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CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE TO AEC GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

3.1.1 SUMMARY DESCRIPTION

This section contains an evaluation of the design basis of the DAEC as measured against the AEC General Design Criteria (GDC) for nuclear power plants, Appendix A, of 10 CFR 50 effective May 21, 1971, and subsequently amended July 7, 1971. The GDC, which are divided into 6 groups and total 55 in number, are intended to establish minimum requirements for the design of nuclear power plants.

It should be noted that the GDC were not written specifically for the BWR; rather, they were intended to guide the design of all water-cooled nuclear power plants. As a result, the criteria are generic in nature and subject to a variety of interpretations. For this reason, there are some cases where conformance of plant design to the interpretation of the criterion is discussed. For each of the 55 criteria, a specific assessment of the plant design is made, and a complete list of references is included to identify where detailed design information pertinent to each criterion is treated in the UFSAR.

Based on the content herein, NextEra Energy Duane Arnold concludes that the nuclear power plant known as the DAEC fully satisfies and is in compliance with the GDC.

3.1.2 CRITERION CONFORMANCE

3.1.2.1 Group I, Overall Requirements

3.1.2.1.1 Criterion 1 - Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

EVALUATION AGAINST CRITERION 1

Particular attention has been directed in the design of the DAEC to the identification of structures, systems, and components important to safety, and special engineering efforts have been expended to ensure the highest quality commensurate with safety-related requirements.

The measures employed to obtain high quality are determined by sound engineering practice drawn from substantial experience in design, procurement, and construction. Individual sections of the UFSAR provide the details of system design, fabrication, erection, and testing.

The codes and standards that have been applied to safety-related structures, systems, and components are broadly accepted in industry and are listed in the applicable UFSAR sections. Codes and standards for nuclear system components are detailed in Section 3.2. Where necessary, supplemental or modified guidelines have been established to ensure that the safety-related components can achieve their design functions.

The comprehensive Quality Assurance Program for the project is described in Chapter 17. It embodies a system of procedures and controls to ensure quality throughout the design, manufacture, and construction processes.

The Quality Assurance Program described in Chapter 17 includes procedures whereby records of the design, fabrication, erection, and testing of structures, systems, and components important to safety are maintained. These measures are documented in a permanent file maintained at the jobsite.

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

EVALUATION AGAINST CRITERION 2

Structures, systems, and components important to safety have been designed to withstand postulated natural phenomena without loss of capability to perform their safety functions. The design bases for safety-related structures with regard to postulated natural phenomena are discussed in Sections 3.2 (seismic), 3.3 (wind and tornado), and 3.4 (flooding).

The most severe natural phenomena that have been reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, have been used to establish the design bases for safety-related structures, systems, and components. The selection of design-basis environmental events is discussed in Sections 2.3 (tornado), 2.4 (flood), and 2.5 (earthquake).

Appropriate combinations of normal operational and accident loadings and loadings due to postulated natural phenomena have been considered in the selection of design bases for safety-related structures, systems, and components and are outlined in Section 3.8.

3.1.2.1.3 Criterion 3 - Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

EVALUATION AGAINST CRITERION 3

The probability and effects of fires have been considered for structures, systems, and components important to safety. The fire protection system is described in Section 9.5.1.

Noncombustible and heat-resistant materials have been used wherever practicable throughout the plant, and combustible materials are stored remotely from the plant to isolate the hazard. Cable separation in fire hazard areas is discussed in Section 8.3.

Fire detection and fighting systems have been designed in accordance with NFPA standards, and the design has been approved by NEPIA.

The high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems are protected by dry pipe water deluge systems so that a break or rupture in the overhead piping at any time, other than during a fire, will not leak water onto the equipment. A failure of the heat-detecting device over the protected equipment will sound an alarm in the control room.

The deluge systems are designed to withstand seismic loadings so that fire fighting pipes will not interfere with these systems during or following a seismic event.

The routing of fire system piping is designed such that a failure of the piping will not cause flooding or water damage to areas where safety-related equipment is located.

3.1.2.1.4 Criterion 4 - Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

EVALUATION AGAINST CRITERION 4

Structures, systems, and components important to safety have been designed to accommodate the effects of and be compatible with conservative combinations of temperature, pressure, humidity, and radiation. In determining the values for these parameters, consideration was given to the radiation exposure the components could receive throughout their design lifetime. Discussions of environmental conditions are found in Sections 6.2, 3.11.5, and in several other sections.

The structures, systems, and components important to safety have been protected from dynamic effects by separating redundant counterparts such that no single event can prevent a required safety function and by routing and locating, to the extent practical, these components to avoid potentially hazardous areas. Components have been selected to the extent practicable to minimize potential sources of missiles (see Section 6.2). The means used to preserve the independence of redundant counterparts of safety-related systems are discussed in Chapters 5, 6, 7, and 8.

Special attention has been directed to the effects of pipe movement, jet forces, and missiles within the drywell. This analysis is presented in Section 6.2.1.3.

Dynamic effects external to the plant that are induced by natural phenomena (e.g., tornado-produced missiles) have been appropriately considered in Section 3.8 and the response
2012-012 to Safety Guide 13, Section 1.8. Additional information to qualify separation for the EDGs relating to missile protection is provided in 8.3.1.3.

3.1.2.2 Group II, Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10 - Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

EVALUATION AGAINST CRITERION 10

The reactor core components consist of fuel assemblies, control rods, incore ion chambers, neutron sources, and related items. The mechanical design is based on the conservative application of stress limits, operating experience, and experimental test results. The fuel is designed to provide high integrity over a complete range of power levels including transient conditions.

The reactor protection system is designed to monitor certain reactor parameters, sense abnormalities, and scram the reactor thereby preventing fuel damage when trip points are exceeded. Scram trip setpoints are selected on operating experience and by the safety design basis. There is no case in which the scram trip setpoints allow the core to exceed the thermal-hydraulic safety limits. Power for the protection system is supplied by its own high-inertia ac motor-generator sets. Alternative electric power is available to the reactor protection system buses. The DAEC also has electrical protection assemblies which monitor the electric power in each of the three sources of power (reactor protection system motor-generator sets A and B and the alternate source) to the reactor protection system. See Section 7.2.1.1.2.

The reactor core and associated coolant, control, and protection systems are designed to ensure that the specified fuel design limits are not exceeded during conditions of normal or abnormal operation and therefore meet the requirements of Criterion 10.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2 .	General Plant Description	1.2
3.	Fuel System Design	4.2
4.	Nuclear Design	4.3
5.	Thermal and Hydraulic Design	4.4
6.	Descriptive Information of Control Rod Drive System	3.9.4.1
7.	Reactor Recirculation System	7.7.5
8.	Reactor Core Isolation Cooling System	5.4.6
9.	Residual Heat Removal System	5.4.7

10. Reactor Trip System 7.2

11. Accident Analyses Chapter 15

3.1.2.2.2 Criterion II - Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

EVALUATION AGAINST CRITERION 11

The reactor core is designed to have a reactivity response that regulates or damps changes in power level and spatial distributions of power production to a level consistent with safe and efficient operation.

The inherent dynamic behavior of the core is characterized in terms of (1) fuel temperature or Doppler coefficient, (2) moderator void coefficient, and (3) moderator temperature coefficient. The combined effect of these coefficients in the power range is termed the power coefficient.

Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it; it contributes to system stability. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void coefficient to Doppler coefficient for optimum load following capability. The BWR has an inherently large moderator-to-Doppler coefficient ratio that permits the use of coolant flow rate for load following.

In a BWR, the moderator void coefficient is of primary importance during operation at power. Nuclear design requires the void coefficient inside the fuel channel to be negative. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative moderator coefficients of reactivity, the BWR has a number of inherent advantages, such as (1) the use of coolant flow as opposed to control rods for load following, (2) the inherent self-flattening of the radial power distribution, (3) the ease of control, and (4) the spatial xenon stability.

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative.

The reactor core and associated coolant system are designed so that in the power operating range prompt inherent dynamic behavior tends to compensate for any rapid increase in reactivity in accordance with Criterion 11.

For further discussion, refer to the following sections:

1.	Principal Design Criteria	1.2 6
2.	Nuclear Design	4.3
3.	Thermal and Hydraulic Design	4.4
4.	(Nuclear System) Stability	4.3.2.7

3.1.2.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

The reactor-core and associated coolant, control, and protection systems shall be designed to ensure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

EVALUATION AGAINST CRITERION 12

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large BWRS, underdamped, unacceptable power distribution behavior could only be expected to occur with power coefficients greater than about 0.01 $\Delta k/k/\Delta P/P$. Operating experience has shown large BWRs to be inherently stable against xenon-induced power instability. The large negative operating coefficients provide the following:

1. Good load following with well damped behavior and little undershoot or overshoot in the heat-transfer response.

2. Load following with recirculation flow control.

3. Strong damping of spatial power disturbances.

The reactor protection system design provides protection from excessive fuel cladding temperatures and protects the nuclear system process barrier from excessive pressure that threatens the integrity of the system. Local abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation.

The reactor core and associated coolant, control, and protection systems are designed to suppress any power oscillations that could result in exceeding fuel design limits. These systems ensure that Criterion 12 is met.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Nuclear Design	4.3
4.	Thermal and Hydraulic Design	4.4
5.	Overpressurization Protection	5.2.2
6.	Reactor Trip System	7.2
7.	Reactor Manual Control System	7.7.3
8.	(Nuclear System) Stability	4.3.2.7
9.	Accident Analyses	Chapter 15

3.1.2.2.4 Criterion 13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

EVALUATION AGAINST CRITERION 13

The fission process is monitored and controlled for all conditions from source range through power operating range. The neutron monitoring system detects core conditions that threaten the overall integrity of the fuel barrier resulting from excess power generation and provides a signal to the reactor protection system. Fission counters, located in the core, are used for the source range through power operating range. The detectors are located to provide maximum sensitivity to control rod movement during startup and to provide optimum monitoring in the intermediate and power ranges.

The source range monitor subsystem (SRMS) provides neutron flux information during reactor startup and low flux level operations. Detectors are inserted into the core for a reactor startup and may be withdrawn after neutron flux is indicated on the intermediate range monitor subsystem (IRMS).

The IRMS monitors neutron flux from the upper portion of the SRMS to the lower portion of the power range monitor subsystem (PRMS). The IRMS is capable of generating a trip signal to block rod withdrawal or to scram the reactor.

The local power range monitor subsystem (LPRMS) consists of fission chambers located throughout the core, the signal conditioning equipment, and trip functions. LPRMS signals are also used in the average power range monitor subsystem (APRMS), rod block monitor subsystem RBMS), and process computer. The RBMS is designed to prevent local fuel damage as a result of a single rod withdrawal error under the worst permitted condition of RBM bypass.

The traversing incore probe (TIP) subsystem provides a signal proportional to the axial gamma flux distribution of the core. This system provides a means to accurately calibrate the LPRM signal by correlation with the TIP signal.

The reactor protection system protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined setpoints are exceeded.

The reactor manual control system consists of the electric circuitry, switches, indicators, and alarm devices required to provide for the manipulation of the control rods and surveillance equipment. The separation of the scram and normal rod control function prevents failures in the reactor manual control circuitry from affecting the scram circuitry.

Reactor vessel instrumentation monitors the transient reactor vessel temperatures, water levels, water flow, internal pressure, and water leakage detection from the top head flange. This information is used to assess conditions existing inside the vessel and the physical condition of the reactor vessel. Reactor vessel temperatures are recorded on a multipoint recorder in the control room. Controlled heating and cooling rates allow thermal stress to be appropriately limited. Reactor vessel water level is also indicated in the control room. Recirculation loop flow, core flow, and differential pressure between the reactor vessel annulus outside of the core and the core inlet plenum are indicated in the control room.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and nuclear system process barrier, the primary containment and reactor vessel isolation control system initiates automatic isolation of appropriate pipelines that penetrate the primary containment whenever monitored variables exceed preselected operational limits (see response to Criteria 55 and 56).

Nuclear system leakage limits are established so that appropriate action can be taken to ensure the integrity of the nuclear system process barrier. Nuclear system leakage rates are classified as identified and unidentified, which correspond respectively to the flow to the equipment drain and drywell floor drains sumps. The permissible total leakage rate limit to these sumps is based on the makeup capabilities of various reactor component systems. Flow integrator and recorders are used to determine the leakage flow pumped from the drain sumps.

A plant process computer system receives input from plant variables including all variables of the reactor protection system. The inputs are scanned and monitored for change of state and provide a quick and accurate determination of the core thermal performance. Certain inputs are annunciated to aid in general plant operation. The data collection and processing, display, print, alarm, and logging functions supplement procedural requirements for control rod display and print during reactor startup and shutdown.

Although the plant process computer is a valuable aid to the operator, it is not required for the safe operation of the plant. The plant process computer also provides core fuel performance analysis and display (see Section 7.7.4.5.6).

As noted above, adequate instrumentation has been provided to monitor system variables in the reactor core, reactor coolant pressure boundary, and reactor containment. Appropriate controls have been provided to maintain the variables in the operating range and to initiate the necessary corrective action in the event of an abnormal operational occurrence or accident. These instrumentation and controls meet the requirements of Criterion 13.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Main Steam Line Isolation System	5.4.5
4.	Detection of Leakage Through Reactor Coolant Pressure Boundary	5.2.5
5.	Containment Systems	6.2
6.	Reactor Trip System	7.2
7.	Primary Containment Isolation and Nuclear Steam Supply Shutoff System	7.3.1
8.	Neutron Monitoring System	7.6.1
9.	Reactor Manual Control System	7.7.3

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

EVALUATION AGAINST CRITERION 14

The piping and equipment pressure parts within the reactor coolant pressure boundary through the outer isolation valves are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2.2 classifies the systems and components within the reactor coolant pressure boundary as Code Group A. The design requirements and codes and standards applied to this code group ensure a quality product in keeping with the safety functions to be performed.

In order to minimize the possibility of brittle fracture within the reactor coolant pressure boundary, the fracture or notch properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness when the system is pressurized to more than 20% of the design pressure. Section 5.3 describes the methods used to control notch toughness properties by selecting and testing fine-grained steels and limiting neutron exposure of materials to acceptable levels. Materials to be impact tested are tested by the Charpy V-notch method in accordance with ASME Boiler and Pressure Vessel (B&PV) Code, Section III. By maintaining a material service temperature of at least 60°F above the nil ductility transition (NDT) temperature for the reactor coolant pressure boundary, adequate protection is further ensured. Where reactor coolant pressure boundary piping penetrates the containment, the fracture toughness temperature requirements of the reactor coolant pressure boundary material apply.

Piping and equipment pressure parts of the reactor coolant pressure boundary are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures are employed which produce welds of complete penetration, of complete fusion, and free of unacceptable defects. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of Section IX of the ASME B&PV Code for the materials to be welded. Qualification records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder, are maintained.

Sections 5.2.1 and 17.1.10 contain the detailed material and examination requirements for the piping and equipment of the reactor coolant pressure boundary before and after its assembly and erection. Leakage testing and surveillance are accomplished as described in the evaluation against GDC 30.

The design, fabrication, erection, and testing of the reactor coolant pressure boundary ensure an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Reactor Vessel	5.3
3.	Reactor Recirculation System	5.4.1
4.	Overpressurization Protection	5.2.2
5.	Reactor Vessel Instrumentation	7.6.4
6.	Analysis of Incidents of Moderate Frequency and Infrequent Incidents	Chapter 15
7.	Pressure Integrity of Piping and Equipment Pressure Parts	3.2
8.	Structural Design and Loading Criteria	3.8
9.	Quality Assurance Program	Chapter 17

3.1.2.2.6 Criterion 15 - Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

EVALUATION AGAINST CRITERION 15

The reactor coolant system consists of the reactor vessel and appurtenances, the reactor recirculation system, the nuclear pressure relief system, the main steam lines, the RCIC system, and the residual heat removal (RHR) system. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards which ensure high integrity of the reactor coolant pressure boundary throughout the plant lifetime. The reactor coolant system is designed and fabricated to meet the following as a minimum:

1. Reactor Vessel - ASME B&PV Code, Section III, Subsection A.

2. Pumps - ASME B&PV Code, Section III, Subsection C.

3. Piping and Valves - ANSI B31.1 and B31.7 as described in Section 3.2.2

The auxiliary, control, and protection systems associated with the reactor coolant system act to provide sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. As described in the evaluation of Criterion 13, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

An example of the integrated protective action scheme that provides sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded is the automatic initiation of the nuclear system pressure relief system on receipt of an overpressure signal. To accomplish over-pressure protection, a number of pressure-operated relief valves are provided that can discharge steam from the nuclear system to the primary containment. The nuclear system pressure relief system also provides for automatic depressurization of the nuclear system in the event of a LOCA in which the vessel is not depressurized by the accident. The depressurization of the nuclear system in this situation allows low-pressure emergency core cooling systems to automatically initiate to supply enough cooling water to adequately cool the core. In a similar manner, other auxiliary, control, and protection systems provide assurance that the design conditions of the reactor coolant pressure boundary are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

The application of appropriate codes and standards and high-quality requirements to the reactor coolant system and the design features of its associated auxiliary, control, and protection systems ensures that the requirements of Criterion 15 are satisfied.

For further description, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Reactor Vessel	5.3
3.	Reactor Recirculation System	5.4.1
4.	Overpressurization Protection	5.2.2
5.	Detection of Leakage Through Reactor Coolant Pressure Boundary	5.2.5
6.	Reactor Vessel Instrumentation	7.6.4
7.	Analysis of Incidents of Moderate Frequency and Infrequent Incidents	Chapter 15
8.	Pressure Integrity of Piping and Equipment Pressure Parts	3.2
9.	Structural Design and Loading Criteria	3.8

3.1.2.2.7 Criterion 16 - Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

EVALUATION AGAINST CRITERION 16

The reactor is housed within a drywell containment vessel made of steel plates of 0.75 to 3 in. thickness. Reinforced concrete ranging in thickness from 4 to 7 ft is placed around the drywell vessel. The ability of the containment vessel to provide a leaktight barrier against the uncontrolled release of radioactivity is verified by periodic leakage tests that are performed throughout the life of the plant. Additional description of the containment is found in Section 6.2.

The containment cooling subsystem of the RHR system has been provided to ensure that containment design conditions will not be exceeded during the course of the postulated accident. This subsystem is discussed in Section 5.4.7.

3.1.2.2.8 Criterion 17 - Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function of each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network on the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to ensure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to ensure that core cooling containment integrity and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

EVALUATION AGAINST CRITERION 17

Both the onsite and offsite electric power systems are able to provide power to the essential distribution system that services the structures, systems, and components important to safety. Each of these sources has adequate capacity and capability to furnish the required power for all postulated operational and accident conditions. Chapter 8 provides a detailed description of the electric power system including capacity and load sequence information.

The onsite electric power sources and distribution systems have adequate independence and redundancy as discussed in the response to Safety Guide 6, Section 1.8. Onsite power source capacity and response are discussed in Chapter 8 and in the response to Safety Guide 9, Section 1.8. Single active failures are considered. Testability is discussed in Criterion 18.

2016-015 [redacted]

Each of the incoming transmission lines will normally be connected to the DAEC switchyard except for short periods during the maintenance of one of the lines. One or more of these lines will be continually connected to the startup transformer to supply power immediately to the essential 4160-V buses in the event of a LOCA. In the event of a subsequent failure of the startup transformer or its incoming line, the emergency 4160-V buses will be automatically transferred to the standby transformer.

The protective relaying and buses in the switchyard are designed so that a fault on one of the transmission lines will be isolated from the other lines and will not interrupt power from the other sources. The transmission system is discussed in Section 8.2.

Stability analyses have been conducted to ensure that a loss of power generated by the DAEC or a loss of power from the transmission network will have a low probability of causing a loss of electrical power from any of the remaining sources (see Section 8.2.2).

Capacity increase analyses have been conducted to determine the necessary var compensation required to provide adequate system voltage levels and maintain a secure network during worst case scenario events. (see Section 8.2.2)

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

EVALUATION AGAINST CRITERION 18

Electric power systems important to safety have been designed to permit periodic inspection and testing. These aspects are discussed in the following sections:

1.	Auxiliary Power System	8.3.1
2.	Standby AC Power Supply and Distribution	8.3.1
3.	DC Power Systems	8.3.2

The inspection and testing program extends from manufacture through installation to operation in order to verify overall quality and compliance. In general, the operability of components is verified by tests of the system operation. Section 8.3 discusses the operational testing of the normal and standby power systems, while other sections discuss the tests that are conducted to verify the operability of that particular safety-related system.

3.1.2.2.10 Criterion 19 - Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

EVALUATION AGAINST CRITERION 19

A control room has been provided in which appropriate controls and instrumentation are located to permit personnel to safely operate the nuclear power unit or maintain it in a safe condition under accident conditions. The radiation protection afforded control room personnel permits the required habitability and is discussed in Section 6.4.

An alternate shutdown capability system has been provided to permit safe shutdown of the plant in the event that the main control room becomes uninhabitable due to a design-basis fire. The system is described in Section 7.4.2.1.

3.1.2.3 Group III, Protection and Reactivity Control Systems

3.1.2.3.1 Criterion 20 - Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

EVALUATION AGAINST CRITERION 20

The reactor protection system is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier. Fuel damage is prevented by the initiation of an automatic reactor shutdown if monitored nuclear system variables exceed preestablished limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The reactor protection system includes the motor-generator power system, sensors, relays, bypass circuitry, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear system high pressure, turbine stop valve closure, turbine control valve fast closure, and reactor vessel low water level will prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding the thermal-hydraulic safety limit during incidents of moderate frequency and infrequent incidents. The response by the reactor protection system is prompt and the total scram time is short.

In addition to the reactor protection system that provides for automatic shutdown of the reactor to prevent fuel damage, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to safety. Systems such as the emergency core cooling system initiate automatically to prevent or limit the extent of fuel damage following a LOCA. Other systems automatically isolate the reactor vessel or the primary containment to prevent or limit the extent of fuel damage following a postulated LOCA and prevent the release of significant amounts of radioactive materials from the fuel and the nuclear system process barrier. The controls and instrumentation for the emergency core cooling systems and the isolation systems initiate automatically when monitored variables exceed preselected operational limits.

The design of the protection system satisfies the functional requirements as specified in Criterion 20.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Descriptive Information of Control Rod Drive System	3.9.4.1
4.	Overpressurization Protection	5.2.2
5.	Main Steam Line Isolation System	5.4.5
6.	Emergency Core Cooling System	6.3
7.	Reactor Trip System	7.2
8.	Primary Containment Isolation and Nuclear Steam Supply Shutoff System	7.3.1
9.	Emergency Core Cooling Systems Control	7.3.2.2
10.	Neutron Monitoring System	7.6.1
11.	Process Radiation Monitoring System	11.5
12.	Accident Analyses	Chapter 15

3.1.2.3.2 Criterion 21 - Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to ensure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

EVALUATION AGAINST CRITERION 21

The reactor protection system design fulfills the single-failure criterion by providing redundant channels. No single component failure, intentional bypass maintenance operation, calibration operation, or test to verify operational availability will impair the ability of the system to perform its intended safety function. In addition, the system design ensures that when a scram trip point is exceeded there is a high scram probability. However, should a scram not occur, other monitored components will scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between trip logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

The reactor protection systems include design features that permit inservice testing. This ensures the functional reliability of the system should the reactor variable exceed the corrective action setpoint.

The reactor protection system initiates an automatic reactor shutdown if the monitored nuclear variables exceed pre-established limits. The protection system consists of two separately powered trip systems. Each trip system has three trip logics, two of which produce an automatic trip signal. The remaining logic is used for a manual trip signal. To produce a scram, at least one logic from each trip system must be tripped. The overall logic scheme is a one-out-of-two twice arrangement.

The reactor protection system can be tested during reactor operation by five separate tests. Manual scram testing is performed by operating one of the two manual scram controls. This tests one trip system. The total test verifies the ability to de-energize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive (CRD) operability can be tested during normal reactor operation. Drive position indicator and incore neutron instrumentation are used to verify control rod movement. Each control rod can be withdrawn one notch and then reinserted to the original position without significantly upsetting the reactor. One control rod is tested at a time. The control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. More importantly, the hydraulic control unit scram accumulator and the scram discharge volume level are continuously monitored.

The main steam line isolation valves may be tested during full reactor operation. They can be closed to 90% of full-open position without affecting the reactor operation. If reactor power is reduced to < 76% of full power, one set of isolation valves may be fully closed assuming all four steamlines are in operation. If the plant is operating on only three steamlines, this testing is limited to < 41 % power (780 MWt). During refueling operations, the valve leakage rate can be determined.

RHR system testing can be performed during normal operation. Main system pumps can be evaluated by taking suction from the suppression pool and discharging through test lines back to the suppression pool. System design and operating procedures also permit testing the discharge valves to the reactor recirculation loops and discharge valves to the containment spray headers. The low-pressure coolant injection (LPCI) mode can be tested after reactor shutdown. Each active component of the emergency core cooling systems provided to operate in a design-basis accident (DBA) is designed to be operable for test purposes during normal operation of the nuclear system.

The high functional reliability, redundancy, and inservice testability of the protection system satisfy the requirements specified in Criterion 21.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Main Steam Line Isolation System	5.4.5
4.	Residual Heat Removal System	5.4.7
5.	Containment System	6.2
6.	Emergency Core Cooling Systems	6.3
7.	Reactor Trip System	7.2
8.	Primary Containment Isolation and Nuclear Steam Supply Shutoff System	7.3.1
9.	Emergency Core Cooling System	7.3.2
10.	Neutron Monitoring System	7.6.1.1
11.	Process Radiation Monitoring System	11.5
12.	Accident Analysis	Chapter 15

3.1.2.3.3 Criterion 22 - Protection System Independence

The protection -system shall be designed to ensure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

EVALUATION AGAINST CRITERION 22

The components of protection systems are designed so that the mechanical and thermal environment resulting from any emergency situation in which the components are required to function will not interfere with that function. Wiring for the reactor protection system outside of the control room enclosures is run in rigid metallic conduit. No other wiring is run in this conduit. The wires from duplicate sensors on a common process tap are run in separate conduits. The system sensors are electrically and physically separated. Only one trip actuator logic circuit from each trip system may be run in the same conduit.

The reactor protection systems are designed to permit maintenance and diagnostic work while the reactor is operating without restricting the plant operation or hindering the output of their safety functions. The diversity in the principle of operation of the protection system which allows operational system testing is the one-out-of-two logic arrangement and the use of an independent trip channel for each trip logic input. When an essential monitored variable exceeds its scram trip point, it is sensed by at least two independent sensors in each trip system. An intentional bypass, maintenance operation, calibration operation, or test will result in a single channel trip. This leaves at least two trip channels per monitored variable capable of initiating a scram. Only one trip channel in each trip system must trip to initiate a scram. Thus, the arrangement of two trip channels per trip system ensures that a scram will occur as a monitored variable exceeds its scram setting.

The protection system meets the design requirements for functional and physical independence as specified in Criterion 22.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Main Stream Line Isolation System	5.4.5
4.	Residual Heat Removal System	5.4.7
5.	Emergency Core Cooling System	6.3

6.	Reactor Trip System	7.2
7.	Primary Containment Isolation and Nuclear Steam Supply Shutoff System	7.3.1
8.	Emergency Core Cooling System Control and Instrumentation	7.3.2
9.	Neutron Monitoring System	7.6.1.1
10.	Process Radiation Monitoring System	11.5
11.	Accident Analyses	Chapter 15

3.1.2.3.4 Criterion 23 - Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

EVALUATION AGAINST CRITERION 23

The reactor protection system is designed to fail into a safe state. The use of an independent trip channel for each trip logic allows the system to sustain any trip channel failure without preventing other sensors monitoring the same variable from initiating a scram. A single sensor or trip channel failure will cause a channel trip. Only one trip channel in each trip system must trip to initiate a scram. Intentional bypass, maintenance operation, calibration operation, or test will result in a single-channel trip. A failure of any one reactor protection system input or subsystem component will produce a trip in one of two channels. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function on another trip.

The environmental conditions in which the instrumentation and equipment of the reactor protection system must operate were .considered in establishing the component specifications. Instrumentation specifications for the reactor and turbine building are based on the worst expected ambient conditions in which the instruments must operate.

The failure modes of the protection system are such that it will fail into a safe state as required by Criterion 23.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Emergency Core Cooling System	6.3
3.	Reactor Trip System	7.2
4.	Primary Containment Isolation and Nuclear Steam Supply Shutoff System	7.3.1
5.	Neutron Monitoring System	7.6.1.1
6.	Electric Power Systems	Chapter 8

3.1.2.3.5 Criterion 24 - Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy and independence requirements of all protection systems. Interconnection of the protection and control systems shall be limited so as to ensure that safety is not significantly impaired.

EVALUATION AGAINST CRITERION 24

There is separation between the reactor protection system and the process systems. Sensors, trip channels, and trip logics of the reactor protection system are not used directly for automatic control of process systems. Therefore, failure in the controls and instrumentation of process systems cannot induce the failure of any portion of the protection system. High scram reliability is designed into the reactor protection system and hydraulic control unit for the control rod drive. The scram signal and mode of operation override all other signals. Primary containment and reactor vessel isolation control systems are designed so that any one failure, maintenance operation, calibration operation, or test to verify operational availability will not impair the functional ability of the isolation control system to respond to essential variables.

The protection system is separated from control systems as required in Criterion 24.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Emergency Core Cooling System	6.3

4.	Reactor Trip System	7.2
5.	Primary Containment Isolation and Nuclear Steam Supply Shutoff System	7.3.1
6.	Emergency Core Cooling System Control and Instrumentation	7.3.2
7.	Neutron Monitoring System	7.6.1.1
8.	Process Radiation Monitoring System	11.5

3.1.2.3.6 Criterion 25 - Protection System Requirements For Reactivity Control Malfunctions

The protection system shall be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

EVALUATION AGAINST CRITERION 25

The reactor protection system provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier. Any monitored variable that exceeds the scram setpoint will initiate an automatic scram and not impair the remaining variables from being monitored; and if one channel fails, the remaining portions of the reactor protection system shall function.

The reactor manual control system is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the reactor manual control system is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Because each control rod is controlled as an individual unit, a failure that results in energizing any of the insert or withdraw solenoid valves can affect only one control rod. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod.

The most serious rod withdrawal errors are considered to be when the reactor is just subcritical and an out-of-sequence rod is continuously withdrawn. The rod worth minimizer would normally prevent the withdrawal of rods with worths higher than programmed. If such a continuous rod withdrawal were to occur, the increase in fuel temperature subsequent to scram would not be sufficient to exceed acceptable fuel design limits.

The design of the protection system ensures that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems as specified in Criterion 25.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Nuclear Design	4.3
4.	Thermal and Hydraulic Design	4.4
5.	Reactor Trip System	7.2
6.	Reactor Manual Control System	7.7.3
7.	Accident Analyses	Chapter 15

3.1.2.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to ensure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

EVALUATION AGAINST CRITERION 26

Two independent reactivity control systems using different design principles are provided. The normal method of reactivity control employs control rod assemblies that contain boron-carbide (B_4C) powder. The control of reactivity is operationally provided by a combination of these movable control rods, burnable poisons, and reactor coolant recirculation system flow. These systems accommodate fuel burnup, load changes, and long-term reactivity changes.

Reactor shutdown by the CRD system is sufficiently rapid to prevent the exceeding of acceptable fuel design limits for normal operation and all abnormal operational transients. The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Because each control rod is controlled as an individual unit, a failure that results in energizing any of the insert or withdraw solenoid valves can affect only one control rod. Two sources of scram energy (accumulator pressure and reactor vessel pressure) provide needed scram performance over the entire range of reactor pressure (i.e., from operating conditions to cold shutdown).

The design of the control rod system includes appropriate margin for malfunctions such as stuck rods in the highly unlikely event that they do occur. Control rod withdrawal sequences and patterns are selected before operation to achieve optimum core performance, and, simultaneously, low individual rod worths. The operating procedures to accomplish such patterns are supplemented by the Rod Worth Minimizer system, which prevents rod withdrawals other than those permitted by the preselected rod withdrawal pattern (except for certain operational activities which require bypassing of the RWM). An additional safety design basis of the control rod system requires that the core in its maximum reactivity condition be subcritical with the control rod of the highest worth fully withdrawn and all other rods fully inserted.

Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods. In the event that a reactor scram is necessary, the unlikely occurrence of a limited number of stuck rods will not hinder the capability of the control rod system to render the core subcritical.

A standby liquid control system containing neutron-absorbing sodium pentaborate solution is the independent backup system. This system has the capability to shut the reactor down from full power and maintain it in a subcritical condition at any time during the core life.

The redundancy and capabilities of the reactivity control systems for the BWR satisfy the requirements of Criterion 26.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information on Control Rod Drive System	4.6.1
3.	Standby Liquid Control System	9.3.4
4.	Reactor Manual Control System	7.7.3
5.	Process Computer System	7.7.4
6.	Rod Worth Minimizer	4.6.2.4

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to ensure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

EVALUATION AGAINST CRITERION 27

There is no credible event applicable to the BWR that requires the combined capability of the control rod system and poison additions by the emergency core cooling network. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by the separation of scram and manual control circuitry, individual control units for each control rod, and fail-safe design features built into the CRD system. Response by the reactor protection system is prompt, and the total scram time is short.

In operating the reactor, there is a spectrum of possible control rod worths, depending on the reactor state and on the control rod pattern chosen for operation. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The Rod Worth Minimizer prevents rod withdrawal other than by the preselected rod withdrawal pattern. The rod worth minimizer function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and other power level operations. As a result of this carefully planned procedure, prompt shutdown of the reactor can be achieved with scram insertion of less than half of the many independent control rods. If accident conditions require a reactor scram, this can be accomplished rapidly with appropriate margin for the unlikely occurrence of malfunctions such as stuck rods.

The reactor core design assists in maintaining the stability of the core under accident conditions as well as during power operation. Reactivity coefficients in the power range that contribute to system stability are (1) fuel temperature or Doppler coefficient, (2) moderator void coefficient, and (3) moderator temperature coefficient. The overall power reactivity coefficient is negative and provides a strong negative reactivity feedback under severe power transient conditions.

The design of the reactivity control systems ensures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool the core is maintained under all postulated accident conditions; thus, Criterion 27 is satisfied.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Nuclear Design	4.3
4.	Thermal and Hydraulic Design	4.4

5.	Reactor Trip System	7.2
6.	Reactor Manual Control System	7.7.3
7.	Process Computer System	7.7.4
8.	Accident Analyses	Chapter 15

3.1.2.3.9 Criterion 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

EVALUATION AGAINST CRITERION 28

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The Rod Worth Minimizer system prevents withdrawal other than by the preselected rod withdrawal pattern (except for certain operational activities which require bypassing of the RWM). The rod worth minimizer function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established control rod procedures for startup, shutdown, and other power level operation.

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity to less than 5 fps. Normal rod movement is limited to six inch rod notch increments, and the rod withdrawal rate is limited through the hydraulic valve to 3 in./sec.

Chapter 15 evaluates the postulated reactivity accidents (limiting faults) as well as incidents of moderate frequency and infrequent incidents, in detail. Analyses are included for rod dropout, steam-line rupture, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. The results of these analyses indicate that none of the postulated reactivity transients or accidents result in damage to the reactor coolant pressure boundary. In addition, the integrity of the core, its support structures, or other reactor pressure vessel internals are maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analysis.

The design features of the reactivity control system that limit the potential amount and rate of reactivity increase ensure that Criterion 28 is satisfied for all postulated reactivity accidents.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Reactor Pressure Vessel Internals	3.9.5
3.	Control Rod Drive System Design	4.6.1
4.	Nuclear Design	4.3
5.	Descriptive Information of Control Rod Drive System	3.9.4.1
6.	Reactor Vessel	5.3
7.	Overpressurization Protection	5.2.2
8.	Main Steam-Line Flow Restrictors	5.4.4
9.	Main Steam Line Isolation System	5.4.5
10.	Process Computer System	7.7.4
11.	Accident Analyses	Chapter 15
12.	Pressure Integrity of Piping and Equipment Pressure Parts	3.2.2
13.	Structural Design and Loading Criteria	Chapter 3

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to ensure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

EVALUATION AGAINST CRITERION 29

The high functional reliability of the protection and reactivity control systems is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and inservice testability. These design features are discussed in detail in Criteria 21, 22, 23, 24, and 26.

An extremely high probability of correct protection and reactivity control systems response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Components important to safety, such as control rod drives, main steam isolation valves, and RHR pumps, are tested using normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering, on one hand, the failure probabilities of individual components and, on the other hand, the reliability effects during individual component testing on the portion of the system not undergoing test. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action setpoint.

The capabilities of the protection and reactivity control systems to perform their safety functions in the event of anticipated operational occurrences are satisfied in agreement with the requirements of Criterion 29.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Information for Control Rod Drive System	4.6.1
3.	Main Steam Line Isolation System	5.4.5
4.	Residual Heat Removal System	5.4.7
5.	Containment System	6.2
6.	Emergency Core Cooling System	6.3
7.	Reactor Trip System	7.2
8.	Primary Containment Isolation and Nuclear Steam Supply	7.3.1

9.	Emergency Core Cooling Systems Control and Instrumentation	7.3.2
10.	Neutron Monitoring System	7.6.1
11.	Process Radiation Monitoring	11.5
12.	Accident Analyses	Chapter 15

3.1.2.4 Group IV, Fluid System

3.1.2.4.1 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

EVALUATION AGAINST CRITERION 30

By using conservative design practices and detailed quality control procedures, the pressure-retaining components of the reactor coolant pressure boundary are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components that comprise the reactor coolant pressure boundary are designed; fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Table 3.2-2. Furthermore, product and process quality planning is provided as described in Chapter 17 to ensure conformance with the applicable codes and standards and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the reactor coolant pressure boundary, further discussion on this subject is treated in the response to Criterion 14.

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases, isolate the reactor coolant pressure boundary from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration in the primary containment atmosphere. In addition to these means of protection, large leaks are detected by changes in flow rates in process lines and changes in reactor water level. The allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant systems leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal ac power concomitant with a loss of feedwater supply, makeup capabilities are provided by the CRD and RCIC systems. While the leak detection system provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The reactor coolant pressure boundary and the leak detection system are designed to meet the requirements of Criterion 30.

For further discussion, see the following sections:

1.	Principal Design Criteria	1.2.6
2.	Reactor Vessel	5.3
3.	Reactor Recirculation System	5.4.1
4.	Overpressurization Protection	5.2.2
5.	Detection of Leakage Through Reactor Coolant Pressure Boundary	5.2.5
6.	Reactor Vessel Instrumentation	7.6.4
7.	Accident Analyses	Chapter 15
8.	Pressure Integrity of Piping and Equipment Pressure Parts	3.2
9.	Structural Design and Loading Criteria	Chapter 3
10.	Quality Assurance Program	Chapter 17

3.1.2.4.2 Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect the consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

EVALUATION AGAINST CRITERION 31

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel, the following steps have been taken:

1. The initial ductile-brittle transition temperature of materials used in the reactor vessel is known by reference or established empirically.

2. Expected shifts in transition temperature during design service life due to neutron flux are determined and employed in the reactor vessel design.

The NDT temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than in a ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1×10^{17} nvt with neutrons of energies in excess of 1 MeV. Since the material NDT temperature dictates the minimum operating temperature at which the reactor vessel can be pressurized, it is desirable to keep the NDT temperature as low as possible.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident on the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant.

The fracture or notch toughness properties and the operating temperature of ferritic materials of the reactor coolant pressure boundary are controlled to ensure adequate toughness when the system is pressurized to more than 20% of the design pressure. Such assurance is provided by maintaining a material service temperature at least 60°F above the NDT temperature.

The reactor coolant pressure boundary is designed, maintained, and tested such that adequate assurance is provided that the boundary will behave in a nonbrittle manner throughout the life of the plant. Therefore, the reactor coolant pressure boundary is in conformance with Criterion 31.

For further discussion, see the following sections:

1.	Reactor Vessel	5.3
2.	Pressure Integrity of Piping and Equipment Pressure Parts	3.2

3.1.2.4.3 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.

EVALUATION AGAINST CRITERION 32

The reactor pressure vessel design and engineering effort include provisions for inservice inspection. Removable plugs in the sacrificial shield and/or removable panels in the insulation provide access for the examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system safety and relief valves, recirculation system, and on the main steam and feedwater systems extending out to and including the first isolation valve outside containment. The inspection of the reactor coolant pressure boundary is in accordance with the ASME B&PV Code, Section XI, 1989 edition. Although the design and engineering effort predates the ASME Code, maximum access has been provided within the limits of drywell design. Section 5.2.4 defines the inservice inspection plan, access provisions, and areas of restricted access.

The reactor recirculation piping and main steam piping are pressure tested in accordance with the Inservice Inspection Plan.

Vessel material surveillance samples will be located within the reactor pressure vessel to enable periodic monitoring of material properties with exposure. The program will include specimens of the base metal and heat-affected zone metal.

The plant testing and inspection programs ensure that the requirements of Criterion 32 will be met.

For further discussion, see the following sections:

1.	Reactor Vessel	5.3
2.	Reactor Recirculation System	5.4.1
3.	Detection of Leakage Through Reactor Coolant Pressure Boundary	5.2.5
4.	Pressure Integrity of Piping and Equipment Pressure Parts	3.2
5.	Inservice Inspection and Testing of Reactor Coolant Pressure Boundary	5.2.4

3.1.2.4.4 Criterion 33 - Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to ensure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

EVALUATION AGAINST CRITERION 33

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases, isolate the reactor coolant pressure boundary from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration in the primary containment atmosphere. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines, and changes in reactor water level. The allowable leakage rates have been based on predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal ac power concomitant with a loss of feedwater supply, makeup capabilities are provided by the CRD and RCIC systems. While the leak detection system provides protection from small leaks, the emergency core cooling system provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges to the extent that fuel clad temperature limits are not exceeded.

The plant is designed to provide ample reactor coolant makeup for protection against small leaks in the reactor coolant pressure boundary for anticipated operational occurrences and postulated accident conditions. The design of these systems meets the requirements of Criterion 33.

For further discussion, see the following sections:

1. Detection of Leakage Through Reactor Coolant Pressure Boundary — 5.2.5

2. Reactor Core Isolation Cooling System — 5.4.6

3. Emergency Core Cooling System 6.3

4. Reactor Vessel Instrumentation 7.6.4

3.1.2.4.5 Criterion 34 - Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION AGAINST CRITERION 34

The RHR system provides the means to

1. Remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed.

2. Supplement the fuel pool cooling system capacity during shutdown to provide additional cooling capacity.

The major equipment of the RHR system consists of two heat exchangers, four main system pumps, and four service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation. The main system pumps are sized on the basis of the flow required during the LPCI mode of operation, which is the mode requiring the maximum flow rate. The heat exchangers are sized on the basis of the required duty for the shutdown cooling function, which is the mode requiring the maximum heat exchanger capacity.

One loop, consisting of a heat exchanger, two main system pumps in parallel, and associated piping, is located in one area of the reactor building. The other heat exchanger, pumps, and piping, forming a second loop, are located in another area of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system. The two loops of the RHR system are cross connected by a single header (with the exception of a small line connecting the loops and the Shutdown Cooling Suction Piping in order to create a differential pressure across the LPCI Inject Check Valves), making it possible to supply either loop from the pumps in the other loop.

The RHR system is designed for the following modes of operation:

1. Shutdown cooling.

2. Containment cooling.

3. Low-pressure coolant injection.

Both normal ac power and auxiliary onsite power systems provide enough power to operate all the auxiliary loads necessary for plant operation. The power sources for the plant auxiliary power system are sufficient in number, and of such electrical and physical independence, that no single probable event could interrupt all auxiliary power at one time.

The plant auxiliary buses supplying power to engineered safety features and reactor protection systems and those auxiliaries required for safe shutdown are connected by appropriate switching to either of two standby diesel-driven generators located in the plant. Each power source, up to the point of its connection to the auxiliary power buses, is capable of complete and rapid isolation from any other source.

Loads important to plant operation and safety are split and diversified between switchgear sections, and means are provided for the detection and isolation of system faults.

The plant layout is designed to effect the physical separation of essential bus sections, standby generators, switchgear, interconnections, feeders, power center, motor control centers, and other system components.

Two full-capacity 2850-kW standby diesel-generators are provided to supply a source of electric power that is self-contained within the plant and is not dependent on external sources of supply. The standby generators produce ac power at a voltage and frequency compatible with the normal bus requirements for essential equipment within the plant. Each of the diesel-generators has sufficient capacity to start and carry the essential loads it is expected to drive. All of the auxiliary loads required for safe and orderly shutdown including components of the RHR system are duplicated and connected to separate buses.

The RHR systems are adequate to remove residual heat from the reactor core to ensure that fuel and reactor coolant pressure boundary design limits are not exceeded. Redundant offsite and onsite electric power systems are provided. The design of the RHR system, including the power supply, meets the requirements of Criterion 34.

For further discussion, see the following sections:

1. Residual Heat Removal System 5.4.7

2. Emergency Core Cooling System 6.3

3.	Emergency Core Cooling Systems Control and Instrumentation	7.3.2
4.	Auxiliary Power System	8.3.1
5.	Standby AC Power Supply and Distribution	8.3.1
6.	Residual Heat Removal Service Water and Emergency Service Water Systems	9.2.3
7.	Accident Analyses	Chapter 15

3.1.2.4.6 Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION AGAINST CRITERION 35

The emergency core cooling systems (ECCS) consist of the following:

1. HPCI system.

2. Automatic depressurization system.

3. Core spray system.

4. LPCI system (an operating mode of the RHR system).

The emergency core cooling systems are designed to limit the fuel-cladding temperature over the complete spectrum of possible break sizes in the nuclear system process barrier including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The HPCI system consists of a steam turbine, a constant-flow pump, system piping, valves, controls, and instrumentation. The HPCI system is provided to ensure that the reactor core is adequately cooled to prevent excessive fuel-cladding temperatures for breaks in the nuclear system that do not result in rapid depressurization of the reactor vessel. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or core spray system operation maintains core cooling. Two sources of water are available from either the condensate storage tank or the suppression pool.

In the case of the capability of the feedwater pumps, control rod drive feedwater pumps, RCIC, and HPCI not being sufficient to maintain the reactor water level, the automatic depressurization system functions to reduce the reactor pressure so that flow from LPCI and the core spray system enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The automatic depressurization system uses several of the nuclear system pressure relief valves to relieve the high-pressure steam to the suppression pool.

Two independent loops are provided as a part of the core spray system. Each loop consists of a centrifugal water pump driven by an electric motor, a spray sparger in the reactor vessel above the core, piping and valves to convey water from the suppression pool to the sparger, and the associated controls and instrumentation.

In the case of low water level in the reactor vessel or high pressure in the drywell, the core spray system automatically sprays water onto the top of the fuel assemblies in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature. The LPCI system starts from the same signals that initiate the core spray and operates independently to achieve the same objective by flooding the reactor vessel.

In the case of low water level in the reactor or high pressure in the containment drywell, the LPCI mode of operation of the RHR system pumps water into the reactor vessel in time to flood the core and prevent excessive fuel temperature. LPCI operation provides protection to the core for the case of a large break in the nuclear system when the feedwater pumps and the HPCI system are unable to maintain reactor vessel water level. Protection provided by the LPCI system also extends to a small break where the automatic depressurization system has operated to lower the reactor vessel pressure so LPCI and the core spray system start to provide core cooling.

Results of the performance of the emergency core cooling systems for the entire spectrum of liquid-line breaks are contained in Chapter 15. Peak cladding temperatures are well below the 2200°F limit.

Chapter 15 provides the necessary analysis to show that the emergency core cooling systems conform to the 10 CFR Appendix K requirements. This analysis shows compliance with the Appendix K requirements with the following results:

1. Peak clad temperatures are well below the 2200°F acceptability limit.

2. The amount of fuel cladding reacting with steam is below the 1% acceptability limit.

3. The clad temperature transient is terminated while core geometry is still amenableto cooling.

4. The core temperature is reduced and the decay heat can be removed for an extended period of time.

The redundancy and capability of the offsite and onsite electric power systems for the emergency core cooling system are represented in the evaluation against Criterion 34.

The emergency core cooling systems provided are adequate to prevent fuel and clad damage that could interfere with effective core cooling and do limit clad metal-water reaction to a negligible amount. Redundant offsite and onsite electric power systems are provided. The design of the emergency core cooling systems, including their power supply, meets the requirements of Criterion 35.

For further discussion, see the following sections:

1.	Residual Heat Removal System	5.4.7
2.	Emergency Core Cooling System	6.3
3.	Emergency Core Cooling Systems Control and Instrumentation	7.3.1
4.	Auxiliary Power System	8.3.1
5.	Standby AC Power Supply and Distribution	8.3.1
6.	Residual Heat Removal Service Water and Emergency Service Water Systems	9.2.3
7.	Accident Analyses	Chapter 15

3.1.2.4.7 Criterion 36 - Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to ensure the integrity and capability of the system.

EVALUATION AGAINST CRITERION 36

The emergency core cooling systems include the RHR, core spray, and HPCI systems connected to the reactor coolant system. The engineering and design efforts for these systems include in-service inspection considerations. The spray rings within the vessel are accessible for inspection during each refueling outage. Removable plugs in the sacrificial shield and/or panels in the insulation provide access for the examination of nozzles for the vessel outside diameter. Removable insulation is provided on the emergency core cooling systems piping out to and including the first isolation valve outside containment. Inspection of the emergency core cooling systems is in accordance with the intent of Section XI of the ASME Code. Section 6.6 defines the inservice inspection plan, access provisions, and areas of restricted access. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time. Components inside the primary containment can be inspected when the drywell is open for access. When the reactor vessel is open, for refueling or other purposes, the spargers and other internals can be inspected. Portions of the emergency core cooling system that are part of the reactor coolant pressure boundary are designed to specifications for inservice inspection to detect defects that might affect the cooling performance. Particular attention will be given to the reactor nozzles, core spray, and feedwater spargers. The design of the reactor vessel and internals for inservice inspection and the plant testing and inspection program ensure that the requirements of Criterion 36 will be met.

For further discussion, see the following sections:

1.	Reactor Pressure Vessel Internals	3.9.5
2.	Reactor Vessel	5.3
3.	Emergency Core Cooling System	6.3
4.	Inservice Inspection and Testing	6.6

3.1.2.4.8 Criterion 37 - Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

EVALUATION AGAINST CRITERION 37

The emergency core cooling system consists of the HPCI, automatic depressurization system, LPCI mode of the RHR system, and the core spray system. Each of these systems is provided with sufficient test connections and isolation valves to permit appropriate periodic pressure testing to ensure the structural and leaktight integrity of its components.

The HPCI, LPCI, and core spray systems, as discussed in Section 6.3.4, and the automatic depressurization system, as discussed in Section 5.2.2, are designed to permit periodic testing to ensure the operability and performance of the active components of each system.

The pumps and valves of these systems will be tested periodically to verify operability. Flow rate tests will be conducted on the core spray, LPCI, and HPCI systems.

The emergency core cooling system will be subjected to tests to verify the performance of the full operational sequence that brings each system into operation.

The operation of applicable portions of the protection system is discussed in Section 7.3.2, the transfer between normal and emergency power sources is discussed in Section 8.3.1, and the operation of the associated cooling water systems is discussed in the response to Criterion 46. Section 5.4.7 and the Technical Specifications contain a more detailed discussion of the tests to which these systems will be subjected.

3.1.2.4.9 Criterion 38 - Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION AGAINST CRITERION 38

In the event of a LOCA within the reactor containment, the pressure suppression system will rapidly condense the steam to prevent containment overpressure. The containment feature of pressure suppression employs two separate compartmented sections of the primary containment: the drywell that houses the nuclear system and the suppression chamber containing a large volume of water. Any increase in pressure in the drywell from leakage in the nuclear system is relieved below the surface of the suppression chamber water pool by connecting vent lines, thereby condensing steam being released to the drywell. The pressure buildup in the suppression chamber is equalized with the drywell by a vent line and vacuum breaker arrangement. Cooling systems remove heat from the reactor core, the drywell, and from the water in the suppression chamber during accident condition, and thus provide continuous cooling of the primary containment.

The emergency core cooling system is actuated to provide core cooling in the event of a LOCA. Low water level in the reactor vessel or high pressure in the drywell will initiate the emergency core cooling system to prevent excessive fuel temperature. Sufficient water is provided in the suppression pool to accommodate the initial energy that can transiently be released into the drywell from the postulated pipe failure.

The suppression chamber is sized to contain this water plus the water displaced from the reactor primary system together with the free air initially contained in the drywell.

Either or both RHR heat exchangers can be manually activated to remove energy from the containment. The redundancy and capability of the offsite and onsite electric power systems for the RHR system are presented in the evaluation against Criterion 34.

The pressure suppression system is capable of rapid containment pressure and temperature reduction following a LOCA to ensure that the design limits are not exceeded. Redundant offsite and onsite electric power systems are provided. The design of the containment heat removal system meets the requirements of Criterion 38.

For further discussion, see the following sections:

1.	Residual Heat Removal System	5.4.7
2.	Containment System	6.2
3.	Emergency Core Cooling System	6.3
4.	Emergency Core Cooling System Control and Instrumentation	7.3.1
5.	Auxiliary Power System	8.3.1
6.	Standby AC Power Supply and Distribution	8.3.1
7.	Residual Heat Removal Service Water and Emergency Service Water Systems	9.2.3
8.	Accident Analyses	Chapter 15

3.1.2.4.10 Criterion 39 - Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to ensure the integrity and capability of the system.

EVALUATION AGAINST CRITERION 39

Provisions are made to facilitate periodic inspections of active components and other important equipment of the containment pressure reducing systems. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time and will be inspected periodically. Components inside the primary containment can be inspected when the drywell is open for access. The testing frequencies of most components will be correlated with the component inspection.

The pressure suppression chamber is designed to permit appropriate periodic inspection. Space is provided outside the chamber for inspection and maintenance. There are two hatches that permit access to the suppression chamber for inspection.

The containment heat removal system is designed to permit periodic inspection of major components both outside and within the primary containment. This design meets the requirements of Criterion 39.

For further discussion, see the following sections:

1.	Residual Heat Removal System	5.4.7
2.	Containment System	6.2
3.	Emergency Core Cooling System	6.3
4.	Emergency Core Cooling System Controls and Instrumentation	7.3.1
5.	Reactor Core Residual Heat Removal and Emergency Equipment Service Water System	9.2.3

3.1.2.4.11 Criterion 40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

EVALUATION AGAINST CRITERION 40

The containment heat removal function is accomplished by the containment cooling mode of the RHR system. This mode is discussed in Section 5.4.7 and consists of the suppression pool cooling subsystem and containment spray subsystem.

The RHR system is provided with enough test connections and isolation valves to permit periodic pressure testing. The containment spray mode is subjected to a periodic air test.

The pumps and valves of the RHR system will be operated periodically to verify operability. The containment spray mode is not fully testable but the operation of the initiation signal and components is verified. The suppression pool cooling mode is not automatically initiated, but the operation of the components is periodically verified.

The operation of applicable portions of the protection system (for containment spray) is discussed in Section 7.3.2., the transfer between normal and emergency power sources is discussed in Section 8.3.1., and the operation of associated cooling water systems is discussed in the response to Criterion 46. Section 5.4.7 and the Technical Specifications contain a more detailed discussion of the tests to which the RHR system will be subjected.

3.1.2.4.12 Criterion 41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to ensure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

EVALUATION AGAINST CRITERION 41

Fission products released into the reactor building following postulated accidents will be automatically processed by the standby gas treatment system (SGTS). The SGTS initiation signal follows the release of radioactivity on the refueling floor or indication of a process-line rupture inside the drywell. The ability of this system to remove radioactivity from the process stream is discussed in Section 6.5.

The SGTS is composed of two trains that are separated physically and electrically so that a single failure will not prevent its function. The redundancy of this system is discussed in Section 6.5.

The SGTS units are connected by a flow orifice downstream of the deep bed filters. This maintains a small continuous flow through the inactive train to ensure cooling for a deep bed potentially loaded with radionuclides. Each train of the SGTS is powered from redundant portions of the emergency ac power system. The trains discharge to a common duct leading to the offgas stack; there are no valves in the common discharge. The suction to the trains is common also; the suction valves that may have to operate after an accident are powered from the emergency ac distribution system and are designed to fail so that containment isolation and reactor isolation and reactor building evacuation via the SGTS are ensured.

Section 6.5 discusses SGTS operation, Table 7.3-1 indicates containment isolation, and Section 8.2 discusses the availability of power.

The hydrogen and oxygen concentration in the drywell is monitored after a severe accident and nitrogen is added to the drywell for dilution as required to maintain a nonflammable mixture inside the drywell.

3.1.2.4.13 Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to ensure the integrity and capability of the systems.

EVALUATION AGAINST CRITERION 42

The inspection of the internal structure of the SGTS members is facilitated by access doors installed in each unit to allow entry to the unit for the visual inspection of structural members and filter faces.

An inspection port is provided for the visual inspection of all filter faces.

Glove ports are provided on all HEPA filters to facilitate scanning with a radiation probe of each HEPA filter bank. Each compartment of both SGTS units is equipped with a gastight lighting fixture. This light is required for the visual inspection of the unit through sight ports.

Each charcoal bed is provided with facilities for taking a sample of charcoal from any section of the bed.

For further discussion of SGTS features, refer to Section 6.5.

3.1.2.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of the components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

EVALUATION AGAINST CRITERION 43

Each unit of the SGTS will be operated periodically to ascertain the operability and performance of the major active components, such as fans, filters, motors, and valves, and the structural integrity of the unit. This test will also verify the operability of the system as a whole and the operability of all associated subsystems. The test is run at design flow, pressure, and temperature. See Section 8.3 for a discussion of the testing of the auxiliary power system.

The leaktightness of the HEPA filters is measured by the DOP (di-octylphalate) test. The deep-bed charcoal filters are checked for bypass with refrigerant 112.

Section 6.5 and the Technical Specifications discuss the testing of the SGTS.

3.1.2.4.15 Criterion 44 - Cooling Water

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operation and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION AGAINST CRITERION 44

The RHR service water system and the emergency service water system perform the function of transferring the heat load from structures, systems, and components important to safety during a plant accident condition as well as during normal shutdown and cooldown. Cooling water is furnished by the river water supply system.

The RHR service water system, emergency service water system, and river water supply systems are each composed of two separate subsystems. The redundancy of these subsystems, components, and features is described in Section 9.2.

No interconnections exist between the two systems nor between the two independent and redundant trains of each system. The interconnection between the RHR service water system and the RHR system is described in Section 9.2.3. The interconnection between the emergency service water and the well water system is described in Section 9.2.3.

To prevent the leakage of water from the RHR system to the RHR service water system, the RHR service water to the heat exchanger is kept at a higher pressure than the RHR system. As additional protection, a process radiation monitor has been placed in the RHR and emergency service water discharge. These systems are discussed in Section 9.2.3.

The power supplies for the isolation valving are such that, assuming a single failure, one subsystem for either the emergency service water, RHR service water, or the river water supply will be available to accomplish the system safety functions. Sufficient redundancy exists in the electric power supply system to ensure a source of power from either onsite or offsite systems. This is discussed in Section 9.2.3.

3.1.2.4.16 Criterion 45 - Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the system.

EVALUATION AGAINST CRITERION 45

To the extent practical and consistent with other design considerations, the components of the RHR service water, emergency service water, and river water supply systems have been located to facilitate visual inspection. Isolation valves and test connections are such as to permit the verification of the integrity of the buried piping. Section 9.2.3 discusses the inspection and testing of these systems.

3.1.2.4.17 Criterion 46 - Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

EVALUATION AGAINST CRITERION 46

The RHR service water, emergency service water, and river water supply systems have been provided with enough test connections and isolation valves to pressure test the integrity of the components within each system.

The pumps and automatic valves will be tested periodically to verify operation. Since the river water supply system is normally in operation, no special tests are required to ensure that the system can operate in an emergency. Periodic tests will be to verify the automatic initiation of the river water supply system and emergency service water system; the RHR service water system has no automatic initiation features, but operation will be verified. The specific tests that are to be conducted are discussed more fully in the Technical Specifications. Chapter 8 discusses the tests that are conducted to ensure the availability of electric power. The pumps and valves of these systems that must operate in an emergency are powered from a standby ac distribution system.

3.1.2.5 Group V, Reactor Containment

3.1.2.5.1 Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

EVALUATION AGAINST CRITERION 50

The containment structure, access opening, heat removal system, and internal compartments are designed to accommodate the pressure and temperature conditions resulting from any LOCA without exceeding the leakage rate incorporated in the Technical Specifications.

As discussed in Section 6.2, the calculated peak drywell pressure of 45.7 psig affords sufficient margin to the maximum internal pressure (62 psig) allowed by the ASME B&PV Code, Section III, Nuclear Vessels.

Figure 15.2-5, “Containment Capability Curve,” indicates the containment vessel has the capability of tolerating arbitrarily large metal-water reactions, thus ensuring sufficient margin in the design to accommodate this postulated phenomenon.

A large body of experimental data has been obtained on BWR suppression containment performance. Furthermore, very conservative assumptions have been used in the containment response analytical model as described in Chapter 15.2 and in General Electric Topical Reports NEDO-10320 and NEDO-21888.

3.1.2.5.2 Criterion 51 - Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to ensure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

EVALUATION AGAINST CRITERION 51

The reactor containment vessel is fabricated to the requirements of the ASME B&PV Code, Section III, Subsection B for nuclear vessels. This Code in Article 12 gives due recognition to the requirement that containment materials behave in a ductile manner for all conditions of service, thus ensuring that the containment's ferritic materials behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. The lowest design service temperature is conservatively taken as 30°F. The actual service temperature is assumed to be approximately 135°F. Thus, sufficient margin is inherent in the design to account for the various uncertainties involved in design and fabrication.

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

EVALUATION AGAINST CRITERION 52

The reactor containment vessel and equipment within the vessel are designed to permit periodic integrated leakage rate testing. A more complete discussion can be found in Section 6.2.6 and in the Technical Specifications.

3.1.2.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

EVALUATION AGAINST CRITERION 53

The reactor containment is designed to optimize the accessibility of important areas to permit required inspection and surveillance.

All penetrations with resilient seals or expansion bellows are the double-seal type. The space between the seals may be periodically pressurized to containment design pressure and their leaktightness verified. This is discussed in Section 6.2.6.

3.1.2.5.5 Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

EVALUATION AGAINST CRITERION 54

Piping systems that penetrate the drywell have been accorded special design considerations to reflect their importance in accomplishing safety-related functions and in achieving isolation, if required. The penetrations are discussed in Section 6.2.1. Both the isolation valving and the system that initiates isolation use components whose quality maximize reliability and are provided with sufficient independence and redundancy to optimize the isolation function should it be required. Containment isolation is discussed in Sections 6.2.1 and 6.2.4, and the system that initiates isolation is discussed in Section 7.3.1.

The operation of remote manual isolation valves will be periodically verified according to the Technical Specifications. Enough test connections are provided to each of these piping systems to ensure that minimal valve leakage is achieved and maintained. This is discussed in Section 6.2.6.

The nuclear power station design has been reviewed, compared to the requirements, and determined to be in compliance with this design criterion.

3.1.2.5.6 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as an automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to ensure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

EVALUATION AGAINST CRITERION 55

The reactor coolant pressure boundary (as defined in 10CFR50.2(v)) consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, valves, and pipes that extend from the reactor pressure vessel up to and including the outermost isolation valve. The lines of the reactor coolant pressure boundary that penetrate the primary containment are capable of isolating the containment, thereby precluding any significant release of radioactivity. Similarly, for lines that do not penetrate the primary containment but that form a portion of the reactor coolant pressure boundary (such as connecting lines up to and including the second isolation valve) , the design ensures that isolation from the reactor coolant pressure boundary can be achieved.

Influent Lines

Influent lines that penetrate the primary containment and connect directly to the reactor pressure vessel are equipped with two isolation valves: one inside the containment, the other outside and located as close to the containment as possible.

Table 3.1-1 lists those influent pipes that comprise the reactor coolant pressure boundary. The purpose of this table is to review the design of each line with respect to the requirements imposed by Criterion 55. The paragraphs referenced in Table 3.1-1 demonstrate that although a word-for-word comparison with Criterion 55 in some cases is not practicable, it is possible to demonstrate adequate isolation provisions on some other defined basis.

Those process lines normally supplying water to the reactor vessel, where the flow is normally into the reactor vessel, are supplied with two check valves in series for isolation.

A test connection is located between these check valves. With the reactor pressurized, the leakage past the inboard check valve can be detected at the leak test connection. With the gate valve downstream of the inboard check valve closed, the space between the two isolation check valves is pressurized and any measured leakage is assumed to be through the outboard check valve. These valves are functionally checked as they are leak tested.

The primary containment isolation valves and check valves are all purchased to the seat leakage requirements spelled out by MSS-SP-61 with 2 cm^3/hr-in. of seat diameter, the maximum allowable.

<u>Effluent Lines</u>

Effluent lines that form part of the reactor coolant pressure boundary and penetrate the primary containment are equipped with two isolation valves: one inside the containment, the other out-side and located as close to the containment as possible. Exceptions are the isolation valves for the reactor water sample lines which connect to jet pump flow-sensing instrument lines; both isolation valves on each sample line are located outside of primary containment.

Table 3.1-2 lists those effluent pipes that comprise the reactor coolant pressure boundary and that penetrate the primary containment.

Aside from the main steam isolation valves, each valve is a motor-operated, automatic and remote manually actuated gate valve capable of providing adequate isolation protection in the event of a break in these lines. The main steam isolation valves are nitrogen-operated, automatic and remote manually actuated globe valves that provide two distinct barriers against containment leakage. On loss of actuating power, all air-operated automatic isolation valves assume the position that provides greater safety. The protection system will initiate automatic isolation under accident conditions for effluent lines that are normally open during operation and are not part of the overall safety system network.

The reactor water sample lines for the postaccident sampling system, which connect to jet pump flow-sensing instrument lines outside of the drywell, have two automatic isolation valves in each line in series located outside the drywell. This deviation from the design criteria is considered safe and adequate for the following reasons:

1. The sample lines are connected to instrument lines which are provided with orifices located inside the drywell . The orifices would minimize the reactor coolant leakage as discussed in Section 6.2.4.2.4 in the event of a postulated failure of piping or any component in the instrument lines or the sample lines outside primary containment.

2. The valves were purchased and installed to the quality standards required for components that are connected to the reactor coolant pressure boundary.

3. The isolation valves have been located as close to the containment as practical.

4. The valves are normally closed because, except for testing, the sample lines are not used during normal plant operation.

<u>Summary</u>

In order to ensure protection against the consequences of accidents involving the release of radioactive material, pipes that form the reactor coolant pressure boundary have been shown to provide adequate isolation capabilities on a case-by-case basis. Adequate isolation capabilities were also demonstrated for pipes that connect to the reactor coolant pressure boundary outside the primary containment. In all cases, a minimum of two barriers were shown to protect against the release of radioactive materials.

In addition to meeting the isolation requirements stated in Criterion 55, the pressure retaining components that comprise the reactor coolant pressure boundary are designed to meet other appropriate requirements that minimize the probability or consequences of an accidental rupture. The quality requirements for these components ensure that they are designed, fabricated, and tested to the highest quality standards of all reactor plant components.

It can therefore be concluded that the design of piping systems that comprise the reactor coolant pressure boundary satisfies Criterion 55.

For further discussion, see the following sections:

1.	Containment Isolation System	6.2.4
2.	Primary Containment Isolation and Nuclear Steam Supply Shutoff System	7.3.1

3.1.2.5.7 Criterion 56 - Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

EVALUATION AGAINST CRITERION 56

Lines that penetrate the primary containment and communicate with the containment interior may be grouped into two categories: (1) pipes that communicate with the drywell or suppression chamber atmosphere and (2) pipes that communicate with the suppression pool.

Lines Which Penetrate Primary Containment and Communicate with the Drywell or Suppression Chamber Atmosphere

Lines that penetrate the primary containment and communicate with the drywell or suppression chamber atmosphere are generally provided with two automatic isolation valves in series located outside the drywell. This deviation from the design criteria is considered safe and adequate for the following reasons:

1. There is limited space within the drywell, and placing these valves inside would seriously impede accessibility for inspection and maintenance.

2. Placing these valves inside the drywell would subject them to an inimical environment and, thus, increase the probability of failure.

3. These valves are purchased and installed to the same quality standards as valves that are connected to the reactor coolant pressure boundary, yet are subjected to a pressure that is approximately 70 times less during normal operation and 15 times less during a design-basis accident.

4. The lines that fall into this category are generally not in use during normal operation and are therefore isolated. Exceptions include the nitrogen-inerting makeup line, the drywell and torus atmosphere monitoring lines, drywell equipment and floor drain discharge lines, containment compressor lines, and HPCI/RCIC exhaust vacuum breaker line.

All automatic valves are capable of remote manual operation and are closed on receipt of an isolation signal as shown in Table 7.3-1, with the following exceptions:

There are two lines that do not have automatic isolation valves: the service air line and the demineralized water line. Each of these penetrations is isolated by a locked-closed manual valve or blank flange located inside the drywell and a locked-closed valve located outside the primary containment.

2016-003 The containment hardened vent line contains two containment isolation valves, CV-4360 and CV-4361, which do not receive an automatic isolation signal. These valves fail closed on loss of pneumatic pressure or loss of power. The valve operators are normally vented and isolated from pneumatic pressure by a locked closed valve. Administrative controls requiring the use of a keylock switch and opening the locked valve will preclude inadvertent opening of CV-4360 and CV-4361. These valves are considered to be “sealed closed” per NRC Standard Review Plan 6.2.4, paragraph II.6.f since they are under administrative control such that they cannot be inadvertently opened. Redundancy is provided such that operators would need to obtain a key for HS-4360, and open locked closed isolation valve V43-0642 prior to opening the primary containment isolation valves. The DAEC Emergency Operating Procedures will control the use of the Hardened Containment Vent isolation valves.

During the interim “abandoned in place” configuration, the Containment Atmosphere Dilution (CAD) system isolation is comprised of a locked-closed manual valve (inboard) and a normally closed, remote manual valve (outboard).

Pipes That Penetrate the Primary Containment and Connect Directly to the Suppression Pool

The rationale for not placing valves inside the suppression pool is similar to the four reasons mentioned in the preceding section. The following discussion provides unique considerations as to the types of valves and isolation capabilities.

Influent Lines to Suppression Pool

1. RCIC and HPCI Turbine Exhaust Lines, HPCI Turbine Condensate Line, RCIC Vacuum Pump Discharge Line
These lines, which penetrate the primary containment and connect to the suppression pool, are equipped with a normally open, stopcheck globe valve located as close to the containment as possible. In addition, there is a simple check valve upstream of each globe valve that provides positive actuation for immediate isolation in the event of a break upstream of this valve. It should be noted that these lines have a leaktight water seal (i.e., any gas that might be present in the air space of the suppression chamber cannot leak through these lines), adding yet another level of protection.

2. Minimum Flow and Test Lines

   These lines have isolation capabilities that are commensurate with the importance to safety of isolating these lines. The RHR, HPCI, RCIC, and core spray minimum flow lines have two valves in series, both of which are located outside the primary containment. The upstream valve is a check valve and the downstream valve is a motor-operated gate valve. The motor-operated valve serves as a flow control valve to ensure proper minimum flow through the pump. Since the operation of the bypass lines is important to the operation of the safety systems, and since automatic isolation valves could degrade the reliability of these systems, no further valving has been incorporated.

The RHR and core spray test lines have a normally closed isolation valve in addition to the water seal to meet the isolation requirements.

Effluent Lines from Suppression Chamber (RHR, Core Spray, HPCI, and RCIC

Criterion 56 requires that these lines have two isolation valves: one inside the containment, the other outside. It should be noted that this criterion does not reflect the consideration of the BWR suppression pool design. For instance, these lines do not have an isolation valve located inside the containment as this would necessitate the placement of the valve underwater. In effect, this would result in introducing a potentially unreliable valve in a highly reliable system thereby compromising design. For this reason these lines incorporate two valves outside of the containment, the first of which is located as close to the containment as possible. These valves are motor-operated, remote manually actuated gate valves. Because of the importance of the suction lines in combating an accident, none of these valves receive an automatic isolation signal.

Summary

In order to ensure protection against the consequences of accidents involving the release of significant amounts of radioactive materials, pipes that penetrate the primary containment have been demonstrated to provide isolation capabilities in accordance with Criterion 56. In all cases, these pipes have provided a minimum of two protective barriers against containment leakage and in some cases more.

In addition to meeting the isolation requirements stated in Criterion 56, the pressure retaining components of these systems are designed to the same quality standards as the containment. In some respects, providing a high-quality system obviates the need for isolation because of the diminished probability of a rupture in these lines.

It can be concluded that the design of piping systems that penetrate the primary containment and connect to the containment interior satisfies Criterion 56.

3.1.2.5.8 Criterion 57 Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

EVALUATION AGAINST CRITERION 57

The DAEC design has been reviewed compared to the requirements and determined to be in compliance with the design criterion. This subject is further discussed in Sections 6.2.4 and 3.6.

3.1.2.6 Group VI, Fuel and Radioactivity Control

3.1.2.6.1 Criterion 60 - Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

EVALUATION AGAINST CRITERION 60

The requirements of this criterion have been met in the design of the DAEC. The description of the radioactive waste treatment system in Chapter 11 details the measures taken to ensure that levels of radioactive materials in effluents released from the plant are kept as low as reasonably achievable in accordance with 10 CFR 20.

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

EVALUATION AGAINST CRITERION 61

1. New-Fuel Storage

New fuel is placed in dry storage in the new-fuel storage vault located inside the secondary containment reactor building. The storage vault within the reactor building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see "Evaluation Against Criterion 62" and Section 9.1.1). The new-fuel storage racks do not require any special inspection and testing for nuclear safety purposes.

2. Spent-Fuel Handling and Storage

The handling of new- and spent-fuel assemblies for reactor refueling is conducted within the reactor building, which is designed to normally provide secondary containment. Fuel storage pool water is allowed to flood the reactor well in order to provide shielding above the reactor and spent fuel. The water height during this operation is approximately 25 ft. This height ensures adequate shielding when spent-fuel assemblies are transferred from the reactor to the spent-fuel pool during the refueling operation. Fuel pool water is circulated through the fuel pool cooling and cleanup system (FPCC) to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental criticality (see "Evaluation Against Criterion 62" and Section 9.1.2).

Reliable decay heat removal is provided by the closed loop FPCC system. It consists of two circulating pumps, two heat exchangers, two filter-demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. The pool water is circulated through the system, suction is taken from the surge tanks, and the flow passes through the heat exchanger and filters and is discharged through diffusers near the bottom of the fuel pool and reactor well. Expected pool water temperatures are discussed in Section 9.1.2.3.2.. If it appears that the fuel pool temperature will exceed 150°F, the FPCC system can be connected to the RHR system. This increases the cooling capacity of the FPCC system.

There are no connections to the fuel storage pool that could allow the fuel pool to be drained below the pool gate between the reactor well and fuel pool. High- and low-level switches indicate pool water level changes in the control room and pump room. Fission product concentration in the pool water is minimized by the use of the filter-demineralizer. This minimizes the release from the pool to the reactor building environment.

No special tests are required because at least one pump, heat exchanger, and filter-demineralizer are continuously in operation while fuel is stored in the pool. Duplicate units are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify the system operability.

Appropriate containment and confinement are provided by the reactor building and the design of the spent-fuel pool. The secondary containment isolation system is described in Section 6.2.3, and the fuel pool design, including a discussion of the separate cask pool, is found in Section 9.1.4.4.5. The response to Safety Guide 13 (Section 1.8) contains additional information regarding the spent-fuel storage facility.

3. Radioactive Waste Systems

The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal, radioactive liquids, gases, and solid waste produced as a result of reactor operation.

Liquid radwastes are classified, collected, and treated as high or low conductivity, chemical, detergent, sludges, or concentrated wastes. Processing includes filtration, ion exchange/or evaporation, analysis, and dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. The solid radwaste system is a continuous part of the liquid radwaste system. Wet solid wastes are packed in an 85-ft^3 steel liner or in 170-ft^3 high integrity containers (HICs), which are contractor furnished and may change in size. Dry solid radwastes are packaged in steel drums or boxes. Required shielding is provided before shipment (see Section 11.4). Gaseous radwastes are monitored, processed, recorded, and controlled so that levels of radioactive material in gaseous effluents are maintained as low as reasonably achievable as discussed in Section 11.3.

Accessible portions of the reactor building, the radwaste building and the low-level radwaste processing and storage facility have sufficient shielding to ensure personnel exposures are within 10 CFR 20 limits. Radioactive waste treatment systems are appropriately confined within plant structures such that inadvertent releases result in radiation levels within annual exposure limits of 10 CFR 20.

The radwaste systems are used on a routine basis and do not require specific testing to ensure operability. Performance is monitored by radiation monitors during operation.

The fuel storage and handling and radioactive waste systems are designed to ensure adequate safety under normal and postulated accident conditions. The design of these systems meets the requirement of Criterion 61.

For further discussion, see the following sections:

1.	Residual Heat Removal System	5.4.7
2.	Containment System	6.2
3.	Radioactive Waste Management	Chapter 11
4.	New-Fuel Storage	9.1.1
5.	Spent-Fuel Storage	9.1.2
6.	Spent-Fuel Pool Cooling and Cleanup System	9.1.3
7.	Air-Conditioning, Heating, Cooling, and Ventilation Systems	9.4
8.	Design of Seismic Category I Structures	3.8

3.1.2.6.3 Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

EVALUATION AGAINST CRITERION 62

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new-fuel and spent-fuel storage is prevented by the fuel storage rack design. There is sufficient spacing between the assemblies to ensure that the array when fully loaded is substantially subcritical. Fuel elements are limited by rack design to only top loading and fuel assembly positions. The new- and spent-fuel racks are Seismic Category I structures.

2012-004 New fuel may be placed in dry storage in the top-loaded new-fuel storage vault. This vault contains a drain to prevent the accumulation of water. The new-fuel storage vault racks (located inside the secondary containment reactor building) are designed to prevent an accidental critical array, even in the event the vault becomes flooded or subjected to seismic loadings. The 6.625-in. center-to-center new-fuel assembly spacing limits the effective multiplication factor of the array to not more than 0.90 for new dry fuel. The K_{eff} will not exceed 0.95 under the abnormal condition if the new fuel is flooded.

2012-004 New fuel and spent fuel are stored under water in the spent-fuel pool. The racks in which these fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Spent fuel is maintained at a subcritical multiplication factor K_{eff} of less than 0.90 under normal condition and 0.95 for abnormal conditions. Abnormal conditions may result from an earthquake, accidental dropping of equipment, or damage caused by the horizontal movement of fuel-handling equipment without first disengaging the fuel from the hoisting equipment.

Refueling interlocks include circuitry that senses conditions of the refueling equipment and the control rods. These interlocks reinforce operational procedures that prohibit making the reactor critical. The fuel handling system is designed to provide a safe, effective means of transporting and handling fuel. The system is designed to minimize the possibility of mishandling or maloperation. The use of geometrically safe configurations for new-fuel and spent-fuel storage and the design of fuel handling systems preclude accidental criticality in accordance with Criterion 62.

For further discussion, see the following sections:

1.	Refueling Interlocks	7.6.2
2.	New-Fuel Storage	9.1.1
3.	Spent-Fuel Storage	9.1.2

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

EVALUATION AGAINST CRITERION 63

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the FPCC system that could result in a loss of RHR capability and excessive radiation levels is alarmed in the control room. Alarmed conditions include high and low fuel pool cooling water pump discharge pressures, high and low levels in the fuel storage pool and skimmer surge tanks, and flow in the drain lines between fuel pool gates between fuel pool and
2014-010 reactor well. System temperature is also continuously monitored in the control room. The reactor building ventilation radiation monitoring system detects abnormal amounts of radioactivity and initiates appropriate action to control the release of radioactive material to the environs. These systems are discussed in Sections 9.1.3 and 6.2.3 and also in the response to Safety Guide 13 in Section 1.8.

Area radiation and tank and sump levels are monitored and alarmed to give indication of conditions that may result in excessive radiation levels in radioactive waste system areas. These systems are discussed in Sections 11.2, 11.3, 11.4, and 12.3.4.

3.1.2.6.5 Criterion 64 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

EVALUATION AGAINST CRITERION 64

Provisions for monitoring radioactivity releases have been made in accordance with the intent of this criterion.

Means have been provided for monitoring the containment atmosphere as described in Section 11.5.5 and spaces containing components for the recirculation of LOCA fluids (Section 3.11). The site environs are monitored for radioactivity releases as described in Section 11.5.6. Effluent discharge paths are monitored as discussed in Chapter 11.

Table 3.1-1

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REACTOR COOLANT PRESSURE BOUNDARY

Influent Lines	Inside Drywell	Outside Drywell
Feedwater[a]	CV	MOCV
a. HPCI return[b]	--	CV-MOV
b. RCIC return[b]	--	CV-MOV
c. Cleanup return[b]	--	CV-MOV
RHR return to recirc.[c]	CV	MOV
Core spray[c]	CV	MOV
CRD return[d]	CV	CV
Standby liquid control[e]	CV	CV
Other[f]	--	--
a. MO1908 bypass	CV	--

Key: CV = check valve
MOCV = motor-operated check valve
MOV = motor-operated valve

[a]That portion of the feedwater line that forms part of the reactor coolant pressure boundary and penetrates the primary containment has two isolation valves. The isolation valve inside the containment is a simple check valve. The isolation valve outside the containment is a stopcheck globe valve. Should a break occur in the feedwater line, the check valves prevent a significant loss of inventory and offer immediate isolation. During the postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, the isolation valve outside of containment does not automatically isolate on signal from the protection system. However, the valve is capable of being remotely closed from the control room to provide long-term leakage protection upon operator judgment that continued makeup from the feedwater source is unnecessary.

[b]Influent lines that form a portion of the reactor coolant pressure boundary but do not penetrate the primary containment must adequately reflect the importance to safety of isolating these piping systems. Pipes of this type include those portions of the RCIC, the reactor water cleanup (RWCU), and the HPCI lines that tie into the feedwater lines. Each of these lines has two isolation valves in series. The first of these is a check valve, and the other is a motor-operated, automatic and remote manually actuated valve. The RCIC and HPCI lines are closed during normal operation, whereas, the RWCU line is open during operation and isolates from the reactor coolant pressure boundary upon receipt of signals from the protection system.

Notes - Table 3.1-1

[c]The RHR return lines to recirculation system and the core spray lines have check valves inside the containment that provide for immediate isolation in the event of a break upstream of these valves. In addition, the isolation valves outside the containment are normally closed, automatic and remote manually actuated valves designed to provide long-term leakage control in the event of a break in these lines. For the postulated LOCA, the protection system will initiate automatic opening of the injection valves at the appropriate time to ensure that acceptable fuel design limits are not exceeded.

[d]That portion of the CRD return line that forms part of the reactor coolant pressure boundary and penetrates the primary containment has two isolation valves. Both valves are simple check valves and are located inside as well as outside the primary containment.

Criterion 55 states that a simple check valve may not be used as the automatic isolation valve outside the containment. During the postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, valves that automatically isolate upon signal from the protection system are not included in the design of this system. Should a break occur in the CRD return line, the check valves would prevent significant loss of inventory and offer immediate isolation.

[e]The standby liquid control line uses a simple check valve as the isolation valve inside as well as outside the primary containment. Criterion 55 states that a simple check valve may not be used as the automatic isolation valve outside the containment; however, should insertion of the liquid poison become necessary, it is imperative that the injection line be open. In the design of this system, it has been the accepted practice to omit an automatic valve that opens on signal as this introduces a possible failure mechanism. As a means of providing assurance for reliable timely actuation, an explosive valve is used. In this manner, the availability of the line is ensured. Because the standby liquid control line is a normally closed, nonflowing line, rupture of this line is very remote; however, should a break occur, the check valves provide positive actuation for immediate isolation.

[f]Other

a. The MO1908 Bypass Lines is a 1/2″ line bypassing MO1908 and completely contained within Primary Containment. This line has a check valve which allows flow only in the influent direction and is intended to relieve pressure between MO1908 and MO1909 during a LOCA. The pressure is relieved back to the Reactor Vessel. MO1908 is located on the RHR Shutdown Cooling Suction Line which is an effluent line.

Notes - Table 3.1-1

CRD Insert and Withdraw Lines

Criterion 55 concerns the reactor coolant pressure boundary penetrating the primary reactor containment. As shown in Table 3.2-4, the CRD insert and withdraw lines are not part of the reactor coolant pressure boundary.

The basis to which the CRD lines are designed is commensurate with the safety importance of isolating these lines. Since these lines are vital to the scram function, their operability is of utmost concern.

In the design of this system, it has been accepted practice to omit automatic valves for isolation purposes as this introduces a possible failure mechanism. As a means of providing positive actuation, manual shutoff valves are used. In the event of a break on these lines, the manual valves may be closed to ensure isolation. In addition, a ball valve located in the insert line is designed to automatically seal this line in the event of a break.

Finally, several breaks and combinations of breaks in the CRD lines have been postulated and analyzed (see Section 4.6.2). The results of these analyses indicate that the worst situation causes a leak rate that is negligible compared to the makeup capability.

TIP System

Since the TIP system lines do not comprise a portion of the reactor coolant pressure boundary, GDC 55 is not directly applicable to this specific class of lines. The basis to which these lines are designed is more closely described by GDC 54, which states in effect, that the isolation capability of a system be commensurate with the safety importance of that isolation. However, since the TIP lines communicate directly with the primary containment via the relief valves on the TIP indexer, conformance with GDC 56 must be addressed. GDC 56 can be satisfied by classifying these lines as nonessential instrument lines, subject to the acceptance criteria of Regulatory Guide 1.11. Such conformance has been demonstrated in NEDC-22253, BWR Owners Group Evaluation of Containment Isolation Concerns, October 1982. These and other safety features are described in the following paragraphs.

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of four valves may be opened at any one time to conduct the calibration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, a signal causes the cable to be retracted and the ball valve to close automatically after completion of cable withdrawal. To ensure isolation capability if a TIP cable fails to withdraw or a ball valve fails to close, an explosive, shear valve is installed in each line. Upon receipt of a signal, this explosive valve will shear the TIP cable and seal the guide tube.

Table 3.1-2

REACTOR COOLANT PRESSURE BOUNDARY

Effluent Lines	Inside Drywell	Outside Drywell
Main steam	NOV	NOV
Reactor water cleanup	MOV	MOV
RHR shutdown cooling	MOV	MOV
Main steam drain	MOV	MOV
RCIC turbine steam	MOV	MOV
HPCI turbine steam	MOV	MOV

Key: NOV = nitrogen-operated valve
MOV = motor-operated valve

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

3.2.0 Structures, Systems, and Components Important to Safety

Certain structures, systems, and components of the nuclear plant are considered important to safety because they perform safety actions required to avoid or mitigate the consequences of abnormal operational transients or accidents. The ways in which structures, systems, and components important to safety work together to avoid the unacceptable results associated with the consequences of various extreme plant events is explained in the IE Nuclear Safety Operational Analysis (NSOA). The purpose of this section is to classify structures, systems, and components according to the importance of the safety function they perform. In addition, design requirements are placed upon such equipment to assure the proper performance of safety actions, when required.

In order to establish the loadings and loading combinations for which each individual structure and system is to be designed, buildings and their contained systems are separated into the seismic or nonseismic categories with respect to seismic design requirements.

3.2.1 Seismic Classifications

3.2.1.1 Seismic Structures, Systems, and Components

Those structures, systems, and components important to safety that are designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional are designated as Seismic Category 1. Tables 3.2-1 and 3.2-3 provide lists of Seismic Classification of Structures, Systems, and Components. Table 3.2-5 shows the relationship between Seismic Classification, Quality Group classification, and Safety Class.

Seismic Qualification Utility Group (SQUG) Methodology was used to verify the seismic adequacy of certain equipment as detailed in Reference 1. The SQUG's Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment provides methodology that relies primarily on the use of earthquake and test experience data to verify the seismic adequacy of generic classes of equipment. The NRC's Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on the GIP, Revision 2, Corrected February 14, 1992 (GIP-2) found the GIP-2 methodology to be an acceptable evaluation method for the USI A-46 plants to verify the seismic adequacy of safe-shutdown equipment and to satisfy the pertinent equipment seismic requirements of General Design Criterion 2 and the purpose of the NRC regulations relevant to equipment seismic adequacy including 10 CFR Part 100.

DAEC committed to the use of SQUG methodology as documented in the GIP-2, to resolve Unresolved Safety Issue (USI) A-46, Seismic Qualification of Equipment in Operating Plants, at the DAEC. The NRC safety evaluation on the resolution of USI A-46 at the DAEC states that the DAEC's A-46 implementation program has, in general, met the purpose and intent of the criteria in GIP-2 and the NRC's SSER No. 2 for the resolution of USI A-46.

