MASTER

CONF-790827--1

NONLINEAR DYNAMIC ANALYSIS OF NUCLEAR REACTOR PRIMARY COOLANT SYSTEMS

By

B. F. Saffell, Jr., R. W. Macek, T. R. Thompson, and R. F. Lippert

EG&G Idaho, Inc.

Box 1625

Idaho Falls, Idaho 83401

ABSTRACT

The ADINA computer code is utilized to perform mechanical response analysis of pressurized reactor primary coolant systems subjected to postulated loss-of-coolant accident (LOCA) loadings. Specifically, three plant analyses are performed utilizing the geometric and material nonlinear analysis capabilities of ADINA. Each reactor system finite element model represents the reactor vessel and internals, piping, major components, and component supports in a single coupled model. Material and geometric nonlinear capabilities of the beam and truss elements are employed in the formulation of each finite element model. Loadings applied to each plant for LOCA dynamic analsis include steady-state pressure, dead weight, strain energy release, transient piping hydraulic forces, and reactor vessel cavity pressurization. Representative results are presented with some suggestions for consideration in future ADINA code development.

INTRODUCTION

Nonlinear dynamic analyses of pressurized water reactor primary coolant systems are being performed for the case of a postulated loss-of-coolant accident (LOCA) at the reactor pressure vessel cold or hot leg nozzle using the ADINA computer code. This postulated event, whose likelihood is very small, is considered because it imposes very severe loads on the reactor pressure vessel (RPV) supports, fuel, and other reactor internal components. Assessment of reactor system structural integrity to this type of event can be accomplished by performing a time history mechanical response analysis. Results of the analyses described herein serve to support Nuclear Regulatory Commission (NRC) licensing evaluation, generic reviews, and operating reactor regulation.

This paper describes the structural analysis of reactor primary coolant systems subjected to a postulated LOCA including the effects of asymmetric loads. A description of a pressurized water reactor (PWR) primary coolant system is presented and followed by an overview of the complete analysis procedure. Subsequently, the finite element representation of a specific reactor primary coolant system is addressed in detail. Discussion of this topic will identify those areas which utilized the nonlinear capabilities of the ADINA computer code. Finally, representative results are presented as are recommendations for consideration in future ADINA code development.

SYSTEM DESCRIPTION

The general layout of typical pressurized water reactor primary coolant loops are shown in Figures 1-3. Light water, pressurized to 15.5 MPa, enters the reactor pressure vessel (RPV) through the cold leg nozzle and flows down the region called the downcomer. This is the area between the RPV and the core barrel, and includes the thermal shield surrounding the nuclear fuel area. The water turns in the lower plenum and flows up through the lower core plate and past the fuel, where it is heated from 290 to 320 deg C and through the upper core plate. From the upper chamber, which contains the core support structure and control rod shrouds, the water flows through the outlet nozzles and along the hot leg to the steam generator, where its heat is transferred through thousands of tubes to boil water and create high pressure steam in the secondary system. The primary coolant water then flows through the crossover leg to the pump, and through the cold leg back to the inlet nozzle of the RPV.

Three plant primary coolant systems have been analyzed using the ADINA computer code -- Indian Point Unit III, San Onofre Unit 2, and Arkansas Nuclear One, Unit 1. Although these plant configurations, Figures 1-3, differ, all have the same essential components. The main structural elements of a reactor primary coolant system are the reactor pressure vessel and internal components, steam generator, pumps, and supports for these components. The RPV internals consist primarily of the reactor core and its support structures. Specifically, the main structural components within the RPV, Figure 4, are the upper support assembly, the core barrel, and the upper and lower core plates. The fuel, although not a structural member, is important in the analysis because of its large mass and the consequences of damage to it.

