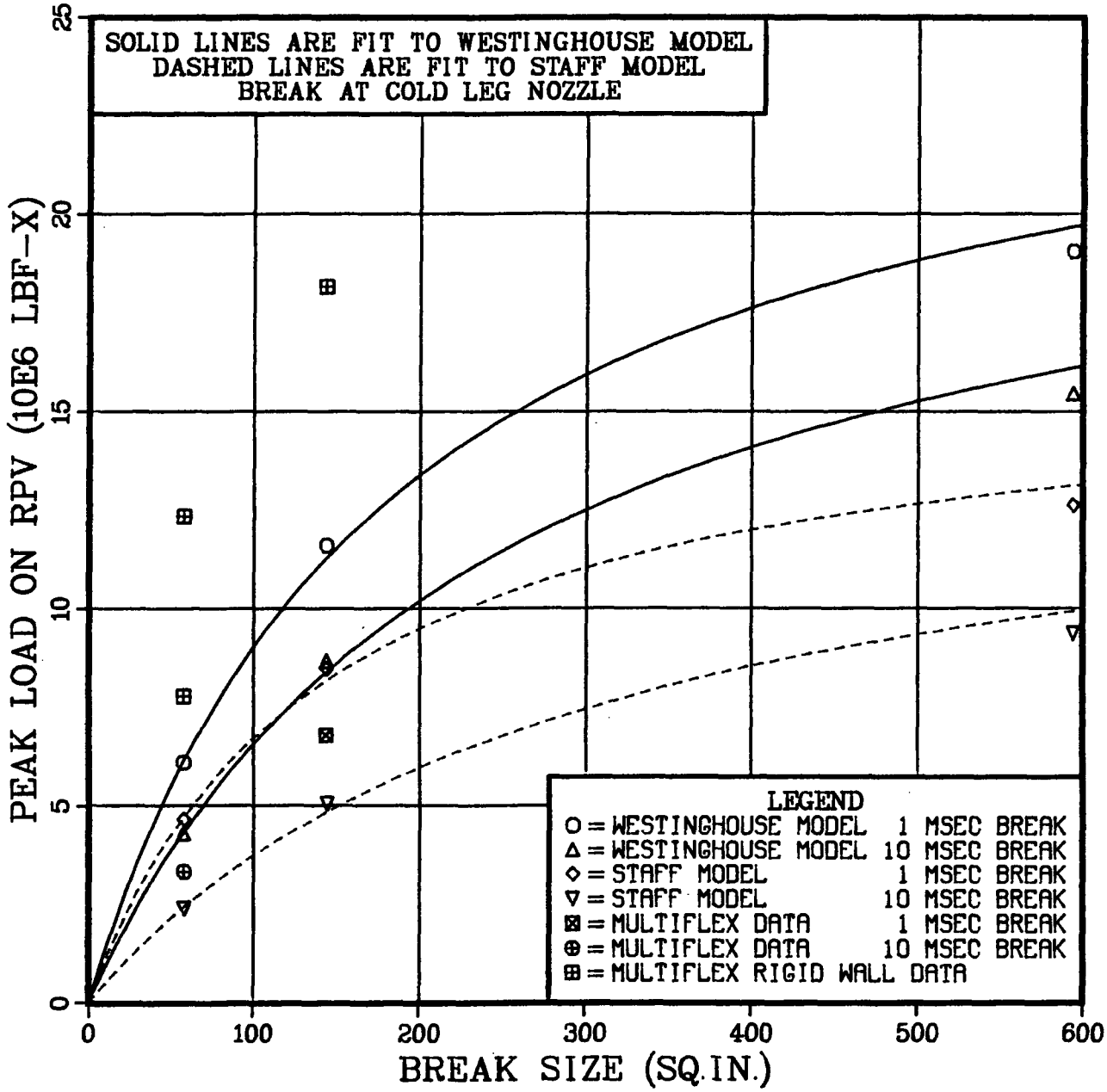

Figure 3. Westinghouse Three-Loop PWR Loads Evaluation, Cold-Leg Nozzle Break.

reactor. However, the physical process is not as straightforward. The resulting motion of the core-support barrel to the decompression wave will feed back into the local fluid pressure through the compressibility of the water. The resulting hydrodynamic loads are reduced when this fluid-structure coupling is considered.

The SOLA-FLX computer program is being developed to supply the staff with a tool for evaluating the coupled system. Figure 4, taken from Reference 9, shows the effect of the fluid-structure-coupled calculation. This figure illustrates the reduced loads associated with a coupled system and also shows the effect of modeling the downcomer width. The results show that the rigid-wall assumption yields higher differential pressure, hence higher peak loads, with a higher frequency response than results in the coupled system; the decompression wave travels faster in the uncoupled case. The thickness of the downcomer annulus can be represented as a single cell and still yield a converged solution. The analytical development of a coupled fluid-structure solution, if employed by the licensee, must also be reviewed.

The review procedure can be broken down into three major categories: (1) analytical development, (2) application and system modeling, and (3) computer-program verification. These items are addressed below in Sections 3.1.2, 3.1.3, and 3.1.4, respectively.

3.1.2 Analytical Development

The staff typically reviews a number of areas as part of the licensee's analytical development of the thermal-hydraulic computer program used for the evaluation of the subcooled blowdown loads. This review begins with the basic conservation equations of mass, momentum, and energy, and with the equation of state.

The assumption generally used for these analyses is homogeneity of the steam/water mixture. The subcooled, transition, and early saturated regimes of the transient, where the homogeneous-fluid assumption is acceptable, are evaluated. The possibility of potential nonequilibrium effects is also considered in the review.

The hydrodynamic loads are directly proportional to the differential pressures applied to a structure. If, as a result of nonequilibrium, the pressure at the break plane falls below the saturation pressure of the fluid, prior to the development of two-phase-flow conditions, an increased loading could result. The effects of nonequilibrium would increase the loading on the core support barrel structure if this pressure "undershoot" is transmitted to the downcomer-annulus region.

The equation of state, the relationship between the fluid pressure and density, and the sonic or acoustic wave speed are also reviewed. The range of pressures from operational pressure to the saturation pressure should be adequately described.

The assumptions used to reduce the basic conservation equations to the form used by the computer program are evaluated, with respect to the problem being

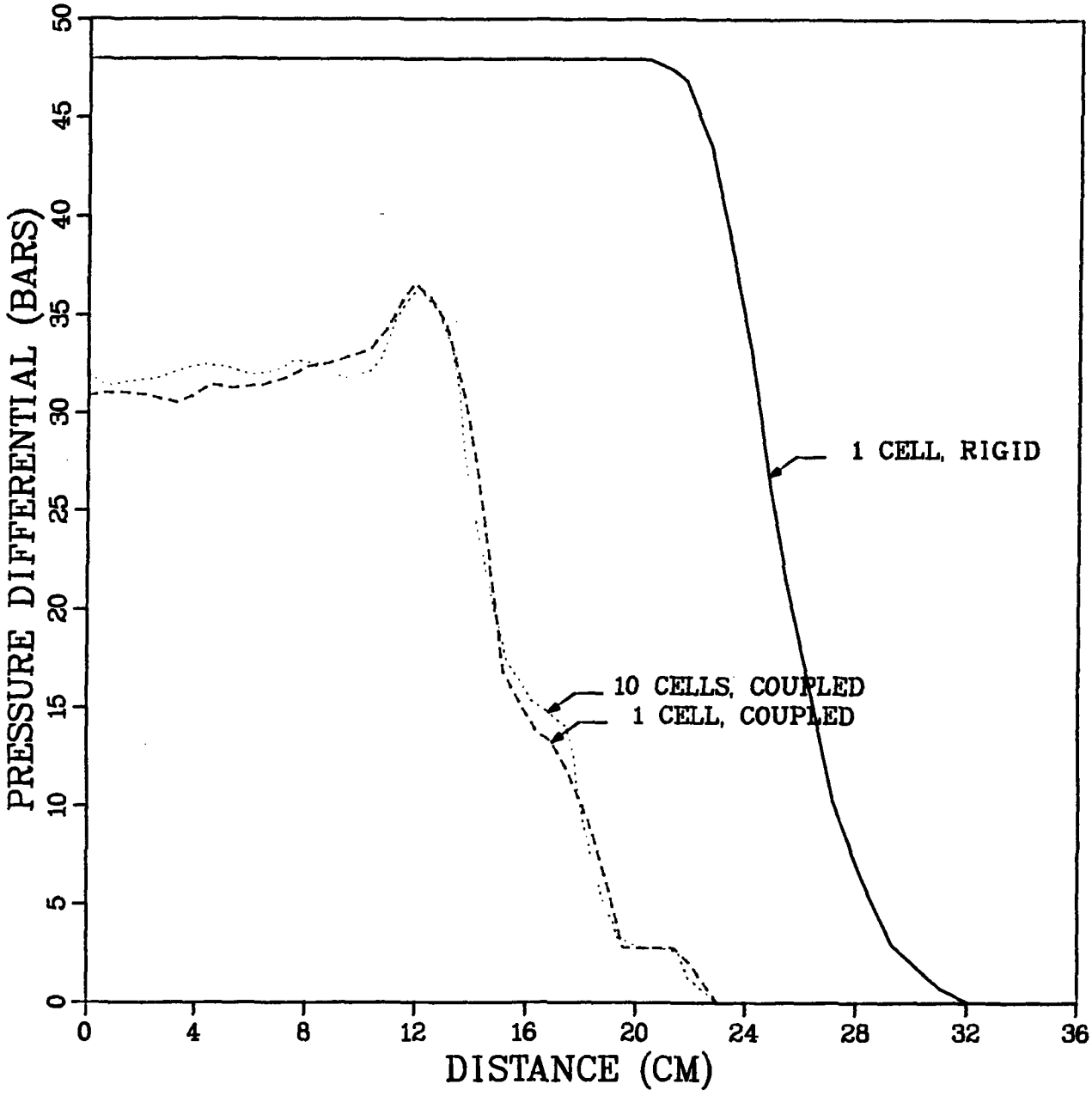

Figure 4. Uncoupled Versus Coupled Fluid-Structure Response, 1/25 Scale PWR Geometry at 240 Microseconds.

solved. The review includes the solution schemes used, convergence criteria employed, steady-state initialization, and the demonstration that conservation of mass, momentum, and energy are achieved. This last item is of primary importance when a one-dimensional computer is used to model a multidimensional region.

For those computer programs which include the fluid-structure coupling, the method of incorporating the moving boundary into the conservation equations is reviewed.

The modeling of system components, such as the heat transfer in the reactor core and steam generator, the pressurizer, the reactor-coolant pumps, and nonrecoverable energy losses are also considered.

The essential boundary conditions reviewed are the break-opening time and area characteristics. The history of the break-opening time and the break area in the nuclear-steam-supply system (NSSS) primary piping system are plant specific and depend on such considerations as the location of the break, the stiffness and mass of the piping system involved, and the type and location of pipe restraints/supports being used. The break geometry postulated for this evaluation is a complete circumferential break of the primary coolant pipe. Appendix C of this report discusses characteristics of pipe breaks in more detail.

The discharge-flow model is an important aspect of this review. The discharge-flow model for the postulated break is the system-forcing function. As such, the treatment of the subcooled critical flow and potential nonequilibrium effects must be properly accounted for in the development of the discharge-flow model.

3.1.3 Application and System Modeling

After the basic solution technique has been selected and equations have been developed, the next stage of the review is oriented to using the computer program to evaluate the decompression transient. The model of the system used to track transmission and reflection of the acoustic waves is reviewed.

The most significant aspect of the review is the modeling of the asymmetric pressure field in the downcomer annulus. At a minimum, a two-dimensional pressure field is required to evaluate adequately the effects of the decompression waves as they travel into and around the downcomer annulus. Since the current generation of computer programs being used for this application is one dimensional, the application of the program to modeling multidimensional regions must be reviewed.

Guidelines for the development of these models, referred to as equivalent pipe networks (EPN) for the solution of the wave equation in characteristic form, may be found in References 10-12. In addition to the characteristic form of solution, the volume-flowpath approach (typical of codes such as RELAP) is also used by some licensees. This approach has been reviewed and can be demonstrated to be nearly equivalent to the vector-momentum equation governing nearly incompressible, low-speed flow for a multidimensional analysis.

The acoustic-wave phenomenon, the transmission and reflection of the pressure waves, is reviewed in terms of the computer-program method of solution. For the characteristic form, the transmission and reflection coefficients are dependent on the area, or the sum of the areas, through which the waves must pass. In the volume-flowpath approach, these coefficients are replaced by the effective inertia of a given flowpath connecting two volumes. These coefficients are reviewed to assure that the wave transmission into the downcomer annulus is conservatively evaluated.

The sensitivity of the model to spatial representation, time-step size, and to the various convergence criteria is also reviewed.

3.1.4 Code Verification

The last part of the review procedure is code verification, to verify the ability of a computer program to predict the subcooled decompression phenomenon or acoustic-wave tracking. By comparing the licensee's program and modeling procedure to selected problems and experimental data, significant modeling assumptions and features can be identified. In addition, the staff has augmented the WHAM computer program (Ref. 13) for use in calculating these same problems and experiments. The program version used by the staff is WHAM/MOD-007 (Ref. 14).

One of the major aspects of the review is the modeling of the downcomer annulus in at least two-dimensional form with the one-dimensional computer programs. Four sample problems were developed to aid in this review. They are described, along with the WHAM/MOD-007 analyses, in Reference 15. Analyses of these problems have also been performed by the staff's consultants at Sandia with the CSQ computer program (Ref. 16). CSQ is a two-dimensional hydrodynamic program. Comparisons of the two sets of analyses demonstrate that the modeling techniques used for the WHAM/MOD-007 analyses are in good agreement with the two-dimensional analyses.

Comparisons with applicable test data covering a wide range of system geometries and scales are also required as part of the code-verification review. A partial list of acceptable test data is given in References 17-24. An HDR (superheated steam reactor) test facility (Ref. 24) analysis is required by all licensees as part of code verification. HDR is a nearly full-scale PWR geometry which has been instrumented to obtain data during a subcooled decompression transient. Westinghouse, Babcock & Wilcox, and Combustion Engineering will be performing analyses of an HDR test to provide additional verification of their methodologies.