Seismic Category “A”, The Main Steam Isolation Valve - Leakage Treatment System (MSIV-LTS) is seismically adequate to withstand the DAEC safe shutdown earthquake and maintain its functionality, and hence, meets the requirements of GDC-2 of Appendix A to 10CFR Part 50. A experience-based methodology was utilized to classify the MSIV Leakage Treatment System as seismically adequate. This seismic classification in Table 3.2-1 is denoted by an “A” in the column marked “Seismic Category”. The experience-based methodology is restricted to its application for ensuring the pressure boundary integrity and functionality of the main steam drain path associated with the MSIV leakage treatment system. The methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at the Duane Arnold Energy Center.

3.2.1.2 Nonseismic Structures, Systems, and Components

Nonseismic structures, systems, and components are those whose failure would not result in the release of significant radioactivity and would not prevent reactor shutdown. All structures, systems, and components not specifically listed as Seismic Category 1 are included in the nonseismic category. The failure of nonseismic structures, systems, or components may interrupt power generation.

Seismic, and nonseismic structures, systems and components are listed in Tables 3.2-1 and 3.2-3.

The equipment and piping classifications meet the general requirements given in Sections 3.2.2 and 5.2.1. They also meet the additional seismic requirements listed in Section 3.7 ("Seismic Design").

3.2.2 System Quality Group Classification

System quality group classifications have been determined for each component of (a) those applicable fluid systems relied upon to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, or to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (b) other associated safety related systems. A tabulation of quality group classification for each component so defined is shown in Table 3.2-1 under the heading "Quality Group Class."

Regulatory Guide 1.26 provides for the use of appropriate construction codes and standards which should be used for Quality Groups A through D. Figure 3.2-1 depicts the relative location of major components and the appropriate DAEC code of construction for these DAEC systems, as well as others, which are listed on Table 3.2-1 or Table 3.2-2. Table 3.2-5 compares the AEC (now NRC) Quality Group classification, Seismic Category and Quality Assurance requirements.

3.2.3 Conditions for Design

Two major categories of conditions might occur at the facility which must be appropriately considered in the design. These include (a) the plant process conditions as may be encountered during normal operation, anticipated operational occurrences, or postulated accidents; and (b) the conditions as may be imposed on the plant from the effects of natural phenomena. This subsection combines the plant process conditions (3.2.3.1) with the safe shutdown earthquake (SSE) and correlates these with design condition categories (normal, upset, emergency, and faulted) for structures within the Reactor Coolant Pressure Boundary (RCPB).

3.2.3.1 Plant Process Conditions (PPC) Considered in Design

The full spectrum of plant process conditions (PPC) are divided into four categories in accordance with their anticipated frequency of occurrence. The four categories of PPC are normal, frequent, infrequent, and limiting. These PPC are defined below and examples of representative process conditions are given.

3.2.3.1.1 Normal PPC

Normal PPC include process conditions which are expected to occur normally or regularly in the course of planned plant operation. Examples of normal PPC include the following:

(1) Refueling;

(2) Startup;

(3) Power Operation;

(4) Hot standby;

(5) Shutdown; and

(6) Routine testing and maintenance of components and systems during any of the above.

3.2.3.1.2 Frequent PPC

Frequent PPC are those incidents which are anticipated to occur occasionally during the life of the plant. Examples of frequent PPC include the following:

(1) Generator trip;

(2) Turbine trip;

(3) Isolation of any or all main steam lines;

(4) Loss of condenser cooling;

(5) Loss of feedwater heating;

(6) Inadvertent moderator cooldown;

(7) Control rod withdrawal error;

(8) Loss of feedwater flow;

(9) Total loss of offsite a-c power;

(10) Trip of any or all recirculation pumps;

(11) Inadvertent pump start in a hot recirculation loop;

(12) Inadvertent opening of a safety/safety relief valve;

(13) Single failure of a control component or an active component such as:

a. Turbine pressure regulator failure

b. Feedwater controller failure

c. Recirculation flow control failure

(14) Single failure in the electrical system; and

(15) Minor reactor coolant system leak which requires plant shutdown.

3.2.3.1.3 Infrequent PPC

Infrequent PPC are those which might occur infrequently during the life of the plant. Examples of infrequent PPC include the following:

(1) Blowdown of reactor coolant through multiple safety or relief valves; loss of reactor coolant from a break or crack which does not depressurize the reactor system, but which requires the safety functions of isolation or containment, emergency core cooling, and reactor shutdown;

(2) Improper assembly of core during refueling;

(3) Seizure of one recirculation pump;

(4) Startup of an idle recirculation pump in a cold loop;

(5) Reactor overpressure with delayed scram; and

(6) Release of radioactive material resulting from radwaste equipment failure.

3.2.3.1.4 Limiting PPC

Limiting PPC are those faults that are not expected to occur, but are postulated because their consequence would include the potential for the release of significant amounts of radioactive material. Limiting PPC are the most drastic process events for which plant protection must be provided. Examples of limiting PPC include the following:

(1) Control rod drop accident;

(2) Fuel handling accident resulting in major clad damage of irradiated fuel;

(3) Major rupture of that portion of the steam line which in not a part of the reactor coolant pressure boundary up to and including a double-ended rupture of the steam line; and

2017-002 (4) Major rupture of any pipe in the reactor coolant pressure boundary larger than that defined as an infrequent PPC and including a double-ended rupture of the largest pipe.

The loading combinations for this event include normal operating loads, plus safe shutdown earthquake loads, plus associated accident loads.

2017-002 3.2.3.2 Natural Phenomena and Environmental Conditions Considered in Design

The full range of natural phenomena and environmental conditions in the vicinity of the site must be evaluated to establish the design bases for structures, systems, and components important to safety. The range of conditions is established on the basis of suitable historical data and environmental conditions as discussed in UFSAR Chapter 2.

3.2.3.3 Design Condition Categories

The postulated combination of plant process conditions and the SSE describe separate design conditions which must be appropriately considered in the design of structures within the Reactor Coolant Pressure Boundary (RCPB) ("structures" as used in the power piping codes, not the same as civil structures such as Reactor Building), as well as systems and components important to safety. Design condition categories commonly used to define such combined effects are the normal, upset, emergency, and faulted conditions as defined in UFSAR Section 3.9.3 for mechanical equipment.

3.2.4 Safety Classes

"Structures within the RCPB," systems and components shall be classified as Safety Class 1, Safety Class 2, Safety Class 3 or Other in accordance with the importance of the safety functions to be performed by such equipment. Equipment is assigned a specific safety class, recognizing that components within a system may be of differing safety importance. A single system may thus have components in more than one safety class. (Supports shall be in the same class as the component supported.)

The safety classes are defined in this section and examples of their broad application are given. Because of specific design considerations, these general definitions are subject to interpretation and exceptions. These interpretations and exceptions are to be obtained from the applicable design specifications. Table 3.2-1, under the heading of "Safety Classes" provides a summary of the safety classes for the principal structures within the RCPB, systems and components of the plant. Table 3.2-5 shows the relationship between Seismic Classification, Quality Group classification, and Safety Class.

Design requirements for components of safety classes are stated in this section. Where possible, reference is made to accepted industry codes and standards which define design requirement commensurate with the safety function(s) to be performed by components of a particular safety class for a given condition of design.

Design requirements for safety related plant structures are considered in UFSAR Sections 2.5 and 3.3 through 3.8.

3.2.4.1 Safety Class 1

3.2.4.1.1 Definition of Safety Class 1

Safety Class 1 (SC-1) applies to components of the RCPB whose failure could cause a loss of reactor coolant, as defined in Subsection 3.2.3.1.3, Infrequent PPC, and Subsection 3.2.3.1.4, Limiting PPC.

2017-002 3.2.4.1.2 Design Requirements for Safety Class 1

Tables 3.2-1 and 3.2-2 list industry code requirements for SC-1 mechanical components and structural foundations.

3.4.2.2 Safety Class 2

3.2.4.2.1 Definition of Safety Class 2

Safety Class 2 (SC-2) applies to those structures within the RCPB, systems and components that are not Safety Class 1 but are necessary to accomplish the safety functions of:

(1) inserting negative reactivity to shut down the reactor;

(2) preventing rapid insertion of positive reactivity;

(3) maintaining core geometry appropriate to all plant process conditions;

(4) providing emergency core cooling;

(5) providing and maintaining containment;

(6) removing residual heat from the reactor and reactor core; and

(7) storing spent fuel.

Safety Class 2 includes the following:

a. Reactor protection system

b. Those components of the control rod system which are necessary to render the reactor subcritical

c. Systems or components which restrict the rate of insertion of positive reactivity

d. The assembly of components of the reactor core which maintain core geometry including the fuel assemblies, core support structure, and core grid plate, as examples.

e. Other components within the reactor vessel such as jet pumps, core shroud and core spray components which are necessary to accomplish the safety function of emergency core cooling

f. Emergency core cooling systems

g. Primary and Secondary Containment

h. Post-accident containment heat removal systems

i. Primary Containment hydrogen control system

j. Initiating systems required to accomplish safety functions, including emergency core cooling initiating system and containment isolation initiating system

k. At least one of the systems which recirculates reactor coolant to remove decay heat when the reactor is not pressurized

l. Spent fuel storage racks and spent fuel pool

m. Electrical and instrument auxiliaries necessary to operation of the above

n. Pipes having a nominal pipe size of 3/4 in. or smaller, that are connected to the reactor coolant pressure boundary.

3.2.4.2.2 Design Requirements for Safety Class 2

In applying industry codes to SC-2 equipment, the codes, except for mechanical equipment, do not fit neatly and automatically into the safety class and design condition designations developed in this section. Therefore, mechanical and structural categories will be treated separately from electrical. Tables 3.2-1 and 3.2-2 list the code requirements for mechanical systems and structures within the RCPB of SC-2 designation.

Design requirements for protection and Class 1E electrical systems (as defined in IEEE-279 and IEEE-308 of SC-2) of SC-2 are shown in Table 3.2-4.

3.2.4.3 Safety Class 3

3.2.4.3.1 Definition of Safety Class 3

Safety Class 3 (SC-3) applies to those structures, systems, and components not included in Safety Class 1 or Safety Class 2, but:

(1) Whose function is to process radioactive wastes and whose failure would result in release to the environment of gas, liquid, or solids resulting in a single-event dose that would be greater than the annual dose from 10CFR20. This is currently interpreted to be 500 mRem at a point on the site boundary; and

(2) Which provide or support any safety system function.

Safety Class 3 includes the following:

a. Portions of the gaseous waste disposal system in accordance with item 1, above

b. Those portions of the radwaste equipment or structures required to prevent an excessive rate of leakage of liquids from the liquid waste disposal system to the environs

c. Cooling water systems required for the purpose of:

1. Removal of decay heat from the reactor

2. Emergency core cooling

3. Post-accident heat removal from the suppression pool

4. Providing cooling water needed for the functioning of emergency systems

d. Fuel supply for the onsite emergency electrical system

e. Emergency equipment area cooling

f. Portions of the compressed gas or hydraulic systems required to support control or operation of safety systems

g. Electrical and instrumentation auxiliaries necessary for operation of the above

3.2.4.3.2 Design Requirements for Safety Class 3

The design requirements for Safety Class 3 mechanical and structural categories (within the RCPB) are listed in Tables 3.2-1 and 3.2-2.

Design requirements for Safety Class 3 electrical equipment are shown in Table 3.2-4.

Safety Class 3 components need not be designed for the SSE if the components are not required to mitigate the consequences of a LOCA or if their failure will not result in release to the environment of radioactive material which exceeds the requirements of Subsection 3.2.4.3.1(1).

3.2.4.4 Other Structures, Systems, and Components

3.2.4.4.1 Definition of Other Structures, Systems, and Components

A boiling water reactor has a number of structures, systems, and components in the power conversion or other portions of the facility which have no direct safety function but which may be connected to or influenced by the equipment within the Safety
2017-002 Classes defined above. Such structures, systems, and components are designated as "other".

3.2.4.4.2 Design Requirements for Other Structures, Systems, and Components

The design requirements for equipment classified as "other" shall be specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where possible, design requirements should be based on applicable industry codes and standards. If these are not available, the designer should rely on accepted industry or engineering practice.

3.2.4.5 Design Requirements for Safety Class 2 and 3 Electrical Systems and Components

Design requirements for Safety Class 2 and 3 electrical systems and components are contained in Table 3.2-4.

3.2.5 Quality Assurance

Structures, systems, and components whose safety functions require conformance to the quality assurance requirements of 10CFR50, Appendix B, are summarized in Table 3.2-1 under the heading, "Quality Assurance Req." and Table 3.2-3. Table 3.2-5 shows the relationship between seismic classification, Quality Group classification, Safety Class, and Quality Assurance requirements.

3.2.6 Correlation of Safety Classes with Industry Codes

The design of plant equipment is commensurate with the safety importance of the equipment. Hence, the various safety classes have a gradation of design requirements. The correlation of safety classes with other design requirements are summarized in Table 3.2-5.

3.2.7 Classification of Piping Systems

Piping and equipment are classified according to service and location. The design, fabrication, inspection, and testing requirements that are defined within this section for each specified classification group ensure the proper pressure integrity for the item listed.

The requirements and provisions of this section are applicable to the nuclear energy system components, as shown in Table 3.2-1 and Figures 3.2-1 and 3.2-2, such as pressure vessels, piping, pumps, valves and their equipment pressure parts such as fittings, flanges, bolts, valve bodies, pump casings, and similar piping system parts that constitute a pressure boundary for the process fluid.

Specifically excluded from the scope of this section are nonpressure parts, such as: shafts, seals, impellers, wear rings, gland followers, seat rings, guides, and operators; any nonmetallic material, such as packing and gaskets; fasteners not in pressure part joints; and washers of any kind.

Piping systems, valves, pumps, heat exchangers, pressure vessels, and equipment pressure parts have been separated into classifications corresponding to applicable codes and industrial standards. These classifications meet code requirements to the extent outlined. Table 3.2-2 gives a summary of major codes applied depending on the component purchase order date. Where information regarding codes for an individual component is available, this is included in Table 3.2-1.

System piping and equipment pressure parts (including those of BWR nuclear systems) are classified as follows:

1. Quality Group A - Piping and equipment pressure parts within the reactor coolant pressure boundary through the outer isolation valves, inclusive.

2. Quality Group B - Piping and equipment pressure parts downstream of the outer isolation valves, extensions of containment, and the emergency core cooling systems.

3. Quality Group C - Auxiliaries to emergency core cooling system or radioactive waste process piping and equipment pressure parts, excluding power generation systems.

4. Quality Group D - Balance-of-plant piping and equipment pressure parts, including power generation systems.

   Group D + QA in the quality group class column and a B in quality assurance requirements column is a critical piping or critical hanger system which requires an Appendix B QA Program. This piping is critical because it has B31.1 quality assurance requirements, and because it has additional nuclear quality requirements.

   Group D + QA in the quality group class column and a D in the quality assurance requirements column is a critical piping or critical hanger system which requires a quality assurance program consistent with good practice for steam power plants. There are no additional requirements identified with its use in a nuclear power plant; therefore, an Appendix B QA Program is not required.

   This means if the Table 3.2-1 entry is “D + QA” and the QA requirement is “B”, the quality level must be 1 or 2. If the Table 3.2-1 entry is “D + QA” and the QA requirement is “D”, the quality level must be 3 or 4.

The Quality Group for an individual structure or component can be found on Table 3.2-1 under the heading “Quality Group Class.”

The piping classes for each particular system are shown on the system P&ID that is included as a figure in the appropriate section of the UFSAR.

Piping classes are designated by a three-letter code on the P&ID for piping not within GE’s scope of supply. The first letter indicates the primary valve and flange rating; the second letter, the type of material; and the third letter, the code to which the piping is designed.

The key to interpreting the first two letters in the line designation is contained in Bechtel Specification 7884-M-190. The third letter in the line designation code can be interpreted as follows:

A - Nuclear Power Piping Code ANSI B31.7, Class I	(ASME Section III (1971), Class 1, for the CRD Hydraulic System)
B - Nuclear Power Piping Code, ANSI B31.7, Class II	(ASME Section (1971), Class 2, for the CRD Hydraulic System)
C - Nuclear Power Piping Code, ANSI B31.7, Class III	(ASME Section III (1971), Class 3, for the CRD Hydraulic System)
D - Nuclear Power Piping Code,	ANSI B31.1.0-1967

E - ASME Nuclear Vessels Code Section III, Class B, Extension of Containment Code Cases 1425, 1426, and 1427 (code date and addenda as noted on Table 3.2-1 or Table 3.2-2)

F - National Fire Protection Code

G - National Plumbing Code

All instrument lines are of the same classification as the system to which they are attached, except that those that contain an excess flow check valve (EFCV) are classified as Group D beyond the EFCV (see Figure 3.2-2). Other requirements for instrument lines are contained in footnote la to Table 3.2-1.

3.2.8 Applicability of Generic Letter 87-11

Generic Letter (GL) 87-11, “Relaxation in Arbitrary Intermediate Pipe Rupture Requirements,” was issued on June 19, 1987 and contains Revision 2 to Branch Technical Position MEB 3-1 of Standard Review Plan (SRP) Section 3.6.2. This revision eliminates all dynamic effects (missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, compartment, subcomponent and cavity pressurization and decompression waves within the ruptured pipe) and all environmental effects (pressure, temperature, humidity and flooding) resulting from arbitrary intermediate pipe ruptures. However, this revision retains the requirements for postulated terminal end pipe ruptures, postulated intermediate pipe ruptures at locations of high stress and high usage factor and for leakage cracks.

GL 87-11's applicability to DAEC is determined on a per system basis; where the system is defined as a section of piping between two anchors. Where GL 87-11 is utilized to eliminate the requirements for arbitrary intermediate pipe ruptures, the basis for such a relaxation will be incorporated into this section.

Stress analyses based on the 1986 Edition of the ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," were performed on the following lines:

- the HPCI Steam Supply Line,
- the RCIC Steam Supply Line, and
- the RWCU return line from the 1E214A regenerative heat exchanger to the feed water piping.

Stress analysis based on the 1977 Edition with Winter 78 Addenda of the ASME Section III Code "Rules for Construction of Nuclear Power Plant Components," were performed on the following lines:

- the Feedwater lines between the drywell penetrations and the outboard whip restraints.

GL 87-11 was found applicable to each of these lines, the intent of the letter has been met via the stress analyses conclusions and the system design bases. Therefore, GL 87-11 has been incorporated for eliminating the arbitrary pipe breaks in these lines.

REFERENCES FOR SECTION 3.2

1. R. Laufer (NRC) letter, to L. Liu (IES), Safety Evaluation on the Resolution of USI A-46 at the DAEC, July 29, 1998.

TABLE 3.2-1
INDEX

Sys #	SysName	SysCode
I	Reactor System	B11
II	Nuclear Boiler System	B21
III	Recirculation System	B31
IV	CRD Hydraulic System	C11
V	Standby Liquid Control System	C41
VI	Neutron Monitoring System	C51
VII	Reactor Protection System	C71
VIII	Process Radiation Monitors	D11
IX	RHR System	E11
X	Low Pressure Core Spray	E21
XI	HPCI System	E41
XII	RCIC System	E51
XIII	Fuel Service Equipment	F11
XIV	Reactor Vessel Service Equipment	F13
XV	In-Vessel Service Equipment	F14
XVI	Refueling Equipment	F15
XVII	Storage Equipment	F16
XVIII	Radwaste System	G11
XIX	Reactor Water Cleanup System	G31
XX	Fuel Pool Cooling and Cleanup System	G41
XXI	Control Room and Remote Shutdown Panels	H11 C61
XXII	Local Panels and Racks	H21
XXIII	Offgas System	N62
XXIV	Emergency Service Water	E13
XXV	RHR Service Water System	E12
XXVI	RBCCW	P42
XXVII	Well Water System	P46
XXVIII	Pneumatic Systems	T48 P50
XXIX	Diesel Generator Systems	R43
XXX	Containment Atmosphere Control System	T48
XXXI	Standby Gas Treatment System	T46
XXXII	ECCS Equipment Area Cooling System	T41
XXXIII	Power Conversion System	N11 N21
XXXIV	Condensate Storage and Transfer System	P11
XXXV	Auxiliary a-c Power System	R20, R22-24
XXXVI	125/250 Volt d-c Power System	R42
XXXVII	River Water Supply	W10
XXXVIII	Not Used	
XXXIX	HVAC	
XXXX	Miscellaneous Components	

TABLE 3.2-1
INDEX (by SysName)

Sys #	SysName	SysCode
XXXVI	125/250 Volt d-c Power System	R42
XXXV	Auxiliary a-c Power System	R20, R22-24
XXXIV	Condensate Storage and Transfer System	P11
XXX	Containment Atmosphere Control System	T48
XXI	Control Room and Remote Shutdown Panels	H11 C61
IV	CRD Hydraulic System	C11
XXIX	Diesel Generator Systems	R43
XXXII	ECCS Equipment Area Cooling System	T41
XXIV	Emergency Service Water	E13
XX	Fuel Pool Cooling and Cleanup System	G41
XIII	Fuel Service Equipment	F11
XI	HPCI System	E41
XXXIX	HVAC	
XV	In-Vessel Service Equipment	F14
XXII	Local Panels and Racks	H21
X	Low Pressure Core Spray	E21
XXXX	Miscellaneous Components	
VI	Neutron Monitoring System	C51
II	Nuclear Boiler System	B21
XXIII	Offgas System	N62
XXVIII	Pneumatic Systems	T48 P50
XXXIII	Power Conversion System	N11 N21
VIII	Process Radiation Monitors	D11
XVIII	Radwaste System	G11
XXVI	RBCCW	P42
XII	RCIC System	E51
VII	Reactor Protection System	C71
I	Reactor System	B11
XIV	Reactor Vessel Service Equipment	F13
XIX	Reactor Water Cleanup	G31
III	Recirculation System	B31
XVI	Refueling Equipment	F15
XXV	RHR Service Water System	E12
IX	RHR System	E11
XXXVII	River Water Supply	W10
XXXI	Standby Gas Treatment System	T46
V	Standby Liquid Control System	C41
XVII	Storage Equipment	F16
XXVII	Well Water System	P46

TABLE 3.2-1

DAEC Classification of Components in System

	Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
I	Reactor System	B11									
1	Reactor vessel	GE/C	1	A	ASME Section III, 1965 Edition, Summer 1967 Addenda	A	B	I	-	1d	No
2	Reactor vessel support skirt	GE	1	A	ASME Section III, 1965 Edition, Summer 1967 Addenda	A	B	I	-	1d	No
3	Reactor vessel appurtenances, pressure retaining portions	GE	1	A	ASME Section III, 1965 Edition, Summer 1967 Addenda	A	B	I	-	1d	No
4	CRD Housing Supports (Shoot-out Steel)	GE	2	-	-	-	B	I	-	-	No
5	Reactor internal structures, engineered safety features	GE	2	A	ASME Section III, 1965 Edition, Summer 1967 Addenda	NA	B	I	-	1d,1u	No
6	Reactor internal structures, other	GE	Other	-	-	NA	B	I	-	1u	No
7	Control rods	GE	2	-	-	NA	B	I	-	-	No
8	Control rod drives	GE	1	1	Section III, Class 1 appurtenances	A	B	I	-	-	No
9	Core support structure	GE	2	A	ASME Section III, 1965 Edition, Summer 1967 Addenda	NA	B	I	-	1d,1u	No
10	Power range detector hardware	GE	2	-	-	B	B	I	-	1a,1b	No
11	Fuel assemblies	GE	2	-	-	NA	B	I	-	-	No
12	RX Vessel Stabilizer	GE	2	A	ASME Section III, 1965 Edition, Summer 1967 Addenda	NA	B	I	-	-	No
13	Refueling bellows	GE	Other	-	-	-	-	NA	-	-	No
II	Nuclear Boiler System	B21									
1	Vessels, level instrumentation condensing chambers	GE	1	-	-	A	B	I	-	-	No
2	Vessels, N2 accumulators	B	2	2	ASME Section III	-	B	I	02/26/73	-	No
3	Piping, relief valve discharge	B	3	3	USAS B31.7-1969	C	B	I	07/30/70	-	No
4	Piping, main steam within outermost isolation valve	GE	1	-	ANSI B31.1.0 + Code Cases N2, N7, N9, N10	A	B	I	12/05/69	1a,1b	No
5	Pipe supports, main steam	GE	1	1	Requirements for Class 1 piping supports in ANSI B31.7.	A	B	I	02/26/71	-	No
6	Pipe restraints, main steam	B	2	-	-	NA	B	I	-	-	No
7	Piping, other within outermost isolation valves	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b	No
8	Piping, instrumentation beyond outermost isolation valves	B	Other	-	USAS B31.1.0-1967	-	B	-	-	1a,1b	No

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	Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
9	Safety valves	GE	1	1	ANSI B31.1.0, Addenda and applicable code cases or NDE standards of B31.1. Code Cases N2, N7, N9, N10 except that the acceptance standards for Class 1 valves in the Draft ASME Code for Pumps and Valves for Nuclear Power may be applied. Use ANSI 16.5 or MSS-SP-66 for design.	A	B	I	12/30/69	1f	No
10	Relief valves	GE	1	1	ASME Code, Section III, 1968 Edition, Article 9, with 1968 Winter Addenda.	A	B	I	12/30/69	1f	No
11	Valves, main steam isolation valves	GE	1	1	ANSI B31.1.0, Addenda and applicable code cases or NDE standards of B31.1. Code Cases N2, N7, N9, N10 except that the acceptance standards for Class 1 valves in the Draft ASME Code for Pumps and Valves for Nuclear Power may be applied. Use ANSI 16.5 or MSS-SP-66 for design.	A	B	I	10/15/69	1f	No
12	Valves, other, isolation valves and within	B	1	1	ASME Code for Pumps and Valves for Nuclear Power	A	B	I	10/16/70	1f	No
13	Valves, instrumentation beyond outermost isolation valves	B	Other	-	USAS B31.1.0-1967	-	-	-	-	1a,1b,1f	No
14	Mechanical modules, instrumentation, with safety function	GE	2	-	-	B	B	I	-	1c	No
15	Electrical modules with safety function	GE	2	-	-	NA	B	I	-	1c	No
16	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
III	Recirculation System	B31									
1	Piping	GE	1	-	ANSI B31.1.0 + Code Cases N2, N7, N9, N10	A	B	I	12/05/69	1a,1b	No
2	Pipe suspension, recirculation line	GE	1	1	Requirements for Class 1 piping supports in ANSI B31.7.	A	B	I	02/26/71	-	No
3	Pipe restraints recirculation line	GE	2	-	-	NA	B	I	-	-	No
4	Pumps	GE	1	1	Draft ASME Code for Pumps and Valves for Nuclear Power or NDE and acceptance requirements of ANSI B31.1 Code Cases N7,N9,N10 + Design Guide for sizing pressure parts in ASME Boiler and Pressure Vessel Code [1968] Section III, Class C	A	B	I	11/22/68	1e	No

TABLE 3.2-1

DAEC Classification of Components in System

	Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
5	Valves	GE	1	1	Requirements for Class 1 valves in Draft ASME Code for Pumps and Valves for Nuclear Power and requirements applicable to valves in ASME Section III [1968 Edition plus addenda], articles 1 and 8.	A	B	I	03/20/70	1f	No
6	Motor, pump	GE	Other	-	NEMA Standards	NA	D	I	-	1x	No
7	Electrical modules, with safety function	GE	2	-	-	NA	B	I	-	1c	No
8	Cable with safety function	B	2	-	-	-	B	I	-	-	No
IV	CRD Hydraulic System	C11									
1	Valves, isolation, water return line	B	1	1	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 1, or ASME Section III, 1971 Edition, Class 1	A	B	I	12/19/72	1f	No
2	Valves, scram discharge volume lines	GE/B	2/2	2	ASME Code for Pumps and Valves for Nuclear Power	B/B	B/B	I/I	12/19/72	1f	No
3	Valves insert and withdraw lines	GE/B	2/2	2	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power or ASME Section III, 1971 Edition, Class 1	B/B	B/B	I/I	12/19/72	1f	No
4	Valves, other	B	Other	2	ASME Section III, 1971 Edition	D	D	NA	12/19/72	1a,1f,1l	Yes
5	Piping, water return line within isolation valves	B	1	1	ASME Section III, 1971 Edition	A	B	I	12/19/72	-	No
6	Piping, scram discharge volume lines	B	2	2	ASME Section III, 1971 Edition	B	B	I	12/19/72	-	No
7	Piping, insert and withdraw lines	B	2	2	ASME Section III, 1971 Edition	B	B	I	12/19/72	-	No
8	Piping, other	B	Other	2	ASME Section III, 1971 Edition	D	D	NA	12/19/72	1a,1b,1l	Yes
9	Hydraulic control unit	GE	2	-	-	Special	B	I	-	-	Yes
10	Electrical modules, with safety function	GE	2	-	-	NA	B	I	-	1c	No
11	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
V	Standby Liquid Control System	C41									
1	Standby liquid control tank	GE	2	-	API-650 and ASME Section VIII, Div. 1	B	B	I	-	-	Yes
2	Pump	GE	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	-	1e	No
3	Pump motor	GE	2		-	NA	B	I	-	-	No
4	Valves, explosive	GE	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	-	1f	No
5	Valves, isolation and within	B	1	1	ASME Code for Pumps and Valves for Nuclear Power	A	B	I	10/16/70	1f	No
6	Valves, beyond isolation valves	B	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	10/16/70	1f	No
7	Piping, within isolation valves	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b	No

TABLE 3.2-1
DAEC Classification of Components in System

	Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
8	Piping, beyond isolation valves	B	2	2	USAS B31.7-1969	B	B	I	07/30/70	1a,1b,1m	No
9	Electrical modules, with safety function	GE	2	-	-	NA	B	I	-	1c	No
10	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
2011-008 11	Test Tank	GE	Other	-	-	D	D	NA	-	-	No
12	Piping, between test tank and its isolation valves	B	Other	-	USAS B31.1.0	D	D	NA	-	-	No
VI	Neutron Monitoring System	C51									
1	Piping, TIP	GE	2	-	-	B	B	I	-	1a,1b	No
2	Valves, isolation, TIP subsystem	GE	2	-	-	B	B	I	-	1f	No
3	Electrical modules, IRM and APRM	GE	2	-	-	NA	B	I	-	1c,1w	Yes
4	Cable, IRM and APRM	B	2	-	-	-	B	I	-	-	No
VII	Reactor Protection System	C71									
1	Electrical modules	GE	2	-	-	NA	B	I	-	1c	No
2	Cable	B	2	-	-	-	B	I	-	-	No
VIII	Process Radiation Monitors	D11									
1	Electrical modules for main steam line and reactor building ventilation monitors	GE	2	-	-	NA	B	I	-	1c	No
2	Cable for main steam line and reactor building ventilation monitors	B	2	-	-	-	B	I	-	-	No
IX	RHR System	E11									
1	Heat exchangers, primary side	GE	2	B	ASME Section III, Class B and TEMA-C	B	B	I	08/15/69	-	No
2	Heat exchangers, secondary side	GE	3	-	ASME Section VIII, Div. 1, and TEMA-C	C	B	I	08/15/69	-	No
3	Piping, within outermost LPCI & shutdown cooling isolation valves	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b	No
4	Piping, other	B	2	2	USAS B31.7-1969	B	B	I	07/30/70	1a,1b	No
5	Pumps	GE	2	2	Draft ASME Code for Pumps and Valves for Nuclear Power or NDE and acceptance requirements of ANSI B31.1 Code Cases N7,N9,N10 + Design Guide for sizing pressure parts in ASME Boiler and Pressure Vessel Code [1968] Section III, Class C	B	B	I	09/17/69	1e	No
6	Pump motors	GE	2	-	-	NA	B	I	-	-	No
7	Valves, isolation, LPCI & shutdown cooling lines	B	1	1	ASME Code for Pumps and Valves for Nuclear Power	A	B	I	10/16/70	1f	No
8	Valves, isolation, other	B	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	10/16/70	1f	No
9	Valves, beyond isolation valves	B	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	10/16/70	1f	No
10	Mechanical modules	GE	2	-	-	B	B	I	-	1c	No

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		Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
	11	Electrical modules, with safety function	GE	2	-	-	NA	B	I	-	1c	No
	12	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
X		Low Pressure Core Spray	E21									
	1	Piping, within outermost isolation valves	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b	No
	2	Piping, beyond outermost isolation valves	B	2	2	USAS B31.7-1969	B	B	I	07/30/70	1a,1b	No
	3	Piping, floodup line to condensate storage tank	B	Other	-	USAS B31.1.0	C	B	I	-	1a,1b	No
	4	Pumps	GE	2	2	Draft ASME Code for Pumps and Valves for Nuclear Power or NDE and acceptance requirements of ANSI B31.1 Code Cases N7,N9,N10 + Design Guide for sizing pressure parts in ASME Boiler and Pressure Vessel Code [1968] Section III, Class C	B	B	I	09/17/69	1e	No
	5	Pump motors	GE	2	-	-	NA	B	I	-	-	No
	6	Valves, isolation and within	B	1	1	ASME Code for Pumps and Valves for Nuclear Power	A	B	I	10/16/70	1f	No
	7	Valves, beyond outermost isolation valves	B	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	10/16/70	1f	No
	8	Valves, floodup line to condensate storage tank	B	Other	-	USAS B31.1.0	C	B	I	-	1f	No
	9	Electrical modules with safety function	GE	2	-	-	NA	B	I	-	1c	No
	10	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
XI		HPCI System	E41									
	1	Piping, within outermost isolation valves	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b	No
	2	Piping, beyond outermost isolation valves	B	2	2	USAS B31.7-1969	B	B	I	07/30/70	1a,1b	No
	3	Piping, return test line to condensate storage tank beyond second isolation valve	B	Other	-	USAS B31.1.0	D	D	NA	07/30/70	1a,1b,1r	No
	4	Pumps	GE	2	2	Draft ASME Code for Pumps and Valves for Nuclear Power or NDE and acceptance requirements of ANSI B31.1 Code Cases N7,N9,N10 + Design Guide for sizing pressure parts in ASME Boiler and Pressure Vessel Code [1968] Section III, Class C	B	B	I	07/31/69	1e	No

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	Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
5	Valves, isolation and within	B	1	1	ASME Code for Pumps and Valves for Nuclear Power	A	B	I	10/16/70	1f	No
6	Valves, return test line to condensate storage beyond second isolation valve	B	Other	-	USAS B31.1.0	D	D	NA	10/16/70	1f	No
7	Vacuum pump discharge line	B	Other	-	USAS B31.1.0	D	D	NA	-	1a,1b	No
8	Valves, other	B	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	10/16/70	1f	No
9	Turbine	GE	2	-	NEMA Standards for Mechanical Drive Steam Turbine	NA	B	I	07/29/69	1g	Yes
10	Electrical modules, with safety function	GE	2	-	-	NA	B	I	-	1c	No
11	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
XII	RCIC System	E51									
1	Piping, within outermost isolation valves	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b	No
2	Piping, beyond outermost isolation valves	B	2	2	USAS B31.7-1969	B	B	I	07/30/70	1a,1b	No
3	Piping, return test line to condensate storage tank beyond second isolation valve	B	Other	-	USAS B31.1.0	D	D	NA	07/30/70	1a,1b,1r	No
4	Pumps	GE	2	2	Draft ASME Code for Pumps and Valves for Nuclear Power or NDE and acceptance requirements of ANSI B31.1 Code Cases N7,N9,N10 + Design Guide for sizing pressure parts in ASME Boiler and Pressure Vessel Code [1968] Section III, Class C	B	B	I	-	1e	No
5	Valves, isolation and within	B	1	1	ASME Code for Pumps and Valves for Nuclear Power	A	B	I	10/16/70	1f	No
6	Valves, return test line to condensate storage beyond second isolation valve	B	Other	-	USAS B31.1.0	D	D	NA	10/16/70	1f	No
7	Vacuum pump discharge line	B	Other	-	USAS B31.1.0	D	D	NA	-	1a,1b	No
8	Valves, other	B	2	2	ASME Code for Pumps and Valves for Nuclear Power	B	B	I	10/16/70	1f	No
9	Turbine	GE	2	-	NEMA Standards for Mechanical Drive Steam Turbine	NA	B	I	07/29/69	1g	Yes
10	Electrical modules, with safety function	GE	2	-	-	NA	B	I	-	1c	No
11	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
XIII	Fuel Service Equipment	F11									
1	Fuel preparation machine	GE	3	-	-	NA	B	I	-	-	No

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2	General purpose grapple	GE	2	-	-	NA	B	I	-	-	No
XIV	Reactor Vessel Service Equipment	F13									
1	Steam line plugs	GE	3	-	-	NA	B	I	-	-	No
2	Dryer and separator sling and head strongback	GE	2	-	-	NA	B	I	-	-	No
XV	In-Vessel Service Equipment	F14									
1	Control rod grapple	GE	2	-	-	NA	B	I	-	-	No
2	Lightweight Work Platform	GE	Other	-	-	NA	D	I	-	-	No
3	360 Degree Work Platform	GE	Other	-	-	NA	B	I	12/31/07	-	No
XVI	Refueling Equipment	F15									
1	Refueling equipment platform assembly	GE	2	-	-	NA	B	I	-	-	No
XVII	Storage Equipment	F16									
1	Fuel storage racks	GE	2	-	-	NA	B	I	-	-	No
2	Defective fuel storage container	GE	3	-	-	-	B	I	-	-	No
XVIII	Radwaste System	G11									
1	Tanks, Atmospheric	GE/B	Oth/Oth		API-650 or AWWA-D100 or ANSI B96.1 or equivalent plus NDE per ASME Section VIII Div. 1.	C&D	D/D	NA/NA	07/16/70	1h	Yes
2	Heat exchangers	GE/B	Oth/Oth	-	-	D/D	D/D	NA/NA	-	-	No
3	Piping and valves, containment isolation	B	2	2	USAS B31.7-1969	B	B	I	-	1a,1b	No
4	Piping, other	B	Other	3	USAS B31.7-1969	C&D	D	NA	-	1a,1b,1j	Yes
5	Pumps	GE	Other	3	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 3, or ASME Section III, 1971 Edition, Class 3	C&D	D	NA	07/03/72	1e,1j	No
6	Valves, flow control and filter system	GE	Other	3	ASME Code for Pumps and Valves for Nuclear Power	C&D	D	NA	-	1f,1j	No
7	Valves, other	B	Other	3	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 3, or ASME Section III, 1971 Edition, Class 3	C&D	D	NA	-	1f,1j	No
8	Mechanical modules	GE	Other	-	-	C&D	D	NA	-	1c,1j	No
XIX	Reactor Water Cleanup System	G31									
1	Vessels: filter/demineralizer	GE	Other	C	ASME Section III	C	B	I	12/17/70	1t	No
2	Heat exchangers	GE	Other	-	ASME Section VIII and TEMA-C	C	B	I	09/25/67	1t	No
3	Piping, within outermost isolation valves	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b	No
4	Piping, beyond outermost isolation valves	B	Other	3	USAS B31.7-1969	C	B	I	07/30/70	1a,1b,1o, 1t	Yes
5	Pumps	GE	Other	3	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 3, or ASME Section III, 1971 Edition, Class 3	C	B	I	12/23/69	1e,1t	No

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6	Valves, isolation valves and within	B	1	1	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 1, or ASME Section III, 1971 Edition, Class 1	A	B	I	-	1f	No
7	Valves, beyond outermost isolation valves	GE/B	Oth/Oth	3	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 3, or ASME Section III, 1971 Edition, Class 3	C/C	B	I	-	1f,1o,1t	No
8	Mechanical modules	GE	Other	-	-	-	B	I	-	1c,1t	No
XX	Fuel Pool Cooling and Cleanup System	G41									
1	Vessels, filter/demineralizers	GE	Other	-	ASME Section VIII, Div. 1.	C	D	NA	-	-	No
2	Vessels, other	B	Other	-	ASME Section VIII, Div. 1.	C	D	NA	-	-	No
3	Heat exchangers	GE	Other	-	ASME Section VIII, Div. 1, and TEMA-C	C	D	NA	12/23/69	-	No
4	Pumps	GE	Other	3	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 3, or ASME Section III, 1971 Edition, Class 3	C	D	NA	-	-	No
5	Piping	B	Other	3	USAS B31.7-1969	C	D	NA	-	1a	No
6	Valves	B	Other	-	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 3, or ASME Section III, 1971 Edition, Class 3	C&D	D	NA	-	1f	No
XXI	Control Room & Remote Shutdown Panels	H11, C61									
1	Electrical modules, with safety function	GE	2	-	-	NA	B	I	-	1c	No
2	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
XXII	Local Panels and Racks	H21									
1	Electrical modules, with safety function	GE/B	2/2	-	-	B/-	B/B	I/I	-	1c	No
2	Cable, with safety function	B	2	-	-	-	B	I	-	-	No
XXIII	Offgas System	N62									
1	Tanks	GE	Other	-	AWWA D100 or API-650	-	D	NA	10/21/71	1h	No
2	Heat exchangers	GE	Other	-	Section III & TEMA-C	-	D	NA	11/20/72	-	No
3	Piping	B	Other	3	ASME Section III-1971	-	D	NA	07/27/72	1k	Yes
4	Pumps	GE	Other	-	-	D	D	NA	-	1e,1k	Yes
5	Valves, flow control	B	Other	-	-	D	D	NA	-	1f,1k	Yes
6	Valves, other	B	Other	-	-	D	D	NA	11/29/71	1f,1k	Yes
7	Mechanical modules	GE	Other	-	-	D	D	NA	-	1c	No
8	Pressure vessels	GE	Other	-	-	D	D	NA	10/21/71	-	No
XXIV	Emergency Service Water	E13									
1	Piping	B	3	-	ANSI B31.1.0-1967	D+QA	B	I	07/30/70	1y	Yes
2	Pumps	B	3	3	ASME Code for Pumps and Valves for Nuclear Power	C	B	I	-	1e	No

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3	Pump motors	B	3	-	-		B	I	-	-	No
4	Valves	B	3	3	ASME Code for Pumps and Valves for Nuclear Power	C	B	I	10/16/70	1f	No
5	Electrical modules, with safety function	B	3	-	-	-	B	I	-	1c	No
6	Cable, with safety function	B	3	-	-	-	B	I	-	-	No
XXV	RHR Service Water System	E12									
1	Piping	B	3	3	USAS B31.7-1969	C	B	I	07/30/70	1p, 1y	No
2	Pumps	B	3	3	ANSI B31.1.0 or ASME Code for Pumps and Valves for Nuclear Power, Class 3, or ASME Section III, 1971 Edition, Class 3	C	B	I	-	1e	No
3	Pump motors	B	3	-	-	-	B	I	-	-	No
4	Valves	B	3	3	ASME Code for Pumps and Valves for Nuclear Power	C	B	I	10/16/70	1f,1p	No
5	Electrical modules, with safety function	B	3	-	-	-	B	I	-	1c	No
6	Cable, with safety function	B	3	-	-	-	B	I	-	-	No
XXVI	RBCCW	P42									
1	Piping, and valves forming part of primary containment boundary	B	2	B	ASME Nuclear Vessels Code Section III, Extension of Containment Code Cases 1425, 1426 & 1427	B	B	I	07/30/70	-	No
2	Piping and valves inside drywell	B	Other	-	USAS B31.1.0	D	B	I	-	1s	No
3	Piping and valves, other	B	Other	-	USAS B31.1.0	D	-	NA	-	-	No
XXVII	Well Water System	P46									
1	Piping and valves forming part of primary containment boundary	B	2	B	ASME Nuclear Vessels Code, Section III, Extension of Containment Code Cases 1425, 1426 & 1427	B	B	I	-	-	No
2	Piping and valves inside drywell	B	Other	-	USAS B31.1.0	D	B	I	-	1s	No
3	Piping and valves, other	B	Other	-	USAS B31.1.0	D	-	NA	-	-	No
XXVIII	Pneumatic Systems	T48, P50									
1	Nitrogen vessels, accumulators, supporting safety-related systems	B	2	2	ASME Section III	-	B	I	02/26/73	-	No
2	Nitrogen piping and valves in lines between above accumulators and safety-related systems	B	2	2	ASME Section III	-	B	I	-	-	No
3	Nitrogen piping and valves forming part of containment boundary	B	2	2	ASME Section III	-	B	I	-	1b	No
4	Instrument air vessels, accumulators, supporting safety-related systems	B	3	-	-	-	B	I	02/26/73	-	No

TABLE 3.2-1
DAEC Classification of Components in System

	Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
5	Instrument air piping and valves in lines between above accumulators and safety-related systems	B	3	-	-	-	B	I	-	-	No
XXIX	Diesel Generator Systems	R43									
1	Day tanks	B	3	-	API-650 or AWWA-D100 or ANSI B96.1 or equivalent plus NDE per ASME Section VIII, Div. 1.	C	B	I	-	-	No
2	Piping and valves, fuel oil system and diesel service water system	B	3	3	USAS B31.7 for pipe and ASME Code for Pumps and Valves for Nuclear Power	C	B	I	-	-	No
3	Pumps, fuel oil system and diesel service water system	B	3	3	ASME Code for Pumps and Valves for Nuclear Power	C	B	I	-	-	No
4	Pump motors, fuel oil system and diesel service water system	B	3	-	-	-	B	I	-	-	No
5	Diesel-generators	B	2	-	-	-	B	I	-	-	No
6	Electrical modules with safety function	B	3	-	-	-	B	I	-	1c	No
7	Cable, with safety function	B	3	-	-	-	B	I	-	-	No
8	Diesel fuel storage tanks	B	Other	-	API-650 or AWWA-D100 or ANSI B96.1 or equivalent plus NDE per ASME Section VIII, Div. 1.	C	B	I	-	-	No
9	Diesel Air Start System	B	2	-	-	-	B	I	-	-	No
XXX	Containment Atmosphere Control System	T48									
1	Piping and valves from primary containment through outer isolation valve	B	2	B	ASME Nuclear Vessels Code Section III, Extension of Containment Code Cases 1425, 1426 and 1427	B	B	I	07/30/70	-	No
XXXI	Standby Gas Treatment System	T46									
1	All components with safety function, including offgas stack dilution fans	B	3	-	-	-	B	I	-	-	No
XXXII	ECCS Equipment Area Cooling System	T41									
1	All components with safety functions	B	3	-	-	-	B	I	-	-	No
XXXIII	Power Conversion System	N11, N21									
1	Main steam piping from outboard MSIV to turbine stop valves and branch line piping up to and including first valve	B	Other	-	USAS B31.1.0	D+QA	B	I	07/30/70	1a	Yes
2	Steam piping and valves, other	B	Other	-	USAS B31.1.0	D+QA	D	NA	07/30/70	1a	No
3	Reactor feedwater piping and valves, RPV to outermost isolation valve	B	1	1	USAS B31.7-1969	A	B	I	07/30/70	1a,1b,1f	No
4	Reactor feedwater piping and valves, other	B	Other	-	USAS B31.1.0	D+QA	D	NA	07/30/70	1a,1f	Yes
5	MSIV-LTS piping and valves	IE	Other	-	USAS B31.1.0	D+QA	D	A	-	-	Yes
XXXIV	Condensate Storage and Transfer System	P11									

TABLE 3.2-1
DAEC Classification of Components in System

	Principle Component	Scope of Supply (a)	Safety Class (b)	Code Class	Construction Code (c)	Quality Group Class	Quality Assurance Req. (d)	Seismic Category (e)(f)	PO Date (g)	Footnotes	Comments (h)
1	Condensate storage tank	B	Other	-	API-650 plus augmented NDE of welds	D+QA	-	NA	07/30/70	1i	Yes
2	Piping and valves	B	Other	-	USAS B31.1.0	D	D	NA	-	1f,1q	No
3	Other components	B	Other	-	-	D	D	NA	-	1q	No
XXXV	Auxiliary a-c Power System	R20, R22-24									
1	All components with safety function	B	2	-	-	-	B	I	-	-	Yes
XXXVI	125/250 Volt d-c Power System	R42									
1	All components with safety function	B	2	-	-	-	B	I	-	-	No
XXXVII	River Water Supply	W10									
1	Piping, pumps and valves	B	3	-	ANSI B31.1.0	D+QA	B	I	-	1f, 1y	Yes
2	Intake traveling screen, trash rakes	B	3	-	-	-	B	I	-	-	No
3	Pump motors	B	3	-	-	-	B	I	-	-	No
XXXVIII	Not Used										
									-	-	No
XXXIX	HVAC										
1	Control room	-	3	-	-	-	B	I	-	-	No
2	Pump house	-	3	-	-	-	B	I	-	-	No
3	Emergency diesel generator room	-	3	-	-	-	B	I	-	-	No
4	Reactor building secondary containment isolation dampers	-	3	-	-	-	B	I	-	-	No
5	Battery rooms	-	3	-	-	-	B	I	-	-	No
6	Intake structure	-	3	-	-	-	B	I	-	-	No
7	Essential switchgear rooms	-	3	-	-	-	B	I	-	-	No
XXXX	Miscellaneous Components										
1	Reactor Building Crane	B	3	-	-	-	B	I	-	-	No
2	Containment Penetrations for Process Piping and Electrical	B	2	-	-	-	B	I	-	-	No
General	General	n/a									
											Yes

TABLE 3.2-1
DAEC Classification of Components in System
Footnotes

Footnote #	Footnote	Systems
a	GE = General Electric; B = Bechtel; C = CB&I; IE = Iowa Electric(=DAEC); DAEC = Duane Arnold Energy Center	
b	1, 2, 3, "other" = safety classes defined in Section 3.2.4; "unc" = unclassified as defined in Section 3.2.4.	
c	The equipment shall be constructed in accordance with the codes listed in Table 3.2-2, if no Code of Construction is provided in this table. The term "construction," as used in this UFSAR, includes provisions for design, materials, fabrication, erection, testing and inspection.	
d	B = The equipment shall meet the quality assurance requirements of 10CFR50, Appendix B, in accordance with the quality assurance program described in Chapter 17. D = The equipment shall be constructed in accordance with the quality assurance requirements consistent with the good practices for steam power plants.	
e	I = The equipment shall be constructed in accordance with the seismic requirements for the safe shutdown earthquake, as described in Section 3.7, Seismic Design. Seismic adequacy of certain equipment was verified by Seismic Qualification Utility Group (SQUG) methodology. The NRC issued a Safety Evaluation on the resolution of USI A-46 at the DAEC on July 29, 1998. NA = The seismic requirements for the safe shutdown earthquake are not applicable to the equipment. A = Seismic Adequate for MSIV-LTS.	
f	Portions of 'non-seismic category I' piping (seismic category NA) passing through rooms containing safeguard equipment are seismically supported as seismic category I.	
g	Date on the purchase order for the component. Where provided, this can be used to establish the code edition and addenda in effect for the component.	
h	A "yes" in this column signifies there is a comment regarding the item at the end of Table 3.2-1.	
1a	The following items are applicable to instrument, sampling or small bore (3/4" NPS and smaller), as noted: **(1)Lines ¾" and smaller which are part of the reactor coolant boundary shall be Safety Class 2. **(2)All instrument lines which are connected to the reactor coolant pressure boundary and are utilized to actuate safety systems shall be Safety Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrument. **(3)All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate safety systems shall be Quality Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation. **(4) All other instrument lines through the root valve shall be of the same classification as the system to which they are attached, except those lines that contain an excess flow check valve (EFCV) are classified as Quality Group D beyond the EFCV. See Figure 3.2-2. **(5) All other instrument lines beyond the root valve, if used to actuate a safety system, shall be the same classification as the system to which they are attached. **(6) All other instrument lines beyond the root valve, if not used to actuate a safety system, shall be quality Group D. **(7) All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system shall be Quality Group D.	I-10, II-7, II-8, II-13, II-4, III-1, IV-4, IV-8, V-7, V-8, VI-1, IX-3, IX-4, X-1, X-3, X-2, XI-1, XI-3. XI-2, XI-7, XII-2, XII-7, XII-1, XII-3, XVIII-4, XVIII-3, XIX-4, XIX-3, XX-5, XXXIII-4, XXXIII-1, XXXIII-3, XXXIII-2,
1b	ANSI B31, Code Case 78 applies for B31.7 Class 1 and Class 2 pipe fittings ¾" nominal pipe size (NPS) and smaller.	I-10, II-7, II-8, II-13, II-4, III-1, IV-8, V-8, V-7, VI-1, IX-4, IX-3, X-2, X-3, X-1, XI-2, XI-3, XI-7, XI-1, XII-3, XII-1, XII-7, XII-2, XVIII-3, XVIII-4, XIX-3, XIX-4, XXVIII-3, XXXIII-3,
1c	A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, and signal processors. Mechanical modules include turbines, strainers, and orifices.	II-14, II-15, III-7, IV-10, V-9, VI-3, VII-1, VIII-1, IX-10, IX-11, X-9, XI-10, XII-10, XVIII-8, XIX-8, XXI-1, XXII-1, XXIII-7, XXIV-5, XXV-5, XXIX-6,
1d	GE Specification 21A1100AS (Ref. 243) adds the following code requirements to the Reactor Vessel: The Winter 1967 Addenda to the ASME Code Section III is not to be included as a basis for purchase of this vessel, except as follows: 1) Charpy impact teats per N-331.2 of the Winter 1967 Addenda will be furnished; 2) Welds are to be ultrasonically examined using the angle beam method described by N-625 of Winter 1967 Addenda; 3) The changes to Article 4-Design by the Winter 1967 Addenda are included; 4) The addition of Appendix IX – Quality Control and Nondestructive Examination Methods in included.	I-1, I-2, I-3, I-5, I-9,

Footnote #	Footnote	Systems
1e	For pump designs, the applicable class, section, or subsection of the referenced ASME B&PV Code is used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting. For example, use ASME Section III, Class C, 1968 Editions, for a design guide for Quality Group A & B pumps. For Quality Group D below 150 psig and/or 212 deg. F, manufacturer's standard pump service intended may be used.	III-4, V-2, IX-5, X-4, XI-4, XII-4, XVIII-5, XIX-5, XXIII-4, XXIV-2, XXV-2
1f	ANSI B16.5 or MSS-SP-66 apply for valves (Note MSS-SP-66-1964 was withdrawn from publication in favor of ANSI B16.34-1973)	II-9, II-10, II-11, II-12, II-13, III-5, IV-1, IV-2, IV-3, IV-4, V-4, V-5, V-6, VI-2, IX-8, IX-9, IX-7, X-7, X-6, X-8, XI-8, XI-6, XI-5, XII-8, XII-6, XII-5, XVIII-6, XVIII-7, XIX-6, XIX-7, XX-6, XXIII-6, XXIII-5,
1g	The RCIC and HPCI turbines do not fall within the applicable design codes. To assure that the turbines are fabricated to the standards commensurate with their safety and performance requirements, General Electric has established specific design requirements for these components.	XI-9, XII-9,
1h	Existing API/AWWA standards and supplementary requirements apply. Tanks are to be constructed to meet the intent of API Standards 620 or 650 or AWWA Standard D100 for those fuel, oil, or water storage tanks.	XVIII-1, XXIII-1,
1i	The condensate storage tank will be designed, fabricated and tested to meet the intent of API Standard 650. In addition, the specifications for this tank will require 100% surface examination of the side wall to bottom joint and 100% volumetric examination of the side wall weld joints.	XXXIV-1,
1j	ASME Section VIII, Division I, and USAS B31.1.0 apply downstream of the outermost isolation valves.	XVIII-4, XVIII-5, XVIII-6, XVIII-7, XVIII-8,
1k	The gaseous radwaste system piping, pumps and valves containing gaseous radwaste shall be constructed in accordance with the applicable codes of Quality Group D	XXIII-3, XXIII-4, XXIII-5, XXIII-6,
1l	Some of this piping was also constructed to B31.1.0	IV-4, IV-8,
1m	Some lines, such as ECB-9 (drain to filter/demineralizer), are class 3, non-seismic.	V-8,
1n	DELETED	
1o	Lines DCB-1 and DCB-2 are nuclear class 3, according to Bechtel Specification M-190	XIX-4, XIX-7,
1p	The RHRSW backwash line (GBD-62 and GBD-63) is non-seismic, according to Bechtel Specification M-190, Sheet 23A.	XXV-1, XXV-4,
1q	Portions of this system which supply suction for HPCI, RCIC, and Core Spray from the condensate storage tank are seismic category I.	XXXIV-2, XXXIV-3,
1r	The return line to the condensate storage tank classified as "Q" by Bechtel in the Q-list (Ref. 225) and was built that way by Bechtel. However, these lines are actually Quality Group D, with no QA requirement. That is the way these lines are classified in this table.	XI-3, XII-3,
1s	The Bechtel Q-list (Ref. 225, item 2.4365) notes this item as Q. However, the entry refers to Bechtel Specification M-119 (Reference 252). This document addresses Seismic Category I supports only. Therefore, only the supports in this item have a requirement for quality assurance and are Seismic Category I.	XXVI-2, XXVII-2,
1t	Bechtel Q-list (Ref. 225, item 2.1510) notes this item as Q. However, only pipe hangers and supports provide a specification (M-119, Ref. 252) as a reference. Therefore, only pipe hangers and supports for this item are seismic category I and have special Quality Assurance requirements.	XIX-1, XIX-2, XIX-4, XIX-5, XIX-7, XIX-8,
1u	See GE document NEDC-31853 (Ref. 2) "Duane Arnold Design Safety Standards", Appendix A, for stress and deformation limits for this item.	I-5, I-6, I-9,
1v	DELETED	
1w	This item includes the 24 volt D.C. Power System.	VI-3,

TABLE 3.2-1
DAEC Classification of Components in System
Footnotes

Footnote #	Footnote	Systems
1x	The recirc pump motors are not seismic category I. However, per 21A9213 sections 4.1.8.3.5 & 4.1.6, they must be seismically supported so as not to breach the integrity of the reactor coolant pressure boundary.	III-6,
1y	De-icing lines located in the Intake Structure are non-seismic Category I.	XXIV-1, XXV-1, XXXVII-1,

TABLE 3.2-1
DAEC Classification of Components in System
Comments

Sys#	Comp#	Comments	
IV	4	1	The design and construction specifications for the hydraulic control unit (HCU) do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example: (1) all welds are liquid-penetrant inspected; (2) all socket welds are checked for minimum engagement and end gap between pipe and socket bottom by a marking technique; (3) all welding was performed by qualified welders; (4)all work was done per written procedures. The following examples are typical of the problems associated with codes designed to control field-assembled components when applied to the design and production of factory fabricated specialty components: **1. The HCU nitrogen gas bottle is a spun forging that is mechanically joined to the accumulator. It stores the energy required to scram a drive at low vessel pressure. It has been code stamped since its introduction in 1966, although its size exempts it from mandatory stamping. It is constructed of a material listed by the ASME B&PV Code, Section VIII, that was selected for its strength and formability. **2. The scram accumulator is joined to the HCU by a split flange joint chosen for its compact design to facilitate both assembly and maintenance. Both the design and construction conform to the B31.1.0 piping code. This joint, which requires a design pressure of 1750 psig, has been proof tested to 10,000 psi. **3. The accumulator nitrogen shutoff valve is a 6,000 psi cartridge valve whose copper alloy material is listed in the ASME B&PV Code, Section VIII. The valve was chosen for this service partly because it is qualified by the U.S. Navy for submarine service. **4. The directional control valves are solenoid pilot-operated valves that are subplate mounted on the HCU. The valve has a body specially designed for the HCU, but the operating parts are identical to a commercial valve with a proven history of satisfactory service. The pressure retaining parts are stainless steel alloys chosen for service, fabrication and magnetic properties. The manufacturer cannot substitute a code material for that used for the solenoid core tube. The foregoing examples are not meant to justify one pressure integrity quality level or another, but to demonstrate that the codes and standards invoked by those quality levels are not strictly applicable to special equipment and part designs. Group D classification is generally applicable because the codes and standards invoked by that classification contain clauses that permit the use of manufacturer's standards and proven design techniques that are not explicitly defined within those codes. This was supplemented by the quality control techniques described above.
IV	4	2	Bechtel built these items to ASME Class 2 standards, based on the piping classes given on the P&ID (see paragraph 3.2.7). However, based on the QA classification of these items, they should have been Class 3. The higher class is shown on this table, although Class 3 is justifiable and would make more sense with the QA Group D designation.
IV	8	1	The design and construction specifications for the hydraulic control unit (HCU) do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example: (1) all welds are liquid-penetrant inspected; (2) all socket welds are checked for minimum engagement and end gap between pipe and socket bottom by a marking technique; (3) all welding was performed by qualified welders; (4)all work was done per written procedures. The following examples are typical of the problems associated with codes designed to control field-assembled components when applied to the design and production of factory fabricated specialty components: **1. The HCU nitrogen gas bottle is a spun forging that is mechanically joined to the accumulator. It stores the energy required to scram a drive at low vessel pressure. It has been code stamped since its introduction in 1966, although its size exempts it from mandatory stamping. It is constructed of a material listed by the ASME B&PV Code, Section VIII, that was selected for its strength and formability. **2. The scram accumulator is joined to the HCU by a split flange joint chosen for its compact design to facilitate both assembly and maintenance. Both the design and construction conform to the B31.1.0 piping code. This joint, which requires a design pressure of 1750 psig, has been proof tested to 10,000 psi. **3. The accumulator nitrogen shutoff valve is a 6,000 psi cartridge valve whose copper alloy material is listed in the ASME B&PV Code, Section VIII. The valve was chosen for this service partly because it is qualified by the U.S. Navy for submarine service. **4. The directional control valves are solenoid pilot-operated valves that are subplate mounted on the HCU. The valve has a body specially designed for the HCU, but the operating parts are identical to a commercial valve with a proven history of satisfactory service. The pressure retaining parts are stainless steel alloys chosen for service, fabrication and magnetic properties. The manufacturer cannot substitute a code material for that used for the solenoid core tube. The foregoing examples are not meant to justify one pressure integrity quality level or another, but to demonstrate that the codes and standards invoked by those quality levels are not strictly applicable to special equipment and part designs. Group D classification is generally applicable because the codes and standards invoked by that classification contain clauses that permit the use of manufacturer's standards and proven design techniques that are not explicitly defined within those codes. This was supplemented by the quality control techniques described above.
IV	8	2	Bechtel built these items to ASME Class 2 standards, based on the piping classes given on the P&ID (see paragraph 3.2.7). However, based on the QA classification of these items, they should have been Class 3. The higher class is shown on this table, although Class 3 is justifiable and would make more sense with the QA Group D designation.

TABLE 3.2-1
DAEC Classification of Components in System
Comments

Sys#	Comp#	Comments	
IV	9	1	The design and construction specifications for the hydraulic control unit (HCU) do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example: (1) all welds are liquid-penetrant inspected; (2) all socket welds are checked for minimum engagement and end gap between pipe and socket bottom by a marking technique; (3) all welding was performed by qualified welders; (4)all work was done per written procedures. The following examples are typical of the problems associated with codes designed to control field-assembled components when applied to the design and production of factory fabricated specialty components: **1. The HCU nitrogen gas bottle is a spun forging that is mechanically joined to the accumulator. It stores the energy required to scram a drive at low vessel pressure. It has been code stamped since its introduction in 1966, although its size exempts it from mandatory stamping. It is constructed of a material listed by the ASME B&PV Code, Section VIII, that was selected for its strength and formability. **2. The scram accumulator is joined to the HCU by a split flange joint chosen for its compact design to facilitate both assembly and maintenance. Both the design and construction conform to the B31.1.0 piping code. This joint, which requires a design pressure of 1750 psig, has been proof tested to 10,000 psi. **3. The accumulator nitrogen shutoff valve is a 6,000 psi cartridge valve whose copper alloy material is listed in the ASME B&PV Code, Section VIII. The valve was chosen for this service partly because it is qualified by the U.S. Navy for submarine service. **4. The directional control valves are solenoid pilot-operated valves that are subplate mounted on the HCU. The valve has a body specially designed for the HCU, but the operating parts are identical to a commercial valve with a proven history of satisfactory service. The pressure retaining parts are stainless steel alloys chosen for service, fabrication and magnetic properties. The manufacturer cannot substitute a code material for that used for the solenoid core tube. The foregoing examples are not meant to justify one pressure integrity quality level or another, but to demonstrate that the codes and standards invoked by those quality levels are not strictly applicable to special equipment and part designs. Group D classification is generally applicable because the codes and standards invoked by that classification contain clauses that permit the use of manufacturer's standards and proven design techniques that are not explicitly defined within those codes. This was supplemented by the quality control techniques described above.
V	1	1	The standby liquid control storage tank is designed, fabricated, inspected, and tested to meet the intent of API Standard 650 and the ASME B&PV Code, Section VIII, Division 1. All butt welds are given spot radiographic examination. Liquid-penetrant inspection is conducted per the ASME Code, Section VIII, Division 1, on the following welds: **(1)All tank nozzle welds below and including the overflow nozzle are examined internally and externally to the tank. **(2)All fillet and socket welds receive a random examination.
V	1	2	The construction of the accumulator is in accordance with the requirements of the ASME B&PV Code, Section VIII, Division 1. An ASME stamp is required. Other codes applied to the accumulator are as follows: **(1)ANSI B16.11 "Forged Steel Fittings, Socket Welded and Threaded". **(2)AND 10050 "Bosses, Standard Dimensions for Gasket Seal Straight Thread".
VI	3	1	See DAEC letter to the USNRC, NG-91-2652, dated 8/27/91, for inclusion of the 24 V D. C. Power Supply with this item.
XI	9	1	The HPCI turbine is categorized as machinery and thus does not fall within the classification groups as earlier identified. To ensure that the turbine was fabricated to the standards commensurate with its performance requirements, General Electric has established specific design requirements for this component, as follows: **(1)All welding was qualified in accordance with Section IX of the ASME B&PV Code. **(2)All pressure retaining castings and fabrications were hydrotested to 1.5 x design pressure. **(3)All high pressure castings were radiographed according to ASTM E-94 (20% coverage, minimum), ASTM E-142 (severity level 3), ASTM-71, ASTM 186, or ASTM-280. **(4)As-cast surfaces were magnetic particle or liquid-penetrant tested according to the ASME B&PV Code, Section III, 1968 Edition, paragraph N323.3 or N323.4. **(5)Wheel and shaft forgings were ultrasonically tested according to ASTM A388. **(6)Butt welds were radiographed according to the ASME B&PV Code, Section III, 1968 Edition, paragraph N624, and magnetic particle or liquid penetrant tested according to ASME Section III, paragraph N626 or N627. **(7)Notification made on any major repairs and records maintained. **(8)Record system and traceability according to the ASME B&PV Code, Section III, 1968 Edition, IX-225. **(9)Control and identification according to the ASME B&PV Code, Section III, 1968 Edition, IX-226. **(10)Procedures conform to the ASME B&PV Code, Section III, 1968 Edition, IX-300. **(11)Inspection personnel are qualified according to the ASME B&PV Code, Section III, 1968 Edition, IX-400. (APED-A61-052, Comment 10)
XII	9	1	The RCIC turbine is categorized as machinery and thus does not fall within the classification groups as earlier identified. To ensure that the turbine was fabricated to the standards commensurate with its performance requirements, General Electric has established specific design requirements for this component, as follows: **(1)All welding was qualified in accordance with Section IX of the ASME B&PV Code. **(2)All pressure retaining castings and fabrications were hydrotested to 1.5 x design pressure. **(3)All high pressure castings were radiographed according to ASTM E-94 (20% coverage, minimum), ASTM E-142 (severity level 3), ASTM-71, ASTM-186, or ASTM-280. **(4)As-cast surfaces were magnetic particle or liquid-penetrant tested according to the ASME B&PV Code, Section III, 1968 Edition, paragraph N323.3 or N323.4. **(5)Wheel and shaft forgings were ultrasonically tested according to ASTM A388. **(6)Butt welds were radiographed according to the ASME B&PV Code, Section III, 1968 Edition, paragraph N624, and magnetic particle or liquid penetrant tested according to ASME Section III, paragraph N626 or N627. **(7)Notification made on any major repairs and records maintained. **(8)Record system and traceability according to the ASME B&PV Code, Section III, 1968 Edition, IX-225. **(9)Control and identification according to the ASME B&PV Code, Section III, 1968 Edition, IX-226. **(10)Procedures conform to the ASME B&PV Code, Section III, 1968 Edition, IX-300. **(11)Inspection personnel are qualified according to the ASME B&PV Code, Section III, 1968 Edition, IX-400. (APED-A61-052, Comment 10)

TABLE 3.2-1
DAEC Classification of Components in System
Comments

Sys#	Comp#	Comments	
XVIII	1	1	Unprocessed liquid radioactive waste piping and equipment pressure parts installed prior to January 1, 1983, were classified as Quality Group C. Unprocessed liquid radioactive waste piping and equipment pressure parts installed subsequent to January 1, 1983, may be included in Quality Group D with added quality control (D+QA) in accordance with the design guidance contained in Regulatory Guide 1.143, Revision 1, modified as follows: **(1)Paragraphs C.1.1.3, C.2.1.3, and C.3.1.3 - The commitment is limited to the seismic design methods used in the original construction of the DAEC and is not upgraded to Regulatory Guide 1.143, Revision 1, requirements. **(2)Paragraph C.4.3 - Systems will be fabricated in accordance with good operability, maintenance, and repairability practices. **(3)Paragraph C.6 - All of paragraph C.6 is replaced in its entirety by the following sentence. "All safety related systems or portions of systems shall be designed, fabricated and installed in accordance with Quality Level II requirements."
XVIII	4	1	Unprocessed liquid radioactive waste piping and equipment pressure parts installed prior to January 1, 1983, were classified as Quality Group C. Unprocessed liquid radioactive waste piping and equipment pressure parts installed subsequent to January 1, 1983, may be included in Quality Group D with added quality control (D+QA) in accordance with the design guidance contained in Regulatory Guide 1.143, Revision 1, modified as follows: **(1)Paragraphs C.1.1.3, C.2.1.3, and C.3.1.3 - The commitment is limited to the seismic design methods used in the original construction of the DAEC and is not upgraded to Regulatory Guide 1.143, Revision 1, requirements. **(2)Paragraph C.4.3 - Systems will be fabricated in accordance with good operability, maintenance, and repairability practices. **(3)Paragraph C.6 - All of paragraph C.6 is replaced in its entirety by the following sentence. "All safety related systems or portions of systems shall be designed, fabricated and installed in accordance with Quality Level II requirements."
XIX	4	1	Segments of the RWCU piping have been replaced with IGSCC resistant materials.
XXIII	3	1	The construction codes used for Offgas pipe, pumps, and valves were USAS B31.7; ASME Sections III and VIII; and the Draft Pump and Valve Code. To identify the correct code for a component it is necessary to research the receiving inspection files (File Q2.321). There are two reasons that several codes and dates were applied. The first reason is that the Offgas design was changed during Procurement/Construction. The second reason is that the codes were changing rapidly during the period of the design. The project correspondence which records when the Offgas System construction code was changed is given in APED-N62-076.
XXIII	4	1	The construction codes used for Offgas pipe, pumps, and valves were USAS B31.7; ASME Sections III and VIII; and the Draft Pump and Valve Code. To identify the correct code for a component it is necessary to research the receiving inspection files (File Q2.321). There are two reasons that several codes and dates were applied. The first reason is that the Offgas design was changed during Procurement/Construction. The second reason is that the codes were changing rapidly during the period of the design. The project correspondence which records when the Offgas System construction code was changed is given in APED-N62-076.
XXIII	5	1	The construction codes used for Offgas pipe, pumps, and valves were USAS B31.7; ASME Sections III and VIII; and the Draft Pump and Valve Code. To identify the correct code for a component it is necessary to research the receiving inspection files (File Q2.321). There are two reasons that several codes and dates were applied. The first reason is that the Offgas design was changed during Procurement/Construction. The second reason is that the codes were changing rapidly during the period of the design. The project correspondence which records when the Offgas System construction code was changed is given in APED-N62-076.
XXIII	6	1	The construction codes used for Offgas pipe, pumps, and valves were USAS B31.7; ASME Sections III and VIII; and the Draft Pump and Valve Code. To identify the correct code for a component it is necessary to research the receiving inspection files (File Q2.321). There are two reasons that several codes and dates were applied. The first reason is that the Offgas design was changed during Procurement/Construction. The second reason is that the codes were changing rapidly during the period of the design. The project correspondence which records when the Offgas System construction code was changed is given in APED-N62-076.
XXIV	1	1	Emergency service water system meets the pressure integrity requirements of Quality Group D, including the additional quality assurance requirements for seismic category I piping. All inspection records will be retained according to the Quality Assurance Program of Chapter 17. These records include data pertaining to the qualification procedures and examination results.
XXXIII	1	1	For Main Steam and Turbine Bypass piping and valves, all inspection records were retained according to the Quality Assurance Program of Chapter 17. These records include data pertaining to the qualification of inspection personnel, examination procedures, and examination results.
XXXIII	1	2	Turbine Stop, Control, and Bypass Valves: A certification was obtained from the vendors of these valves indicating that all cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective have been examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards are used as an alternative to radiographic methods.
XXXIII	1	3	The main steam piping between the outermost containment isolation valves up to the turbine stop valves, the main turbine bypass piping up to the turbine bypass valves and all branch line connected to these portions of the main steam and turbine bypass piping up to the first valve capable of timely actuation are classified as Quality Group D and meet the additional quality assurance requirements for "critical" piping, as stated in Section 17.1.8.1, Schedule IV.
XXXIII	1	4	The first valve capable of timely actuation in branch lines connected to the main steam lines between the outermost containment isolation valves and turbine stop valves and connected to the turbine bypass valves meets all of the pressure integrity requirements of Quality Group D, including the additional quality assurance requirements for "critical" piping, as stated in Section 17.1.8.1, Schedule IV.

TABLE 3.2-1
DAEC Classification of Components in System
Comments

Sys#	Comp#	Comments	
XXXIII	1	5	All inspection records for the main steam and turbine bypass piping and the first valve in the branch lines connected to this piping were retained according to the Quality Assurance Program of Chapter 17. These records include data pertaining to the qualification of inspection personnel, examination procedures, and examination results.
XXXIII	4	1	Materials used in feedwater control valves are as follows: **(1)Valve body is ASTM A105 Gr. II. **(2)Valve bonnet is ASTM A105 Gr. II and A234 Gr. WPB.
XXXIII	4	2	Examination and testing requirements for the feedwater control valves are as follows: **(1)All pressure retaining castings are radiographed, after final heat treatment, in accordance with the ASME B&PV Code, Section III, Appendix IX, paragraph 330 and ASTM E142. Discontinuities are judged by ASTM E71, E186, and E280. **(2)All accessible surfaces of all pressure retaining castings are examined in finished condition, after final heat treatment, by either liquid penetrant methods per paragraph N323.4 or magnetic particle methods per paragraph N323.3, with acceptance criteria per paragraph N323.4 of the Summer 1969 addenda to ASME Section III. **(3)All pressure retaining forgings are examined in the as-furnished condition by the ultrasonic method per paragraph N322 of the Summer 1969 Addenda to ASME, Section III.
XXXIII	5	1	Seismic Adequate Category (A) as added for MSIV-LTS was based on original construction of pipe as "critical" using the same records as seismic category I pipe. A seismic evaluation based on walkdowns and comparative analysis was performed in lieu of a formal seismic analysis.
XXXIII	5	2	The QA Requirements for MSIV-LTS are driven by the SER for Amendment 207 to License No DPR 49 (Docket No. 50-331).
XXXIV	1	1	The condensate storage tank was designed, fabricated, and tested to meet the intent of API Standard 650. In addition, the specifications for this tank require (1) 100% surface examination of the side wall to bottom joint and (2) 100% volumetric examination of the side wall weld joints.
XXXIV	1	2	Page T3.2-5 of the UFSAR (Ref. 233) says that the CST is non-seismic.
XXXV	1	1	The Auxiliary a-c Power System is composed of the 4160 VAC Switchgear, the 480 VAC Load Centers, the 480 VAC Motor Control Centers, and the Instrument AC Control Power System.
XXXVII	1	1	The River Water Supply System meets the pressure integrity requirements of Quality Group D, including quality assurance requirements for seismic category I. Inspection records will be retained according to the Quality Assurance Program of UFSAR Chapter 17. These records include data pertaining to the qualification procedures and examination results.
General		1	B31.1.0 and B31.7 were originally published as USA Standards (USAS), but are now designated as ANSI Standards.
General		2	See additional material examination requirements of Section 17.1.8.1 for piping and valves.
General		3	Code effective date is obtained by the Purchase Order date for the particular component (see Table 3.2-1) or by referring to Table 3.2-2.

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TABLE 3.2-2

CLASSIFICATION AND CODE COMPLIANCE REQUIREMENTS

For items which do not have a specific construction code listed In Table 3.2-1, the following codes, including their addenda and applicable code cases, in effect at the time of component purchase order date, have been applied. In case of a conflict between this table and Table 3.2-1, Table 3.2-1 shall govern.

Safety Class (SC) or Quality Group Classification[c]	Components[d]	Components Ordered Before Jan. 1, 1970	Components Ordered on or after Jan. 1, 1970 and before July 1, 1971	Components Ordered on or After July 1, 1971
SC-1 Group A	Vessels	ASME Section III, '68 Ed., Classes A, C;	ASME Section III, '68 Ed., Class A;	ASME Section III, '71 Ed., Class 1; NA&NB subsections;[h]
	Piping	USAS B31.1.0;[a][l]	USAS B31.7, Class I;[b][k][l][h]	
	Pumps[e] & valves[g]	USAS B31.1.0;[a][l]	ASME NP&VC Class I;[m]	
	Heat Exchangers	TEMA Code[f]	TEMA Code[f]	TEMA Code[f]
SC-2 Group B	Vessels	ASME Section III, '68 Ed., Classes B, C;[i]	ASME Section III, '68 Ed., Classes B, C;[i]	ASME Section III, '71 Ed., Class MC[i] or 2; NA&NC subsections;
	Piping	USAS B31.1.0;[a][l]	USAS B31.7, Class II;[b][k][l]	
	Pumps[e] & valves[g]	USAS B31.1.0;[a][l]	NP&VC Class III;[m]	
	Heat Exchangers	TEMA Code;[f]	TEMA Code[f]	TEMA Code[f]
	Tanks	[j]	[j]	[j]
SC-3 Group C	Vessels	ASME Section VIII, '68 Ed., Div. 1;	ASME Section VIII, '68 Ed., Div. 1;	ASME Section III, '71 Ed., Class 3; NA&ND subsections;
	Piping	USAS B31.1.0;[a][l]	USAS B31.7, Class III;[b][k][l]	
	Pumps[e] & valves[g]	USAS B31.1.0;[a][l]	NP&VC Class III;[m]	
	Heat Exchangers	TEMA Code[f]	TEMA Code[f]	TEMA Code[f]
	Tanks	[j]	[j]	[j]
Power Plant (piping systems) Group D	Vessels	ASME Section VIII, '68 Ed., Div. 1;	ASME Section VIII, '68 Ed., Div 1;	ASME Section VIII, '71 Ed., Div 1;
	Piping	USAS B31.1.0;[a][l]	USAS B31.1.0;[a][l]	USAS B31.1.0;[a][l]
	Pumps[e] & valves[g]	USAS B31.1.0;[a][l]	USAS B31.1.0;[a][l]	USAS B31.1.0;[a][l]
	Heat Exchangers	TEMA Code[f]	TEMA Code[f]	TEMA Code[f]
	Tanks	[j]	[j]	[j]

TABLE 3.2-2

CLASSIFICATION AND CODE COMPLIANCE REQUIREMENTS

a USAS B31.1.0-1967 plus applicable code cases. Requirements of ANSI Nuclear Code Cases N-2, N-7, N-9, and N-10 are applicable for Group A (RCPB) components ordered before January 1, 1970.

b USAS B31.7-1969 plus applicable code cases

c For detailed piping/equipment classification, refer to Table 3.2-1.

d Components required to be stamped to Section III of the ASME B&PV Code are stamped with the applicable ASME Code symbol, and the required third-party inspection was performed by an authorized Inspector.

e For pump designs, the applicable class, section, or subsection of the referenced ASME B&PV Code is used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting. For example, use ASME Section III, Class C for Group A&B pump design guide. For Group D below 150 psig and/or 212°F, Manufacturer's Standard pump for service intended may be used.

f Tubular Exchanger Manufacturer's Association (TEMA) Code Requirements were applied using classes appropriate to each heat exchanger's duty cycle.

g ANSI B16.5 or MSS-SP-66 apply for valves (note MSS-SP-66 1964 was withdrawn from publication in favor of ANSI B16.34-1973).

h Class I nuclear piping, pumps, and valves purchased after January 1, 1970, will meet the provision of ASME B&PV Code Section III, paragraph N-153 for stamping and third-party inspection.

i Metal containment vessel and penetrations (extensions of containment) are ASME, Section III, stamped Class B or MC (subsection NA&NE), and the required third-party inspection shall be performed by an authorized inspector.

j Existing API/AWWA standards and supplementary requirements apply. Tanks are to be designed, fabricated, constructed, and tested to meet the intent of API Standards 620 (Recommended Roles for Design and Construction of Large Welded Low Pressure Storage Tanks) or 650 (Welded Steel Tanks for Oil Storage) or AWWA Standard D100 (standard for steel tanks, stand pipes, reservoirs, and elevated tanks for water storage) for these fuel, oil, or water storage tanks.

k ANSI B31, Code Case 78 applies for B31.7 Class I and Class II pipe and fittings 3/4 inch nominal pipe size and smaller.

l These codes were originally published as USA Standards (USAS), but are now designated as ANSI Standards.

m ASME Code for Pumps and Valves for Nuclear Power (or Nuclear Pump and Valve Code). This was incorporated into ASME Section III after the construction of DAEC.

TABLE 3.2-3

SEISMIC CATEGORY I STRUCTURES

Reactor Building

Drywell (including reactor vessel pedestal)

Wetwell (torus)

Control Building

Intake Structure

Turbine Building (portion containing emergency diesel generators)

Pump House (portion containing residual heat removal and emergency service water systems)

Offgas Stack

Notes:

(a) Structures, systems, and components not listed as Seismic Category I in Table 3.2-1 or above are nonseismic.
(b) 10 CFR 50 Appendix B QA program is applied to Seismic Category I structures.

TABLE 3.2-4

DESIGN REQUIREMENTS FOR SAFETY CLASSES 2 AND 3 ELECTRIC SYSTEMS AND COMPONENTS

PROTECTION				CLASS 1E								
Components	Modules	Sensors	Systems[a]	Cable	Connectors	Switch Gear	Transformers	Diesel	Systems	Motors	Valve Actuators	Penetrations
IEEE-323	IEEE-323	IEEE-323	IEEE-279	(b)	(b)	IEEE-344	IEEE-344	(b)	IEEE-308	IEEE-323	IEEE-323	IEEE-344
										IEEE-334(c)		
	IEEE-344	IEEE-344								IEEE-344		

Notes:

(a) IEEE-279 shall apply only to those Safety Class 2 or 3 systems and components which actuate reactor trip or, in the event of a serious reactor accident, actuate engineered safeguards.

(b) Design requirements had not been developed for this Condition of Design by the applicable code at the time DAEC was designed. Design requirements are to be developed for the specific component.

(c) GE Scope of Supply

TABLE 3.2-5

SUMMARY OF SAFETY CLASS DESIGN REQUIREMENTS

	Safety Class				
Design Requirements	1	2	3	Other	
Quality Group Classification[(a)]	A	B	C or D + QA	D + QA	C or D
Quality Assurance Requirement[(b)]	B	B	B	B	B or D
Seismic Category[(c)]	I	I	I	NA	I or NA

Notes:

(a) The equipment shall be constructed in accordance with the indicated code listed in Table 3.2-1 or 3.2-2.

(b) B – The equipment shall be constructed in accordance with the quality assurance requirements of 10CFR50 Appendix B and the Quality Assurance Program described in Chapter 17.

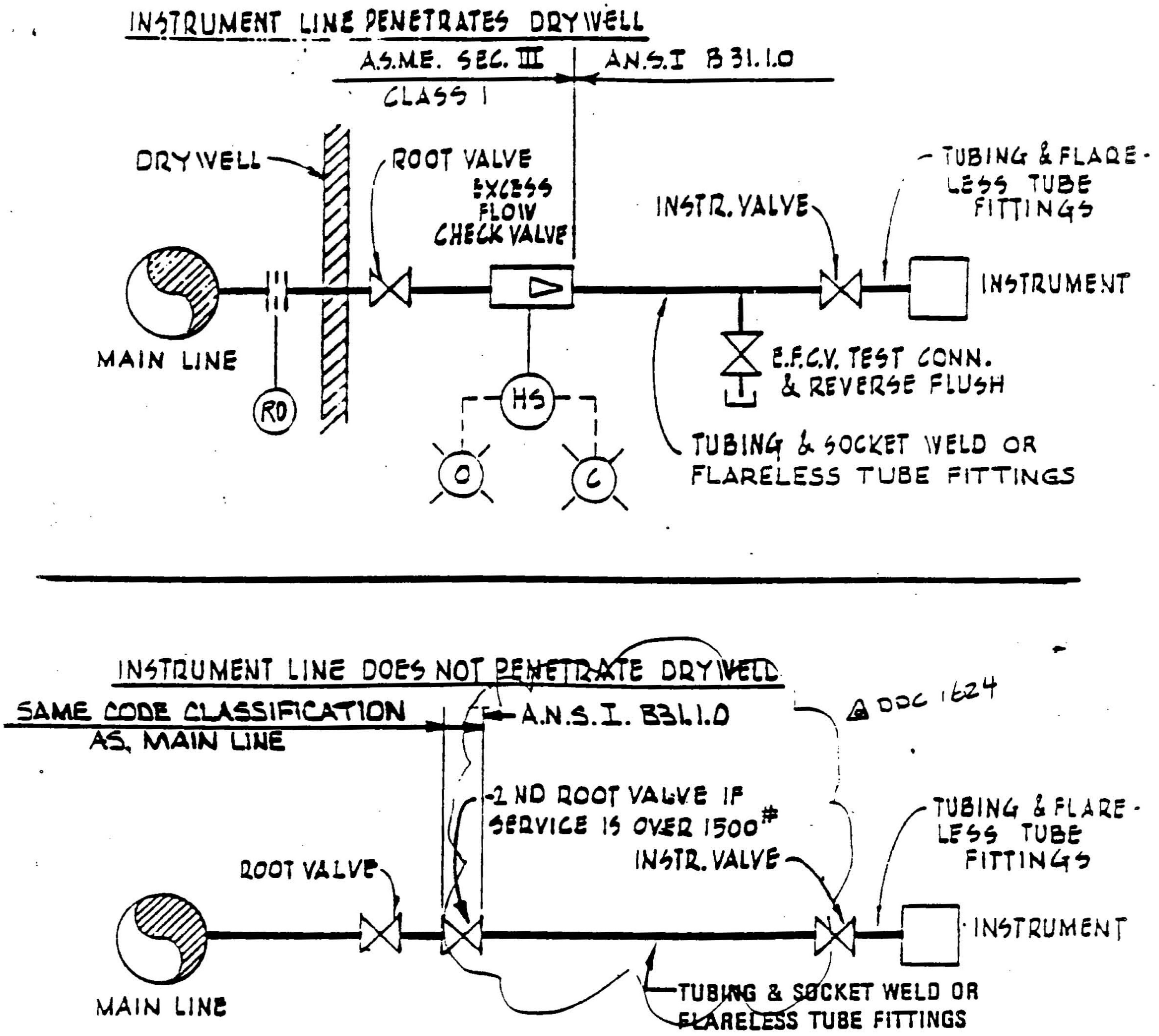
D – The equipment shall be constructed in accordance with the Quality Assurance requirements consistent with good practice for steam power plants.

(c) I – The equipment is this seismic category shall be constructed in accordance with the seismic requirements of the safe shutdown earthquake as described in Subsection 3.2.1 and Section 3.7.

NA – The seismic requirements for the safe shutdown earthquake are not applicable to the equipment of this classification.

[redacted]

INSTRUMENT PROCESS LINE CODE CLASSIFICATION



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Instrument Process Line Code Classification

Figure 3.2-2

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

All structures are designed for wind loads in accordance with ASCE Paper 3269 "Wind Forces on Structures."[1] Wind pressure as it applies to the structures is as follows (See also Section 2.3):

			Wall Load		
Height (ft)	Basic Velocity (mph)	Dynamic Pressure (Including 1.1 Gust Factor) q (psf)	Pressure 0.9 q (psf)	Suction 0.4 q (psf)	Roof Load Suction 0.6 q (psf)
0-50	105	34	31	14	20
50-150	125	48	43	19	29
150-400	145	65	59	26	39

Whenever wind loads are combined with other loads, a 33% increase in allowable stresses is permitted.