ANALYSIS OVERVIEW

Structural analysis of a reactor primary coolant system for a postulated loss-of-coolant accident event requires an understanding of the problem, the hydraulic transient calculation methods, primary system finite element model formulation, structural analysis methods, and interpretation of results. First, as previously stated, a postulated break at the RPV nozzle imposes very severe loads on the RPV supports. These loads are due to the lateral loads on the core barrel and reactor vessel which are associated with this postulated event. Any cold leg LOCA will cause asymmetric decompression of the downcomer annulus, Figure 5, and, therefore, lateral core barrel loading. If the cold leg rupture is postulated to occur at the RPV nozzle, the cavity region between the reactor vessel and biological shield, Figure 5, is asymmetrically pressurized, resulting in a lateral reactor vessel loading.

In order to perform a transient structural analysis, the hydraulic transients must first be calculated to define the forcing functions associated with a LOCA. Forcing functions for the subject event are defined in two stages. First, a one-dimensional network is formulated to represent the flow path through each primary coolant loop. The multidimensional flow regions within the reactor vessel are also represented utilizing one-dimensional pipe elements. These various models are combined into a single system model for use in a computer code, such as WHAM[1], which is designed to define the pressure transients throughout the system for the particular event being considered. The second stage, which is required when a postulated LOCA within the biological shield is being considered, is to formulate a model of the reactor vessel cavity. This input model is required to define the asymmetric loading on the reactor vessel. Completion of both stages results in definition of the forcing functions associated with a postulated LOCA within the biological shield. Typical results from the hydraulic analyses are presented in the form of the subcooled, core barrel lateral load transient, Figure 6, and the cavity pressure force on the reactor vessel, Figure 7.

A finite element representation of the reactor primary coolant system is required to define the system mechanical response to any forcing function or applied load. Finite element model details for the Indian Point Unit 3 plant are presented in the following section.

The finite element model with the hydraulic forcing function serves as input to a structural analysis computer code which is utilized to define the system mechanical response. Either linear-elastic or nonlinear, elastic-plastic time history analysis may be performed using the ADINA[2] computer program. Primary coolant system dynamic analysis yields results in the form of forces, moments, and displacements. This information is then utilized to evaluate the structural integrity of the primary coolant piping and component supports or to serve as input to other more detailed subassembly analyses.

FINITE ELEMENT REPRESENTATION

The Indian Point Unit 3 finite element model[3] is used to illustrate the details of primary coolant system structural representations. The Indian Point Unit 3 primary system is represented by an assemblage of linear elastic, nonlinear elastic, and elastic-plastic truss and beam elements. As shown in Figure 8, the steam generator supports are represented by equivalent elastic beam structures. The pump supports are also 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 9, 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 10 and 11 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 10 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.

MECHANICAL RESPONSE ANALYSIS AND RESULTS

The loading on the Indian Point Unit 3 primary system during a postulated LOCA caused by a primary pipe rupture near an RPV inlet nozzle consists of pressure transients throughout the primary system, asymmetric external pressure on the RPV, strain energy release loads and steady-state gravity loads. The pressure transients are represented by time dependent forces at all locations within the primary system where the fluid flow changes direction. The external asymmetric pressure on the RPV is integrated to obtain concentrated forces and moments acting on the RPV. The strain energy release loads or the loads due to the disturbance of static equilibrium are represented by initial displacements. The gravity loading in the dynamic analysis is represented by initial displacements and a step load.

The response of the system to the LOCA loads was determined for 0.1 second. Shaded areas of the piping, Figure 12, indicate regions where plasticity occurred in the piping during the transient response. Figure 13 indicates that the RPV supports yield early in the transient and undergo significant plastic deformation. The vertical response of the RPV is shown in Figures 14 and 15. As can be seen, the vessel lifts off its supports at approximately 0.02 second into the transient and reimpacts the supports with 11.36 meganewtons of force at 0.06 second. The slight phase difference between the plots of Figures 14 and 15 reflects the rigid body rotation or the "rocking" of the RPV on its supports.

The effect of including gaps in the finite element model was evaluated in the San Onofre Unit 2 analysis and is summarized in Figure 16. All components except for F_b, which is an axial component, exhibit significant differences. For each load component, except for F_b, the linear elastic calculations result in lower valves than those for the nonlinear elastic calculation.