Audit comparison of a PWR subcooled decompression analysis with the current staff models and computer program is also performed as part of the verification program.

3.1.5 Summary

The staff review of the methodology used by the licensee to evaluate the response of the PWR to a subcooled decompression transient involves a number of items important to the development of the hydraulic loads. Although each

analysis, including code formulation and code predictions of standard subcompartment problems. The licensee must also provide ancillary evidence, in the form of nodalization-sensitivity studies, to justify that the nodalization used in the design-basis analysis is adequate for calculating the subcompartment pressure transients. This requirement applies irrespective of the licensee's adherence to the subcompartment guidelines contained in this report.

If the licensee has used a subcompartment computer code not previously found acceptable by the staff, the staff will perform confirmatory analyses and compare the results with those obtained by the applicant. Of primary interest are the differential pressures across subcompartment walls and the transient asymmetric pressure loads and their resultant location on enclosed components (such as the reactor pressure vessel). Evaluation of the adequacy of structures and component supports should be based on the more severe of the loading conditions calculated by the staff and the licensee following staff review and acceptance of the nodalization model and input data.

3.2.2 Analytical Methods

Conventional computer codes for subcompartment analyses utilize a number of assumptions to facilitate the solution of the equations that describe the fluid-flow phenomena occurring within a subcompartment. Codes developed for current use by the staff include the COMPARE code (Ref. 25) and the RELAP-4 MOD 5 Code (Ref. 26). These codes predict (in a conservative manner) experimental data obtained from subcompartment tests. The principal assumptions made in currently accepted subcompartment-analysis codes are addressed below.

Containment subcompartments (for example, the reactor cavity) are analyzed by subdividing them into a number of control volumes or nodes, that is, by nodalization of the subcompartment. This network of control volumes interconnected by junctions or flowpaths describes the geometry of the subcompartment.

All thermodynamic properties within a control volume are assumed to be uniform; as a result, pressure distributions are represented by mean values. Furthermore, control-volume constituents--a two-phase, two-component, air-steam-water mixture--are assumed completely mixed at stagnation conditions and at the same temperature. Thus, thermodynamic calculations are based on the assumption of homogeneous equilibrium. Additionally, the water expelled from the break is assumed to be completely entrained in the downstream flow mixture.

The subsonic rates of mass flow between control volumes are obtained by solving the incompressible one-dimensional momentum equation. Compressibility effects in the vent flow are accounted for by adjusting vent-flow densities. The momentum equation includes terms that account for inertia effects and pressure loss because of wall friction and form losses. The calculation of critical-mass flows between control volumes is based on either the frictionless Moody correlation with a multiplier of 0.6 or the homogeneous equilibrium model (HEM) correlation as outlined in Standard Review Plan Section 6.2.1.2.

It should be noted that the use of the HEM critical-flow correlation and the Moody correlation with a 0.6 multiplier has, in the past, been found acceptable

part may be considered on its own, the entire package must be considered in determining the acceptability of the method and of the computer program. The goal is to provide assurance that the development of the pressure and velocity fields is applicable to the PWR geometry and that the loads obtained are conservative and consistent with safe plant-shutdown requirements.

3.2 Cavity-Pressure Loads

A subcompartment is a fully or partially enclosed volume within the containment that houses or adjoins high-energy piping systems and restricts the flow of fluid to the main containment volume in the event of a postulated pipe rupture. Analyses of pressure transients in containment subcompartments involve the evaluation of the thermodynamic consequences of a postulated pipe rupture in these compartmentalized regions of the containment. A spatially dependent, short-term-pressure transient will occur inside a subcompartment following a pipe rupture, and pressure differentials across subcompartment structures and equipment will reach their maximum values generally within the first second after blowdown begins. The safety issue deals with the localized pressure gradients within a subcompartment and whether the subcompartment walls and component supports for the primary reactor-coolant system can withstand these gradients so that the reactor can be brought to a safe condition in the event of a pipe rupture in the reactor system.

Therefore, subcompartment pressure analyses must be performed to determine the asymmetric pressure loadings on major components and on subcompartment walls. A discussion of the analytical methods, assumptions, and procedures for performing these analyses is provided in this section. The discussion also includes a summary of the staff's review procedures and acceptance guidelines for evaluating the subcompartment analyses.

3.2.1 Review Procedures

The staff procedure for evaluating a subcompartment analysis will follow one of two paths depending on the subcompartment-analysis code used by an applicant.

An expedited review is possible if a licensee has used a subcompartment code that has been reviewed by the staff and found acceptable. In this instance, the staff will review the subcompartment-modeling procedures in light of the recommendations contained in this report. The staff will review the nodalization of the subcompartment and the input parameters used to determine the adequacy of the analytical model for calculating the pressure gradients and component loads. If a licensee has adhered to these guidelines, sensitivity studies are not required. NRC staff will also review the adequacy of the model used to determine the component loads if it differs from the subcompartment model used to calculate pressure gradients.

If a licensee has used a subcompartment code not previously reviewed and found acceptable by the staff, additional information must be provided to justify the acceptability of the subcompartment analysis. In this regard, the licensee should provide a description of the subcompartment code used for

because these models in general calculate low-mass-flow rates conservatively, over the spectrum of fluid conditions. The limiting loads on structures or equipment are for most cases determined by the loads induced by the rupture of a high-energy line immediately adjacent to the structure or equipment in question. Therefore, critical-flow correlations which yield a low estimate of mass flow will result in higher pressures in the vicinity of the break and lower pressures in regions further removed from the break. This translates to maximizing the loads on the structures or components.

In those situations, however, where the component is most vulnerable to a loading induced by the rupture of a pipe not immediately adjacent to the component or where the worst loading results from an overturning moment created by loads away from the break, it is incumbent upon the designer to consider the effects of a flow correlation which would yield corresponding high values of choked flow.

Additionally, the critical-flow models employed in state-of-the-art codes are unable to mechanistically model vena contracta effects. Therefore, the licensee should demonstrate that the vena contracta effects are unimportant to the analysis or that this phenomenom is adequately treated in the flow calculation.

In summary, the staff will require that subcompartment analysis codes that are used for calculating transient pressure gradients incorporate the following features:

(1) Thermodynamic capabilities of mixture of air, steam, and water

(2) Homogeneous equilibrium of control-volume constituents

(3) Complete water entrainment for the determination of flow mixtures

(4) Subsonic flow calculations based on the incompressible momentum equation, including terms for:

    (a) temporal change of momentum

    (b) pressure loss from friction, area change, turning losses, and

    (c) compressibility effects (variation of upstream density)

(5) Vent-flow checks based on selection of minimum flow calculated from subsonic calculation and critical-flow calculation

(6) Critical-flow calculation based on the HEM correlation or the Moody correlation with a 0.6 multiplier

Subcompartment Modeling--Subcompartment modeling involves the nodalization of the physical region to be analyzed and the calculation of the requisite input data describing the nodalization arrangement for use in computer-code analyses. In this regard, the following topics need to be discussed:

(1) Subcompartment nodalization

(a) general nodalization procedures

(b) nodalization of blowdown volume(s)

(2) Subcompartment-code input data

(a) control-volume data

(b) junction-flowpath parameters

(c) time steps and blowdown data

(d) insulation considerations

(3) Force and moment calculations

3.2.2.1 Subcompartment Nodalization

Subcompartment nodalization is the subdividing of a subcompartment into a network of control volumes or nodes to assure an accurate prediction of the transient-pressure response within the subcompartment. Consequently, the staff's review of subcompartment analyses will include an evaluation of the basis for nodalization of a subcompartment. The following discussion will address the general procedure for nodalization of a containment subcompartment, with special emphasis on the reactor cavity, which is the annular volume between the reactor vessel and the primary shield wall. Since it is not feasible to prescribe an acceptable, specific nodal arrangement applicable to all configurations of the reactor cavity, nodalization in the reactor cavity will be discussed in detail to identify a modeling technique that will result in acceptable nodalization. The recommendations on subcompartment nodalization are based on a study (Ref. 27) of several different designs of reactor cavities; this study investigated the sensitivity of the results to nodalization arrangements and other subcompartment-code input parameters.

General Nodalization Procedures--The nodalization of a subcompartment should represent, as much as practicable, the physical geometry and flowpaths in a manner that is consistent with the assumptions used in the subcompartment-analysis code. Boundaries between control volumes, which represent junctions or vent paths, should be located at physical discontinuities where geometric influences are expected to create pressure differentials. In the subcompartment of a reactor cavity, there are many geometric variations that merit attention when developing the nodalization model. Examples of these are listed below:

(1) Nozzles and piping of reactor vessels (for example, hot leg, cold leg, core flood line)

(2) Supports of reactor vessels

(3) Instrumentation tubes (for example, neutron detectors)

(4) Shielding devices (for example, shield blocks, plugs, shield ring, or shield tank)

(5) Variations in diameter of reactor vessels

(6) Variations in profile of primary shield wall

(7) Entrances and exits of penetration of shield wall

In addition to nodalization in regions where there are flow-area changes, it is also necessary to define nodal boundaries and, hence, junctions in regions of constant cross-sectional area so as to account for frictional or inertial effects which may create significant pressure losses. These frictional effects include wall friction (fl/d) and turning losses.

A sample nodal model for a reactor cavity is shown in Figure 5. The reactor cavity as shown is folded out to represent the entire 360° circumferential span. The nodalization in Figure 5 was developed for design of a reactor cavity where only two geometric discontinuities were assumed to exist; namely, the flow-area change past the hot and cold legs of the reactor-coolant system and the variation in flow area encountered at the location of the change in diameter of the reactor vessel. These two geometric influences result in horizontal nodal boundaries at elevations A and B. The reason for including the horizontal boundary at elevation C was to permit a more accurate calculation of the pressure gradient in the long, constant-area, annular region below the nozzles of the reactor pressure vessel. The boundary at elevation C is established at the midpoint between elevations B and D. The junctions at the bottom would correspond to the area change in flowing to the lower reactor cavity. The top boundaries would represent either a closed top or area change for flow to volume(s) above the annulus of the reactor cavity.

Sensitivity studies (Ref. 27) show that the horizontal boundary at elevation C (shown in Figure 5) can be eliminated for certain cavity designs, when the boundary is established in the absence of geometric restrictions. These studies also showed that the resultant forces and moments (calculated about a horizontal axis through the nozzles--elevation A) changed by less than 5%. However, NRC is unable to conclude that such consolidation of nodes is uniformly acceptable for all cavity designs, since the impact of such consolidation depends on specific cavity dimensions and flowpaths. Therefore, an applicant should demonstrate a similar insensitivity by performing noding studies to preclude any need for this level of nodalization for a specific plant application.

The above discussion dealt with the criteria for establishing horizontal nodal boundaries in a model of a reactor cavity. In establishing vertical boundaries to account for pressure differentials created by circumferential flow about the reactor pressure vessel, attention should be directed again at modeling geometric discontinuities. For the design of the reactor cavity represented in Figure 5, the only geometric variations in the circumferential direction are those created by the piping of the coolant system. Therefore, the control

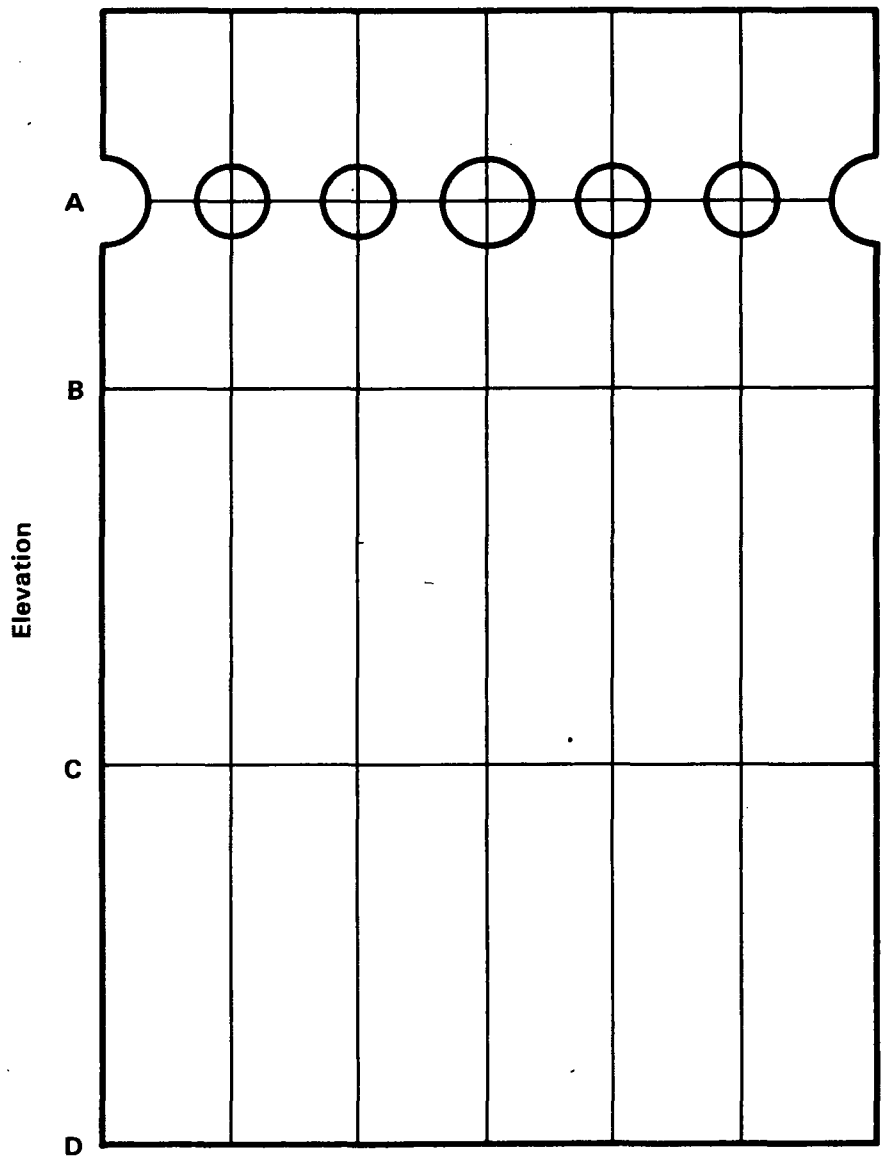

Figure 5. Sample Nodal Model for a Reactor Cavity.

volumes in Figure 5 encompass equal 60° segments about the reactor cavity region. This same level of noding detail was retained in areas located away from the nozzle region to adequately account for inertia and frictional effects (including wall friction and turning loss). Also, the greater degree of nodalization reduces the impact of the typical computer-code assumption that flowpaths are one-dimensional in nature. Cavity designs which include restrictions to circumferential flow such as support columns or instrument tubes of the pressure vessel should be modeled so that nodal boundaries reflect the presence of these restrictions.