3.3.2 TORNADO LOADINGS

3.3.2.1 Applicable Design Parameters

The design-basis tornado consists of a tornado with a maximum tangential velocity of 300 mph as shown in Figure 3.3-1, traveling with a maximum transverse velocity of 60 mph. The loadings created by the design-basis tornado are reflected in the following two tornado design criteria used in the design of tornado-resistant structures:

1. The velocity components are applied as a uniform 300 mph wind on the structure.

2. The pressure differential is applied as a 3 psi positive (bursting) pressure occurring in 3 sec.

Although the design-basis tornado consists of a tornado with a maximum transverse velocity of 60 mph, the tornado phenomenon is so complex that considerable engineering judgment is necessary in the determination of a wind-loading criterion. Therefore, the design-basis tornado velocity components are conservatively applied as a 300 mph wind on the structure using the applicable portions of the wind design methods described in ASCE 3269[1] particularly for shape factors. Variation of wind velocity with height is not used. The average wind-loading approach is chosen since it is almost impossible for a structural designer to apply a vortex-shaped loading on a building.

Shape factors from ASCE Paper 3269 are used since they are the best currently available information, even though velocities as high as 300 mph are not considered in the paper. The radial and vertical components are neglected since they act parallel to the walls of the structure. In order to determine the conservatism of this approach, the tornado criteria have been checked for various critical structures against the design-basis tornado (an extrapolation to the design velocities of Hoecker's[2] velocity profiles derived from movies of the Dallas tornadoes of April 2, 1957). The average wind loading on the DAEC reactor building due to the design-basis tornado with a transverse velocity of 60 mph is approximately that due to a 215-mph wind, which is 85 mph below the design wind loading. Figure 3.3-2 shows a comparison of the tornado wind loading criteria to the design-basis tornado superimposed on a reactor building wall.

The tornado model used on past jobs specifies a maximum tangential velocity of 300 mph and a maximum translational velocity of 60 mph as shown in Figure 3.3-1. The tangential velocities vary with height and distance from the tornado center. When this tangential velocity distribution is superimposed on the reactor building, and the wind loads integrated over the surface of the structure, the resultant average wind load is about 155 mph. The 215-mph wind loading mentioned accounts for the 60-mph additional force due to the translational velocity. The above forces represent the actual loads on the structure if the tornado model actually struck the building. The actual loading condition is shown in Figure 3.3-2 by the solid lines.

The above discussion is provided only as a justification of the 300-mph wind load criterion. The tornado model with a 300-mph maximum velocity component results in an average wind load of 215 mph. This 215 mph is considered to be a realistic value of the effective wind force on the structure. The increase in the criterion to 300 mph accounts for the possible excess load effects due to model assumptions, assumptions in the structural analysis, simplifications in calculations, and effects of construction sequence and methods. Since wind pressure is a function of velocity squared (V^2), the increase from 215 to 300 mph results in a pressure increase from 120 to 230 psf. Thus, the margin of safety implied by the increase in loading criterion is about 2.0. In addition, an increase in design loads is provided by simultaneously applying a differential (bursting) pressure due to the 3-psi pressure drop. This pressure adds to the suction force of the wind to provide one of the critical loading conditions. The wind and depressurization forces are directly combined even though they do not occur simultaneously. This simplification adds to the margin of safety of the structure.

Where failure could affect the operation and functions of the primary containment and reactor primary system, and for structures housing equipment necessary for the safe shutdown of the reactor, the following tornado effects are considered in the design of these structures (see also Sections 2.3 and 3.5.1):

1. External wind forces resulting from a tornado funnel having a horizontal peripheral tangential velocity of 300 mph and a transverse velocity of 60 mph. The wind force on the structure was considered as a static load and calculated on the basis of a 300-mph horizontal wind applied over the full height of the projected area.

2. Differential pressure between inside and outside of fully enclosed areas of 3 psi (bursting). Means are included in the actual design of the structures to limit excessive pressure differentials.

3. Missile equivalent to a 4 in. x 12 in. x 12 ft long wood plank (108 lb) traveling end-on at 300 mph, or a passenger auto (4000 lb) flying through the air at 50 mph and at not more than 25 ft aboveground with a contact area of 20 ft^2.

4. A torsional moment resulting from applying the wind specified in item 1 above on one-half of the structure and a wind velocity equal to one-half that specified in item 1 above applied to the other half of the building in the opposite direction.

Note: The effects of items 1, 2, and 3 were considered as acting simultaneously.

All structures housing equipment necessary for a safe shutdown are designed to withstand a tornado-induced depressurization rate of 1 psi/sec for 3 sec. To accomplish this design objective, all nonvented compartments are checked to verify that they are capable of withstanding an internal (bursting) pressure of 3 psi.

The margin of safety in the tornado wind criteria is in the assignment of the large wind force and the simultaneous loading application. A concrete building is designed using the normal ACI code provisions and methods for ultimate strength design. This specifically includes the appropriate capacity reduction factor (0). The only modification to the ACI provisions is in the assignment of load factors. The load factors for the design equation are always assigned as 1.0.

The implied margin of safety is verified by the lack of structural damage to buildings that have withstood tornadoes. Steel structures are designed using traditional elastic methods of analyses and allowable stresses of 1.5 f_s with 0.9 F_y as the upper limit. This is consistent with the design philosophy of structures under the DBE.

3.3.2.2 Determination of Forces on Structures

Once the loads have been established, the design requires a determination of the pressure coefficients. The forces on a square-sided building are 185 psf pressure and 115 psf suction. The outward pressure due to depressurization ranges from 50 psf to over 400 psf depending on compartment geometry. Thus, the total suction pressure on a wall ranges from 165 psf to over 600 psf. Normal wind loads specify a suction pressure of about 20 psf. Using the load factors of the ACI for normal wind loads and those proposed for the tornado wind loads, the resultant loads are 25 psf (1.25 x 20 psf) and 165 to 600 psf, respectively. The damage of concrete structures by tornadoes is almost nonexistent. Buildings and structures designed for normal wind loads have been in the direct path of tornadoes and sustained no structural damage. Thus by increasing design loads by at least 6.6 times (165/25 = 6.6), the structural integrity of the building is ensured.

For those compartments that are vented, a flow analysis of all air volumes and interconnecting vent areas was performed, and the maximum transient pressure differential across every wall, floor, and roof was calculated using the principles of fluid mechanics to determine its maximum transient pressure differential. Finally, each structural component was checked to ensure that it could withstand the maximum calculated transient pressure differential that it would experience.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

The structural steel frames of the reactor building upper superstructure are designed to withstand the pressure corresponding to a 300-mph wind.

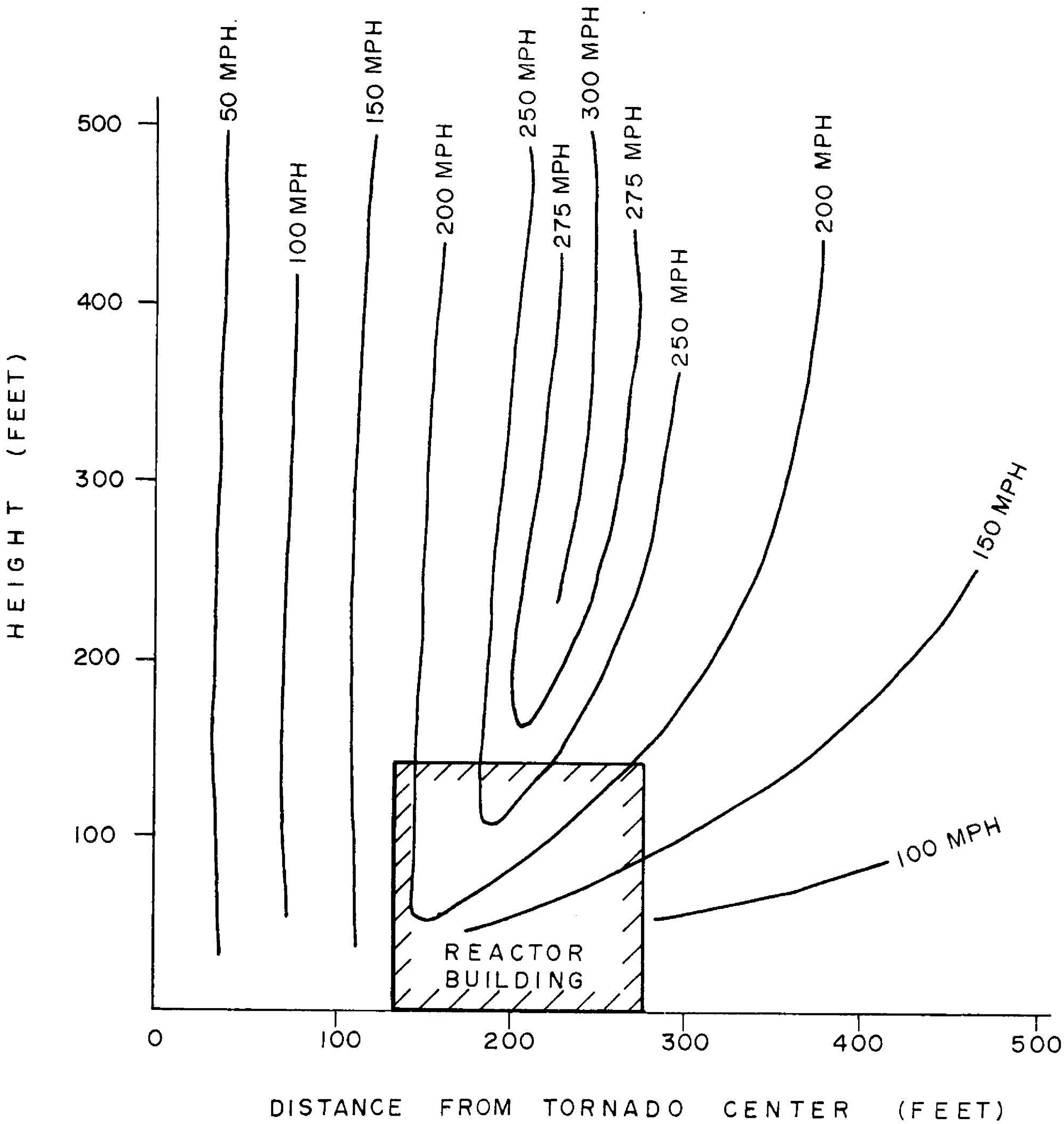
The reactor building siding and roof decking, however, have been designed for the normal wind loading. When this design velocity is appreciably exceeded, the siding and decking may blow off and expose the refueling floor and parts of the reactor building not required for safe shutdown.

If tornado winds traverse the site, the reactor is capable of being shut down and secured in a safe shutdown mode. Superstructure damage could be incurred to the reactor building, turbine building, and incoming power lines without affecting the ability to shut down the reactor, to maintain the integrity of the primary containment, and to provide adequate core cooling. Simultaneous damage to all of these structures is not expected. Components that directly affect the ultimate safe shutdown of the plant are located either under the protection of reinforced concrete or underground.

REFERENCES FOR SECTION 3.3

1. American Society of Civil Engineers, "Wind Forces on Structures Paper No. 3269, Final Report of Task Committee on Loads and Stresses, Structural Division," Transactions ASCE, 1961.

2. W. H. Hoecker, Jr., "Wind Speed and Air Flow Patterns in the Dallas Tornado of April 2, 1957, "Monthly Weather Review, 1960.



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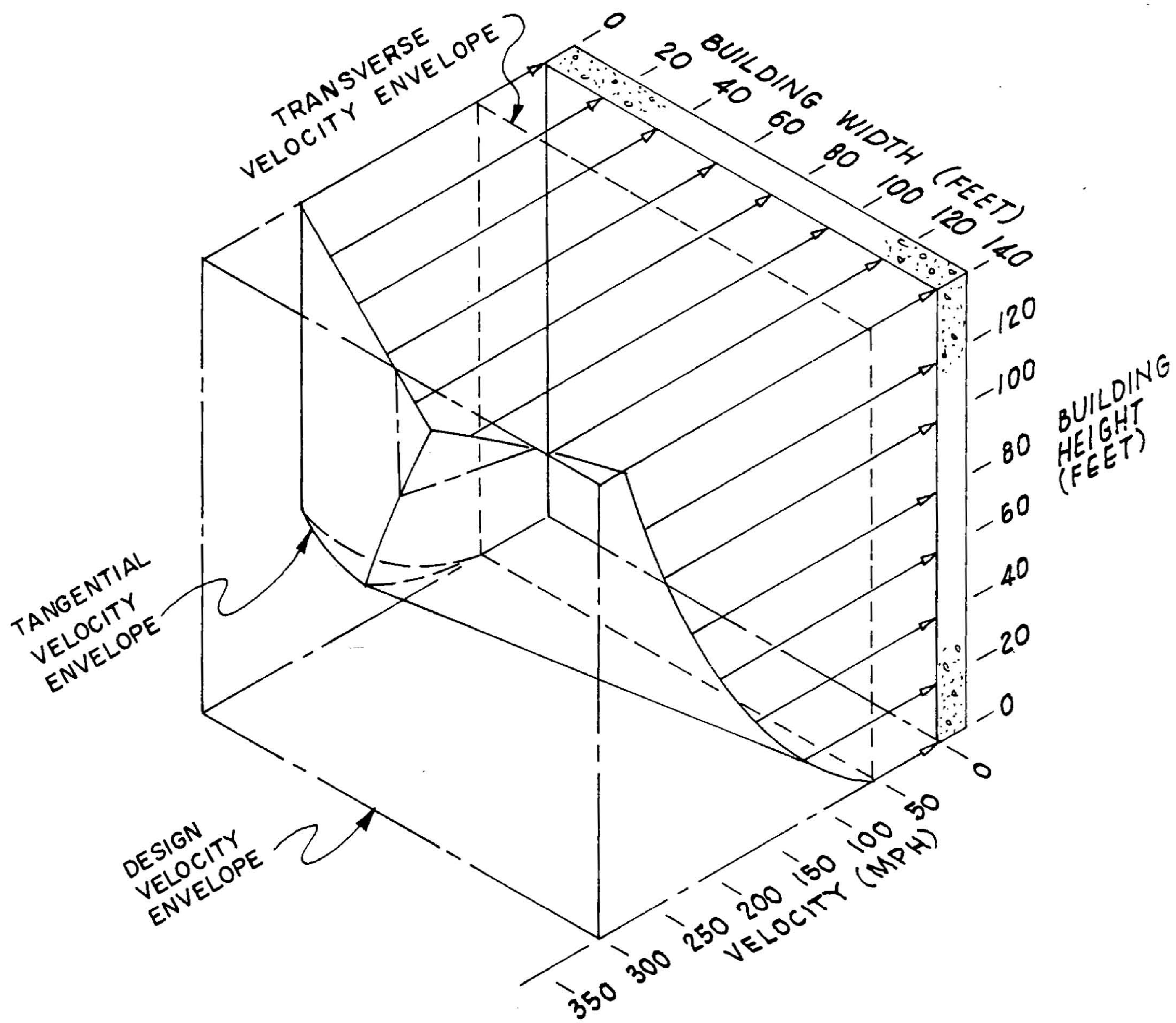
Tangential Velocity Distribution

Figure 3.3-1



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Comparison - Design-Basis Tornado and Tornado Wind Loading on Reactor Building

Figure 3.3-2

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

3.4.1.1 Flood Protection Measures for Seismic Category I and Nonseismic Structures

3.4.1.1.1 Introduction

All Seismic Category I structures and Nonseismic structures housing Seismic Category I equipment are designed to withstand the hydraulic head resulting from the "maximum probable flood" to which the site could be subjected. Under this condition, the water level may reach an elevation of 764.1 ft msl. To allow for wave action and free board, the facility was designed (during the construction permit period of review) to resist flood waters to elevation of 767.0 feet msl. Further review of the wave action and runup caused by winds resulted in additional requirements by the Commission.

Temporary protection for openings in the exterior walls up to the following levels is provided: elevation 770.5 feet msl on the northerly side of safety-related buildings; elevation 773.7 ft msl on the southerly side of safety-related buildings, and to 769 feet msl on all other sides of safety-related buildings. Openings below the flood level are either watertight or are provided with means to control the inflow of water in order to ensure that a safe shutdown can be achieved and maintained.

3.4.1.1.2 Waterproofing

A waterproofing system was used on the exterior surfaces of all Seismic Category I structures below grade requiring protection, see specification for materials (BECH-MRS-C007-S). The system is a fluid-applied membrane that was applied to the specified minimum dry film thickness on vertical wall surfaces below grade. The membrane bonds tightly to the concrete surface to which it is applied, thus preventing lateral migration of water between the membrane and concrete surface in the event it is punctured. Thus, in the event a leak is detected within the structure, only local repair procedures in the area of the leak would be required and the integrity of the entire system would not be jeopardized. The membrane surface was protected from puncture by the placement of celotex boards against the walls before backfill placement.

Joints between adjacent structures were protected by the embedment of a 6-in. center bulb-type water stop that was run continuously below grade. The waterproof membrane was applied up to and across the water stop on both structures. All exterior wall surfaces of the reactor building were protected to just below grade with waterproof membrane. In addition to the 6-in. center bulb water stop between structures, a 1-in. minimum silicon rubber base sealant, General Electric, No. 1300, was applied as a secondary backup protection system. The integrity of the watertightness in this area may be checked by vertical access from above.

Pipe and conduit penetrations through the exterior walls below elevation 757 ft 0 in. were coated with the waterproof membrane system for a distance of 1 ft from the structure. At all pipe conduit and concrete construction joints, the minimum dry film thickness was increased to 100 mils for a distance of 1 ft from the joint.

3.4.1.1.3 Design Criteria

The following criteria were investigated in order to establish flood protection methods to be applied to all structures housing Seismic Category I equipment in the event of a maximum probable flood.

1. The structural safety of all buildings for the resulting hydrostatic loading.
2. An inventory of all openings in the buildings below elevation 769.0 ft.*
3. Modifications to buildings required to withstand the hydrostatic loading and/or methods for closing openings below elevation 769.0 ft.*

Flood data were obtained from a report prepared by Commonwealth Associates, Inc., Jackson, Michigan, on the probable maximum flood of the Cedar River near Palo, Iowa (Appendix H of the DAEC PSAR). The probable maximum flood discharge was determined to be 316,000 cfs and to have a corresponding peak stage of elevation 764.1 ft msl. The flood would result from meteorological conditions that could occur during late winter or early spring and would reach maximum river level in about 6.4 days after the beginning of the storm. The maximum flood of record at the site occurred in 1961 and rose to elevation 746.5 ft. The "Standard Project Flood" as determined by the U.S. Army Corps of Engineers would flood the plant site to elevation 754.5 ft.

The site natural grade level in the vicinity of the plant varies from about elevation 746.0 ft to elevation 750.0 ft. As a consequence of the "Standard Project Flood," the plant site finished grade is at elevation 757.0 ft.

Major equipment penetrations in the exterior walls are located above elevation 767 ft. Personnel doors and railroad and truck openings at or near grade would require protection in the event of a flood above elevation 757.0 ft. All structures have been designed in accordance with the provision for Ultimate Strength Design ACI-318-63 to withstand the hydrostatic loadings resulting from the flood conditions. The hydrostatic load was treated as a dead load using the following load factors:

1.5 x OL for high water level at elevation 757.0 ft
1.0 x OL for high water level at elevation 767.0 ft

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* As discussed in the DAEC SE (by Directorate of Licensing, U.S.A.E.C) dated January 23, 1973, protection had been provided to elevation 769.0 ft using stop logs at the accesses to safety-related buildings. The Commission requested the DAEC to provide flood protection to elevation 770.5 feet msl on the northerly side of safety-related buildings; to 773.7 on the southerly side of safety-related buildings; and to 769 feet msl on all other sides of safety-related buildings. The protection consists of stop-logs, augmented with plastic sheeting to be held in place with sand bags to reduce inleakage.

All buildings were also checked against uplift (buoyancy) for a flood level at elevation 767.0 ft, and the minimum factor of safety used was 1.2.

3.4.1.1.4 Plant Structures

The plant buildings that were reviewed for the maximum probable flood of elevation 767.0 ft are the following:

1. Reactor building (including HPCI structure).
2. Turbine building.
3. Intake structure.
4. Control building.
5. Radwaste building.
6. Pump house.
7. Recombiner room.
8. Low-level Radwaste Processing and Storage Facility (Storage Portion).

The arrangement of the structures on the site is shown in Figure 1.2-1.

All stoplogs, caulking, and bracing required are maintained at the site. As approximately 6.4 days exist from the start of the storm to maximum flood stage, enough time exists to make all flood preparations.

3.4.1.1.4.1 Reactor Building

This building is a reinforced-concrete structure from the foundation [redacted] Grade around the building is generally at elevation 757.0 ft. The building has a factor of safety against buoyancy (considering dead loads and equipment loads only) of 2.0.

There are no openings below elevation 757 ft 6 in. that require protection against flooding. The following access doors require protection against flooding:

1. [redacted]

2. [redacted]

[redacted] will be provided with stoplogs to prevent the flooding of the structure. Additional caulking and temporary bracing of gap material will be provided because of the 1-in. gap between structures adjoining the reactor building. For the north side the 1 inch gap is next to [redacted] and for the south side the 1 inch gap is next to [redacted]. A waterstop is installed in the reactor and turbine buildings' foundation walls, which will prevent water from entering the gap between the two buildings. There are no ducts that exit the structure below elevation 767.0 ft. Piping penetrating the exterior walls below elevation 757.0 ft is embedded in the concrete with a ring plate that ensures against water seepage. Above elevation 757.0 ft, all piping is caulked in the wall and some minor seepage may be expected. Minor seepage from both piping and at doors may be easily controlled by sump pumps at the mat elevation and through the use of additional portable water pumps. Typical stoplog arrangements are shown in Figure 3.4-1.

3.4.1.1.4.2 <u>Turbine Building</u>

The substructure of this building is of reinforced concrete from the [redacted] and precast concrete panels [redacted] The building has a factor of safety against buoyancy (considering dead loads and equipment loads only) of 1.3.

There are no openings below elevation 757 ft 6 in. that would require protection against flooding. The following access doors require protection against flooding:

1. [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

2015-011 [redacted] will be protected by a [redacted] will be provided with stoplogs to prevent the flooding of the building. [redacted] will be provided with additional caulking and temporary bracing of gap material between the turbine and control buildings. There are no ducts that exit below elevation 767 ft 0 in. Piping penetrations are similar to that in the reactor building. Minor seepage will be controlled by sump pumps and through the use of additional portable water pumps.

[redacted] This penetration is gasketed to ensure no water seepage into [redacted] The transformers can be allowed to flood, after the lower fans have been removed and control cabinets are sealed and braced.

[redacted] The space between the 24-in.- diameter pipe and 36-in.- diameter sleeve will be caulked and braced.

3.4.1.1.4.3 Intake Structure

The intake structure is basically a chambered box of reinforced concrete from its foundation at [redacted] to the top deck [redacted] Grade around the building is at elevation 750 ft 0 in.

Seismic Category I equipment contained within the intake structure is located above the peak stage of the flood at [redacted] Therefore, no flood protection is required. The structure is allowed to be flooded resulting in no uplift.

3.4.1.1.4.4 Control Building.

[redacted] The stresses in the walls resulting from hydrostatic loading are very low, and the building has a factor of safety against buoyancy of 1.8.

The control room is located above the maximum flood level [redacted] with access to the administration building will be provided with a stoplog to prevent the battery rooms and switchgear rooms from flooding.

3.4.1.1.4.5 Radwaste Building

The radwaste building is a reinforced-concrete structure with top of mat [redacted] Because of possible contamination of flood waters with radwaste, this structure will be maintained in a dry condition. The stresses in the walls resulting from hydrostatic loading are very low, and the building has a factor of safety against buoyancy of 2.1.

Minor seepage will be controlled by sumps located in the building and the water contained in the radwaste system.

The 1-in. gap between the radwaste building and the reactor building will be provided with additional caulking and temporary bracing of gap material.

3.4.1.1.4.6 Pump House

The pump house is a reinforced-concrete structure. [redacted] The building has a factor of safety against buoyancy (considering dead loads and equipment loads only) of 1.6.

1. [redacted]

[redacted] [redacted]

3.4.1.1.4.7 Recombiner Room

[redacted] which will be provided with stoplog protection against flood waters. A portable water pump will be available to protect against minor seepage at the door opening. Stresses and safety against buoyancy are very conservative.

3.4.1.1.4.8 Diesel-Generator Rooms

Provisions are made to keep the entire [redacted] dry during the maximum probable flood. Thus the interior walls of the diesel-generator rooms are not required to be watertight. Power and control cables from the diesel-generator room to the control building are routed in conduits embedded under the [redacted] floor. Like any embedded conduit or duct bank, these conduits are not considered waterproof but the cable routed through them is designed and manufactured for service in indoor and outdoor duct banks and for direct burial in wet and dry locations. Standard factory tests include moisture absorption tests and immersion tests.

3.4.1.1.4.9 Low-level Radwaste Processing and Storage Facility (LLRPSF)

The storage section of the LLRPSF is a reinforced concrete structure with top of mat [redacted] To prevent possible contamination of flood waters, this portion of the facility is protected against the maximum probable flood. The stresses in the walls resulting from hydrostatic loading are very low, and this section of the building has a factor of safety against buoyancy of 2.6.

The following doors have been provided with steel stoplogs to prevent flooding of this section of the facility:

1. [redacted]

[redacted] [redacted]

[redacted] [redacted]

Minor seepage into the facility will be detected and monitored by the sump system located in this portion of the facility.

A waterproof membrane is installed for all exterior wall construction joints and at all corners below elevation 767'-6". In addition, a continuous waterstop to provide a watertight boundary between the existing radwaste building and the storage portion of the facility has been installed below elevation 767'-6". Finally, the gap between the storage section and the existing radwaste building is filled with ethafoam.

Protection against the maximum probable flood is not considered necessary for the processing section of the LLRPSF. There will be a 6.4-day period before flood peak. This will provide time to move [redacted] into an area protected against the maximum probable flood.

3.4.1.1.5 Roof Structures

In the event of rains as severe as those that could cause a local probable maximum flood, some local flooding could occur because of the fact that the site storm drainage system is designed to accommodate the runoff from a 10-yr storm, but this flooding will have no adverse effect on any safety-related structures or equipment.

A review of the DAEC structural design has shown that all safety-related structures are capable of supporting a water accumulation on their roofs to the depth of the parapet without failure. Above this depth, the water will spill over the parapet and down the side of the building.

All roof penetrations on all plant safety-related structures extend above the roof to a height greater than the height of the parapet, thus precluding any flooding of the interior of any building from excessive precipitation. Therefore, it may be concluded that the probable maximum precipitation storm will not cause the failure of any safety-related structures or equipment.

The emergency diesel-generator air intakes are located [redacted] are watertight at the penetrations and they ensure that water will not accumulate on the roof and flood the emergency diesel-generator air intakes.

3.4.1.1.6 Emergency Procedures

Emergency measures to be taken to protect safety-related structures and equipment from the consequences of a probable maximum flood and coincident wave action are discussed in Section 3.4.1.1.4.

To eliminate possible leakage at plant access openings during periods when flood protection is required, sheet plastic held in place with sandbags will be used in addition to the stoplogs shown in Figure 3.4-1.

The Technical Requirements Manual includes the requirement to begin plant shutdown when, under severe flood conditions, the stillwater level reaches elevation 757 ft (plant grade).

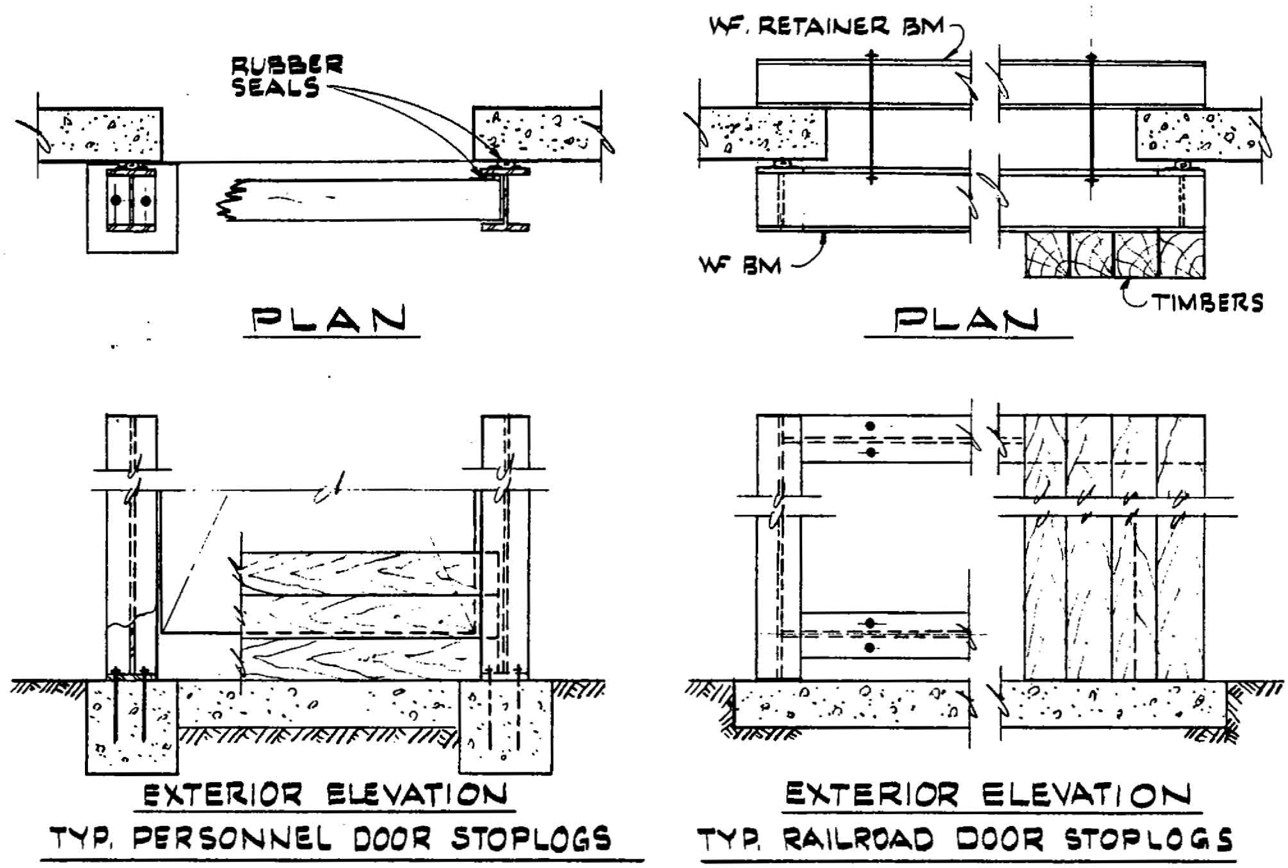
It should be noted, however, that the plant, including its safety-related components, is in fact protected to an elevation of at least 769 ft. In view of this fact, DAEC will consult with the NRC before any required shutdown to determine whether shutdown should be implemented, considering all relevant circumstances, including the need for power, which may be created by the emergency itself.

3.4.1.2 Permanent Dewatering System

The DAEC does not have a permanent dewatering system.

3.4.2 ANALYTICAL AND TEST PROCEDURES

All Seismic Category I structures and structures housing Seismic Category I equipment are designed to withstand the loads of a probable maximum flood and coincident wind-generated activity. Methods used to incorporate these loads into the design are discussed in Section 3.8.2.3.5.



DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Typical Stoplog Arrangements

Figure 3.4-1

3.5 MISSILE PROTECTION

3.5.1 MISSILE SELECTION AND DESCRIPTION

3.5.1.1 Internally Generated Missiles (Outside Containment)

Wheel average tangential stresses in the HPCI and RCIC turbines are sufficiently low that wheel failure is not predicted even at the theoretical runaway condition of 200% rated speed. Therefore, failure of these turbine wheels is considered so improbable as to be of no consequence with respect to becoming a potential missile or affecting safe shutdown of the plant. In addition, the HPCI and RCIC are located in separate concrete rooms within the reactor building.

3.5.1.2 Internally Generated Missiles (Inside Containment)

3.5.1.2.1 Recirculation Pump Overspeed Missile

3.5.1.2.1.1 Description

The subject of the effects of destructive overspeed of the recirculation pump following a design-basis LOCA, which is an ACRS concern,[1] is discussed below.

The DAEC drywell piping layout has been examined to determine the effect of missiles formed in the recirculation system following a postulated pipe rupture. The source of these postulated missiles is the breakup of either the recirculation pump motor or the recirculation pump impeller. The effects of the overspeed of the recirculation pump motor could be avoided by means of a decoupling device between the recirculation pump and the pump motor. From the results of an extensive analysis, the probability that any serious damage would result from the pump motor or impeller missiles is considered to be negligibly low.

3.5.1.2.1.2 Effects of Recirculation Pump Overspeed

General Electric has submitted a licensing topical report[2] to the NRC, which contains a discussion of the theoretical potential for damage resulting from missiles originating in a recirculation system pump due to destructive overspeed caused by the pump acting as a turbine following a postulated pipe rupture in the recirculation system. Although the report indicates that destructive overspeed of the pump and motor could occur and missiles resulting from the overspeed could be generated, the probability of these missiles causing-damage is extremely low. The report also indicates that the effects of overspeed of the motor could be avoided by incorporating a decoupling device between the pump and motor, but this has not been done. In order to evaluate the effects of missiles produced by pump overspeed, it is necessary to determine if such missiles could escape the piping system and, if so, to determine whether the missiles would strike safety-related components with enough energy to penetrate or otherwise cause enough damage to impair operation. Such an evaluation was made including a study of the probability of serious damage from the pump missiles. [redacted] The DAEC plant layout has been compared with [redacted] for which the probability of such an event was calculated.

[redacted] and therefore, the results of the study are considered applicable to the DAEC. In addition, it must be noted that the results given in this evaluation are actually conservative if applied to the DAEC for the following reasons:

1. The DAEC has an improved pipe restraint system as compared to the evaluation model[2].

2. The DAEC recirculation pipe diameter of 22 in. is smaller than the evaluation model of 28 in., resulting in smaller missiles and less energy.

Consequently, the results given in Reference 2 represent the bounded values for the DAEC.

3.5.1.2.1.3 Analysis

The probabilistic analysis of the release of radioactive material to the environment due to the formation of recirculation pump missiles involves a hypothetical series of events that include the following:

1. The instantaneous double-ended break of the recirculation line at a specific location followed by,

2. The formation and escape of a missile of very high energy which is,

3. Directed at the containment or a pipe at a very specific angle and,

4. Then impacts at its sharpest edge orientation, thus possibly endangering containment integrity.

This highly unlikely series of events has such a low probability of occurrence that it is comparable to other low- probability events that are considered negligible.

Given the design-basis LOCA, consider the random nature of the subsequent events of a missile-induced pipe break. First, the recirculation-line break must occur at one of several locations out of over 40 locations at which it is equally probable that a break could occur. Next, the pump impeller must break up in a certain way, and a large missile must enter the piping. The missile must then leave the pipe in a certain direction out of a virtually infinite number of possible directions within a cone of dispersion. The missile must not lose energy due to impact with nonvital piping or other objects, or the containment, and must finally strike a pipe at a certain angle and specific orientation. The total probability of damage to piping due to recirculation pump missiles is presented in Table 3.5-1. The unlikely and random occurrence of the missile-induced pipe break is of the same probability as the random failure of a second active component.

The probability of this series of events occurring in conjunction with a LOCA and releasing radioactive material to the environment is summarized in Table 3.5-2. For comparison, the probability of radioactive material release from all other causes is also included.

It can be seen from Table 3.5-2 that the probability of radioactive material release to the atmosphere is only slightly increased when recirculation pump missiles are considered and that the overall probability of release is still very low.

The probability of the dual-break event (recirculation pipe break and containment break or other pipe break) is comparable to the probability of other events normally considered negligible. For example, the probability of the LOCA plus a single failure of the LPCI injection valve plus the break of a core spray line by a randomly directed recirculation pump missile is calculated to the 1.1×10^{-10} per year. The corresponding probability of a LOCA plus a single failure of the LPCI injection valve plus the random failure of another active component in the core spray system is 2.3×10^{-10} per year. Thus, the dual-break probability is smaller than the probability of other random double active component failures.

In summary, the breaking of a vital pipe by a hypothetical recirculation pump missile randomly directed is of the same nature as the random failure of two active components. The low probability of this failure in conjunction with the "double-failure" nature of the occurrence makes it outside the scope of the design basis of the DAEC.

3.5.1.3 Turbine Missiles

[redacted] are designed to withstand the loading due to missiles generated by the failure of the turbine-generator. Further details on missile generation probability, missile characteristics, penetration ability of missiles, and protection against missiles are provided below.

The GE turbine for the DAEC is a three-casing, tandem-compound, four flow exhaust, 1800-rpm unit with 38-in. last-stage blades. Connected to the turbine shaft is the ac generator, and coupled to the generator is the exciter.

The original low-pressure rotors have been replaced by the “monoblock” design. This design has wheels which are integral to the shaft, while the original design had separate wheels which were shrunk onto the shaft. The missile analysis below, which considers a last stage wheel failure, was applicable to the original design. The probability of

wheel missile generation is significantly lower with the monoblock design, and any postulated wheel missile would have less potential for causing damage since the monoblock design precludes failures which would result in large wheel fragments. Therefore the analysis below represents a highly conservative upper bound for the monoblock rotor. The bucket vane analysis below is applicable to the monoblock rotor,
2013-019 since it uses the same bucket design as the original rotor. The bucket vane analysis is also applicable to the long shank bucket design installed on the 5^{th} stage turbine wheels of both low pressure rotors. The long shank design has the same type of connection as the original bucket design.

The probability of the occurrence of turbine missiles has been evaluated by GE to be very low. Proven control system reliability and equipment redundancy make a turbine runaway highly improbable. Even assuming a turbine runaway, modern test procedures of disk forgings virtually eliminate the possibility of wheel failure, except at above 169% of rated speed.

Based on the GE report[4], the following 40-year interval probabilities of a wheel failure have been calculated:

Low speed failure (below 127% speed), 3.5×10^{-7}
Runaway failure (at higher speed), 2.1×10^{-7}
Total lifetime failure probability, 5.6×10^{-7}

The corresponding average annual probability of a wheel failure is found to be 1.4×10^{-8}.

However, recognizing the potentially serious consequences of turbine missiles, it was postulated in this study that turbine missiles are generated and their effect on the plant safety was evaluated.

In the case of the GE low-pressure turbine, two missiles were postulated. The first, and potentially the most dangerous because of its size, was the last-stage wheel[5]. Typically, it is also the most highly stressed and hence the most probable candidate for failure. The second missile considered was the bucket vane. There is a good possibility that it would lose most of its energy and would be deformed after penetrating the inner casing and the hood.

General Electric studies indicate that in the case of a last-stage wheel, 120-degree fragments are potentially more damaging than either 90- or 180-degree fragments[5]. The properties of a 120-degree wheel fragment are shown in Table 3.5-3 and the properties of a bucket vane are shown in Table 3.5-4.

Local penetration effects were determined by the modified Petry Formula[6]. According to the formula, a missile will penetrate a reinforced-concrete slab of infinite thickness the following amount:

$$D = K\,A_P\,V' \qquad (3.5\text{-}1)$$

where

K = penetration coefficient of concrete experimentally determined (see Table 3.5-5)

A_P = sectional pressure of the missile obtained by dividing the missile weight by the frontal area (lb/ft)

V = striking velocity of the missile (ft/sec)

$$V^1 = \text{Log}_{10}\, 1 + \frac{V^2}{215{,}000}$$

Amirikian reports Navy experiments that resulted in a formula to calculate penetration into concrete slabs of finite thickness:

$$D' = D\,[1 + e^{-4(T/D\,-\,2)}] \qquad (3.5\text{-}2)$$

where

D = penetration depth in an infinite slab, ft
D' = penetration depth in a finite thickness slab, ft
T = slab thickness, ft

The rearrangement of this equation shows that D = T/2 gives complete penetration. Therefore, from Equation 3.5-1, the thickest slab that will be perforated by a missile is:

$$T = 2\,K\,A_P\,V' \qquad (3.5\text{-}3)$$

Both low trajectory missiles (LTM), where the direction of impact is predominately horizontal, and high trajectory missiles (HTM), where the direction of impact is vertical, were considered. Air drag, which has the effect of significant energy loss for the missile, was considered for the HTM.

The results of the penetration ability of the postulated turbine missiles are summarized in Table 3.5-6.

The penetration depths shown in Table 3.5-6 represent the minimum concrete thickness required to stop the respective missiles.

All equipment required for a safe shutdown of the plant was reviewed regarding its vulnerability to the postulated turbine missiles.

Table 3.5-6 lists the minimum concrete wall thickness to prevent penetration by the missiles under consideration. All LTM generated [redacted] will be contained by the heavy beams of the turbine foundation. A summary of protection provided for safety-related equipment located in the various buildings is given in the following sections.

3.5.1.3.1 Reactor Building

The exterior wall of the reactor building [redacted] directly in the path of a turbine missile varies in thickness [redacted] which will provide adequate protection. See also Section 10.2.

3.5.1.3.2 Spent-Fuel Storage Pool

Damage of the pool by low trajectory turbine-generated missiles is not possible due to the structural and shielding requirements that result in [redacted]

[redacted] As noted in Section 3.5 1.3, the total lifetime probability of a wheel failure is 5.6 x 10^{-7}. [redacted] In view of the low probability, no further analysis is required.

3.5.1.3.3 Control Building

The control building is located outside the path of a turbine-generated missile. [redacted] provide adequate protection.

Based on the conservative assumption that a secondary missile, spalling of concrete, will be generated from a postulated turbine missile intersecting the reactor building east-wall, the 40-year interval probability of occurrence is 5.6 x 10^{-7} as evaluated above.

Recognizing the potential serious consequence of secondary missiles, all systems and equipment required for safe shutdown of the plant were reviewed regarding their vulnerability to a postulated secondary missile.

An inventory of all systems required for safe shutdown of the plant shows that there are none that are vulnerable to a secondary missile.

3.5.1.3.4 Diesel-Generator Rooms

[redacted] Accordingly, no turbine missile will disable the diesel-generator.

3.5.1.3.5 Area of the Drywell Shield Plugs

The only missiles that can strike in the area of the shield plugs are the 4 in. x 12 in. x 12 ft plank at 300 mph and the missile generated by a turbine-generator failure (see Section 3.5.1.2). The plank missile is postulated to travel in a horizontal direction and thus cannot strike the horizontal exposed surface of the plug. An analysis for the unlikely event of a turbine-generator failure has been made. Using the modified Petry Formula, it was found that [redacted] thus penetration is not possible.

The lower surface of the bottom plug is sheated with 0.25-in. stainless steel plate so that any localized concrete spalling will not result in concrete falling into the drywell.

3.5.1.4 <u>Missiles Generated by Natural Phenomena</u>

A missile equivalent to a 4 in. x 12 in. x 12 ft long wood plank (108 lb) traveling end-on at 300 mph and a passenger auto (4000 lb) flying through the air at 50 mph and at not more than 25 ft aboveground with a contact area of 20 ft^2 generated by the design basis tornado were considered in the design of the DAEC (see Section 3.3.2.1).

3.5.1.5 <u>Missiles Generated by Events Near the Site</u>

The closest commercial rail line to the DAEC which could carry hazardous material on a routine basis is approximately 3.5 miles west of the site. No hazardous material is manufactured in the vicinity of the site.

There are no known mineral mines or petroleum wells located within 5 miles of the plant site. There is only one operational quarry within 5 miles of the site. It is located approximately 3 miles southwest of the DAEC site. The potential hazard from the quarry is judged to be insignificant (see Section 2.2.3).

A survey of offsite facilities at the time of initial FSAR submission (1972) concluded that there were no offsite facilities that might have a significant effect on the safe operation of the plant.

[redacted] The hazard analysis contained in Reference 2 determined the gas pipelines around the DAEC site present negligible risk to the safe operation of DAEC.

3.5.1.6 Aircraft Hazards

Aircraft hazards from airports, military aviation and federal airways in the vicinity of the DAEC are discussed in Section 2.2.2.5, 2.2.2.5.1 and 2.2.2.5.2.

At the time of the initial FSAR, the private airport for light planes were approximately 2 miles west in Shellsburg, Iowa. Such a facility was designated on the most recent edition of the Dubuque Sectional Aeronautical Chart and also on the 1968 USGS topographical map, Shellsburg, Quadrangle. Investigation and discussion with Cedar Rapids area aviation organizations indicated that a private restricted airstrip was operated at one time at the above mentioned location by an individual who owned a single light airplane, but the owner had dismantled the hangar and later died. The air strip was no longer in use, but was still designated on the aeronautical charts because apparently no cancellation had been initiated. During the course of the investigation, it was determined that another small landing strip not shown on the most recent aeronautical charts exists at a location approximately 4 miles southeast of the plant. Discussion with the airfield manager indicated that approximately 10 single engine light airplanes were stationed at the strip. The maximum gross weight of these aircraft was estimated to be 3000 lb. Runway orientation is north-south (300U ft turf) and east-west (2600 ft turf) according to the "Iowa Airport Directory 1972-73," published by the Iowa Aeronautical Commission. The facility was an uncontrolled field, and accordingly there was no record kept of the number of takeoffs and landings. An estimate of the number of takeoffs and landings was made based upon the assumption that between May 1 and November 1 each of the 10 planes averaged four movements per weekend. During the winter months it was estimated that four of the planes are active on six weekends, again at an average of four movements per weekend. These assumptions were felt to be conservative and were based upon discussions held with the owner. Accordingly, it was conservatively estimated that there were less than 1500 movements per year from this airfield. The airfield owner indicated that at the time of contact there were no plans for significantly expanding the scope of operations at this facility. No other airfields were known at that time to exist within 5 miles of the DAEC. At the time of the initial FSAR the potential for aircraft accidents at the facility was considered extremely improbable due to the relatively small number of airplane movements at the small landing strip described in Section 2.2.2.5.

[redacted]

[redacted] However, under no conditions would the ability to safely shut down the reactor be compromised. Thus, the effects of an aircraft accident would be possible loss of generating capacity and [redacted] However, the capability to safely shutdown the reactor would be maintained.

Aircraft hazards from airport , military aviation and federal airways in the vicinity of the DAEC are discussed in Section 2.2.2.5, 2.2.2.5.1 and 2.2.2.5.2. The conditional probability of core damage from an aircraft crash into the site is evaluated and documented in Appendix B of Reference 7. This analysis was prepared in response to Generic Letter 88-20 and submitted to the NRC as part of DAEC's Individual Plant Examination External Events. This analysis found the core damage frequency to be below the NUREG-1407 screening criterion of 1E-6/yr. Therefore, the analysis concludes that the design of the DAEC plant is appropriate of its sitting (i.e., with respect to aviation hazards) such that the contribution to overall plant risks from aircraft crashes in and around the plant is not significant.

3.5.1.7 <u>Propane Storage Tank</u>

An auxiliary boiler propone gas pilot supply tank [redacted]. Evaluation of the tank shows that the tank does not present an immediate risk to plant safety, but an accident sequence (vehicle impact) could conservatively be postulated with a frequency greater than the NUREG-1407 screening criterion of 1E-6/yr. [redacted] thereby reducing the overall risk contribution to less than the NUREG-1407 screening criterion of 1E-6/yr.

3.5.2 STRUCTURES, SYSTEMS, AND COMPONENTS TO BE PROTECTED FROM EXTERNALLY GENERATED MISSILES

Reactor building walls and slabs and control room walls and slabs are designed to withstand the loading due to missiles generated by the failure of the turbine-generator (see Section 3.5.1.3). The thickness of the structural components is adequate to prevent complete penetration by a missile.

Structures housing equipment necessary for the safe shutdown of the reactor are designed to withstand the design-basis tornado-generated missiles (see Section 3.3.2.1).

3.5.3 BARRIER DESIGN PROCEDURES

Within containment the following measures have been taken to ensure that the damage caused by any single missile that might be generated from a component failure will not remove more than one redundant subsection of a vital safety system from service.

The separation of redundant safety-related components has been used as a basic criterion throughout the design of this plant to optimize the independence of these components. The design intent was to separate redundant counterparts by 180º in azimuth and, where this was not possible, to maintain the maximum possible separation. An elaborate truss system has been used inside the drywell to restrain large pipes; this truss system has also furnished a considerable amount of structural material which acts as a protective barrier and prevents a missile from traveling a very great distance. Section 6.2.1 discusses in more detail the means by which primary containment integrity is protected.

2012-012 [redacted] Missile protection for the EDGs is accomplished by separation, as described in Section 8.3.1.3.

REFERENCES FOR SECTION 3.5

1. ACRS letter on Duane Arnold Energy Center Operating License from H. G. Mangelsdorf to Dixy Lee Ray, dated March 13, 1973.

2. General Electric Company, Analysis of Recirculation Pump Overspeed n a Typical General Electric Boiling Water Reactor, NEDO-10677, October 1972

3. "Probabilistic Analysis of the Effects of Missiles Formed in the Recirculation System Following Postulated Pipe Failure," filed with James A. Fitzpatrick Nuclear Power Plant, Docket No. 50-333, from Asa George, of PASNY, to J. F. Stolz, AEC, dated April 10, 1973.

4. General Electric Company, Probability of Turbine-Generator Rotor Failure Leading to Ejection of External Missiles, memo report, February 1971.

5. E. E. Swicky, An Analyses of Turbine Missiles Resulting from Last-Stage Wheel Failure, TR675L211, General Electric, Schenectady, New York, 1967.

6. A. Amirikian, Design of Protection Structures, Nav. Docks P-51, Bureau of Yards and Docks, Department of the Navy, Washington, DC, 1959.

7. IES Utilities Inc., Individual Plant Examination of External Events, Transportation and Nearby Facility Hazards Evaluation of the DAEC Site, December 1995.

Table 3.5-1

TOTAL PROBABILITY OF DAMAGE TO PIPING
DUE TO RECIRCULATION PUMP MISSILES
(with LOCA probability of 1 x 10^{-5} per year)

Occurrence	Total Probability (per year)
Penetration of main steam line	0.63 x 10^{-8}
Disablement of main steam isolation valve	1.0 x 10^{-8}
Penetration of feedwater line	2.0 x 10^{-7}
Penetration of core spray line	6.5 x 10^{-8}
Disablement of high-pressure coolant injection	1.5 x 10^{-8}
Disablement of reactor core isolation cooling	0.5 x 10^{-8}

Table 3.5-2

PROBABILITY OF RELEASE OF
RADIOACTIVE MATERIAL
TO ENVIRONMENT DUE TO RECIRCULATION PUMP MISSILES

Events	Probability per Year of Recirculation Pump Missiles (with LOCA[a])	All Other Causes Except Pump Missiles
Loss of containment integrity	30×10^{-9}	1×10^{-9}
Fuel damage[b] and loss of containment integrity	63×10^{-13}	2.1×10^{-13}

[a]Given a design-basis LOCA probability of 1×10^{-5} per year.
[b]Fuel damage probability based on "realistic" core heatup assumptions is 2.1×10^{-4} per year.

Table 3.5-3

PROPERTIES OF A 120-DEGREE SEGMENT FROM A 38-IN. WHEEL[a]

Parameter	Value
Fragment angle	120 degrees
Fragment weight	5944 lb
Minimum proj. area	3.657 ft^2
Maximum proj. area	8.368 ft^2
Failure speed, % of 1800 rpm	169
Initial velocity	666.8 ft/sec
Energies	
Initial, transition	41.0 x 10^6 ft-lb
Initial, rotation	23.5 x 10^6 ft-lb
Outside turbine casing, translation	20.5 x 10^6 ft-lb
After air drag (vertical trajectory)	16.3 x 10^6 ft-lb

[a]See Reference 5.

Table 3.5-4

PROPERTIES OF A BUCKET VANE

Parameter	Value
Vane weight	27.25 lb
Vane tip area	1.41 in.^{2}
Vane root area	4.36 in.^{2}
Vane maximum thickness	0.920 in.
Failure speed, % of 1800 rpm	169
Failure velocity	1600 ft/sec
Failure energy	1 x 10^6 ft-lb
Estimated velocity after penetrating casing	1100 ft/sec
Estimated energy after penetrating casing	0.5 x 10^6 ft-lb

Table 3.5-5

PENETRATION COEFFICIENTS

Material	K
3000-psi reinforced concrete	0.00476
6000-psi reinforced concrete	0.00282

Table 3.5-6

PENETRATION ABILITY OF POSTULATED TURBINE MISSILES

Concrete Compressive Strength (psi)	38-in. Wheel LTM	38-in. Wheel HTM	Bucket Vane LTM	Bucket Vane HTM
6000	1 ft 9 in.	1 ft 6 in.	0 ft 6 in.	0 ft 6 in.
3000	3 ft 0 in.	2 ft 6 in.	1 ft 0 in.	1 ft 0 in.

[redacted]

[redacted]

[redacted]

[redacted]

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

3.6.1.1 Design Bases

Not required for FSAR.

3.6.1.2 Description

A detailed analysis has been made of all high-energy lines located outside containment at the DAEC to determine if the rupture of any such line would impair the ability of the plant to be shut down and maintained in a safe shutdown condition.

This analysis was conducted in accordance with the criteria set forth in the attachment to the AEC letter of December 15, 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," and supplemented by analysis utilizing the revised guidance contained in NRC Generic Letter 87-11 as well as the loss of offsite power criterion in subsection 3.b.(1) from the Branch Technical Position SPLB 3-1 attachment to Standard Review Plan No. 3.6.1, Revision 2.

[redacted] These figures are not revised to show subsequent changes.

3.6.1.2.1 Assumptions

1. The assumed modes of pipe failure are as follows:

   a. Circumferential breaks, Figure 3.6-42, are those breaks that are perpendicular to the pipe axis. The break area is taken to be the same as the internal cross-sectional area of the pipe unless the pipe is adequately restrained to prevent relative motion of the two sides of the break. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause whipping in any direction normal to the pipe axis, unless the pipe is restrained to prevent such motion.

   b. Longitudinal breaks, Figure 3.6-43, are those breaks that are parallel to the pipe axis. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis, unless the pipe is restrained to prevent such motion.

c. Critical size cracks are those breaks that are taken to be one-half the pipe diameter in length and one-half the wall thickness in width.

2. Circumferential and longitudinal breaks have been assumed to occur at the following locations in each piping run or branch run of Seismic Category I piping. Only a single break has been assumed to occur.

a. Terminal ends.

b. Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 ($S_h + S_A$), where S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code, Section III, Winter 1972 Addenda, and S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the Code for Pressure Piping, ANSI B31.1.0, 1967, or the expansion stresses exceed 0.8 S_A.

c. Two intermediate locations in addition to those determined by the above, selected on the basis of highest stress determined by taking the sum of normal operation stresses and seismic stresses.

Note: The above criteria for intermediate breaks has been relaxed by use of the criteria in NRC GL 87-11. This applies to the RCIC and HPCI steam supply piping, RWCU return piping, and the Feedwater lines between the drywell penetrations and the outboard whip restraints.

3. For Nonseismic high-energy piping, a single circumferential and longitudinal break has been postulated at any location.

4. A critical size crack has been postulated to occur at any location along the length and at any point around the circumference of a pipe carrying high-energy fluid. This criterion is not applied to the HPCI and RCIC steam supply lines or the RWCU return piping from the regenerative heat exchanger to the feedwater system, or the Feedwater lines between the drywell penetrations and the outboard whip restraints, which have been analyzed pursuant to the revised criteria in NRC GL 87-11.

5. The postulated break has been conservatively assumed to occur during normal steady-state operating conditions at rated power.

6. Loss of offsite ac power has been assumed to occur concurrently with the postulated failure of the high-energy pipe except where the revised criteria of NRC Generic Letter 87-11 have been applied using the criterion in the Standard Review Plan (SRP) as indicated in Section 3.6.1.2. This is, in

general, a conservative assumption. However, in some cases, a loss of normal ac power actually mitigates the effects of the postulated pipe break (e.g., failure in the feedwater system). In those cases, it has been conservatively assumed that offsite power is not lost.

Use of the loss of offsite power guidance from the SRP with the implementation of Generic Letter 87-11 is a departure from the conventional practice used in analyses at DAEC; however, it is consistent with the guidance in the Standard Review Plan, Revision 2. This methodology was followed since Generic Letter 87-11 was initiated to provide a revision to the guidance given in the SRP and by following these guidelines, greater consistency resulted between the DAEC practice and the methodology provided in the SRP. This practice may be followed if other opportunities arise for performing revisions to earlier analyses.

7. No other accident has been assumed to occur concurrently with the pipe failure outside the containment.

8. A single failure of an active component has been assumed to occur in analyzing the accident and the ability to safely shut down the plant.

3.6.1.2.2 General Approach

The analysis has been conducted using the following general procedure:

1. The high-energy piping systems were identified using the criteria that the service temperature is greater than 200ºF and the design pressure is greater than 275 psig. The systems meeting those criteria are the following:

    a. Main steam.

    b. Feedwater.

    c. HPCI steam.

    d. RCIC steam.

    e. Reactor water cleanup (RWCU).

    f. High-energy sampling and instrument sensing lines.

    g. HPCI discharge piping in the steam tunnel upstream of the normally closed inboard isolation valve (MO2312).

2. The systems required for safe shutdown of the reactor for the postulated pipe failure of each of the high-energy systems were identified. By verifying that these systems are maintained operational in the event of the postulated

failure of the piping systems, it is ensured that the plant can be safely shut down and maintained in a safe shutdown condition.

3. physical arrangement of each high-energy piping system was investigated to determine the potential effects of pipe whip or jet impingement on structures, systems, and components required for safe shutdown of the plant.

4. The electric cables that could be broken by either pipe whip or jet impingement were identified. Each cable that could be broken was tabulated and the effects of its loss analyzed with respect to the ability to safely shut down. The loss of required redundancy was considered. The results of that study were incorporated in the analyses of shutdown capability that were included in the study of each high-energy system.

5. The effects of steam pressurization of the compartments that could be pressurized by the failure of any of the identified high-energy lines were investigated.

6. The environmental effects of postulated ruptures of any of the high-energy lines were evaluated.

7. A site study was made of each area of concern to verify that no other potential problems exist. Vent areas between compartments were quantitatively checked.

8. For those problem areas identified by the above process, alternative corrective measures were considered and a method of correction decided upon.

9. The modification to install a HPCI High Pressure Keep Fill system introduced the possibility for the HPCI discharge piping, in the steam tunnel, upsteam of the normally closed inboard isolation valve (MO2312), to at times, meet the criteria of a high energy line. The maximum possible energy release as a result of a HPCI discharge line break is bounded by that of a feedwater line break in the steam tunnel, and therefore, does not require analysis.

3.6.1.2.3 Inherent Safety Features of the DAEC

[redacted] The seismic classification of the various structures and systems is given in Section 3.2.1.

Safeguard equipment is located in Seismic Category I structures with redundant equipment being physically separated by distance as well as Seismic Category I walls. [redacted]

By the nature of this type of reactor plant, [redacted] The shielding walls also mitigate the effects of a high-energy pipe failure by providing physical separation between the high-energy lines and other equipment.

[redacted] Using the criteria previously outlined, it has been determined that no postulated high-energy pipe failure can cause damage from pipe whip, jet impingement, external overpressurization, or environmental conditions to the control room complex.

[redacted] There are no high-energy lines in the vicinity of the diesel-generator rooms. [redacted] No postulated high-energy pipe failure can cause damage from the pipe whip, jet impingement, or environmental conditions to the onsite emergency ac power supply.

Information on pressure integrity of piping, applicable codes, specifications, and piping classification is presented elsewhere in this chapter. The quality control and inspection programs that have been used for piping systems outside the containment are specified in this chapter and in Chapter 17.

The following general comments apply to the design of the DAEC relating to the environmental effects of a postulated high-energy pipe failure in addition to those in Section 3.11.

1. All Class 1E cabling used throughout the DAEC has been selected to comply with the environmental specifications as outlined in Section 7.3.5 (safety-related) based on testing and/or analysis.

2. [redacted]

3. The valve motor operators used outside the primary containment are [redacted] similar to those used inside the primary containment. The [redacted] used within the primary containment under high-temperature saturated steam conditions is described in Section 3.11.

4. Safeguard equipment is protected from the direct effects of postulated pipe failures which could cause the environment in which it is located to become environmentally harsh per Reference 9 of Section 3.11. Plant specific high energy-line break evaluations have demonstrated that the possibility exists for [redacted] This scenario however, does not prevent the mitigation of the event since the redundant train of RHR is

available. No high-energy line unrelated to the equipment is routed through a safeguard equipment room.

5. The criteria for routing of electric cabling are presented in Chapter 8. Factors such as physical separation and encasement of cabling in conduit further mitigate the effects of the postulated pipe failure on the ability of the plant to be safely shut down.

6. There are no safeguard instrument panels located in compartments through which high-energy piping is routed, nor are there direct line-of-sight paths of communication (doors, wall penetrations, etc.) between high-energy pipes and safeguard panels. Therefore, the direct effects of pipe whip and jet impingement on such panels can be neglected.

7. [redacted] As such, it is highly unlikely that steam from a high-energy pipe failure would be drawn into the control room.

3.6.1.2.4 Analysis

3.6.1.2.4.1 Structural Loading. The methods used to evaluate the adequacy of the structures that could be affected by the postulated failure of a high-energy line outside the primary containment are the same as those presented in Section 3.8.

3.6.1.2.4.2 Jet Impingement Loading. The following formulas were used to evaluate the various categories of jet impingement loading:

1. Jet impingement from a circumferential high-energy pipe failure as a function of distance x from the break:

$$P_x = \frac{K_j \{P_o - P_a\}}{[1 + 2x \tan\varnothing/D]^2}$$

where

P_x = pressure of jet as a function of distance
Kj = thrust coefficient
= 1.26 for steam or two-phase flow
= 2.0 for subcooled water such as feedwater or cleanup flow
P_o = pressure in pipe just upstream of break (psi)
x = distance from break (in.)
D = pipe inside diameter (in.)
P_a = ambient pressure outside of pipe before break (psi)
$\varnothing$ = half angle of jet dispersion
= 10 degrees ($\tan \varnothing = 0.17633$)

2. Jet impingement from a longitudinal high-energy pipe failure as a function of distance x from the break:

$$P_x = \frac{K_j \{P_o - P_a\}}{(1 + 2x \tan \varnothing/\ell)(1 + 2x\tan \varnothing/w)}$$

where the terms are as defined above and

ℓ = crack length (in.) = 2D
w = crack width (in.) = A/ℓ

3. Jet impingement from a critical crack failure in a high-energy pipe as a function of distance x from the break is assumed to be the same as for a longitudinal failure.

It has been assumed that jet impingement forces are developed in zero time and that any changes in the forces as a function of time are negligible. Jet impingement is assumed to cease when the high-energy line is isolated.

3.6.1.2.4.3 Factors to Account for Target Shape.

1. For flat or concave surfaces, the effective area of jet interaction was taken to be equal to the projected area of the target normal to the jet or the expanded area of the jet normal to the target, whichever was smaller.

2. For circular targets, such as pipes, the effective area of interaction was taken to be 0.6 of the cross-sectional area of the target normal to the jet or the expanded area of the jet normal to the target, whichever was smaller. Therefore, the force on a circular target is given by:

$$F = 0.6\, P_X A_T$$

where

P_x = jet impingement pressure as calculated by the appropriate formula discussed above
A_T = cross-sectional area of the target normal to the jet

3.6.1.2.4.4 Jet Thrust Forces. The reaction forces acting on the pipe caused by the momentum change of fluid flowing through the break were calculated using the following relationship:

$$F_j = K_j A_b (P_o - P_a)$$

where K_j, P_o, and P_a are as defined above and

F_j = jet reaction force (lb)
A_b = area of break (in.^{2})

3.6.1.2.4.5 <u>Compartment Pressure Analysis Model</u>. A postulated high-energy pipe rupture is the expulsion of high-energy steam or a steam-water mixture out of the ruptured pipe into the surrounding compartment. As the pressure builds up within the compartment, the steam-air-water mixture flows through openings to relieve the resultant pressure. The equilibrium pressure achieved depends on the number and shape of the openings between the compartments, the volume of each compartment, and the blowdown rate from the broken pipe. Differential-pressure analyses were made to calculate the pressure responses of two compartments during the postulated event. The calculations include mass and energy balances of the two-phase, two-compartment, steam-air-water mixture as high-energy fluid enters the compartments during the event and passes through the various compartment vent openings. There are no provisions in the calculations for heat transfer since heat transfer would have a negligible effect on compartment pressures for the short time following the rupture within which peak differential pressure occurs.

The analyses conducted yielded pressure versus time relationships that were used to determine if the structures surrounding the affected compartments or vital equipment within the compartments were capable of withstanding the pressurization. In all the cases the analyses indicated that sufficient vent area was available to prevent overpressurization.

3.6.1.3 <u>Safety Evaluation</u>

In each of the system evaluations described below, the following systems were analyzed with regard to their required functions for the safe shutdown of the plant. The applicability of each of these systems was determined using the criteria set forth in Sections 6.3 and 7.3 and Chapter 15.

1. Pressure relief system.

2. Reactor scram protection.

3. Reactor vessel isolation.

4. Required core cooling (RHR, HPCI, RCIC, ADS, and emergency service water).

In all cases, the system evaluation included the electric power supplies and the instrumentation associated with a particular system.

[redacted]

[redacted]

Pipe failure in the main steam system outside the primary containment is discussed in Chapter 15. The design of the main steam lines, flow restrictors, and isolation valves is presented in Chapter 5. As described in Section 7.3, the main steam lines will automatically isolate in the event of a postulated failure.

With respect to pipe whip, the postulated main steam line break locations are the terminal ends and the two highest stressed intermediate points. Specifically, these locations are the following:

1. The connections at the downstream side of the outboard main steam isolation valves. It is assumed that at these locations either a longitudinal or a circumferential break could occur at the junction of the steam pipe to the isolation valve.

2. [redacted]

3. The connections at the turbine stop valves.

As part of the design of the DAEC to ensure the protection of the outboard main steam isolation valve in the unlikely event of a pipe rupture in the steam tunnel or the turbine building, pipe whip restraints have been placed in the steam tunnel around the main steam and feedwater piping as shown in Figure 3.6-44. These restraints preclude damage from pipe whip due to a pipe rupture in this area from having any unacceptable effect on the ability of the plant to be safely shut down.

To mitigate the effects of a pipe break down stream of the outboard isolation valve, restraining structures have been placed in the area immediately external to the containment.

The major component in this system of restraints is the [redacted] it is capable of restraining a full spectrum of loading conditions, including the rupture of the main steam/feedwater line.

[redacted] to control the movement of the piping in the unlikely event of a pipe rupture in this area. These structures are capable of restraining the full thrust force of the ruptured pipe.

As discussed in Section 7.3, the HPCI, RCIC, and main steam lines are equipped with differential-pressure switches that isolate the respective lines on high flow. In the event of a
2013-012 feedwater line break, a low suction pressure alarm and feedwater pump trip is initiated and a reactor scram will occur on low water level.

With regard to the possible effect of jet impingement, [redacted] The possible effects of jet impingements on this conduit have been analyzed; no protection is required.

It is concluded that whipping of the main steam lines from the postulated failure will not result in the failure of any other high-energy lines or equipment required for safe shutdown of the plant.

It has been determined by analysis that the steam tunnel structure can withstand the combined effects of dead loads and steam tunnel peak pressurization plus the jet impingement loads from the steam line break.

A blowout panel in the entrance of the steam tunnel to the turbine building prevents any excessive pressure buildup in the tunnel.

Jet impingement and environmental effects from a critical crack in the main steam lines have been investigated. The critical cracks have been assumed to occur at any point along the main steam pipes and at any point on the circumference of the pipe. It has been assumed that such breaks can occur at the most adverse location with respect to structures or components required for safe shutdown. It has been determined that no such failures in the main steam lines will adversely affect the ability to safely shut down the plant. The analysis leading to this conclusion included a study of the effects of jet impingement on the adjacent pipes; on the floor, walls, and ceiling of the tunnel; and on the [redacted]

It is concluded that the plant can be safely shut down following a postulated main steam line failure outside the primary containment.

3.6.1.3.2 Turbine Building

As the main steam lines enter the turbine building, they are routed as shown in [redacted] The reactor protection cabling for the low condenser vacuum and turbine stop valve fast-closure signals for main steam isolation valves [redacted] This situation presents no problem, however, as backup isolation signals exist for the main steam isolation valves as discussed in Section 7.3.

As the main steam lines approach the turbine stop valves, their configuration is such that a rupture could cause a line to whip toward the control building. This situation has been analyzed, and the presence of two reinforced-concrete block walls, a corridor, the reinforced-concrete wall of the control building, and the floor slab at [redacted] ensures that no loss of structural integrity of the control building will occur.

The only other potential problem area in the turbine building was found to be [redacted]

2013-016 [redacted] For areas potentially subject to adverse effects of a postulated feedwater pipe break, this cabling is routed in rigid steel conduit outside zones of potential pipe impact and with sufficient separation/protection to preclude detrimental damage from jet impingement.

It is concluded that the plant can be safely shut down following a postulated
2013-016 feedwater pipe break outside the containment.

3.6.1.3.3 Reactor Building

In the reactor building, the HPCI and RCIC steam lines were analyzed and break locations determined as shown in [redacted] Based upon the revised criteria for determining break locations, which is contained in NRC GL 87-11, the HPCI and RCIC steam lines are postulated to break only at the terminal ends. Thus, only the steam turnnel and HPCI and RCIC equipment rooms are affected. It was found that at various points ruptures could cause damage to RHR service water and/or emergency service water systems. In all cases, the rupture would result in having to shut down the equipment located in one corner room (one-half of the RHR system) as a result of a loss of cooling water or service water to the equipment.

The loss of the redundant service water loop and thus the ability to shut down and cool down the plant is dependent on the ability to manually operate the RHR heat exchanger discharge valve in the redundant loop. This valve is the only single active component in each of the service water trains whose failure would result in the loss of that train. These valves are located so that they would be accessible in the event of a rupture that would disable the other train. The location of these valves and the cross-connection capability of the RHR and emergency service water systems as described in Section 5.4.7 ensure that sufficient time would be available to restore the ability to shut the plant down and maintain it in a safe shutdown condition.

A rupture of the RCIC steam line in the RCIC room would result in damage only to the RCIC system itself.

A rupture of the HPCI steam line inside the HPCI room could result in damage to the emergency service water system or one of the RHR service water valves described previously. In this case, the above discussion relative to the RHR heat exchanger valve would also apply.

The RWCU system is described in Section 5.4.8. The system contains high-energy fluid in only a portion of its piping. [redacted]

A postulated failure in the cleanup system will result in a single-ended failure. A check valve in the return line immediately upstream of the connection into the feedwater piping would prevent backflow from the return side of the break. Automatic isolation of the cleanup system is initiated by various line break sensing instruments. These are identified in Section 5.4.8.

It has been determined that pipe whip or jet impingement from pipe failure at the postulated break points will not adversely affect the ability to safely shut down the plant. The substantial shielding walls surrounding each of the equipment cells prevent damage to any equipment outside the cell containing the postulated break. The cleanup system equipment cells do not contain any components not related to the cleanup system that are required for safe plant shutdown.

Critical cracks in the RWCU line have been evaluated and will not adversely affect the ability to safely shut down the plant.

It is concluded that the plant can be safely shut down and maintained shut down following a postulated RWCU system line failure outside the primary containment.

All instrument and sample lines in the DAEC are routed such that their failure would not result in any damage to equipment necessary for safe shutdown.

The postulated rupture of an instrument or sample line would be a single-ended failure that would not result in any direct effects on structures, systems, or components required for shutdown. Pipe whip and jet impingement effects would be negligible due to the small size of these lines and the fact that they are completely enclosed in protective channels. Therefore, the only potential effects of a postulated line failure are environmental in nature and would not have any adverse effect on the ability of the plant to be safely shut down.

Instrument line isolation considerations and the effect of an instrument line rupture are discussed in Section 6.2.4.

From the discussions presented above, it is concluded that the design of the high-energy piping systems outside containment on the DAEC is such that their failure would not result in the inability of the plant to be shut down and maintained in a safe shutdown condition.

3.6.2 POSTULATED PIPING FAILURES IN FLUID SYSTEMS INSIDE CONTAINMENT

3.6.2.1 Design Bases (PSAR)

Not required for FSAR.

3.6.2.2 Description

Pipe failures inside containment can give rise to pipe movement and jet impingement effects. Protecting against pipe movement and jet impingement involves the consideration of the “source" and the "target." The source investigation must include the determination of where a pipe may fail and the means to compute the resultant forces. The failed pipe can result in a pipe movement in reaction to the expulsion of a high-pressure fluid or jet-like impingement of the fluid itself. The target investigation must include identification of components or systems that are considered essential, and the means to determine how much force or energy a target can receive without the impairment of function.

Table 3.6-1 is a matrix that was developed to display the interrelationships that exist between the identified potential sources of pipe movement and the systems, components, and structures important to safety. These interrelationships were investigated using design drawings and a model of the DAEC drywell.

[redacted] as an aid in picturing the restraint system, pipe routing, and physical arrangement within the drywell area.

3.6.2.2.1 Assumptions

1. General Assumptions

   a. The pipe break was assumed to occur in combination with the DBE and loss of offsite power.

      (1) Direct current and standby ac power were available.

      (2) No credit was taken for the integrity of any piping or the functioning of any system not classified as Seismic Category I.

   b. The single failure of an active component (i.e., the single-failure criterion) was applied after the given rupture and its immediate effects had occurred.

   c. The plant was in normal operation before the initiating event.

d. In this analysis, the loop selection logic is assumed to be inoperable. Since it is required for the recirculation-line break only, protecting the core spray loops from a recirculation-line break is sufficient.

2. Source Piping Assumptions

a. All pipes were assumed to be susceptible to either a circumferential or slot failure throughout that portion of their length that is normally exposed to reactor pressure. Failures were assumed to be nonmechanistic; no attempt was made to specify how they occurred.

b. In those portions of systems where the disposition of valves and/or check valves is such that a sustained jet will not result subsequent to a pipe rupture, pipe movement or jet spray was not considered.

c. Pipe trajectories were based on the reaction of the source pipe to a circumferential rupture. Various pivotal points for the pipe were chosen and the resultant movement evaluated.

3. Target Impingement Assumptions

a. A line of a given size and schedule will not cause a line of equal or larger size and schedule to fail to perform its function if the smaller should strike the larger.

b. Any component (except the containment vessel wall) within the jet spray of the postulated rupture was not considered functional following the event. Ten pipe diameters radially from the rupture and a 10-degree half-angle of dispersion were used as guideline values for determining the extent of the jet spray.

c. Valves that are closed during normal operation were not required to have protection to ensure closure during the isolation process.

d. The local temperature of the primary steel containment under jet impingement was taken as 300▪F. In areas where local deformation is not permitted, the allowable stresses corresponding to this temperature were used.

3.6.2.2.2 General Approach

The following systems were identified as essential to plant safety:

1. Safety-related systems required to shut down the reactor.

2. Safety-related systems required to isolate the reactor vessel and primary containments.

3. Safety-related systems required for cooling the core.

4. Safety-related systems required for containment integrity.

The study was conducted by analyzing, for each potential source, the consequences of a rupture as it affected the essential targets. In general, the following four solutions satisfied the objective:

1. The source pipe is restrained.

2. The source and target are separated by a great enough distance such that the source cannot affect the target.

3. The source cannot affect the target due to the interference provided by intervening structures.

4. If contact can be made by a jet or the pipe traveling on a hypothesized trajectory, the consequences will not prevent the safety-related functions from being accomplished, that is, the containment vessel wall will not be penetrated or adequate system/component redundancy exists to ensure the completion of a required function.

3.6.2.2.3 Inherent Safety Features of the DAEC

There are five inherent design features that further minimize the effects of pipe failures:

1. Based on conservative piping design using proven engineering practices, the proper choice of piping materials and loadings, and conservative quality control standards and procedures for piping, materials, fabrication, and installation, it is most unlikely that pipes will catastrophically fail.

2. The structures, including the primary containment, are conservatively designed. The primary containment vessel is completely enclosed in a reinforced-concrete structure having a thickness of 4 to 7 ft.

3. A dimensionally controlled gap is provided to permit the growth of containment due to temperature and pressure. A nominal gap of 2 in. is provided up to the drywell head so that the containment may expand or deform the distance without suffering a "tear." This feature has also been evaluated in areas subject to jet force and has been determined to be adequate. Since concrete is not available at the vent openings, deflection plates have been placed across these openings for jet protection.

4. If a pipe leak should occur, means for detecting even small leaks are available so that proper action can be taken before they develop into appreciable breaks.

5. The design provides for the consideration of system redundancy to optimize the reliability of accomplishing essential functions. The drywell design has used this concept and the preliminary study to ensure that the independence of redundant systems is maintained.

3.6.2.2.4 Analysis

3.6.2.2.4.1 Jet Force. The jet force is that force resulting from the sudden expansion of steam or water through a given rupture as measured at the jet exit plane.

$$F_J = KP_VA_B$$

where

F_J = jet force

K = thrust multiplication factor to account for the effect of the change in momentum of the escaping medium. K = 1.2 for saturated steam or water

P_v = reactor vessel normal operating pressure (1050 psia)

A_B = area of break, assumed equal to free flow area of the pipe

The jet is assumed to expand in the shape of a truncated cone with a half-angle of dispersion equal to 10 degrees. This dispersion spreads the total force over a greater area as radial distance from the rupture increases. No quantitative use was made for the reduction in jet forces due to flow restrictions, and, since a loss of function was assumed, no use was made of the reduction in jet forces due to target shape.

3.6.2.2.4.2 Restraint Spacing. Pipe movement is restricted by limiting the distance between pipe restraints, where required, to some dimension less than the critical plastic hinge length of the pipe.

Restraints have been added on any pipe which, by analysis, could circumferentially fail and either hit the drywell liner with an unacceptable amount of energy or hit some component whose loss is unacceptable. The restraints added to the RHR suctions and discharge on the recirculation loop fall into this category.

1. Rupture at an elbow

Where the assumed rupture is at a pipe elbow or fitting for a circumferential pipe break, the critical plastic hinge length (L_l) is determined by the moment-resisting capabilities of the pipe (M_p) and the magnitude of the blowdown jet thrust (F_j), assuming the pipe acts as a simple cantilevered member. The critical plastic hinge length of this condition is determined as

$$L_1 = \frac{M_p}{F_J(DLF)}$$

The dynamic load factor (DLF) used in the loading design of these systems is dependent on the ratio of maximum allowable deflection to yield deflection (u), and the ratio of load duration to the period of the structure. Based on the structures and loads expected, a value of 1.25 was determined to be conservative and was used in the design.

M_p is equal to the plastic moment of the pipe based on the following formulas:*

$$M_P = \frac{Sy\ I}{R}\ 1.27 \text{ (recirculation line)}$$

$$M_P = Sy\ 4/3\ (R^3 - r^3) \text{ (other restrained piping)}$$

where

I = Moment of inertia of the pipe

$$= \frac{\pi}{4}(R^4 - r^4)$$

Sy = yield strength of the pipe material at operating conditions.

R = pipe outside radius

r = pipe inside radius

2. Longitudinal Rupture

The second case considered is a longitudinal pipe rupture located between the restraints. The most severe loading condition for this analysis is when the rupture occurs midway between the restraint brackets. The critical plastic hinge length (L_2) is analyzed if the pipe is considered a simply supported member, determined by $L_2 = \frac{M_p}{F_J(DLF)}$

The pipe restraints are designed to allow for normal pipe movements due to temperature and OBE requirements.

* The two formulas produce results that are virtually identical.

3.6.2.2.4.3 Restraint Loading. The magnitude of the pipe restraint loads for the bracket design is determined by the following formula:

$$F_2 = KPA \times C$$

where

k = thrust multiplication factor for the primary two-phase steam-water mixture of 1.20

P = operating pressure of the reactor vessel of 1050 psia

A = free flow area of the pipe

C = load factor

3.6.2.2.4.4 Recirculation Line Restraint Criteria. The recirculation piping loops are restrained against pipe movement, in the event that the recirculation-line ruptures, by a system of pipe restraint brackets. Both longitudinal (axial) and circumferential (guillotine) type pipe ruptures were considered in the design of the pipe restraint system. The pipe ruptures were assumed to occur anywhere in the system, and the consequential damage of the whipping pipe was evaluated. The restraints are located and spaced in such an arrangement on the recirculation piping system as to protect the primary containment pressure boundary, to ensure that the DBA break area is not exceeded, and to ensure essential component protection with sufficient emergency core cooling capability.

The spacing of the pipe restraints is established at less than the distance between that restraint and the primary containment drywell shell plate. The ruptured recirculation loop pipe is restrained from touching the containment shell plate everywhere within the drywell.

The pipe restraint system is also designed to limit excessive pipe movements in areas where the ruptured recirculation loop pipe could strike an adjacent pipe which, if ruptured, would exceed the DBA break area. In this analysis, the recirculation pipe is not allowed to touch an adjacent pipe of sufficient size so that the total combination of break areas of each pipe (including any other adjacent pipes which the second or third piping systems might rupture due to cascading effects) does not exceed the DBA break area. This excessive pipe movement is restricted in these areas by limiting the distance between restraints to some dimension less than the critical plastic hinge length of the pipe.

For the recirculation line, the dynamic load factor was found to vary for each pipe size, pipe system arrangement, pipe restraint location, and various other conditions such as locations of hangers, pumps, valves, etc. This value varied from 1.1 to 1.7 for various configurations modeled, and an average value of 1.50 was established to standardize restraint hardware design and construction.

The pipe restraint hardware design was established to limit the maximum allowable stresses in the bracket to 150% of AISC Code allowable and 90% of the breaking strength of the cables for this infrequent loading condition. The restraint brackets are fabricated of A-36 material, and the cables are Type 6 x 26 G IWRC extra-improved plow steel construction.

3.6.2.2.4.5 Main Steam, Feedwater, and HPCI Steam Line Restraint System. The main steam lines, feedwater lines, and HPCI steam line are restrained to prevent excessive pipe movement, that is, restraint spacing is such as to prevent the formation of a plastic hinge.

Piping restraints and supports installed for the express purpose of preventing pipe motion are designed so that the stresses in the restraint and its supporting structure remain below the material minimum specified yield stress for the combinations of live and dead loads discussed in Section 3.8.

3.6.2.2.4.6 Containment Impingement. Since the physical integrity of the containment is paramount, no pipe is permitted to strike the containment with enough energy to penetrate it. Restrained lines cannot move, and thus meet this criterion; all other lines were analyzed to ensure that a moving pipe could not travel a sufficient distance to acquire the energy required to penetrate the containment.

1. Energy imparted to a moving pipe

   a. The energy of the moving pipe at the point of contact is $E_T = F_jS$

      where

      F_j = the jet force = KP_VA_B

      S = distance traveled by the pipe, i.e., arc of motion of the pipe = $L\theta$

      L = unattached length of pipe

      θ = angle through which the pipe can move

      No credit was taken for the energy dissipated due to strain hardening.

b. The energy dissipated after the formation of the plastic hinge is

$$E_p = M_p\theta$$

c. Impact energy (E_I)

The impact energy is the energy available at the point of impact

$$E_I = E_T - E_p$$

2. Energy required to penetrate containment

The Stanford Research Institute (SRI) formula[1, 2] for the penetration of steel plates was used.

$$U = D_m T_u(O.344\ t^2 + 0.00806\ Wt)$$

where

$U =$ critical penetration energy, ft-lb_f

D_m missile diameter; when the outer circumference of the pipe is tangent to the outer surface of the plate, the missile diameter is assumed to be the chord formed by the outer surface of the plate and the outside diameter.

$= 2\ (2RT - T^2)^{1/2}$ for $R<T$

$= 2R$ for $R \Downarrow T$

$T_u =$ ultimate strength (psi) = 70,000 psi

$t =$ plate thickness (in.)

$W =$ window width = $8D_m$

$R =$ outside radius of pipe

3. Penetration of the Containment

The potential for penetration of the containment was determined by equating the energy required to penetrate the containment and the impact energy. For each given pipe size, containment thickness, and unattached length of pipe, there is a minimum distance the pipe must move to satisfy this condition. This value provided an estimate of the piping drywell separation that is critical and indicated where more detailed evaluation was required.

3.6.2.2.4.7 Separation. Maintaining the independence of redundant safety systems and components is greatly enhanced by separating the redundant components so that no single postulated event can prevent the safety-related functions from occurring.

The concept of the separation was used throughout the design of the plant. As a specific example, the emergency core cooling systems were analyzed to identify precisely wherein the redundancy exists. The core cooling function requires both depressurization of the reactor and low-pressure injection. For large breaks, the reactor will depressurize itself; for small breaks the HPCI-automatic depressurization system (ADS) combination is redundant. Low-pressure coolant injection is by the core spray-LPCI combination; for recirculation line breaks, core spray loop A, core spray loop B, and LPCI functions are redundant; for all other breaks, core spray A, core spray B, LPCI-A, and LPCI-B are redundant. The protection requirements are summarized below:

1. HPCI-ADS

    a. No small steam or water line break can be allowed to cause the loss of two ADS valves or cause the loss of high-pressure coolant injection and one ADS valve.

    b. No HPCI break can be allowed to cause the loss of any ADS valve.

    c. No intermediate size water break can be allowed to cause the loss of any ADS valve.

2. Core Spray-LPCI Systems

    a. No recirculation line or LPCI rupture can be allowed to cause the loss of either core spray loop.

    b. No core spray rupture can be allowed to cause the loss of the LPCI function.

    c. No rupture can be allowed to cause total failure of the core spray system or the loss of one core spray loop and the LPCI function simultaneously.

For any pipe break, at least one RHR loop, including its RHR heat exchanger and service water pump, must remain operable for suppression pool cooling.

For the DAEC, the following apply:

1. The small steam break is less than 0.09 ft^2 (4.1 in. I.D.).

2. The small water break is less than 0.05 ft^2 (3 in. I.D.).

3. The intermediate water break is greater than 0.05 ft^2 and less than 0.17 ft^2 (5.6 in. I.D.).

3.6.2.2.4.8 Pipe Ruptures Within the Reactor Shield. Incorporated into the design of the reactor shield is the capability to withstand, without failure, the internal pressure and coincident jet impingement loads resulting from failures of high pressure lines in the shield space region (from the outside diameter of the reactor vessel). A failure of the reactor vessel (including nozzles) is not considered credible; however, the consequences of safe-end failures are given full consideration. Safe-ends, even though attached by the reactor vessel manufacturer, are not considered to be an integral part of the reactor vessel but are regarded as a transition piece between the reactor vessel and the primary piping. Although steps have been taken to effectively preclude safe-end failures, the design criteria developed for the reactor shield consider a full spectrum of breaks up to a double-ended recirculation-line break at the nozzle to safe-end weld.

Maximum Internal Pressure Buildup

Maximum internal reactor shield static pressure, for any credible event, occurs following a circumferential rupture of the recirculation outlet line at the nozzle to safe-end weld (shown in Figure 3.6-47). This break area is 2.18 ft^2 (22-in. pipe I.D.).

The penetrations are designed to prevent a full break from pressurizing the shield space. Since this break occurs at about 12 in. from the outside diameter of the shield, only a fraction of the break flow from the cross-sectional area finds its way into the annular space between the shield and the reactor vessel. Based on the maximum permitted movement of the pipe due to limits set by the shield penetration, as shown in Figure 3.6-48, the effective area of the cross section blowing down into the annulus is 1.61 ft^2 (74% of the cross-sectional area). A conservative break size of 2.18 ft^2 was assumed.

The parameters used in the pressure response analysis of the reactor shield included a reactor cavity volume of 3136 ft^3. The vent area to the drywell was conservatively taken as 133 ft^2. This area is based on the assumption that all openings in the shield have been closed with penetration plugs similar in configuration to those represented in Figure 3.6-46. Only a portion of the openings have plugs, however, so that the actual computed vent area is approximately 269 ft^2. As a further conservatism, all vent openings were treated as orifices.

Moody's slip-flow model (Reference 3) of two-phase maximum flow with subcooled liquid was used to determine a maximum blowdown rate of 9316 lb/sec ft^2. This equates to a constant blowdown rate of 14,999 lb/sec for the 1.61 ft^2 flow area considered. The enthalpy of the flow was 529 Btu/lb.

A conservative calculation of the vent area from the shield space combined with these criteria results in a peak pressure of 18.98 psi.