CONCLUSIONS AND RECOMMENDATIONS

The ADINA computer code has been utilized to perform nonlinear (geometric), elastic-plastic, time history dynamic analysis of nuclear reactor primary coolant systems. Based on the experience gained from three plant analyses, the following recommendations are made:

(a) Revise the elastic-plastic beam element to include the pressure hoop stress (for hollow circular beams) in the yield criterion and plastic deformation.

(b) Include a pipe elbow element ultimately with pressure effects and material plasticity.

(c) Modify the elastic-plastic truss element to permit representation of gaps.*

(d) Include a linear arbitrary stiffness and mass matrix element.**

---

* This has been incorporated into ADINA at the Idaho National Engineering Laboratory (INEL) by EG&G Idaho.

** This modification is in the process of being made.

(e) Revise the equilibrium iteration to permit updating of the stiffness matrix during the iteration.

(f) Include multinode constraints (slaved degrees of freedom).

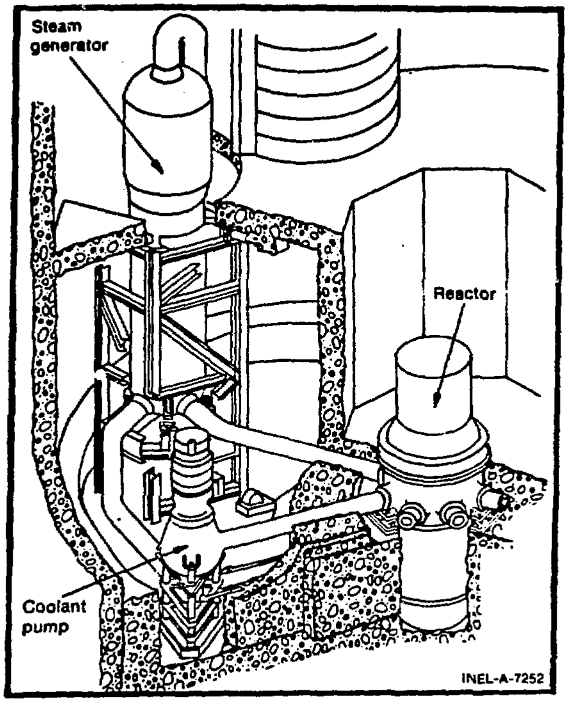
Recommendations (a) and (b) would expand applicability of ADINA further in the piping analysis area. Recommendation (c) could be used to model boundaries where gaps exist and the boundary yields. The arbitary mass and stiffness matrix element could be used for static condensation, substructure representation and fluid structure interaction analysis. Stiffness matrix updating during an iteration may improve the stability of the equilibrium iteration for hardening systems.

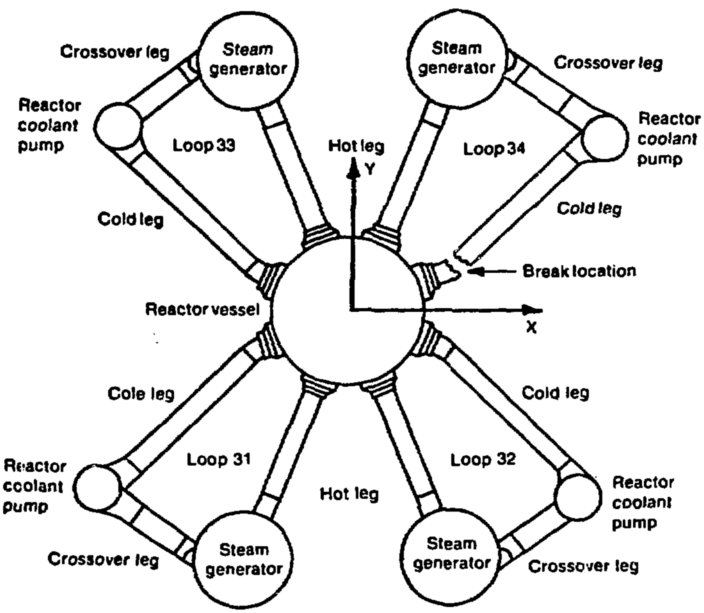
REFERENCES

1. S. Fabic, "Computer Program WHAM for Calculation of Pressure, Velocity, and Force Transients in Liquid Filled Piping Networks", Report No. 67-49-R, Kaiser Engineers, Oakland, California, November 1967.