Sensitivity studies (Ref. 27) have also been performed to evaluate the impact of the level of detail in circumferential nodalization. These studies demonstrated that nodalization could be made much coarser; that is, fewer circumferential nodes, without substantial changes in the calculated peak loads and moments on the reactor vessel. However, recognizing that sensitivity to nodalization detail will depend upon plant-specific geometry, it is recommended that circumferential noding for a design-basis model should include vertical boundaries through all nozzles of reactor pressure vessels for the height of the reactor cavity. Sensitivity studies on a specific plant may be performed by applicants in order to demonstrate acceptability for a model with less detail.

There are alternative approaches to development of an acceptable nodalization arrangement for subcompartment analysis of the reactor cavity. One approach would be to model the reactor cavity with detailed nodalization accounting for all obstructions, and resolution of detail in large, constant-area regions in the manner described above. In this case, if an approved subcompartment-analysis is used, further sensitivity studies are not required.

Another approach would be to model the cavity in less detail, but account for all pressure losses, and provide sensitivity studies demonstrating the acceptability of the design-basis model. Such sensitivity studies must compare the results of interest, whether it be the loading on subcompartment walls or the forces and moments on components. These studies may be performed on a model of either the plant under study or another plant having the same reactor cavity configuration and approximately the same dimensions.

Nodalization of Blowdown Volumes--A critical aspect of modeling the reactor cavity region is nodalization of the volume(s) into which the break effluent mass and energy are released. Development of criteria for the modeling of this control volume, commonly referred to as the break node or blowdown volume, is made more complex by the diversity of designs of reactor cavities.

For reactor cavity designs where the gap between the primary shield wall and the pressure vessel is large (approximately 2 ft) or where the shield wall is recessed in the vicinity of the nozzles, the break effluent should be assumed to enter the adjoining control volumes which have a projected area on the reactor pressure vessel. For an arrangement similar to that shown in Figure 5, the blowdown from a postulated guillotine rupture, neglecting jetting effects, would be assumed to enter the four control volumes surrounding the ruptured pipe.

There are some designs in which the location of the terminal ends of the piping of the reactor-coolant system is within nodes or control volumes that are away from the annular region of the reactor cavity. Thus, in the event of a postulated pipe rupture, the blowdown effluent is assumed to enter a control volume inside the primary shield wall (for example, pipe-penetration volume) or otherwise radially removed from the annular space immediately adjacent to the reactor vessel.

For models of reactor cavities where the break node is not assumed to be directly projected onto the reactor vessel, the applicant must clearly demonstrate the acceptability of the nodalization in the vicinity of the pipe break. This includes justification with sufficiently detailed drawings of the region of the vent areas away from the break node.

In selecting the break node(s), consideration must be given to the potential for, and effects of, fluid jetting from the site of the pipe rupture. This is important for all reactor-cavity designs but especially in those configurations where the pipe rupture is postulated to occur within a primary shield-wall penetration. For such designs the applicant must either justify that direct jetting into the reactor-cavity annulus would not occur or include the effect of jetting in calculating the loading condition on the reactor vessel. An acceptable approach is to demonstrate, if practicable, that pipe restraints would limit lateral separation in the event of pipe rupture to less than the pipe thickness.

3.2.2.2 Subcompartment Code Input Data

The following is a discussion of the data that are required for input into the computer codes used in subcompartment analysis. The data can be described as either control-volume data or flowpath data. Additional input parameters addressed are time-step data, blowdown data, and assumptions regarding the treatment of thermal insulation.

Control-Volume Data--The pertinent control-volume data are the free volume and the initial temperature, pressure, and relative humidity of control volumes. Initial conditions should be selected, within the range of anticipated values, to conservatively predict the result of interest; that is, loads on structures or loads and moments on primary system components.

The studies (Ref. 27) we have performed showed that the sensitivity to initial pressure, temperature, and relative humidity variations is insignificant (less than 5%). However, this study was done for a specific reactor cavity design and may not apply to other subcompartment (for example, steam-generator enclosure) analyses or analysis involving different reactor cavity designs.

Uncertainties in the calculated free volume of each node shall be applied in such a way as to obtain the most severe loading condition of interest.

Flowpath Parameters--The flowpath data of interest in modeling a flowpath between control volumes typically include the flow area, loss coefficients, and the inertia term (L/A ft-1).

The flow area that is used as input to subcompartment codes should be based on the minimum flow area between control volumes. This is consistent with the basic premise of subcompartment nodalization which is to place boundaries at flow restrictions. Also, critical flow models are based on the minimum flow area.

Uncertainties in flow areas should be applied in a manner that yields conservative results of interest. For example, if one would calculate lateral loads on the reactor vessel, a low estimate of the flow areas should be used for flowpaths that relieve the pressure near the break, for example, circumferential flowpaths. However, if the vertical location of the resultant lateral force, that is, the moment, is of concern, then maximization of appropriate flow areas for vertical flow on the same side of the reactor vessel as the break should be considered.

If the subcompartment model includes vent areas that become available as a result of fluid flow or pressurization effects after initiation of the accident, the following criteria apply:

(1) The modified flowpath parameters (for example, vent area, resistance) as a function of time or pressure after the break should be based on analysis of the subcompartment pressure response.

(2) Availability of vent areas resulting from pressure or flow effects should be justified analytically or experimentally.

(3) The potential for, and effects of, missiles that may be generated during the transient should be evaluated.

Loss coefficients are conventionally used to account for total pressure losses due to changes in flow area, turning losses, and wall friction. Coefficients of hydraulic loss are available in the literature; the most commonly used comprehensive itemization is found in Reference 28. Applicants should provide a discussion of the methods used to determine loss coefficients and a representative sample calculation of the loss coefficients.

Consideration should be given to the direction of flow when selecting loss coefficients for a given flowpath, since values may change depending on the direction of flow. If the analysis is done with a subcompartment code that does not account for effects of flow direction, the specified flow direction that is used as input must be verified by the analysis.

Attention should also be given to ensuring the consistency between the flow area and loss coefficients that are used as input data. For loss coefficients based on an area other than that used as input to the code, the loss coefficients must be appropriately modified (by the ratio of the areas squared). It is recognized that loss coefficients used in subcompartment analyses are based on hydraulic, steady-state flow and, as such, do not reflect consideration of the complexities associated with a compressible two-phase, two-component, transient-flow problem. However, based on the reactor-cavity studies (Ref. 27) NRC staff has performed to date, the sensitivity of reactor-cavity analysis to

the uncertainties associated with the steady-state, hydraulic loss coefficients is small. Therefore, the current method of selecting loss coefficients in subcompartment analysis of reactor cavities is judged to be acceptable.

Fluid inertia has a significant effect on the pressure gradients calculated in subcompartment analyses (Ref. 27). Inertia is typically accounted for by use of the geometric (L/A ft-1) as input, which results from solution of the momentum equation. Currently, acceptable practice for calculation of the inertia term is outlined in Reference 27.

Time steps used to calculate the transient distribution of mass from the pipe rupture should be sufficiently small (a decrease in the time-step size should be done with the aid of sensitivity studies on a specific plant or by comparison with studies on similar plants).

3.2.2.3 Other Factors

Insulation Behavior Considerations--Thermal insulation on piping and vessels must be considered when performing subcompartment analysis. The relative importance of assumed insulation behavior is magnified for the reactor-cavity analysis since thermal insulation may comprise a significant volume within the reactor-cavity annulus and the penetrations through the shield wall.

Assumptions related to postaccident insulation behavior must either be shown to be conservative or, if insulation behavior is treated mechanistically, such behavior must be justified analytically or experimentally.

In either case, the applicant must provide justification for the assumed behavior of insulation and a detailed discussion of how insulation affects volumes and flow areas both in the reactor-cavity annulus and in those flowpaths away from the reactor cavity (for example, penetrations of the shield wall).

Blowdown Mass and Energy--The driving potential for asymmetric pressure loads is the blowdown from a postulated pipe rupture. The blowdown characteristics are typically input to the subcompartment computer codes in the form of a mass-flow addition rate as a function of time and a corresponding enthalpy- or energy-addition rate as a function of time. On the basis of studies performed by the staff (Ref. 27) and a review of applicants' analyses, we find that the blowdown mass-and-energy release rates are important parameters in the calculation of asymmetric loads and moments. Analyses have shown that the sensitivity of peak calculated loads and moments of reactor pressure vessels can be directly proportional to the short-term mass-and-energy release rates. Therefore, subcompartment analysis should be based on conservatively derived blowdown mass-and-energy release rates.

In general, calculations of the mass-and-energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment subcompartment response. The analytical approach used to compute the mass-and-energy release profile for the first several seconds of blowdown for use in subcompartment analysis is acceptable if the volume noding of the piping

system is similar to those of an approved emergency-core-cooling-system (ECCS) analysis or if appropriate noding-sensitivity studies have been made. The computer programs that are currently acceptable include SATAN-V (Ref. 29), CRAFT-2 (Ref. 30), CE FLASH-4A (Ref. 31), and RELAP4 (Ref. 32), when a flow multiplier of 1.0 is used with the applicable choked-flow correlations for the above computer codes. An alternate acceptable approach is to assume a constant blowdown profile using the initial conditions with the choked-flow correlations in the above references.

3.2.2.4 Force-Moment Calculations

The calculation of forces and moments, as applicable, on components of reactor systems or on subcompartment structures is the culmination of the subcompartment analysis and, therefore, requires the same detailed consideration as that for subcompartment modeling.

The following requirements apply to the translation of calculated pressure gradients into forces and moments.

(1) The subcompartment nodal model used for the calculation of forces and moments should be the same model as that found acceptable for the calculation of pressure gradients. If the model used to calculate forces and moments is different from the model used to calculate pressure gradients, then the calculated forces and moments shall be shown to be equivalent to those forces and moments that would have been calculated had the other model been used.

(2) Projected areas onto a curved surface shall be based on projected planar areas. Then multiplication of the projected area by the calculated nodal pressure gives the force acting normal to the surface through the area centroid.

(3) Force calculations for components such as the reactor pressure vessel shall include loads resulting from differential pressures acting across the vessel piping and nozzles.

4 STRUCTURAL EVALUATION

The structural evaluation of the primary system for asymmetric LOCA loads involves the development of detailed finite element models, the application of the loading transients (discussed in Section 3) to these models, and finally the assessment of the resulting structural response against appropriate acceptance criteria. The finite element models required for this evaluation are typically large and complex and may incorporate both nonlinear (geometric) and inelastic (material) modeling aspects. Based on these considerations, the staff chose an audit approach to verify various modeling assumptions and sensitivity.

The staff evaluated six PWR primary systems both to gain a better understanding of the primary-system response and to develop the capability for performing an audit of plant-specific analyses. The six plants chosen represented two from each NSSS vendor: one using conservative elastic-analysis techniques and one incorporating inelastic-and nonlinear-analysis techniques. A detailed discussion of each of these six plant-audit calculations, listed below, is provided in References 33 through 38.

- Westinghouse Lead Plants

  North Anna 1--elastic analysis (Ref. 33)
  Indian Point 3--inelastic analysis (Ref. 34)

- Combustion Engineering Lead Plants

  San Onofre--elastic analysis (Ref. 35)
  St. Lucie 1--inelastic analysis (Ref. 36)

- Babcock and Wilcox Lead Plants

  Erie--elastic analysis (Ref. 37)
  Arkansas 1--inelastic analysis (Ref. 38)

A discussion of modeling considerations and review procedures for the structural evaluation are provided in Section 4.1 and structural acceptance criteria, although not developed as part of this task action plan, are provided in Section 4.2.

4.1 Primary-System Analytical Development

4.1.1 Modeling Aspects

Structural analysis of reactor primary coolant systems is performed to demonstrate that the plant can be safely shut down in the unlikely event of a postulated LOCA. Structural analysis for purposes of this discussion is defined as the specification of mechanical response (loads, deflections, and stresses) of the primary coolant piping, major components and their supports, and reactor internals and fuel. Detailed finite element models are required to predict these mechanical responses accurately, although more-conservative simplified techniques may be applicable in special cases.

Modeling techniques for this analysis are not unique. They vary depending on the computer codes, the geometry, and the analytical assumptions. Primary systems responses, in general, are nonlinear in nature due to geometrical gaps and various design characteristics of supports, such as cables which act in only one direction. The analytical models may also require inelastic considerations to reflect material nonlinearities.

A single model representing the entire primary system and incorporating sufficient detail is not practical within current computer and economic limits. The structural analysis is, therefore, completed in phases in which the responses from a general system model are used as input to detailed structural models (subsystems) of critical parts of the primary system. The system model typically includes a representation of the reactor pressure vessel, its core barrel, the internals and fuel assemblies, the primary coolant piping, the steam generator and supports, and the primary-coolant pump and supports. Typical subsystems which are evaluated using input from the systems model are: attached ECCS piping, detailed models of RPV and component supports, and detailed evaluation of the fuel integrity.

Basically, the preceding paragraphs have been oriented toward a finite element model of a single reactor system. Finite element representation of the primary coolant system using a number of completely decoupled models may also constitute an acceptable approach. As an example, a reactor-coolant system could be represented by the following three separate finite element submodels:

(1) Reactor vessel internals and vessel supports
(2) Broken piping loop, components, and component supports
(3) Intact loop, component, and component supports

If this approach is pursued, the interaction of the various submodels must be accounted for in the evaluation.

A general discussion of modeling aspects is provided in this section as well as a detailed discussion of one acceptable modeling technique (used by NRC staff for the Indian Point 3 audit) to illustrate modeling considerations.