Combination Jet Loads and Internal Pressure Buildup

As indicated by Figure 3.6-47, a mechanism whereby significant direct impingement loads can arise on the structural components of the recirculation line outlet shield penetration cannot be identified. However, for smaller reactor coolant lines, it is conceivable that a nozzle to safe-end weld will be located far enough inside the shield penetration plugs to allow a significant impingement load. It also should be noted that only the recirculation nozzles and the two reactor water level instrumentation nozzles at the top of the core will have shield penetration plugs. The shield openings at other nozzles away from the core region can be made large enough for inservice inspection without removable plugs.

All lines penetrating the reactor shield are listed in Table 3.6-2. For the recirculation inlet line configuration, shown in Figure 3.6-49, a jet is postulated that emanates from a double-ended rupture at the safe-end weld and impinges on the penetration shield plug in an outward manner with an average angle of approximately 31 degrees from the horizontal. As an additional margin of conservatism, it was assumed that the jet force impinging on the shield plugs results from the jet pressure times the projected cross-sectional area of the line.

The impingement force on the blockout is given by the following:

$$F_I = K_d K_t K_a P_v A_b$$

where

F_I = impingement force
K_d = dynamic load factor
K_t = thrust multiplication factor
K_a = correction for target orientation ($\cos \theta$)
P_v = pressure 1065 psia (dome pressure of 1055 psia + 10 psi hydrostatic head)
A_b = effective area of break
θ = angle of impingement

The structural rigidity of the shield plug results in a uniformly distributed jet load over the surface area of the inside face of the plug. Thus, for the recirculation inlet line break described above, a jet load of 112,590 lb is distributed over a surface area of 2000 in.^{2} . The peak stagnation pressure in the annulus resulting from 100% of this break pressurizing the shield annulus is 3.8 psi. The coincident jet and pressure loading is 60.2 psi. For smaller lines, the coincident pressure and jet force loadings are less severe. Thus, it may be concluded that an equivalent design pressure of 70 psi ensures an adequate capability to withstand credible jet force and pressure loadings without intolerable consequences, as shown on Table 3.6-3.

This configuration was conservatively analyzed to determine the capability of the shield wall to withstand pressures generated in the annulus between the reactor pressure vessel and the shield. The following criteria are used to estimate shield wall capability:

1. That only the two steel plates, acting as a thin cylindrical shell, resist the pressure forces with no credit for wide flange beam or concrete strength.

2. That the shear stress in the welds is taken as 1.5 times the normal code allowable.

3. That the pressure differential across the shield wall is a constant load although the differential pressure would continually decrease as the drywell is pressurized.

For these assumptions, the shield wall is capable of withstanding a differential pressure of 96 psig.

Based on the above, the static pressures and appropriate jet impingement pressures have been evaluated on all shield plugs. These values are shown in Table 3.6-3. The calculated values were increased by 20% to give design values with an additional margin of safety. All plugs are restrained for the design values.

3.6.2.2.5 Conclusions

Table 3.6-1 summarizes how the essential targets are protected from potential sources of pipe movement.

1. Restraint Design

The central feature of the drywell restraint system is a Vierendeel truss network located around the pressure vessel, cantilevered from the biological shield. The main steam and feedwater pipes are run between the vertical members of the truss. The HPCI steam line has ring-type restraints attached to the end vertical truss member. All restraints are anchored either to the biological shield or the truss system.

The largest loading on the truss network results from a rupture of a main steam line. For this situation, the calculated loading at the limiting point is within the allowable stress (90% of yield strength).

Figure 3.6-46, Sheets 1 through 4, provides photographs of the restraint system within the DAEC drywell.

2. Pipe Penetrations

The containment pipe penetrations are designed to withstand the normal containment environmental conditions that may prevail during plant operation and to retain their integrity during all postulated accidents.

3. Jet Force on Containment Wall

The drywell for the DAEC meets the following requirements for jet forces:

Location	Jet Force Maximum (lb)	Interior Area Subjected to Jet Force, (ft^2)	Force per Unit Area (psi)
Spherical portion of drywell	393,000	2.19 (recirc. line)	1245
Cylinder and sphere to cylinder transition	325,000	1.80 (main steam)	1245

Assuming:

a. 1050 psia reactor pressure.
b. 1.2 thrust multiplication factor.
c. Interior area = inside area of pipe considered.

4. Pipe Penetration of Containment Wall

To penetrate the containment, a pipe must acquire a kinetic energy greater than that which can be absorbed by the containment itself. The energy consideration concept is used because of the impulse-like characteristic of the impact.

Using this approach, and assuming a drywell thickness of 0.75 in. (the thinnest plate any potential source passes through), the following table has been derived:

Pipe Size (in.) Sch. 80	Distance Required To Penetrate Containment (ft)
1	26.4
2	12.1
3	8.0
4	5.7
6	3.6
8	2.6
10	2.0
12	1.6
16	1.3
20	1.0
22	0.9

For pipes with an unsupported span less than 6 ft or for thicker containment plates, the allowed travel distances are greater. These situations were evaluated on a case basis.

The above table is based on the following assumptions:

a. The length of a pipe is the straight-line distance between restraints or between penetrations about which a pipe can hinge. If a pipe ruptures circumferentially, it will travel in a path similar to an arc of a circle with the unattached length acting as the radius. The actual unattached length depends on the hinge point and the particular pipe configuration.

b. The distance required to penetrate the containment represents the distance the end of a pipe must travel to pose a hazard to containment integrity. If the end of the pipe can travel a greater distance, the pipe must be restrained.

c. A single value of the minimum distance to travel can be used for longer pipe lengths because of the asymptotic nature of the distance required versus the length of pipe curves of the various sizes of pipe.

5. Containment Overpressure

The containment will not be overpressurized as long as the combined pipe break areas are less than the design-basis accident of 2.523 ft^2. On the DAEC, only the recirculation line, main steam line, and RHR suction and discharge to the recirculation line have the minimum inside diameter that can create an overpressure condition in combination with another smaller pipe break.

Line	Nominal Diameter (in.)	Inside Diameter (Sch. 80) (in.)	Inside Diameter of Maximum Permitted Secondary Break (in.)
Recirculation line	22	19.75	8.5
Main steam line	20	18.0	11.8
RHR discharge	20	18.0	11.8
RHR suction	18	16.1	14.2

Since these lines will be restrained, they will not be able to sever another line and possibly overpressurize the containment.

6. Small Lines

Lines 1 in. or less in diameter are not considered as potential sources of pipe movement or jet spray. Based on the previously discussed criteria, a 1-in. line would have to travel 26.4 ft unimpeded to penetrate a 0.75-in. containment plate. A circumferential rupture of 1-in. diameter would cause a jet to form, which, according to the model on jet forces, will decrease to 45 psi at a distance of 1 ft from the break.

Based on the inherent design features, the conservatism used throughout the analyses as confirmed by the independent evaluation, and the restraint system used, it was found that the effects of a pipe rupture inside the drywell can be accommodated so that containment integrity and adequate emergency core cooling are ensured.

3.6.2.2.6 Reanalysis of Recirculation Piping Restraints

General Electric conducted a series of tests[5] on restraint cables to compare actual performance with predicted characteristics. The force-deflection curves for 1-in. cables were plotted and scaling relationships determined such that this data could be extrapolated to other cable sizes. This extrapolated data was then compared to the characteristics used in the previous FSAR analysis. This comparison found a significant discrepancy between assumed and actual characteristics, with actual characteristics having considerably less strain energy capability to arrest the motion of a whipping pipe.

Portions of the recirculation loop pipe-whip analysis were then reperformed to determine if the cables would still be adequate to perform their design function. Fluid blowdown forces were recalculated using current technology, as a refinement to the approximate methods used in the earlier analysis. For circumferential breaks at terminal points, simplified, nonlinear, time-history response of the pipe and cables to the revised blowdown forces was determined, using actual cable property data. This dynamic analysis found that the cables could fail under the applied loads.

Upgrade Program

Since the previous evaluation was performed, pipe rupture criteria have undergone considerable evolution. Currently, for nuclear grade piping systems, pipe ruptures are postulated on a mechanistic basis. Terminal ends, tees, and intermediate high stress points are considered the most likely locations where the pipe would break if an accident were to occur. It was thus considered prudent to upgrade those restraints, which on the basis of break probability, would be the most likely restraints to be loaded. On this basis the following restraints have been upgraded:

Nine cable restraints on the loops A & B suction, header, and riser lines were replaced by a new design which uses A304 stainless steel rods as the energy absorbing element.

Two cable restraints on the discharge risers were removed and the existing frame supports were shimmed. (These frames also function as pipe-whip restraints for the main steam pipe.) The final gap between the pipe and shimmed frame is consistent with other pipe-whip frames on the system which are structurally similar and designed for the appropriate loads.

The new pipe-whip restraints have been designed to appropriate regulatory and industry standards. All work (including material procurement, fabrication, and installation) was performed to quality standards commensurate with those for Seismic Category I structures.

Figures 3.6-50 through 3.6-54 show the new modifications for the restraints; [redacted]

3.6.3 INDEPENDENT EVALUATION OF THE MAIN STEAM AND RECIRCULATION LINE RESTRAINTS INSIDE CONTAINMENT

This section presents the results of the analytical studies performed by Nuclear Services Corporation to evaluate the adequacy of existing pipe rupture restraint systems in 1972 that were designed by Bechtel and GE for the DAEC. Restraint designs were evaluated by considering circumferential and longitudinal breaks of the main steam line for Bechtel restraints and of the recirculation lines for GE restraints.

Nonlinear dynamic analyses of the main steam and recirculation lines were performed to evaluate the adequacy of existing pipe restraints. The restraints are required to protect the primary containment from the results of postulated circumferential and longitudinal breaks in the piping systems. The time histories of maximum strain in the pipe, maximum reaction force in the restraints, and the envelope of maximum deflections of the pipe were determined.

The coupled nonlinear dynamic analysis of the pipe-restraint system takes into account the dynamic nature of the rupture force, impact effects due to the clearance between the restraint and pipe, and elastic-plastic deformation of the pipe-restraint system.

3.6.3.1 Break Locations

Break locations on the piping systems were postulated based on the following guidelines:

a. Locations that yield potential damage to the containment integrity.

b. Locations that maximize the loading on the restraint.

c. Locations that maximize the unsupported span for the pipe.

d. Locations that help to evaluate the adequacy of different designs of the restraint system, and for each design evaluate different load-resisting paths.

e. Locations that contain representative points of maximum stresses in the piping system. [redacted] The combined primary plus secondary stresses at all locations in the main steam lines due to these same load combinations are less than 1.8 S_h.

Larger numbers of break locations were postulated for the recirculation lines than for the main steam lines because of the higher stresses and greater number of different configurations of the pipe-restraint system in the recirculation lines.

Ten break locations were considered in the analyses and are described as follows:

1. Bechtel Restraint Design

   Main steam line

   a. Circumferential break at the reactor vessel nozzle (Ml).

   b. Circumferential break at the horizontal header (M2).

   c. Longitudinal break in the radial direction adjacent to restraint (M3).

   d. Longitudinal break in the tangential direction adjacent to restraint (M4).

   Refer to Figure 3.6-57 for break location.

2. General Electric Restraint Design

   Recirculation Suction Line

   a. Circumferential break at the reactor vessel nozzle (Rl).

   b. Longitudinal break in the radial direction adjacent to restraint (R2).

   c. Longitudinal break in the tangential direction adjacent to restraint (R3).

   Recirculation Discharge Line

   a. Circumferential break at the discharge elbow (R4).

   Recirculation Riser

   a. Circumferential break at the reactor vessel nozzle (R5).

   Recirculation Header

   a. Circumferential break at the riser/header tee (R6).

   Refer to Figure 3.6-58 for break locations.

3.6.3.2 Restraint System Description

3.6.3.2.1 Bechtel Restraint Design

The typical Bechtel restraint design evaluated in the pipe rupture analyses is shown in Figure 3.6-59. The restraint is a closed horseshoe-shaped structural bracket around the pipe. The bracket cross section consists of inner and outer flanges and a web, all fabricated from A-441 structural steel plates. These restraints are anchored either to the shield wall or to structural steel framing.

3.6.3.2.2 General Electric Restraint Design

The typical GE restraint design, shown in Figure 3.6-60, consists of two 6 x 26 G IWRC cables, around the pipe and anchored to a structural steel bracket at the cable anchor plate. The structural steel bracket is fabricated from A-36 steel plates. The movement of the pipe in the outward (radial) direction is resisted by the cable, whereas the structural steel bracket provides resistance to pipe movement in the lateral direction.

3.6.3.3 Design Criteria

3.6.3.3.1 Systems Criteria

The routing of the piping and the location of the pipe rupture restraints, and the structural steel floor framing providing the resistance to pipe movement, were taken from Bechtel and GE DAEC design drawings. Pipe properties, system temperature, pressure, and material specifications are summarized in Table 3.6-4 through 3.6-20. The stress-strain curves used in the analyses for the different materials are based on information from standard references. The actual and the idealized stress-strain curves for the slow rate of the loading are shown in Figures 3.6- 61 through 3.6-63. For rapid strain rate effects, the ordinates of the stress-strain curves for the ductile materials have been increased by 15% per References 6 and 7. The clearances between the piping and the restraints that are used in the analyses are summarized below.

Pipe Rupture Restraint Clearances

Location	Clearance (in.)
Main steam line and restraint	0.5
Main steam line and structural steel floor framing at [redacted]	13.0
Recirculation line and cable restraint	0.9

Recirculation line and structural steel bracket restraint	1.2
Recirculation discharge line and structural steel floor framing at [redacted]	18.0

A damping value of 2% of critical damping is used in the analyses.[8]

3.6.3.3.2 Loading Criteria

The rupture force time curves for the circumferential and longitudinal breaks in the piping systems were generated using information furnished in Reference 9, and are shown in Figures 3.6-64 through 3.6-71. The time-dependent nature of the rupture force is due to the effects of wave propagation phenomena on the piping. In general, there are three principal periods of interest as described below:

1. Initial period, from the instantaneous occurrence of the break until the flow front reaches the first change in direction in the piping system. During this period the rupture force is given by

$$F_1 = PA_e \qquad (3.6\text{-}1)$$

where

F_1 = initial rupture force

P = system normal operating pressure

A_e = effective flow area of the piping system = KCA

K = break area factor relating effective area A_e to the pipe flow area A

= 1.0 for circumferential break
= 2.0 for longitudinal break

C = discharge coefficient

= 1.0 for circumferential break
= 0.6 for longitudinal break

The time span for the initial period is given by

$$t_1 = \frac{L}{V} \qquad (3.6\text{-}2)$$

where

t_1 = duration of the initial period

L = distance from break point to first elbow for a circumferential break

= twice the break length for a longitudinal break

V = sonic velocity

= 1600 fps for steam
= 4000 fps for water

2. Intermediate period, during which time there is an unbalanced force in the piping system from the break point to the first elbow. This force is due to the momentum stored in the pipe and is given by

$$F_2 = A_e \overset{L}{\ } \frac{\delta(\rho v)}{\delta t}\, dz \qquad (3.6\text{-}3)$$

where

F_2 = rupture force during intermediate period

$$A_e \overset{L}{\ } \frac{\delta(\rho v)}{\delta t}\, dz = 0.7\, F_1 \text{ for saturated steam}$$

$$= \frac{P_{sat}}{P} F_1, \text{ for saturated water}$$

The intermediate period exists until a reflected wave returns from the reservoir. Therefore the duration for the intermediate period is given by

$$t_2 = \frac{L_2}{V} \qquad (3.6\text{-}4)$$

where

L_2 = twice the distance from the first change of direction to the reservoir

For the longitudinal breaks, there are two intermediate periods corresponding to the force time curve for each direction. The force time curve for each direction has been superimposed thus giving the overall force time history for this type of break.

3. Steady-state period takes place at the end of the intermediate period. For frictionless steady-state flow the rupture force is given by

$$F_3 = 1.26\ F_1 \text{ for saturated steam} \qquad (3.6\text{-}5)$$
$$= 1.26\ F_1 \text{ for saturated water}$$

To account for the frictional losses and the presence of flow restrictions in the pipe, the coefficients in the above equations were reduced using the procedure described in Reference 10.

The detailed calculation of the rupture force time history for various break locations is summarized in Tables 3.6-13 through 3.6-20.

3.6.3.4 Analytical Procedure

3.6.3.4.1 Description of the Pipe Rupture Phenomena

The piping system responds to the rupture by moving in the direction of the rupture force. The pipe and restraint system will undergo elastic and/or plastic deformation until either the input energy due to the time-dependent force equals the energy absorbed by the restraint system or the system fails. The elastic and/or plastic deformation of the pipe and restraint system varies in degree, depending on the geometry (spacing of the restraints, clearances, etc.) and the load- or energy-carrying capacity of the system.

3.6.3.4.2 Mathematical Model

1. Lumped Parameter Idealization

The continuous piping system was idealized as an assembly of arbitrary segments that are assumed to be rigid. The mass and mass moment of inertia of each segment were lumped at the center of gravity of each segment while the elastic properties of segments were concentrated at points between segments. Load deformation characteristics of the piping system were represented by the shear and bending springs. The stiffness properties of these springs are dependent on the effective shear area and moment of inertia of the pipe.

2. Boundary Conditions and External Restraints

The stiffness characteristics of the springs at the ends of the model represent the flexibility of the remainder of the piping system. The stiffness of piping elbows and branch connections was modified to account for local deformation effects by the flexibility factors suggested in Reference 11. The pipe rupture restraints were represented by their load deflection characteristics. The mathematical models of the pipe-restraint system for the various types of breaks analyzed are shown on Figures 3.6-72 through 3.6-79.

3.6.3.4.3 Mathematical Formulation of the Analysis

Considering the translation and rotational degrees of freedom of the nonlinear spring-mass system and assuming a viscous form of damping (velocity proportional), the equation of equilibrium is expressed in matrix form as follows:

$$MX + C^TDCX + C^T[f(CX)] = F_{(t)} \qquad (3.6\text{-}6)$$

where

M = diagonal mass matrix

X = absolute coordinate of mass m_i

C = rectangular matrix relating relative (spring) coordinate to absolute coordinate

D = diagonal matrix of damping coefficients

$f(CX)$ = force-deflection relationships for springs

$F_{(t)}$ = forcing function

The equations of motion were integrated using Newmark's "BETA" parameter method [12]. In this step-by-step integration method, the value of ß = 0.25 was selected. This corresponds to a uniform value of acceleration during the time interval to the mean of the initial and final values of acceleration. Time intervals were determined using a test on the number of significant figures desired for accuracy in the iterative procedure that converges on acceleration. The above-mentioned calculations were carried out using the NONLIN computer program[13]. The output of the analysis consists of displacements, velocities, accelerations, and spring forces as a function of time. The envelopes of these quantities were summarized at the end of the output. From the displacement time histories, the time histories of the maximum strains in the pipe were calculated.

3.6.3.4.4 Load Deflection Characteristics for Nonlinear Springs

3.6.3.4.4.1 Piping. The load deflection characteristics, for the lumped bending and shear springs representing the pipe, were calculated using the idealized stress-strain curves for the pipe material and assumed stress distribution across the pipe cross section as shown in Figures 3.6-80 and 3.6-81. In developing these load deflection characteristics, the pipe cross section was assumed to remain circular.

1. Bending Spring

The moment carrying capacity of the pipe, at the stress level corresponding to the yield and ultimate strain, is given by

$$M_p = \sigma_y z_p \qquad (3.6\text{-}7)$$

$$M_{1.0\varepsilon_u} = S\left[\left(\frac{Z_p}{S} - 1\right)\sigma_y + \sigma_{1.0\varepsilon_u}\right] \qquad (3.6\text{-}8)$$

where

M_p = plastic moment carrying capacity the pipe

$M_{1.0\varepsilon_u}$ = moment carrying capacity of the pipe at ultimate strain

σ_y = stress at yield

$\sigma_{1.0\varepsilon_u}$ = stress corresponding to ultimate strain

Z_p = plastic section modulus of the pipe
$= 4r^2t$

S = elastic section modulus of the pipe =
$\pi r^2 t$

The rotation of the bending spring for the moments calculated above is given by

$$\varnothing_p = \frac{M_p}{K_\theta} \qquad (3.6\text{-}9)$$

$$\varnothing_{1.0\varepsilon_u} = \frac{(M_{1.0\varepsilon_u} - M_p)}{E_{st}I} L + \varnothing p \qquad (3.6\text{-}10)$$

where

$\varnothing_p$ = rotation of the bending spring at yield

$\varnothing_{1.0\varepsilon_u}$ = rotation of bending spring at ultimate strain

K_θ = elastic stiffness of bending spring

$$\frac{EI}{L}$$

E = elastic modulus

L = distance between two mass points

I = moment of inertia of the pipe

E_{st} = strain hardening modulus

2. Shear Spring

The load carrying capacity of the shear spring for the shear stress distribution as shown in Figure 3.6-81 is given by

$$F_y = \swarrow_y A/2 \qquad (3.6\text{-}11)$$

$$F_{1.0\searrow u} = {}^{\swarrow}1.0_{\searrow u} A/2 \qquad (3.6\text{-}12)$$

where

F_y = shear force carrying capacity of the pipe at yield

$F_{1.0\searrow u}$ = shear force carrying capacity of the pipe at ultimate strain

A = effective shear area of the pipe cross section

The deflection of the shear spring, corresponding to its load carrying capacity as computed by equations 3.6-11 and 3.6-12, is given by

$$\delta_y = \frac{F_y}{K_s} \qquad (3.6\text{-}13)$$

$$\delta_{1.0\searrow u} = \frac{1.0_{\searrow u}}{y}\ \delta_y \qquad (3.6\text{-}14)$$

where

K_s = elastic stiffness of shear spring

= $\frac{AG}{L}$

G = shear modulus

δ_y = deflection of the shear spring at yield

$\delta_{1.0\searrow u}$ = deflection of the shear spring at ultimate strain

$\searrow u$ = ultimate strain value

3.6.3.4.4.2 Restraint System

1. Bechtel Restraint Design

   The load deflection characteristics for the typical restraint design were calculated by first idealizing the restraint system as an equivalent frame structure and then performing elastic and plastic analyses of this equivalent frame. The elastic stiffness values, used in calculating the load deflection characteristics for the longitudinal and lateral direction, were furnished by Bechtel.

2. General Electric Restraint Design

   The movement of pipe in a radial direction away from the reactor vessel is resisted by the cable. The load deflection characteristics of the restraint for this direction of pipe movement were calculated by idealizing the restraint system as a series of springs, representing the cable, cable anchor plate, restraint structure (GE), and the support structure (Bechtel), through which the restraint is anchored to the shield wall. As the load is increased, each spring will undergo elastic-plastic deformation to a different degree depending on its own load deflection property. This interaction effect between various springs was considered in calculating the total load deflection characteristics for the restraint.

   The movement of pipe in the tangential direction with respect to the reactor vessel is resisted by the structural steel bracket. The effects of combined shear and bending were included in calculating the load deflection characteristics of the restraint for this direction of pipe movement.

   The typical load deflection curves for the pipe and restraints are shown in Figures 3.6-82 through 3.6-91. The clearance between pipe and restraint is included in the load deflection curves for the restraints.

3.6.3.5 Discussion of Results

The maximum response results of the pipe and the restraint system are summarized on Tables 3.6-21 through 3.6-30. The load carrying capacity of the pipe and restraint at different strain levels is noted for comparison purposes.

3.6.3.5.1 Bechtel Restraints

The time histories of maximum strain in the pipe, maximum reaction force in the restraint, and the envelope of maximum deflections for the circumferential and longitudinal breaks of the main steam line are shown in Figures 3.6-92 through 3.6-103. The cyclic high-frequency nature of the elastic response should be noted in Figures 3.6-98, 3.6-99, 3.6-101, and 3.6-102. In addition, in comparing Figures 3.6-66 and 3.6-102, it should be noted that the maximum reaction force on the restraint is 2.7 times the rupture force. This magnification factor accounts for the impact effect on the elastic system due to the clearance between the pipe and restraint system. The summary of the response results as shown on Tables 3.6-21 through 3.6-24 indicates that the maximum strain in the pipe and restraint is within 50% of their ultimate values.

3.6.3.5.2 General Electric Restraints

1. Recirculation Suction Line

   The time histories of the pipe and restraint system response for the circumferential and longitudinal breaks of the recirculation suction line are shown in Figures 3.6-104 through 3.6-112. The response results are summarized on Tables 3.6-25 through 3.6-27. It can be seen that their maximum values are within the load carrying capacity of the pipe and the restraint system.

2. Recirculation Discharge Line

   The response results for the recirculation discharge line are shown in graphic form in Figures 3.6-113 and 3.6-114 and are summarized in tabular form on Table 3.6-28. Although the maximum strain in the pipe approaches the ultimate strain for the pipe material, the movement of the pipe will not degrade the containment integrity or the adequacy of the emergency core cooling system due to the inherent distance of the pipe's travel path from these systems.

3. Recirculation Riser

   The time histories of the response results for the recirculation riser are shown in Figures 3.6-115 and 3.6-116. The summary of the results is shown on Table 3.6-29. Although the maximum strain in the pipe approaches the ultimate strain value for the pipe material, the movement of the pipe will not degrade the containment integrity.

4. Recirculation Header

   The response results for the recirculation header are shown in graphic form in Figures 3.6-117 through 3.6-119 and are summarized in tabular form on Table 3.6-30. It can be seen that the maximum strain in the pipe and the restraint system is within 50% of their ultimate values.

REFERENCES FOR SECTION 3.6

1. C.V. Moore, "The Design of Barricades for Hazardous Pressure Systems," Nuclear Engineering and Design, Vol. 5, pp. 81-97, 1967.

2. A.L. Gluckman, "Some Notes on Dynamic Structural Problems in the Design of Nuclear BWR Stations," Nuclear Structural Engineering, Vol. 2, 1965.

3. F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," ASME Paper No. 64-HT-35.

4. Not Used.

5. General Electric Design Report 22A4046, Recirculation System Pipe Whip Restraint for BWR 4, 218 and 215 Mark I and Mark II Product Line Plant, General Electric Document No. 22A4046.

6. N. R. Rao, M. Lohrmann, and L. Tall "Effects of Strain Rate on Yield Stress of Structural Steels," Journal of Materials, 1966.

7. N. M. Newmark, Design of Structures for Dynamic Loads Including the Effects of Vibration and Ground Shock, Bechtel Corporation Lectures, 1969.

8. N. M. Newmark, Seismic Design Criteria for Nuclear Reactor Facilities, Fourth World Conference on Earthquake Engineering, Chile, 1969.

9. General Electric Company, Systems Criteria and Application for Protection Against the Dynamic Effects of Pipe Break, Document No. 22A2625, 1972.

10. F. J. Moody, Prediction of Blowdown Thrust and Jet Forces, ASME Paper No. 69HT-31, 1969.

11. American Society of Mechanical Engineers, "Nuclear Power Piping Code," ANSI B31.7, 1969.

12. N. M. Newmark, "A Method of Computation for Structural Dynamics," Journal of Engineering Mechanics Division, ASCE, 1959.

13. Nuclear Services Corporation, "NONLIN" Computer Code for Nonlinear Dynamic Analysis, 1972.

14. Generic Letter 87-11, “Relaxation in Arbitrary Intermediate Pipe Rupture Requirements,” June 19,1987.

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Table 3.6-1

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Table 3.6-2

PRINCIPAL LINES PENETRATING THE REACTOR SHIELD

Size (in.)	Number	System	Reactor Flow Direction
22	2	Recirculation	Out
10	8	Recirculation	In
10	4	Feedwater	In
8	2	Core spray	None[a]
2-1/2	1	CRD hydraulic	None[a]
2	5	Instrumentation	None[a]

Note: The jet pump instrument penetration was omitted from this listing since it does not fit the nozzle line definitions considered here. In any case, it represents no significant source for jet force.

[a] Normally no flow.

Table 3.6-3

SUMMARY OF PRESSURES ON SACRIFICIAL SHIELD PLUGS

Penetration	Static Pressure	Jet Pressure	Total Pressure	Design Pressure
Recirc. outlet N1	18.98	0.0	18.98	20 psi
Recirc. inlet N2	3.8	56.4	60.2	70 psi
Inst. N16	0.0	4.6	4.5	20 psi[a]

[a] N1 failure governs.

Table 3.6-4

PIPE DATA: SECTION PROPERTIES

Description	Schedule	Size O. D. (in.)	Wall Thickness (in.)	Weight of Pipe (lb/ft)	Weight of Water (lb/ft)	Weight of Insulation (lb/ft)	Total Weight (lb/ft)
Main steam line	80	20	1.031	209	--	24	233
Recirculation suction line	--	22	0.903	204	138.5	26	368.5
Recirculation discharge line	--	22	1.038	232	135.7	26	393.7
Recirculation riser	--	10.75	0.542	59.1	31.8	15.3	106.2
Recirculation header	--	16	0.771	125.4	71.1	20.2	216.7

PIPE DATA: MECHANICAL AND PHYSICAL PROPERTIES

Description	Normal Operating Pressure (psig)	Temp. (▪F)	Matl. ASTM Spec.	MOD. OF ELAST. E. COLD (10^6psi)	E. HOT (10^6psi)	Yield Stress at Oper. Temp. ↙$_u$ (ksi)	Ultimate Stress ↙1.0↘$_u$ (ksi)
Main steam line	1020	562	A155 KCF 70	27.9	26	29.45	47.0
Recirculation suction line	1032	532	A358 TP 304	28.3	25.8	18.8	63.0
Recirculation discharge line	1200	532	A358 TP 304	28.3	25.8	18.8	63.0
Recirculation riser	1200	535	A358 TP 304	28.3	25.8	18.8	63.0
Recirculation header	1200	535	A358 TP 304	28.3	25.8	18.8	63.0

Table 3.6-5

MAIN STEAM LINE
MATHEMATICAL MODELS (M1), (M3), AND (M4)
MASS AND STIFFNESS PROPERTIES

Mass No.	Spring No.	Elevation (ft)	Mass (lb. sec.^{2}) in.	Mass Moment of Inertia (lb sec^2 in.)	Spring Constants Lb x 10^6 in.	Spring Constants Lb in. x 10^6 rad
1		[redacted]	1.86			
2				108.5		
	1				13.28	
	2					2252.3
3			3.12			
4				659.0		
	3				8.67	
	4					1470.9
5			2.71			
6				793.4		
	5				7.46	
	6					1264.0
7			3.02			
8				1056.0		
	7				7.08	
	8					1201.0
9			3.02			
10				1056.0		
	9				6.83	
	10					1158.0
11			3.24			
12				1285.0		
	11				6.59	
	12					1117.0
13			3.24			
14				1285.0		
	13				7.42	
	14					1258.9

MAIN STEAM LINE
MATHEMATICAL MODELS (M1), (M3), AND (M4)
MASS AND STIFFNESS PROPERTIES

					Spring Constants	
Mass No.	Spring No.	Elevation (ft)	Mass $\frac{(lb. sec.^2)}{in.}$	Mass Moment of Inertia $(lb\ sec^2\ in.)$	$\frac{Lb \times 10^6}{in.}$	$\frac{Lb\ in. \times 10^6}{rad}$
15		[redacted]	2.51			
16				648.0		
	15				8.95	
	16					1517.0
17			2.26			
18				494.4		
	17				9.44	
	18					1601.0
19			2.26			
20				494.4		
	19				8.09	
	20					1372.8
21			3.02			
22				1057.0		
	21				9.44	
	22					1601.6
23			2.51			
24				189.0		
	23				0.002	
	24					0.086
	30				33.3	
	M3					
	31					231.0
	M3					
	30				1.96	
	M4					
	31					1201.2
	M4					

Table 3.6-6

MAIN STEAM LINE
MATHEMATICAL MODELS (M2)
MASS AND STIFFNESS PROPERTIES

					Spring Constants	
Mass No.	Spring No.	Elevation (ft)	Mass (lb. sec.^{2}) / in.	Mass Moment of Inertia (lb sec^2 in.)	Lb x 10^6 / in.	Lb in. x 10^6 / rad
1		[redacted]	2.51			
2				189.0		
	1				9.44	
	2					1601.6
3			3.02			
4				1057.0		
	3				8.09	
	4					1372.8
5			2.26			
6				4.94		
	5				9.44	
	6					1601.6
7			2.26			
8				494.4		
	7				8.95	
	8					1517.0
9			2.51			
10				648.0		
	9				7.42	
	10					1258.9
11			3.24			
12				1285.0		
	11				7.59	
	12					1117.0
13			3.24			
14				1285.0		

MAIN STEAM LINE
MATHEMATICAL MODELS (M2)
MASS AND STIFFNESS PROPERTIES

Mass No.	Spring No.	Elevation (ft)	Mass (lb. sec.^{2}) in.	Mass Moment of Inertia (lb sec^2 in.)	Spring Constants Lb x 10^6 in.	Spring Constants Lb in. x 10^6 rad
	13	[redacted]			6.83	
	14					1158.0
15			3.02			
16				1056.0		
	15				7.08	
	16					1201.0
17			3.02			
18				1056.0		
	17				7.46	
	18					1264.0
19			2.71			
20				793.4		
	19				8.67	
	20					1470.9
21			3.12			
22				659.0		
	21				13.28	
	22					2252.3
23			1.86			
24				108.5		
	23				1.96	
	24					1201.2

Table 3.6-7

RECIRCULATION SUCTION LINE
MATHEMATICAL MODEL (R1)
MASS AND STIFFNESS PROPERTIES

Mass No.	Spring No.	Elevation (ft)	Mass (lb. sec.^{2}) / in.	Mass Moment of Inertia (lb sec^2 in.)	Spring Constants: Lb x 10^6 / in.	Spring Constants: Lb in. x 10^6 / rad
1		[redacted]	6.32			
2				825.40		
	1				14.9	
	2					3121.9
3			3.93			
4				513.30		
	3				13.7	
	4					2861.7
5			2.387			
6				311.70		
	5				7.68	
	6					1604.7
7			5.728			
8				2792.80		
	7				6.23	
	8					1300.87
9			4.773			
10				1696.90		
	9				6.85	
	10					1460.87
11			4.773			
12				1696.90		
	11				6.85	
	12					1430.87
13			4.773			
14				1696.90		
	13				6.85	
	14					1430.87
15			4.773			
16				1696.90		
	15				6.85	
	16					1430.87
17			4.773			
18				1696.90		
	17				8.74	
	18					1826.6
19			2.386			
20				311.60		
	19				0.067	
	20					295.0

Table 3.6-8

RECIRCULATION SUCTION LINE
MATHEMATICAL MODEL (R2)
MASS AND STIFFNESS PROPERTIES

Mass No.	Spring No.	Elevation (ft)	Mass (lb. $sec.^2$) in.	Mass Moment of Inertia (lb sec^2 in.)	Spring Constants Lb x 10^6 in.	Spring Constants Lb in. x 10^6 rad
1		[redacted]	6.32			
2				825.40		
	1				14.9	
	2					3121.9
3			3.93			
4				513.30		
	3				13.7	
	4					2861.7
5			2.387			
6				311.70		
	5				7.68	
	6					1604.7
7			5.728			
8				2792.80		
	7				6.23	
	8					1300.87
9			4.773			
10				1696.90		
	9				6.85	
	10					1430.87
11			4.773			
12				1696.90		
	11				6.85	
	12					1430.87
13			4.773			
14				1696.90		
	13				6.85	
	14					1430.87
15			4.773			
16				1696.90		
	15				6.85	
	16					1430.87
17			4.773			
18				1696.90		
	17				8.74	
	18					1826.6
19			2.386			
20				311.30		
	19				0.067	
	20					295.0
	25				3.5	
	26					508.0

Table 3.6-9

RECIRCULATION SUCTION LINE
MATHEMATICAL MODEL (R3)
MASS AND STIFFNESS PROPERTIES

Mass No.	Spring No.	Elevation (ft)	Mass (lb. sec.^{2}) / in.	Mass Moment of Inertia (lb sec^2 in.)	Spring Constants: Lb x 10^6 / in.	Spring Constants: Lb in. x 10^6 / rad
1		[redacted]	6.32			
2				825.40		
	1				14.9	
	2					3121.9
3			3.93			
4				513.30		
	3				13.7	
	4					2861.7
5			2.387			
6				311.70		
	5				7.68	
	6					1604.7
7			5.728			
8				2792.80		
	7				6.23	
	8					1300.87
9			4.773			
10				1696.90		
	9				6.85	
	10					1430.87
11			4.773			
12				1696.90		
	11				6.85	
	12					1430.87
13			4.773			
14				1696.90		
	13				6.85	
	14					1430.87
15			4.773			
16				1696.90		
	15				6.85	
	16					1430.87
17			4.773			
18				1696.90		
	17				8.74	
	18					1826.6
19			2.386			
20				311.60		
	19				0.067	
	20					295.0
	25				0.59	
	26					508.0

Table 3.6-10

RECIRCULATION SUCTION LINE
MATHEMATICAL MODEL (R4)
MASS AND STIFFNESS PROPERTIES

Mass No.	Spring No.	Elevation (ft)	Mass (lb. sec.^{2}) in.	Mass Moment of Inertia (lb sec^2 in.)	Spring Constants Lb x 10^6 in.	Spring Constants Lb in. x 10^6 rad
1		[redacted]	4.96			
2		[redacted]		647.80		
	1	[redacted]			15.65	
	2	[redacted]				3227.2
3		[redacted]	2.55			
4		[redacted]		333.03		
	3	[redacted]			15.65	
	4	[redacted]				3227.2
5		[redacted]	2.55			
6		[redacted]		333.03		
	5	[redacted]			15.65	
	6	[redacted]				3227.2
7		[redacted]	2.55			
8		[redacted]		333.03		
	7	[redacted]			10.47	
	8	[redacted]				2151.5
9		[redacted]	5.10			
10		[redacted]		1813.56		
	9	[redacted]			7.82	
	10	[redacted]				1613.6
11		[redacted]	5.10			
12		[redacted]		1813.56		
	11	[redacted]			8.53	
	12	[redacted]				1760.0
13		[redacted]	4.25			
14		[redacted]		1122.0		
	13	[redacted]			0.013	
	14					2120.0

Table 3.6-11

RECIRCULATION SUCTION LINE
MATHEMATICAL MODEL (R5)
MASS AND STIFFNESS PROPERTIES

					Spring Constants	
Mass No.	Spring No.	Elevation (ft)	Mass $\frac{(\text{lb. sec.}^2)}{\text{in.}}$	Mass Moment of Inertia (lb sec^2 in.)	$\frac{\text{Lb} \times 10^6}{\text{in.}}$	$\frac{\text{Lb in.} \times 10^6}{\text{rad}}$
1		[redacted]	.645			
2				67.2		
	1				3.51	
	2					230.77
3			.549			
4				165.2		
	3				3.73	
	4					243.3
5			.549			
6				165.2		
	5				3.73	
	6					243.3
7			.549			
8				165.2		
	7				3.73	
	8					243.3
9			.549			
10				165.2		
	9				3.73	
	10					243.3
11			.549			
12				165.2		
	11				3.73	
	12					243.3
13			.549			
14				165.2		
	13				3.73	
	14					243.3
15			.549			
16				165.2		
	15				0.004	
	16					21.7

Table 3.6-12

RECIRCULATION SUCTION LINE
MATHEMATICAL MODEL (R6)
MASS AND STIFFNESS PROPERTIES

					Spring Constants	
Mass No.	Spring No.	Elevation (ft)	Mass (lb. sec.^{2}) in.	Mass Moment of Inertia (lb sec^2 in.)	Lb x 10^6 in.	Lb in. x 10^6 rad
1		[redacted]	.888			
2				52.461		
	1				10.01	
	2					1452.20
3			.5614			
4				23.015		
	3				15.85	
	4					2299.32
5			.5614			
6				23.015		
	5				15.85	
	6					2299.32
7			.5614			
8				23.015		
	7				15.85	
	8					2299.32
9			.5614			
10				23.015		
	9				15.85	
	10					2299.32
11			.4679			
12				17,465		
	11				19.02	
	12					2759.18
13			.5614			
14				23.015		
	13				15.85	
	14					2299.32
15			.5614			
16				23.015		
	15				15.85	
	16					2299.32
17			.5614			
18				23.015		
	17				15.85	
	18					2299.32
19			.795			
20				42.197		
	19				0.72	
	20					479.0

Table 3.6-13

MAIN STEAM LINE
CIRCUMFERENTIAL BREAK M1 AT REACTOR VESSEL NOZZLE
RUPTURE FORCE TIME HISTORY

Data: P_o = 1020 psi
A = 252.7 in.^{2}, K = 1.0; C = 1.0; A_e = KCA = 252.7 in.^{2}
A_R = Restriction flow area (velocity limiter) = 0.25A
L_1 = Distance from break to first elbow = 4.25 ft
L_2 = Twice the distance from break to first change of direction (at velocity limiter) = 174 ft

Assumptions:
1. Neglect frictional losses
2. Choking occurs at the velocity limiter

a. Initial Thrust

$$F_1 = 1.0\ P_o\ A_e = \frac{1020 \times 252.7}{1000} = 258 \text{ kips}$$

(Table 11-1, Reference 9)

$$t_1 = \frac{L_1}{V} = \frac{4.25}{1600} = 0.0027 \text{ sec}$$

b. Intermediate Thrust

$F_2 = 0.7\ P_oA_e = 0.7 \times 2.58 = 180$ kips (Figure 11-6B, Reference 9)

$$t_2 = \frac{L_2}{V} + t_1 = \frac{174}{1600} + 0.0027 = 0.11 \text{ sec}$$

c. Steady-state Thrust

$F_3 = 0.4\ P_oA_e = 0.4 \times 258 = 103$ kips (Figure 11-6B, Reference 9)

Table 3.6-14

MAIN STEAM LINE
CIRCUMFERENTIAL BREAK M2 AT HEADER
RUPTURE FORCE TIME HISTORY

Data: P_o = 1020 psi
A = 252.7 in.^{2}; K = 1.0; A_e = KCA = 252.7 in.^{2}
L_1 = Distance from break to first elbow = 3.0 ft
L_2 = Twice the distance form break to first change of direction (at reactor vessel) = 100 ft

a. Initial Thrust

$$F_1 = 1.0\, P_oA_e = \frac{1.0 \times 1020 \times 252.7}{1000} = 258 \text{ kips}$$

(Table 11-1, Reference 9)

$$t_1 = \frac{3.0}{1600} = 0.00188 \text{ sec}$$

b. Intermediate Thrust

$F_2 = 0.7P_oA_e = 0.7 \times 258 = 180$ kips (Figure 11-6B, Reference 9)

$$t_2 = \frac{L_2}{V} + t_1 = \frac{100}{1600} + 0.0018 = 0.0625 \text{ sec}$$

c. Steady-State Thrust

Friction losses:

Pipe (50 ft length) $\frac{L}{D} = 33$

Three elbows (3 x 20) = $\underline{60}$

$$\nearrow \frac{L}{D} = 99$$

$$\nearrow \frac{FL}{D} = 0.012 \times 99 = 1.2$$

$$F_3 = \frac{160{,}000}{144} \times 252.7 = 281 \text{ kips (Figure 11-4, Reference 9)}$$

MAIN STEAM LINE
LONGITUDINAL BREAK M3 & M4
RUPTURE FORCE TIME HISTORY

Data: $P_o =$ 1020 psi

$A =$ 252.7 in.^{2}; K = 2.0 ; C = 0.6; $A_e = KCA = 303$ in.^{2}

$A_R =$ Restriction flow area (velocity limiter) = 0.25 A

$L_1 =$ Twice the break length = 2 x 5 = 10 ft

$L_{21} =$ Twice the distance form break to reactor vessel = 2 x 20 = 40 ft

$L_{22} =$ Twice the distance form break to velocity limiter = 134 ft

Assumptions:
1. Neglect frictional losses
2. Choking occurs at the velocity limiter

a. Initial Thrust

$$F_1 = P_oA_e = \frac{1020 \times 303}{1000} = 309 \text{ kips}$$

$$t_1 = \frac{10}{1600} = 0.0063 \text{ sec}$$

b. Intermediate Thrust

First intermediate thrust from reactor vessel side

$F_{21} =$ 0.7 P_oA_e = 0.7 x 309 = 216 kips (Figure 11-6B, Reference 9)

$$t_{21} = \frac{L_{21}}{V} + t_1 = \frac{40}{1600} + 0.0063 = 0.031 \text{ sec}$$

Second intermediate thrust from reactor vessel side and velocity limiter side

$F_{22} =$ 1/2(1.26 P_oA_e + 0.7 P_oA_e)
(Table 11-2 and Figure 11-6B, Reference 9)

$$F_{22} = 1/2\ 1.96 \times \frac{1020 \times 303}{1000} = 303 \text{ kips}$$

$$t_{22} = \frac{L_{22}}{V} + t_1 = \frac{134}{1600} + 0.0063 = 0.09 \text{ sec}$$

MAIN STEAM LINE
LONGITUDINAL BREAK M3 & M4
RUPTURE FORCE TIME HISTORY

c. Steady-State Thrust

Combination of steady-state thrust from reactor vessel side and velocity limiter side

$F_3 = 1/2(1.26\ P_oA_e + 0.4\ P_oA_e)$
(Table 11-2 and Figure 11-6B, Reference 9)

$= 1/2\ \frac{1.66 \times 1020 \times 303}{1000} = 257$ kips

Table 3.6-16

RECIRCULATION SUCTION LINE
CIRCUMFERENTAIL BREAK R1 AT REACTOR VESSEL NOZZLE
RUPTURE FORCE TIME HISTORY

Data: P_o = 1032 psi; T = 532▪F

P_{sat} = 900 psi; h = 526; h_{sat} = 547; h_{sub} = 21

A = 320 in.^{2}; K = 1.0; C= 1.0; A_e = KCA = 320 in.^{2}

A_R = Restriction flow area (jet pumps) = 55.4 in.^{2}

L_1 = Distance from break to first elbow = 6 ft

L_2 = Twice the distance from break to first change of direction (at jet pumps) = 212 ft

Assumptions:
1. Saturated water blowdown since $h_{sub} < 22$ (page 11-7, Reference 9)
2. Choking occurs at jet pump
3. Neglect frictional losses in the line
4. Neglect losses in recirculation pump

a. Initial Thrust

$$F_1 = 1.0\, P_o A_e = \frac{1032 \times 320}{1000} = 330 \text{ kips}$$ (Table 11-1, Reference 9)

$$t_1 = \frac{L_1}{V} = \frac{6}{4000} = 0.0015 \text{ sec}$$

b. Intermediate Thrust

$$F_2 = \frac{P_{sat}}{P_o} F_1 = \frac{900}{1032} \times 330 = 288 \text{ kips}$$ (Table 11-1, Reference 9)

$$t_2 = \frac{L_2}{V} + t_1 = \frac{212}{4000} + 0.0015 = 0.0545 \text{ sec}$$

c. Steady-State Thrust

$$F_3 = 0.72\, P_o A_e = \frac{0.72 \times 1032 \times 320}{1000} = 238 \text{ kips}$$ (Figure 11-5, Reference 9)

RECIRCULATION SUCTION LINE
LONGITUDINAL BREAK R2 AND R3
RUPTURE FORCE TIME HISTORY

Data: P_o = 1032 psi; T = 532▪F

P_{sat} = 900 psi; h = 526 Btu/lb; h_{sat} = 547; h_{sub} = 21

A = 320 in.^{2}, K = 2.0; C = 0.6; A_e = KCA = 384 in.^{2}

A_R = Restriction flow area (jet pumps) = 55.4 in.^{2}

L_1 = Twice the break length = 10 ft

L_{21} = Twice the distance from break to reactor vessel = 40 ft

L_{22} = Twice the distance from brake to jet pump = 180 ft

Assumptions:
1. Saturated water blowdown since $h_{sub} < 22$
2. Choking occurs at jet pump
3. Neglect frictional losses in the line
4. Neglect losses in recirculation pump

a. Initial Thrust

$F_1 = P_oA_e \frac{1032 \times 384}{1000} = 396$ kips (Table 11-1, Reference 9)

$t_1 = \frac{L_1}{V} = \frac{10}{4000} = 0.0025$ sec

b. Intermediate Thrust

First intermediate thrust

$F_{21} = \frac{P_{sa}}{P_o} F_1$ (Table 11-1, Reference 9)

$= \frac{900}{1032} \times 396 = 346$ kips

$t_{21} = \frac{L_{21}}{V} + t_1 = \frac{40}{4000} + 0.0025 = 0.0125$ sec

RECIRCULATION SUCTION LINE
LONGITUDINAL BREAK R2 AND R3
RUPTURE FORCE TIME HISTORY

Second intermediate thrust from reactor vessel nozzle side and jet pump side

$$F_{22} = \frac{1.26}{2} P_o A_e + \frac{P_{sat}}{P_o} \frac{F_1}{2} \quad \text{(Table 11-2 and Table 11-1, Reference 9)}$$

$$= \frac{1.26 \times 1032 \times 384}{2 \times 1000} + \frac{900}{1032} \times \frac{396}{2}$$

$$= 250 + 173 = 423 \text{ kips}$$

$$t_{22} = \frac{180}{4000} + 0.0025 = 0.0475 \text{ sec}$$

c. Steady-State Thrust

Combination of thrust from reactor vessel side and jet pump side

$$F_3 = 1.26 \frac{P_o A_e}{2} + 0.72 \frac{P_o A_e}{2} \quad \text{(Table 11-2 and Figure 11-5, Reference 9)}$$

$$= \frac{1.26 \times 1032 \times 384}{2 \times 1000} + \frac{0.72 \times 1032 \times 384}{2 \times 1000}$$

$$= 250 + 142 = 392 \text{ kips}$$

Table 3.6-18

RECIRCULATION DISCHARGE LINE
CIRCUMFERENTIAL BREAK R4 AT DISCHARGE ELBOW
RUPTURE FORCE TIME HISTORY

Data: P_o = 1032 + pressure rises in the pump = 1200 psi

T = 535▪F

P_{sat} = 923 psi, h = 530; h_{sat} = 528; h_{sub} = 2

A = 311.6 in.^{2}; K = 1.0; C = 1.0; A_e = KCA = 311.6 in.^{2}

A_R = Restriction flow area (jet pump) = 55.4 in.^{2}

L_1 = Distance from break to first elbow = 3 ft

L_2 = Twice the distance form break to first change of direction (at jet pump) = 88 ft

Assumptions:

1. Saturated water blowdown since h sub < 22 (page 11-7, Reference 9)
2. Neglect frictional losses in the line
3. Choking occurs at jet pumps

a. Initial Thrust

$F_1 = P_oA_e = \frac{1200 \times 311.6}{1000} = 374$ kips (Table 11-1, Reference 9)

$t_1 = \frac{3}{4000} = 0.0008$ sec

b. Intermediate Thrust

$F_2 = \frac{P_{sat}}{P_o} F_1 = \frac{923}{1200} \times 374 = 287$ kips (Table 11-1, Reference 9)

$t_2 = \frac{88}{4000} + 0.0008$ sec = 0.0228 seconds

c. Steady-State Thrust

$F_3 = 0.72\ F_1$ (Figure 11-5, Reference 9)

$F_3 = 0.72 \times 374 = 269$ kips

Table 3.6-19

RECIRCULATION RISER
CIRCUMFERENTAIL BREAK R5 AT REACTOR VESSEL NOZZLE
RUPTURE FORCE TIME HISTORY

Data: P_o = 1200 psi; T = 532▪F

h = 530; P_{sat} = 923 psi; h_{sat} = 528; h_{sub} = 2

A = 73.34 in.^{2}; K = 1.0; C = 1.0

A_e = KCA = 73.34 in.^{2}

L_1 = Distance from break to first elbow = 1.5 ft

L_2 = Twice the distance from break to first change of direction (at recirculation header) = 31 ft

Assumptions:
1. Saturated water blowdown
2. Neglect frictional losses in the line
3. Recirculation header as the reservoir

a. Initial Thrust

$F_1 = P_oA_e$ (Table 11-1, Reference 9)

$$F_1 = \frac{1200 \times 73.34}{1000} = 88 \text{ kips}$$

$$t_1 = \frac{1.5}{4000} = 0.0004 \text{ sec}$$

b. Intermediate Thrust

$$F_2 = \frac{P_{sat}}{P_o} F_1 \text{ (Table 11-1, Reference 9)}$$

$$= \frac{923}{1200} \times 88 = 68 \text{ kips}$$

$$t_2 = \frac{31}{4000} + 0.0004 = 0.00815 \text{ sec}$$

c. Steady-State Thrust

$F_3 = 1.26\ P_oA_e$ (Table 11-2, Reference 9)

$$= 1.26 \times \frac{1200 \times 73.34}{1000} = 111 \text{ kip}$$

Table 3.6-20

RECIRCULATION HEADER
CIRCUMFERENTIAL BREAK R6 AT RISER/HEADER TEE
RUPTURE FORCE TIME HISTORY

Data:
P_o = 1200 psi; T = 532▪F
P_{sat} = 923 psi; j = 530; h_{sat} = 528; h_{sub} = 2
A = 73.34 $in.^2$; K = 1.0;C = 1.0
A_e = KCA = 73.34 $in.^2$
L1 = Distance from break to header

Assumptions:
1. Saturated water blowdown
2. Recirculation header as the reservoir
3. Neglect intermediate thrust

a. Initial Thrust

$F_1 = P_oA_e$ (Table 11-1, Reference 9)

$$F_1 = \frac{1200 \times 73.34}{1000} = 88 \text{ kips}$$

$$t_1 = \frac{1.5}{4000} = 0.0004 \text{ sec}$$

b. Steady-State Thrust

$F_3 = 1.26\ P_oA_e$ (Table 11-2, Reference 9)

$$= \frac{1.26 \times 1200 \times 73.34}{1000} = 111 \text{ kips}$$

Table 3.6-21

RESTRAINT DESIGN - BECHTEL
MAIN STEAM LINE
CIRCUMFERENTIAL BREAK (M1) AT REACTOR VESSEL NOZZLE

Parameter	0.5-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	13.5×10^3
Percent of moment carrying capacity at yield ($M_p = 12.6 \times 10^3$ kips in.)	107%
Percent of moment carrying capacity at 1.0➘u ($M_{1.0➘u} = 17.5 \times 10^3$ kips in.)	77%
Maximum strain level (in./in.)	0.0343
Percent of ultimate strain (➘u = 0.18 in./in.)	20%
Maximum deflection (in.)	18
Restraint	
Maximum restraint load (kips)	615
Percent of load carrying capacity of restraint at yield (930 kips)	66%
Percent of load carrying capacity of restraint at 1.0➘u (1190 kips)	52%
Maximum strain level (in./in.)	0.0013
Percent of ultimate strain (➘u = 0.18 in./in.)	0.72%

Table 3.6-22

RESTRAINT DESIGN - BECHTEL
MAIN STEAM LINE
CIRCUMFERENTIAL BREAK (M2) AT HORIZONTAL HEADER

Parameter	0.5-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	13.3×10^3
Percent of moment carrying capacity at yield ($M_p = 12.6 \times 10^3$ kips in.)	106%
Percent of moment carrying capacity at 1.0➘u ($M_{1.0➘u} = 17.5 \times 10^3$ kips in.)	76%
Maximum strain level (in./in.)	0.0213
Percent of ultimate strain (➘u = 0.18 in./in.)	12%
Maximum deflection (in.)	14
Restrainment	
Maximum restraint load (kips)	646
Percent of load carrying capacity of restraint at yield (1150 kips)	56%
Percent of load carrying capacity of restraint at 1.0➘u (1344 kips)	48%
Maximum strain level (in./in.)	0.0012
Percent of ultimate strain (➘u = 0.18 in./in.)	0.67%

Table 3.6-23

RESTRAIN DESIGN - BECHTEL
MAIN STEAM LINE
LONGITUDINAL BREAK (M3) IN RADIAL DIRECTION

Parameter	0.5-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	12.0×10^3
Percent of moment carrying capacity at yield ($M_p = 12.6 \times 10^3$ kips in.)	95%
Percent of moment carrying capacity at 1.0➘u ($M_{1.0➘u} = 17.5 \times 10^3$ kips in.)	69%
Maximum strain level (in./in.)	0.0014
Percent of ultimate strain (➘u = 0.18 in./in.)	0.8%
Maximum deflection (in.)	0.69
Restraint	
Maximum restraint load (kips)	793
Percent of load carrying capacity of restraint at yield (930 kips)	85%
Percent of load carrying capacity of restraint at 1.0 u (1190 kips)	67%
Maximum strain level (in./in.)	0.0017
Percent of ultimate strain (➘u = 0.18 in./in.)	0.94%

Table 3.6-24

RESTRAINT DESIGN - BECHTEL
MAIN STEAM LINE
LONGITUDINAL BREAK (M4) IN TANGENTIAL DIRECTION

Parameter	0.5-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	12.0×10^3
Percent of moment carrying capacity at yield ($M_p = 12.6 \times 10^3$ kips in.)	95%
Percent of moment carrying capacity at $1.0\varepsilon_u$ ($M_{1.0\varepsilon_u} = 17.5 \times 10^3$ kips in.)	69%
Maximum strain level (in./in.)	0.0009
Percent of ultimate strain ($\varepsilon_u = 0.18$ in./in.)	0.6%
Maximum deflection (in.)	0.72
Restraint	
Maximum restraint load (kips)	850
Percent of load carrying capacity of restraint at yield (1050 kips)	81%
Percent of load carrying capacity of restraint at $1.0\varepsilon_u$ (1344 kips)	63%
Maximum strain level (in./in.)	0.0016
Percent of ultimate strain ($\varepsilon_u = 0.18$ in./in.)	0.89%

Table 3.6-25

RESTRAINT DESIGN - GENERAL ELECTRIC
RECIRCULATION SUCTION LINE
CIRCUMFERENTIAL BREAK (R1) AT REACTOR VESSEL NOZZLE

Parameter	0.9-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	15.38×10^3
Percent of moment carrying capacity at 1.0➘u ($M_{1.0➘u} = 24.7 \times 10^3$ kips in.)	62%
Maximum strain level (in./in.)	0.124
Percent of ultimate strain (➘u = 0.39 in./in.)	32%
Maximum deflection (in.)	20.3
Restraint	
Cable	
Maximum restraint load (kips)	554
Maximum load/cable (kips)	148
Percent of breaking strength (159 kips)	93%
Percent of ultimate strength (173 kips)	86%
Cable anchor plate	
Maximum load/plate (kips)	148
Percent of load carrying capacity at 1.0➘u (196 kips)	76%
Maximum strain level (in./in.)	0.075
Percent of ultimate strain (➘u = 0.18 in./in.)	42%

Table 3.6-26

RESTRAINT DESIGN - GENERAL ELECTRIC
RECIRCULATION SUCTION LINE
LONGITUDINAL BREAK (R2) IN RADIAL DIRECTION

Parameter	0.9-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	8.8 x 10_3
Percent of moment carrying capacity at $1.0\varepsilon u$ ($M_{1.0\varepsilon u}$ = 24.7 x 10^3 kips in.)	36%
Maximum strain level (in./in.)	0.0019
Percent of ultimate strain (εu = 0.39 in./in.)	0.5%
Maximum deflection (in.)	1.6
Restraint	
Cable	
Maximum restraint load (kips)	536
Maximum load/cable (kips)	143
Percent of breaking strength (159 kips)	90%
Percent of ultimate strength (173 kips)	83%
Cable anchor plate	
Maximum load/plate (kips)	143
Percent of load carrying capacity at $1.0\varepsilon u$ (196 kips)	73%
Maximum strain level (in./in.)	0.056
Percent of ultimate strain (εu = 0.18 in./in.)	31%

Table 3.6-27

RESTRAINT DESIGN - GENERAL ELECTRIC
RECIRCULATION SUCTION LINE
LONGITUDINAL BREAK (R3) IN TANGENTIAL DIRECTION

Parameter	0.9-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	11.9×10^3
Percent of moment carrying capacity at yield ($M_p = 8.7 \times 10^3$ kips in.)	137%
Percent of moment carrying capacity at 1.0➘u ($M_{1.0➘u} = 24.7 \times 10^3$ kips in.)	48%
Maximum strain level (in./in.)	0.00087
Percent of ultimate strain (➘u = 0.39 in./in.)	0.2%
Maximum deflection (in.)	2.1
Restraint	
Maximum restraint load (kips)	518
Percent of load carrying capacity of restraint at yield (492 kips)	105%
Percent of load carrying capacity of restraint at 1.0➘u (917 kips)	56%
Maximum strain level (in./in.)	0.044
Percent of ultimate strain (➘u = 0.18 in./in.)	24%

Table 3.6-28

RESTRAINT DESIGN - GENERAL ELECTRIC
RECIRCULATION DISCHARGE LINE
CIRCUMFERENTIAL BREAK (R4) AT DISCHARGE ELBOW

Parameter	1.2-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	Load carrying capacity of pipe exceeded.[a]
Percent of moment carrying capacity at 1.0➘u ($M_{1.0➘u}$ = 28.0 x 10^3 kips in.)	[a]
Maximum strain level (in./in.)	[a]
Percent of ultimate strain (➘u = 0.39 in./in.)	[a]
Restraint	
Maximum restraint load (kips)	≅500
Percent of load carrying capacity of restraint at yield (492 kips)	≅102%
Percent of load carrying capacity of restraint at 1.0➘u (917 kips)	≅54%
Maximum strain level (in/in.)	0.031
Percent of ultimate strain (➘u = 0.18 in./in.)	17%

[a] Due to the high local strain and collapse of the pipe cross section, as the load carrying capacity of pipe is exceeded, the pipe will hinge around the restraint traveling in a path similar to an arc of a circle with the unattached length acting as the radius. This travel path of the recirculation discharge line will not degrade the containment integrity or the adequacy of the emergency core cooling system.

Table 3.6-29

RESTRAINT DESIGN - GENERAL ELECTRIC
RECIRCULATION RISER
CIRCUMFERENTIAL BREAK (R5) AT REACTOR VESSEL NOZZLE

Parameter	0.9-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	Load carrying capacity of pipe exceeded.[a]
Percent of moment carrying capacity at 1.0➘u ($M_{1.0➘u}$ = 3.4 x 10^3 kips in.)	[a]
Maximum strain level (in./in.)	[a]
Percent of ultimate strain (➘u = 0.39 in./in.)	[a]
Restraint	
Cable	
Maximum restraint load (kips)	220
Maximum load/cable (kips)	64
Percent of breaking strength (79 kips)	81%
Percent of ultimate strength (86 kips)	74%
Cable anchor plate	
Maximum load/plate (kips)	64
Percent of load carrying capacity at 1.0➘u (94 kips)	68%
Maximum strain level (in./in.)	0.038
Percent of ultimate strain (➘u = 0.18 in./in.)	21%

[a] Due to the high local strain and collapse of the pipe cross section, as the load carrying capacity of pipe is exceeded, the pipe will hinge around the restraint traveling in a path similar to an arc of a circle with the unattached length acting as the radius. This travel path of the recirculation risers will not degrade the containment integrity or the adequacy of the emergency core cooling system.

Table 3.6-30

RESTRAINT DESIGN - GENERAL ELECTRIC
RECIRCULATION HEADER
CIRCUMFERENTIAL BREAK (R6) AT RISER/HEADER TEE

Parameter	1.2-in. Clearance 2% Damping
Pipe	
Maximum moment (kips in.)	7.7×10^3
Percent of moment carrying capacity at yield ($M_p = 3.9 \times 10^3$ kips in.)	197%
Percent of moment carrying capacity at 1.0➘u ($M_{1.0\searrow u} = 11.0 \times 10^3$ kips in.)	70%
Maximum strain level (in./in.)	0.145
Percent of ultimate strain (➘u = 0.18 in./in.)	37%
Maximum deflection (in.)	37.8
Restraint	
Maximum restraint load (kips)	291
Percent of load carrying capacity of restraint at yield (283 kips)	103%
Percent of load carrying capacity of restraint at 1.0➘u (440 kips)	66%
Maximum strain level (in./in.)	0.038
Percent of ultimate strain (➘u = 0.18 in./in.)	21%

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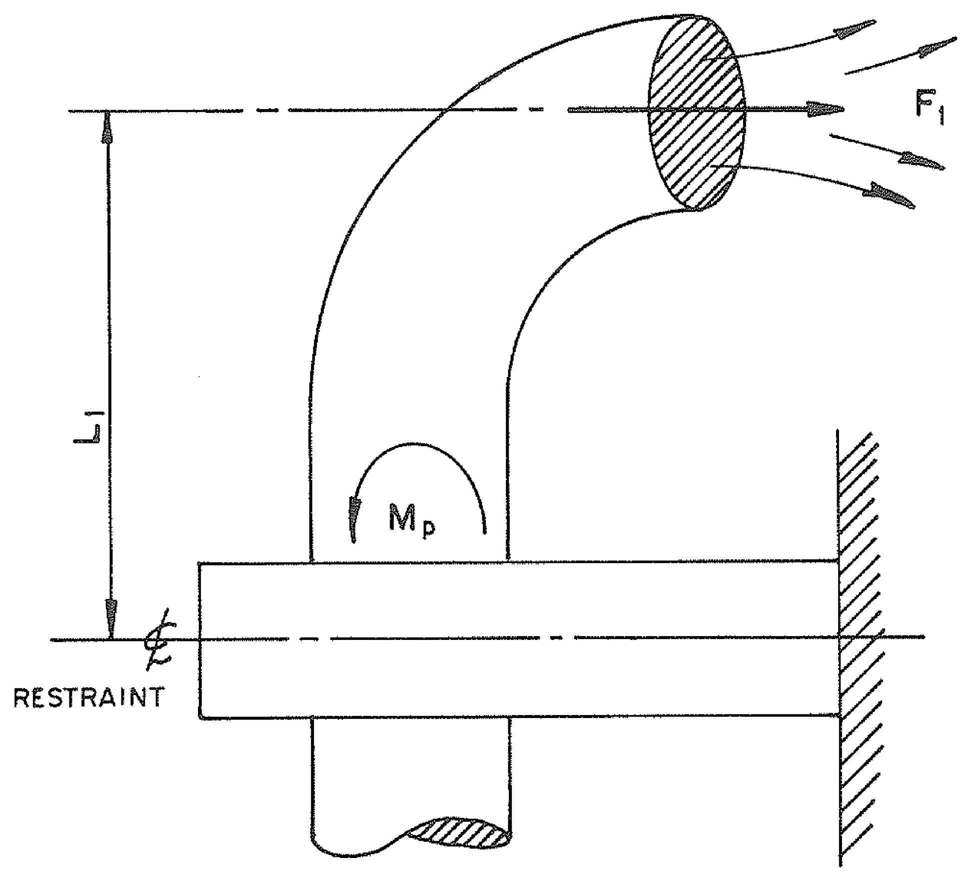

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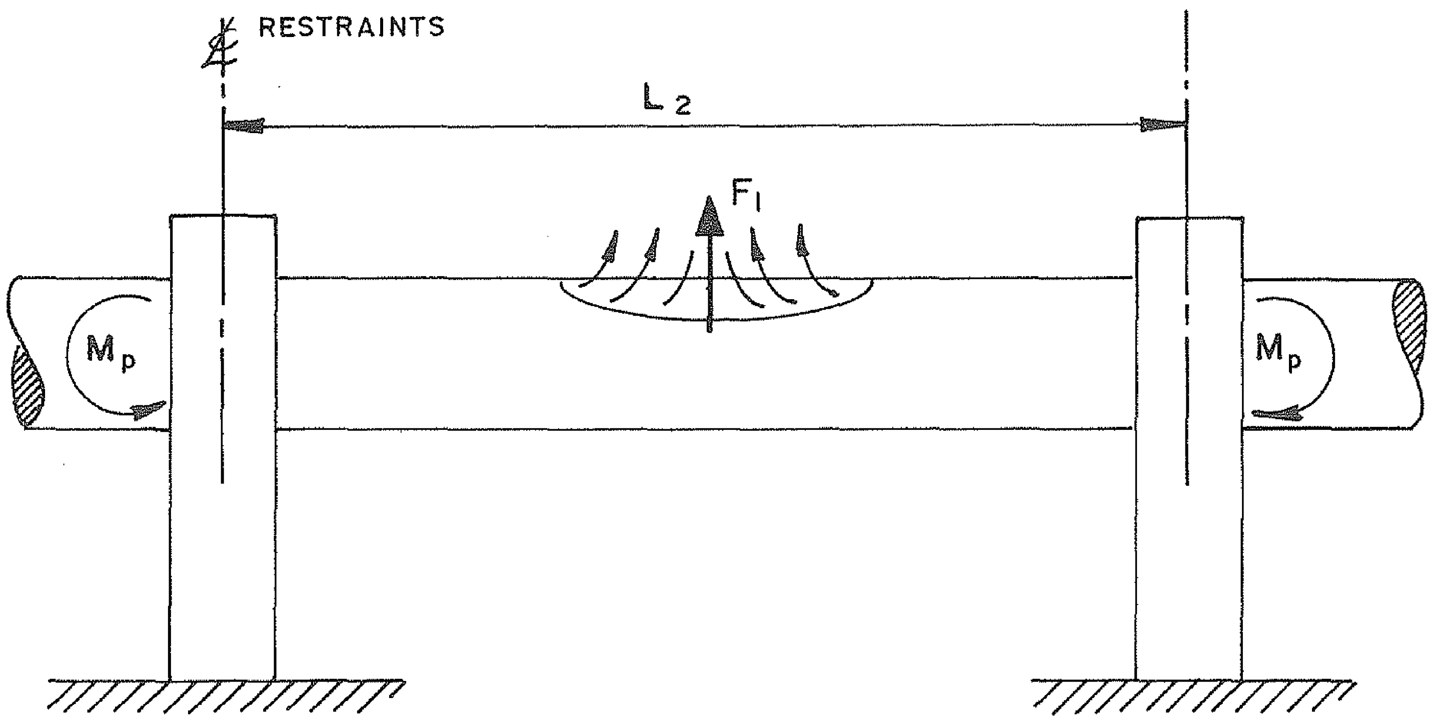
Circumferential Pipe Rupture

Figure 3.6-42



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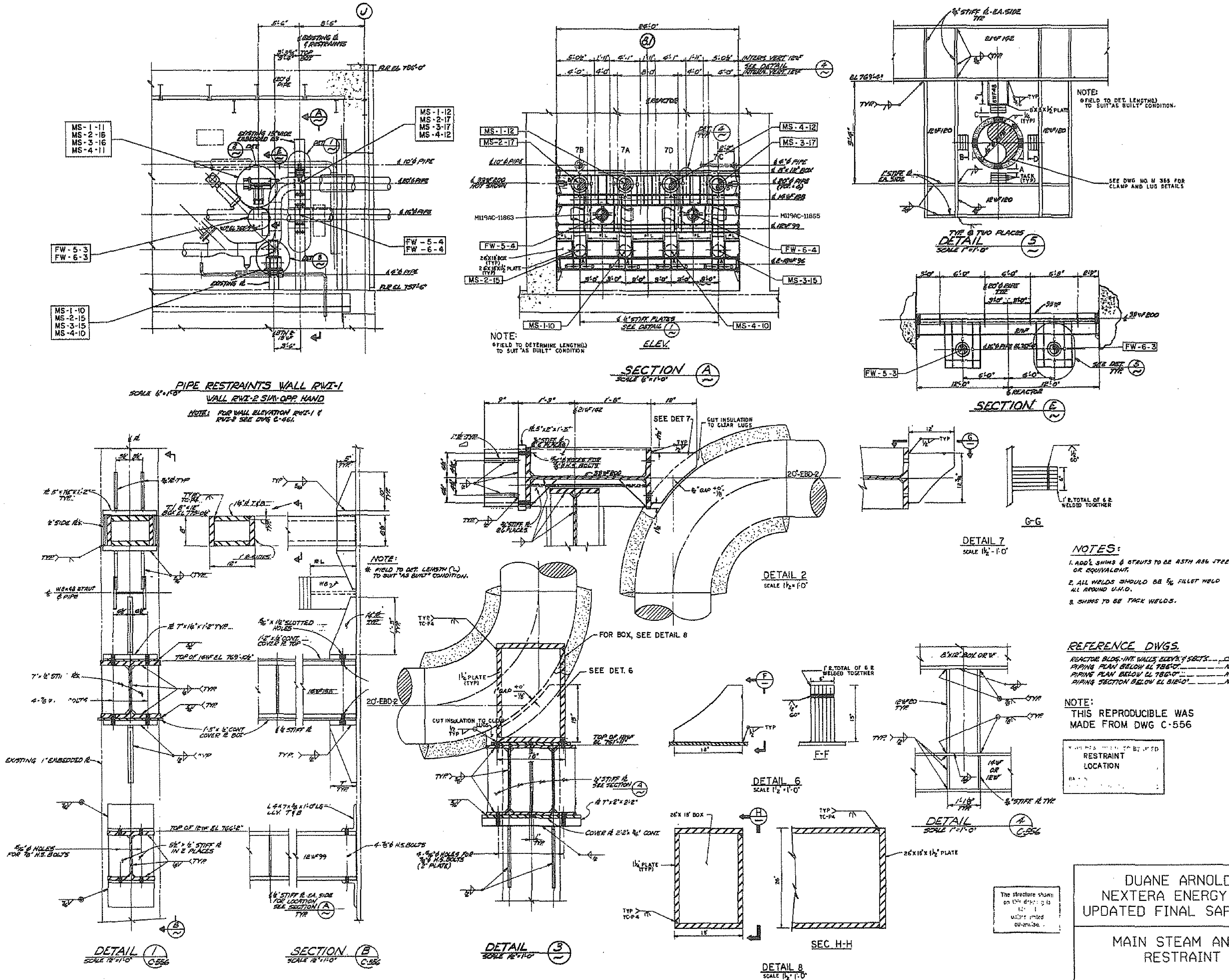
Longitudinal Pipe Rupture

Figure 3.6-43



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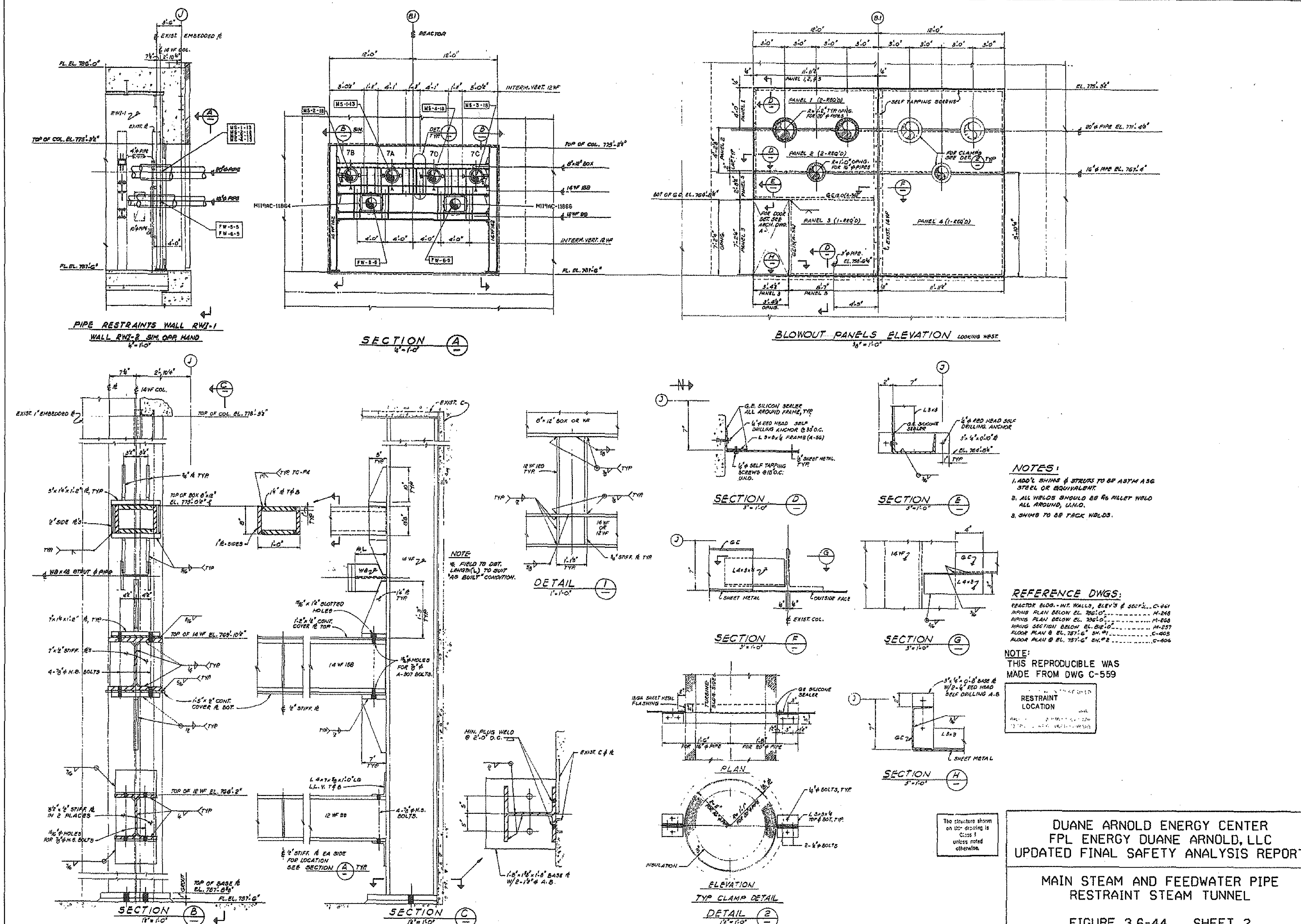
MAIN STEAM AND FEEDWATER PIPE RESTRAINT STEAM TUNNEL

FIGURE 3.6-44 SHEET 1



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FPL ENERGY DUANE ARNOLD, LLC
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MAIN STEAM AND FEEDWATER PIPE
RESTRAINT STEAM TUNNEL

FIGURE 3.6-44 SHEET 2

BECH-M367 REV. 2

REVISION 19 - 09/07

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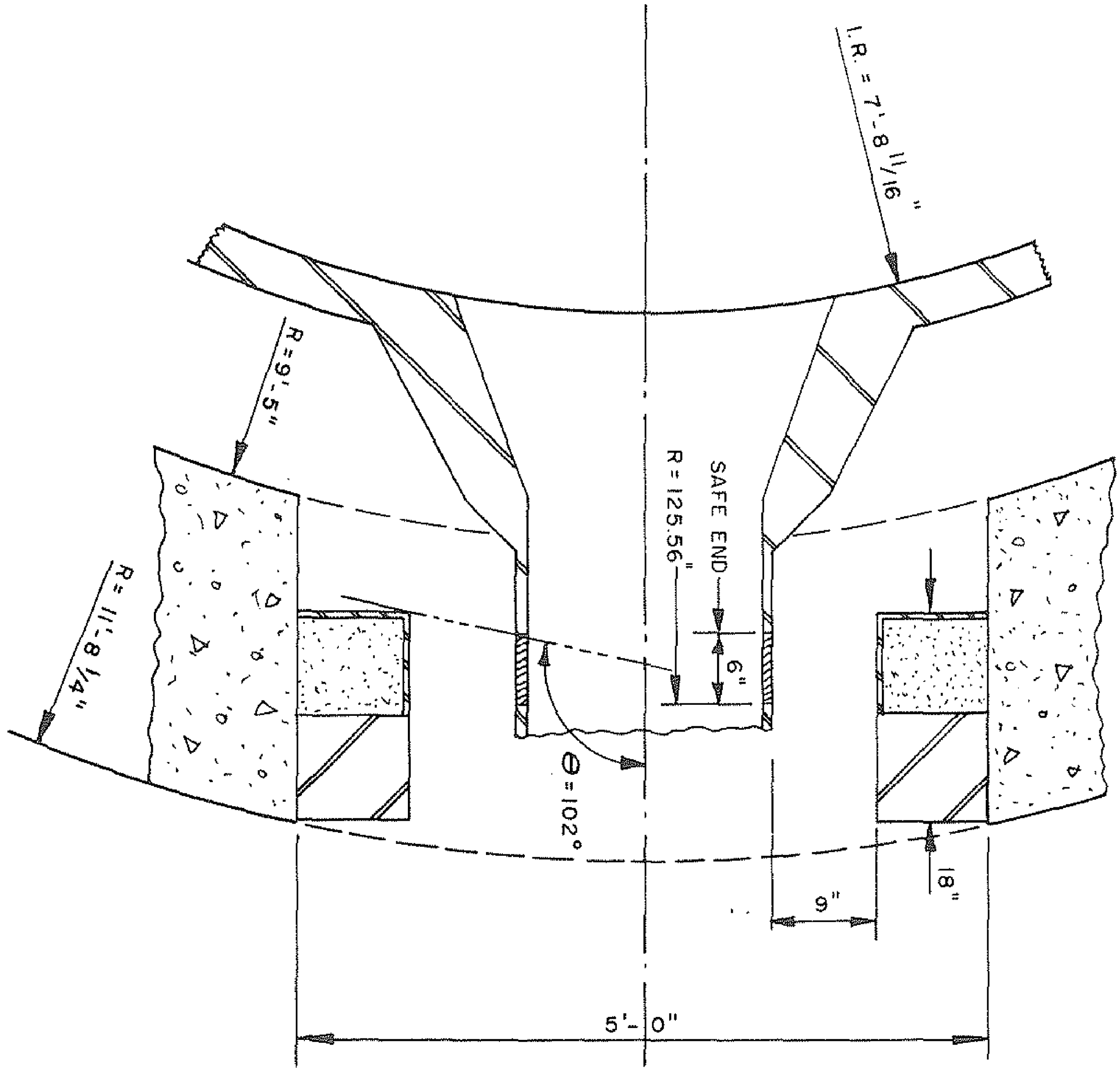

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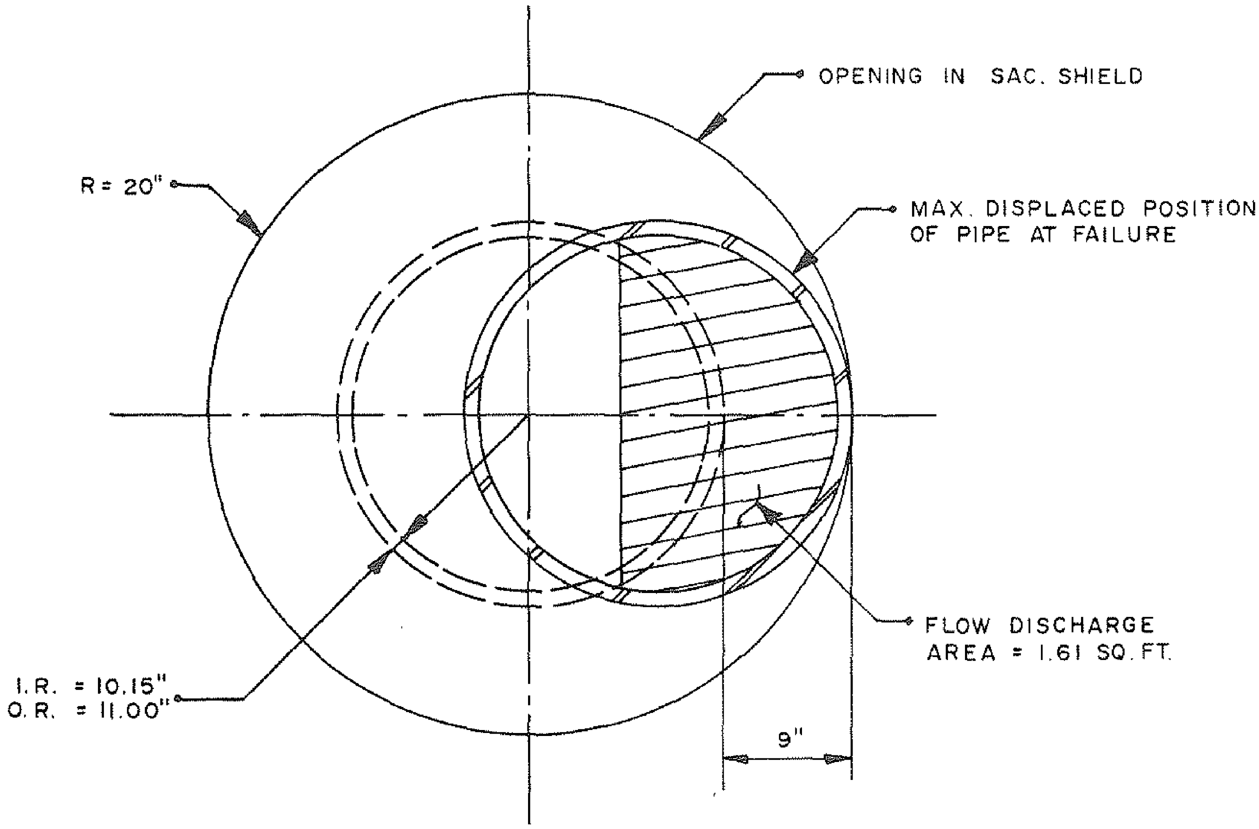
Recirculation Line Penetration of Sacrificial Shield

Figure 3.6-47



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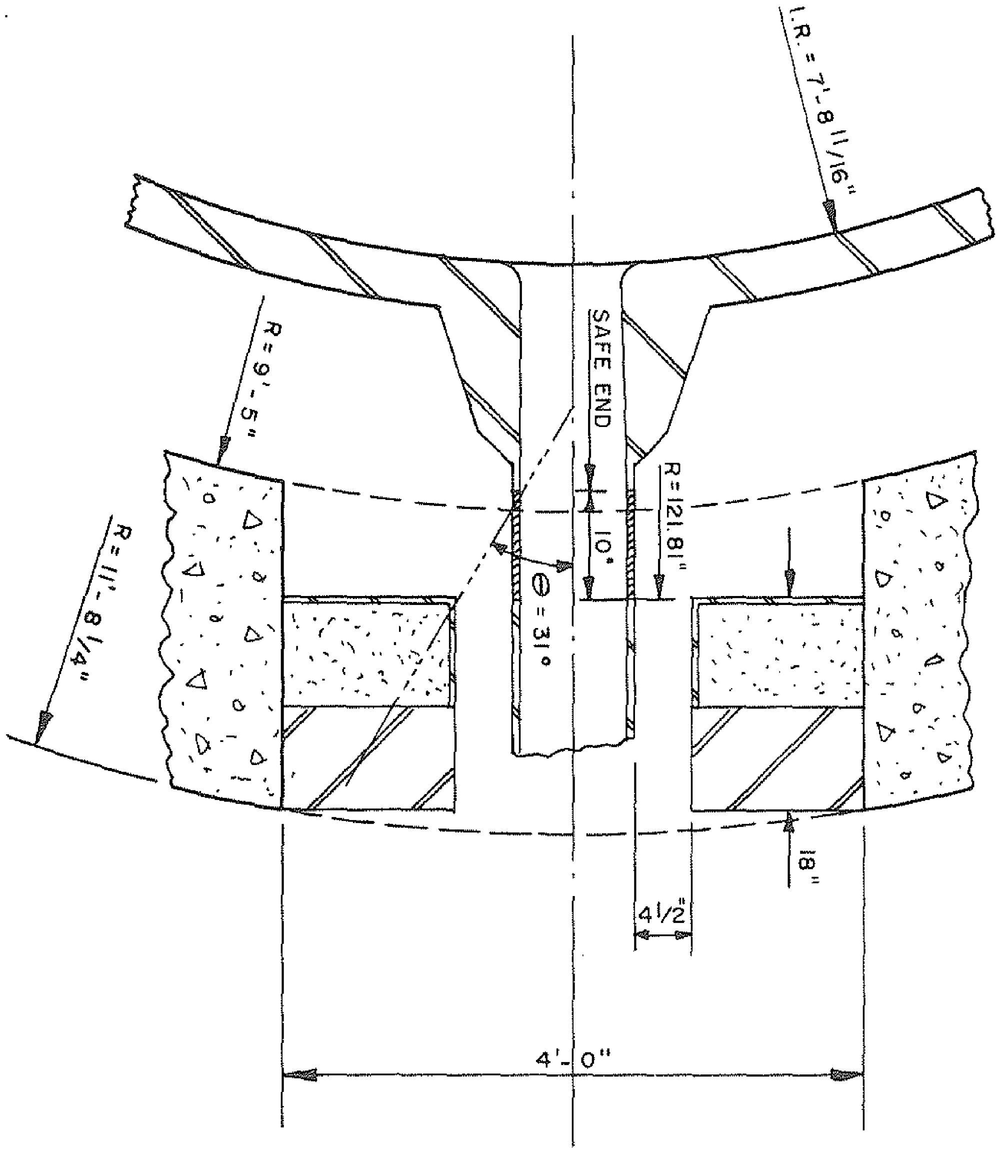
Pipe Break in Reactor
Shield Region