2. K. J. Bathe, "ADINA - A Finite Element Program for Automatic Dynamic Incremental Nonlinear Analysis", Report 82488-1, Accoustics and Vibration Laboratory, Mechanical Engineering Department, Massachusetts Institute of Technology, Cambridge, Massachusetts, September 1975.

3. R. W. Macek, "Nonlinear, Elastic-Plastic, Dynamic Analysis of A Four Loop Pressurized Water Reactor Primary Coolant System Subjected to Postulated Loss-Of-Coolant Accident Transient Loads", ENS/ANS International Topical Meeting On Nuclear Power Reactor Safety, October 1978, Brussels, Belgium.





Note: RPV supports are beneath cold leg nozzles of loops 31 and 34 and beneath hot leg nozzles of loop 32 and 33.

Fig. 1 Indian Point Unit 3 primary coolant system

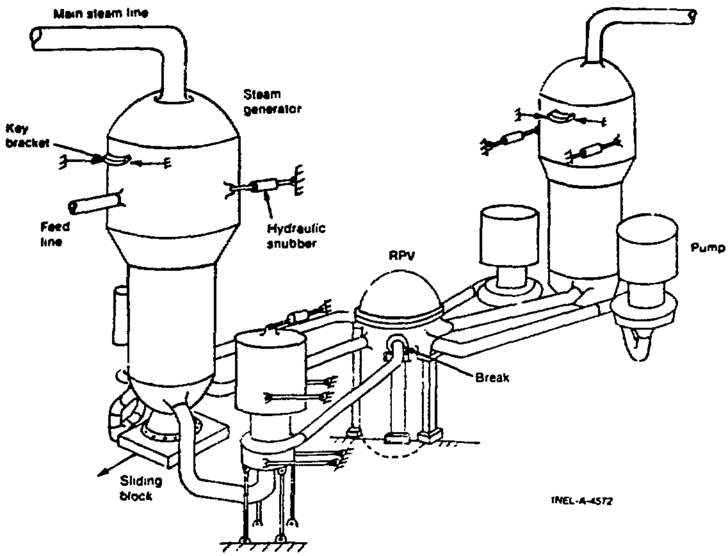


Fig. 2 San Onofre Unit 2 primary coolant system

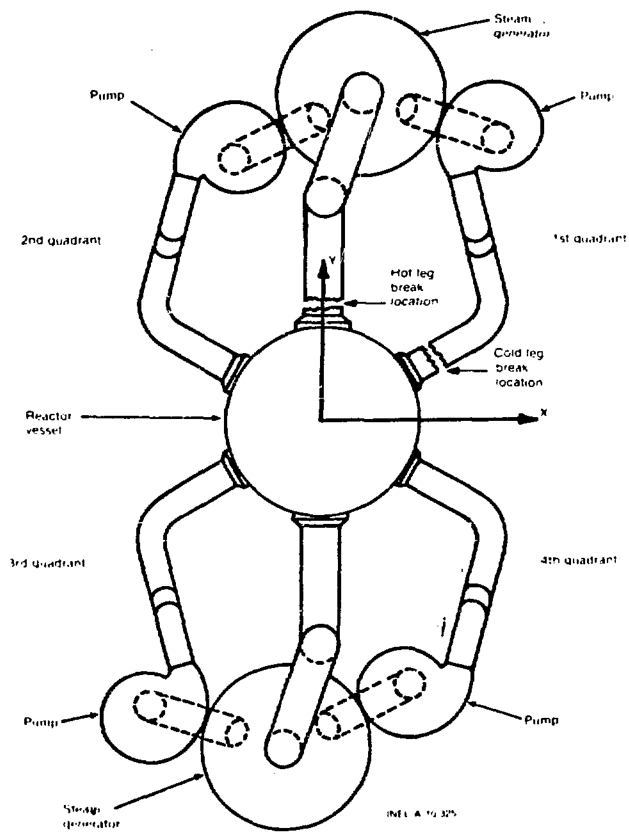


Fig. 3 Arkansas Nuclear One, Unit 1 primary coolant system

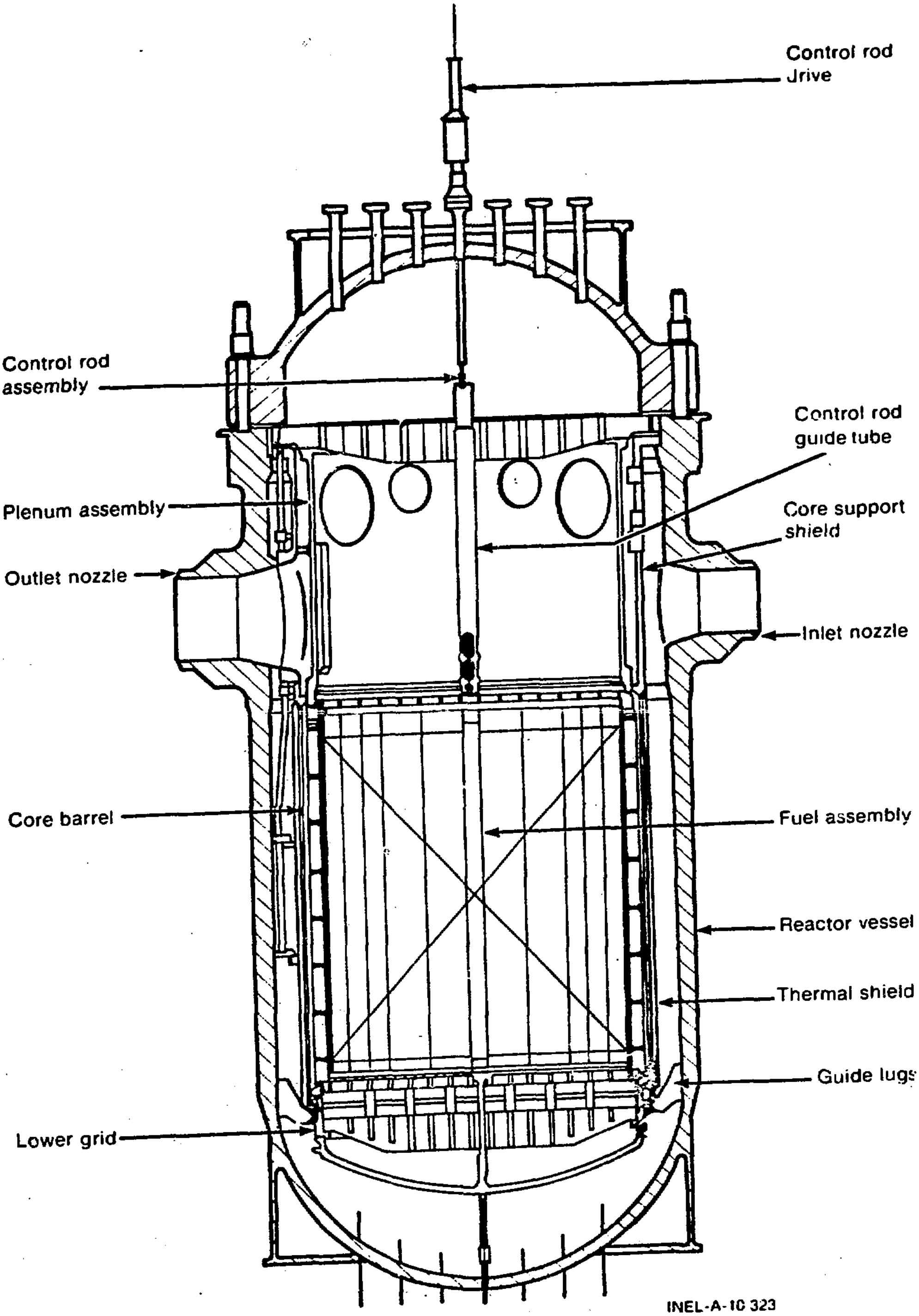
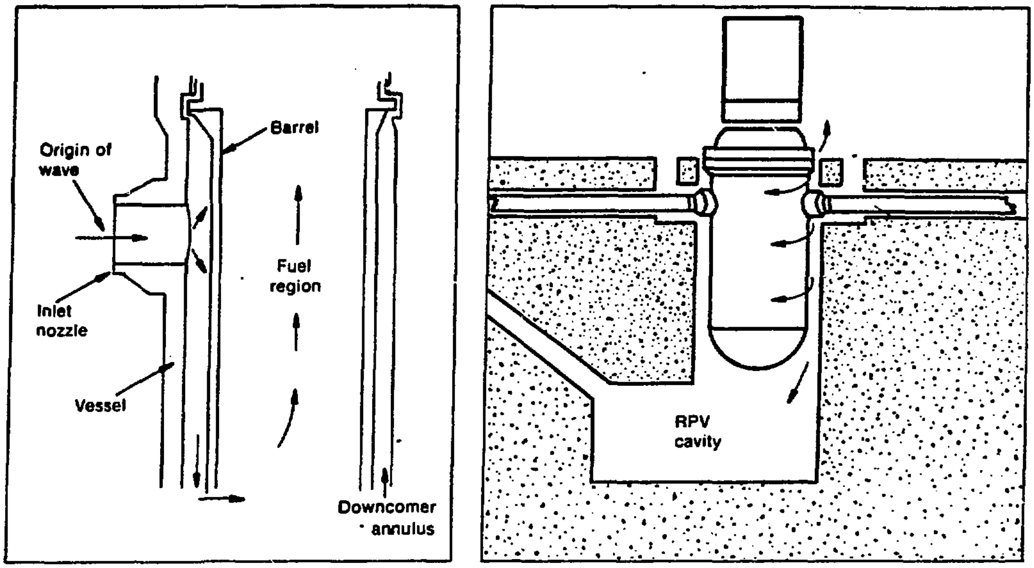