4.1.1.1 Vessels

Rigid-body representation of the reactor vessel and steam-generator vessel is acceptable for the LOCA analysis. Models of the vessel shell should include sufficient noding detail to allow application of the time-history loads, incorporation of supports and vessel internals, and determination of boundary conditions for attached ECCS systems. Nozzle flexibilities should also be included at appropriate locations in the models. The models should represent the mass distribution and inertia characteristics of the vessels. Mechanisms of the control-rod drive should also be included in the RPV model.

4.1.1.2 Component Supports

Major component supports are, in some cases, very complex structures. Simplified representation of these structures in an analysis of the reactor-coolant system is acceptable (where significant nonlinearities do not exist) and, in

some cases, necessary. Constraints in the size of the finite element model may make it necessary to incorporate a simplified representation of the steam generator and pump supports.

In modeling the various component supports, substructuring may be a valid means of reducing the size of the overall model. To explain the term "substructuring" as used in this report, consider the following example for a main coolant pump with a complex structure or framework of beams comprising its support to the foundation. A separate analysis of the support structure can be conducted with loads applied at the locations where the pump attaches to its support structure. If force versus deflection curves and moment versus rotation curves can be developed for this "substructured" support and the mass characteristics (natural frequencies) can be reasonably represented, then the complex support system can be replaced with a simpler support system whose load versus deflection, moment versus rotation, and mass characteristics are the same as those for the complex system. The net result is then a simpler, smaller model for the system analysis.

4.1.1.3 Internals

The goal of the internal portion of the system model is twofold; the first is to determine the structural response of the core barrel and core supports; the second is to provide appropriate input for a detailed evaluation of the fuel (discussed in Chapter 5). The internal structures and terminology may vary, depending upon NSSS vendor designs; but, in general, the internals model should include the following:

(1) Core barrel
(2) Thermal shield
(3) Core-support structure
(4) Core
(5) Upper plenum assembly
(6) Guide-tube assembly

Modeling of these structures in the system model should include sufficient detail to represent the mass distribution and inertia effect. Attention should be given to the review of interface areas (such as the core-barrel-to-RPV connection) which are part of the load path, to ensure an accurate modeling expectation.

4.1.1.4 Primary Piping

Representation of the piping should be sufficiently detailed to calculate the maximum stress, permit representation of pipe supports, and provide input for structural analysis of ECCS and other piping subsystems that connect to the primary coolant system.

4.1.1.5 Indian Point Unit 3 Audit Model

A summary of the modeling techniques used for the Indian Point Unit 3 (Ref. 34) lead plant audit is provided to illustrate the modeling detail and development. The primary system of Indian Point Unit 3 is represented by an assemblage of

linear elastic, nonlinear elastic, and elastic-plastic truss and beam elements. Finite element models used in staff audit of Indian Point Unit 3 are represented in Figures 1-9. As shown in Figure 6, the steam-generator supports are represented by equivalent beams analogous to the steam-generator supports. These equivalent structures were derived from the flexibility matrices and vibration modes of detailed support models. The primary piping loop models, Figure 7, consist of an elastic-plastic beam representation of the primary piping with an elastic beam representation of the pump, steam generator, and main steam piping. Figures 8 and 9 show the RPV and internals and core models. The RPV and internals are represented by elastic beams, and the core is represented by elastic beams interconnected by nonlinear elastic truss elements or gap elements. With this model, the fuel, represented by elastic beams, is allowed to impact both the core barrel and adjacent fuel. The RPV supports shown in Figure 8 are represented by a horizontal elastic-plastic truss element and a vertical nonlinear elastic or compression-only truss element. The remaining structural components, the hot stops and snubbers on the steam-generator supports, the base restraints of the pump supports, the tie rods, and the primary piping restraints are represented by either nonlinear elastic or elastic-plastic truss elements.

4.1.2 Review Procedure

The structural evaluation of the primary system must include, as a minimum, an analysis of the following:

(1) Reactor pressure vessel
(2) Fuel assemblies, including grid structures
(3) Control and rod drives
(4) ECCS piping that is attached to the primary coolant piping
(5) Primary coolant piping
(6) Reactor-vessel, steam-generator, and pump supports
(7) Reactor internals
(8) Biological shield wall and neutron shield tank (where applicable)
(9) Steam-generator compartment wall

The review and determination of the adequacy of a plant-specific analysis for asymmetric loads by the staff should include an assessment in the following areas: the structural computer codes, the modeling techniques and assumptions, and the acceptability of the responses.

The structural computer codes and modeling techniques and assumptions are reviewed and approved as a package. Two approaches for performing this review have been developed by the staff:

(1) Review and approve a general analytical technique applicable to more than one plant.

(2) Perform an independent audit of a specific plant evaluation.

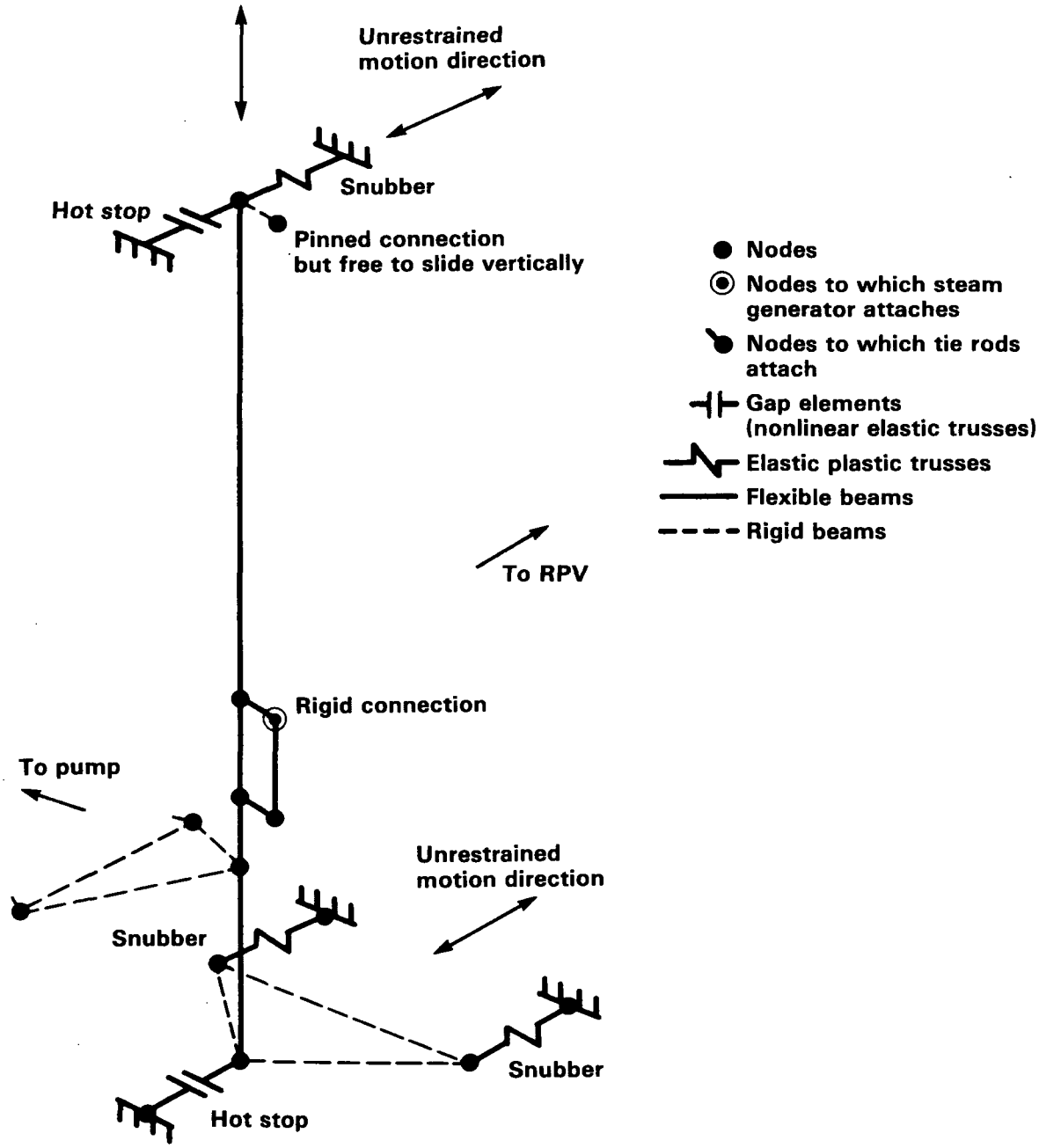

Figure 6. Steam Generator Support.

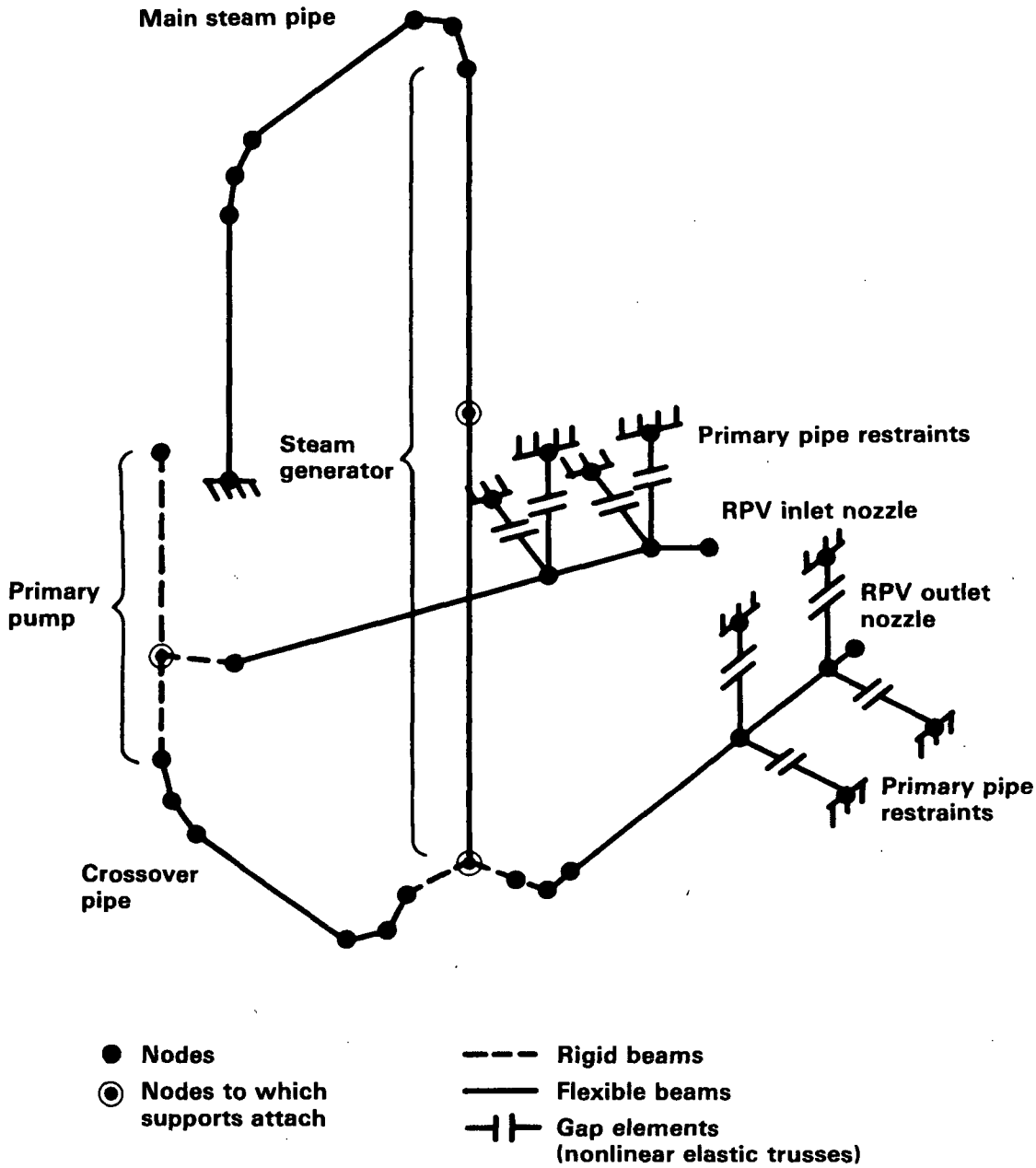

Figure 7. Typical Piping-Loop Model.

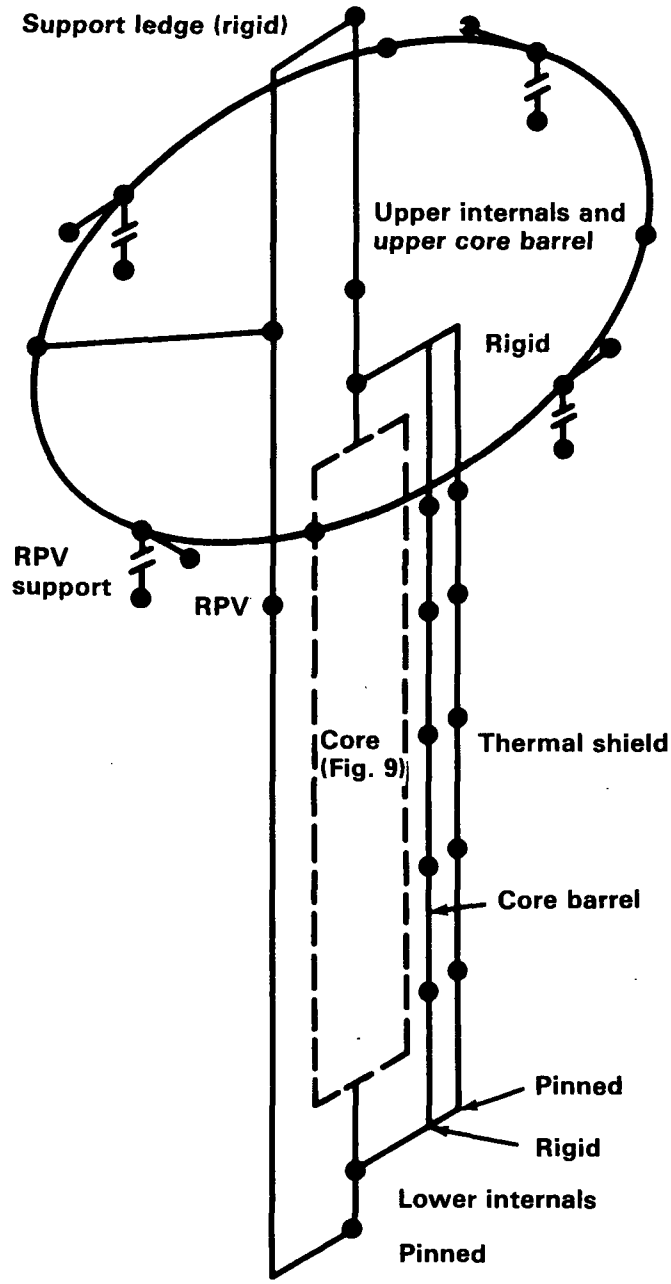

Figure 8. Reactor Pressure Vessel and Internals Model.