Figure 3.6-48



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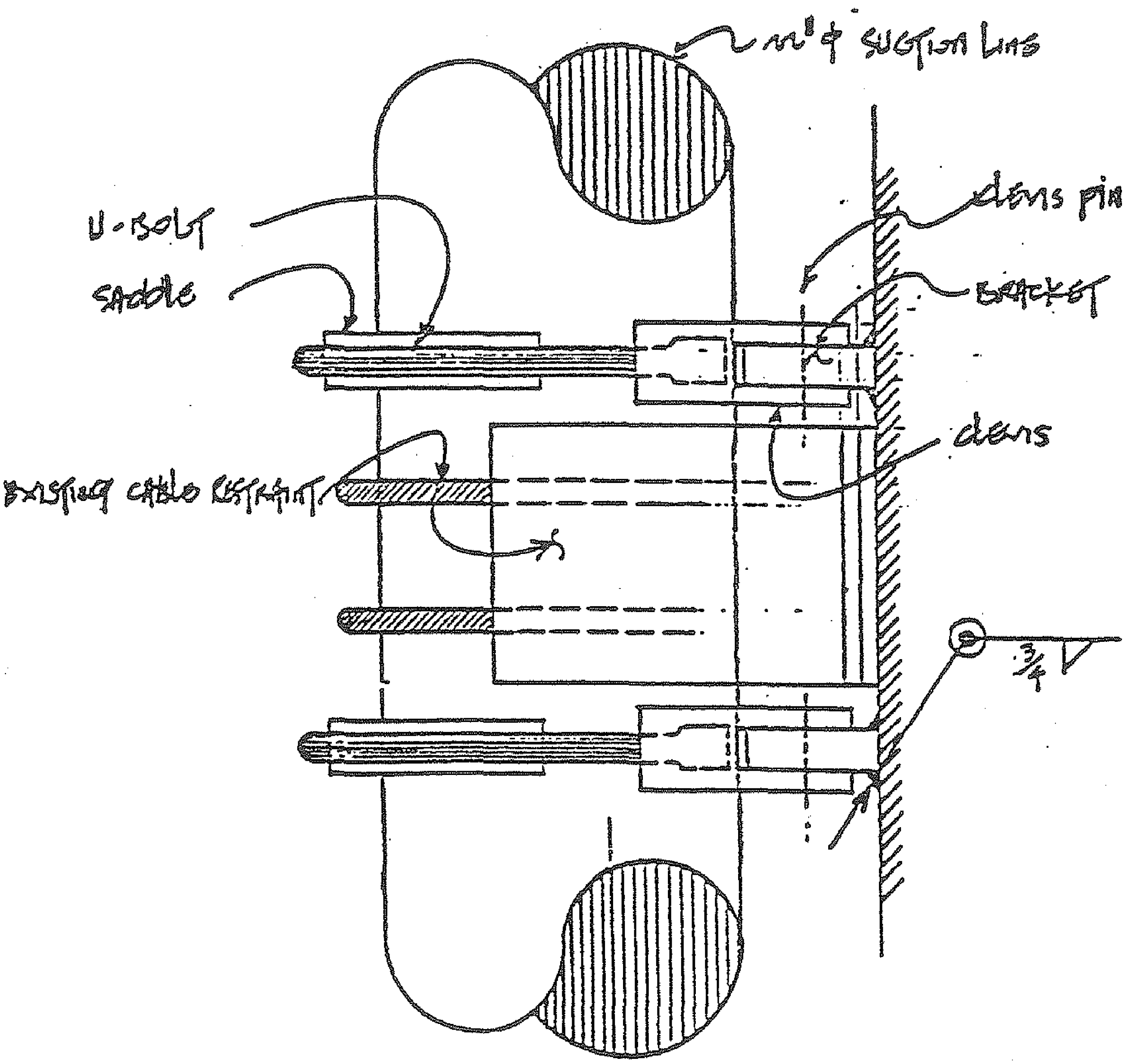
Recirculation Inlet Penetration of Sacrificial Shield

Figure 3.6-49



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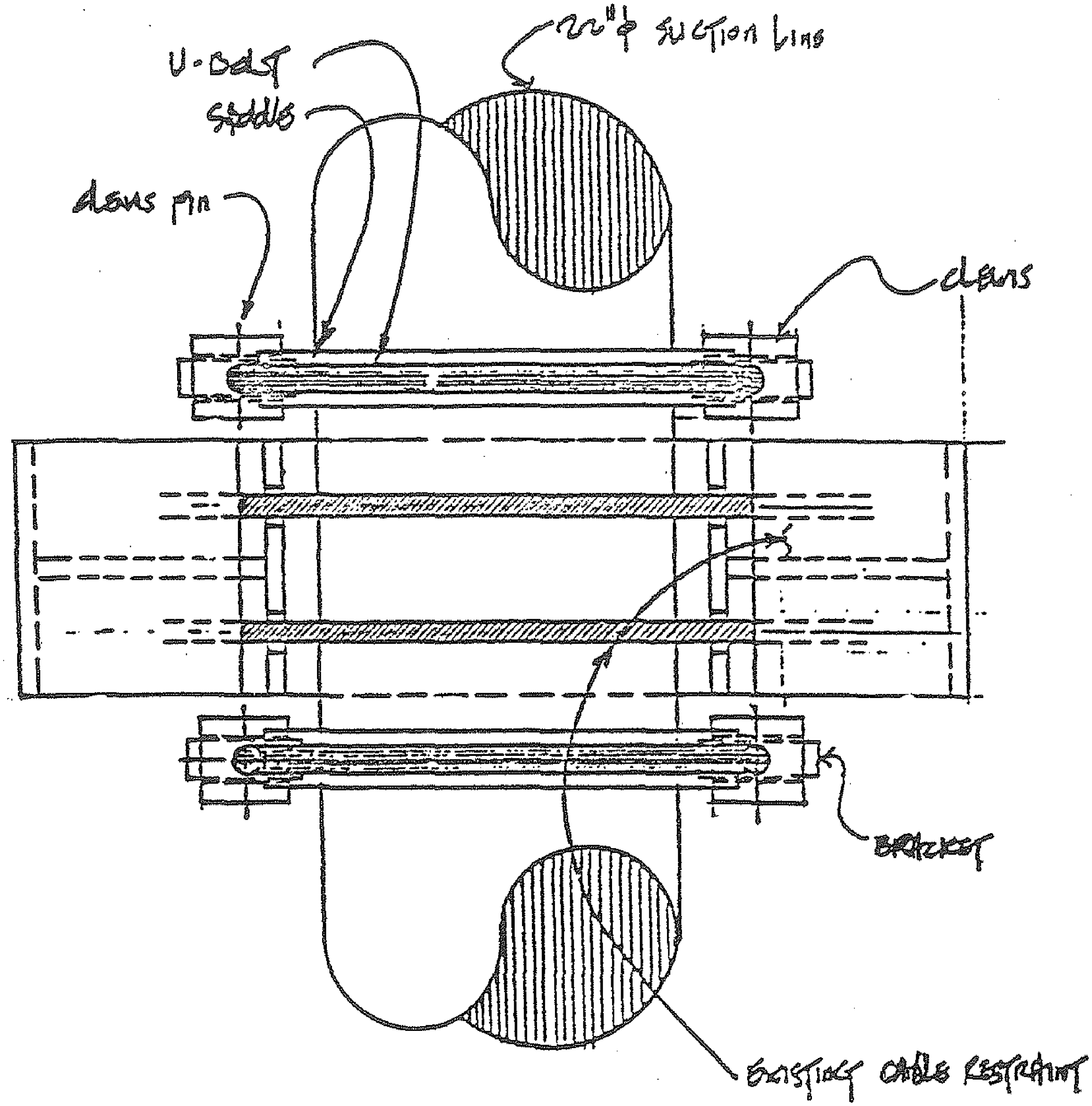
U-Bolt Type Pipe Whip Restraint - Suction Line - Side Elevation

Figure 3.6-50



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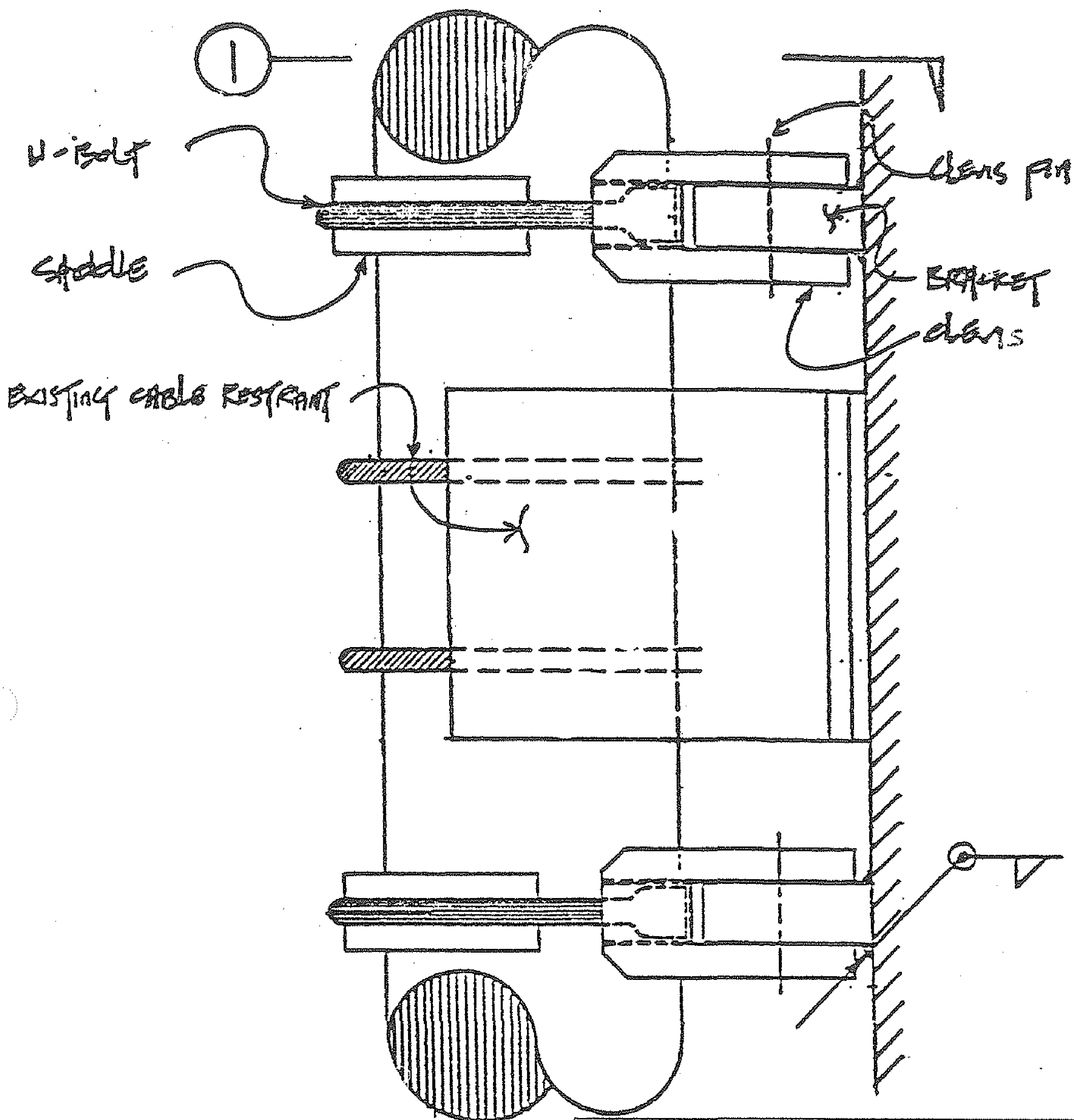
U-Bolt Type Pipe Whip Restraint - Suction Line - Front Elevation

Figure 3.6-51



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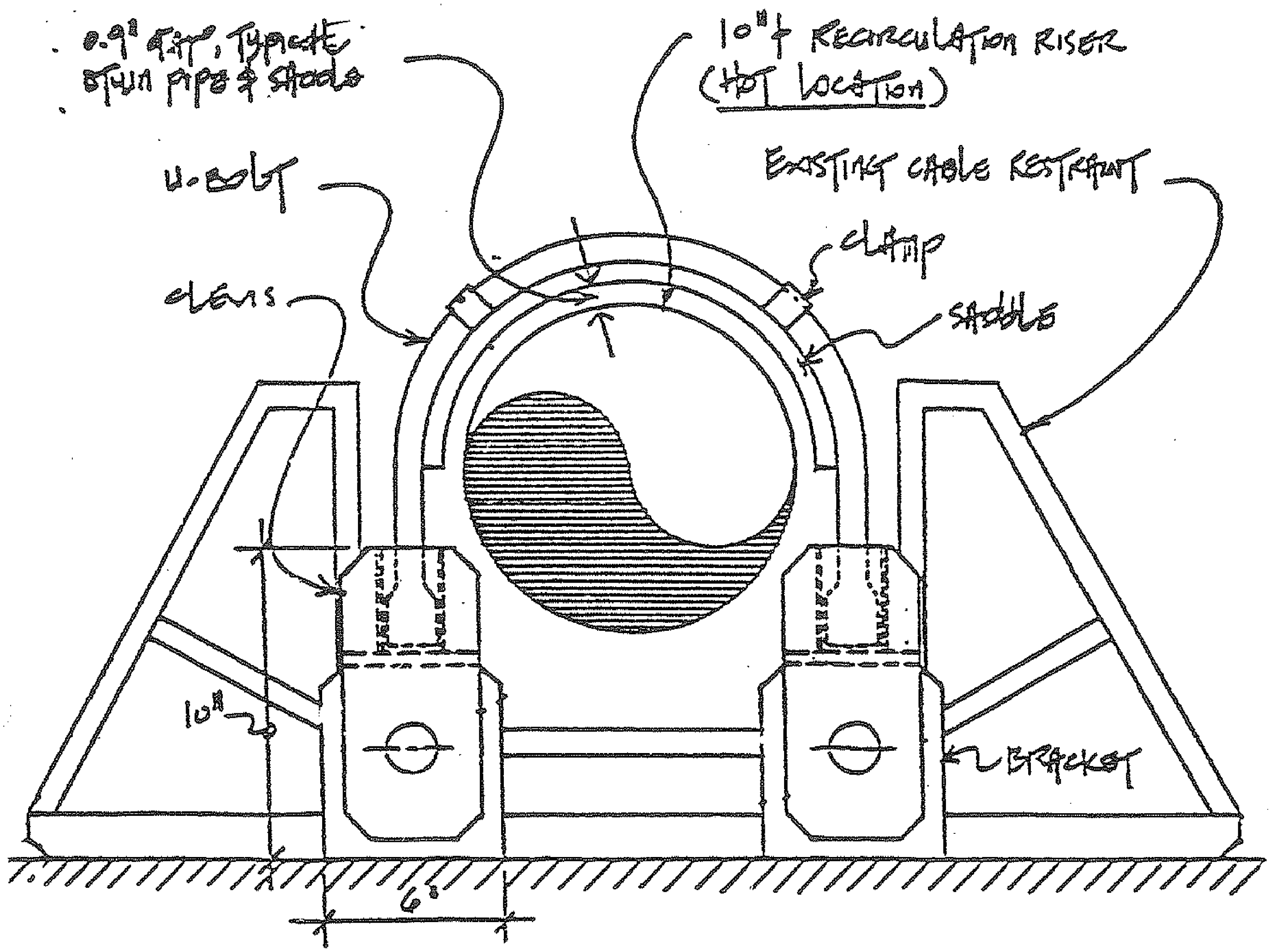
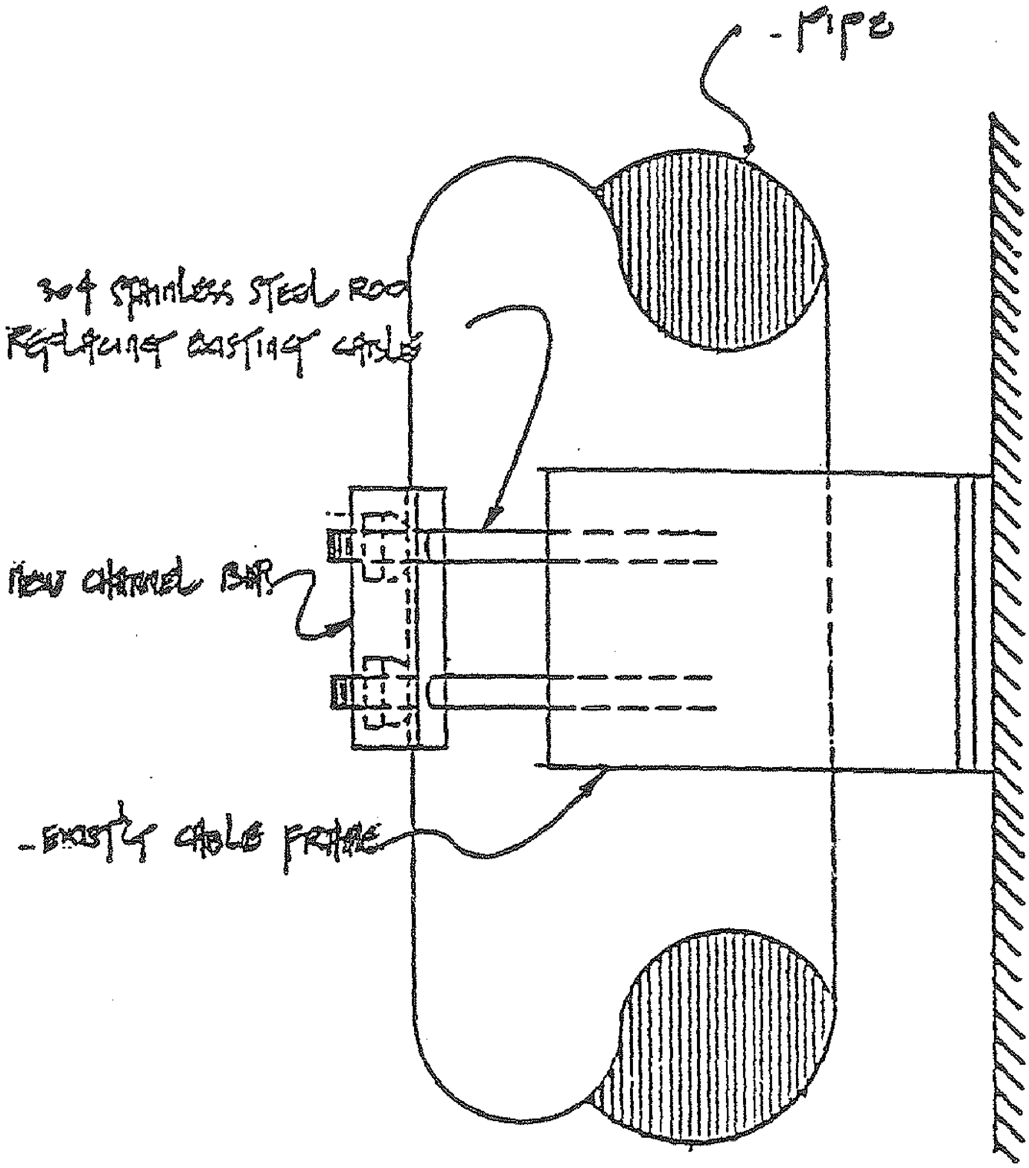
U-Bolt Type Pipe Whip Restraint - Riser Line - Side Elevation

Figure 3.6-52



U-Bolt Type Pipe Whip Restraint - Riser Line - Section

Figure 3.6-53



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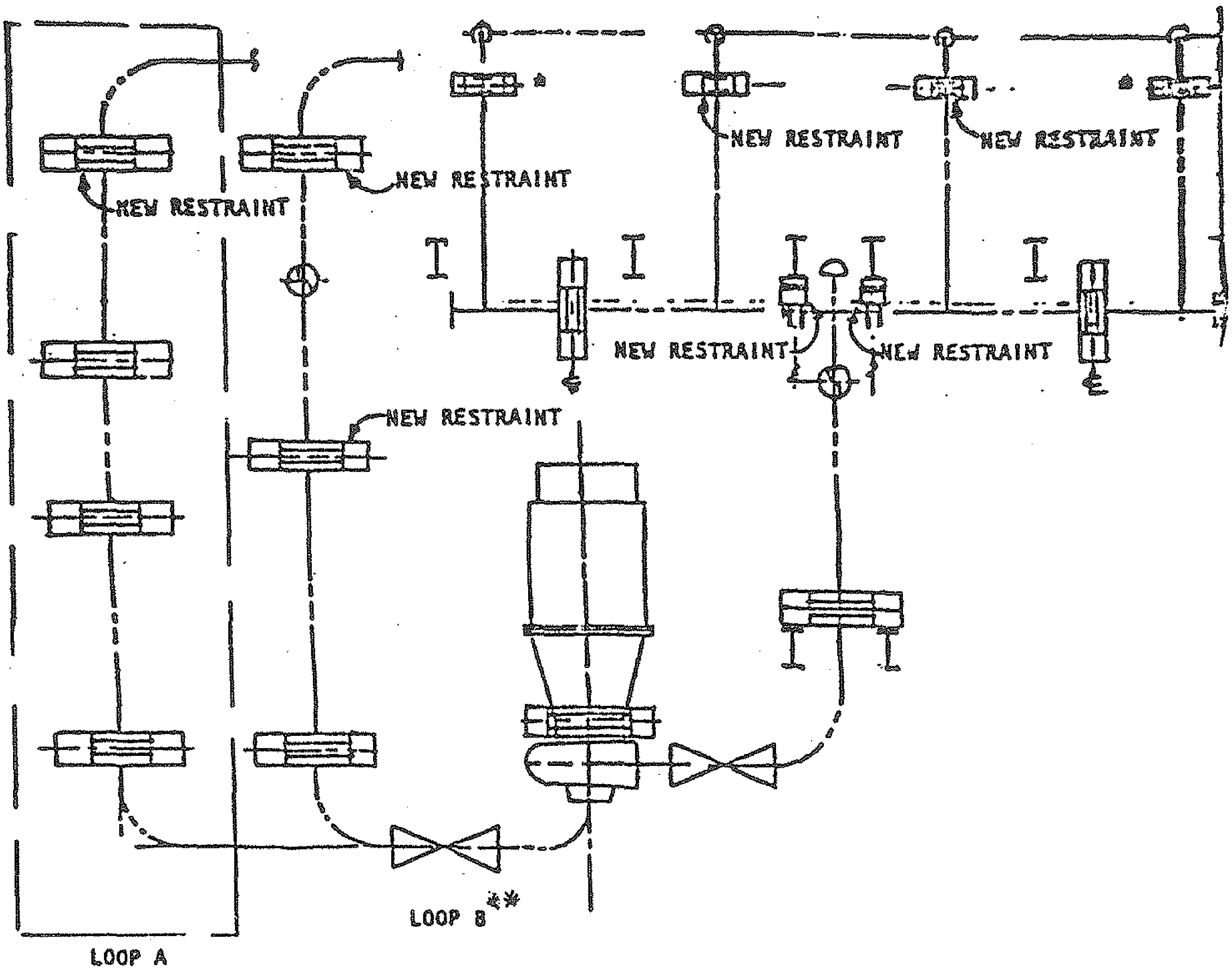
304 Stainless Steel Rod/Channel Beam Type Pipe Whip Restraint - Side Elevation

Figure 3.6-54



*Existing restraints are fabricated from beam sections only (e.g., no cables used). Therefore, these restraints will not be modified.

**Loop B same as Loop A unless noted otherwise.

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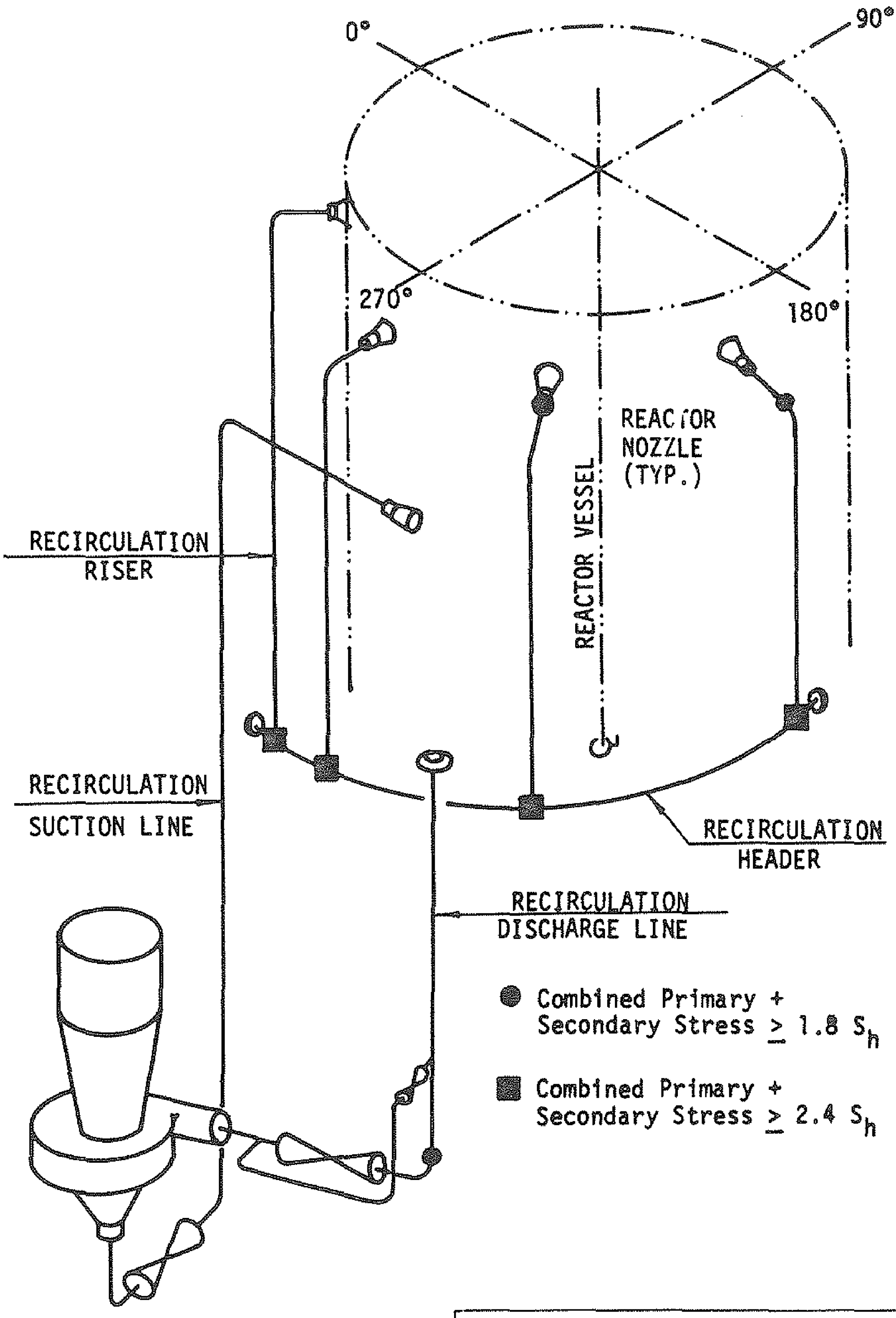
Location for New Restraints to Replace Existing Cables

Figure 3.6-55



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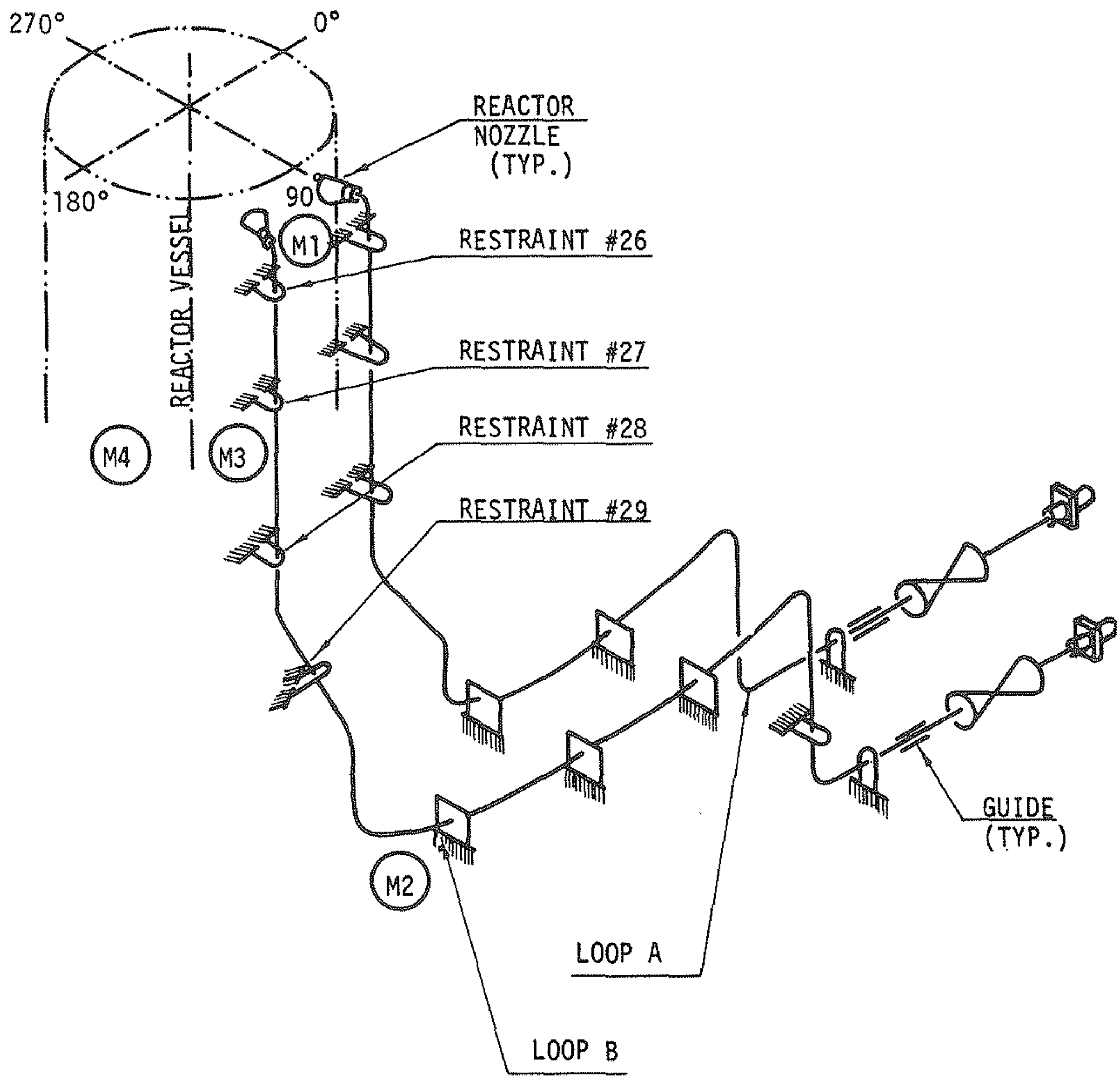
Isometric Sketch, Recirculation Suction and Discharge Line

Figure 3.6-56



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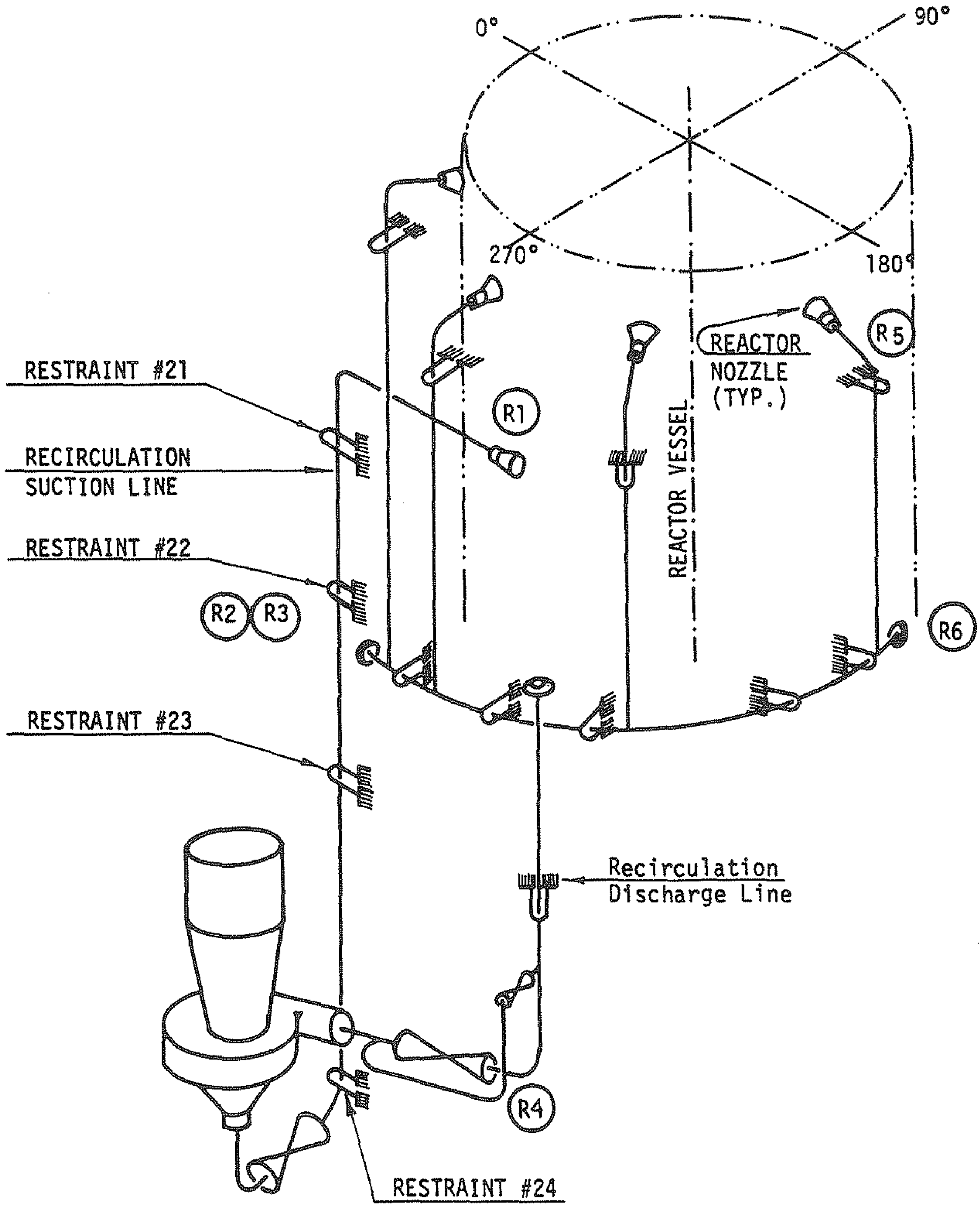
Restraint Design,
Bechtel Corporation,
Main Steam Line Isometric Sketch

Figure 3.6-57



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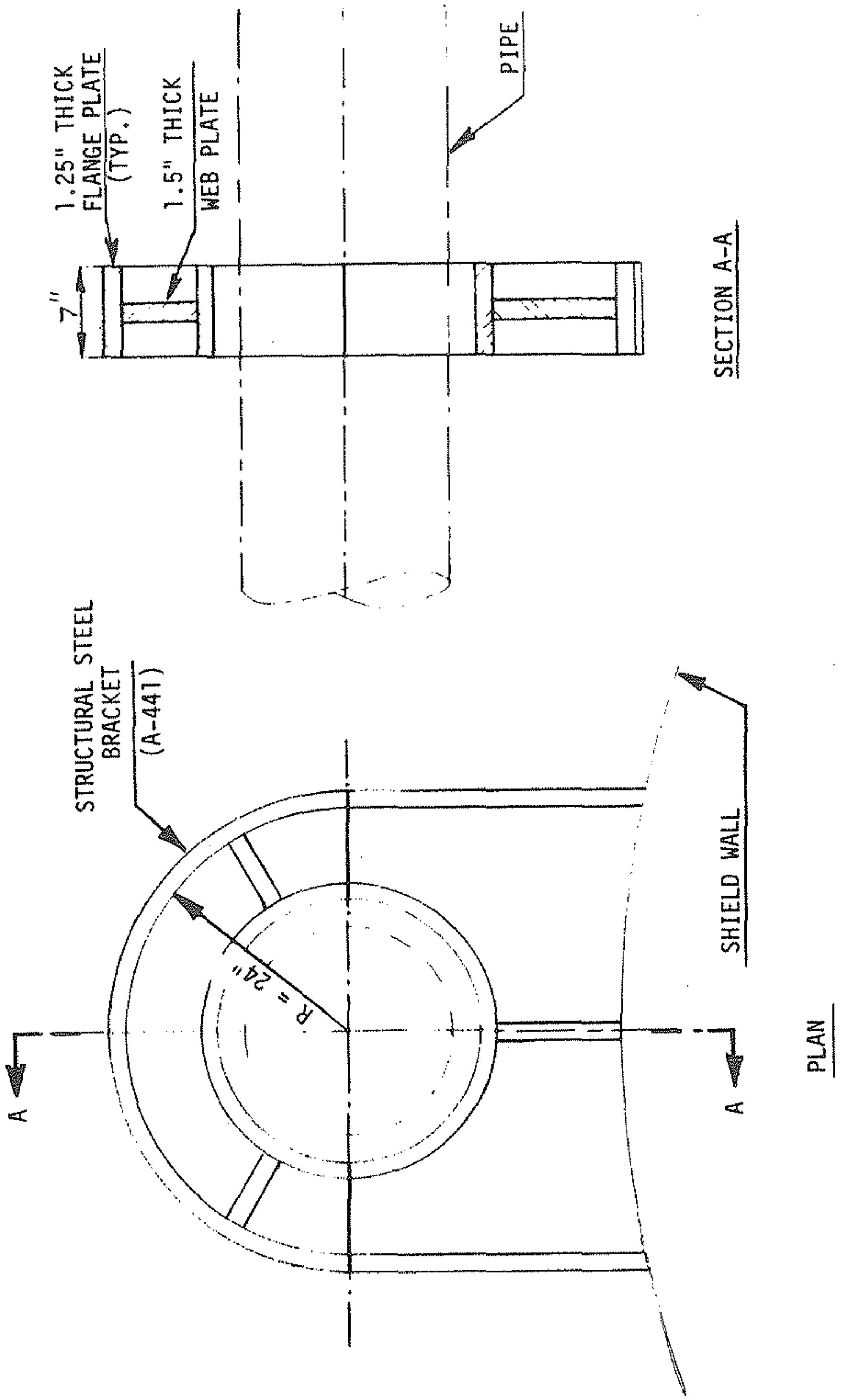
Restraint Design, General Electric Company, Recirculation Line Isometric Sketch Figure 3.6-58



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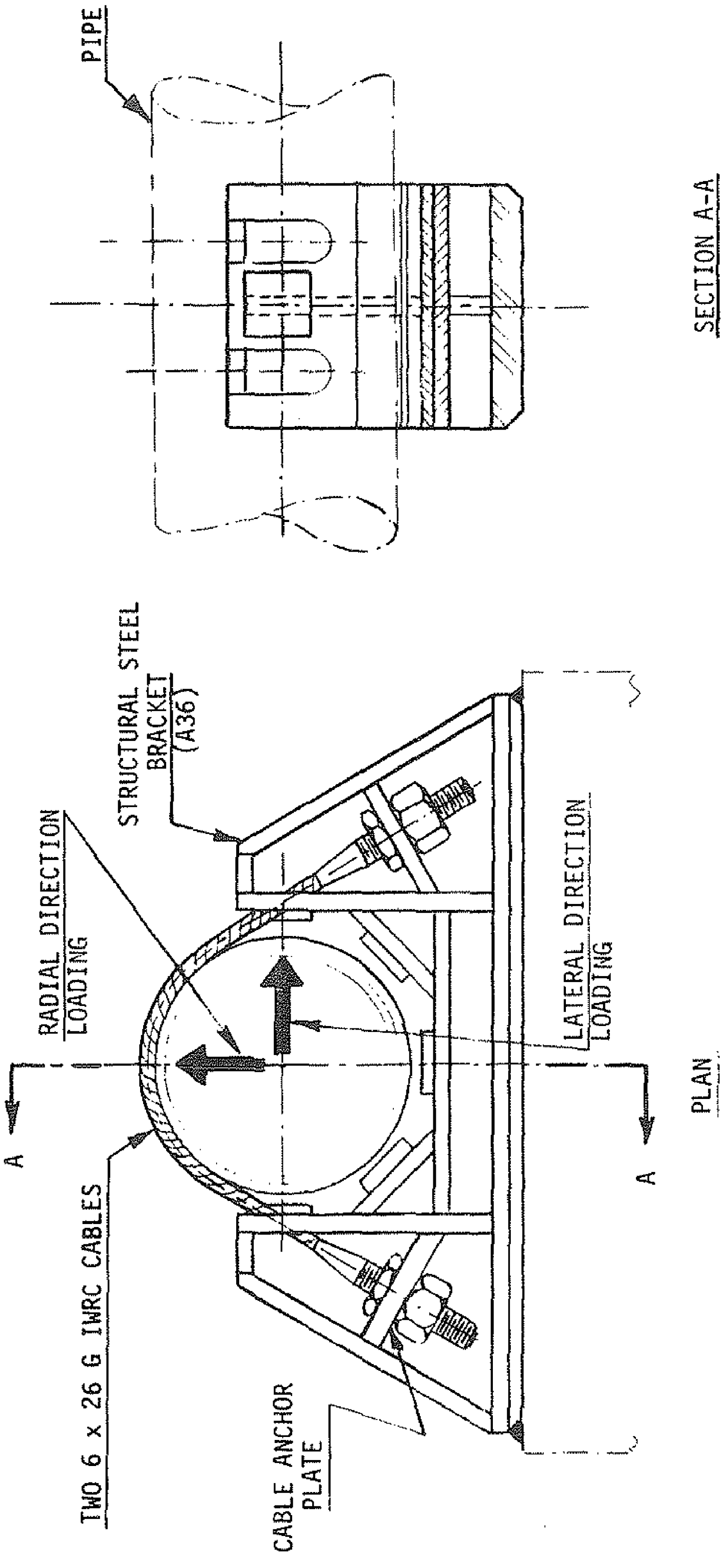
Main Steam Line Restraint Design,
Bechtel Corporation

Figure 3.6-59



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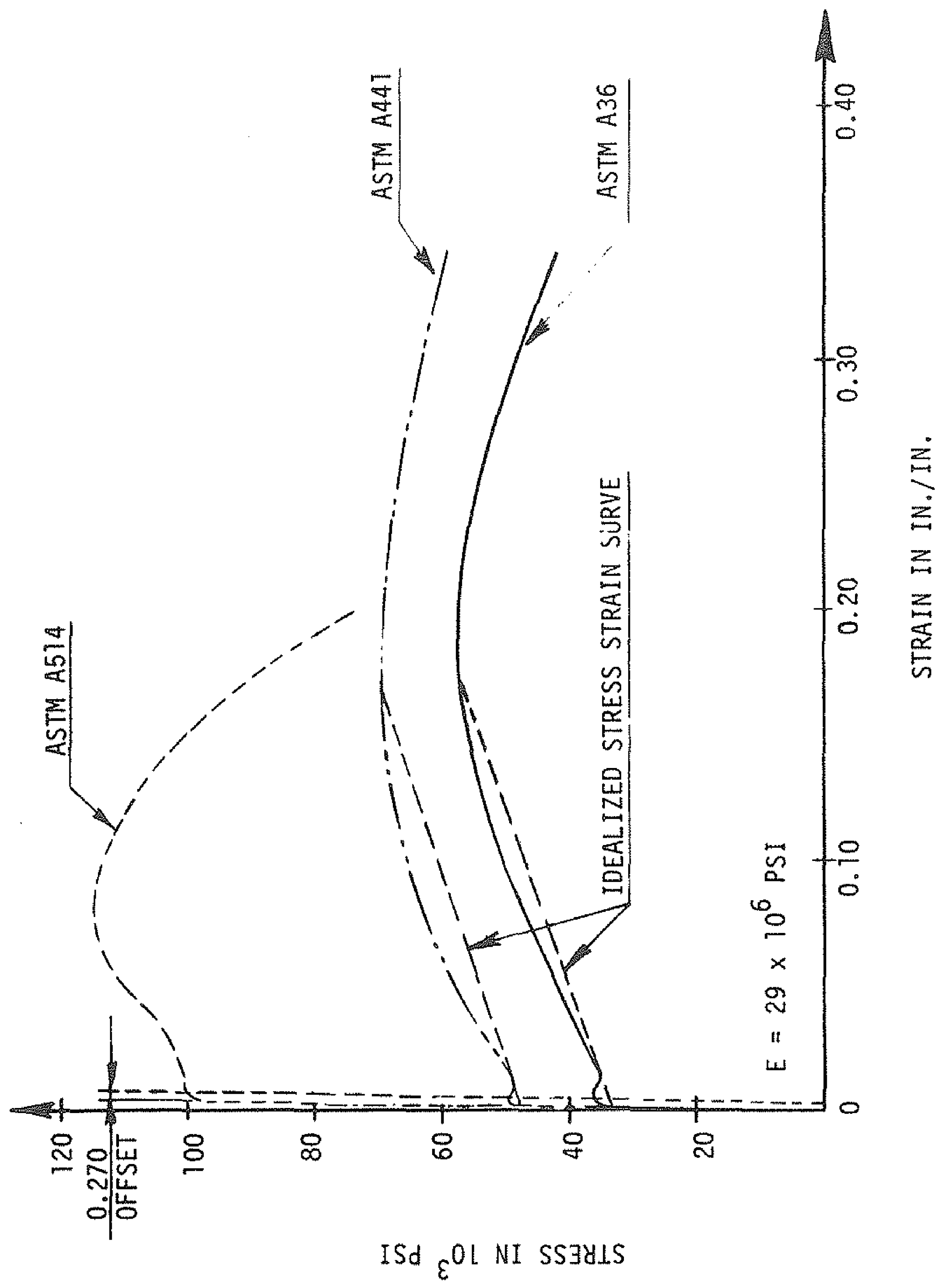
Recirculation Line Restraint Design,
General Electric Company

Figure 3.6-60



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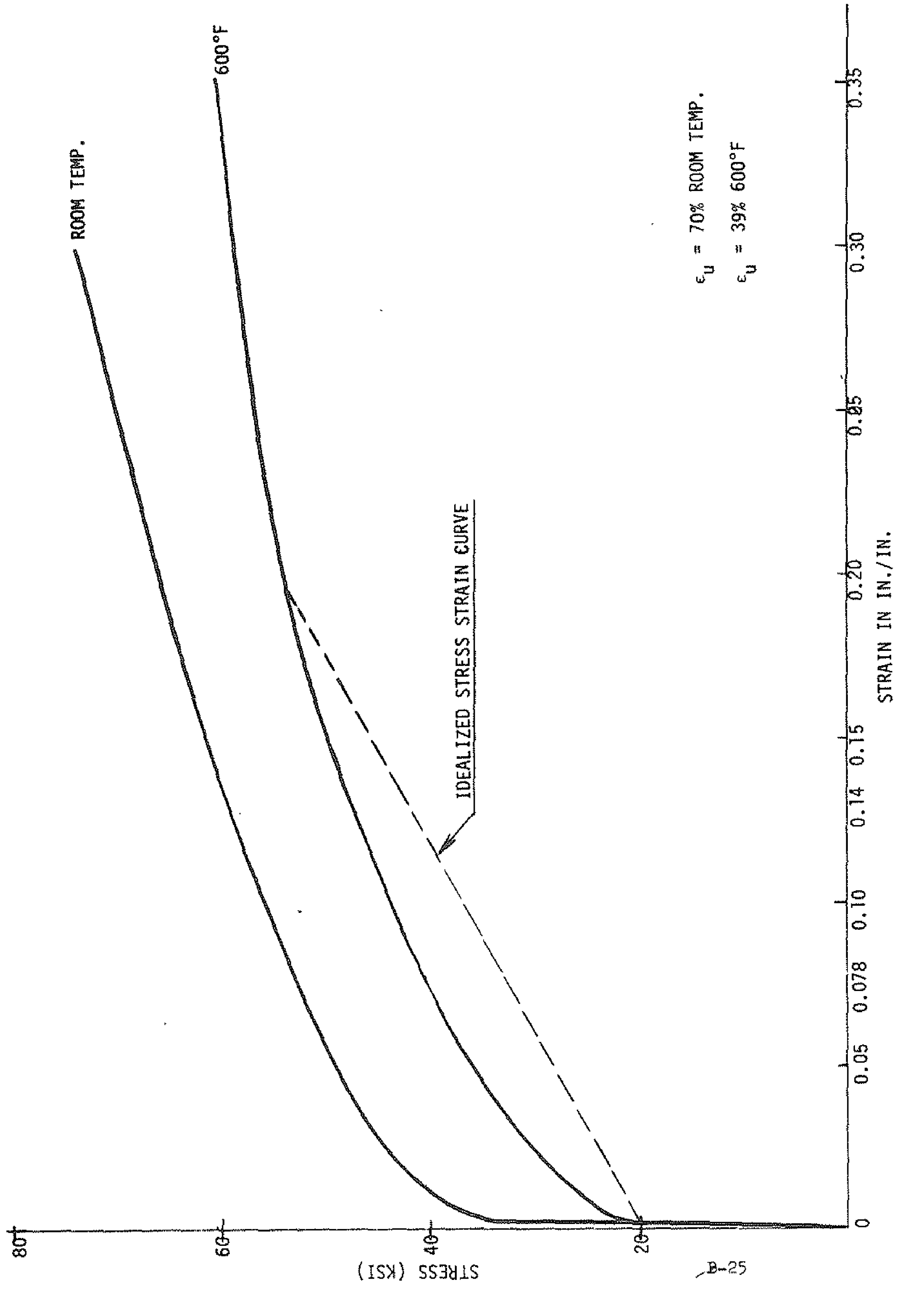
Stress Strain Curves
for Structural Steel

Figure 3.6-61



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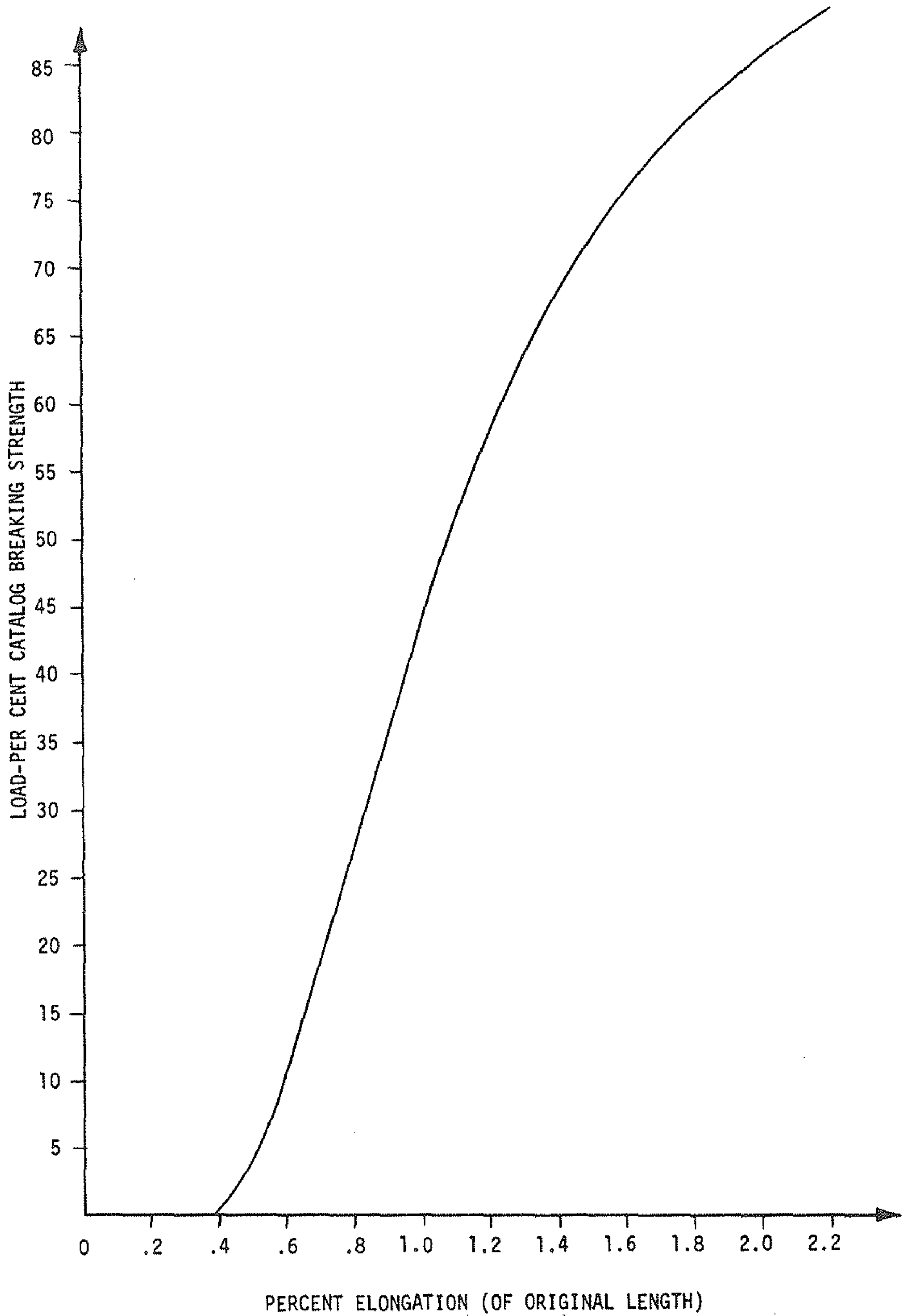
Stress Strain Curve for
304 Stainless Steel

Figure 3.6-62



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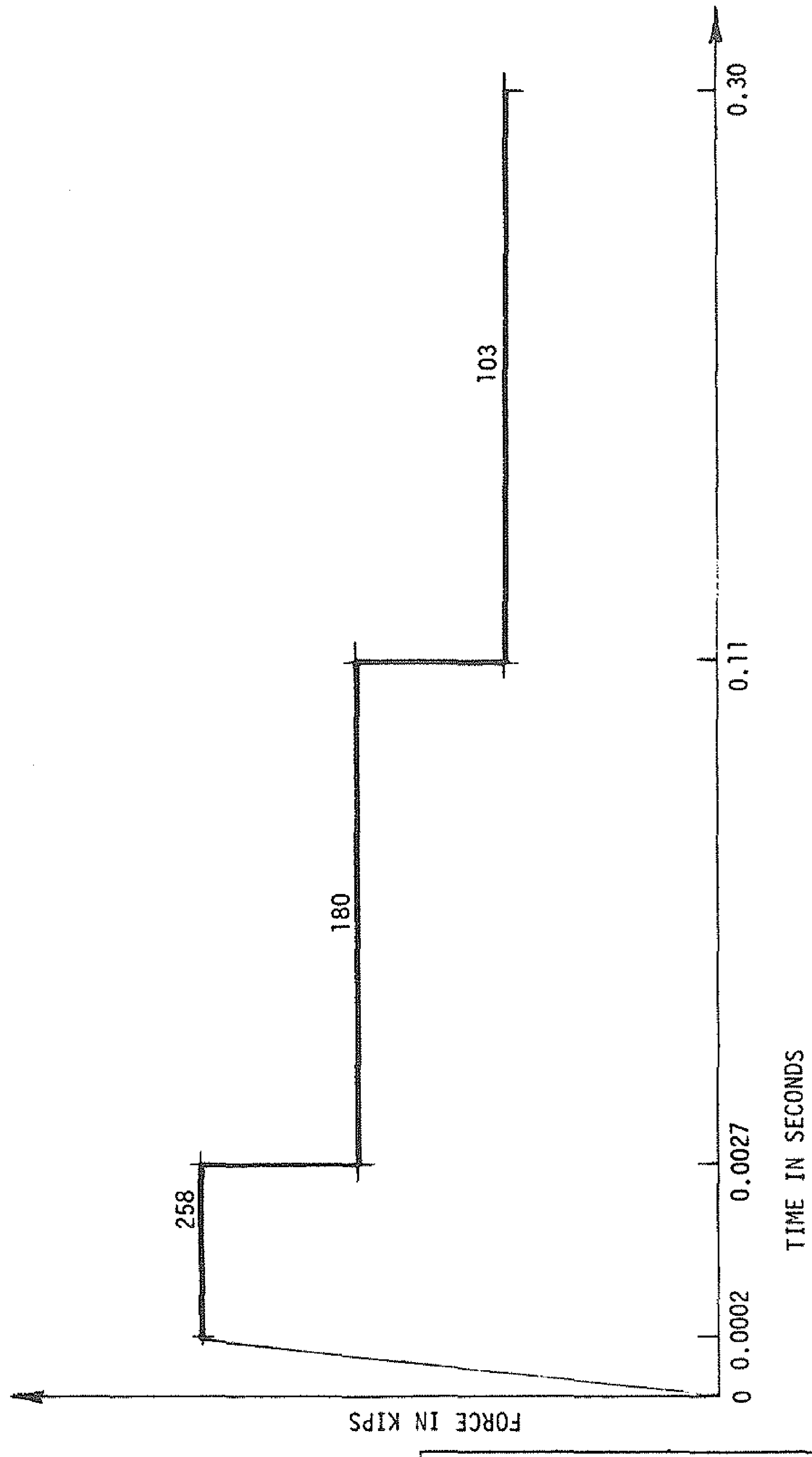
Load-Elongation Curve for
6 x 26 G IWRC Cable

Figure 3.6-63



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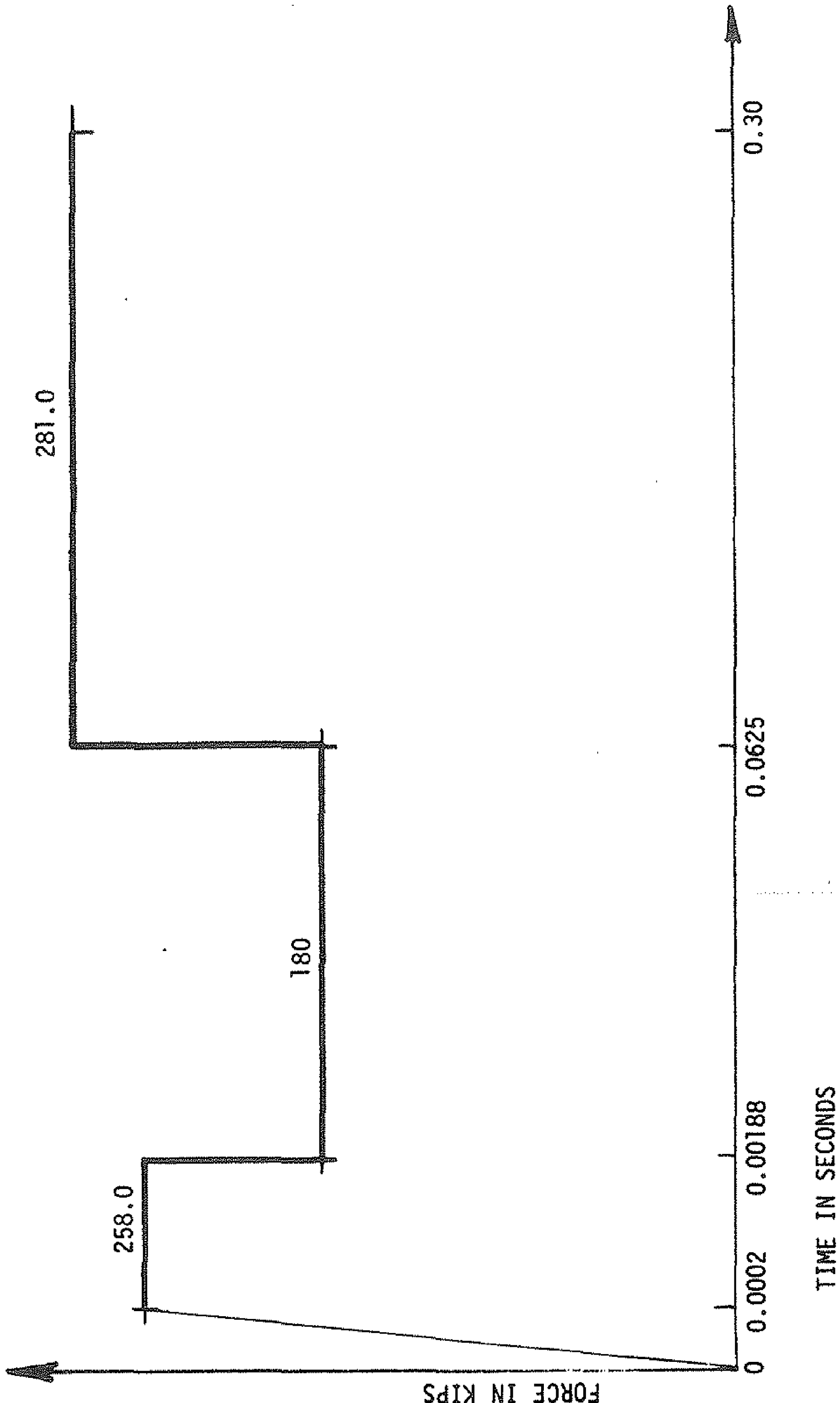
Main Steam Line Circumferential
Break M1 at Reactor Vessel
Nozzle, Rupture Force Time Curve

Figure 3.6-64



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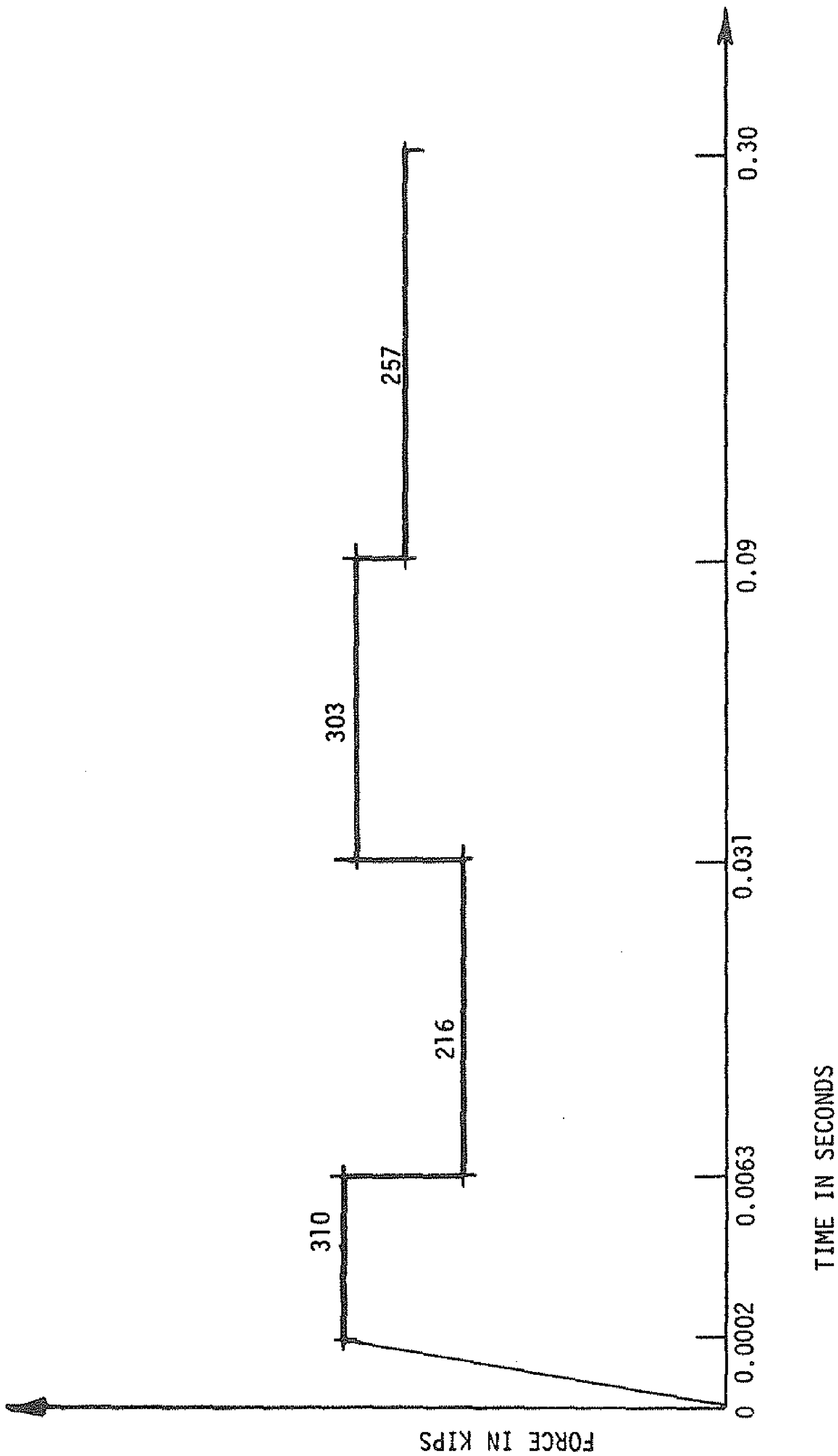
Main Steam Line Circumferential Break M2 at Horizontal Header, Rupture Force Time Curve

Figure 3.6-65



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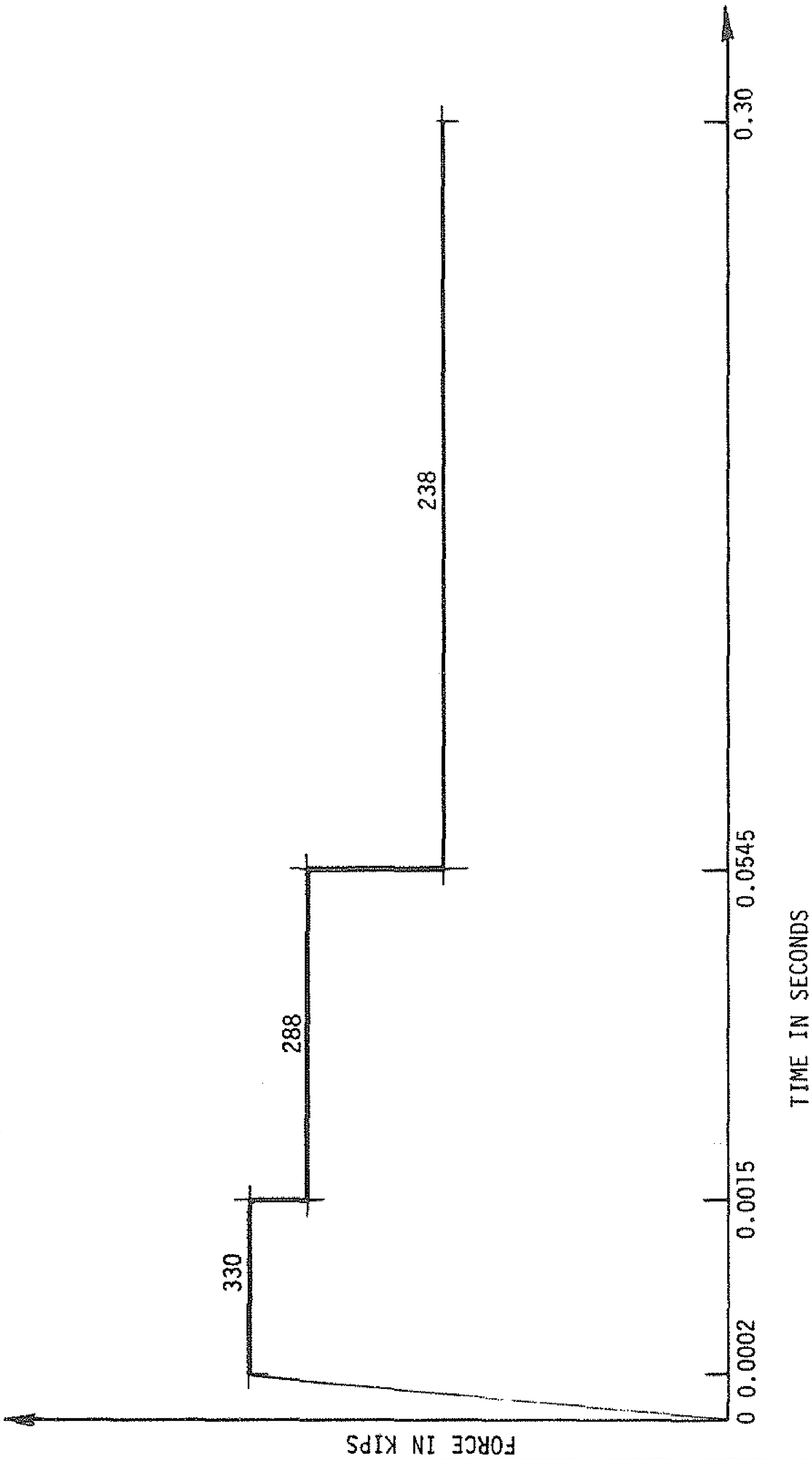
Main Steam Line Longitudinal
Break M3 and M4, Rupture
Force Time Curve

Figure 3.6-66



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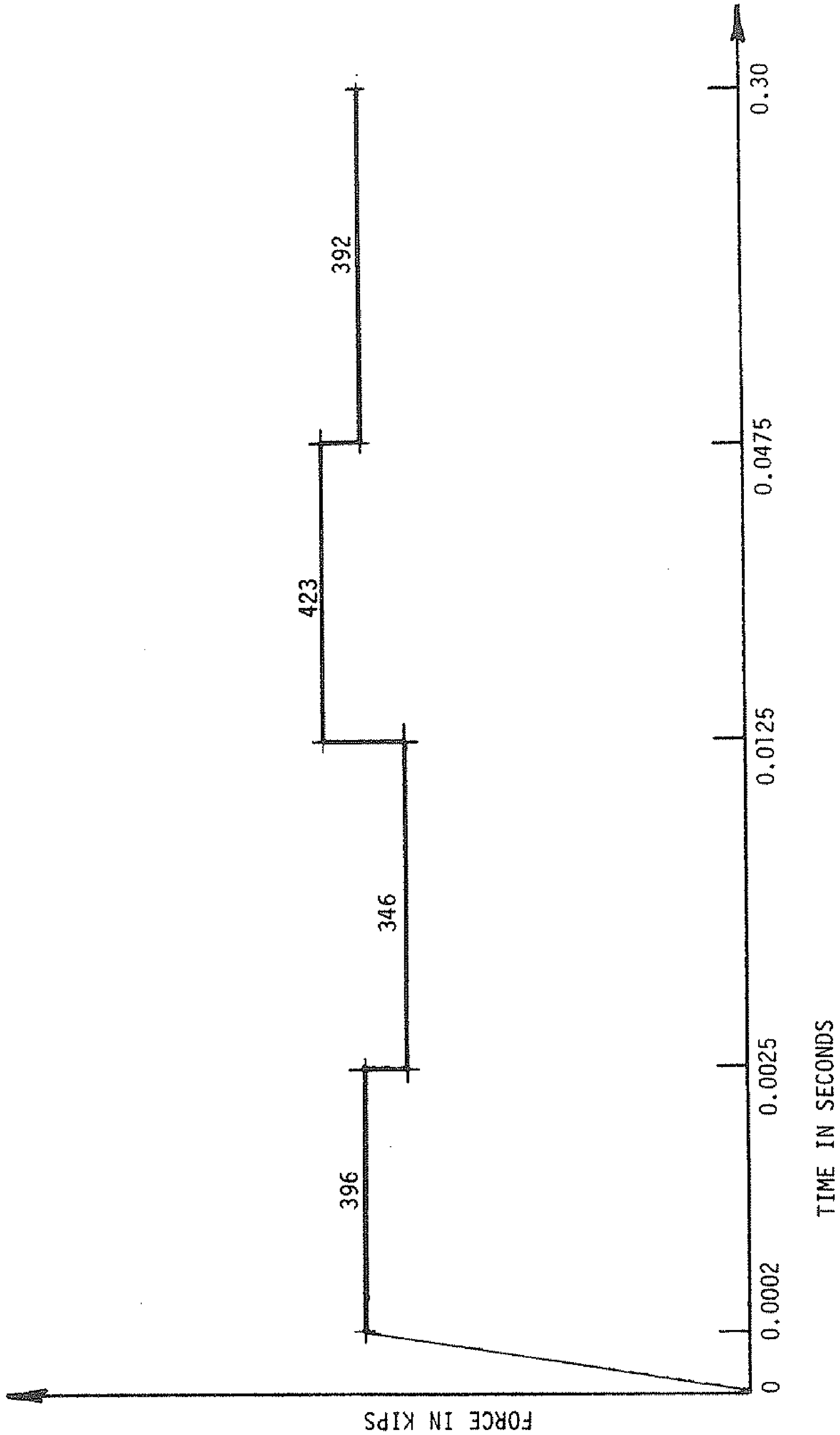
Recirculation Suction Line Circumferential Break R1 at Reactor Vessel Nozzle, Rupture Force Time Curve

Figure 3.6-67



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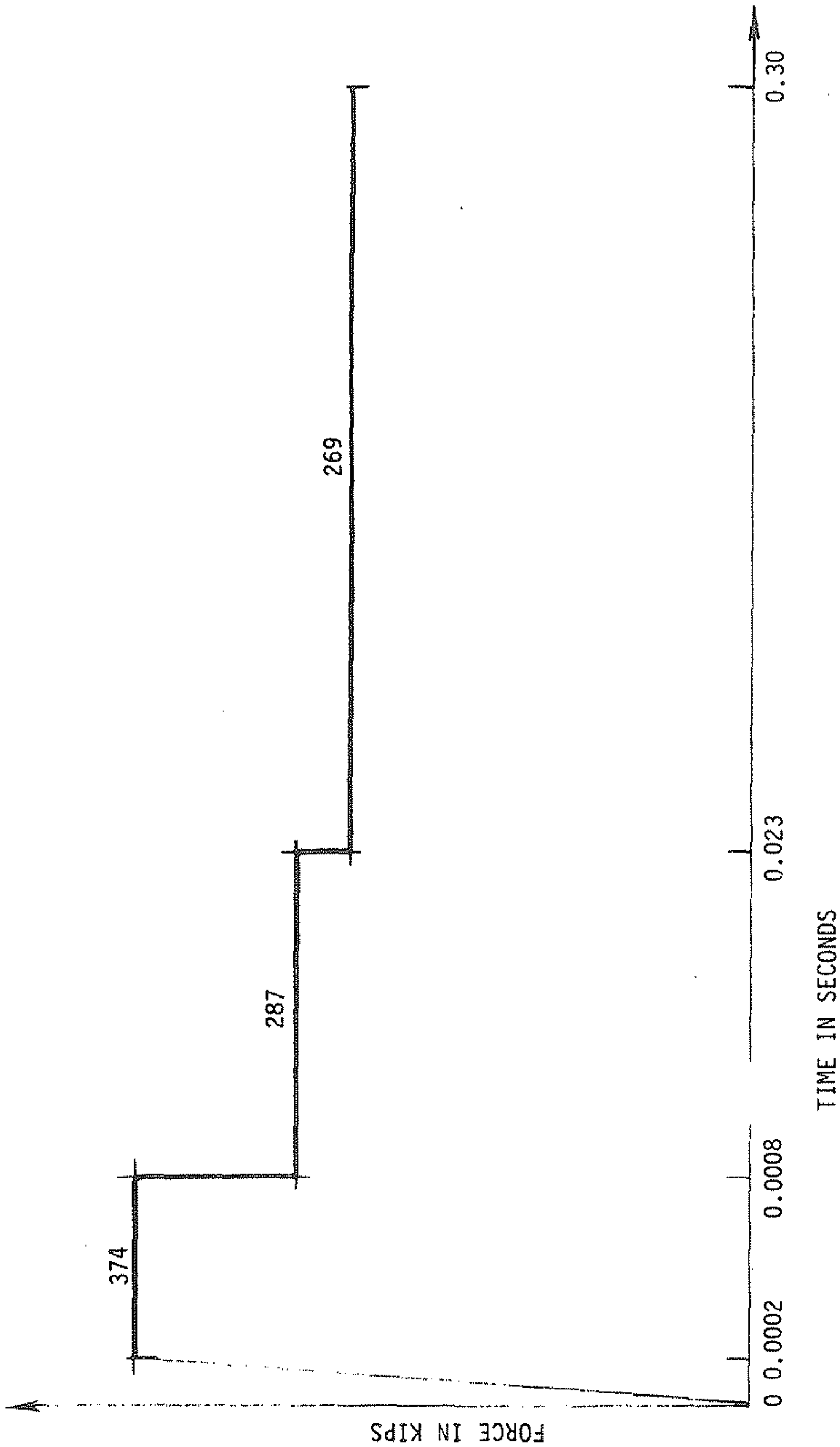
Recirculation Suction Line
Longitudinal Break R2 and R3,
Rupture Force Time Curve

Figure 3.6-68



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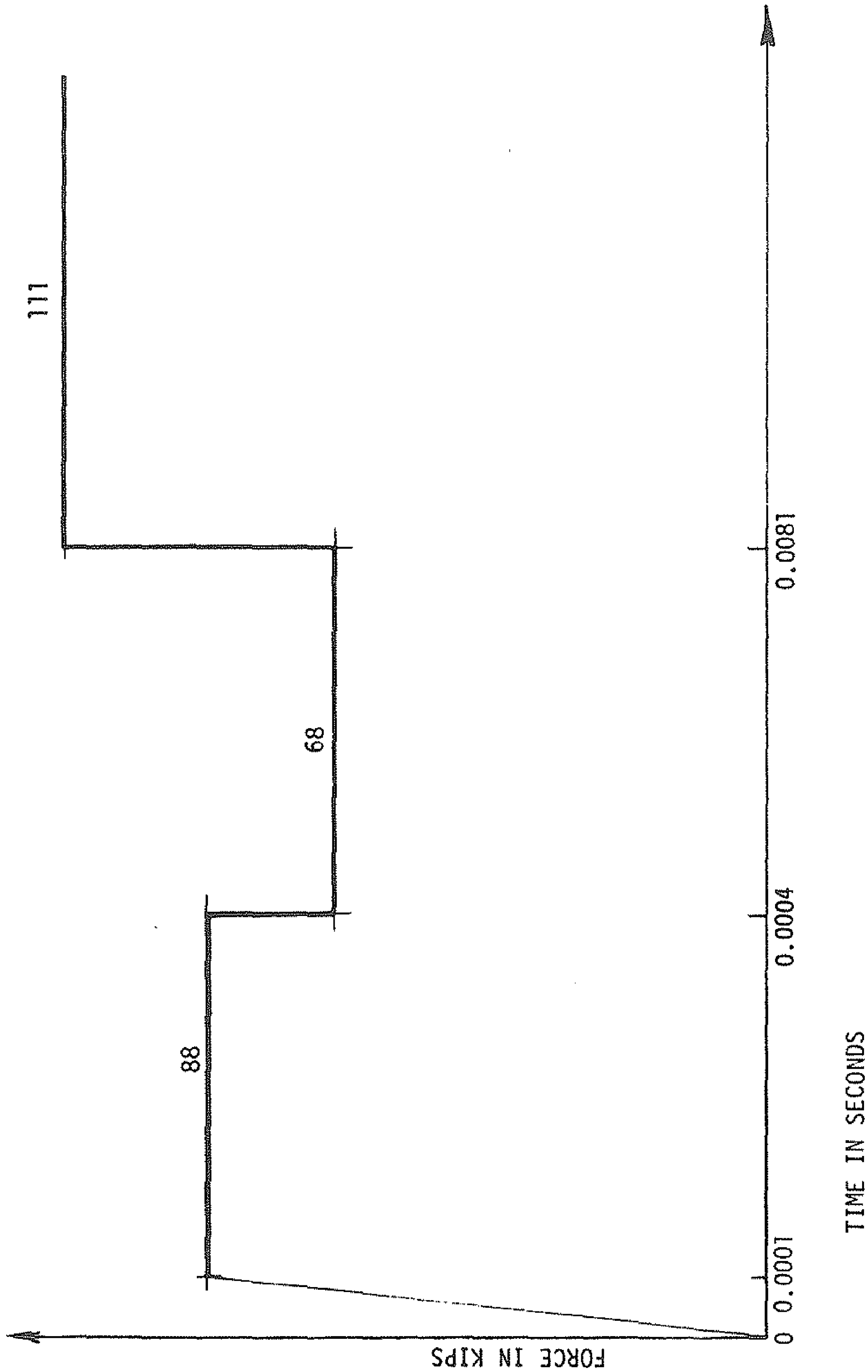
Recirculation Discharge Line
Circumferential Break R4 at Discharge
Elbow Rupture,Force Time Curve

Figure 3.6-69



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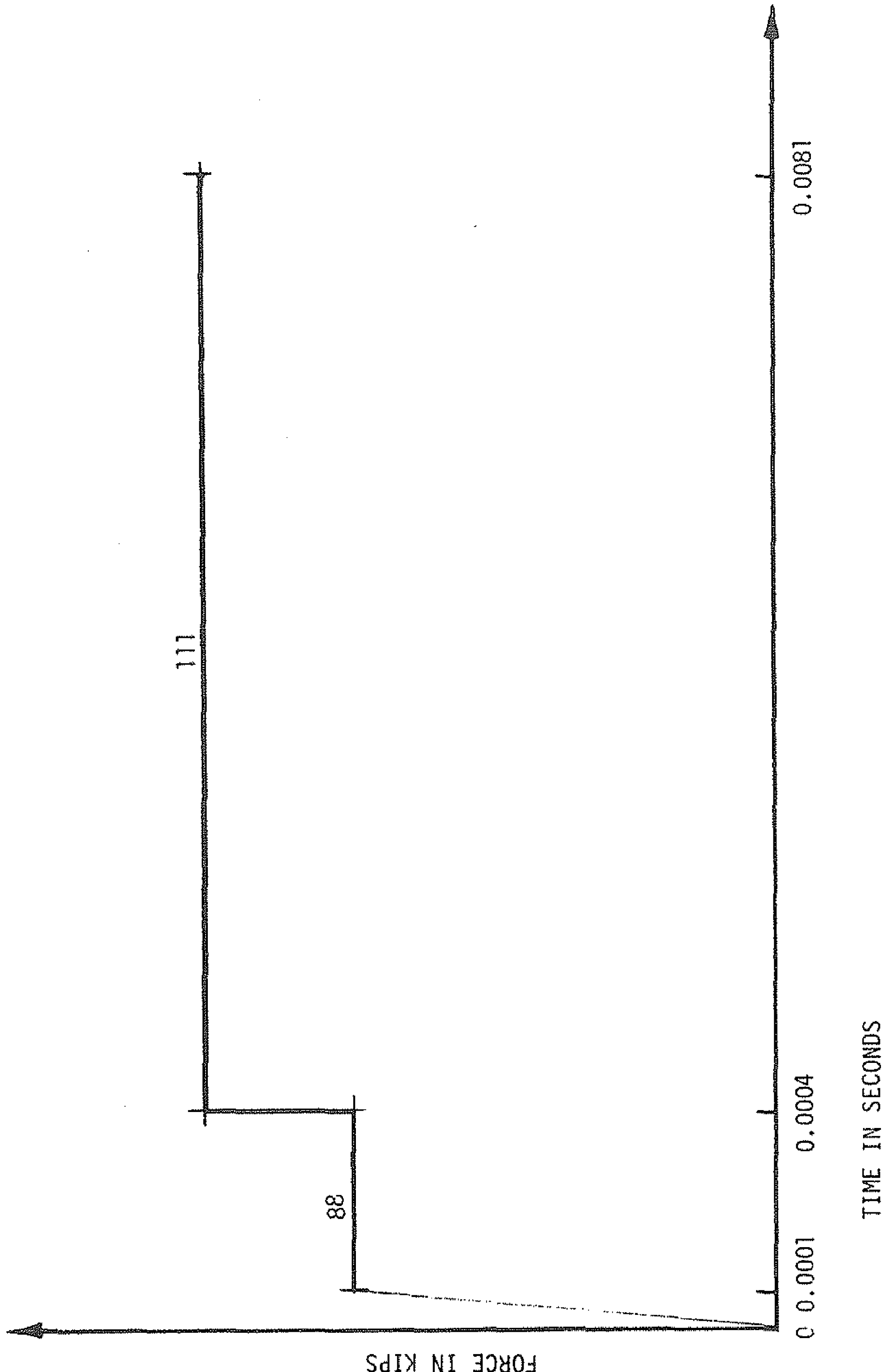
Recirculation Riser Circumferential
Break R5 at Reactor Vessel
Nozzle, Rupture Force Time Curve

Figure 3.6-70



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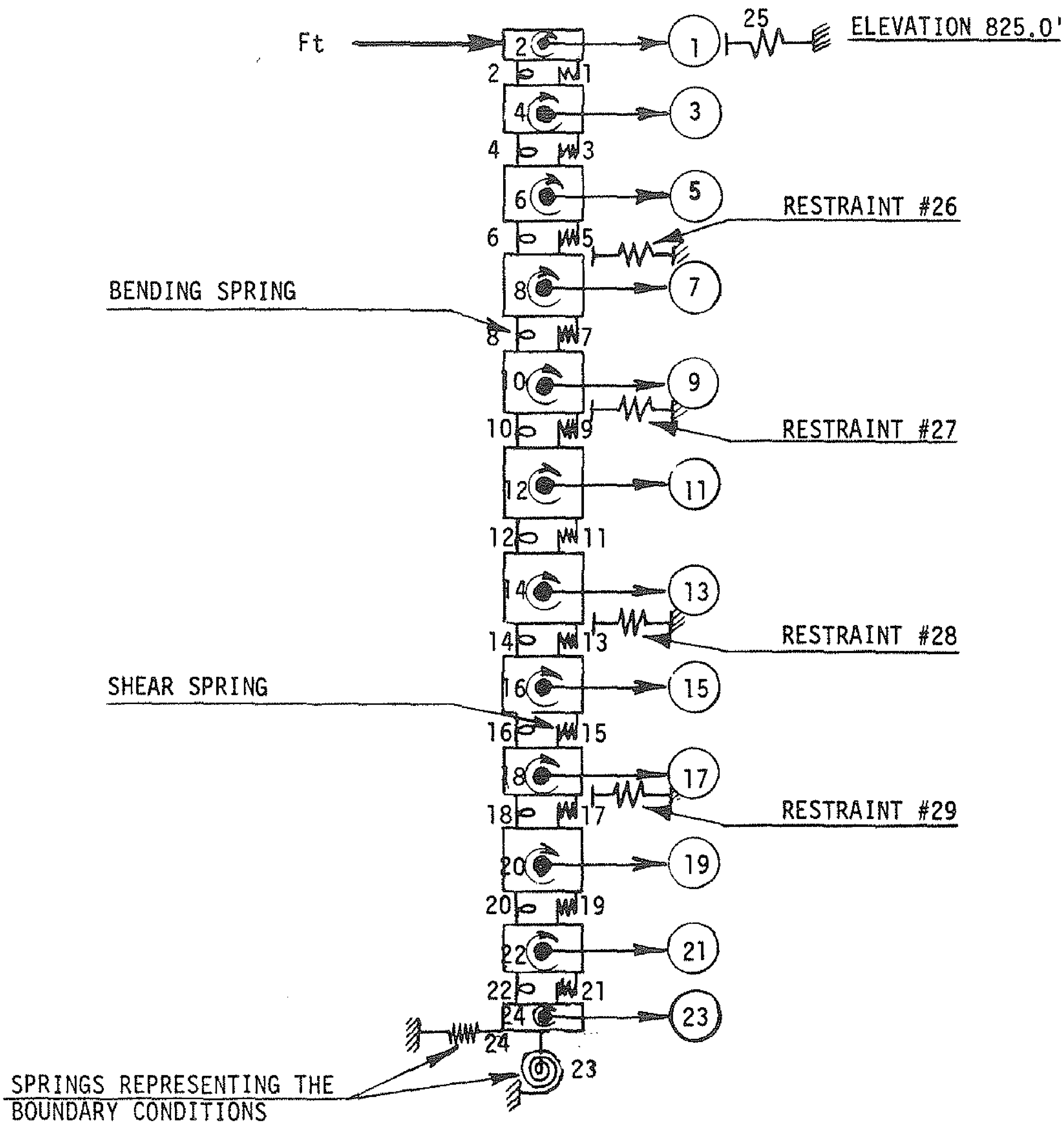
Recirculation Header Circumferential
Break R6 at Riser/Header
Tee, Rupture Force Time Curve
Figure 3.6-71