Fig. 4 Reactor vessel and internals



INEL-A-7249 INEL-A-7250

Fig. 5 Schematic of downcomer annulus and reactor vessel cavity

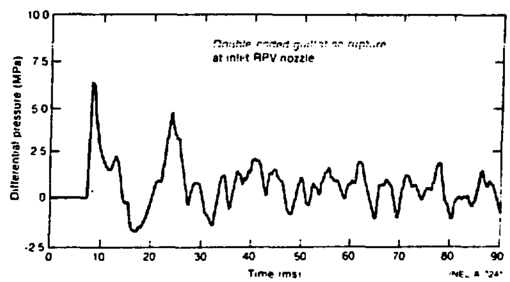


Fig. 6 Circumferential differential pressure across a PWR core barrel at the inlet nozzle centerline

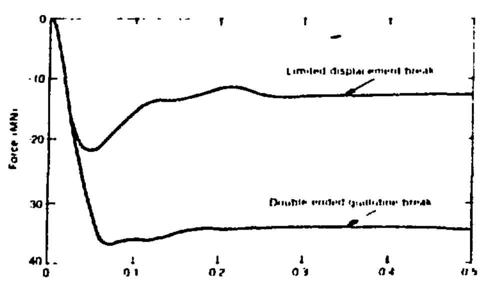


Fig. 7 Cavity pressure force on RPV due to inlet nozzle LOCA

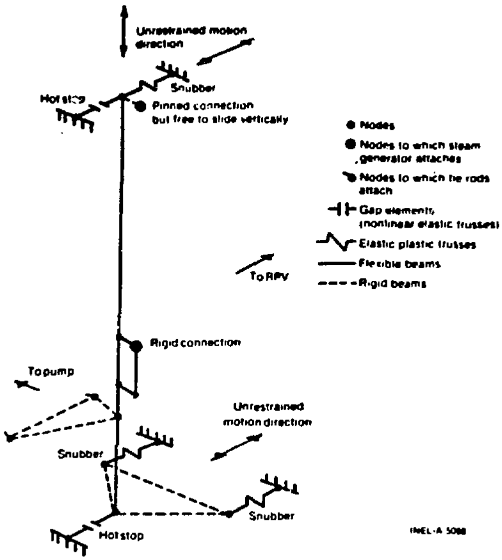


Fig. 8 Steam generator support

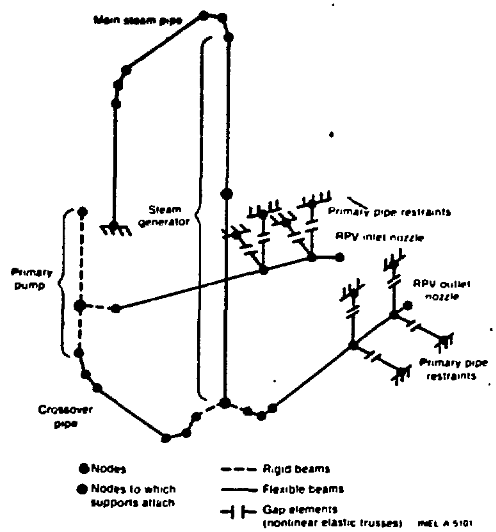


Fig. 9 Typical piping loop model

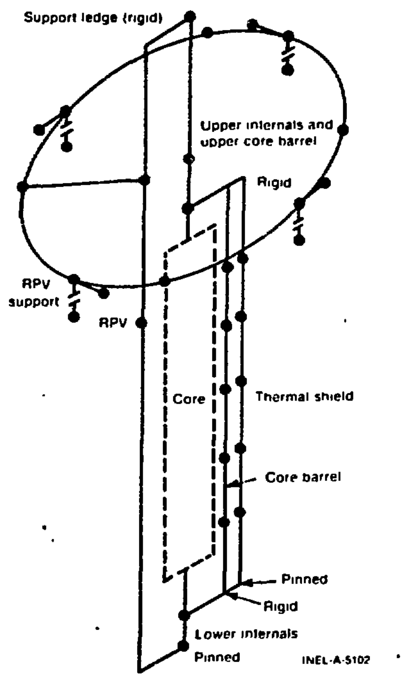


Fig. 10 RPV and internals model

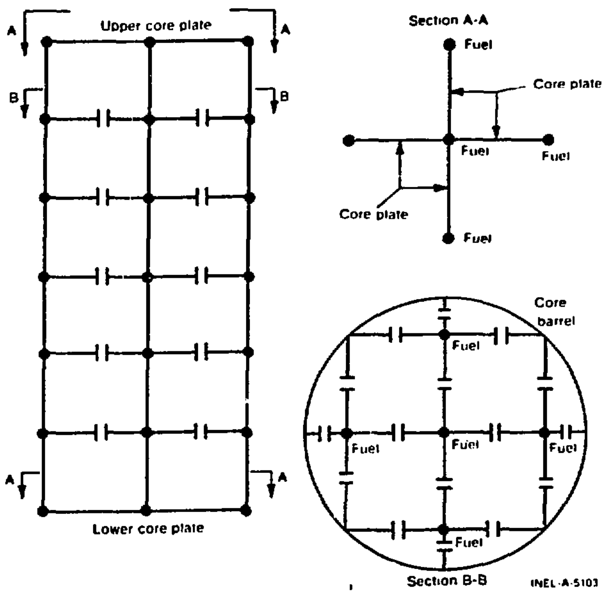


Fig. 11 Core model

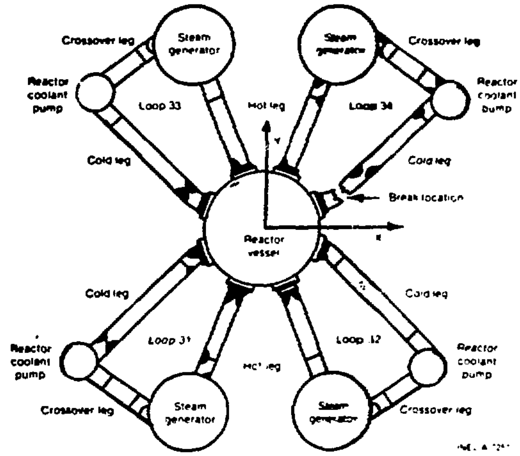