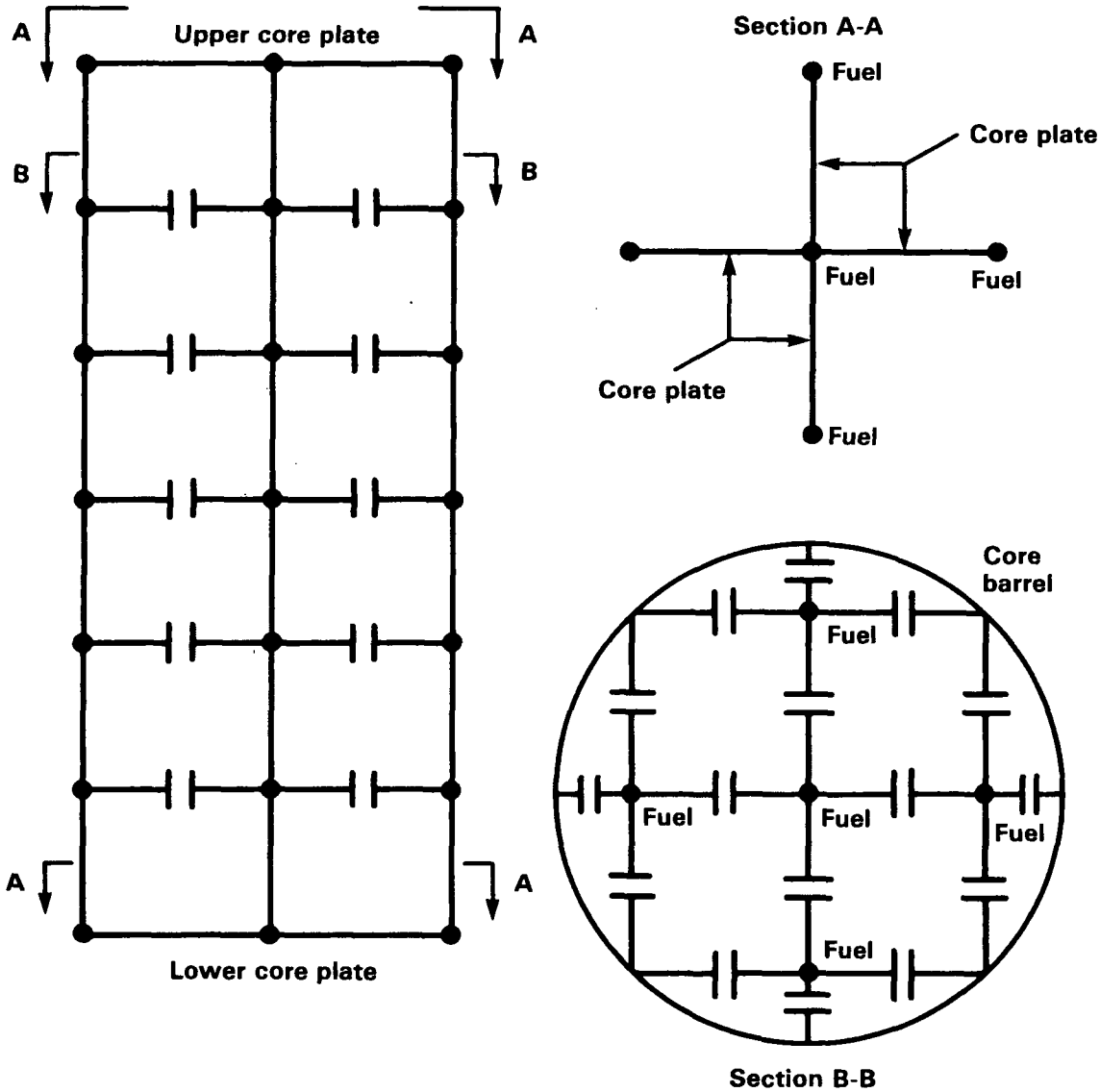

Figure 9. Core Model.

4.1.2.1 General Analytical Technique

Review and approval of a general analytical technique is applicable when a number of plants are evaluated using the same computer codes and modeling considerations. This approach was chosen by the majority of operating plants who engaged their NSSS vendors to perform these evaluations. The review is performed in two steps: first, the general analytical technique is reviewed and approved and, second, each plant analysis is reviewed for consistency with the approved general approach.

Typically, the general analytical technique consists of the structural computer codes, the analytical modeling, and the result from a representative analysis. The representative analysis may be of a specific plant or a generic plant and should identify selected user options in the computer code and provide the modeling detail and assumptions. The staff typically determines the acceptability of the general analytical technique by performing an independent audit evaluation, discussed in Section 4.1.2.2. The audit evaluation may not be required if sufficient noding-sensitivity studies are performed on the structural models and the computer codes have been verified as acceptable by the staff.

The staff has currently reviewed and approved the general analytical approach submitted by Westinghouse for the Indian Point 3 nuclear plant (Ref. 39). Details of the audit evaluation are presented in an EG&G report (Ref. 34). The staff has developed audit models for both a Combustion Engineering plant (Ref. 34) and a Babcock and Wilcox plant (Ref. 38). Their general analytical approach will be evaluated when their respective plant reports are submitted for staff review.

Basically, evaluation of each plant submittal involves reviewing the structural models for detail and completeness and applying the structural computer codes for consistency with the approved general approach. Review of the modeling should include the relative noding detail, the boundary conditions, and the noding assumptions. The computer code version and user-specific options should also be verified to be consistent with the approved approach.

4.1.2.2 Independent Staff Audit

The second approach is to perform an independent audit of the plant using a staff-approved computer code and staff-developed models. Plant analyses which either do not use an approved general analytical approach or modify an approved approach significantly, may require an independent staff audit. Exceptions may exist where sufficient supporting documentation is provided to enable the staff to make an evaluation. Review for acceptability is fairly simple in this approach and typically requires a comparison of the response characteristics. The information required from the licensee to perform the audit is summarized below.

Piping--Geometric layout dimensions for piping centerline are required so connectivity from the RPV to the steam generator (hot-leg segment), steam-generator-to-pump inlet (crossover segment), pump outlet to RPV inlet (cold-leg segment), can provide for proper interactions of the piping with its attached

components. Section properties for each segment of piping must be available from the drawings so proper stiffness effects can be included. Materials lists for all segments should be provided along with normal operating conditions (pressure and temperature) for each segment. Pipe supports must be well defined with regard to location and type (material, stiffness, load limit, behavior above load limit--failure versus constant load, action--one directional only or either direction, geometric gaps).

Steam Generators--Generally, section properties of the shell must be available so the effective stiffness of the steam generators can be modeled. Additionally, weights of steam-generator components should also be provided (upper and lower head tube sheets, tubes, water contained). Total weight of the steam generator (dry and including normal operation water) should be available to provide reasonable mass distribution for the steam generator. Steam-generator support details must also be available, including support-to-foundation interface to provide a useful form of input for the analyst.

Pumps--The weight (including water weight), center-of-gravity location, and rotary inertia should be provided for the pumps in order to represent their effects on the piping. Pump supports should be defined and located including snubbers and their behavior (that is, their physical characteristics: load rating, behavior on exceeding rated load and failure load, and preload, if any). For complex pump supports to foundation, details must be provided to allow mass and stiffness properties to be included for that part of the system.

Reactor Pressure Vessel--Details of the RPV shell must be available since stiffness properties for the RPV are normally based on properties of the shell section. Weights should be provided (including water weights and their distribution) to represent the proper mass distribution of the RPV. Support details and support-to-foundation details and information should be provided, as for steam generators and pumps.

RPV Internals--Detailed and assembly drawings should be supplied so interactions of various, major internal components to the RPV can be modeled. These drawings should include section properties and material properties for the fuel modules and internals of the core barrel. Weights for these components should also be supplied. In addition, weights of core plates and of remaining internals should be supplied, so proper mass distribution in the RPV can be included.

Water weights for the various regions inside the RPV and fuel frequencies should be supplied.

The final review consideration is the review of the structural response against applicable acceptance criteria (Section 4.2).

4.2 Structural Acceptance Criteria

The structural integrity of the primary system including the reactor pressure vessel, RPV internals, primary coolant loop, and components must be evaluated against appropriate acceptance criteria to determine if acceptable margins of

safety exist. Allowable limits and appropriate loading combinations are set forth in standard review plans (SRPs), which are listed in the table that follows. The staff recognizes that in some specific cases, where "as built" designs are being reevaluated for asymmetric LOCA loads, these design limits may be exceeded. Acceptance of alternative allowable limits will be based on a case-by-case evaluation of the safety margins.

Load combination criteria in general were not addressed as part of this study. Currently the staff requires that seismic (SSE) and LOCA response be combined, along with responses due to other loading as specified by the SRP. An acceptable method for combining elastically generated seismic and LOCA responses is provided in NUREG-0484 (Ref. 40). Acceptable methods for combining response generated by an inelastic LOCA analysis and elastic seismic analyses will be evaluated on a case-by-case basis.

Item	SRP
Reactor pressure vessel	3.9.3
Reactor internals	3.9.5, 3.9.1
Primary coolant loop piping	3.9.3
ECCS piping	3.9.3
RPV, SG, pump supports	3.8.3
Biological shield wall	3.8.3
Steam-generator compartment wall	3.8.3
Neutron-shield tank	3.8.3

5 FUEL-ASSEMBLY EVALUATION

Although core components are included in the structural analysis of the primary system, a more-detailed structural analysis of the reactor fuel assembly is performed to show that the fuel is maintained in a cool condition and that insertion of control rods is not prevented. These are basic requirements that follow from the General Design Criteria (10 CFR Part 50) and are described explicitly in Section 4.2 of the Standard Review Plan (SRP).

To perform the detailed analysis for fuel-assembly structural response, input information is obtained from results of the structural analysis of the primary system. The input information includes primary-system motions (for example, core plate, core shroud, and fuel-alignment plate) and transient pressure differences as described earlier. Since the basic requirements for fuel integrity and the input motions are similar to those for the seismic analysis, and since the fuel analysis is done separately from the primary-system analysis, unified guidelines for the fuel structural analysis have been developed. These guidelines are described in a proposed Appendix A to SRP Section 4.2. The SRP Appendix is included as Appendix E of this report and has been issued for public comment (Federal Register, Vol. 45, No. 40, 2/27/80, p. 12939).

Appendix A of SRP Section 4.2 addresses seismic loads of fuel assembly and structural loads of LOCAs as well as offering an acceptable method for combining these responses for PWRs and BWRs. It thus includes, but goes beyond, requirements related to asymmetric blowdown loads of Task A-2. The SRP Appendix details analytical methods for predicting structural loads on fuel assembly components (for example, spacer grids, fuel rods, and guide tubes), determination of component strengths, and acceptance criteria for comparing predicted loads with component strengths. The main points of the acceptance criteria are summarized below; see Appendix E for details.

(1) Fuel rod (poison rod, control rod, guide tube, etc.) fragmentation does not occur. Allowable material strength values used in design should bound a large percentage (about 95%) of the distribution of strength values.

(2) LOCA temperature and oxidation limits of Title 10 of CFR 50.46 for the fuel-rod cladding are not exceeded. If the fuel-assembly grid-crushing strength P_{crit} (the force required to permanently deform the grid) is not exceeded, the standard LOCA analysis is performed; otherwise the effects of a deformed grid are accounted for in the LOCA analysis. The allowable value of P_{crit} is a mean of the distribution of measured P_{crit} values.

(3) Seismic and LOCA loads are combined by taking the square root of the sum of the squares (SRSS).

Technical background for the SRP Appendix is given in three reports referenced in that appendix.

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APPENDICES

APPENDIX A

TASK ACTION PLAN A-2

ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEM

Lead NRR Organization:	Division of Operating Reactors (DOR)
Lead Supervisor:	Darrell G. Eisenhut, Acting Director, DOR
Task Manager:	Steve B. Hosford, EB/DOR
Applicability:	Light-Water Reactors (PWRs)
PWR Project Completion Date:	October 1979

1. DESCRIPTION OF PROBLEM

On May 7, 1975, the NRC was informed by Virginia Electric & Power Company that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by Westinghouse or Stone and Webster in the original design of the reactor vessel support system for North Anna, Units 1 and 2. In the event of a postulated LOCA at the vessel nozzle, asymmetric LOCA loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more detailed analytical models, it became apparent to Westinghouse that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports, thereby affecting their integrity. Although first identified at the North Anna facility, this concern has generic implications for all PWRs.

Upon postulation of a break in a reactor coolant pipe, at the above mentioned locations, several rapidly occurring events could cause internal and external transient loads to act upon the reactor vessel. For the reactor vessel pipe break at the inlet nozzle, asymmetric pressure changes take place in the annulus between the core barrel and the vessel. Decompression could occur on the side of the vessel annulus nearest the pipe break before pressure on the opposite side changes. The momentary momentary difference in pressure across the core barrel could induce lateral loads in opposite directions on the core barrel and the reactor vessel. Vertical loads could also be applied to the core internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, as fluid escapes through the break, the annulus between the reactor vessel and biological shield wall could become asymmetrically pressurized resulting in a difference in pressure across the vessel causing additional horizontal and vertical external loads on the vessel. In addition, the vessel could be loaded by the effects of initial tension release and blowdown thrust at the pipe break. The loads occur simultaneously. For a reactor vessel outlet break the same type of loadings could occur, but the internal loads would be predominantly vertical due to more rapid decompression of the upper plenum.