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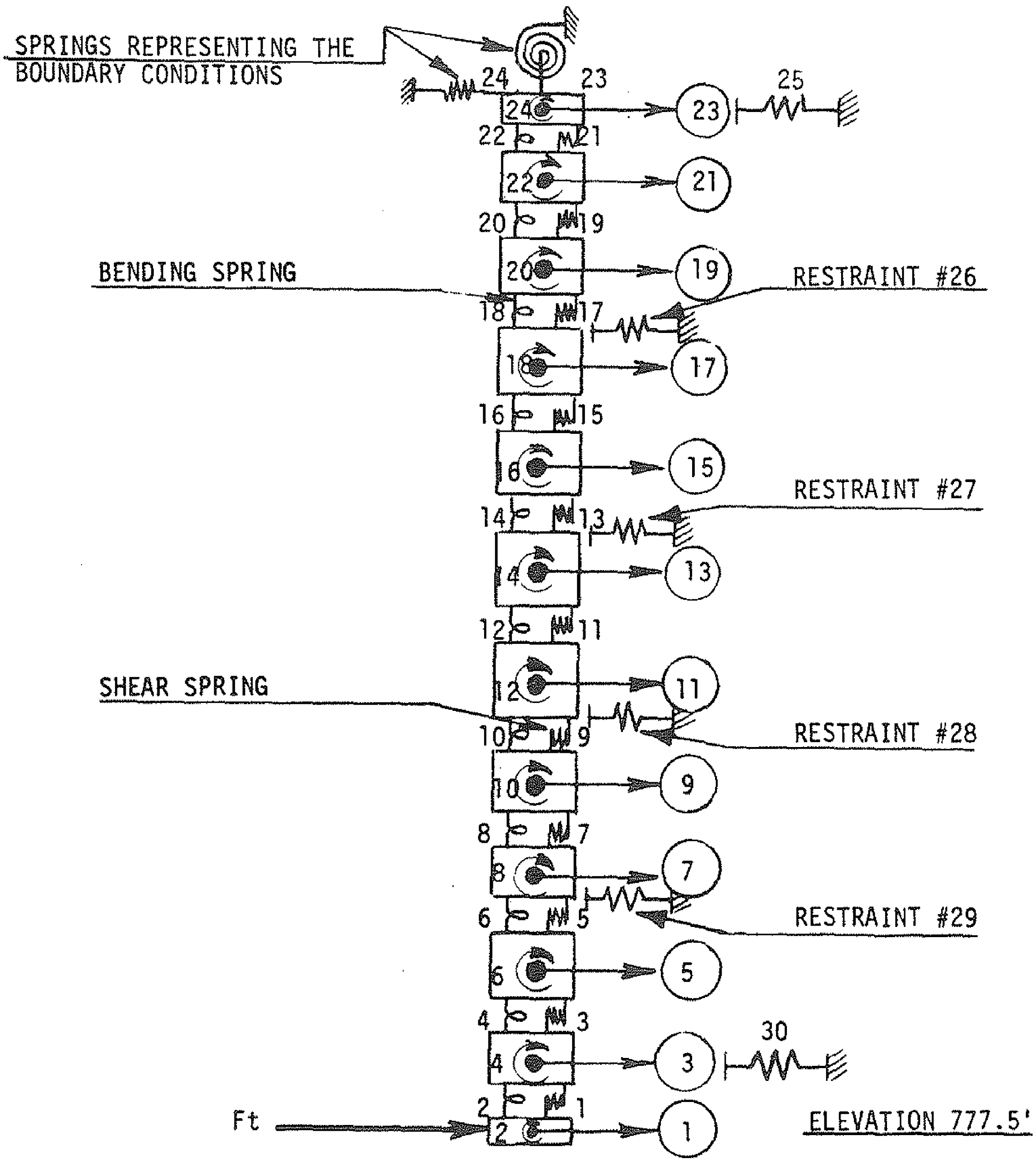
Main Steam Line Circumferential Break at Reactor Vessel Nozzle, Mathematical Model M1

Figure 3.6-72



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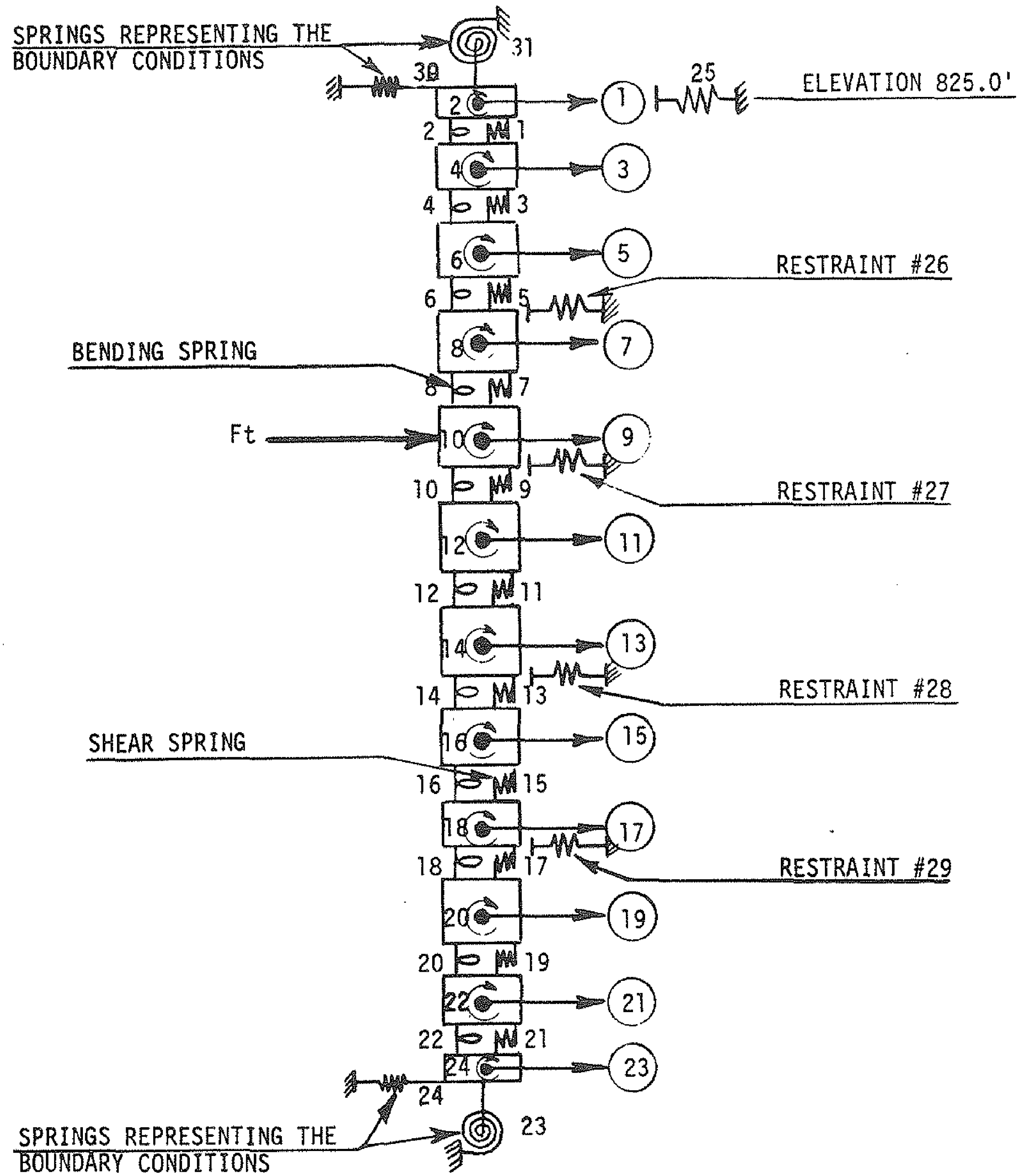
Main Stream Line Circumferential
Break at Horizontal
Header, Mathematical Model M2
Figure 3.6-73



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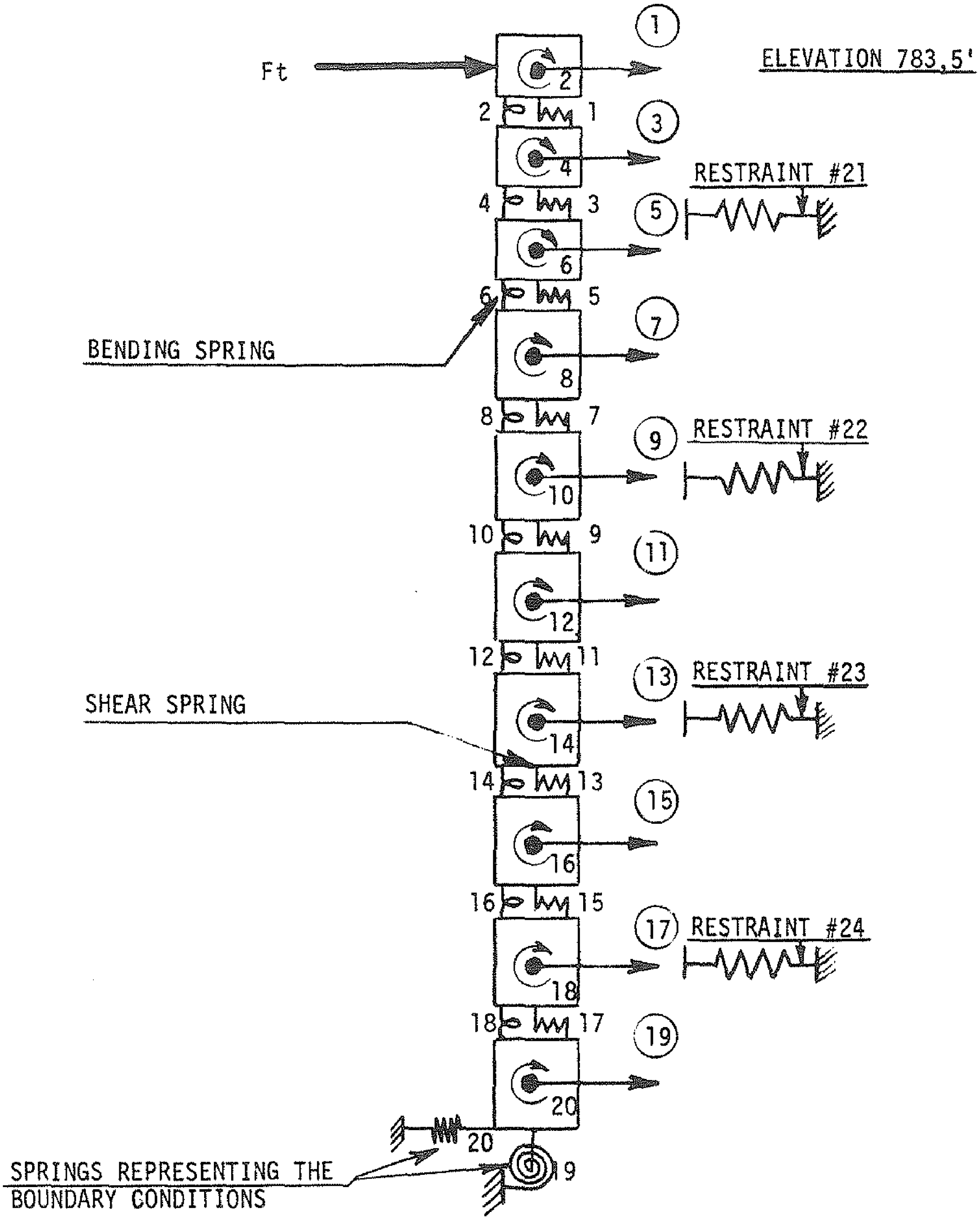
Main Steam Line Longitudinal Break at Mass Point 9, Mathematical Model M3 and M4 Figure 3.6-74



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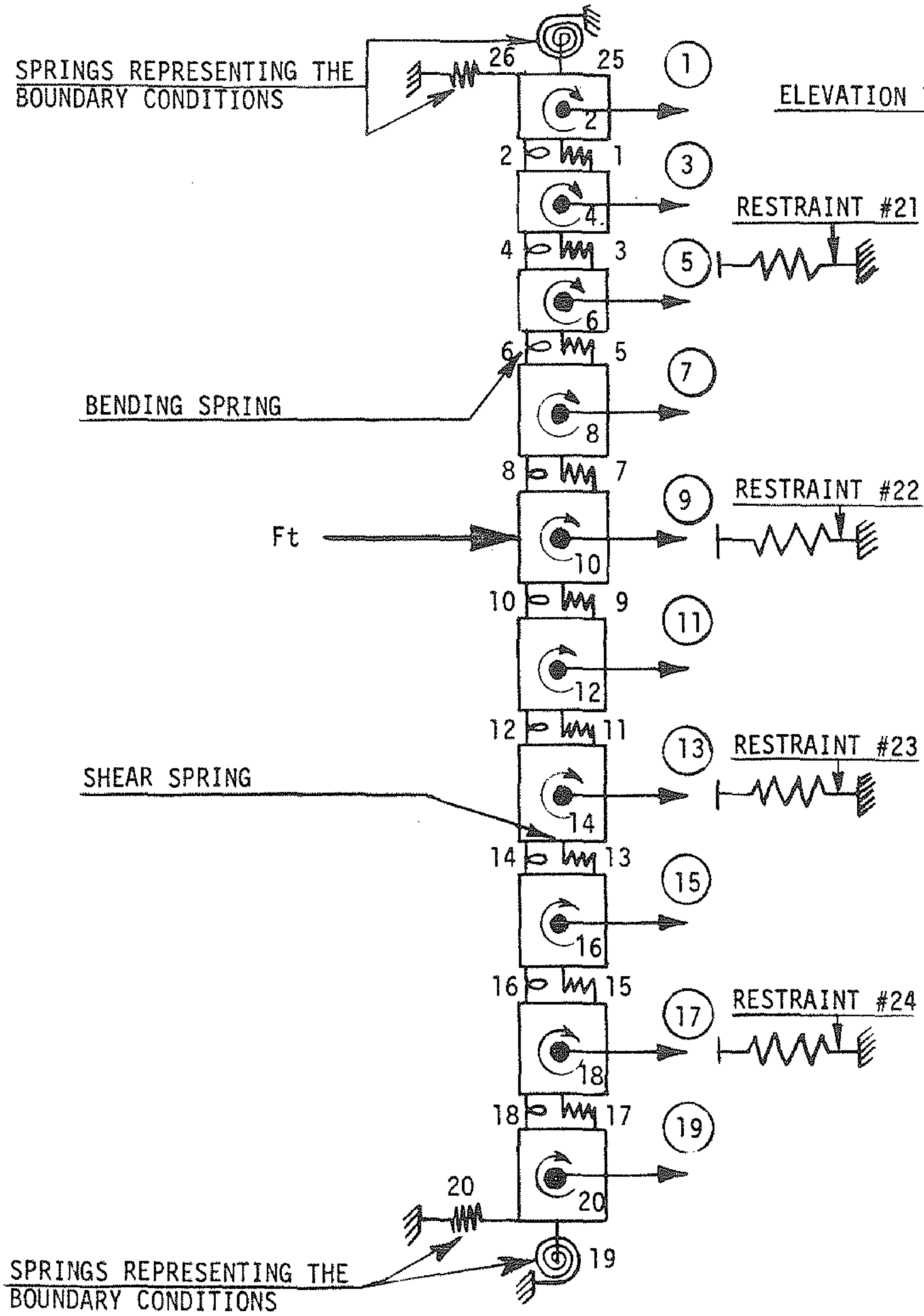
Recirculation Suction Line
Circumferential Break at Reactor
Vessel Nozzle, Mathematical Model R1

Figure 3.6-75



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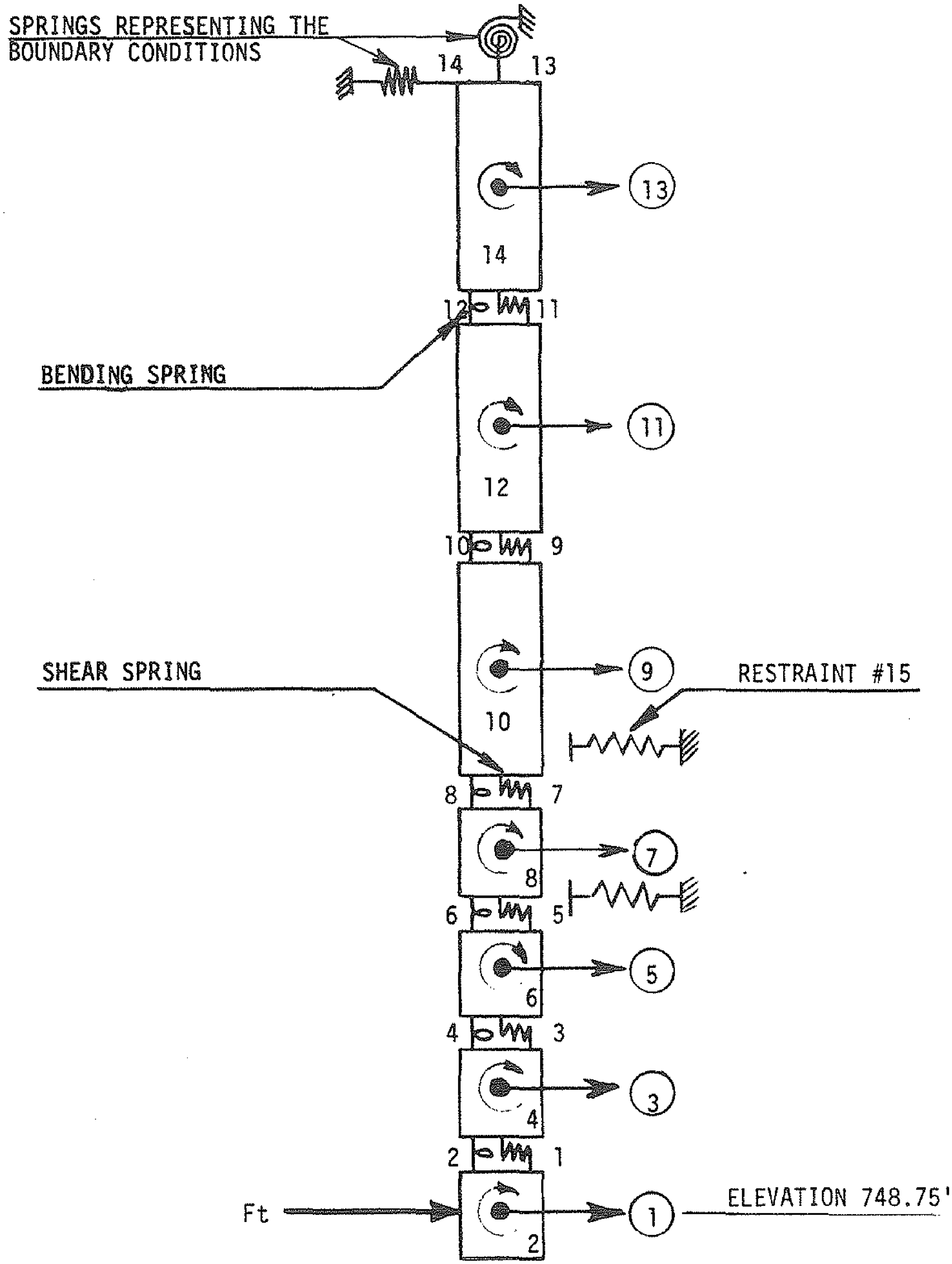
Recirculation Suction Line Longitudinal Break at Mass Point 9, Mathematical Model R2 and R3

Figure 3.6-76



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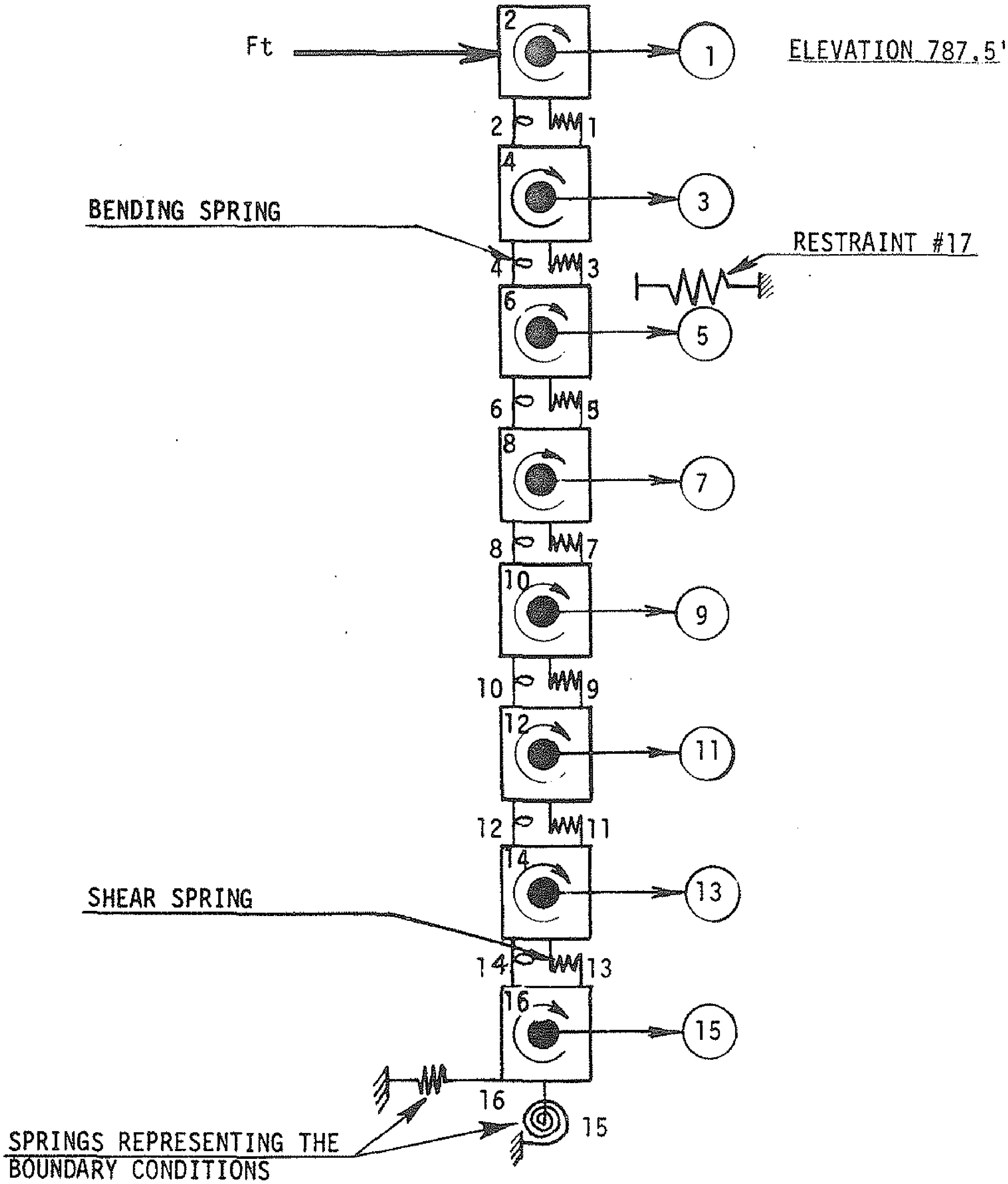
Recirculation Discharge Line
Circumferential Break at
Discharge Elbow, Mathematical Model R4
Figure 3.6-77



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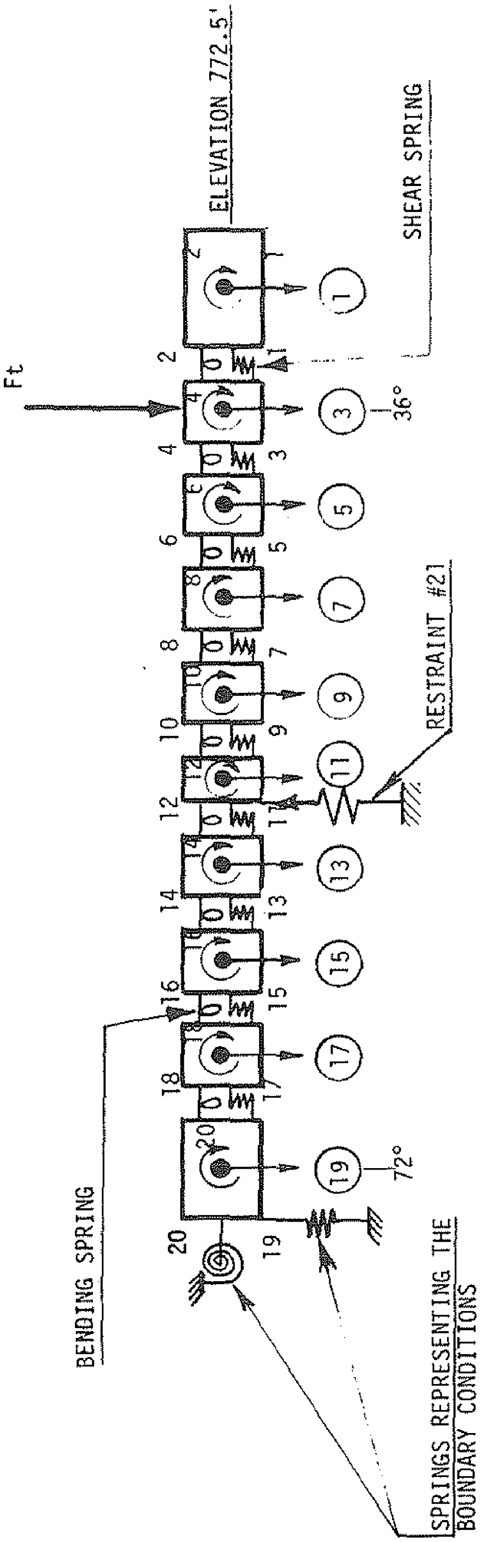
Recirculation Riser Circumferential Break at Reactor Vessel Nozzle, Mathematical Model R5

Figure 3.6-78



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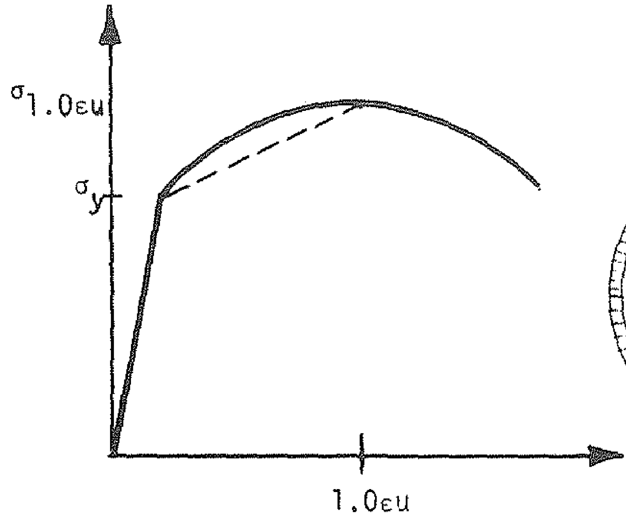
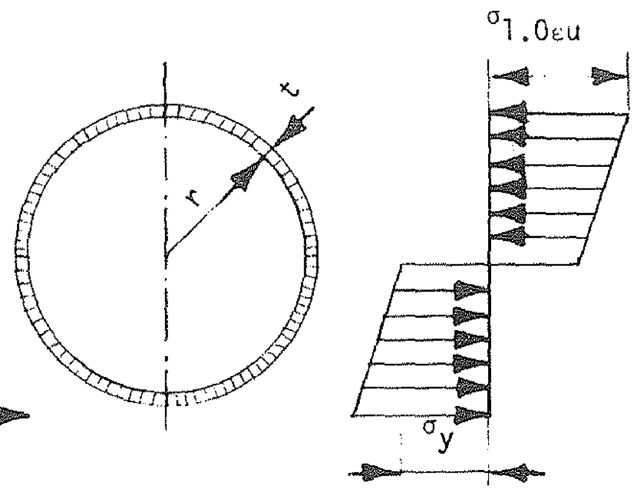
Recirculation Header Circumferential Break at Riser/Header Tee, Mathematical Model R6

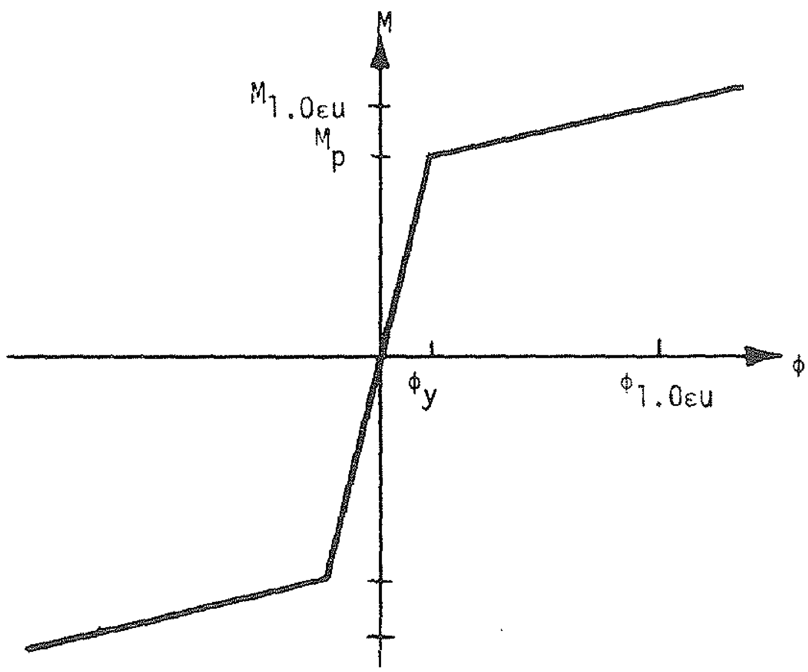
Figure 3.6-79



Idealized Stress Strain Curve



Assumed Bending Stress Distribution



Moment-Rotation Curve

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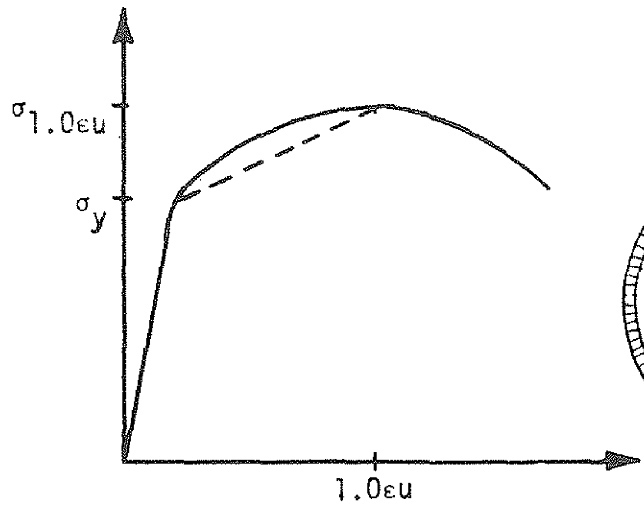
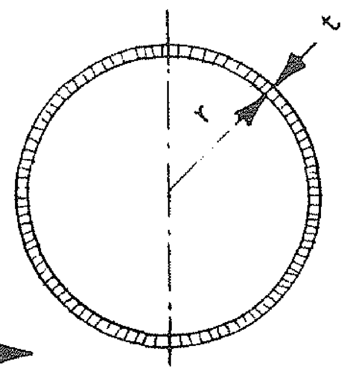
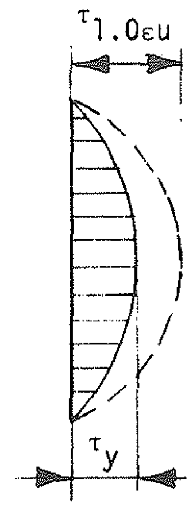
Load Deflection Curve for
Nonlinear Spring Bending Spring

Figure 3.6-80

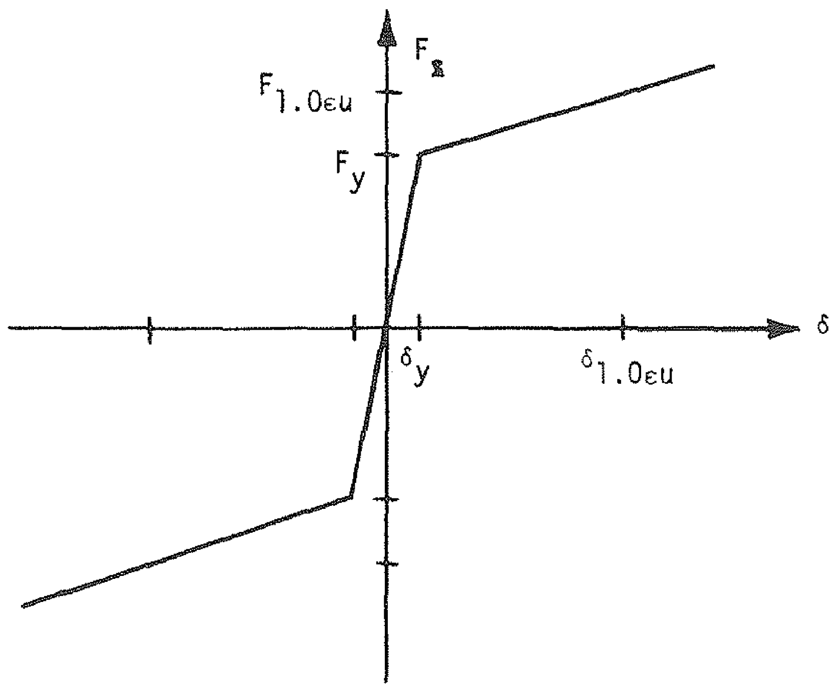




Idealized Stress Strain Curve



Assumed Shear Stress Distribution



Load-Deflection Curve

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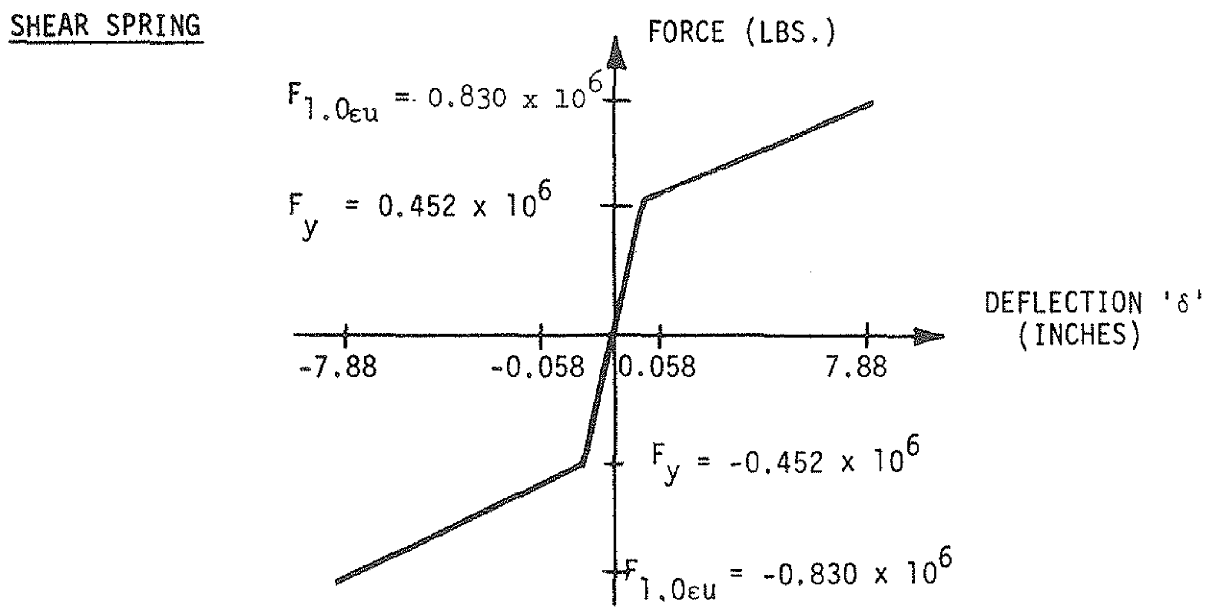
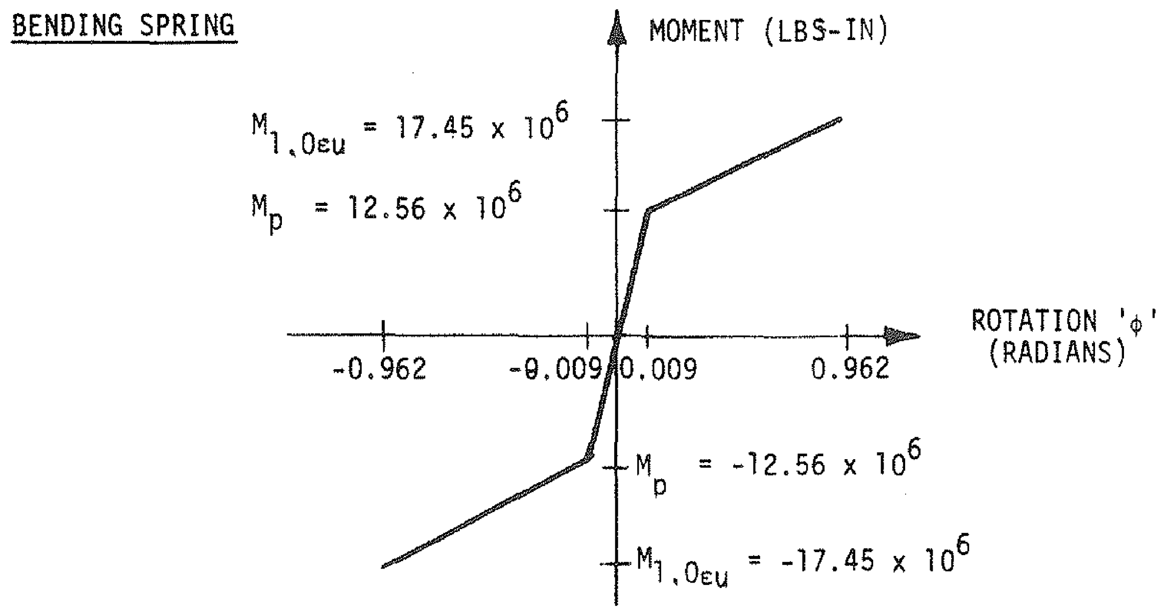
Load Deflection Curve for
Nonlinear Spring Shear Spring

Figure 3.6-81





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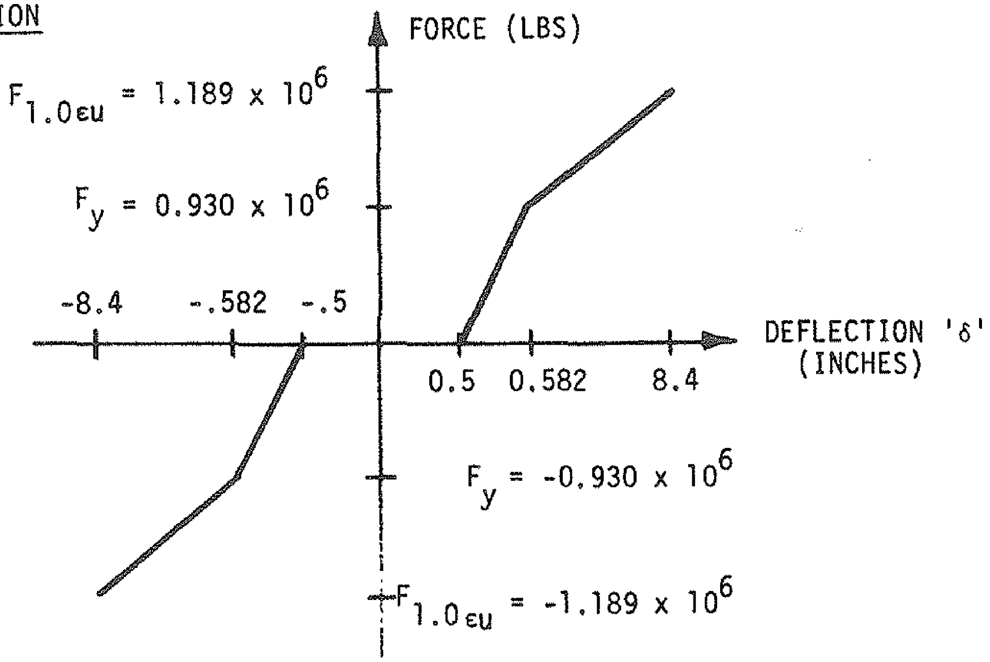
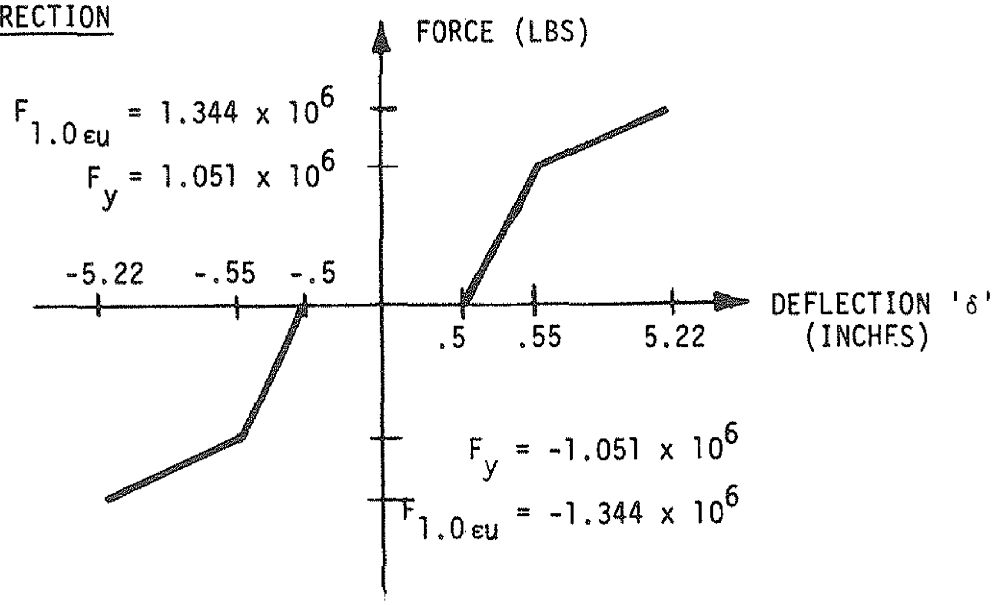
Main Steam Line Typical Load Deflection Curve for Pipe

Figure 3.6-82

RADIAL DIRECTION



TANGENTIAL DIRECTION



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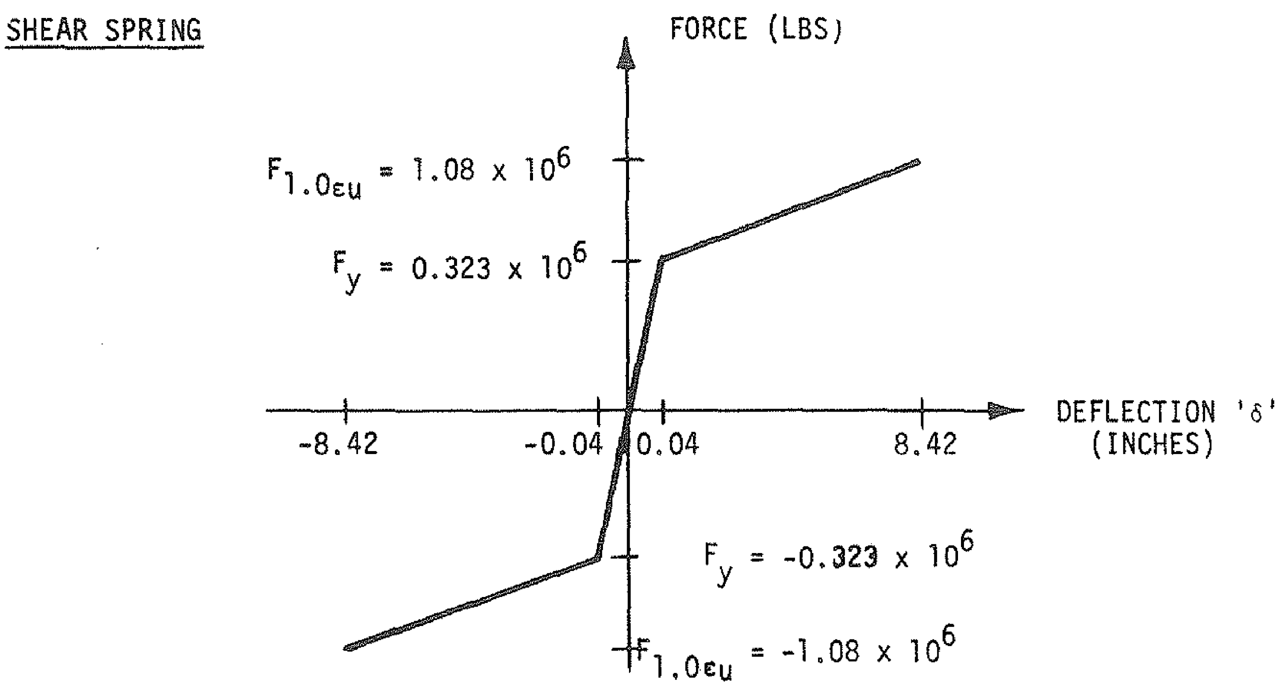
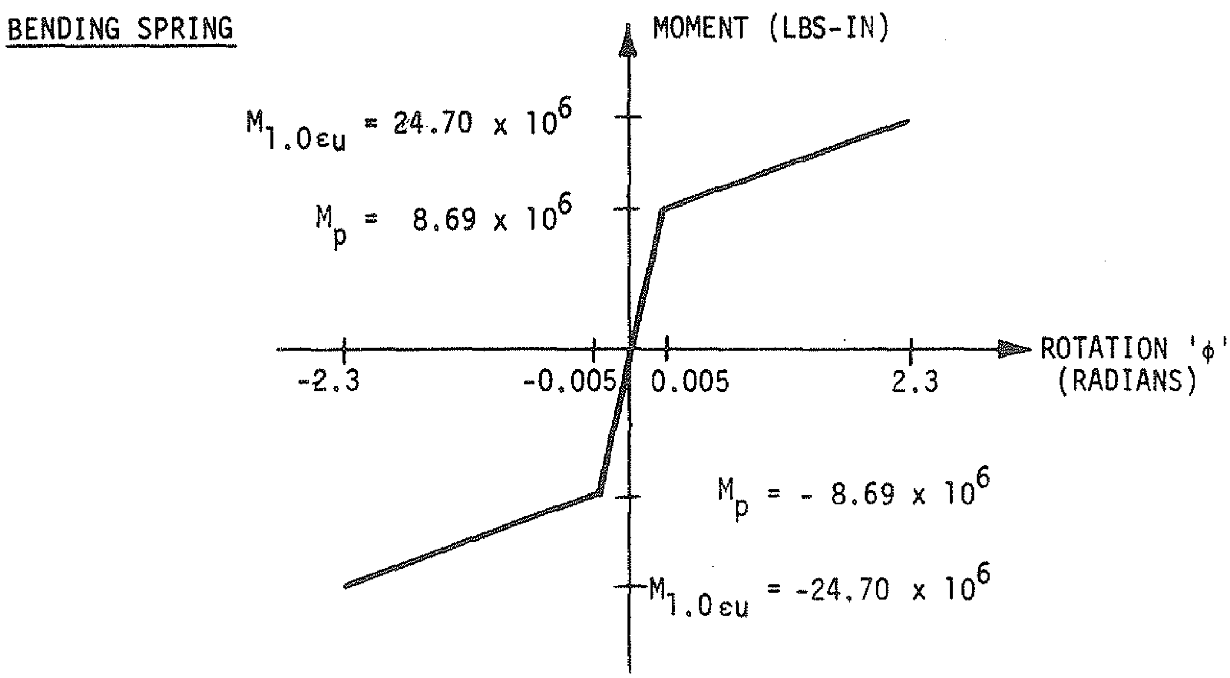
Main Steam Line Typical Load
Deflection Curve for Restraint

Figure 3.6-83





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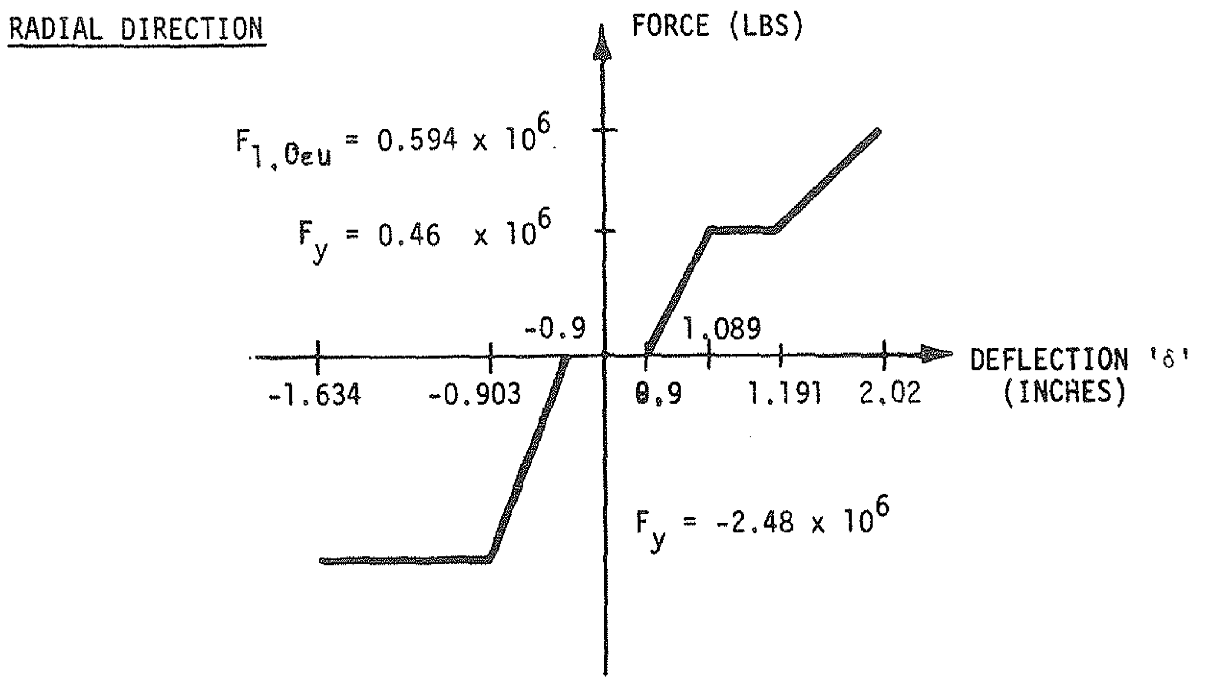
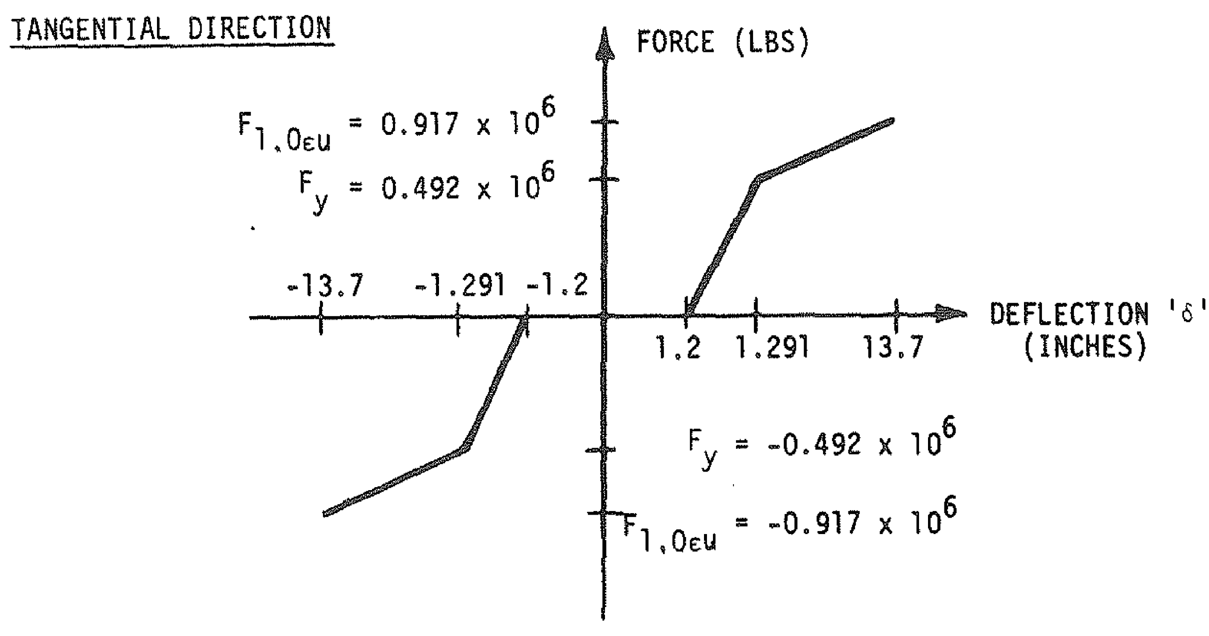
Recirculation Suction Line Typical
Load Deflection Curve for Pipe

Figure 3.6-84





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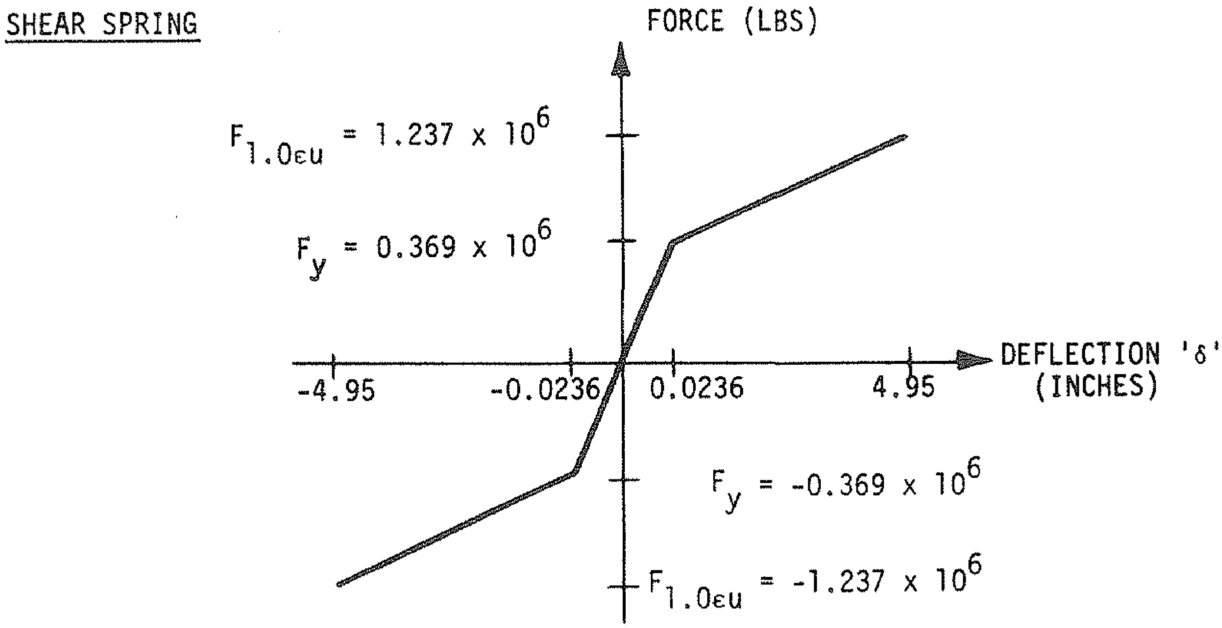
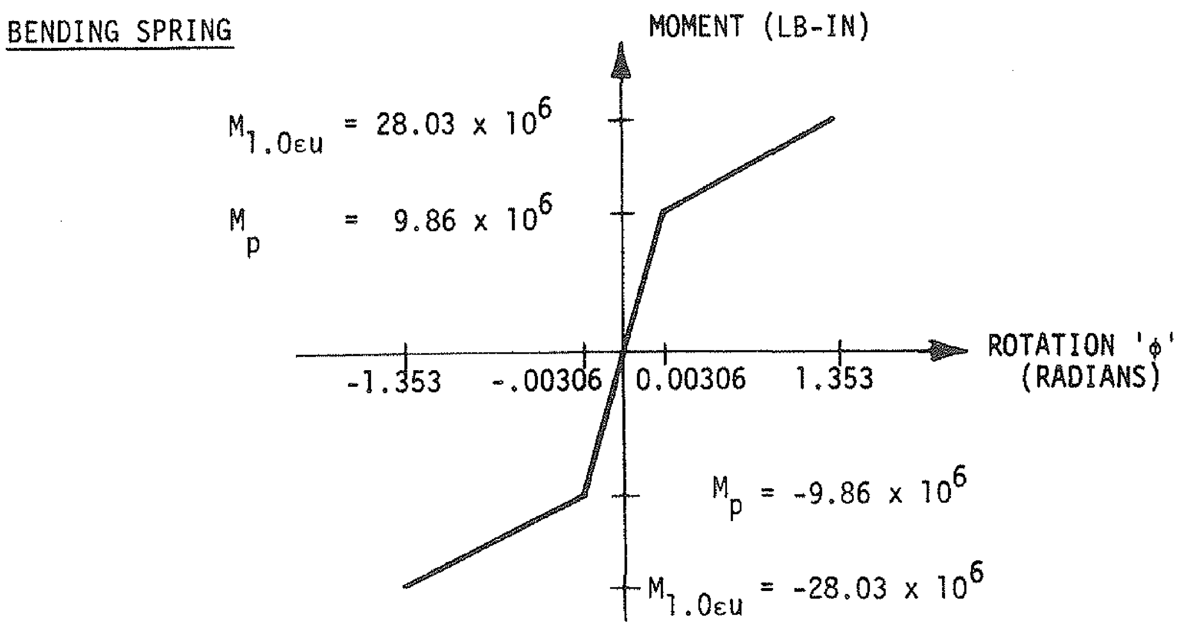
Recirculation Suction Line Typical Load Deflection Curve for Restraint

Figure 3.6-85





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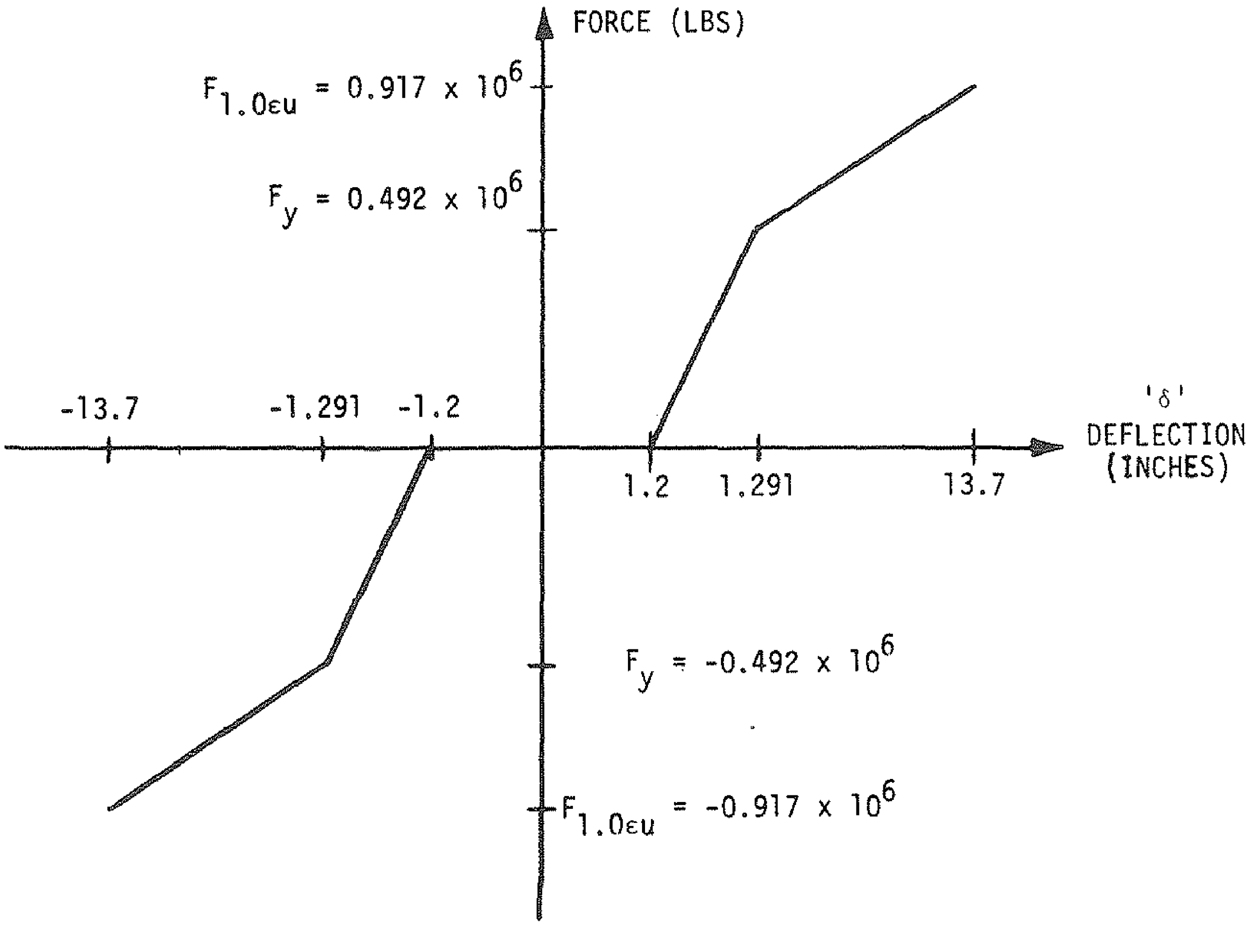
Recirculation Discharge Line Typical Load Deflection Curve for Pipe

Figure 3.6-86



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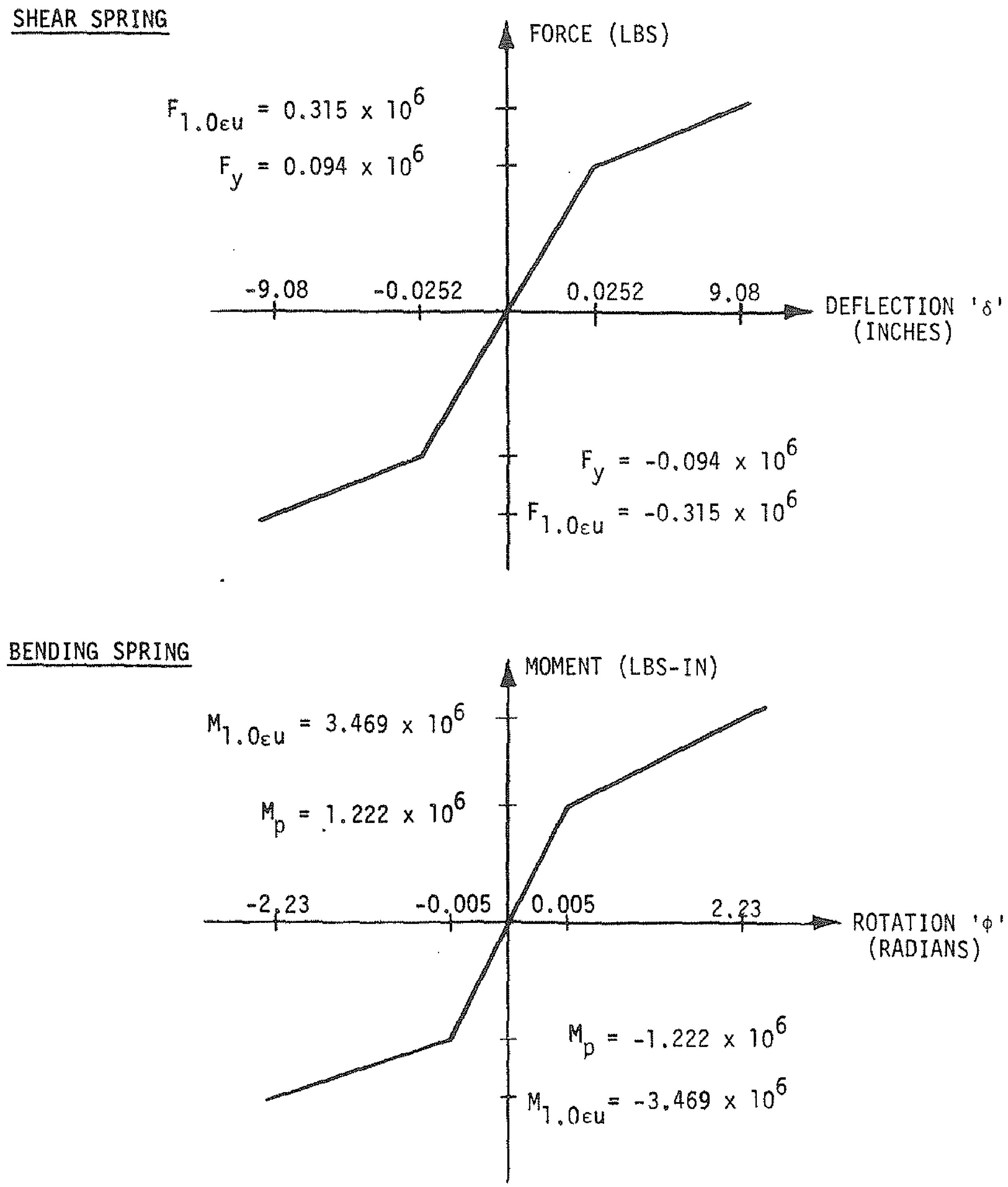
Recirculation Discharge Line Typical
Load Deflection Curve for Restraint

Figure 3.6-87



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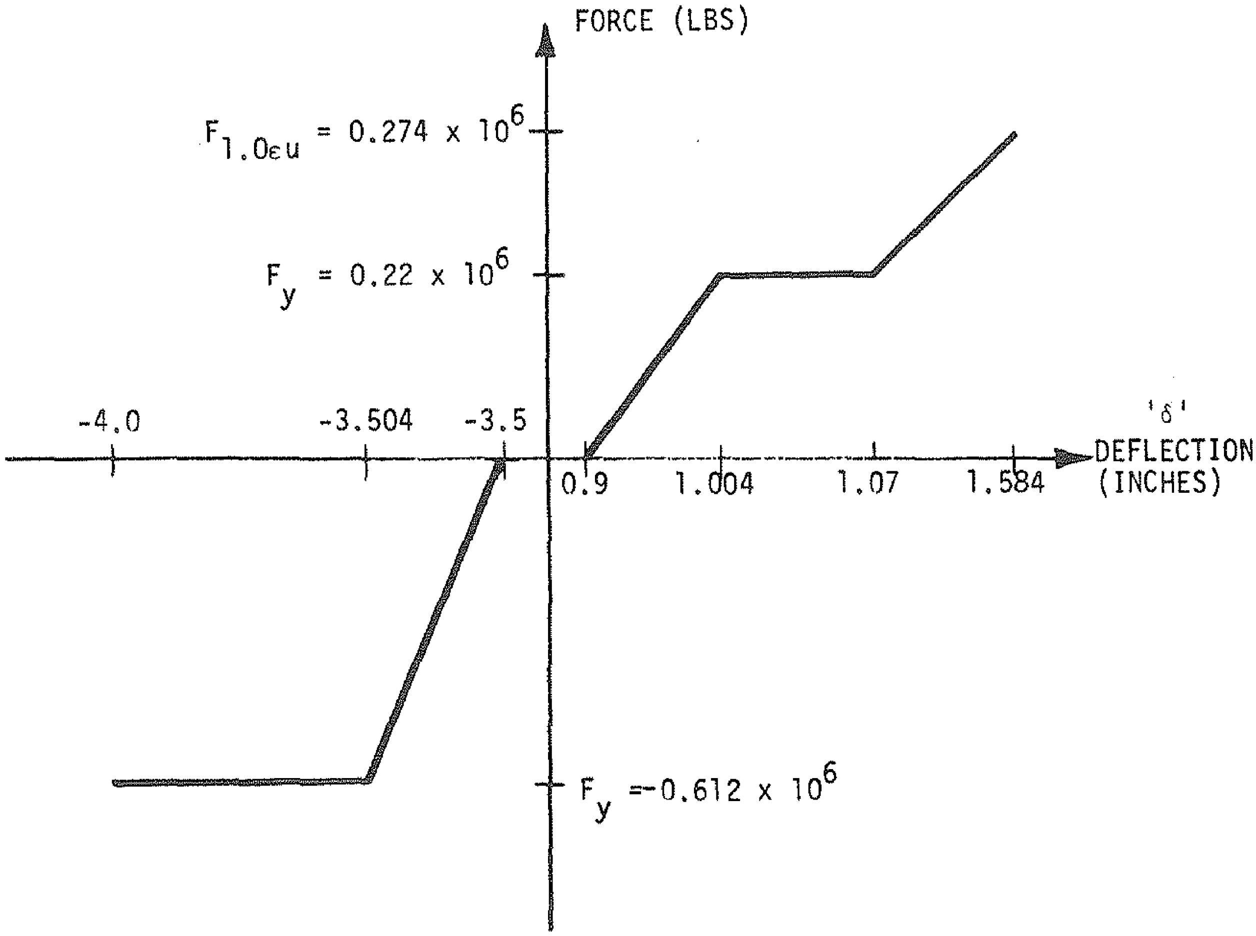
Recirculation Riser Typical Load Deflection Curve for Pipe

Figure 3.6-88



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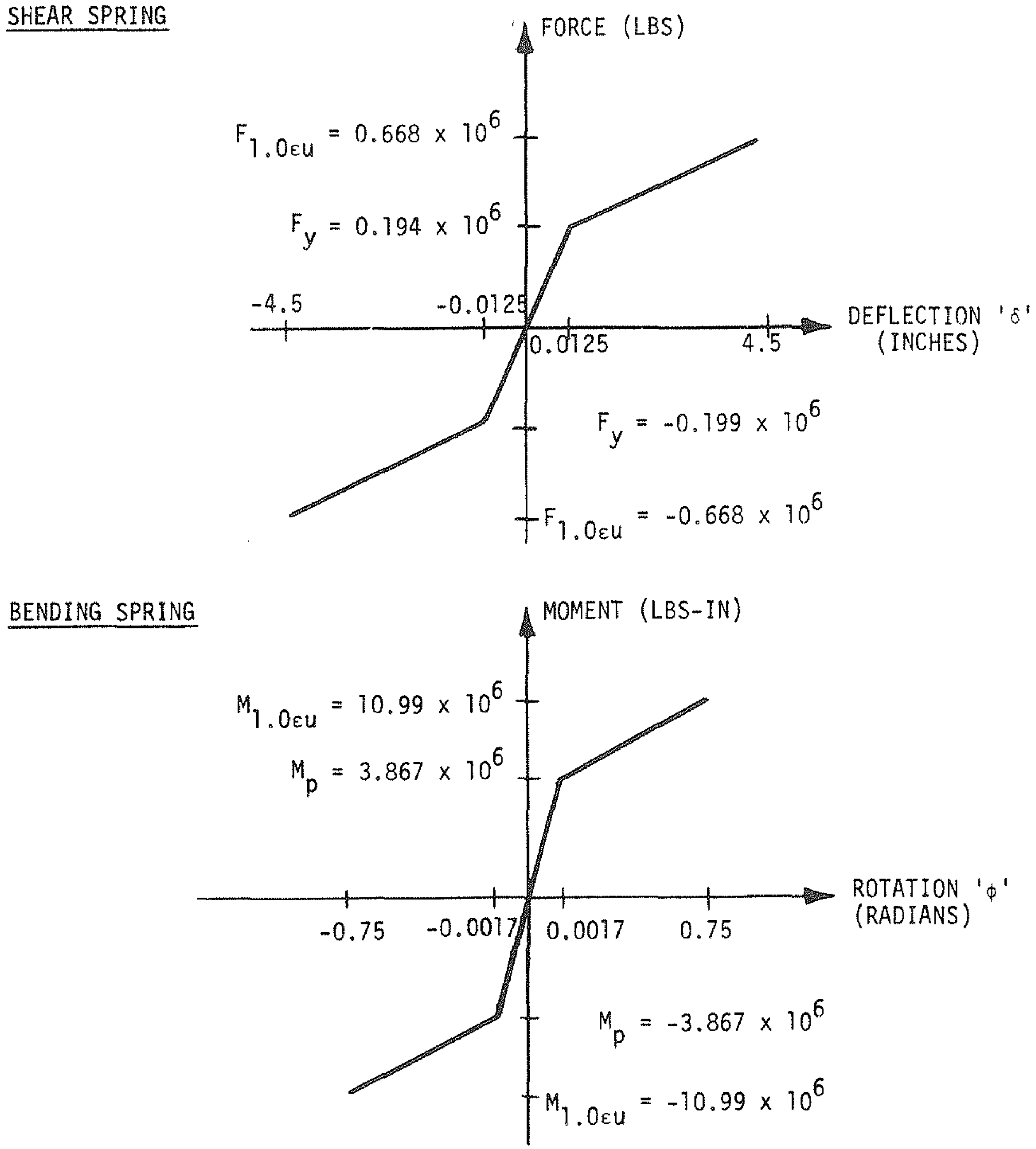
Recirculation Riser Typical Load Deflection Curve for Restraint

Figure 3.6-89



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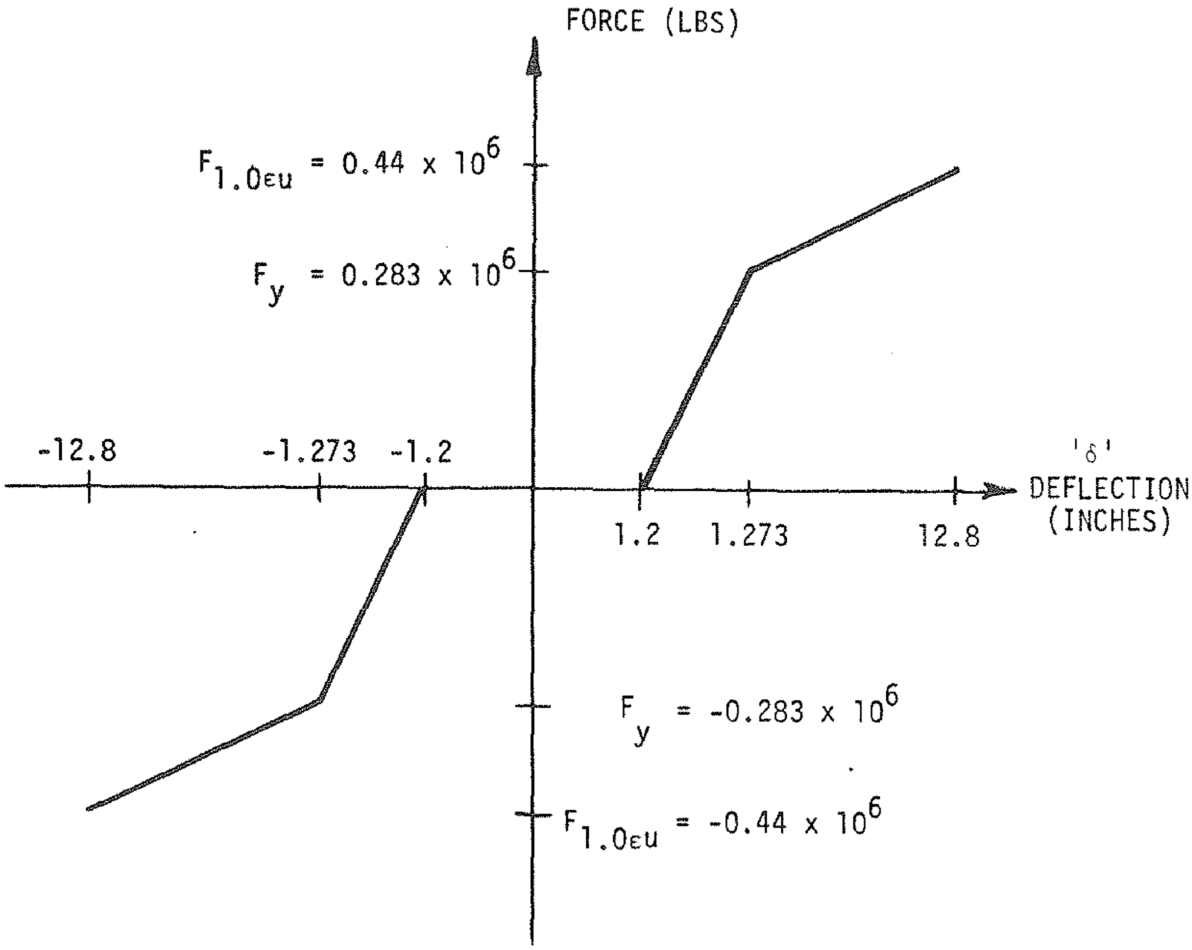
Recirculation Header Typical Load Deflection Curve for Pipe

Figure 3.6-90



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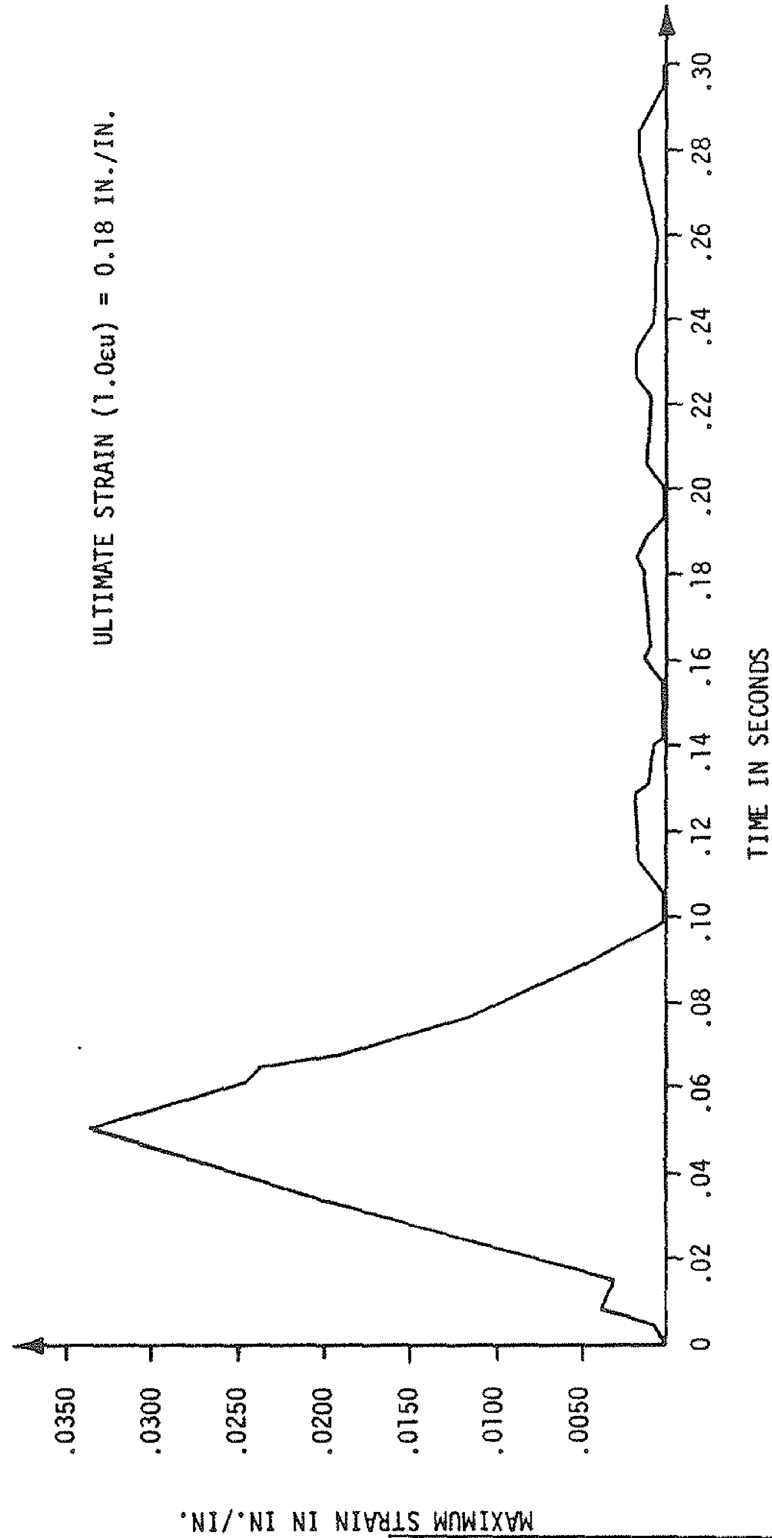
Recirculation Header Typical Load Deflection Curve for Restraint

Figure 3.6-91



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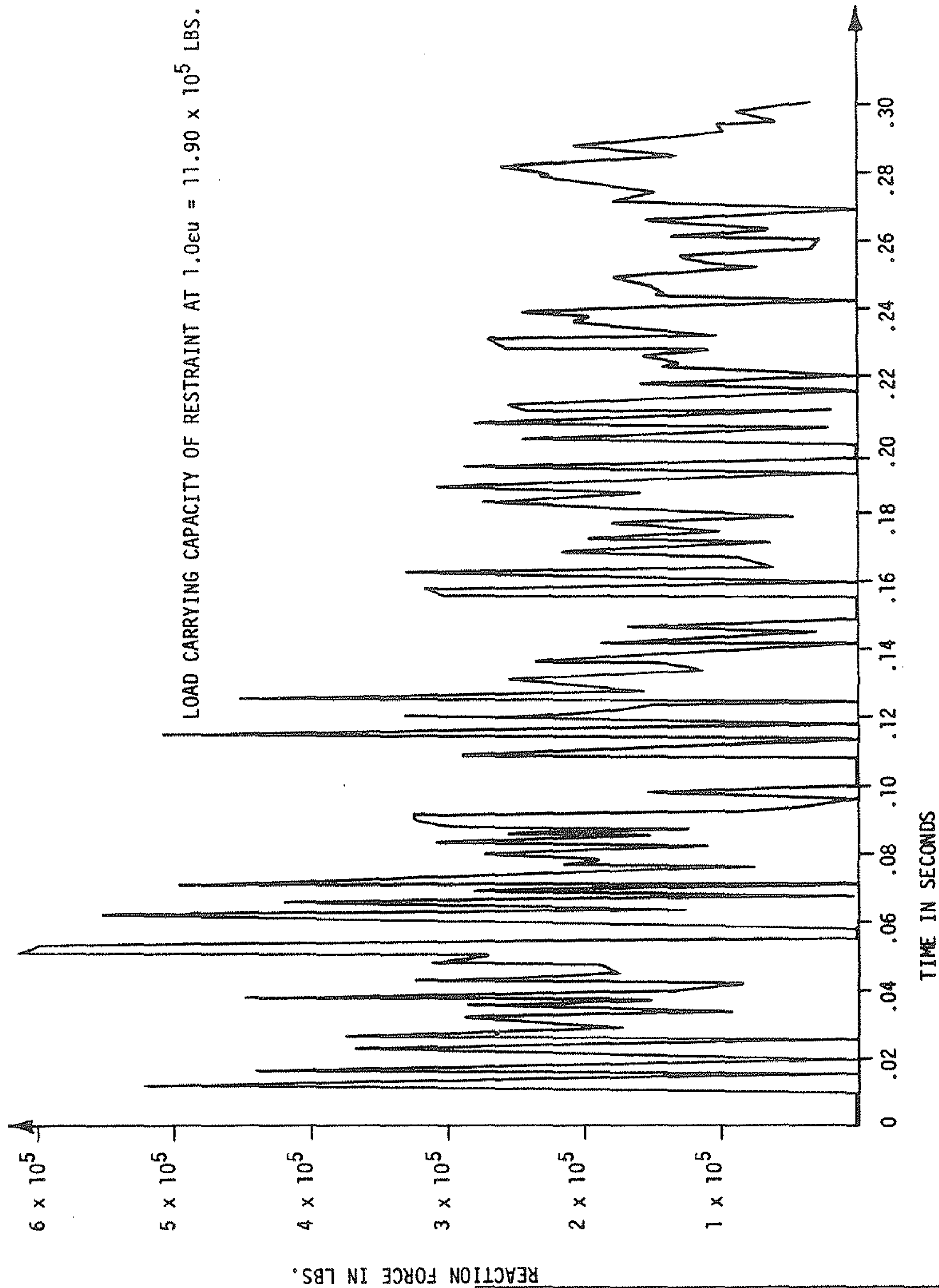
Main Steam Line Circumferential Break
M1 at Reactor Vessel Nozzle,
Maximum Strain Time History in Pipe
Figure 3.6-92



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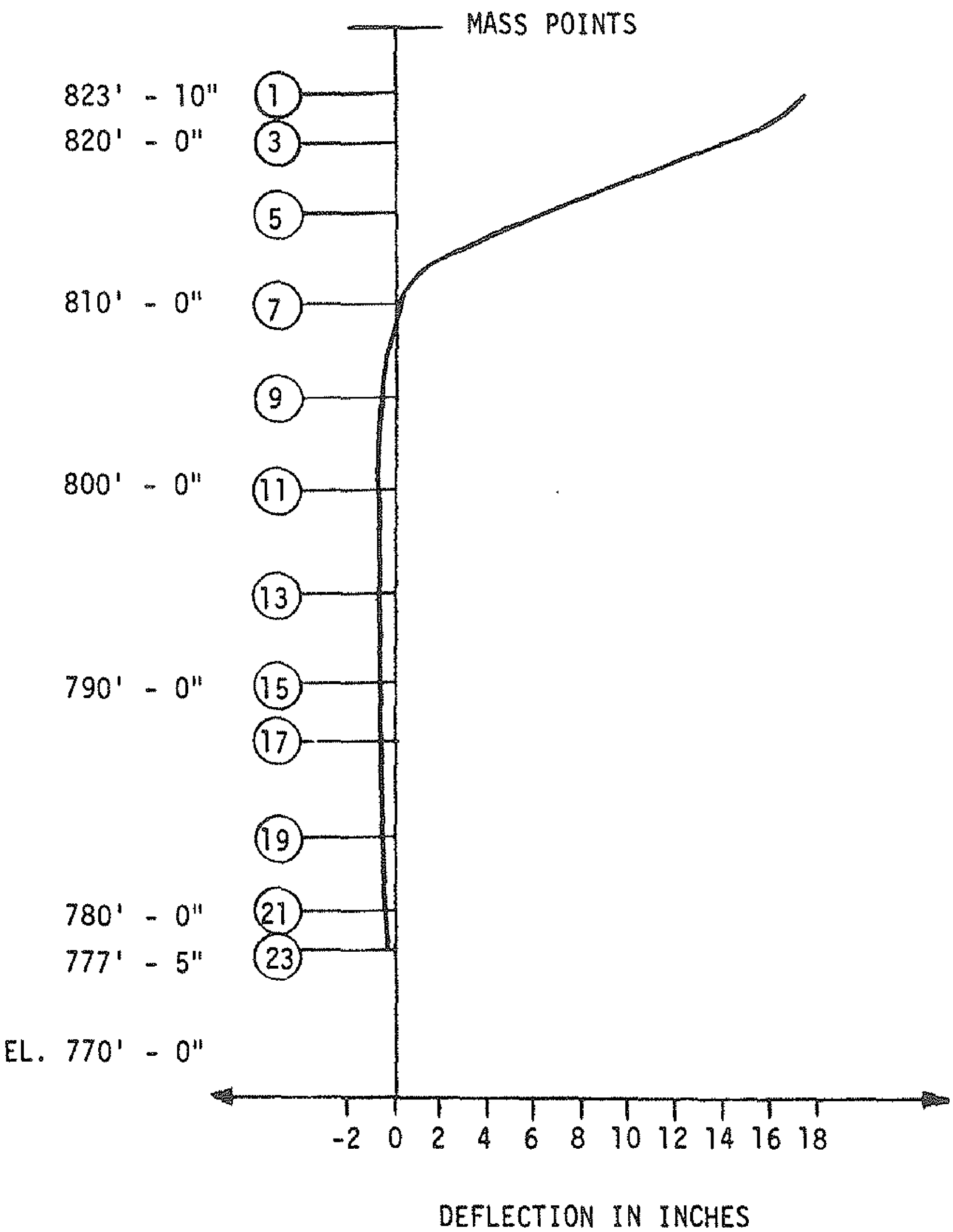
Main Steam Line Circumferential Break M1 at Reactor Vessel Nozzle, Maximum Reaction Force Time History in Restraint No. 26