Fig. 12 Plan view of primary system with areas of plasticity shown

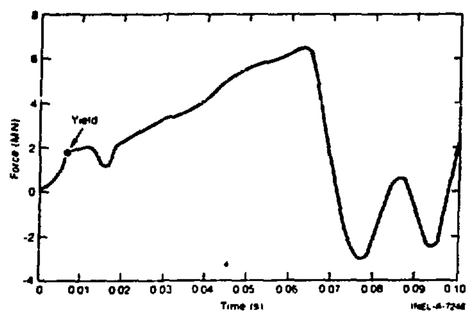


Fig. 13 Horizontal force in RPV support on loop 32

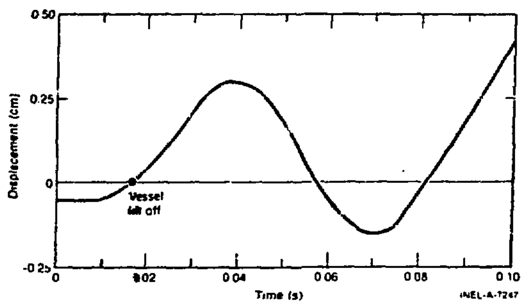


Fig. 14 Vertical displacement of RPV at nozzle elevation

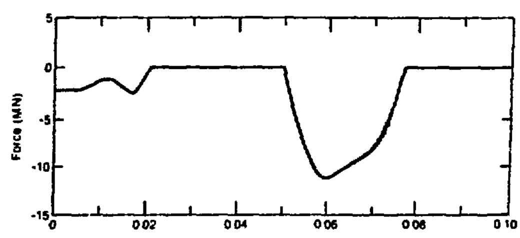
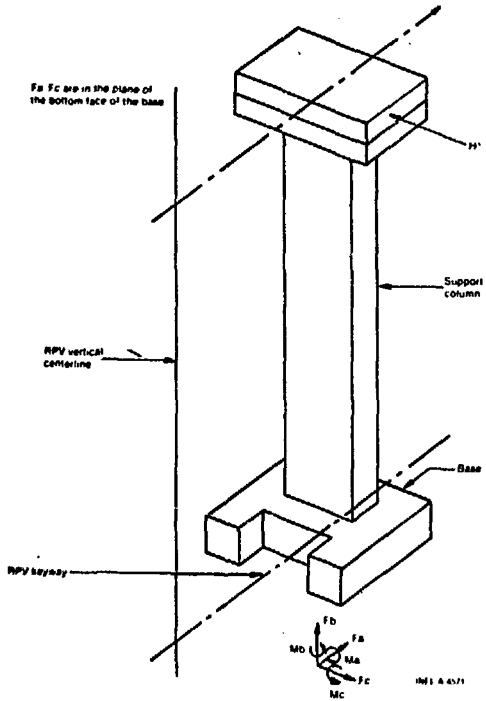


Fig. 15 Vertical force on RPV support in loop 31



	No Gaps	With Gaps
H	13.3	17.5
F_a	0.013	0.053
F_b	12.5	11.3
F_c	0.053	0.236
M_a	0.205	0.689
M_b	0.232	0.307
M_c	0.434	0.502

Forces are in meganewtons; moments are in meganewton-meters.

Fig. 16 Comparison of RPV support loads using linear and nonlinear analysis techniques