For each of the above-mentioned postulated breaks, the time history of the reactor vessel support reactions due to the complete set of simultaneous horizontal and vertical loads should be calculated.

In the event that such loadings would result in a significant degree of failure within the reactor pressure vessel support system and consequent vessel movement, there is a potential that this could (1) result in damage to the ECCS lines, (2) affect the capability of the control rods to function properly, and (3) result in damage to other reactor coolant

system components (pump and steam generator supports). In addition, the differential pressures occurring during subcooled blowdown could result in stresses on fuel assemblies caused by lateral core barrel and core plate motion. This could degrade heat transfer capability if fuel spacer grids are deformed by impacting either each other or the core baffle. (This loading can occur independently of vessel support failure.)

The above-described phenomena may also exist to some degree in BWR plants but is expected to have a lesser safety significance due to the lower operating pressures. Currently, a plan of resolution for the BWR plants is being developed by DOR:EB and will be considered as a separate issue outside the scope of this TAP.

2. PLAN FOR PROBLEM RESOLUTION

A. Background

Following disclosure of this problem during the OL review of North Anna Units 1 and 2, a survey of all operating PWR reactors was conducted in October 1975. That survey showed that neither of the above described transient differential pressures had been considered in the design of the reactor vessel supports for any operating PWR facility.

In June 1976, the NRC requested all operating PWR licensees to proceed to assess the adequacy of the reactor vessel supports at their facilities with respect to these newly-identified loads. Most licensees having a common NSSS vendor took identical or similar positions with respect to this request and did not respond as requested.

Most licensees with Westinghouse plants proposed an augmented inservice inspection program (ISI) of the reactor vessel safe-end-to-end pipe welds in lieu of providing the detailed analysis we requested. Licensees with Combustion Engineering plants submitted a probability study (prepared by Science Applications, Inc.) in support of a conclusion that a break at this location has such a low probability of occurrence that no further analysis is necessary. B&W licensees have engaged Science Applications, Inc. for a similar study.

When the W and CE owners group reports were received in September 1976, DOR formed a review Task Group consisting of members from DOR, DSS and EDO to evaluate these alternate proposals. In addition, EG&G Idaho, Inc. was contracted to perform an independent review of the submitted probability study. A short review schedule was established since it appeared that most licensees would hold off on further analysis pending our consideration of their submittals.

Our review of the proposed alternates has been completed. The Task Group and EG&G independently reached the same conclusion: that the alternate proposals set forth in these reports should not be accepted

in lieu of the requested analyses. The basis is that a sufficient data base does not exist within the nuclear industry to provide satisfactory answers to many information needs we identified. This information would be needed to support this "no-break" approach. Further investigation of pipe break probabilities is planned by the staff, see item (3) below.

B. Plan

(1) Letters will be sent to all licensees and applicants stating that an analysis must be undertaken to assess the design adequacy of the reactor vessel supports and other structures to withstand the loads when asymmetric LOCA forces are taken into account.* We will point out that it may be possible to group plants such that only a limited number of plants need be analyzed, and that it may be possible to provide a simple "fix" (e.g., pipe restraints) which will permit bounding the problem. Therefore, the letters will request licensees and applicants to submit their schedule for completion of the task.

(2) The staff will meet with the licensees constituting both the W and the CE owners group to explain why the probability study reports could not be accepted. We will also provide them all the questions that have been generated to date as a result of our review of the W and CE topical reports. (We will not issue formal requests for additional information on these topicals to these groups of licensees.)

(3) We will review and approve vendor models and codes prior to plant-specific application to the extent possible. The staff or its consultants will audit the calculations on a plant-specific basis where a pre-approved code is not used (the case-specific reviews are not part of this TAP).

(4) The staff will develop explicit guidelines and acceptance criteria for the asymmetric LOCA load analysis, including load combinations and acceptable alternatives where, depending on the construction or operating status of a given plant, application of the guidelines per se could require modifications that are judged to be a practical impossibility. Such alternative guidelines would be designed to provide an adequate and acceptable LOCA load generic issue consistent with safe plant shutdown requirements.

(5) The staff will conduct a pipe break probability study that will encompass (a) advances that are being made in nondestructive

*Including an assessment of asymmetric loads produced by large pipe breaks outside the reactor vessel cavity.

examination techniques, (b) an improved flaw distribution data base of actual NSSS materials, and (c) development of a realistic break opening model to describe pipe break characteristics. The pipe break probability study will be used to confirm the adequacy of staff decisions related to the continued operation of plants for the interim period until an analysis of these loads is conducted.

(6) The staff will perform a series of sensitivity studies to independently evaluate the effect of noding upon the magnitude and distribution of pressures within typical reactor cavity designs. Results of sensitivity studies will be utilized to prepare guidelines for the evaluation of the volumes within the confines of the reactor cavity.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

The safety issue addressed by this Task Action Plan (TAP) is primarily applicable to pressurized light water reactors (PWRs). As discussed in Section 1, the potential asymmetric loading conditions for boiling water reactors (BWRs) appear to be of a lesser safety significance. The potential loads are not expected to be as large, since the reactor vessel pressure and sub-cooling are substantially lower than for PWR plants (i.e., 1000 psia and approximately 40 Btu/lb, respectively for BWRs compared to 2200 psia and 100 Btu/lb for PWR plants).

With respect to PWRs, each NSSS vendor (i.e., Westinghouse, Babcock and Wilcox, and Combustion Engineering) has submitted topical reports which describe analysis methods for assessing the internal asymmetric blowdown loads. In accordance with the schedule for completion of this TAP, staff review and approval of these topical reports will be completed by December 1978 (the Westinghouse analysis methods have already been approved by the staff).

Topical reports have not been submitted by all the vendors and architect engineers that deal with asymmetric cavity and reactor vessel external loads in PWRs. These reviews are conducted by the staff on a case by case basis. Technical assistance programs are underway to resolve those aspects related to the reactor vessel cavity asymmetric loads and structural response methods and criteria. It is expected that these programs will be completed in FY 79 and will provide acceptable methods and models for independent staff analysis. Current staff evaluations include case-by-case reviews of applicants' mass and energy blowdown calculations, reactor cavity loads including pressurization and vessel internal loads, and the resulting structural response. The results and conclusions drawn from these ongoing staff activities are subject to confirmation in the course of development of the generic methods and models that will be the output of this generic task.

For PWRs currently under licensing review for a Construction Permit, the staff is requiring applicants to commit to address this safety issue as part of the subsequent applications for Operating Licenses. The CP applicants are also required to perform plant-specific analyses of mass and energy release and reactor cavity pressurization for purposes of establishing asymmetric design loads at the CP stage while the generic methods are being reviewed and developed as part of this task. Since staff approval of the analysis methods is anticipated well in advance of the time for consideration of this safety issue at the Operating License review stage for plants now under CP review and since necessary modifications,* if any, to the plant design can be accomplished while the plant is being constructed, the staff has concluded that, pending completion of this task, Construction Permits can be granted with reasonable assurance that (1) there will be a satisfactory resolution of this concern prior to operation, and (2) operation will not present undue risk to the health and safety of the public.

CP applicants are required to perform plant specific analyses using the best available methodology for their designs. Also, the staff reviews and performs audit calculations to ensure the reasonableness of the currently available models; e.g., mass and energy blowdown calculations, cavity pressure loads, and methodology for load combinations.

For PWRs under review for an Operating License, the staff will require applicants to perform a plant specific analysis of their facility using the best available methodology and criteria for their design and to implement any design modifications which are necessary prior to the issuance of an Operating License. Applicants are also required to commit to perform an evaluation of their facility following staff approval of the generic analysis method should the methodology or criteria developed from this task warrant such reevaluation. Based upon this commitment and the rationale presented below for continued operation of licensed facilities, we have concluded that, pending completion of this task, Operating Licenses can be granted for facilities within this category with reasonable assurance that operation will not present an undue risk to the health and safety of the public.

*Such modifications, i.e., installation of physical restraints to limit the postulated break area within the reactor cavity and/or to provide additional piping support, have been implemented late in the construction stage of several facilities (North Anna Units Nos. 1 and 2, Farley Unit No. 1). Modifications of this type can also be accomplished after construction is completed, as evidenced by the modifications proposed by the licensee for Indian Point Unit No. 3, an operating facility. It should be noted, however, that physical space (access) limitations at certain older operating reactors may preclude the accomplishment of such modifications.

For PWRs currently licensed for operation, we have concluded that there is reasonable assurance that continued operation, pending completion of this task, does not constitute an undue risk to the health and safety of the public for the following reasons.

As discussed below, the likelihood of occurrence of an initiating event of sufficient magnitude to seriously challenge the structural adequacy of the vessel support members or other structures is low.

The disruptive failure of a reactor vessel itself has been estimated to lie between 10^{-6} and 10^{-7} per reactor year -- so low that it is not considered as a design basis event. The rupture probability of pipes is estimated to be higher. The data base used by WASH-1400 indicates a median value of 10^{-4} for LOCA initiating ruptures per plant-year for all pipe sizes 6" and greater (with a lower and upper bound of 10^{-5} and 10^{-3}, respectively). We believe that considering the large size of the pipe in question (up to 50" O.D. and 4-1/8" thick), a median value nearer 10^{-5} than 10^{-4} is more appropriate using the same data base. In addition, the quality control of the piping used in nuclear power plants is somewhat better than that of the piping used in the WASH-1400 data base.

Because (1) the break of primary concern must be large (and therefore of low probability); (2) only certain break locations lead to high loads; and (3) these welds are currently subject to preservice and inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI, we conclude that the probability of a pipe break resulting in substantial transient loads on the vessel support system or other structures is acceptably small and that reactor operation and licensing of facilities for operation can continue during the interim period of approximately 2 years while this matter is being resolved.

It is anticipated that the plant-unique analyses, which will be performed following staff approval of the generic analysis methods, will indicate that design modifications may be necessary to restore the originally intended safety margins at certain operating facilities. Such modifications, i.e., installation of physical restraints to limit the postulated break area within the reactor cavity, have already been shown to be feasible. In the event that design modifications are judged to be impossible for specific operating facilities, alternative solutions, including such things as augmented inservice inspection, will be required to ensure adequate safety margins.

4. NRR TECHNICAL ORGANIZATIONS INVOLVED

A. Analysis Branch, Division of Systems Safety. Has lead responsibility for review of vendor hydrodynamic analysis methods and codes.

Manpower Estimates: 0.5 man-months in FY 1979

B. Core Performance Branch, Division of Systems Safety. Has lead responsibility for reviewing vendor analysis methods for calculating loads on fuel assemblies resulting from decompression for plants under CP and OL review.

Manpower Estimates: 1.75 man-months in FY 1979

C. Containment Systems Branch, Division of Systems Safety. Responsible for reviewing vendor models and methods for calculating asymmetric cavity loads for all plants, and associated vendor models.

Manpower Estimates: 4.6 man-months in FY 1979

D. Mechanical Engineering Branch, Division of Systems Safety. Responsible for review of structural aspects of vendor analysis methods and codes for plants not licensed for operation. Responsible for developing structural acceptance criteria (with Engineering Branch, DOR).

Manpower Estimates: 1.5 man-months in FY 1979

E. Engineering Branch, Division of Operating Reactors. Responsible for review of structural aspects of analysis methods and codes applicable to operating reactors (including loads on fuel assemblies). Responsible for development of structural acceptance criteria (with Mechanical Engineering Branch, DSS).

Manpower Estimates: 1.0 man-months in FY 1979

F. Engineering Branch, Division of Operating Reactors. Responsible for the coordination and management of this Technical Activity.

Manpower Estimates: 1.5 man-months in FY 1979

5. TECHNICAL ASSISTANCE

A. Managed by DOR (Engineering Branch):

Contractor: EG&G Idaho, Inc.

Funds Available: $70K in FY 1979.

This is an NRC program to independently model representative Westinghouse 4-loop (Indian Point 3), B&W (Arkansas Nuclear One Unit 1), and CE (St. Lucie 1) plants for the purpose of assessing the loads on all major structures and components resulting from asymmetric LOCA loads. The purpose of this program is to develop an independent NRC capability for performing inelastic dynamic analyses. Sensitivity studies will be performed to evaluate the effects of various break opening times, effects of component stiffness, and three-dimensional coupling effects.

B. Managed by DSS (Mechanical Engineering Branch):

Contractor: EG&G Idaho, Inc.

Funds Available: 80K in FY 1979

This is an NRC/DSS program to provide the staff with the analytical tools necessary to independently verify the selection of design basis pipe rupture locations; and to verify that the criteria for assurance of integrity under LOCA & SSE loads for reactor coolant piping, the reactor vessel, steam generators, main coolant pumps and the supports for these components have been implemented correctly. Verification analyses for a B&W plant (Erie), and a BWR plant (Zimmer 1) will be run to verify results reported by the applicants. Support models will be designed to be revised as necessary to represent various support configurations utilized by Architect/ Engineers of the plants under CP/OL review.

C. Managed by DSS (Core Performance Branch)

Contractor: EG&G, Inc.

Funds Available: 60K in FY 1979

This is an NRC/DSS program to perform independent calculations of fuel assembly response from LOCA as well as seismic loads. Audit calculations are performed to assist NRC in reviewing vendor analytical methods. Evaluation on Westinghouse and General Electric Co. analytical methods for fuel assembly response have been completed.

In addition, the contractor will assist DSS/CPB in developing acceptance criteria for the fuel assembly evaluation.

6. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

A. W Owners Group of licensees

The W owners group of licensees is an ad hoc organization of most (but not all) owners of operating W plants, formed for the purpose of sponsoring and proposing the augmented inservice inspection program (WCAP-8802) in lieu of furnishing the detailed analysis requested by NRC. This group of licensees has engaged Westinghouse Electric Corporation as its principal consultant.

With the advent of the NRC decision to request all licensees for a detailed analysis and to set aside - at least for the present--the ISI proposal, the continued role of this licensee group is undetermined.

B. CE Owners Group of Licensees

The CE owners group of licensees is also an ad hoc organization of most owners of operating CE plants. This group sponsored the probability study prepared by Science Applications, Inc., which concluded that the probability of severe pipe breaks that could trigger the loads under consideration is below the threshold of concern. The future role of this licensee group is also undetermined.