Figure 3.6-93



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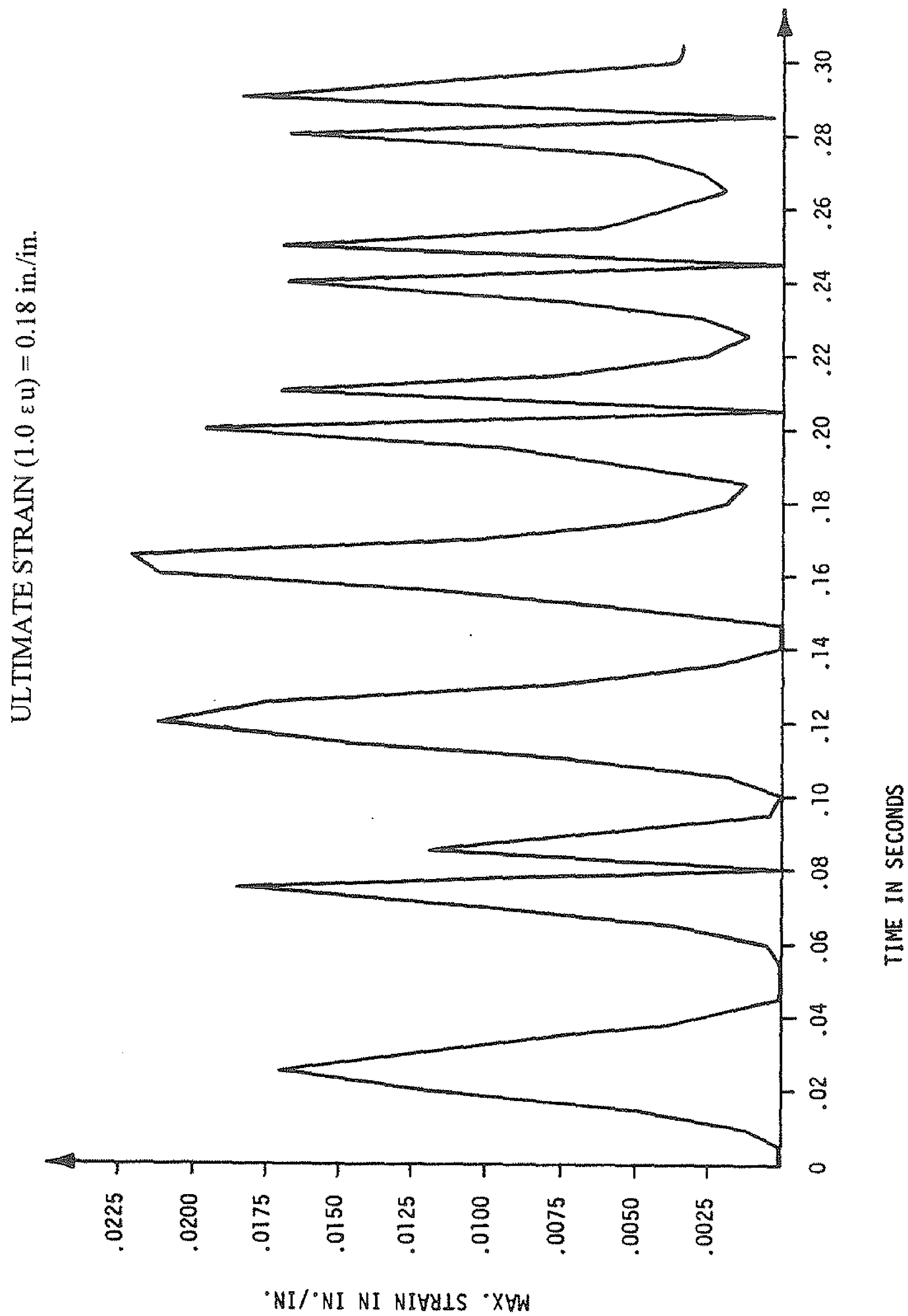
Main Steam Line Circumferential Break M1 at Reactor Vessel Nozzle, Envelope of Maximum Deflection

Figure 3.6-94



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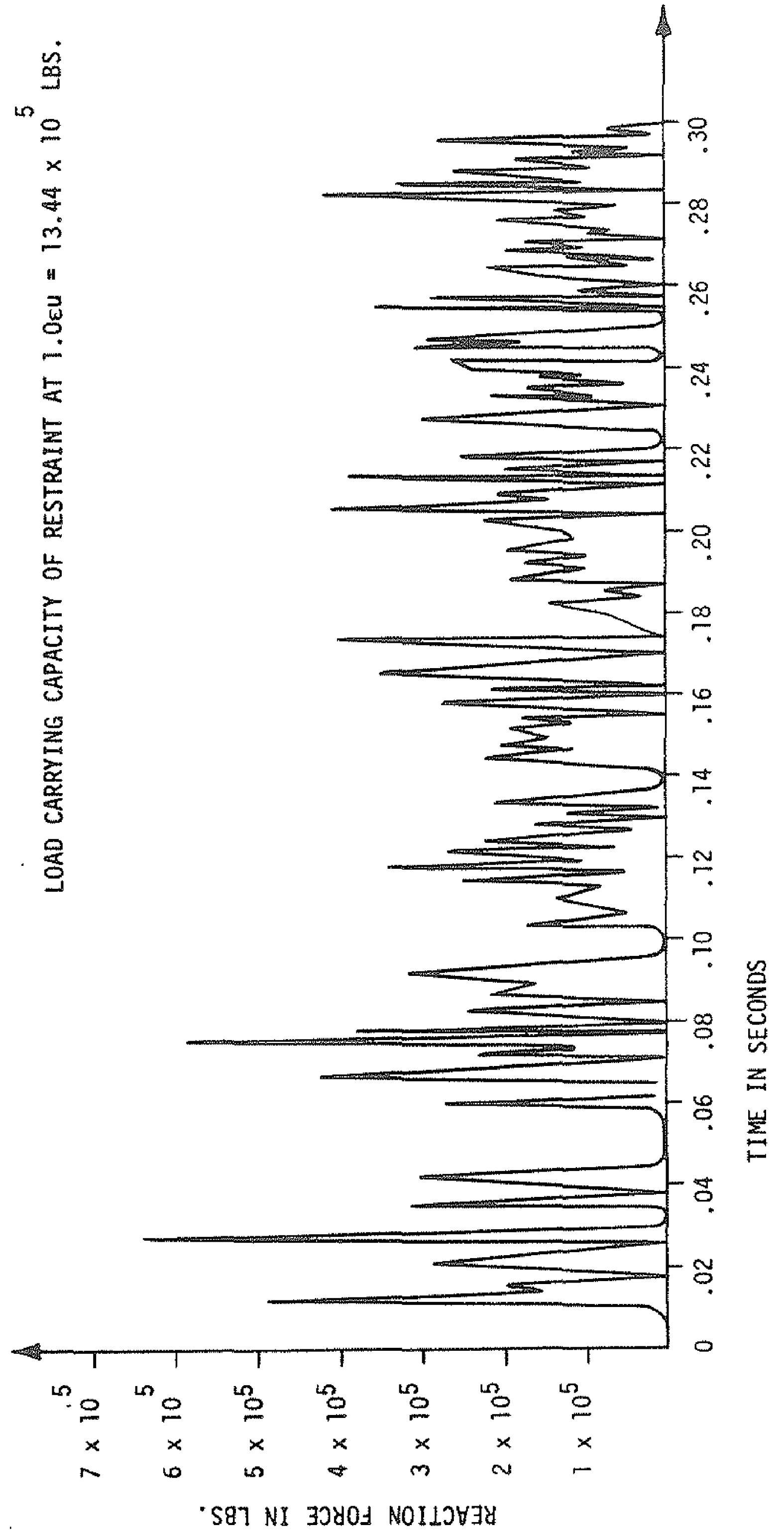
Main Steam Line Circumferential
Break M2 at Horizontal Header,
Maximum Strain Time History in Pipe

Figure 3.6-95



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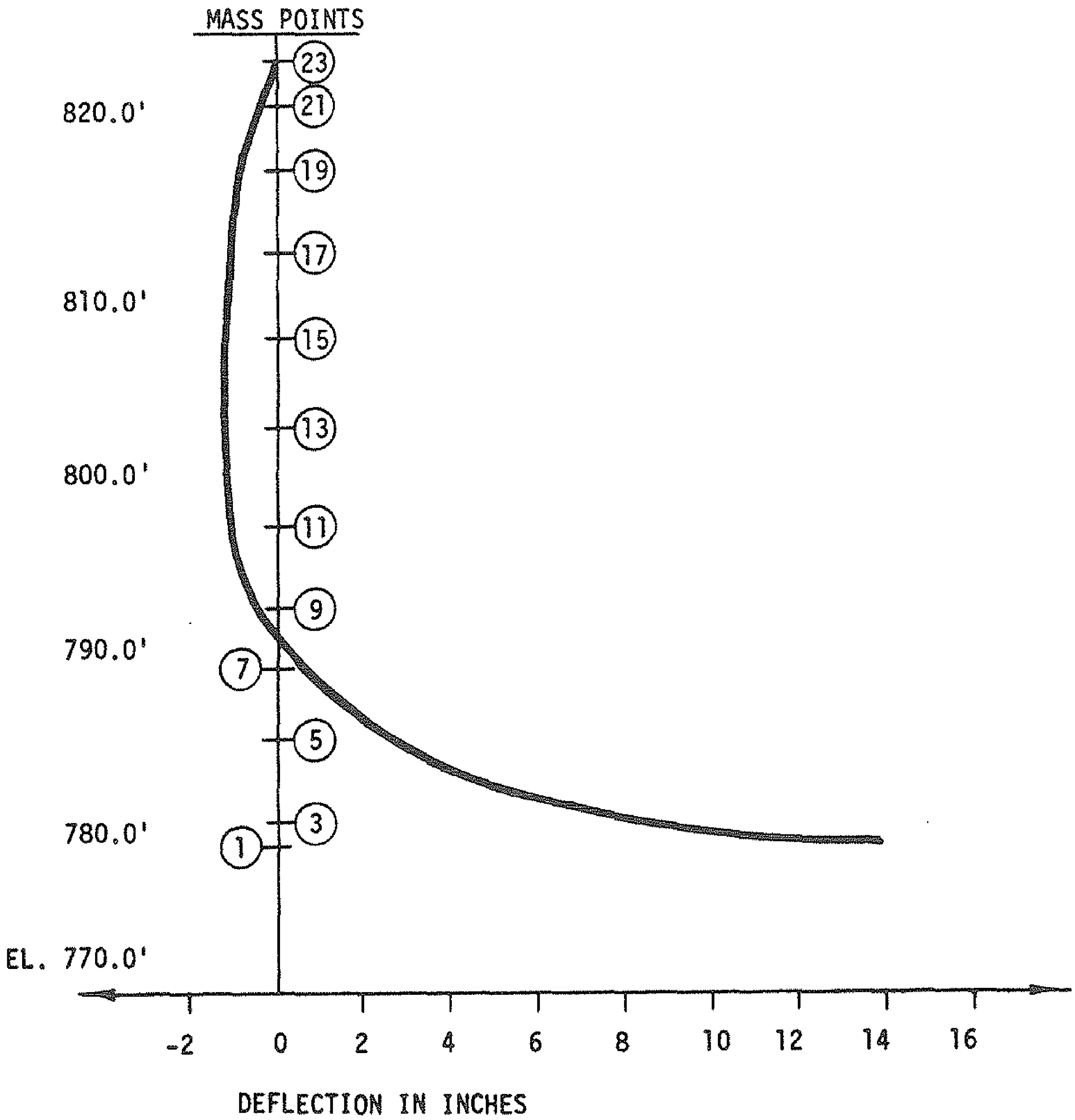
Main Steam Line Circumferential Break M2 at Horizontal Header, Maximum Reaction Force Time History in Restraint No. 29

Figure 3.6-96



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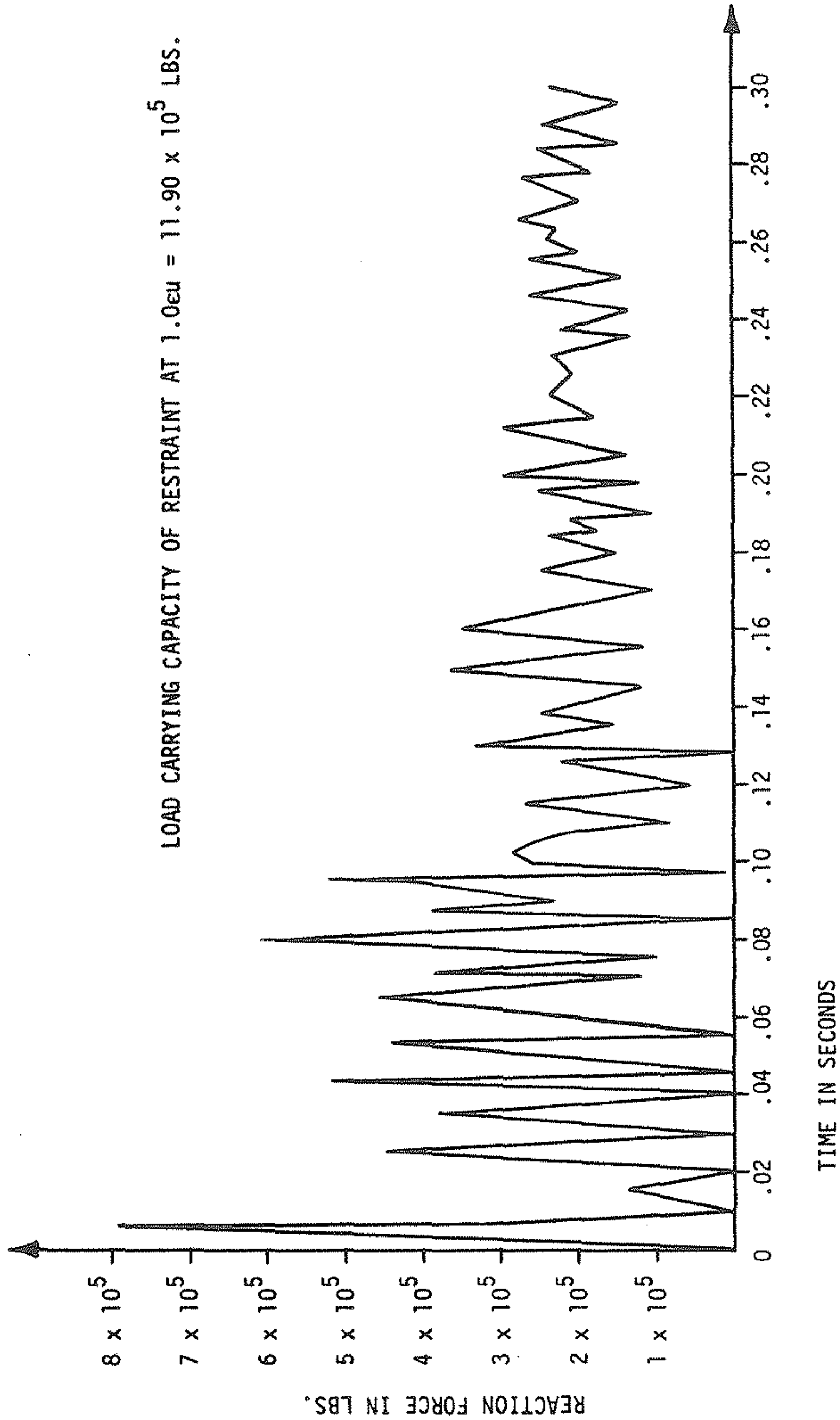
Main Steam Line Circumferential Break M2 at Horizontal Header, Envelope of Maximum Deflection

Figure 3.6-97



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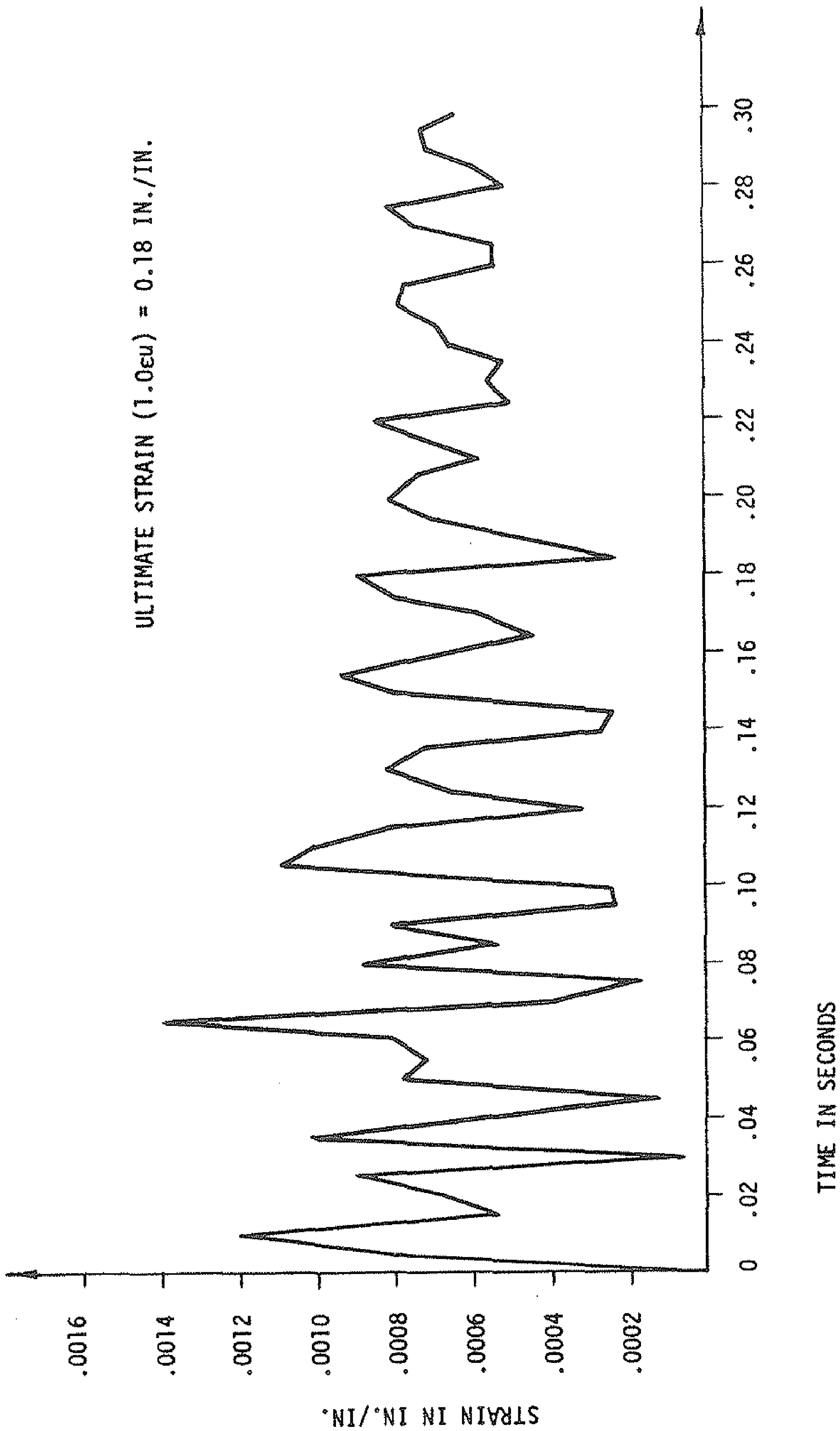
Main Steam Line Longitudinal Break M3
in Radial Direction, Maximum Reaction Force
Time History in Restraint No. 27

Figure 3.6-98



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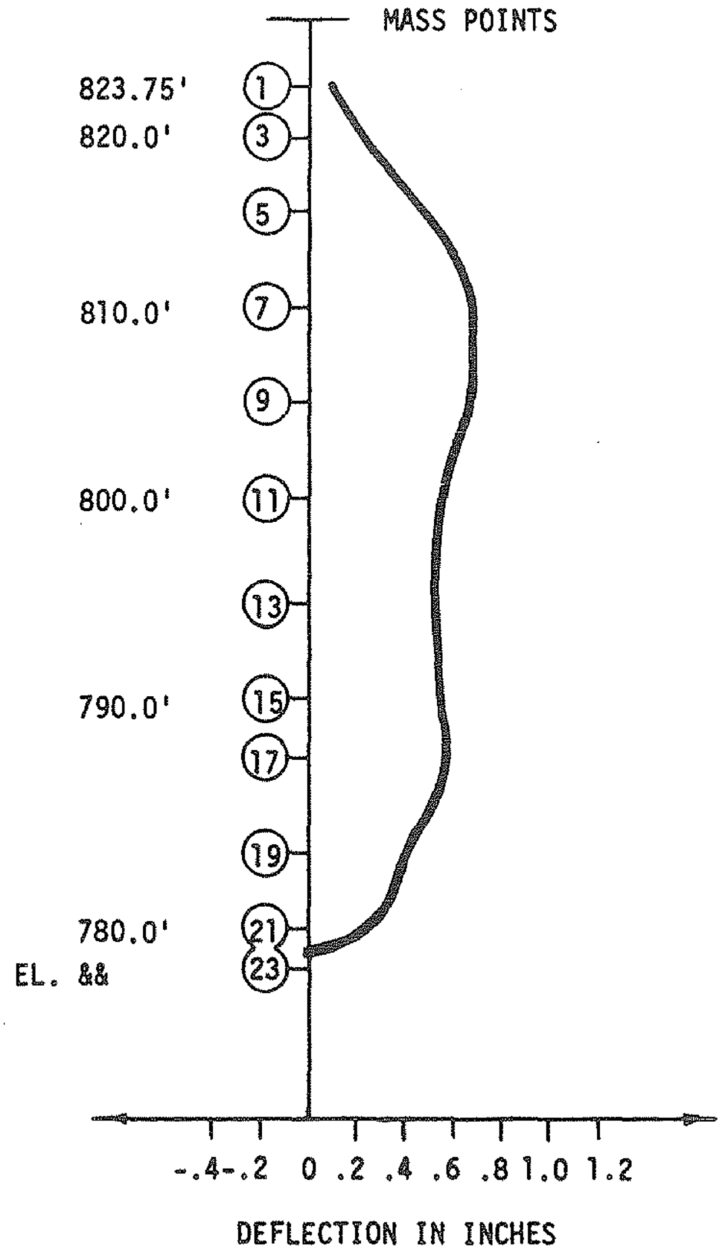
Main Steam Line Longitudinal
Break M3 in Radial Direction,
Maximum Strain Time History in Pipe

Figure 3.6-99



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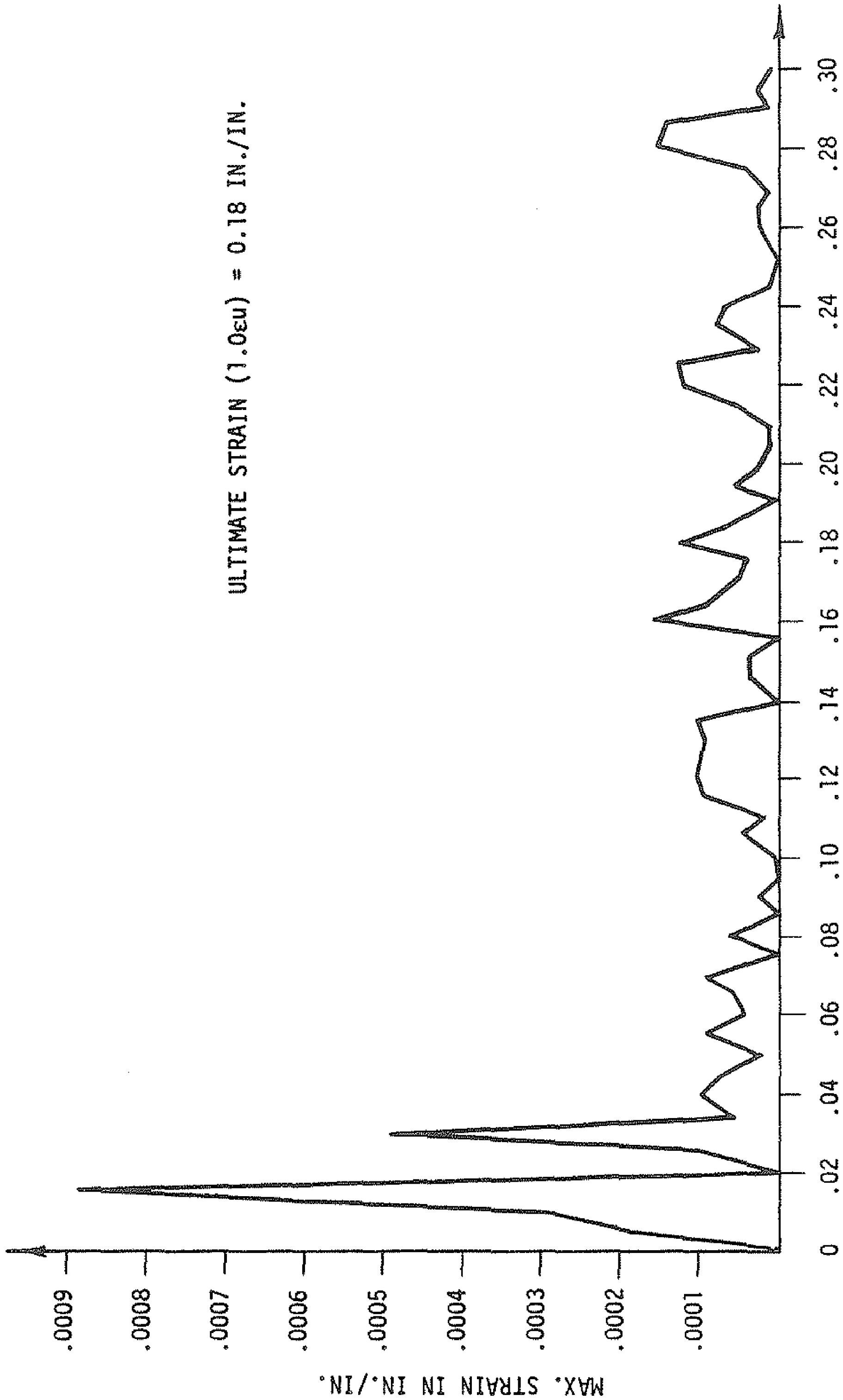
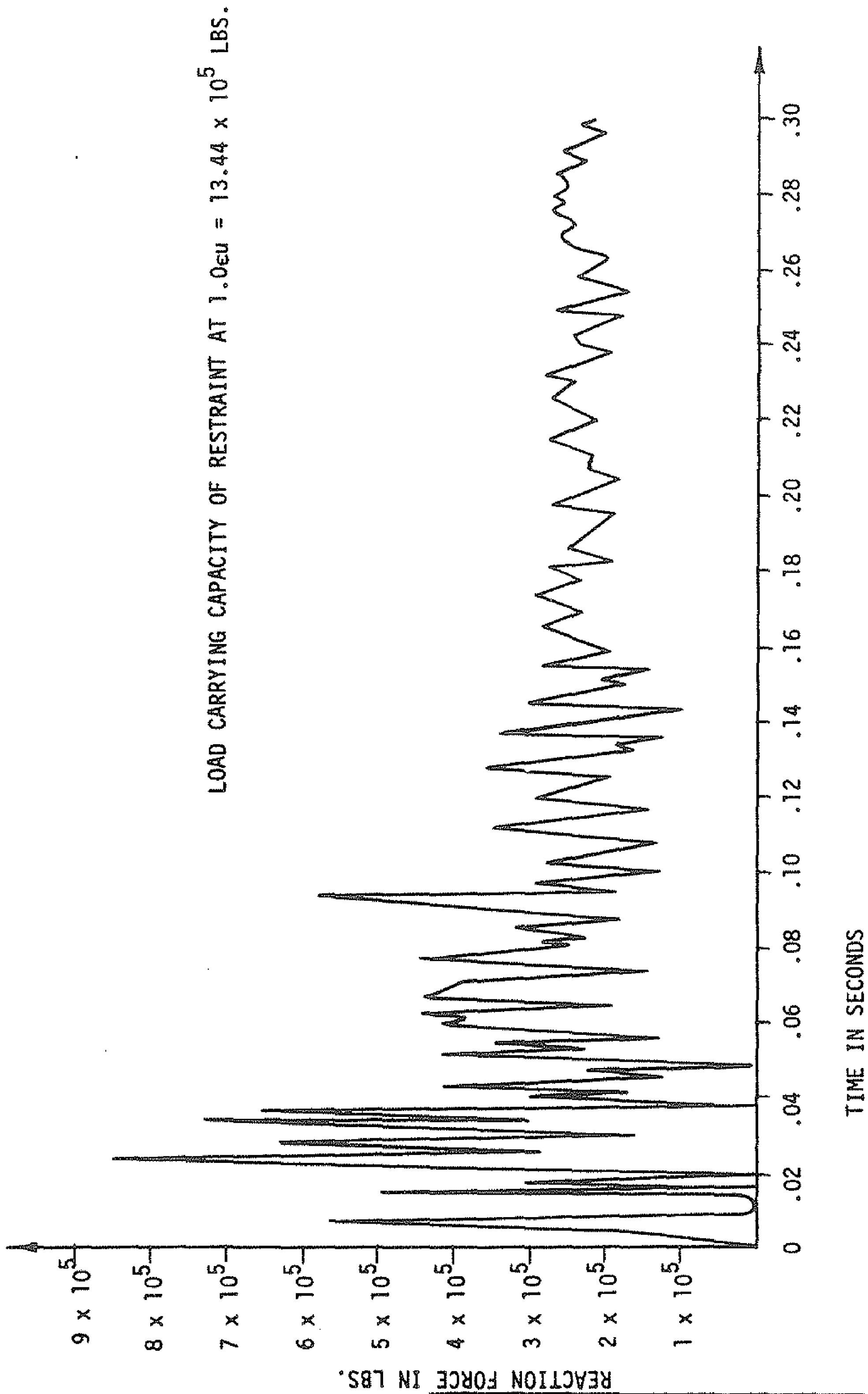
Main Steam Line Longitudinal Break M3 in Radial Direction, Envelope of Maximum Deflection

Figure 3.6-100



DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT & POWER COMPANY UPDATED FINAL SAFETY ANALYSIS REPORT
Main Steam Line Longitudinal Break M4 in Tangential Direction, Maximum Strain Time History in Pipe Figure 3.6-101



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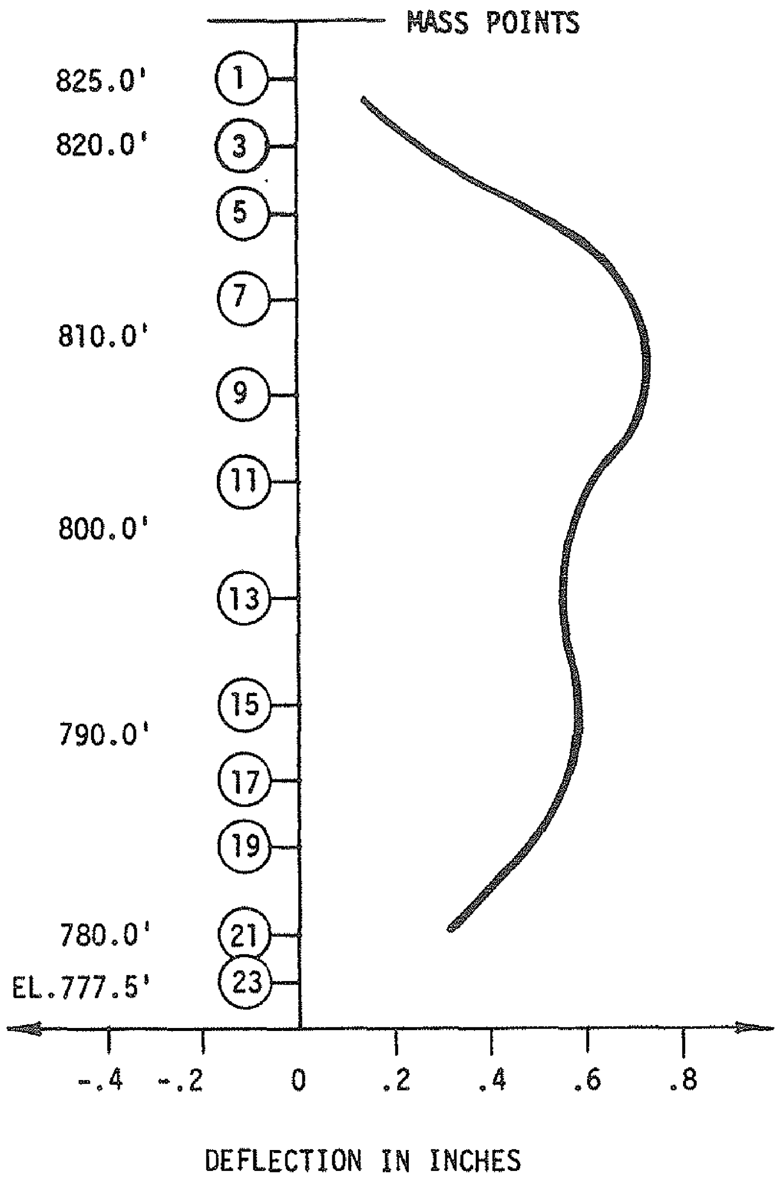
Main Steam Line Longitudinal Break M4
in Tangential Direction, Maximum
Strain Time History in Restraint No. 27

Figure 3.6-102



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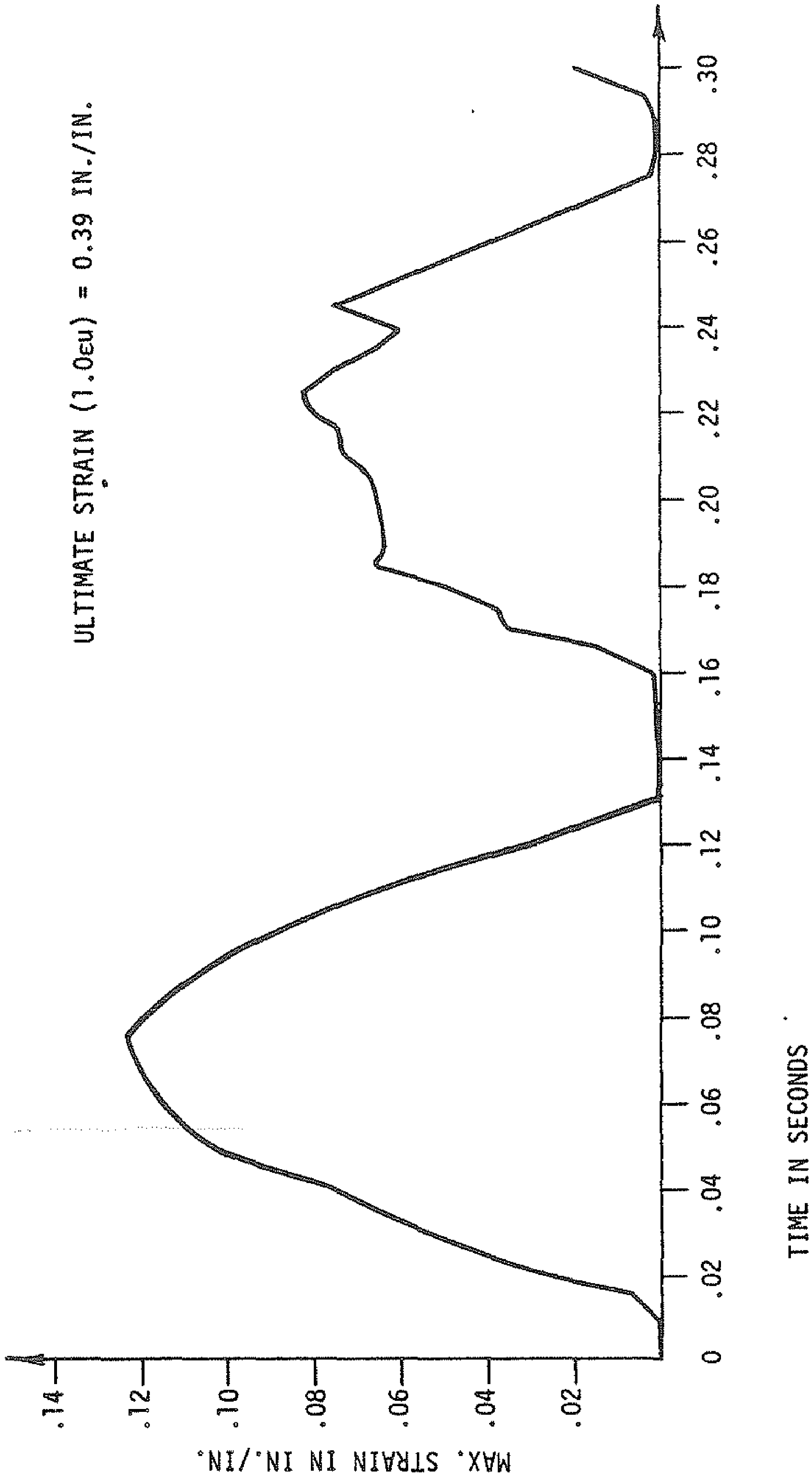
Main Steam Line Longitudinal Break M4 in Tangential Direction, Envelope of Maximum Deflection

Figure 3.6-103



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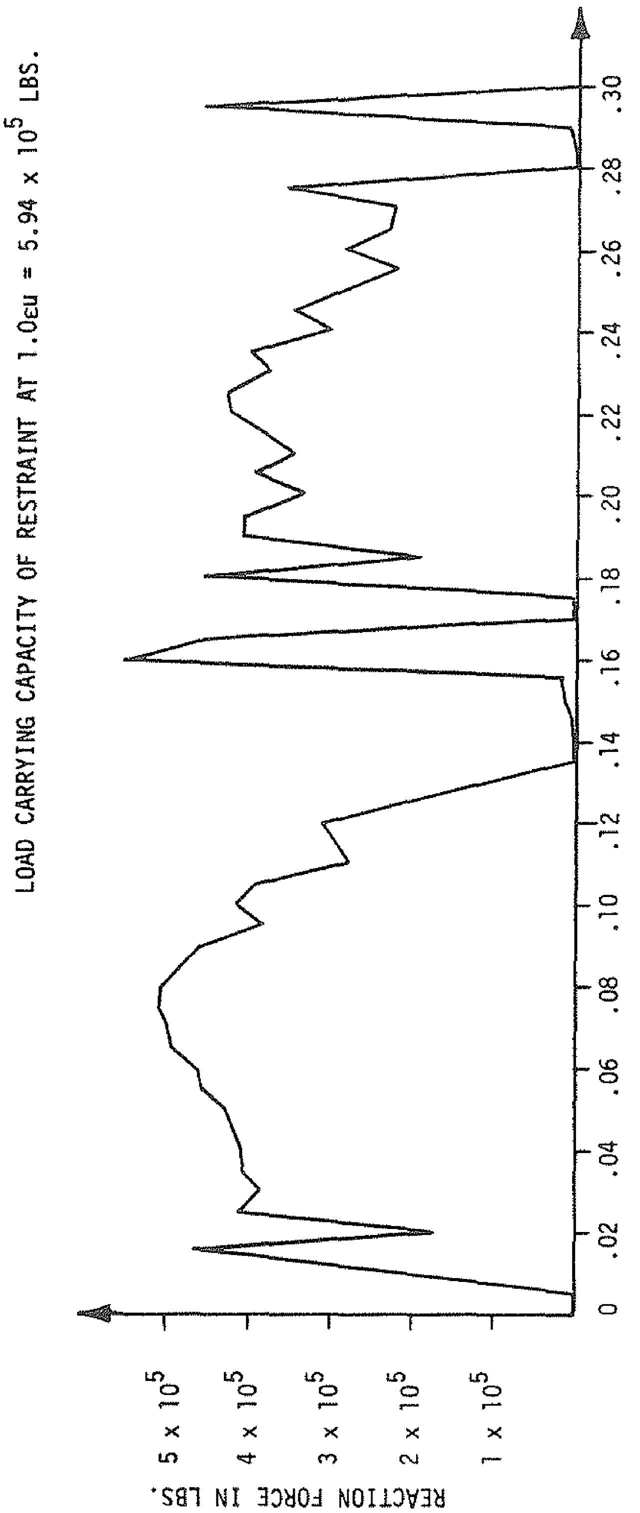
Recirculation Suction Line Circumferential Break R1 at Reactor Vessel Nozzle, Maximum Strain Time History in Pipe

Figure 3.6-104



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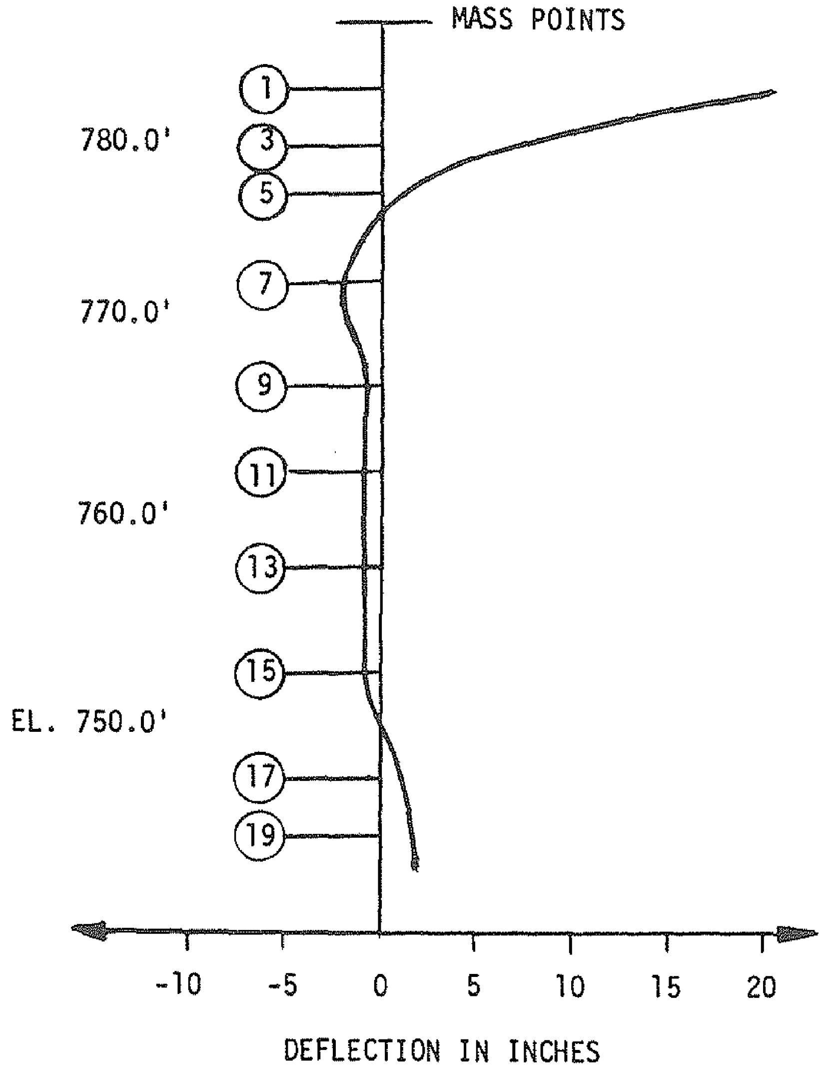
Recirculation Suction Line Circumferential Break R1 at Reactor Vessel Nozzle, Maximum Reaction Force Time History in Restraint No. 21

Figure 3.6-105



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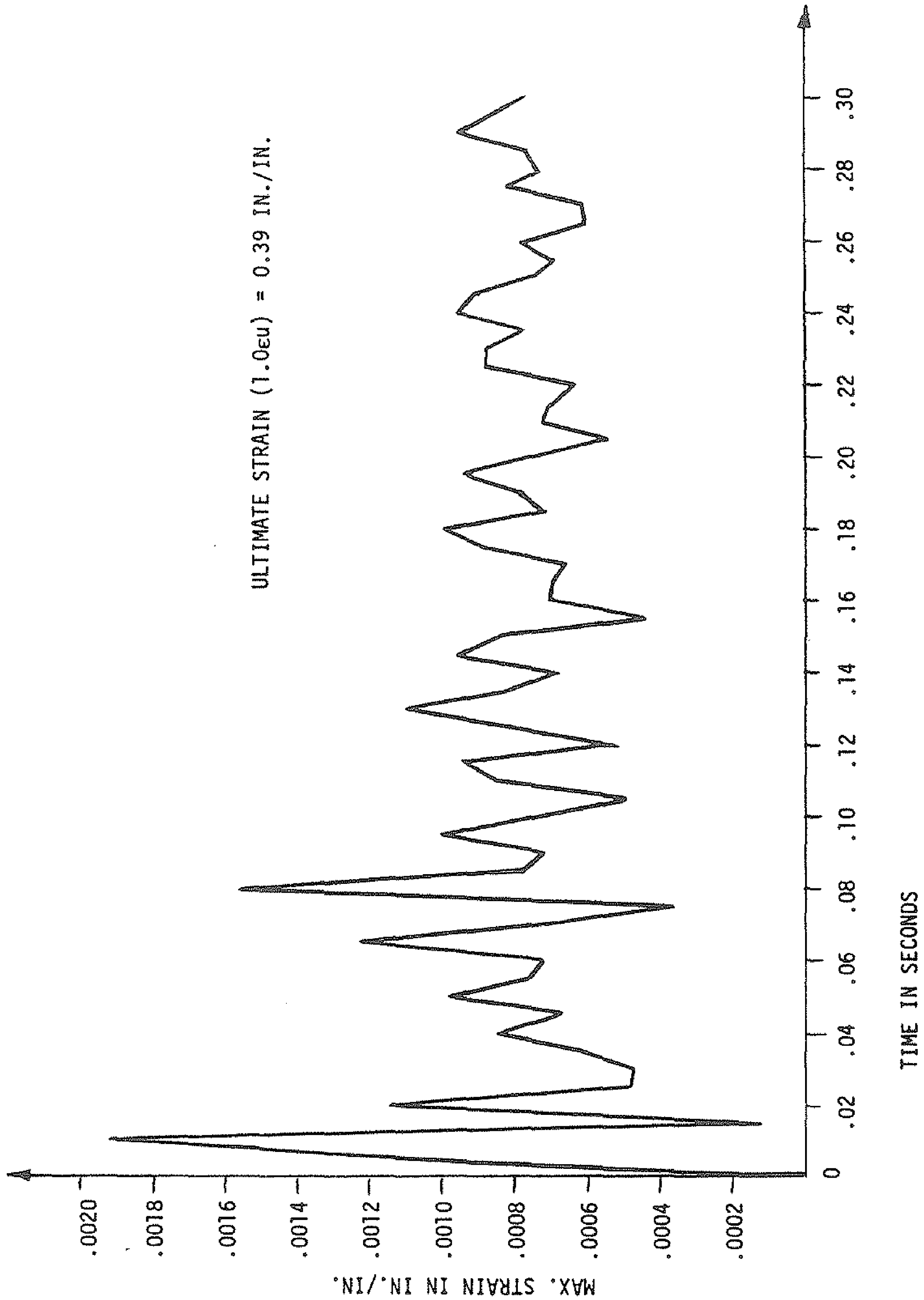
Recirculation Suction Line Circumferential Break R1 at Reactor Vessel Nozzle, Envelope of Maximum Deflection

Figure 3.6-106



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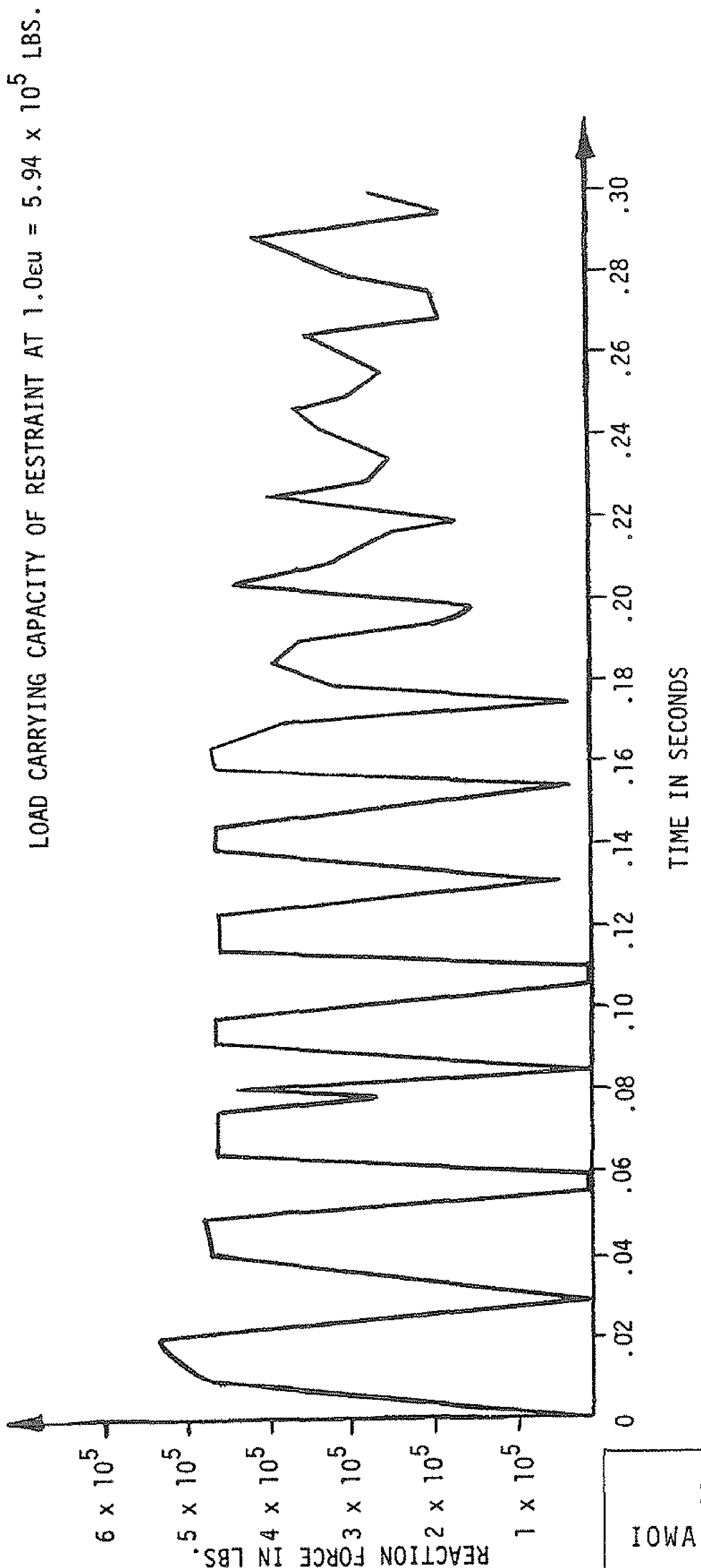
Recirculation Suction Line Longitudinal
Break R2 in Radial Direction,
Maximum Strain Time History in Pipe
Figure 3.6-107



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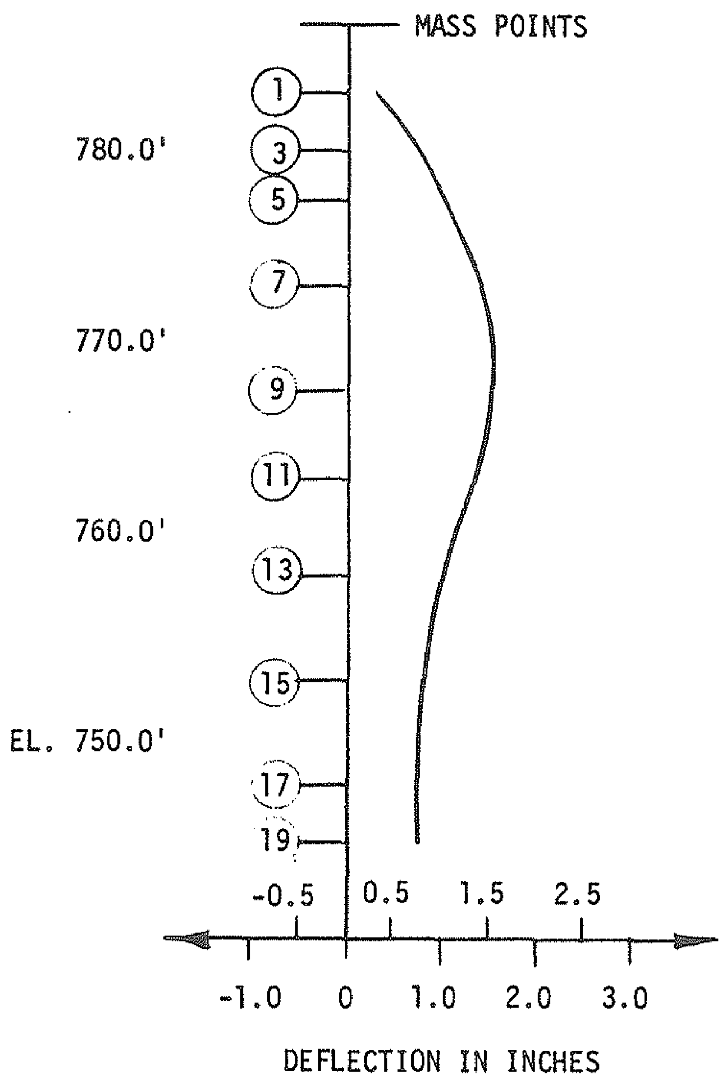
Recirculation Suction Line Longitudinal Break R2 in Radial Direction, Maximum Reaction Force Time History in Restraint No. 22

Figure 3.6-108



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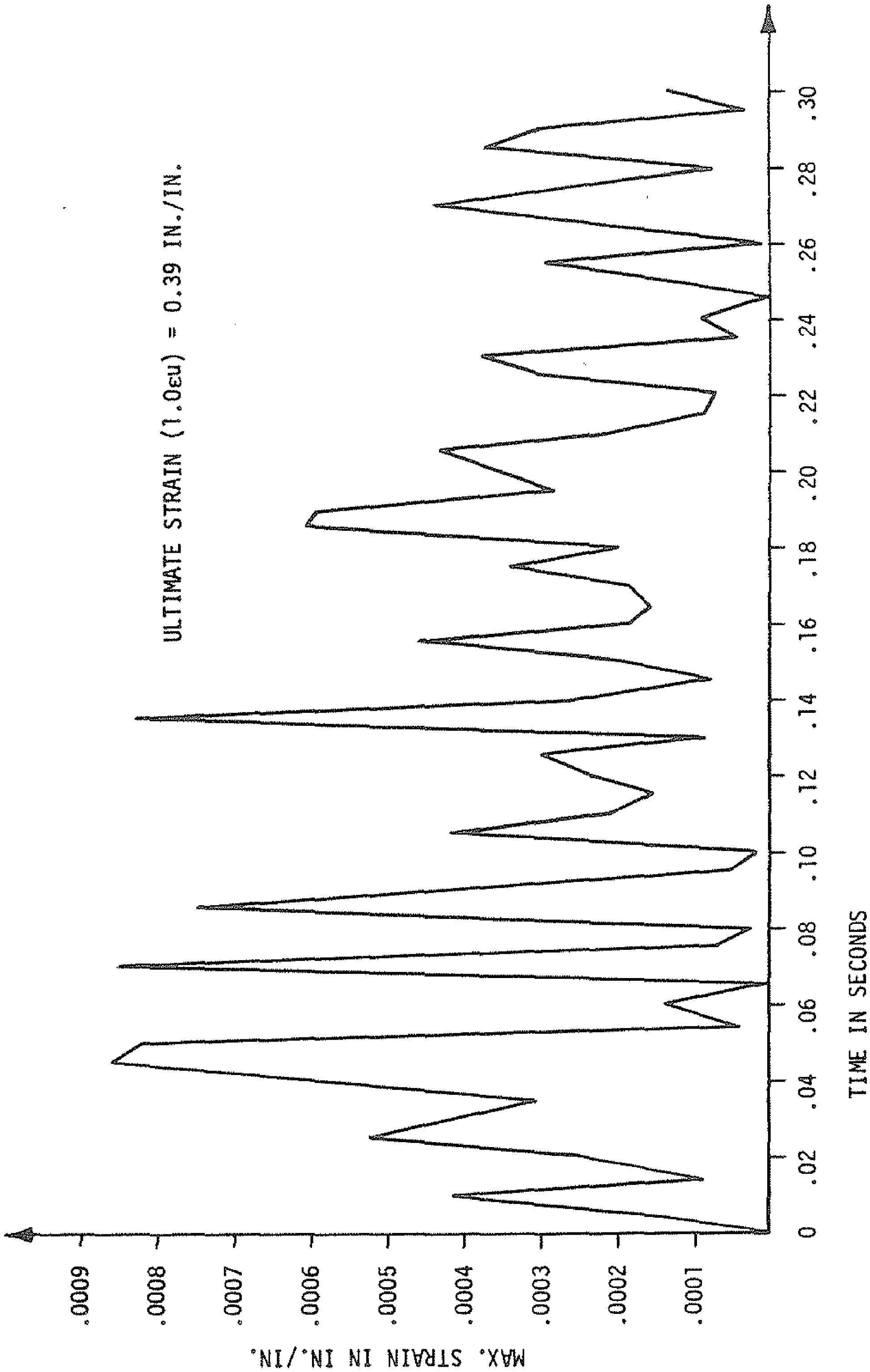
Recirculation Suction Line
Longitudinal Break R2 in Radial
Direction, Envelope of Maximum Deflection

Figure 3.6-109



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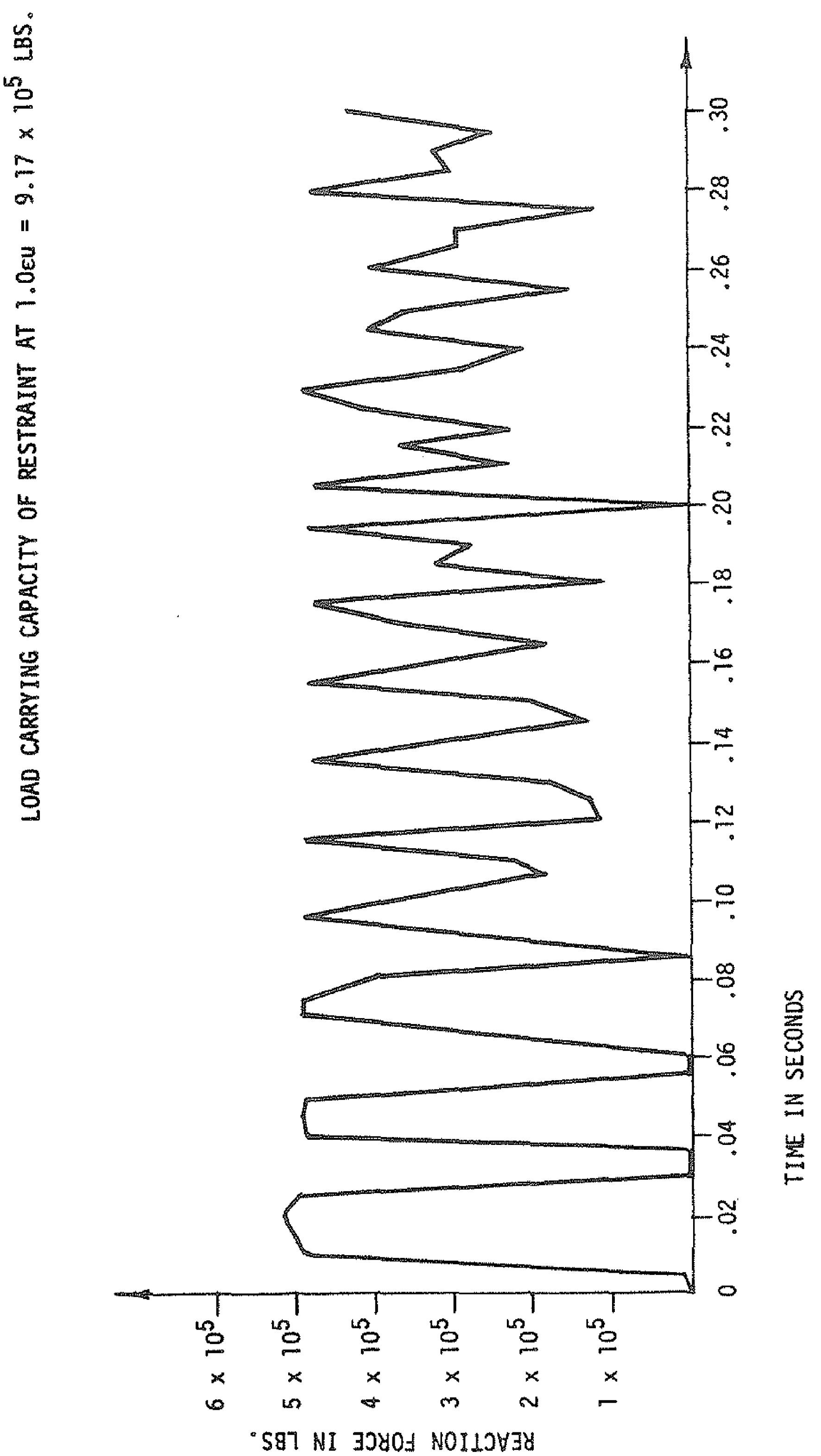
Recirculation Suction Line Longitudinal Break R3 in Tangential Direction, Maximum Strain Time History in Pipe

Figure 3.6-110



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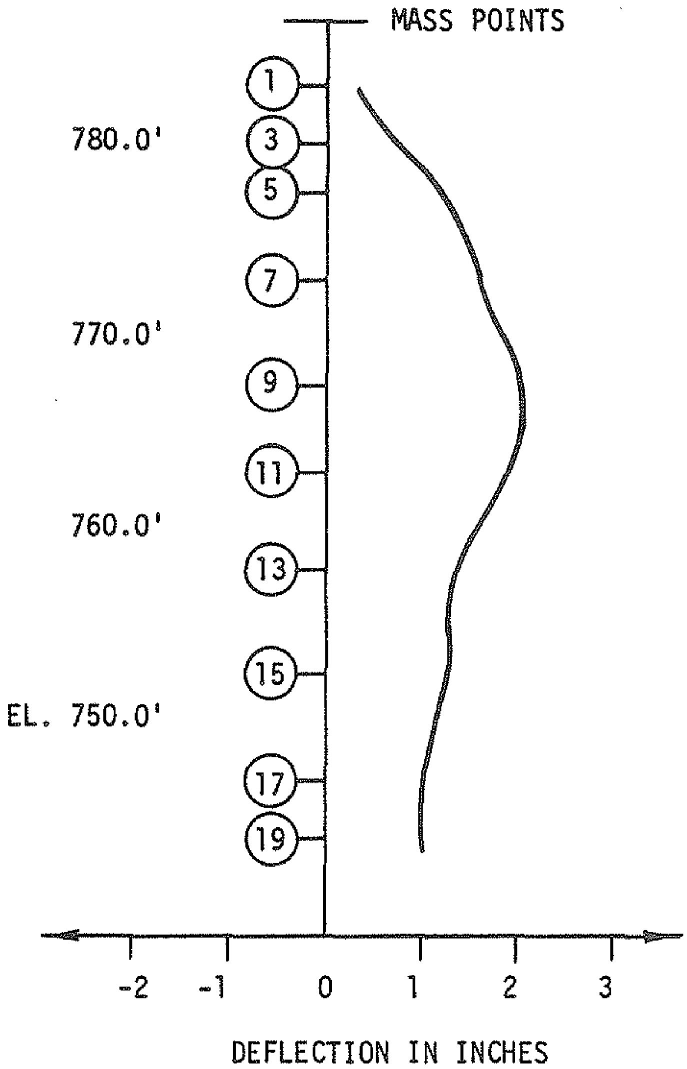
Recirculation Suction Line Longitudinal Break R3 in Tangential Direction, Maximum Reaction Force Time History in Restraint No. 22

Figure 3.6-111



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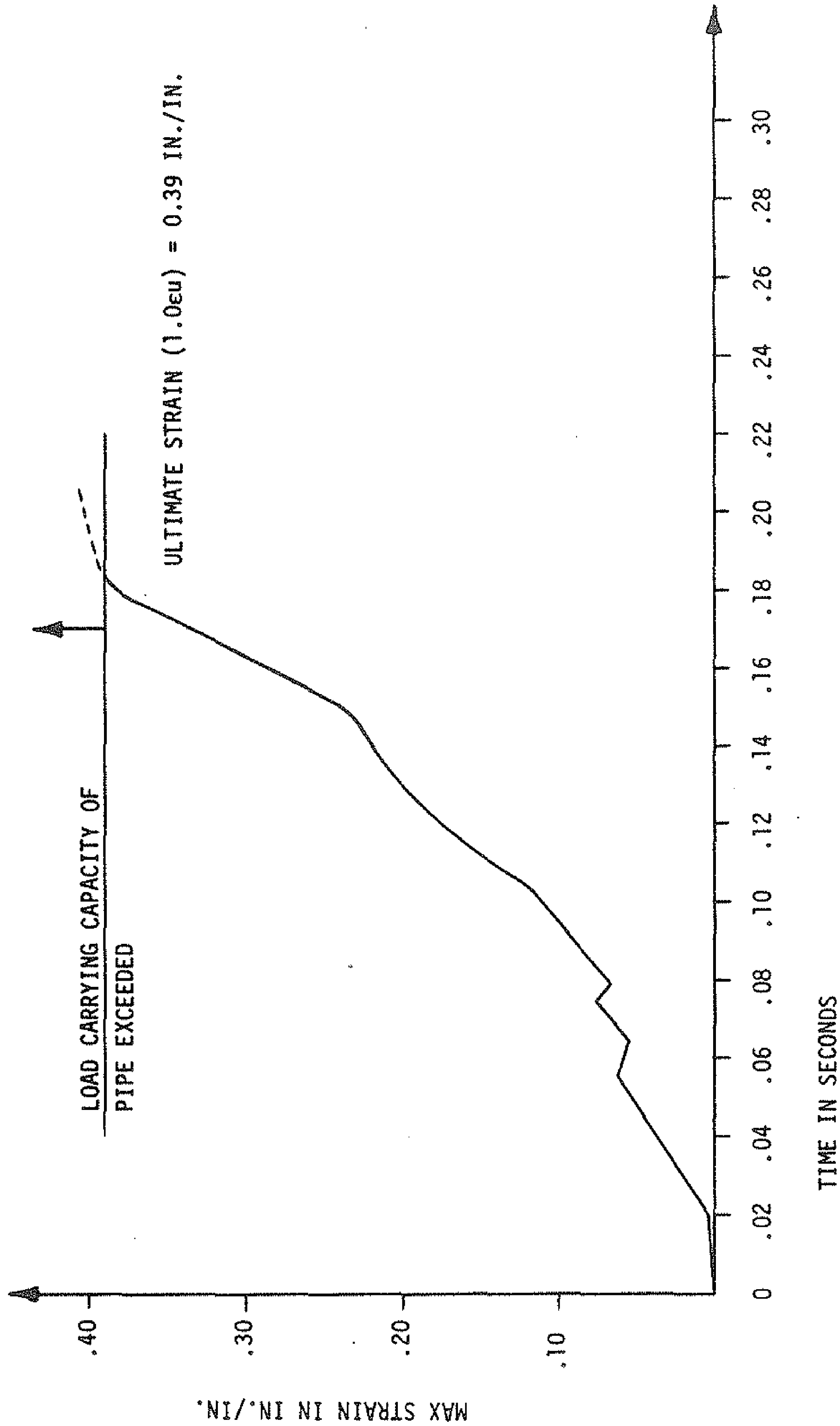
Recirculation Suction Line Longitudinal Break R3 in Tangential Direction, Envelope of Maximum Deflection

Figure 3.6-112



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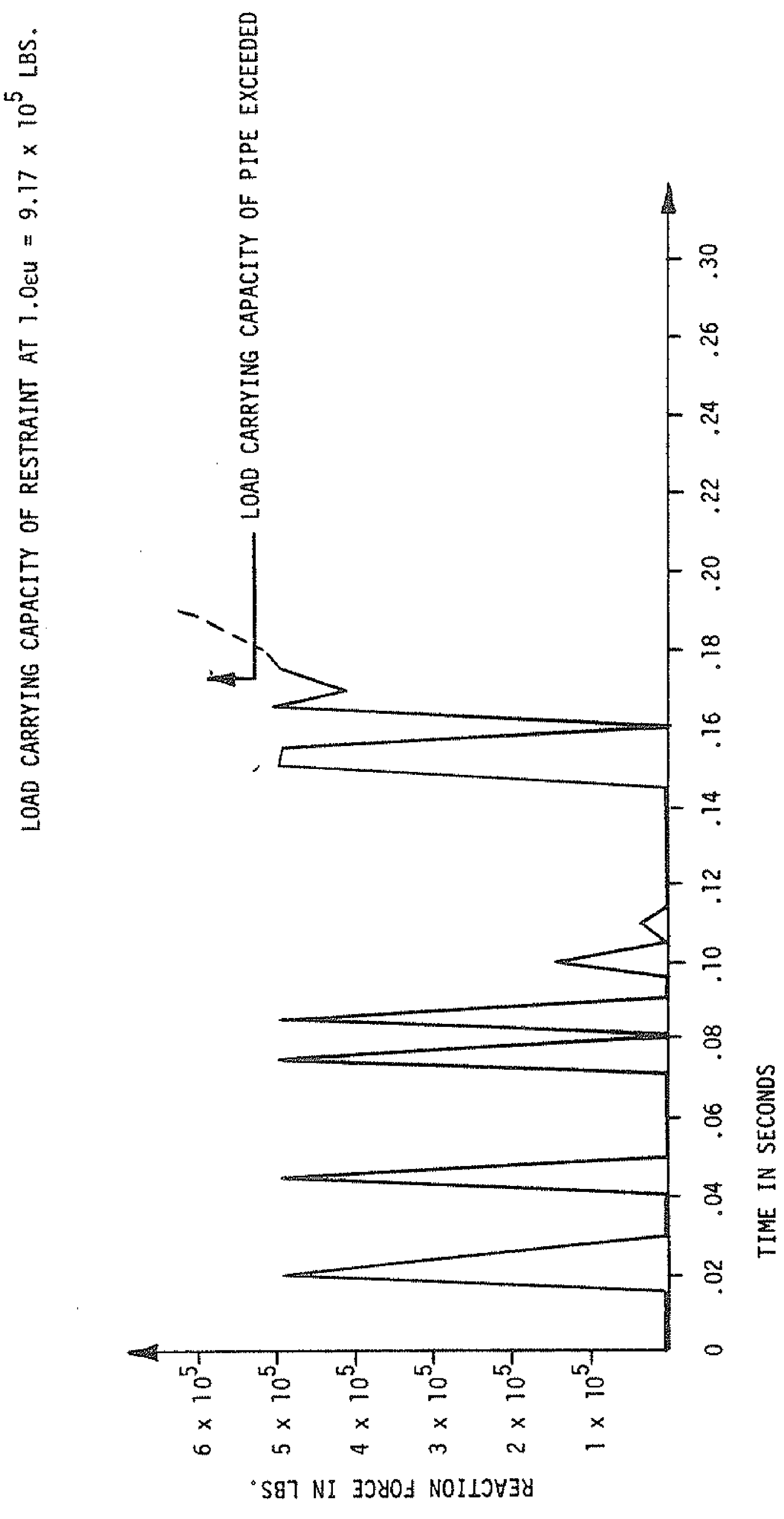
Recirculation Discharge Line
Circumferential Break R2 at Discharge
Elbow, Maximum Strain Time History in Pipe

Figure 3.6-113



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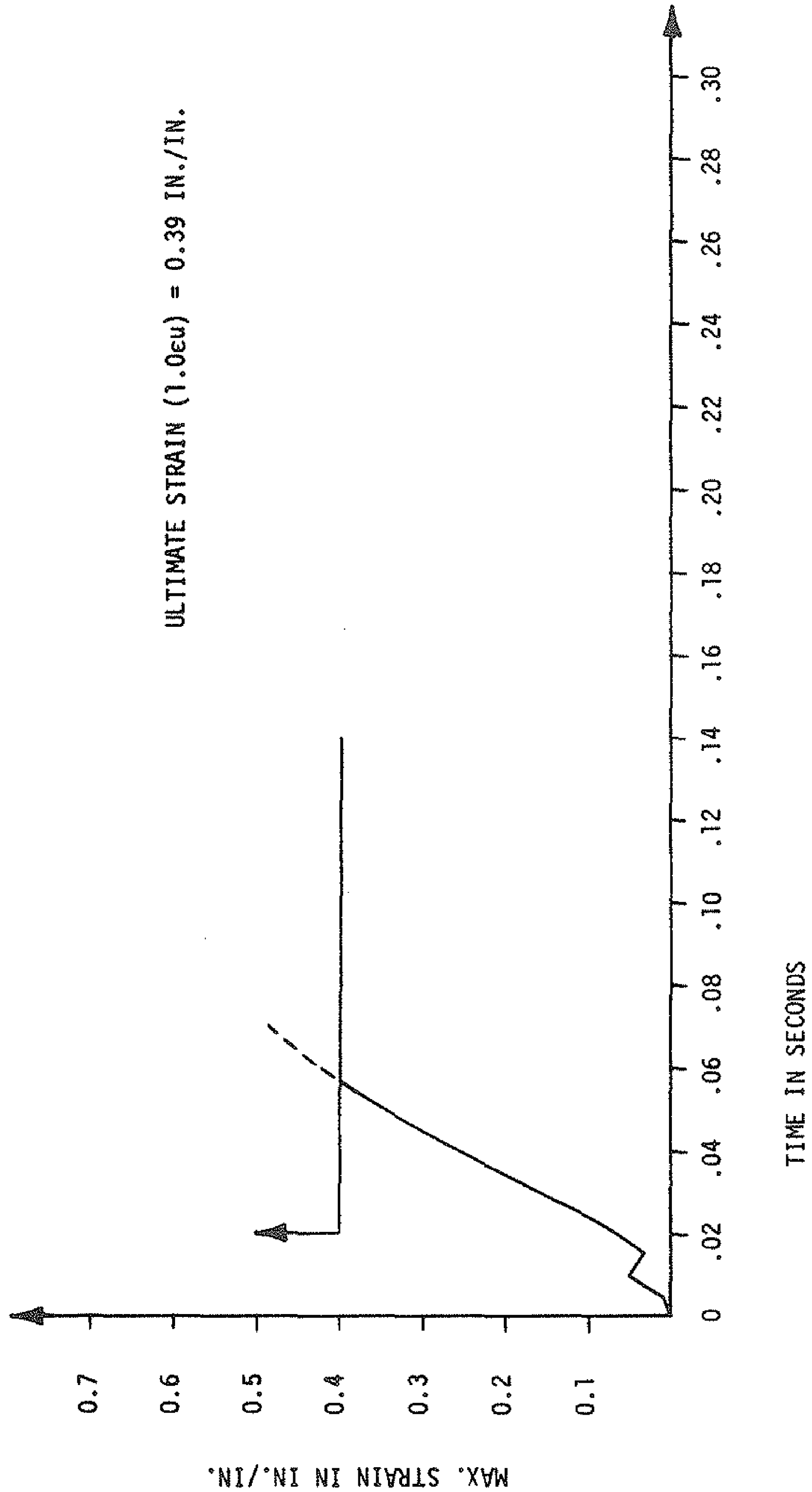
Recirculation Discharge Line Circumferential Break R4 at Discharge Elbow, Maximum Reaction Force Time History in Restraint No. 15

Figure 3.6-114



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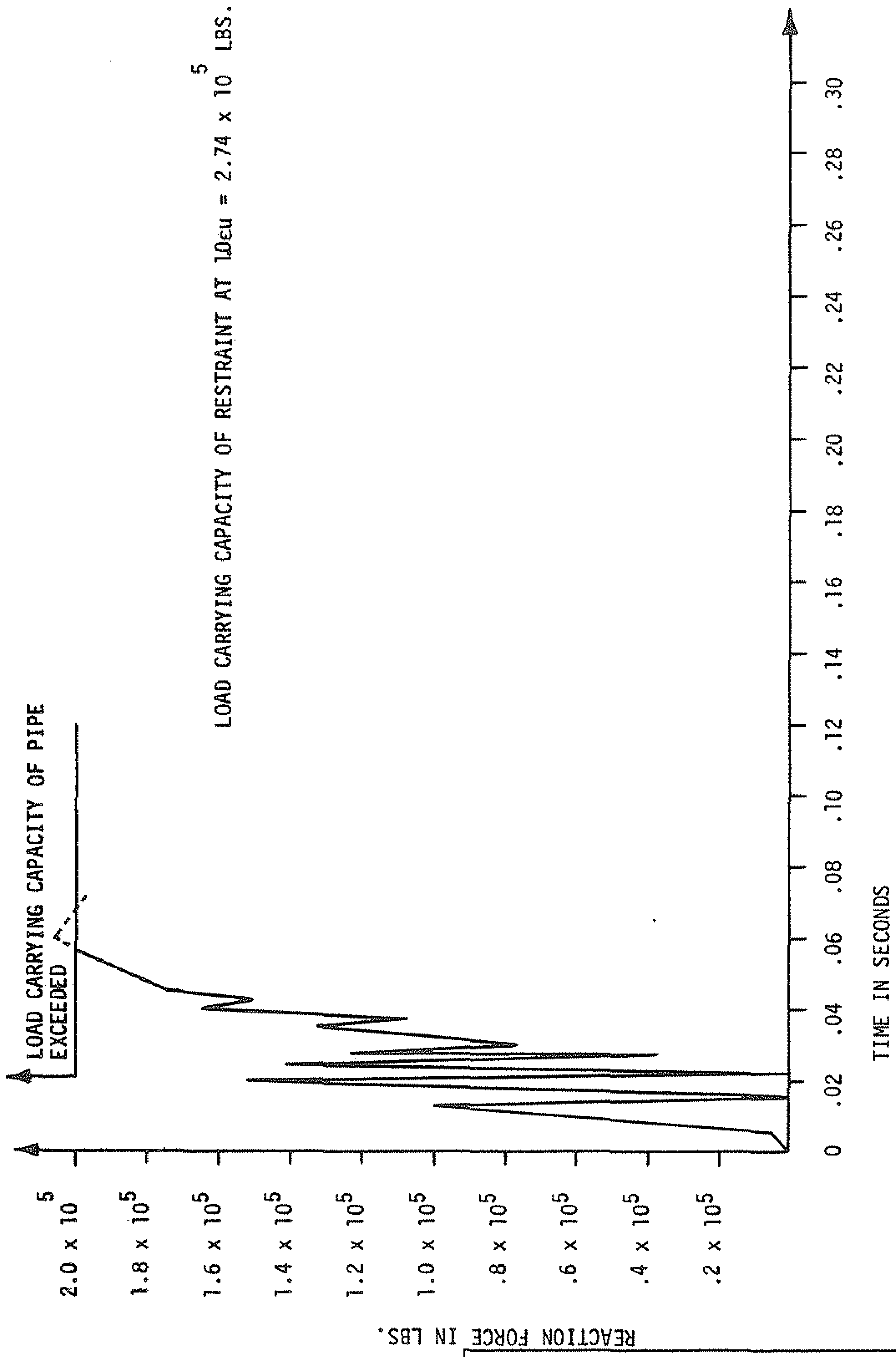
Recirculation Riser Circumferential
Break R5 at Reactor Vessel Nozzle,
Maximum Strain Time History in Pipe
Figure 3.6-115



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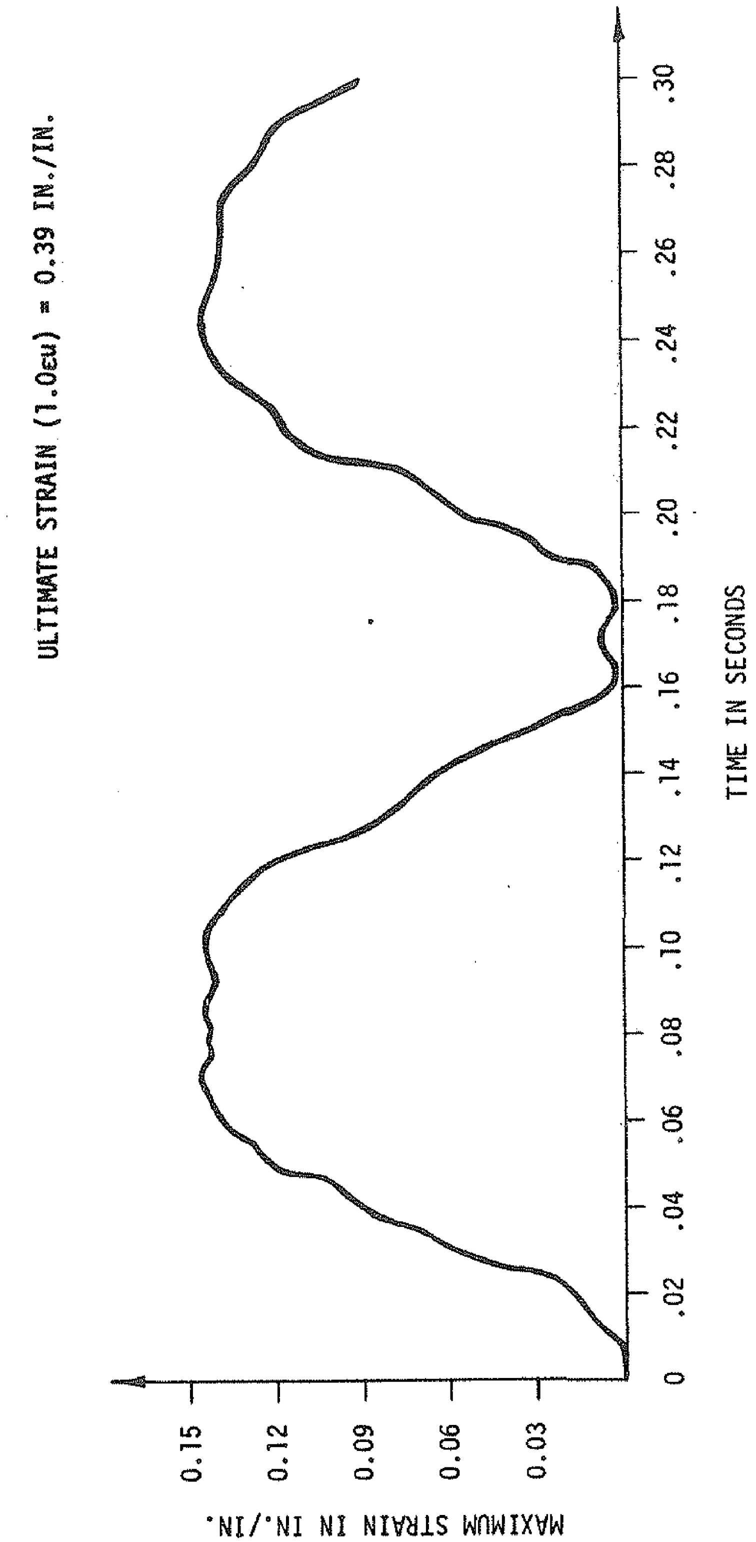
Recirculation Riser Circumferential Break R5 at Reactor Vessel Nozzle, Maximum Reaction Force Time History in Restraint No. 17 Figure 3.6-116



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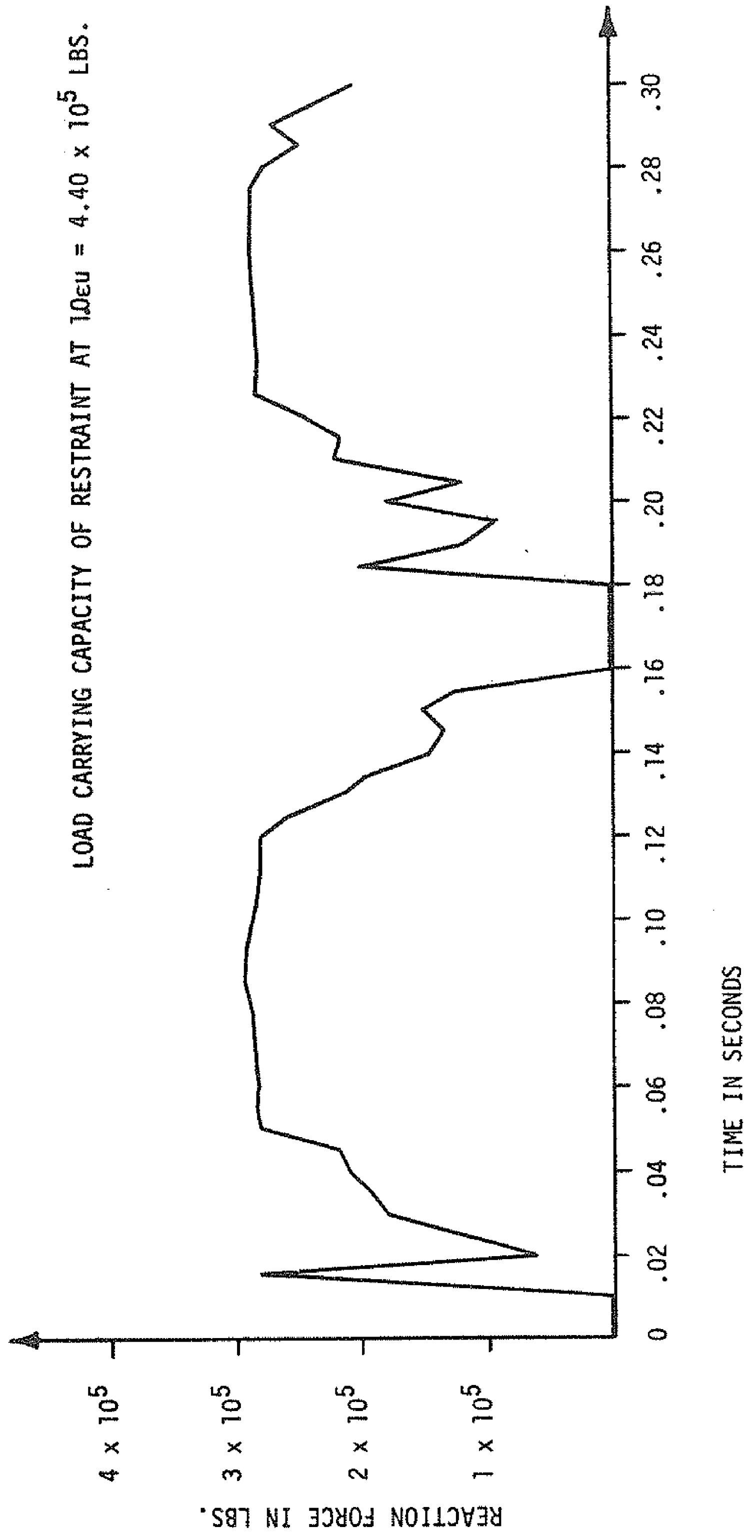
Recirculation Header Circumferential Break R6 at Riser/Header Tee, Maximum Strain Time History in Pipe Figure 3.6-117



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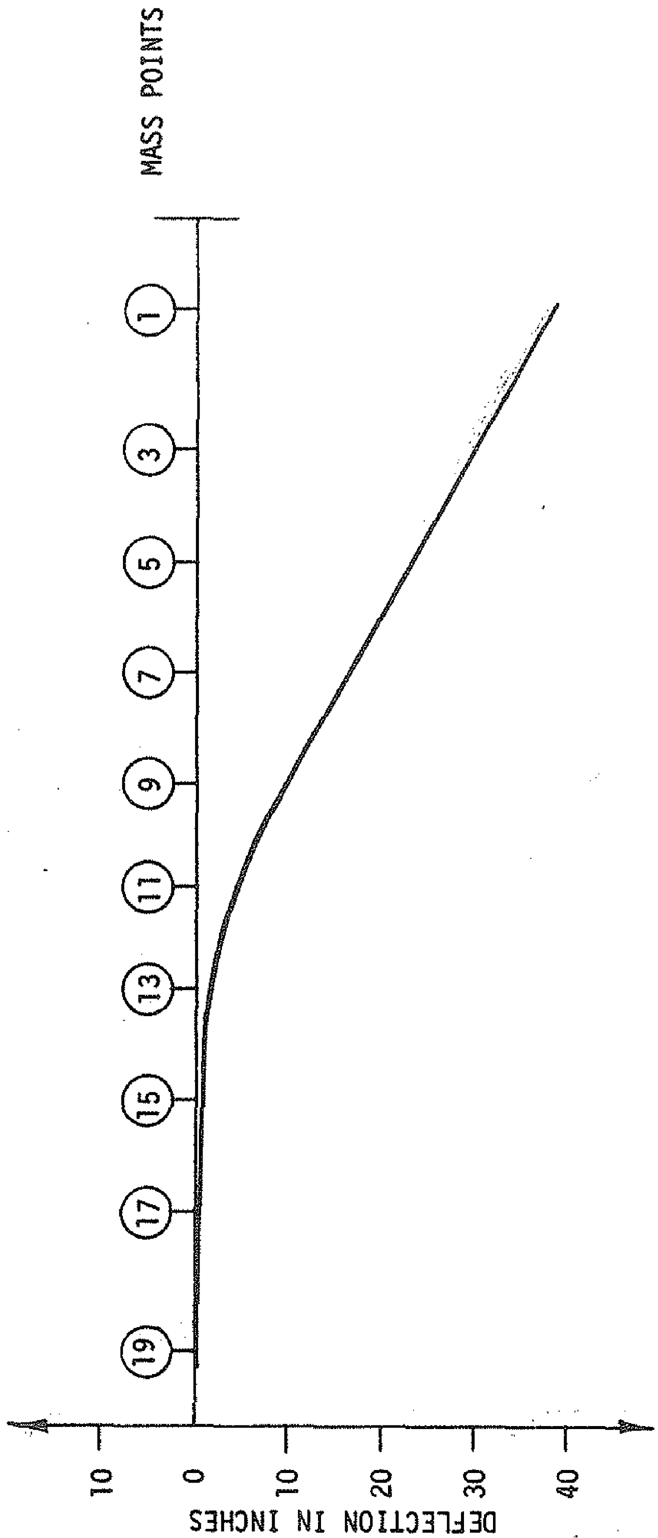
Recirculation Header Circumferential Break R6 at Riser/Header Tee, Maximum Reaction Force Time History in Restraint No. 21

Figure 3.6-118



DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Recirculation Header Circumferential
Break R6 at Riser/Header
Tee, Envelope of Maximum Deflection

Figure 3.6-119

3.7 SEISMIC DESIGN

General Electric (GE) and Bechtel Corporation were responsible for the seismic design of all structures, systems and components of the plant that were related to safety.