C. B&W Owners Group of Licensees

This group is composed of owners of B&W plants having nuclear steam supply systems of the same design (177 fuel assemblies, skirt supported vessels.) This group has engaged SAI and B&W as its consultant for the preparation of a probability study similar to the one done by SAI for the CE owners group. This report has not yet been submitted.

D. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

7. ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

None.

8. POTENTIAL PROBLEMS

A. Three owners groups representing most operating PWRs have been formed and either will propose or have proposed solutions different from the requested analysis (augmented ISI, probability studies). Therefore, strong industry resistance to our request for some form of analysis is possible.

B. Rigorous application of the generic structural acceptance criteria may require modifications that are judged to be impossible for some older plants. For these cases, alternative solutions may be required.

APPENDIX B

STATUS OF OPERATING PLANT EVALUATION
November 30, 1980

Plant	Licensee Evaluation Submitted	Projected Safety Evaluation Schedule
Arkansas 1	Yes	5/81
Arkansas 2	Yes	10/81
Beaver Valley 1	Yes	5/81
Calvert Cliffs 1	Yes	7/81
Calvert Cliffs 2	Yes	7/81
Cook 1	Yes	3/81
Cook 2	Yes	3/81
Crystal River 3	Yes	5/81
Davis-Besse 1	Yes	5/81
Farley 1	Yes	9/81
Fort Calhoun 1	Yes	7/81
Ginna	Yes	3/81
Haddam Neck	Yes	3/81
Indian Point 1*	No	--
Indian Point 2	Yes	12/80
Indian Point 3	Yes	12/80
Kewaunee**	Yes	5/81
Maine Yankee**	Yes	10/81
Millstone 2	Yes	7/81
North Anna	Yes	5/81
Oconee 1	Yes	5/81
Oconee 2	Yes	5/81
Oconee 3	Yes	5/81
Palisades	Yes	7/81
Point Beach 1	Yes	3/81
Point Beach 2	Yes	3/81
Prairie Island 1	Yes	5/81
Prairie Island 2	Yes	5/81
Rancho Seco	Yes	5/81
Robinson 2	Yes	3/81
Salem 1	Yes	9/81
Salem 2	Yes	9/81
San Onofre 1	Yes	3/81
St. Lucie 1	Yes	5/81
Surry 1	Yes	3/81
Surry 2	Yes	3/81
TMI-1	Yes	5/81
TMI-2	Yes	5/81
Trojan	Yes	9/81
Turkey Point 3	Yes	3/81
Turkey Point 4	Yes	3/81
Yankee Rowe 1	Yes	3/81
Zion 1	Yes	3/81
Zion 2	Yes	3/81

* Shut down.
** Modification complete.

APPENDIX C

CHARACTERISTICS OF PIPE BREAKS

The staff performed a study at EG&G to investigate pipe-break area and time characteristics. Pipe-break flow area and the time associated with development of flow area are major considerations in the development of asymmetric LOCA loads. Historically, the staff has required conservative assumptions in these areas to be used in the development of loading transients. A task in the generic action plan A-2 required the staff to investigate realistic characteristics of pipe breaks.

The study performed by EG&G included a review of pipe-break models submitted by each PWR NSSS vendor and the development of pipe-break characteristics at four specific plants. A summary of the EG&G report* and its conclusions documenting that study are presented below.

This report documents an analysis performed by EG&G Idaho on break-opening times and associated flow areas due to pipe separation during a postulated loss-of-coolant accident (LOCA). This effort supports the resolution of the U.S. Nuclear Regulatory Commission (NRC) Category A Task Action Plan A-2 on asymmetric LOCA loads and provides NRC Division of Operating Reactors (DOR) the capability of independently verifying vendor analyses.

- A postulated LOCA for a nuclear-power plant is initiated by a severance, or break, in the piping of the primary coolant system, inducing severe transient loads on piping, major components, and supports in the system.

- The severity of the depressurization transient is directly dependent upon the combination of break size and length of time required for pipe separation. In general, LOCA thermal hydraulic analyses have taken the conservative approach by assuming break openings to occur instantaneously (that is, 10 msec or less). If it can be shown that this approach is unrealistic, (that is, break openings are not instantaneous), the loads associated with the LOCA transients may be greatly reduced.

- In this report, break-opening times and areas are examined on a plant-specific basis. The analysis covers four nuclear-power plants designed by three different vendors: San Onofre Unit 2 and St. Lucie Unit 1 designed by Combustion Engineering (CE), Arkansas Nuclear One Unit 1 designed by Babcock and Wilcox (B&W), and Indian Point Unit 3 designed by Westinghouse. These four plants were chosen for two reasons: (1) mechanical dynamic response analyses of all four plants have previously been performed for postulated LOCA applied loads by EG&G Idaho in support of NRC licensing audits, and (2) the four plants represent a good cross-section of nuclear-plant design. Information from the previous analyses was utilized in

*R. F. Lippert, EG&G Idaho, Inc., "Study of Break Opening Area/Time Determination for Postulated LOCA Guillotine Pipe Ruptures of Nuclear Primary Coolant Systems," RE-A-79-093, July 1979.

expediting development of models of the necessary segments of the primary coolant loops and in providing appropriate hydraulic transients. The structural models include material and geometric nonlinearities when applicable.

Subsequent sections of the report include: (1) an investigation of various techniques for geometrically calculating break-opening areas, (2) a description of the several analytical structural models employed in the analysis, (3) a description of the applied LOCA loadings associated with the various postulated pipe breaks analyzed, and (4) a presentation of the resulting pipe-break areas as a function of time and associated observations. In addition, the appendices include vendor break-area procedures, plant geometries, and plots of break-area time-history results.

The EG&G analysis of four nuclear plants to study break opening areas and times for postulated LOCA pipe ruptures presented in this report provides the following observations. Times for break-opening areas to develop are significantly greater than the 1- and 10-msec times presently assumed, and, in the cases of limited breaks, the break areas developed were significantly less than those to which the plants were designed. Both results indicate significant conservatism in the design and analysis approaches generally employed. For the results of this study to be applicable to nuclear-power plants other than those analyzed herein, careful consideration must be given to the particular plant's size, power, dimensions, and geometry as well as to the type of structural analysis and LOCA load development.

APPENDIX D

LETTERS TO APPLICANTS/LICENSEES

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

All PWR Licensees

Gentlemen:

In October of 1975, the NRC staff notified each licensee of an operating PWR facility of a potential safety problem concerning the design of the reactor pressure vessel support system. Those letters requested each licensee to review the design basis for the reactor vessel support system for each of its PWR facilities to determine whether certain transient loads, which were described in the enclosure to the letter, had been appropriately taken into account in the design. Furthermore, these letters indicated that, on the basis of the results of licensees' reviews, a reassessment of the reactor vessel support design for each operating PWR facility may be required.

Licensee responses to that request indicated that these postulated asymmetric loads have not been considered in the design basis for the reactor vessel support system, reactor internals including the fuel, steam generator supports, pump supports, emergency core cooling system (ECCS) lines, reactor coolant system piping, or control rod drives.

Subsequently in June 1976, the NRC staff informed each PWR licensee that a reassessment of the reactor vessel support system design for each of its facilities was required. While the emphasis of these letters was primarily focused on the need to reassess 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, we indicated that our generic review may extend to other areas in the nuclear steam supply system (NSSS) and that further evaluation may be required.

For your information, Enclosure 1 is a summary of the background and current status of our review efforts related to this generic concern.

We have now determined that an assessment of the potential for damage to other NSSS component supports (e.g., steam generator and pump supports), the fuel assemblies, control rod drives, and ECCS piping attached to the reactor coolant system due to loadings associated with postulated coolant system piping breaks is required. Our request for additional information transmitted to you in June 1976 has been revised both to clarify our original request and to identify the extension of our concerns to other areas in the NSSS, as identified above. A copy of this revised request for additional information is provided as Enclosure 2.

The revised request for additional information identifies a requirement that your assessment of potential damage to the reactor vessel and other NSSS component supports, reactor vessel, fuel and internals, attached ECCS lines and the control rod drives should include consideration of breaks both inside and outside of the reactor pressure vessel cavity. This assessment should be made for postulated breaks in the reactor coolant piping system, (secondary systems are not to be included), including the following locations:

a) Reactor vessel hot and cold leg nozzle safe ends
b) Pump discharge nozzle
c) Crossover leg
d) Hot leg at the steam generator (B&W and CE plants only)

A number of licensees, have presented to the NRC staff alternate proposals, other than to conduct a detailed analyses, to resolve this concern. Based upon our review of these proposals, we have concluded that these alternative proposals do not establish an acceptable basis for long term operation without a detailed assessment of the risk resulting from these postulated transient loading conditions. We have, however, concluded that the low probability for occurrence of an event which could result in these loads establishes an adequate basis to justify continued operation for a short term period.

The NRC staff will consider an analysis that is applicable to more than one specific plant if it can be adequately demonstrated that such an analysis is either representative or bounding for each plant concerned.

Additional guidance regarding loading combinations (safe shutdown earthquake loads, loss of coolant accident loads), will be provided by about March 1, 1978, following the conclusion of staff investigations in this area.

Please respond within 30 days of receipt of this letter, indicating your intent to proceed with an evaluation of the overall asymmetric loss of coolant accident (LOCA) loads as described herein. In addition, please submit to us, within 90 days, your detailed schedule for providing the required evaluation. Your schedule should be consistent with our desire to resolve this problem within two years and should clearly state your intent to demonstrate the safety of long term continued operation.

We are transmitting information copies of this letter to the Westinghouse, Combustion Engineering and Babcock & Wilcox Companies. If you have any questions or want any clarification on this matter, please call your NRC Project Manager.

Sincerely,

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
1. Background and Current Status
2. Revised Request for Additional Information

cc w/enclosure:
See attached listing

ENCLOSURE 1

BACKGROUND AND CURRENT STATUS OF THE NRC STAFF REVIEW OF ASYMMETRIC LOCA LOADS FOR PWR FACILITIES

On May 7, 1975, the NRC was informed by Virginia Electric & Power Company that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by Westinghouse or Stone & Webster in the original design of the reactor vessel support system for North Anna, Units 1 and 2. It had been identified that in the event of a postulated instantaneous, double-ended offset LOCA at the vessel nozzle, asymmetric loading could result from forces induced on the reactor internals by transient differential pressure across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more detailed analytical models, it became apparent that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports and on other components, thereby possibly affecting their integrity. Although this potential safety concern was first identified during the review of the North Anna facilities, it has generic implications for all PWRs.

Upon closer examination of this situation, it was determined that postulated breaks in a reactor coolant pipe at vessel nozzles were not the only area of concern but rather that other pipe breaks in the reactor coolant system could cause internal and external transient loads to act upon the reactor vessel and other components. For the postulated pipe break in the cold leg, asymmetric pressure changes could take place in the annulus between the core barrel and the vessel. Decompression could occur on the side of the vessel annulus nearest the pipe break before the pressure on the opposite side of the vessel changes. This momentary differential pressure across the core barrel could induce lateral loads both on the core barrel and on the reactor vessel. Vertical loads could also be applied to the core internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, for vessel nozzle breaks, the annulus between the reactor and biological shield wall could become asymmetrically pressurized resulting in a differential pressure across the vessel causing additional horizontal and vertical external loads on the vessel. In addition, the vessel could be loaded by the effects of initial tension release and blowdown thrust at the pipe break. These loads could occur simultaneously. For a reactor vessel outlet break, the same type of loadings could occur, but the internal loads would be predominantly vertical due to more rapid decompression of the upper plenum.

Although the NRC staff's original emphasis and concern were focused primarily on the integrity of the reactor vessel support system with respect to postulated breaks inside the reactor cavity (i.e., at a nozzle), it has since become apparent that significant asymmetric forces can also be generated by postulated pipe breaks outside the cavity and that the scope of the problem is not limited to the vessel support system itself. For such outside-cavity postulated breaks, the aforementioned concerns, such as the integrity of fuel assemblies and other structures, need to be examined.

In June 1976, the NRC requested all operating PWR licensees to evaluate the adequacy of the reactor system components and their supports at their facilities with respect to these newly-identified loads.

In response to our request, most licensees with Westinghouse plants proposed an augmented inservice inspection program (ISI) of the reactor vessel safe-end-to-end pipe welds in lieu of providing an evaluation of postulated piping failures. Licensees with Combustion Engineering plants submitted a probability study (prepared by Science Applications, Inc.) in support of their conclusion that a break at a particular location (vessel nozzle) has such a low probability of occurrence that no further analysis is necessary. A similar study has been recently submitted by Science Applications, Inc. (SAI) for B&W plants.

When the Westinghouse and CE owners group reports were received in September 1976, the NRC formed a special review task group to evaluate these alternative proposals. In addition, EG&G Idaho, Inc., was contracted to perform an independent review of the SAI probability study submitted for the CE owners group.

This review effort resulted in a substantial number of questions which previously have been provided to representatives of each group. Based on the nature of these questions and other factors to be discussed later in this report, we cannot accept these reports in their present form as a resolution for the asymmetric LOCA load generic issue. Based on our review, we have concluded that a sufficient data base does not presently exist within the nuclear industry to provide satisfactory answers to these information needs. Several long-term experimental programs would be required to provide much of this information. Although the probability study recently submitted by SAI for certain B&W owners does respond to some of the informal questions raised during our review of the SAI report prepared by CE plants, the more fundamental questions remain. Therefore, this conclusion also applies to the SAI topical report for B&W plants (SAI-050-77-PA).

A second - and equally important - reason for not accepting probability/ISI approaches as a solution at this point concerns our need and industry's need to gain a better understanding of the problem. We consider it essential that an understanding of the important breaks and associated consequences be known before applying any remedy - be it pipe restraints, probability, ISI, or some combination of these measures. Only in this way will we have a basis on which to judge the importance of the remedy with respect to what it is designed to prevent.

Although we have many questions on each of these topical reports, this does not mean that we view the probabilistic/ISI approach as completely without merit. In fact, the results of a probabilistic evaluation serves as the basis for continued operation and licensing of nuclear plants during this interim period while additional evaluations can be performed by vendors and licensees.

We believe that the justification for continued plant operation has as its basic foundation the fact that the event in question, i.e., a hypothetical double-ended instantaneous rupture of the main coolant pipe at a particular location, has a very low probability of occurrence.

The disruptive failure probability of a reactor vessel itself has been estimated to lie between 10^{-6} and 10^{-7} per reactor year - so low that it is not considered as a design basis event. The rupture probability of pipes is estimated to be higher. WASH-1400 used a median value of 10^{-4} for LOCA initiating ruptures per plant-year for all pipes sizes 6" and greater (with a lower and upper bound of 10^{-5} and 10^{-3}, respectively). We believe that considering the large size of the pipes in question (up to 50" O.D. and 4-1/8" thick), the lower bound is more appropriate since these pipes are more like vessels in size. In addition, the quality control of this piping is the best available and somewhat better than that of the piping used in the WASH-1400 study.

These factors, coupled with the facts that (1) the break of primary concern must be very large, (2) it must occur at a specific location, (3) the break must occur essentially instantaneously, and (4) these welds are currently subject to inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI, lead us to conclude that the probability of a pipe break resulting in substantial transient loads on the vessel support system or other structures is acceptably small such that continued reactor operation and continued licensing of facilities for operation can continue while this matter is being resolved.

In support of the above, the staff has developed a short-term interim criterion to determine if an acceptable level of safety exists for operating PWRs under conditions of a postulated pipe break. This interim criterion is based on a simplified probabilistic model that incorporates elastic fracture mechanics techniques to estimate the probability of a pipe break. Critical flaw size and subcritical flaw growth rates were determined assuming the presence of a surface flaw located in a circumferential weld of a thick-walled pipe, Determination of the critical flaw size was based on an estimated fracture toughness value of K_{IC} at a minimum temperature of 200 F and a uniform tensile stress equal to the consideration of various operating conditions producing elastically calculated stresses ranging in value from 1 to 3 times the material minimum yield strength.

Then, using the calculated critical flaw size, the subcritical growth rate, and an estimated probability distribution of an undetected flaw in thick-walled pipe welds, the upper bound probability of pipe break was estimated to be 10^{-4}. This value is also supported by a recent publication by Dr. S. H. Bush* which states that actual failure statistics confirm rates of 10^{-4} to 10^{-6} per reactor-year in large pipes, with higher rates as the pipe size decreases. Considering these analyses, we conclude that our conservative estimate on a pipe break in the primary coolant system is in the range of 10^{-4} to 10^{-6}. This estimated pipe break probability is considered acceptably low to justify short-term operation of nuclear power plants.

In view of all previous discussions concerning this issue, the NRC staff has concluded that an evaluation must be undertaken to assess the design adequacy of the reactor vessel supports and other affected structures and systems to withstand asymmetric LOCA loads, including an assessment of the effects of asymmetric loads produced by various pipe breaks both inside and outside the reactor cavity. On performing these evaluations the staff will permit the grouping of plants, where adequate justification for such grouping exists, in order to limit the number of plants to be analyzed. Alternatively, the staff will permit the analyzing of a "prototypical" plant, which is sufficiently representative of a group of plants, to provide the necessary information. Both of these concepts have been discussed with the Westinghouse and CE Owners Groups, and we believe that such approaches could save a significant amount of time and effort in obtaining results on which to base any needed corrective measures. The NRC staff is prepared to meet with PWR licensees to discuss such approaches, and has already done so. For example, we met with the Westinghouse owners group on October 19, 1977 for the purpose of discussing a generic solution for breaks outside the reactor cavity. It is expected that a similar meeting will be held in the near

*"Critical Factors in Blowdown Loads in the PWR Guillotine Nozzle Break (Volume 2 - the Asymmetric Load Problem)", dated June 6, 1977

future to address breaks located inside the cavity. This "phased" approach is acceptable to us, provided that it sheds light on and serves to expedite consideration of the more limiting inside-cavity breaks.

For your information, the NRC has a technical assistance contract with EG&G Idaho, Inc., to independently model representative Westinghouse, B&W, and CE plants for the purpose of assessing the loads on all major structures and components resulting from asymmetric LOCA loads. We believe that the results of this program which will include sensitivity studies, will provide significant confirmatory information related to this generic safety concern.

Although, as stated earlier, we believe that continued operation and licensing of facilities for the short term is justified, we also believe that efforts to resolve this issue should proceed without delay, with the objective of both completing the necessary assessments and installing any necessary plant modifications within two years. In making this statement, we wish to make it clear that plant modifications, if indicated by licensee assessments, is the preferred approach. At the same time, we recognize that there may be cases wherein appropriate modifications may be judged to be unwarranted based on the consideration of overall risk. In such cases, and only in such cases, we will be prepared to give further consideration to alternate approaches, such as probability ISI. We feel, however, that ISI techniques as they exist today could be considerably improved, and, to the extent that such improvements could have a direct bearing on this problem as well as an impact of nuclear safety in general, we would welcome their development.

ENCLOSURE 2

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 Compartment Wall

The following information should be included in your reassessment of 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 postulated breaks at the reactor vessel hot and cold leg nozzle safe ends, pump discharge nozzle and crossover leg that result in the most severe loading conditions for the above-mentioned

systems.* Provide an assessment of the effects of asymmetric pressure differentials on these systems/components in combination with all external loadings including asymmetric cavity pressurization for both the reactor vessel and steam generator which might result from the required postulate. This assessment should consider:

a. limited displacement break areas where applicable
b. consideration of fluid-structure interaction
c. use of actual time-dependent forcing function
d. reactor support stiffness
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 exceeded.

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:

*B&W and CE plant licensees should also consider breaks in the hot leg at the steam generator inlet.

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 of 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 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 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. Specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design, or both.

(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.

Describe the 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 portions 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.

(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.

APPENDIX E

FUEL-SYSTEM DESIGN

(Appendix A to SRP 4.2)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Appendix E

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
PROPOSED ADDITION OF
APPENDIX A
EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE
TO EXTERNALLY APPLIED FORCES
TO
STANDARD REVIEW PLAN PSRP-4.2, REVISION 2, DRAFT 1

A. BACKGROUND

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This Appendix describes the review that should performed of the fuel assembly structural response to seismic and LOCA loads. Background material for this Appendix is given in Refs. 1-3.

This proposed revision of the Standard Review Plan and the support value/impact statement have not received a complete staff review and approval and do not represent an official NRC staff position. Public comments are being solicited on both the revision and the value/impact statement (including any implementation schedules) prior to a review by the Regulatory Requirements Review Committee and their recommendation as to whether this revision should be approved. Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. All comments received by _____ will be considered by the Regulatory Requirements Review Committee. A summary of the meeting of the Committee at which this revision is considered, the Committee recommendations and all of the associated documents and comments considered by the Committee will be made publicly available prior to a decision by the Director, Office of Nuclear Reactor Regulation, on whether to implement this revision.

B. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from results of the primary coolant system and reactor internals structural analysis, which is reviewed by the Mechanical Engineering Branch.

Input for the fuel assembly response to a LOCA should include (a) motions of the core plate, core shroud, fuel alignment plate, or other relevant structures; these motions should correspond to the break that produced the peak fuel assembly loadings in the primary coolant system and reactor internals analysis, and (b) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a non-linear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the SSE; if a linear response is produced, a spectral analysis may be used (see Regulatory Guide 1.60).

2. Methods

Analytical methods used in performing structural response analyses must be reviewed. Justification should be supplied to show that the numerical solution techniques are appropriate.

Linear and non-linear structural representations (i.e., the modeling) must also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when pratical.

A sample problem of a simplified nature must be worked by the applicant and compared by the reviewer with either hand calculations or results generated by the reviewer with an independent code (2). Although the sample problem should use a structural representation that is as close as possible to the design in question (and, therefore, would vary from one vendor to another), simplifying assumptions may be made (e.g., one might use a 3-assembly core region with continuous sinusoidal input).

The sample problem should be designed to exercise various features of the code and reveal their behavior. The sample problem comparison is not, however, designed to show that one code is more conservative than another, but rather to alert the reviewer to major discrepancies so that an explanation can be sought.

3. Uncertainty Allowances

The fuel assembly structural models and analytical methods are probably conservative and input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any non-conservatisms, two precautions should be taken: (a) If it is not explicitly evaluated, impact loads from the PWR

LOCA analysis should be increased (by about 30%) to account for a pressure pulse, which is associated with steam flashing that affects only the PWR fuel assembly analysis. (b) Conservative margin should be added if any part of the analysis (PWR or BWR) exhibits pronounced sensitivity to input variations.

Variations in resultant loads should be determined for ±10% variations in input amplitude and frequency; variations in amplitude and frequency should be made separately, not simultaneously. A factor should be developed for resultant load magnitude variations of more than 15%. For example, if ±10% variations in input magnitude or frequency produce a maximum resultant increase of 35%, the sensitivity factor would be 1.2. Since resonances and pronounced sensitivities may be plant-dependent, the sensitivity analysis should be performed on a plant-by-plant basis until the reviewer is confident that further sensitivity analyses are unnecessary or it is otherwise demonstrated that the analyses performed are bounding.

4. Audit

Independent audit calculations for a typical full-sized core must be performed by the reviewer to verify that the overall structural representation is adequate. An independent audit code (2) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

General Design Criterion 2 requires an appropriate combination of loads from natural phenomena and accident conditions. Loads on fuel assembly components should be calculated for each input (i.e., seismic and LOCA) as described above in Paragraph 1, and the resulting loads should be added by the square-root-of-sum-of-squares (SRSS) method. These combined loads should be compared with the component strengths described in Section C according to the acceptance criteria in Section D.

C. DETERMINATION OF STRENGTH

1. Grids

All modes of loading (e.g., in-grid and through-grid loadings) should be considered, and the most damaging mode should be represented in the vendor's laboratory grid strength tests. Test procedures and results should be reviewed to assure that the appropriate failure mode is being predicted. The review should also confirm that (a) the testing impact velocities correspond to expected fuel assembly velocities, and (b) the crushing load P_{crit} has been suitably selected from the load-vs- deflection curves. Because of the potential for different test rigs to introduce measurement variations, an evaluation of the grid strength test equipment will be included as part of the review of the test procedure.

The consequences of grid deformation are small. Gross deformation of grids in many PWR assemblies would be needed to interfere with control rod insertion during an SSE (i.e., buckling of a few isolated grids could not displace guide tubes significantly from their proper location), and grid deformation (without channel deflection) would not affect control blade insertion in a BWR. In a LOCA, gross deformation of the hot channel in either a PWR or a BWR would result in only small increases in peak cladding temperature. Therefore, average values are appropriate, and the allowable crushing load P_{crit} should be the 95% confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. While P_{crit} will increase with irradiation, ductility will be reduced. The extra margin in P_{crit} for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond Pcrit.

2. Components Other than Grids

Strengths of fuel assembly components other than spacer grids may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components (e.g., fracturing of guide tubes or fragmentation of fuel rods) could be more serious than grid deformation, allowable values should bound a large percentage (about 95%) of the distribution of component strengths. Therefore, ASME Boiler and Pressure Vessel Code values and procedures may be used where appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should include consideration of buckling and fatigue effects.

D. ACCEPTANCE CRITERIA

1. Loss-of-Coolant Accident

Two principal criteria apply for the LOCA: (a) fuel rod fragmentation must not occur as a direct result of the blowdown loads, and (b) the 10 CFR 50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisfied if the combined loads on the fuel rods and components other than grids remain below the allowable values defined above. The second criterion is satisfied by an ECCS analysis. If combined loads on the grids remain below P_{crit}, as defined above, then no significant distortion of the fuel assembly would occur and the usual ECCS analysis is sufficient. If combined grid loads exceed P_{crit}, then grid deformation must be assumed and the ECCS analysis must include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed grids) may be made unless other assumptions are justified.

Control rod insertability is a third criterion that must be satisfied. Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P_{crit} as defined above, then significant deformation of the fuel assembly would not occur and control rod insertion would not be interfered with by lateral displacement of the guide tubes. If combined loads on the grids exceed P_{crit}, then additional analysis is needed to show that deformation is not severe enough to prevent control rod insertion.

For a BWR, several conditions must be met to demonstrate control blade insertability: (a) combined loads on the channel box must remain below the allowable value defined above for components other than grids; otherwise, additional analysis is needed to show that deformation is not severe enough to prevent control blade insertion, and (b) vertical liftoff forces must not unseat the lower tieplate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

2. Safe Shutdown Earthquake

Two criteria apply for the SSE: (a) fuel rod fragmentation must not occur as a result of the seismic loads, and (b) control rod insertability must be assured. The first criterion is satisfied by the criteria in Paragraph 1. The second criterion must be satisfied for SSE loads alone if no analysis for combined loads is required by Paragraph 1.

E. REFERENCES

1. R. L. Grubb, "Review of LWR Fuel System Mechanical Response with Recommendations for Component Acceptance Criteria," Idaho National Engineering Laboratory, NUREG/CR-1018, September 1979.

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3. R. L. Grubb, "Technical Evaluation of PWR Fuel Spacer Grid Response Load Sensitivity Studies," Idaho National Engineering Laboratory, NUREG/CR-1020, September 1979.

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16. ABSTRACT (200 words or less)

NRC staff, after being informed of newly identified asymmetric loadings resulting from postulated ruptures of primary piping, initiated a generic investigation, Task Action Plan A-2, limited to pressurized-water-reactor (PWR) plants because of their higher primary system pressures. The intent of the investigation was to develop acceptable criteria and guidelines for evaluating plant analyses.

The staff concludes that an acceptable basis is provided in this report for performing and reviewing plant analyses. Criteria were developed for evaluating loading transients, structural components, and the fuel assembly.

The staff approved computer programs and modeling techniques submitted by each PWR vendor for development of the subcooled blowdown and cavity-pressure loading transients Audit models were developed to evaluate the structural computer programs and modeling techniques. Methods have been approved for the structural-analysis method submitted by Westinghouse for the Indian Point Unit 3 plant. Criteria and guidelines are provided to perform a detailed evaluation of the fuel assembly. Acceptance criteria are also provided so deformed fuel-assembly spacer grids may be evaluated.

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