The GE Atomic Power Equipment Department (APED) organizations who had responsibility for the seismic design of safety-related systems and structures in the NSSS were Requisition Engineering and Design Engineering. The seismic design responsibility was assigned to the functional groups for component designs or systems designs who are responsible for the equipment and/or structure design. Those functional groups were responsible to the Manager, Design Engineering.

The Bechtel DAEC project design organization consisting of the Mechanical Group, Layout Group, Civil Group, and the Electrical Group, in parallel with the Bechtel Power and Industrial Piping Stress Group, had the responsibility for the seismic design of all balance-of-plant (BOP) structures, systems, and components related to safety. Dames and Moore performed the site seismology studies. These studies were reviewed and checked by Bechtel's Soils and Geology Department. Chicago Bridge and Iron performed the primary containment stress analyses. John A. Blume and Associates, Engineers, performed the dynamic analysis of all Seismic Category I structures and other structures housing Seismic Category I equipment.

Design organizations of GE APED were responsible for proper application of seismic design loads and conditions to the design of equipment components and piping in the NSSS scope. Analytical assistance was available within GE Design Engineering from analytical components specialized in seismic design. An Engineering Practices and Procedures Manual defined explicitly in writing all Design Engineering and Application Engineering responsibilities, including seismic. The Manager, Design Engineering, had overall responsibility for the adequacy of the seismic design of the GE product.

The dynamic analysis of plant structures was performed by John A. Blume Engineering after the location of major component masses and structural concepts was determined by the involved Bechtel project groups. The resulting floor response spectrum curves were submitted to Bechtel for review and approval. On approval the seismic report was transmitted to the GE APED Project Organization and other project groups within Bechtel through the Civil Group, which coordinated and had overall responsibility for the BOP seismic design. The Civil group also had responsibility for plant structural design. The Mechanical and Electrical Groups had responsibility for obtaining vendor seismic design analyses or test results of safety-related equipment and instrumentation; the Layout Group provided input on plant piping layout to the Piping Stress Group, which performed piping stress calculations for Seismic Category I piping systems.

The mechanism for the interchange of needed design information and changes thereto and the coordination of the various facets of the seismic design among the involved design organization components and/or groups is shown in Figure 3.7-1.

The system shown in Figure 3.7-1 is a pattern of interrelationships and checks from which an interactive process evolved that ensured proper plant seismic design for structures, systems, and components related to safety.

Within GE APED Design Engineering, the design engineer was ultimately responsible for the implementation of the seismic design requirements. Within the Bechtel DAEC project organization, the engineer responsible for the safety-related equipment, supported by engineers qualified in seismic analysis within the Civil Group and occasionally by qualified engineers of John A. Blume Engineering, was responsible for the implementation of the seismic design requirements.

For GE APED components, the adequacy of seismic design was the responsibility of the individual design engineer. Within Bechtel, the seismic certification of safety-related equipment was the responsibility of the design group procuring the equipment. Within each group, one or more engineers coordinated the transfer of vendor seismic certification (analyses, tests, or documentation of suitable performance in comparable vibrational environments) to the Civil Group for engineering review and approval.

Seismic design requirements were incorporated in the purchase specifications of all safety-related equipment.

For Bechtel-supplied Seismic Category I equipment, the supplier was given three options. First, the supplier determined the fundamental natural frequency: if it was found to be greater than 33 Hz, he could then use the floor horizontal acceleration values; if found to be less than 33 Hz, he was provided with the appropriate acceleration value from the response curve for 1% critical damping. Second, if the supplier chose not to determine the natural frequency, he was provided and instructed to use the peak value of the 1% critical damping response curve for molded supports, and 2% critical damping response curves for bolted supports, for the appropriate floor level. In all cases, the vertical seismic component was two-thirds of the foundation level horizontal acceleration for the operating-basis earthquake (OBE) or design-basis earthquake (DBE), applied simultaneously with the horizontal component. Third, the supplier was allowed to carry out a dynamic test.

The vendor was required by the purchase specification to submit the test data and/or the seismic analysis for the buyer’s approval as a condition of acceptance of the equipment for the intended function. The vendor could use test data on the particular components or equipment, applicable data from previously tested comparable equipment, performance data from comparable equipment which, during normal operating conditions, had been subjected to equal or greater shock loadings, and/or suitable analytical results.

This data was then referred, for NSSS equipment, to the responsible design engineer within GE APED or, for BOP safety-related equipment, to the Bechtel Civil Group, which reviewed the methods, procedures, and results for compliance with the criteria. If rejected, the vendor was requested to resubmit an amended report and/or perform structural modifications to the equipment. The whole sequence of submittal and review was repeated, if necessary, to achieve conformance with the design criteria.

3.7.1 SEISMIC INPUT

The foundation level accelerations for the OBE and the DBE are described in Section 2.5.2.

3.7.1.1 Design Response Spectra

In addition to the identification of the maximum foundation level acceleration that might occur at the site, the distribution of frequencies of the foundation level motion was considered. To represent the frequency distribution of foundation level motion resulting from postulated earthquakes, the concept of the response spectrum was used.

A response spectrum is a plot of the maximum value of some parameter, such as acceleration, experienced by a single-degree-of-freedom damped spring mass system subjected to an input motion. The maximum value is expressed as a function of the natural period and damping of the system. The foundation level response spectra for the OBE and DBE are shown in Figure 2.5-8.

3.7.1.2 Design Time History

One earthquake time history was developed for use as the input motion in performing the seismic analysis of structures. The time history was developed so that response spectra generated from it would yield no significant deviations below the OBE response spectra for the range of damping considered in the analyses. Comparative plots of the acceleration response spectra generated from the modified earthquake time history and the OBE response spectra are shown for damping ratios of 0.005, 0.010, 0.020, and 0.050 in Figures 3.7-2 through 3.7-5. These comparative plots were presented in the PSAR and accepted by the NRC.

This time history represents the site specific (bedrock) earthquake motion and was used as the input for the seismic analysis of structures. The mathematical model for each seismically analyzed structure included the soil-structure interaction effects as described in Section 3.7.2.4. Therefore, the use of one input motion for the seismic analysis of structures is acceptable.

The DBE spectrum is a multiple of the OBE spectrum. Therefore, the DBE values are obtained by multiplying the OBE values by the same multiple, thus ensuring the same match with the site spectrum for the DBE as was obtained for the OBE.

3.7.1.3 Damping Values

For structures applying to BOP systems comprised of elements possessing different damping characteristics, a composite damping value was selected for each natural mode of vibration. The composite damping values were determined by first plotting and inspecting the mode shapes. Then weighted damping values were determined based on the estimated degree of participation of each element in each mode. These weighted, or composite, damping values were used in the dynamic analysis of the system by the modal superposition method.

For NSSS systems, to account for damping in different elements of a coupled system, a modal damping value for each mode of the coupled system was obtained by using the kinetic energy of the different elements as a weighting function.

Structures were analyzed using modal superposition techniques. Element or material-associated damping values are given in Table 3.7-1. "Composite" or modal damping values in structural systems comprised of different element material types were selected on the basis of an inspection of the significant mode shapes. If a particular mode indicated the response to be of a single-element type, the damping value corresponding to that element type was assigned to that mode. If a particular mode indicated the response to be of several element types, the damping value for that mode was estimated on the basis of the degree of participation of the different elements. In all cases, the damping values were selected conservatively.

3.7.1.4 Supporting Media for Seismic Category I Structures

The reactor building mat was designed to span a 16-ft hypothetical cavity at a depth of 2 ft below the rock surface in accordance with Section 1.8 of the PSAR, which was submitted with PSAR Amendment 10 on November 26, 1969.

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2.1 Seismic Analysis Methods

The design of the Seismic Category I structures, the reactor pressure vessel, and the drywell is based on a dynamic analysis using the time-history method. The design of other Seismic Category I equipment supported within the structures is typically based on a dynamic analysis using the response spectrum method.

The structures were analyzed for the foundation level accelerations described in Section 2.5.2.

In the dynamic analyses of Seismic Category I structures and components, equivalent discrete-mass mathematical models were developed to represent the physical characteristics of the structural systems. The models included rocking and translation of the structures on their foundation materials. In the analyses, all modes that contributed significantly to the total response were used. The earthquake conditions were applied to the structures in the direction of each of their principal axes. Stresses resulting from horizontal and vertical accelerations were considered to be acting simultaneously and added algebraically.

Structures, including the reactor pressure vessel and the drywell, were analyzed by the time-history method. These analyses considered the effects of the dynamic coupling between buildings and equipment. In the time-history method of dynamic analysis, the response of each mass for each individual mode at each increment of time is computed, and the total response for each increment of time is obtained by adding together the response of each mode at a particular instant of time. The maximum response is then the maximum absolute value of the time history of the total response.

Other Seismic Category I mechanical and electrical equipment, and piping supported in or on major structures, were generally analyzed by the response spectrum method of dynamic analysis. The response spectrum method uses the single-mass response spectra. In this method, the maximum response for each quantity for each mode is computed and the modal responses are combined to determine the total response.

3.7.2.1.1 Mathematical Procedures

The equation of motion of a multidegree-of-freedom discrete-mass damped system subjected to an arbitrary ground motion $\ddot{v}g(t)$, can be written as

$$m\ddot{v}(t) + c\dot{v}(t) + kv(t) = -m\rho_o \ddot{v}g(t) \qquad (3.7\text{-}1)$$

where

m	= mass matrix
c	= damping matrix
k	= stiffness matrix
ρ_o	= unit vector
$\ddot{v}g(t)$	= ground acceleration
$v(t), \dot{v}(t), \ddot{v}(t)$	= matrices of displacement, velocity and acceleration, respectively

Using the orthogonality relations and expressing the displacement, velocities and accelerations in normal coordinates (i.e., $v(t) = \varnothing Y(t)$, $\dot{v}(t) = \varnothing \dot{Y}(t)$, and $\ddot{v}(t) = \varnothing Y(t)$,), the above couple equations of motion (Equation 3.7-1) may then be rewritten as the following uncoupled, normal equation of motion:

$$M_r \ddot{Y}_r(t) + 2\lambda_r M_r \omega_r \dot{Y}_r(t) + K_r Y_r(t) = -\psi_r M_r \ddot{v}g(t) \qquad (3.7\text{-}2)$$

where

$M_r = \varnothing^t_r m \varnothing \gamma$ = generalized mass for the r^{th} mode

$\lambda_r = \dfrac{\varnothing_r^t c \varnothing r}{2\omega_r \varnothing_r^t m \varnothing_r}$ = damping ratio for the r[th] mode

$K_r = T_r K_r$ = generalized stiffness for the r[th] mode

$\Psi_r = \dfrac{\varnothing_r^t m \rho_o}{\varnothing_r^t m \varnothing_r}$ = participation factor for the r[th] mode

ω_r = undamped circular frequency of the r[th] mode

$\varnothing_r$ = mode shape matrix for the r[th] mode

$\varnothing_r^t$ = transposition of $\varnothing_r$

The undamped circular frequencies, ω, may be calculated from

$$\left| K - \omega_r^2 m \right| = 0 \tag{3.7-3}$$

and the mode shape matrix for the r[th] mode can be obtained from

$$\left[K - \omega_r^2 m \right] \varnothing_r = 0 \tag{3.7-4}$$

The solution of the differential Equation 3.7-2, for the case of at-rest initial conditions is

$$Y_r(t) = \frac{-\psi r}{\omega_r \sqrt{1-\lambda^2}} \int_0^t \ddot{v}_g(\tau) e^{-\lambda_r \omega_r (t-\tau)} \sin\left[\omega_r \sqrt{1-\lambda^2}\,(t-\tau) \right] d\tau \tag{3.7-5}$$

For small damping ratios, λ_r, the above solution may be approximated by

$$Y_r(t) = \frac{-\psi_r}{\omega_r} \int_o^t \ddot{v}_g(\tau) e^{-\lambda_r \omega_r (t-\tau)} \sin\left[\omega_r (t-\tau) \right] d\tau \tag{3.7-6}$$

There are basically two methods of dynamic analysis that can be used to solve multidegree-of-freedom problems: the time-history method and the response spectrum method.

3.7.2.1.1.1 Time-History Method of Analysis

If the ground motion acceleration time history, $\ddot{v}g(t)$, is known, Equation 3.7-6 can be solved by a numerical step-by-step integration procedure. $Y_r(t)$ is computed as a function of time for r=1,2,3 . . . n, where n is the total number of degrees of freedom of the system. The modal displacements, $v_r(t)$, at time t for the r^{th} mode, can then be calculated from

$$V_r(t) = \varnothing_r Y_r(t) \tag{3.7-7}$$

The total displacement, v(t), of the structure at any time t may be obtained by adding the individual modal displacements at time t

$$V(t) = V_1(t) + V_2(t) + \ldots\ldots V_n(t) \tag{3.7-8}$$

Once the time histories of the displacements have been determined, the time histories of shears and moments can be determined by conventional structural analysis procedures. The maximum values of the time histories of the shears and moments are determined and then used for design.

3.7.2.1.1.2 Response Spectrum Method of Analysis

If the ground motion time history is not available and the design earthquake is specified in terms of a response velocity spectrum, Equation 3.7-6 can be written

$$\left|Y_r(t)\right|_{\max} = \frac{\psi_r S_{vr}}{\omega_r} \tag{3.7-9}$$

where

S_{vr} = spectral velocity for the r^{th} mode.

$$S_{vr} = \int_0^t \ddot{v}_g(\tau) e^{-\lambda_r \omega_r (t-\tau)} \sin\left[\omega_r (t-\tau)\right] d\tau \quad \max \tag{3.7-10}$$

The maximum modal displacements, $V_{r\max}$, for the r^{th} mode are

$$v_{r\max} = \varnothing_r \frac{\psi_r S_{vr}}{\omega_r} \tag{3.7-11}$$

If the design earthquake is specified in terms of a response acceleration spectrum instead of a velocity spectrum, the maximum modal displacements, $V_{r\max}$, of the structure for the r^{th} mode are

$$v_{r\max} = \varnothing_r \frac{\psi_r S_{ar}}{\omega_r^2} \qquad (3.7\text{-}12)$$

where
S_{ar} = spectral acceleration for the r^{th} mode

With maximum modal displacements known, the other modal quantities such as shears and moments can be computed for each mode by conventional structural analysis procedures.

3.7.2.1.2 Example of Building Analysis

Figure 3.7-6 shows the model that was used in the analysis of the reactor building and the major equipment housed within it. The building was represented as a series of masses lumped at the floor levels and connected by weightless elastic elements representing the flexural characteristics of the building. The deformation of the building foundation material was represented by translational and rocking springs. The spring constants were calculated based on formulas published by Whitman and Richart.[1] The drywell, reactor pressure vessel, shield wall, and pedestal were represented by lumped-mass models, and elastic and rigid elements were used to represent the interconnections among the components of the resulting composite structure.

The natural mode shapes and periods of vibration were determined as described in Section 3.7.2.1.1. Damping values were assigned to each mode based on the relative deformations of the different materials within the structure as described in Section 3.7.1.3. For this example, the natural periods of vibration and assigned damping values for the significant modes of the reactor building model are shown in Table 3.7-2..

By applying the appropriate earthquake time history at the base of the model, time histories of the individual modal responses were determined for the structure. From these results, time histories of the structural accelerations, deflections, shears, and moments were determined. The maxima of these values were used in the design of the structure. For this example, the maximum accelerations, displacements, shears, and moments resulting from applying the OBE time history to the reactor building model are shown in Tables 3.7-3 through 3.7-7.

Floor response spectra for building floors and other support points within the structure were generated as described in Section 3.7.2.5. For this example, the response spectra for damping ratios of 0.005, 0.010, 0.020, and 0.050 derived from the motions of the reactor building floor at elevation 855 ft 0 in. (mass point 2) caused by the OBE are shown in Figure 3.7-7.

3.7.2.2 <u>Natural Frequencies and Response Loads</u>

Typical determination of the natural periods of vibration at various locations in the reactor building and the associated movements, forces, and moments is described in Section 3.7.2.1.2.

3.7.2.3 Procedure Used for Modeling

A typical model is described in Section 3.7.2.1.2.

3.7.2.4 Soil-Structure Interaction

The seismic analysis of structures must account for the site specific (bedrock) earthquake motion and the effects of the load transfer system. This can be accomplished with a model that is anchored at bedrock with the soil, structure and other mass/stiffnesses explicitly included. Soil-structure interaction effects are represented by the equivalent springs between the base of the model and the building foundation (see Figures 3.7-6, 3.7-8 and 3.7-9 for examples). This allows the bedrock motion to be the input for the analysis, with the amplification effects being directly calculated. The bedrock response spectra are assumed to be the same as the foundation level response spectra for structures supported on bedrock (i.e., no amplification).

As with any analysis, simplification of the model is allowed if appropriate conservatism is applied. For the seismic analysis of a building, a simple model is one that is anchored at the foundation level with the input motion defined for the foundation level to account for the effects of soil-structure interaction. This simple model eliminates the need to model the soil properties. Appropriate conservatism is included in the foundation level response spectra for structures supported on soil.

The foundation level response spectra are described in Section 2.5.2.

3.7.2.5 Development of Floor Response Spectra

Floor response spectra were derived for use in the analysis and design of Seismic Category I mechanical and electrical equipment and piping supported within the structures. The floor response spectra were obtained as follows: the building was subjected to the developed earthquake time history, and the corresponding output acceleration time histories at the floors or points of interest were determined. These acceleration time histories were then used to derive single-degree-of-freedom system response spectra, which are the floor response spectra, for each floor or point of interest.

The expected variations in the parameters used in the analyses were considered for both the design of the structures and the determination of floor response spectra. A minimum shift in building frequency of ±10% from the variation in structural properties was included. A possible variation in the soils moduli (for till and backfill) of ±50% was used. The use of these values resulted in a conservative design.

3.7.2.6 Three Components of Earthquake Motion

The methods used to combine the components of earthquake motion are described in Section 3.7.2.1.

3.7.2.7 Combination of Modal Responses

Since it is not known at which time the maximum modal responses computed in Section 3.7.2.1.1.2 occur, an approximate method had to be used to combine the modal responses to obtain the total response. One method of combination is to compute the sum of the absolute values of the response of each mode. However, the most commonly used method of combination is the root-mean-square method. In this approximation, the total response is obtained by computing the square root of the sum of the squares of the modal maxima. For example, the total displacements of the structure, v_{tot}, are computed from

$$v_{tot} = \left[\sum v_{r\ max}^2 \right]^{1/2} = \left[v_{1\ max}^2 + v_{2\ max}^2 .....+ v_{n\ max}^2 \right]^{1/2} \qquad (3.7\text{-}13)$$

The approximate values of the total shear and moment can be computed in a similar manner.

If several controlling frequencies in an eigenvalue solution are found to lie close together, the root-mean-square method of modal combination was used. In these cases, the modal maxima were combined by direct summation, or the system was analyzed by the time-history method. A technical justification for the use of the root-mean-square method is presented in Section 3.9.3.1.5.

3.7.2.8 Interaction of Nonseismic Structures with Seismic Category I Structures

Seismic design of Nonseismic structures follows prudent engineering practice, in accordance with the Uniform Building Code (UBC), 1970. Although the site is in Seismic Zone 0, criteria for Zone 1 were applied.

Allowable stresses for Nonseismic structures are as specified in Section 3.9.4.

Seismic Category I to Nonseismic structure interfaces are designed so that there will be no functional failure of Seismic Category I structures because of the possible failure of Nonseismic structures. This is accomplished by physically separating Nonseismic structures from Seismic Category I structures.

Those portions of Nonseismic structures housing Seismic Category I equipment were designed in accordance with Seismic Category I design criteria. In addition, the structures are investigated as a whole to ensure that failure in other areas would not endanger the areas housing Seismic Category I equipment.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The effect of uncertainties in building frequencies and in soil properties was considered as described in Section 3.7.2.5.

3.7.2.10 Use of Constant Vertical Static Factors

For those structures that were determined to be rigid in the vertical direction, vertical accelerations were applied as static coefficients. For those items that are not rigid in the vertical direction, dynamic analyses were performed using the time-history method or response spectrum method, as appropriate.

Structures and components with a fundamental natural frequency greater than 30 Hz were considered to be rigid and were designed to resist the maximum acceleration of their support applied as a static coefficient.

A structure's vertical response was determined by analyzing a mathematical model developed to represent the vertical physical characteristics of the soil-structure systems. Soil-structure interaction effects are represented by the equivalent springs shown at the base of each model (see Figures 3.7-6, 3.7-8 and 3.7-9 for examples). The masses in the model include the weights of the building structure and appropriate equipment loads. The stiffness of the springs between masses was based on the stiffness of the walls and columns between floors. The structures were determined to be rigid in the vertical direction, which resulted in no amplification of vertical accelerations.

3.7.2.11 Method Used to Account for Torsional Effects

For asymmetrical buildings, the translational and torsional seismic responses were determined by performing a dynamic analysis where the building was represented by a three-dimensional lumped-mass model.

As an example, Figure 3.7-9 shows the model that was used in the analysis of the control building. This model consists of the building masses lumped at the floor levels and interconnected by flexible shear elements representing the walls and by rigid diaphragms representing the floors and roof. The structure is nearly symmetrical about the north-south axis, but is asymmetrical about the east-west axis because of the braced frame on the south wall. The properties of the sand fill on which the building is founded were modeled by a series of springs that represent the resistance of the sand fill to forces imposed on it by the motion of the building. Horizontal, vertical, and rotational motions of the foundation were considered in determining the foundation stiffness. The spring constants used were computed from formulas derived from elastic half-space theory and published by Whitman and Richart.[1] The range of spring values used was based on a possible variation of soils moduli of ±50% as described in Section 3.7.2.5.

The earthquake motions postulated for the DAEC site, as presented in the PSAR and approved by the NRC, did not include spectra for torsional motions. Therefore, no torsional dynamic analyses of symmetrical buildings were performed. However, the symmetrical structures were designed for the effects of a torsional moment equal to the story shear acting with an eccentricity between the centers of mass and rigidity of 5% of the appropriate building dimension. The torsional effects had no significant influence on the designs.

3.7.2.12 Comparison of Responses

Time-history analyses were performed for all structures. Modal response spectrum analyses were not performed because the time history produced spectra that were conservative relative to the criteria response spectra.

3.7.2.13 Methods for the Seismic Analysis of Dams

This subject does not apply to the DAEC. See Section 2.4.4.

3.7.2.14 Analysis Procedure for Damping

Structures were analyzed using modal superposition techniques. Element or material-associated damping values are given in Table 3.7-1. “Composite” or modal damping values in structural systems comprised of different element material types were selected on the basis of an inspection of the significant mode shapes and on the assumption that the contribution of each material to the composite effective modal damping is proportional to the elastic energy induced in each material. The following criteria and procedures were applied on a mode-by-mode basis to evaluate and conservatively determine composite damping values:

1. Where a particular mode primarily indicated the response to be of a single element type, the damping value corresponding to that element type was assigned to that mode. Where all but a negligible amount of the elastic energy was induced in, for example, concrete or soil, the damping value appropriate to these materials was applied. Similarly, where a lightly damped material exhibited a major portion of the elastic energy to be of the mode, a conservative choice was made to use the damping value of that material for that mode. In most cases for the DAEC, the modes were well defined according to material types, and composite damping values could be selected by a visual inspection of mode shapes with no numerical computations required.

2. In a few instances, the above criteria could not be applied because a particular mode indicated the response to be of several element types. The damping value for that mode was conservatively estimated on the basis of the degree of participation of the different elements. The elastic energy induced in the different elements was estimated, and composite damping values were assigned in proportion to the elastic energy.

The approach described above is consistent with currently accepted techniques, and in all cases, the damping values were selected conservatively. The use of this approach has resulted in a design that can conservatively resist the seismic motions postulated.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

Seismic Category I equipment was examined to ensure its ability to withstand design loading requirements. Experienced designers examined the equipment using appropriate techniques to determine which specific portions of the systems and components required further examination. These techniques fit into two general categories: (1) normal analytical techniques, consisting of empirical design methods as defined by appropriate design codes; and (2) special techniques, used to supplement code calculations or to cover conditions not considered by existing codes.

1. Normal Design Techniques

   Seismic Category I equipment (e.g., piping, valve bodies, and pump cases) was designed in accordance with applicable industrial codes. Some codes used empirical design methods for equipment that could not be sized by conventional rational stress analysis methods and did not require a detailed stress analysis for primary design work. This equipment was designed to meet a detailed functional requirement specification. The design is supported by field and test experience.

2. Special Supplemental Methods

   Some complex equipment is normally sized by rational stress analysis techniques and requires supplemental criteria in areas where industrial codes do not apply.

For piping systems designed to the rules of ANSI B31.7, that is, Nuclear Classes II and III and Seismic Category I Piping, the stresses resulting from earthquake loads have been considered as follows:

1. OBE

   The vectorial combination of all longitudinal primary stresses does not exceed 1.2 times the hot allowable stress (S_h).

2. DBE

   The vectorial combination of all longitudinal primary stresses does not exceed the material yield stress at temperature unless higher allowable limits can be calculated by the methods outlined in Section 3.8.

The seismic evaluation of piping is discussed in detail in Section 3.7.3.8.

3.7.3.1 <u>Seismic Analysis Methods</u>

3.7.3.1.1 Bechtel-Supplied Equipment

All BOP Seismic Category I mechanical equipment was analyzed according to the General Project Seismic Requirements that were developed by Bechtel and the seismic consultants on the DAEC.

In most cases, the analysis of the equipment was made by the equipment vendor and reviewed by Bechtel; however, a few items were analyzed by Bechtel in the manner of a typical calculation.

The basis for ensuring proper functioning of the equipment during a seismic event was that the equipment should not fail or be subject to misoperation or malfunction during or after a seismic disturbance resulting from the OBE, nor should the equipment fail or be subject to misoperation or malfunction after a seismic disturbance resulting from the DBE. Operability during an OBE event was ensured by requiring that the maximum stresses from combined seismic and normal loads should not exceed the applicable code allowable stresses without the usual one-third increase of allowable stresses for short-term loadings. The induced displacements were also required to not exceed those required for safe operation. Operability after a DBE event was ensured by requiring that the maximum stress from combined seismic and normal loads should not exceed 90% of the yield stress of the material and that the induced displacements should not cause failure, malfunction, or prevent safe shutdown.

Three methods were permitted to ensure compliance with the project seismic requirements: frequency-not-determined analysis, frequency-determined analysis, or verification by testing. In all cases, the vertical seismic component was two-thirds of the foundation level horizontal acceleration for the OBE or DBE, applied simultaneously with the horizontal component.

For the frequency-not-determined method of analysis, the equipment was analyzed for a static coefficient equal to 1.5 times the peak acceleration of the floor response spectra. An additional discussion of this analysis method is found in Section 3.7.3.10. The stresses due to this static coefficient were combined with the normal stresses as indicated above.

For the frequency-determined method of analysis, the fundamental natural frequency of the equipment was determined by analysis. If the natural frequency was found to be greater than 33 cycles/sec (Hz), the equipment was categorized as "rigid" and analyzed for a constant acceleration equal to the peak support acceleration. If the natural frequency was found to be less than 33 cps, the equipment was categorized as "flexible" and analyzed for accelerations determined from a response acceleration spectrum appropriate for the mounting location of the equipment. Whether the equipment was rigid or flexible, the stresses due to the seismic accelerations were combined with the normal stresses as indicated above.

For items that were analyzed by testing, the equipment was subjected to vibration levels equivalent to those represented by the appropriate response acceleration spectrum for the mounting location of the equipment. For the OBE accelerations, the equipment was required to be operable during and after testing. For the DBE accelerations the equipment was required to be operable after the test.

It was found that certain types of equipment were most readily analyzed using the frequency-not-determined method. The Seismic Category I heating and ventilating duct systems typify this equipment. The complex geometries of a duct system make frequency determination extremely difficult; however, the small component weights make the induced seismic loads low. Thus, the Seismic Category I heating and ventilating ducts could be easily analyzed using the frequency-not-determined method and the resulting seismic and normal stresses kept within permitted values.

Equipment that was best analyzed by the frequency-determined method is typified by the several small Seismic Category I pumps used throughout the plant. The components of these pumps are relatively simple, and fundamental natural frequencies were determined using conventional methods of analysis. Invariably, the natural frequencies of these items fell in the rigid range with the resulting low seismic accelerations. Because of the inherent conservatism in the design of this type of equipment, and the low seismic coefficients required, the seismic stresses were of little consequence in the design of this equipment.

As of December 1972, no Seismic Category I mechanical equipment had been analyzed by testing; however, should testing be required, the methods used will be as described above.

3.7.3.1.2 General Electric-Supplied Equipment

The design of the Seismic Category I mechanical equipment must meet the design criteria set forth in this section. Proper functioning of Seismic Category I equipment during a seismic event is ensured by the fact that such equipment is analyzed for combinations of dead loads, live/operating loads, and seismic loads. The stresses resulting from such load combinations are less than allowable design stresses. For the MSIV, the structural acceptance criteria 3.8.4.5 is used for non-pressure boundary components. The results of the various loads analyses are included in Tables 3.7-8 through 3.7-20.

For dynamic analysis, Seismic Category I equipment is represented by models that consist of discrete masses connected by weightless springs. The criteria used to lump masses are the following:

1. Masses are chosen so that all significant modes are included.

2. A mass is located at each significant concentrated weight.

3. If the equipment has an overhang span whose flexibility is significant, an overhanging mass is used.

4. When a mass is located between two supports, it is located near the point of maximum displacement. This tends to conservatively lower the natural frequencies of the equipment. Similarly, in the case of live loads or variable support stiffness, the location of the load and the magnitude of support stiffness are chosen so as to lower the frequency. This ensures conservative dynamic loads.

3.7.3.2 Determination of Number of Earthquake Cycles

The stress in the piping is assumed to be occurring with a frequency of 20 Hz. This value is considered to be conservative, since it is greater than the significant natural frequencies of both the containment structures and the piping systems. The time duration of both the OBE and the DBE is assumed to be 15 sec. Two occurrences of the OBE and one occurrence of the DBE are assumed.

These values are also considered to be conservative. The number of earthquake cycles is, therefore, 600 for the OBE and 300 for the DBE. These numbers are specified in the Bechtel design specification for ASME Code, Section III, Nuclear Class I piping.

The following criteria were used in the design of Class I systems:

1. DBE

    a. Number of assumed DBE in the life of piping system is ≤ 1.

    b. Duration of strong motion vibration for each DBE is 15 sec.

    c. Number of cycles of the piping system for each DBE is 300.

    d. Total lifetime number of cycles of the piping system is 300.

2. OBE

    a. Expected number of equivalent OBE in the life of the piping system is 2.

    b. Average duration of strong motion vibration OBE is 15 sec.

    c. Average number of cycles of the piping system for each OBE is 300.

    d. Total lifetime number of cycles of the piping system is 600.

The OBE was considered to act concurrently with upset condition when making the ANSI B31.7 analysis, and the DBE was considered an emergency condition.

3.7.3.3 Procedure Used for Modeling

Analytical models used for structures and components are described in Sections 3.7.2.1 and 3.7.3.1. Models used for piping are described in Sections 3.7.3.8.

3.7.3.4 Basis for Selection of Frequencies

Frequencies for seismic evaluations were selected in accordance with Sections 3.7.2.1.1, 3.7.3.1 and 3.7.3.8.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

The conservative simplification of using equivalent static loads is described in Sections 3.7.3.1.1 and 3.7.3.8.4.

3.7.3.6 Three Components of Earthquake Motion

The method used to combine the components of earthquake motion are described in Section 3.7.2.1.

3.7.3.7 Combination of Modal Responses

Responses were combined as described in Section 3.7.2.7.

3.7.3.8 Analytical Procedures for Piping

The piping has been analyzed for the effects of thermal loads combined with deadweight and external forces. The calculated bending and torsional stresses are combined in accordance with the requirements of ANSI B31.1.0, 1967, Power Piping Code. Flexibility and stress intensification factors were applied in accordance with ANSI B31.1.0. Several thermal cycles were evaluated, and all critical points were compared to the expansion stress limits of ANSI B31.1.0. In addition, events with very low probability of occurrence were analyzed, and stresses of all critical points were evaluated to the limits defined in this section. The point of highest stress for each load combination is located, and the calculated stress is compared to the allowable stress in Tables 3.7-21 through 3.7-26.

The piping systems were dynamically analyzed using the response spectrum method. Input to the dynamic analysis was damped acceleration response spectra for the applicable floor elevation. The percentage of critical damping for all modes is 0.5% for the OBE and 1.0% for the DBE as given in Table 3.7-1.

The continuous piping system was mathematically idealized as an assembly of elastic structural members connecting discrete nodal points. Nodal points were placed in such a manner as to isolate particular types of piping elements, such as straight runs of pipe, elbows, and valves, for which force deformation characteristics can be categorized. Nodal points were also placed at all discontinuities, such as piping supports, concentrated weights, branch lines, and other critical points where stress calculation was desired.

The mathematical model was a lumped mass, multi-degree of freedom model. The distributed piping mass was “lumped” at the system nodal points. Valves were considered as lumped masses on the pipe, and valve operators are considered as lumped masses acting at the center of gravity of the operator. Inertia forces at the lumped masses resulted from the seismic-induced accelerations of the system supports.

Total system response, in terms of forces, moments, and seismic stresses, was obtained by the root-mean-square combination of the individual modal values for each direction of earthquake excitation input. The vertical direction earthquake was assumed to act simultaneously with either direction horizontal earthquake. The results of the seismic analyses were combined with other loading conditions, and combined stresses were computed in accordance with ANSI B31.1.0.

Constant load factors were not used in a multimass dynamic analysis such as piping seismic analysis. Inputs to such dynamic analyses are taken from the reactor building floor response spectra.

3.7.3.8.1 Balance-of-Plant Piping Analysis

BOP safety-related piping systems used the response spectrum technique to compute shears, moments, stresses, deflections, and/or accelerations for each seismic-excited piping mode. The piping system was idealized as a mathematical model consisting of lumped masses separated by elastic members. The lumped masses were carefully loaded so as to adequately represent the dynamic and elastic properties of the piping system. The three-dimensional stiffness matrix of the mathematical model was determined by the direct stiffness method.

The mass matrix was also calculated. After the stiffness and mass matrices of the mathematical model were calculated, the natural frequencies of vibration and corresponding mode shapes were determined using the following equation:

$$\left[\underline{K} - W_N^2 \underline{M}\right] \varnothing_N = \underline{0} \qquad (3.7\text{-}14)$$

where

$\underline{K}$	= stiffness matrix
W_N	= natural circular frequency for the Nth , mode
$\underline{M}$	= mass matrix
$\underline{0}$	= zero matrix
$\varnothing_N$	= mode shape matrix for the Nth mode

The mode shapes were normalized according to the following equation:

$$\varnothing_N^T \underline{M} \varnothing_N = 1 \qquad (3.7\text{-}15)$$

The maximum response of each mode was found through the following equation:

$$\underline{Y}_N(t)_{\max} = \frac{\varnothing_N^T \underline{MD}Sa}{W_N^2 M_N} \tag{3.7-16}$$

where

Sa = Spectral acceleration value for the N^{th} mode

$\underline{D}$ = earthquake vector matrix, used to introduce earthquake direction to the response analyses

$\varnothing_N^T$ = transposition of the N^{th} mode shape

M_N = generalized mass of the N^{th} mode shape

$\underline{Y}_N$ = generalized coordinate for the N^{th} mode

Using the maximum generalized coordinates for each mode, the maximum deflections associated with each mode are calculated using

$$V_N = \underline{\varnothing}_N Y_N(t)_{\max} \tag{3.7-17}$$

The root-mean-square method is used to combine the total modal responses as indicated by

$$Vi = \left[Vi_1^2 + Vi_2^2 + \ldots\ldots + Vi_N^2 \right]^{1/2} \tag{3.7-18}$$

where

Vi = deflection at i^{th} point due to the response of N modes

Vi_N = deflection at i^{th} point due to N^{th} mode

Once the appropriate deflections were determined for each mass and each mode, the effective applied forces for each mode were computed using the following equation:

$$\underline{Q}_N = \underline{KV}_N \tag{3.7-19}$$

where

$\underline{Q}_N$ = inertial forces due to mode N

The accelerations for each mode were calculated by the following equation:

$$\underline{a}_N = \underline{M}^{-1}\underline{Q}_N \tag{3.7-20}$$

where

$\underline{a}_N$ = accelerations due to N^{th} mode

M^{-1} = the inverse of mass matrix

After the effective forces have been determined, the internal forces (thrusts and shears) and moments for each mode were calculated using

$$\underline{S}_N = \underline{bQ}_N \tag{3.7-21}$$

where

$\underline{S}_N$ = internal forces and moments due to the N^{th} mode

$\underline{b}$ = force transformation matrix

The internal forces (thrusts and shears) and moments were combined on the same basis as Equation 3.7-18. The stresses were then computed from the internal forces and moments and in accordance with ANSI B31.7.

The percentage of critical damping for all modes is 0.5 for the OBE and 1.0 for the DBE.

The criteria for the selection and location of snubbers and dampers for Seismic Category I piping were as follows:

1. The use of snubbers or dampers was limited to those locations where unacceptable thermal expansion stresses would result from the use of a rigid translational restraint.

2. The snubbers and dampers have provision for thermal movement and can limit translational movement during the earthquake.

3. The snubbers and dampers were selected to sustain the seismic reaction resulting from the DBE at the point of attachment to the piping.

3.7.3.8.2 Computer Codes

The computer codes that were used by the San Francisco Power Division of Bechtel in the seismic stress analysis of safety-related piping are the following:

1. ME 632 - “Seismic Analysis of Piping Systems,” Bechtel.

2. ME 101 - “Leap” - “Linear Elastic Analysis of Pipe,” Bechtel.

3. PISOL - EDS Nuclear, Inc.

4. NUPIPE - Nuclear Services Corporation.

5. SAPIPE - PMB Systems Engineering, Inc.

6. TPIPE - PMB Systems Engineering, Inc.

The computer programs listed above have been verified as follows:

1. ME 632 has been verified using PISOL, PIPESD, and TPIPE.

2. ME 101 has been verified using ME 632, TPIPE, and SUPERPIPE.

3. PISOL has been verified using NUPIPE, PIPESD, ADLPIPE, and ME 101.

4. NUPIPE has been verified using ADLPIPE. (In the verification, the algebraic summation option in ADLPIPE was not used.)

5. SAPIPE has been verified using PISOL.

6. TPIPE has been verified using PISOL and ME 632.

3.7.3.8.3 Recirculation Piping and Nozzle Analysis

3.7.3.8.3.1 Recirculation Piping

The dynamic analysis that was performed on the recirculation piping systems determined the inertia effects of seismic loading. The primary objective of this analysis was to demonstrate that the recirculation piping loop A and loop B and their RHR piping meet the requirements of ANSI B31.1.0, 1967.

Because dynamic effects due to flow-induced vibration are insignificant, an analysis of this type was not performed. Water-hammer-effects analysis in the recirculation piping system was also considered and found to be negligible.

The seismic analyses were performed by using the method of response spectrum superposition. In this method, the maximum acceleration response of each mass of the mathematical model was computed for each significant mode using an appropriate acceleration response spectrum corresponding to the reactor building motion criteria. The computation for the acceleration response spectrum of each mass, which is the function of the maximum acceleration parameter versus the period of vibration, was based on a single-degree-of-freedom system subjected to the appropriate force vibration for an assumed critical damping ratio. All the significant mode shapes with frequencies less than 33 Hz were included in determining the seismic responses. Each seismic response parameter (inertia forces, displacements, member forces, reactions, etc.) was calculated using the sum of squares methods to determine the resultant value of the respective response parameters.

The seismic (inertia loads) cases were as follows:

1. NCASE 1 and 2

   OBE cases for vertical accelerations (Y direction) acting concurrently with horizontal north-south accelerations (Z direction) and for the same vertical accelerations acting concurrently with the horizontal east-west accelerations (X direction), respectively. The OBE cases assume an 0.5% damped acceleration response spectrum.

2. NCASE 3 and 4

   DBE cases for the same combined accelerations as OBE NCASE 1 and 2, respectively, except that the acceleration response spectra are for the DBE which are two times the OBE, using 1.0% of critical damping.

   The seismic anchor movement load case was based on the OBE and is considered in the analyses to determine the effects of relative earthquake displacements on the piping and supports.

   The horizontal displacements for the reactor pressure vessel nozzles and the anchors located just outside the drywell penetrations were based on the maximum value at the corresponding elevation. There were no relative vertical displacements.

   The loading conditions for thermal, weight/pressure, external forces, and seismic loadings for various normal and upset loadings, and loadings having a very low probability of occurrence, are individually considered and subsequently combined as to satisfy the rules of ANSI B31.1.0, 1967, and the following stress criteria:

3.7.3.8.3.2 Stress Criteria

See Tables 3.7-21 through 3.7-26.

3.7.3.8.3.3 Drawings

The following drawings provide piping isometric views of the recirculation line.

Drawing No.
C-518
M-116
M-332
M-333
M-338
M-339
M-340
M-341
M-352
M-353
M-357
APED-B11-2655-97
APED-B31-9(l)
APED-B31-9(2)
APED-B31-15(l)
APED-B31-15(2)
APED-B31-23(l)

3.7.3.8.3.4 Nozzle Analysis

Vessel Design Conditions

1. Design pressure, 1250 psig at vessel elevation 0.0.

2. Design temperature, 575°F.

3. Normal operating pressure - 1005 psig at top of vessel.

4. Normal operating temperature - Saturation temp at 1005 psig.

The calculated reactions are shown in Table 3.7-27.

The recirculation inlet nozzle design was found adequate in accordance with the requirements of ASME Code, Section III, Nuclear Vessels, and of General Electric Specification 2lA1100AS.

3.7.3.8.4 Equivalent Dynamic Analysis

There are two types of analysis in this category, as follows:

1. Analysis using first mode greater than the peak value.

2. Analysis using a modified spectrum curve.

Both of these approaches result in charts and tables showing span lengths and restraint forces for various building elevations.

3.7.3.8.4.1 First Mode Greater than Spectrum Peak

A piping system may be considered seismically acceptable if it can be divided into a series of simple spans. These spans are limited by guides that are specified in the form of vertical and lateral restraints at each change of direction, at all concentrated masses (e.g., valves), at all extended masses, at each tee, and at a maximum spacing on straight runs of piping defined by the following criteria. The fundamental frequency of multispan piping systems supported as stated above is greater than or equal to the fundamental frequency of a simple beam of maximum seismic span that is calculated using

$$f = \frac{\pi}{2}\sqrt{\frac{EI}{mL^4}} \qquad (3.7\text{-}28)$$

where

f	= fundamental frequency
E	= modulus of elasticity
I	= moment of inertia
m	= mass per unit length
L	= maximum seismic span (maximum distance between two seismic guides)

The frequency is chosen so that it is 20% larger than the frequency that defines the rigid side of the spectrum curve as shown in Figure 3.7-10. This is done on a case basis for each elevation. The simple beam formula can be used for the static equivalent load analysis with the peak value and at the spectrum curve and yields conservative results. A static load is then applied to the span to determine the maximum displacement, moment, and restraint force, which is calculated by

$$V = \frac{5}{384}\frac{mL^4 Sa}{EI} \qquad (3.7\text{-}29)$$

$$M = 0.125mL^2 Sa \qquad (3.7\text{-}30)$$

$$R = mL^2 Sa \qquad (3.7\text{-}31)$$

where

S_a	= the peak value of spectrum curve
V	= maximum displacement
M	= maximum moment
R	= maximum restraint force

Even though restraints are specified to ensure that the system will be on the rigid side of the curve, the spectral acceleration associated with the peak of the curve is used to obtain restraint loading and piping stresses.

A dynamic analysis was performed to verify the conservatism of the approach.

The piping system chosen for analysis was a general model that included various piping configurations. The model and results are given in Section 3.7.3.8.4.5. A sample chart of limiting span is shown in Table 3.7-28.

3.7.3.8.4.2 Modified Spectrum Method

The piping system is restrained as described above, except that the maximum spacing between two seismic guides on a straight run was determined by dynamic calculations using a modified spectrum curve. The spectrum curve for a particular building elevation was modified such that the flexible side of the peak of the curve remained constant at the peak spectral acceleration for decreasing frequencies (see Figure 3.7-11). A sample chart of limiting span is shown in Table 3.7-29.

If a dynamic analysis were performed using the above spectrum, the results would, by inspection, be conservative.

The fundamental frequency of the piping system supported as stated above is greater than the fundamental frequency of a simple beam of maximum seismic span, which was calculated using equation 3.7-28.

A dynamic analysis was then performed on a simply supported beam. The justification of this approach as well as a study demonstrating conservatism by comparing results of this approach with a dynamic analysis of a random piping system are presented later.

The following is a description of the development of the "modified spectrum method."

The circular frequency of a simple beam is calculated using

$$W_N = (N\pi)^2 \frac{EI}{mL^4} \qquad (3.7\text{-}32)$$

where

W_N = natural circular frequency for the N^{th} mode

The response spectrum method is then used to find the maximum response of each mode

$$V_{2N-1} = \frac{4S_{a(2N-1)}}{\pi(2N-1)W^2_{2N-1}}, V_{2N} = 0 \qquad (3.7\text{-}33)$$

where

S_{aN} = spectral acceleration value of the modified N spectrum curve for the N^{th} mode

V_N = maximum displacement (at mid-span) due to N^{th} mode

The SRSS method is used to combine the modal responses as described in Section 3.7.2.7.

$$V = \sqrt{V_1^2 + V_2^2 + \ldots V_N^2} \qquad (3.7\text{-}34)$$

where

V = maximum displacement due to the response of N modes

After the maximum displacement for each mode is determined, the maximum moment (at mid-span) and the maximum restraint force (at the support) are determined by

$$M_{2N-1} = \frac{4mL^2 Sa_{2N-1}}{\pi^3 (2N-1)^3}, M_{2N} = 0 \qquad (3.7\text{-}35)$$

$$R_{2N-1} = \frac{4mLSa_{2N-1}}{\pi^2 (2N-1)^2}, R_{2N} = 0 \qquad (3.7\text{-}36)$$

where

M_N = maximum moment (at mid-span) due to the N^{th} mode

R_N = maximum restraint force (at the support) due to the N^{th} mode

The moments and restraint forces are also combined on the same basis as Equation 3.7-34.

From the nature of the modified spectrum curve, the spectral acceleration for the first mode is always the largest value of the spectral accelerations of any mode. The first mode frequency for a given span is calculated and the resultant spectral acceleration is obtained. This maximum acceleration is then applied to all higher modes giving conservative results. The variables in Equations 3.7-34 through 3.7-36 can then be eliminated and the equations reduced to

$$V = 0.0131 \frac{mL^4 Sa}{EI} \qquad (3.7\text{-}37)$$

$$M = 0.1291 mL^2 Sa \qquad (3.7\text{-}38)$$

$$R = 0.8164 mLSa \qquad (3.7\text{-}39)$$

where

Sa = spectral acceleration of modified spectrum curve corresponds to the fundamental frequency of the maximum seismic span

V = maximum displacement

M = maximum moment

R = maximum restraint force

The analysis is made for both horizontal and vertical excitation. The horizontal and vertical responses are then combined by the square root of the sum of the squares.

The modified spectrum method as described above uses dynamic techniques that give results equal to a full dynamic analysis of a simply supported beam. It has been demonstrated that continuous beams restrained as described above will result in a response lower than that of the simply supported beam.

The modified spectrum method is therefore a dynamic analysis and the SRSS method is applied to the combination of responses. The larger of the square root of the sum of the squares of one horizontal and the vertical is used in the computations.

The static approach as described in Section 3.7.3.8.4.1 does, however, use the absolute sum method for the combination of responses.

Highest stressed regions are checked and verified for compliance with applicable codes.

3.7.3.8.4.3 General Guidelines

Although it is difficult to categorize as to which analytical classification a specific piping system will fit, certain generalizations can be made.

The major parts of the larger-diameter piping systems were analyzed using a full dynamic analysis. This was especially true where process fluid temperatures were high. For these systems, the large number of restraints required with the other techniques described would lead to thermal expansion difficulties. This also applies to small-diameter high-temperature systems. For other piping, namely large diameter-low temperature and small diameter-low temperature, one of the equivalent dynamic approaches was used.

Rigid-range piping techniques were typically reserved for instrumentation and some small-diameter piping. As previously stated, many conditions affect the selection of the appropriate technique. For example, a large-diameter system-low temperature system may be given a rigorous dynamic analysis to reduce the number of restraints required if the system is located such that installation of the restraints would be difficult.

3.7.3.8.4.4 Verification of Simplified Approach

The simplified approach is verified if it can be shown that the fundamental frequency of a piping system restrained as stated above is greater than or equal to the fundamental frequency of a simply supported beam (pin connected ends called SSB) of maximum seismic span (L).

The fundamental frequency of a multi-equal-span continuous beam is equal to the fundamental frequency of a simply supported beam of the single span length of the continuous beam.[2] For a multi-unequal-span continuous beam, the fundamental frequency of the maximum span using the SSB formula is less than the fundamental frequency of the multi-unequal-span continuous beam. This can be easily proved by considering a three-span continuous beam. Suppose that the middle span is longer than the two side spans: When the side spans are made smaller, the system approaches the fixed-fixed end case. Suppose that one of the side spans is the longest span: When the middle span is made smaller, the system is approaching the hinged-fixed end case. From the analytical results of a single simply supported beam with various end conditions, it can be concluded that the SSB formula gives the smallest frequency value of the three cases.[2,3] The same argument can be applied to multispan continuous beam. Therefore, the fundamental frequency of a piping system restrained as described above is greater than or equal to the fundamental frequency of a simply supported beam of the maximum seismic span (L).

Peak of response curve method is conservative for piping supported in accordance with Section 3.7.3.8.4.1.

When seismic spans are limited in length by methods described in Section 3.7.3.8.4.1, the first mode (fundamental) frequency falls on the rigid side of the peak of the spectrum curve (Figure 3.7-10). A safety factor of 1.2 is used to ensure that the first mode is on the rigid side of the peak.

Referring to the spectrum response curve (Figure 3.7-12), it can be seen that the nature of the curve is such that the lst mode will experience a higher spectral acceleration than the higher modes.

It can be shown that the first mode dominates the response for the simply supported beam.

The higher modes contribute less than 4% to the total system response. The coefficient of response is 0.129 for the combination of all modes for bending moments. A static case (consideration of only the lst mode) shows a coefficient of 0.125. The percentage difference is

$$\frac{0.129 - 0.125}{0.125} = 3.2\% \qquad (3.7\text{-}40)$$

Therefore, the use of the 1^{st} mode response differs from the total dynamic response by less than 4%.

In the interest of conservatism, however, the spectral acceleration corresponding to the peak of the spectrum response curve is used to determine piping stresses. This practice results in factors of conservatism of 3 to 8, that is,

$$\frac{S_{ap}}{S_{ar}} = 3 \text{ to } 8 \qquad (3.7\text{-}41)$$

where

S_{ap} = spectral acceleration corresponding to the peak
S_{ar} = spectral acceleration corresponding to rigid range

In actuality, the dynamic analysis using the response spectrum method results in a safety factor of 10.

The modified spectrum method is conservative for piping supported in accordance with Section 3.7.3.8.4.1.

Owing to the characteristics of the modified spectrum curve (see Figure 3.7-11), the span with the lowest fundamental frequency will always have a spectral acceleration equal to or greater than spans with a higher frequency.

It can be demonstrated by the same technique as used above that both the fixed-fixed end and fixed-hinged end case result in a lower dynamic response than for the simply supported beam.

Dynamic analyses were done for beams having five, six, and seven equal spans, and in all cases the stresses resulting were smaller than those obtained when the simple beam formula was used. The following is a comparison of the results of the dynamic analysis and the static equivalent load method of a typical problem.

3.7.3.8.4.5 Typical Problem (Example)

As a typical example, a piping system (Figure 3.7-13) has been analyzed by the dynamic analysis using the response spectrum method. The following results show that the static equivalent load method described in Section 3.7.3.8.4.1 yields a very conservative result:

	Dynamic Analysis	Static Equivalent Load Method
Fundamental frequency (cps)	12.10	11.10
Maximum stress (psi)	200.00	2,031.00
Maximum displacement (in.)	0.01	0.111
Maximum reaction (lb)	10.00	55.80

This typical example analyzed through both the dynamic analysis and the static equivalent load method uses the following criteria:

Number of degrees of freedom = 130
Number of modes considered in dynamic analysis = 30
Least significant period = 0.03 sec

The following is a comparison of the results of the dynamic analysis and modified spectrum method of a typical problem.

As a typical problem, a piping system (Figure 3.7-14) has been analyzed through the dynamic analysis using the response spectrum method. The following results have shown that the modified spectrum method yields a very conservative result:

	Dynamic Analysis	Modified Spectrum Method
Fundamental frequency (cps)	11.05	6.47
Maximum stress (psi)	1300.00	5703.00
Maximum displacement (in.)	0.05	0.438
Maximum reaction (lb)	16.00	22.00

This typical example analyzed by both the dynamic analysis and the modified spectrum methods (Figure 3.7-12) uses the following criteria:

Number of degrees of freedom	= 112
Number of modes considered in dynamic analysis	= 20
Least significant period	= 0.03 sec

The information presented in this section is contained in Bechtel Corporation Topical Report BP-TOP-1.

3.7.3.8.5 NSSS Piping

NSSS Class I piping and equipment supplied by General Electric, mainly fall under four categories as follows:

1. Piping (recirculation loop and primary steam piping).

2. Equipment such as pumps, heat exchangers, and tanks.

3. Instrumentation.

4. Reactor pressure vessel internals.

The methods of seismic analysis for the above categories are discussed below.

Piping

The piping systems are dynamically analyzed using the response spectrum method. For each of the piping systems, a mathematical model consisting of lumped masses at discrete joints connected together by weightless elastic elements is constructed. For the piping runs, the number of lumped masses is normally adequate to include all the vibration modes with frequencies less than 33 Hz. In addition, masses are lumped at the points where concentrated weights, like valves, etc. are located. Stiffness matrix and mass matrix are then generated, and natural periods of vibration and corresponding mode shapes are determined. Input to the dynamic analyses are the acceleration response spectra for the support locations. The increased flexibility of the curved segments of the piping systems is considered. The torsional effects of valves and other eccentric masses are normally considered in the development of a dynamic model.

If the stresses due to these effects can readily be shown to be less than 500 psi, the torsional effects of these components are often neglected in the dynamic model. The results for earthquakes acting in the X and Y (vertical) directions simultaneously and the Z and Y directions simultaneously are computed separately. Maximum joint displacements, member forces, and support reactions are determined by a square root of the sum of the square combination of each of these parameters for each mode and for each set of earthquake directions. The member forces thus obtained are combined with the member forces produced by other loading conditions to compute the stresses. The applicable stress and deformation criteria for all the loads including seismic loads are defined in Section 3.9.

3.7.3.8.6 Piping with Multiple Input Attachments

The relative displacement between piping and equipment under seismic excitation is interpreted as the relative displacement between the supports of piping systems. The following method was used to determine the effect of differential end and support motion on piping systems. The seismic displacements at the ends and at restraints are known from the building analyses. These displacements were applied to the piping restraints and anchors corresponding to the maximum differential displacements that could occur. The analysis was made twice: once for north-south differential displacements and once for east-west differential displacements. For each response quantity considered (i.e., moments or displacements at a point, and restraint force or moment), the largest value of the two analyses was chosen. The displacements and restraint forces and moments were combined with the corresponding quantity from the inertia load analysis. The basis of combination was root mean square, since the maximums of the two quantities would not occur at the same time. The internal moments were used to calculate the stresses in accordance with ANSI B31.7.

The results of the differential displacement analysis were usually insignificant compared to the results of the inertia force analysis. The differential displacements were usually very small, and most piping systems (especially hot ones) had enough flexibility that these small displacements caused very little distress.

The above discussion applies to BOP equipment and piping; the paragraph below applies to the NSSS equipment and piping.

Criteria and methods of analysis are included in Tables 3.7-21 through 3.7-26. The results obtained from the maximum seismically-induced relative anchor displacements are accounted for by adding to the results from normal seismic analyses.

For piping systems that go between different structures, such as between the reactor building and auxiliary buildings, the spectrum curves at the end points may be radically different. For these pipes, if intermediate restraints exist between the pipe and one of the two structures, a spectrum curve associated with the restraining structure was used. If no intermediate restraints existed, the average of the spectrum curves associated with the end points was used.

In 1985, the loads at seismic supports were reanalyzed as a part of IE Bulletin 79-14 activities. The support calculations showed that all supports were operable, but at some supports the component reactions exceeded the design rating. Those supports have been modified to restore the original design factor of safety.

3.7.3.8.7 Seismic Analyses of As-Built Safety-Related Piping Systems

In response to IE Bulletin 79-14, the DAEC inspected as-built safety-related piping systems and found no nonconformances. The inspections described in IE Bulletin 79-14, Items 2 and 3, were conducted for the DAEC by Bechtel using its Procedure for Verifying Conformance of Seismic Analysis to Actual Configuration of Safety-Related Piping Systems. The procedure and the results of the inspections were included in Reference 4.

3.7.3.9 Multiple-Supported Equipment Components With Distinct Inputs

The only significant components subject to multiple simultaneous seismic input are certain piping subsystems. The evaluation of the effects of such configurations and inputs is presented in Section 3.7.3.8.6.

3.7.3.10 Use of Static Coefficients

Equipment specifications for certain mechanical and electrical equipment allow a static analysis using a coefficient of 1.5 times the peak of the applicable floor response spectrum. This coefficient was derived from a study of different structure configurations and was shown to be conservative.

The study included the following representative simple structural configurations:

1. Uniform cantilever comprised of shear elements.
2. Uniform cantilever comprised of bending elements.
3. Uniform beam with fixed ends.
4. Uniform beam with pinned ends.
5. Rigid cantilever with a flexible rocking base.

For all cases, it was assumed that the spectral accelerations of the first mode were equal to the peak of the spectrum and that the spectral accelerations of the higher modes were equal to one-half that value. These assumptions are conservative relative to the floor response spectra for the plant. The resultant mass accelerations for each mode were computed using participation factors and then combined by calculating the square root of the sum of the squares of the modal values. The maximum resulting acceleration in the structure represents the acceleration that is applied to the item of piping or equipment for design purposes. This acceleration is expressed as a coefficient times the peak spectral accelerations in the following tabulation. (Also included are the coefficients that would be obtained if only the first mode response is considered.)

	Coefficient	
Case	First Mode Only	All Significant Modes
1	1.27	1.33
2	1.46	1.50
3	1.29	1.31
4	1.25	1.26
5	1.43	1.46

The maximum coefficient of 1.50 for a slender cantilever represents a reasonable upper bound.

For the Seismic Category I piping that was analyzed using a static load equivalent, the analysis was based on the simple beam analysis of the maximum seismic span of piping system providing a seismic guide (restraining the pipe laterally) at each change of direction, at all concentrated masses (e.g., valves), at each tee, and at a maximum spacing (which was determined by the following method) on a straight run of piping.

The fundamental frequency of the small-diameter piping system supported as stated previously is greater than the fundamental frequency of a simple beam of maximum seismic span that is calculated using

$$f = \frac{\pi}{2}\sqrt{\frac{EI}{mL^4}} \tag{3.7-42}$$

where
f = fundamental frequency
E = modulus of elasticity
I = moment of inertia
m = mass per unit length
L = maximum seismic span (maximum distance between two seismic guides)

The static load is equivalent to the peak value of the appropriate spectrum curve. The seismic span is limited by the fundamental frequency (Equation 3.7-42) to be greater than the frequency of the peak area of the spectrum curve. The piping system is then in the rigid side of the peak area of the spectrum curve. The simple beam formula can be conservatively used for the static equivalent load analysis with the peak value. The maximum displacement, moment, and restraint force are calculated by equations 3.7-29 through 3.7-31.

The analysis is made for both horizontal and vertical excitation. The horizontal and vertical responses are then combined by the sum of the absolute values method. The stresses are computed in accordance with ASME Code, Section III.

A typical case has been dynamically analyzed using the response spectrum method, with the following results:

Maximum stress	= 250 psi
Maximum displacement	= 0.01 in.
Maximum reaction	= 10 lb

The simplified analysis as described above shows the following:

Maximum stress	= 2031 psi
Maximum displacement	= 0.111 in.
Maximum reaction	= 55.8 lb

It is evident from the above discussion that the simplified analysis is conservative.

3.7.3.11 Torsional Effects of Eccentric Masses

The effect of eccentric masses such as valve operators is discussed in Section 3.7.3.8.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

Buried Seismic Category I pipes are laid in a prepared trench and backfilled with select material. The backfill material is compacted to 95% of maximum density as determined by the AASHO T180 Method “D”. Field quality control is performed in accordance with AASHO T147.

Where Seismic Category I piping enters the building near the base, differential movement between the building and soil at the location of pipe penetrations may be considered to be zero. Where Seismic Category I piping enters the secondary containment near the ground surface, flexible or rigid seals are provided as required.

Seismic Category I piping is designed for the maximum relative differential movement that could occur at the support points. Because the stresses resulting from this maximum relative differential movement are not likely to occur in phase with the maximum stresses due to dynamic response of the pipe, if any, the two were combined on a root-mean-square basis.

The seismic stresses are combined in accordance with the design rules of Section 3.8.

3.7.3.13 Seismic Analysis for Reactor Internals

General Electric used the response spectrum method for seismic analyses of piping systems and reactor vessel and internals. The resultant displacements, loads (namely shears and moments), and stresses were computed by the square root of the sum of the square of the corresponding individual model responses (displacement, loads, or stresses).

A complete discussion of the seismic analysis of the reactor vessel and internals is presented in Section 3.9.5.2.3.

3.7.3.14 Analysis Procedure for Damping

Damping analysis is discussed in Section 3.7.2.14.

REFERENCES FOR SECTION 3.7

1. Whitman, R. V., and Richart, R. E., Jr., "Design Procedures for Dynamically Loaded Foundations," J. Soil Mech. and Fnd. Div., ASCE, SM6, November, 1967.

2. Biggs, J. M., "Introduction to Structural Dynamics," McGraw-Hill, 1964.

3. Hurty, W. C. and Rubinstein, M. F., "Dynamics of Structures," Prentice-Hall, 1964.

4. Letter from L. D. Root, Iowa Electric, to J. G. Keppler, NRC, Region III, Subject: Final Response to IE Bulletin No. 79-14 Concerning Seismic Analyses for As-built Safety-related Piping Systems, dated October 17, 1980 (LDR-80-284).

5. Roark, R. J., "Formulas for Stress and Strain," McGraw-Hill, 1954.

6. Holtec Report, HI-92889, Licensing Report for Spent Fuel Pool Storage Capacity Expansion DAEC, Transmitted to NRC along with RTS-252, NG-93-0566, dated March 26, 1993.

Table 3.7-1

DAMPING VALUES[a]

	Percent of Critical Damping	
	Operating-Basis Earthquake	Design-Basis Earthquake
Containment structure and all internal concrete structures	2.0	5.0
Other conventionally reinforced concrete structures; such as shear walls or rigid frames	5.0	5.0
Welded structural steel assemblies	1.0	1.0
Bolted or riveted steel assemblies	2.0	2.0[b]
Piping systems	0.5	1.0
Foundations, rock or lean concrete backfill, soil	5.0	5.0

[a] The damping values listed are the damping values originally used. The damping values found in NRC Regulatory Guide 1.61 are also acceptable for use in seismic analyses.

[b] Seismic analysis was performed on certain cable trays (1L3A, 1M3A, 1M5A and 1N5A) after fireproofing was added in response to 10 CFR 50, Appendix R. A damping value of 5% for a DBE was used for the analysis.

Table 3.7-2

EXAMPLE OF BUILDING ANALYSIS
NATURAL PERIODS OF VIBRATION AND ASSIGNED DAMPING
VALUES FOR SIGNIFICANT REACTOR BUILDING MODES

Mode Number	Period of Vibration (sec)	Damping Ratio (%)
1	0.288	2
2	0.228	5
3	0.120	1
4	0.072	5
5	0.059	2
6	0.047	2
7	0.043	2
8	0.036	2
9	0.032	2

Table 3.7-3

EXAMPLE OF BUILDING ANALYSIS
MODAL RESPONSE, REACTOR BUILDING

Mass Point	Elevation	Acceleration (g units)	Displacement (in.)	Shear (kips)	Moment (ft-kips)
1	[redacted]	0.500	0.581	- -	0
2		0.165	0.122	561	23600
3		0.141	0.099	2580	81000
4		0.117	0.078	4920	176000
5		0.090	0.050	6960	362000
6		0.080	0.017	8250	598000
7		0.070	0.002	8780	955000
Base		0.070	0.002	11240	1070000

Table 3.7-4

EXAMPLE OF BUILDING ANALYSIS
MODAL RESPONSE, DRYWELL

Mass Point	Elevation	Acceleration (g units)	Displacement (in.)	Shear (kips)	Moment (ft-kips)
8	[redacted]	0.292	0.117	- -	0
9		0.179	0.102	17	249
10		0.148	0.098	26	354
11		0.133	0.089	30	586
12		0.110	0.070	450	4820
13		0.103	0.049	454	9360
14		0.102	0.045	457	11940
15		0.094	0.034	461	16460
16		0.092	0.023	465	21550
17		0.081	0.011	466	26760
18		0.077	0.006	468	31400
7		0.070	0.002	468	35600

Table 3.7-5

EXAMPLE OF BUILDING ANALYSIS
MODAL RESPONSE, REACTOR PRESSURE VESSEL

Mass Point	Elevation	Acceleration (g units)	Displacement (in.)	Shear (kips)	Moment (ft-kips)
19	[redacted]	0.212	0.115	- -	0
20		0.181	0.107	15	115
- -		0.157	0.091	115	1870
21		0.137	0.083	209	1550
22		0.138	0.077	125	1980
23		0.135	0.069	75	2520
24		0.128	0.060	36	2690
25		0.132	0.050	47	2520
26		0.112	0.038	70	1980

Table 3.7-6

EXAMPLE OF BUILDING ANALYSIS
MODAL RESPONSE, SACRIFICIAL SHIELD

Mass Point	Elevation	Acceleration (g units)	Displaceme nt (in.)	Shear (kips)	Moment (ft-kips)
- -	[redacted]	0.149	0.090	- -	0
27		0.148	0.082	63	687
28		0.144	0.072	20	820
29		0.133	0.057	30	668
26		0.112	0.038	63	444

Table 3.7-7

EXAMPLE OF BUILDING ANALYSIS
MODAL RESPONSE, PEDESTAL

Mass Point	Elevation	Acceleration (g units)	Displaceme nt (in.)	Shear (kips)	Moment (ft-kips)
26	[redacted]	0.112	0.038	- -	2360
30		0.098	0.027	154	1210
31		0.082	0.017	179	1460
32		0.077	0.007	197	2850
7		0.070	0.002	208	4300

Table 3.7-8 (a) Sheet 1 of 2

STRESS SUMMARY FOR NEW FUEL AND PaR SPENT FUEL RACKS

Criteria	Loading Condition	Location	Stress (psi) Allowable	Calculated[a]
New fuel storage racks	Emergency:	Column	16,000	2,950
Stresses due to normal, upset, or emergency loading shall not cause the racks to fail so as to result in a critical fuel array	Dead loads Full fuel load in rack Design-basis earthquake	Base to column welds	11,000	100
Primary stress limit				
Paper number 3341 and 3342, Proceedings of the ASCE, Journal of the Structural Division, December 1962 (task committee on light-weight alloys)		Channel	20,000	3,150
		Support channel to column weld	6,000	2,650
PaR Spent fuel storage racks[b]	Emergency "A"	At column to base welds	11,000	760
Stresses due to normal, upset, or emergency loading shall not cause the racks to fail so as to result in a critical fuel array	Dead loads Full fuel load in rack Design-basis earthquake	Support beam	35,000	33,500
Primary stress limits				
Paper numbers 3341 and 3342, Proceedings of the ASCE, Journal of the Structural Division, December 1962 (task committee on lightweight alloys)				

[a] These values are calculated for 1.5g static seismic coefficient applied horizontally to the rack. Actual earthquake loads give much lower than these due to the structural stiffness.
[b] Applies to original spent fuel storage racks that were replaced with high-density, PaR racks.

Table 3.7-8 (a) Sheet 2 of 2

STRESS SUMMARY FOR NEW FUEL AND PaR SPENT FUEL RACKS

Criteria	Loading Condition	Location	Stress (psi) Allowable	Stress (psi) Calculated[a]
Emergency conditions	Emergency "B"			
Stress limit = yield Strength at 0.2% offset				

Note: Emergency condition "B":

1. Loading. In addition to testing the capability of the racks to withstand the loading conditions given in this table, the racks were tested and analyzed to determine their capability to safely withstand the accidental, uncontrolled drop of the fuel grapple from its full retracted position into the weakest portion of the rack.

2. Method of Analysis. The displacement of the vertical column at the ends of the racks was determined by considering the effect of the grapple kinetic energy on the upper structure. The energy absorbed in the shearing of the rack longitudinal structural member welds was determined. The effect of the remaining energy on the vertical columns was analyzed. Equivalent static load tests were made on the structure to ensure that the criteria were met.

3. Results of Analysis. All criteria were met. Analysis showed that the grapple would shear the welds in the area where the impact occurred. The longitudinal structural member bends, but it does not fail in shear. Grapple penetration into the rack is not sufficient to cause the vertical columns to deflect the fuel into a critical array. Static load testing showed that forces in excess of those resulting from a grapple drop are required to cause the columns to deflect to the extent that the criteria are violated.

[a] These values are calculated for 1.5g static seismic coefficient applied horizontally to the rack. Actual earthquake loads give much lower than these due to the structural stiffness.

Table 3.7-8 (b) Sheet 1 of 2

STRESS SUMMARY FOR HOLTEC SPENT FUEL STORAGE RACKS (REF. 6)

Criteria	Loading Condition	Location	Stress (psi) Allowable	Stress (psi) Calculated[a]
Holtec Spent Fuel Storage Racks	Emergency “C”	Rack to Baseplate Weld	29820	9277
	Dead Loads	Baseplate to Pedestal Weld (Dimensionless Limit Load Ratio)	1.0	.283
	Full Fuel Load in rack	Cell to Cell Weld	5271	1222
	Design Basis Earthquake	Bearing Pad	2380	719
Primary Stress Limit				
ASME SA-240-304 (Rack Mat’l), ASTM-240, Type 304 (upper part of support feet), ASTM 564-630 (lower part of support feet, age hardened at 1100 °F)				
Emergency Conditions	Emergency “D”			
Stress Limit = ASME Code, Section III Subsection NF, Yield Strength at 200°F (Max. Pool Temp.)	See Note✸			

[a] These values are calculated for 1.5g static seismic coefficient applied horizontally to the rack. Actual earthquake loads give much lower than these due to the structural stiffness.

Table 3.7-8 (b) Sheet 2 of 2

STRESS SUMMARY FOR HOLTEC SPENT FUEL STORAGE RACKS (REF. 6)

✶ These values are based on dynamic analysis using design base earthquake (DBE) response spectrums (horizontal and Vertical). DBE response spectrums were generated by multiplying by two the operating base earthquake (OBE) response spectrums of the Reactor Building floor elevation 812'. The values shown are extracted form the largest rack in the spent fuel pool.

✷ Note: Emergency Condition "D":

1. Loading. In addition to testing the capability of the racks to withstand the loading conditions given in this table, the free standing Holtec spent fuel storage racks are also analyzed to determine their capability to safely withstand the drop of fuel assembly (680 lbs.) with associated handling equipment (120 lbs.), being carried 18" above the spent fuel rack.

2. Method of Analysis. An 800 lbs. fuel assembly plus handling equipment is dropped from 18 " above the top of a storage location and impacts the base of the module. The rack design should ensure that gross structural failure does not occur and the subcriticality of the adjacent fuel assemblies is not violated.

3. Results of Analysis. Calculated results show that there will be no change in the spacing between cells. Local deformation of the baseplate will in the neighborhood of the impact will occur, but the dropped assembly will be contained and not impact the liner. In the case when the fuel assembly with the channel is dropped from 18" above the top of the rack, and impacts the top of the rack, it is shown that the damage, if it occur, will be restricted to a depth of less than or equal to 1.09" below the top of the rack. This is above the active fuel region. Analysis of the Local buckling of the fuel cell walls, and in-rack welded joints shows that the maximum stresses are within allowable limits.

Table 3.7-9 Sheet 1 of 2

STRESS SUMMARY FOR RHR PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
<u>Closure bolting</u>	Bolting loads and stresses calculated per "Rules for Bolting Flange Connections," ASME Code, Section VIII, Appendix II	Maximum allowable stress = 20,000 psi	Maximum calculated stress = 9990 psi
<u>Loads</u>:			
<u>Normal and upset</u>: Design pressure and temperature, design gasket load			
<u>Bolting stress limit</u>: Allowable working stress per ASME Code, Section VIII			
<u>Wall thickness</u>	Per rules of ASME Code, part UG, Section VIII	Maximum allowable stress - main pump = 14,000 psi	Maximum calculated stress = 12,154 psi
<u>Loads</u>:			
<u>Normal and upset</u>: Design pressure and temperature			
<u>Stress limit</u>: ASME Code, Section III			
<u>Nozzle</u>	For the maximum moment due to pipe reaction, the maximum force shall not exceed the allowable	Force in lb Moment in ft-lb	
<u>Loads</u>:			
<u>Normal plus upset</u>: Design pressure and temperature, deadweight, thermal expansion, and operating-basis earthquake	Total nozzle stress with these criteria does not exceed stress limits	<u>Normal plus upset</u>: <u>Suction</u> F= 83,000-1.7M	<u>Normal plus upset</u>: <u>Suction</u> Force = 7,376 lb Moment = 13,894 ft-lb

Table 3.7-9 Sheet 2 of 2

STRESS SUMMARY FOR RHR PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Normal and upset (Continued):		Discharge F = 62,000-2.23M	Discharge Force = 4985 lb Moment = 14,329 ft-lb
Loads:			
Emergency: Design pressure and temperature, deadweight, thermal expansion, and design-basis earthquake		Emergency: Suction F = 125,000- 1.7M	Emergency Suction Force = 9,900 lb Moment = 19,563 ft-lb
Stress limit: ASME Code, Section VIII, for normal and upset, 1.5 of allowable stress for emergency		Discharge F = 92,000 - 2.23M	Discharge Force = 7116 lb Moment = 18,609 ft-lb

Table 3.7-10 Sheet 1 of 2

STRESS SUMMARY FOR RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
<u>Closure bolting</u>	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections," ASME Code, Section VIII, Appendix II		
	a. Shell cover bolts	25,000 psi	23,700 psi
	b. Channel cover bolts	25,000 psi	23,900 psi
<u>Loads</u>:			
<u>Normal and upset</u>: Design pressure and temperature, design gasket load			
<u>Bolting stress limit</u>: Allowable working stress per ASME Code, Section VIII			
<u>Wall thickness</u>	Shell side, ASME Code, Section III, and TEMA Class A		
<u>Loads</u>:			
<u>Normal and upset</u>: Design pressure and temperature	Tube side, ASME Code, Section VIII, and TEMA Class C		
<u>Stress limit</u>: ASME Code, Section VIII	a. Shell	0.6478 in.	0.875 in.
	b. Shell cover	0.6407 in	0.875 in.
	c. Channel ring	0.6478 in.	1.000 in.
	d. Tubes	0.0548 in.	18 BWG
	e. Channel cover	5.1857 in	5.500 in.
	f. Tube sheet	5.2815 in.	5.500 in.

Table 3.7-10 Sheet 2 of 2

STRESS SUMMARY FOR RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Nozzle	For the maximum moment due to pipe reaction, the maximum force shall not exceed the allowable	Force in lb Moment in ft-lb (maximum in any direction)	
Loads:			
Normal plus upset: Design pressure and temperature, deadweight, thermal expansion and operating-basis earthquake	Primary stress less than $1.5S_m$ and secondary stress less than $3.0S_m$	Normal plus upset:	
		N1 F = 48,000-1.1M	Force = later Moment = later
		N2 F = 48,000 - 1.1M	Force = later Moment = later
		N3 F = 45,000-0.94M	Force = 6430 lb Moment = 15,257 ft-lb
		N4 F = 62,000 - 1.0M	Force = 3313 lb Moment = 14,043 ft-lb
Loads:			
Emergency: Design pressure and temperature, deadweight, thermal expansion, and design-basis earthquake	Primary stress less than $1.8S_m$	Emergency:	Emergency:
		N1 F = 74,000 -0.74M	Force = later Moment = later
		N2 F = 74,00 - 0.74M	Force = later Moment = later
Stress limit: ASME Code, Section VIII		N3 F = 72,000 - 0.6M	Force = 7378 lb Moment = 20,390 ft-lb
		N4 F = 108,000 - 0.68M	Force = 4959 lb Moment = 19,339 ft-lb

Table 3.7-11 Sheet 1 of 2

STRESS SUMMARY FOR CORE SPRAY PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Closure bolting	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections," ASME Code, Section VIII, Appendix II	Maximum allowable stress = 20,000 psi	Maximum calculated stress = 9030 psi
Loads:			
Normal and upset: Design pressure and temperature, design gasket load			
Bolting stress limit: Allowable working stress per ASME Code, Section VIII			
Wall thickness	Per rules of ASME Code, Part UG, Section VIII	Maximum allowable stress - main pump = 14,000 psi	Maximum calculated stress = 11,431 psi
Loads:			
Normal and upset: Design pressure and temperature			
Stress limit: ASME Code, Section III			
Nozzle	For the maximum moment due to pipe reaction, the maximum force shall not exceed the allowable	Force in lb Moment in ft-lb	
Loads:			
Normal plus upset: Design pressure and temperature, deadweight, thermal expansion, and operating-basis earthquake	Total nozzle stress with these criteria does not exceed stress limits	Normal plus upset: Suction F = 81,000 - 1.56M	Normal plus upset: Suction Force = 1437 lb Moment = 6545 ft-lb

Table 3.7-11 Sheet 2 of 2

STRESS SUMMARY FOR CORE SPRAY PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
		Discharge	Discharge
		F = 61,000 - 2.23M	Force = 1497 lb Moment = 2789 ft-lb
Loads:			
Emergency: Design pressure and temperature, deadweight, thermal expansion, and design-basis earthquake		Emergency: Suction F = 122000 - 1.56M	Emergency: Suction Force = 1716 lb Moment = 8207 ft-lb
		Discharge	Discharge
		F = 92,000 - 2.23M	Force = 2879 lb Moment = 5640 ft-lb
Stress limit: ASME Code, Section VIII, for normal and upset; 1.5 of allowable stress for emergency			

Table 3.7-12 Sheet 1 of 3

STRESS SUMMARY FOR RCIC TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Closure bolting	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections," ASME Code, Section VIII, Appendix II	Maximum allowable stress = 20,000 psi	Maximum calculated stress = 6400 psi
Loads:			
Normal and upset: Design pressure and temperature, design gasket load			
Bolting stress limit: Allowable working stress per ASME Code, Section VIII			
Casing wall thickness	Per rules of ASME Code, Part UG, Section VIII	Maximum allowable stress = 17,500 psi	Maximum calculated stress = 12,700 psi
Loads:			
Normal and upset: Design pressure and temperature			
Stress limit: ASME Code, Section III			
Nozzle	For the resultant moment due to pipe reaction, the resultant force shall not exceed the allowable	Force in lb Moment in ft-lb	
Loads: Normal: Design pressure and temperature, deadweight, and thermal expansion	Detailed design analysis has demonstrated the limits	Inlet F = (2620 - M)/3 Exhaust F = (6000 - M)/3	Inlet Force = 470 lb Moment = 1797 ft-lb Exhaust Force = 356 lb Moment = 2527 ft-lb

Table 3.7-12 Sheet 2 of 3

STRESS SUMMARY FOR RCIC TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Loads:			
Normal plus upset: Design pressure and temperature, deadweight, thermal expansion, and operating-basis earthquake		Normal plus upset: Inlet F = (3000 - M)/2.5 Exhaust F = 3(6000 - M), but not >8370 lb	Normal plus upset: Inlet Force = 633 lb Moment = 2679 ft-lb Exhaust Force = 849 lb Moment = 3357 ft-lb
Loads:			
Emergency: Design pressure and temperature, deadweight, thermal expansion, and design-basis earthquake		Emergency: Inlet F = (4500 - M)/2.5 Exhaust F = 3(9000 - M), but not >12,555 lb	Emergency: Inlet Force = 803 lb Moment = 3586 ft-lb Exhaust Force = 1355 lb Moment = 8376 ft-lb
Stress limit: Specified by vendor for normal loads; ASME Code, Section VIII, for upset loads; increased 20% for emergency loads			
Turbine mounting bolt (turbine to baseplate)	Vertical and horizontal forces on mounting bolts calculated as the sum of seismic accelerations on the turbine and the pipe reaction forces and moments on the nozzles		

Table 3.7-12 Sheet 3 of 3

STRESS SUMMARY FOR RCIC TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Loads:			
Normal and upset: Operating-basis earthquake, nozzle loads for operating-basis earthquake, deadweight, and thermal expansion		Tensile and shear stress for bolting materials are specified in ASME Code, Section VIII	By meeting the nozzle load criteria above, the detailed seismic analyses indicate the mounting bolts satisfy the allowable stress requirements
Loads:			
Emergency: Design-basis earthquake, nozzle loads for design-basis earthquake, deadweight, and thermal expansion		Tensile stress less than 0.9 yield and shear stress less than twice allowable by ASME Code, Section VIII	By meeting the nozzle load criteria above, the detailed seismic analyses indicate the mounting bolts satisfy the allowable stress requirements
Stress limits: ASME Code, Section VIII, allowables for normal and upset loads; for emergency loads 0.9 yield and twice allowable shear			

Table 3.7-13 Sheet 1 of 2

STRESS SUMMARY FOR RCIC PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Closure bolting:	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections," ASME Code, Section VIII, Appendix II	Maximum allowable stress = 25,000 psi	Maximum calculated stress = 19,288 psi
Loads:			
Normal and upset: Design pressure and temperature, design gasket load			
Bolting stress limit: Allowable working stress per ASME Code, Section VIII			
Wall thickness	Per rules of ASME Code, Part UG, Section VIII	Maximum allowable stress - main pump = 14,000 psi	Maximum calculated stress = 11,960 psi
Loads:			
Normal and upset: Design pressure and temperature			
Stress limit: ASME Code, Section III	Volute stress is calculated per Roarck[5]	Maximum allowable stress - main pump = 14,000psi	Maximum calculated stress = 11,913 psi
Nozzle	For the maximum moment due to pipe reaction, the maximum force shall not exceed the allowable	Force in lb Moment in ft-lb	
Loads:			
Normal plus upset: Design pressure and temperature, deadweight, thermal expansion, and operating-basis earthquake	Total nozzle stress with these criteria does not exceed limits	Normal plus upset: Suction F = 9400 - 2.50M	Normal plus upset: Suction Force = 1029 lb Moment = 2916 ft-lb

Table 3.7-13 Sheet 2 of 2

STRESS SUMMARY FOR RCIC PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
		Discharge F = 9400 - 4.33M	Discharge Force = 756 lb Moment = 1231 ft-lb
Loads:			
Emergency: Design pressure and temperature, deadweight, thermal expansion, and design-basis earthquake		Emergency: Suction F = 19,000 - 2.42M	Emergency: Suction Force = 1949 ft Moment = 5594 ft-lb
		Discharge F = 19,000 - 5.05M	Discharge Force = 1211 lb Moment = 1358 ft-lb
Stress limits: ASME Code, Section VIII, for normal and upset loads; 1.5 allowable stress for emergency loads			

Table 3.7-14 Sheet 1 of 3

STRESS SUMMARY FOR HPCI TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Closure bolting	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections," ASME Code, Section VIII, Appendix II	Maximum allowable stress = 20,000 psi	Maximum calculated stress = 18,290 psi
Loads:			
Normal and upset: Design pressure and temperature, design gasket load			
Bolting stress limit: Allowable working stress per ASME Code, Section VIII			
Casing wall thickness	Per rules of ASME Code, Part UG, Section VIII	Maximum allowable stress = 17,500 psi	Maximum calculated stress = 7200 psi
Loads:			
Normal and upset: Design pressure and temperature			
Stress limit: ASME Code, Section III			
Nozzle	For the resultant moment due to pipe reaction , the resultant force shall not exceed the allowable	Force in lb Moment in ft-lb	
Loads:			
Normal: Design pressure and temperature, deadweight, and thermal expansion	Detailed design analysis had demonstrated the acceptability of these values	Normal: Inlet F = (7570 - M)/3	Normal: Inlet Force = 1760 lb Moment = 6219 ft-lb

Table 3.7-14 Sheet 2 of 3

STRESS SUMMARY FOR HPCI TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
		Exhaust F = (9930 - M)/3	Exhaust Force = 1,476 lb Moment = 4,684 ft-lb
Loads:			
Normal plus upset: Design pressure and temperature, and operating-basis earthquake		Normal plus upset: Inlet F = (20,000 - M)/2.5, but not > 5000 lb	Normal plus upset: Inlet Force = 2872 lb Moment = 12,496 ft-lb
		Exhaust F = (20,000 - M)/0.8, but not > 11,500 lb	Exhaust Force = 2,611 lb Moment = 11,763 ft-lb
Loads:			
Emergency: Design pressure and temperature, deadweight, thermal expansion, and design-basis earthquake		Emergency Inlet F = (30,000- M)/2.5, but not > 7500 lb	Emergency Inlet Force = 4016 lb Moment = 20,153 ft-lb
		Exhaust F = (30,000 - M)/0.8, but not > 17,250 lb	Exhaust Force =4,586 lb Moment = 17,150 ft-lb
Stress limit: Specified by vendor for normal loads; ASME Code, Section VIII, for upset loads; increased 20% for emergency loads			
Turbine mounting bolt (turbine to baseplate)	Vertical and horizontal forces on mounting bolts calculated as the sum of seismic acceleration on the turbine and the pipe reaction forces and moments on the nozzle		

Table 3.7-14 Sheet 3 of 3

STRESS SUMMARY FOR HPCI TURBINE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Loads:			
Normal and upset: Operating-basis earthquake, nozzle loads for operating-basis earthquake, deadweight, and thermal expansion		Tensile and shear stress for bolting materials are specified in ASME Code, Section VIII	By meeting the nozzle load criteria above, the detailed seismic analysis indicate the mounting bolts satisfy the allowable stress requirements
Loads:			
Emergency: Design-basis earthquake, nozzle loads for design-basis earthquake, deadweight, and thermal expansion		(Same as for normal and upset loads above.)	
Stress limit: ASME Code, Section VIII, allowables for normal and upset loads; for emergency loads, 0.9 yield and twice allowable shear			

Table 3.7-15 Sheet 1 of 2

STRESS SUMMARY FOR HPCI PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Closure bolting	Bolting loads and stresses calculated per “Rules for Bolted Flange Connections,” ASME Code, Section VIII, Appendix II	Maximum allowable stress - main pump = 20,000 psi; booster pump = 20,000 psi	Maximum calculated stress - main pump = 16,960 psi; booster pump = 7180 psi
Normal and upset: Design pressure and temperature, design gasket load			
Bolting stress limit: Allowable working stress per ASME Code, Section VIII			
Wall thickness	Per rules of ASME Code, Part UG, Section VIII	Maximum allowable stress - main pump = 14,000 psi; booster pump = 14,000 psi	Maximum calculated stress - main pump = 8700 psi; booster pump = 3360 psi
Loads:			
Normal and upset: Design pressure and temperature	Nozzle stress Volute stress is calculated per Roarke[5]	Main pump = 14,000 psi; booster pump = 14,000 psi	Main pump = 7840 psi; booster pump = 2610 psi
Stress limit: ASME Code, Section III			
Nozzle:	For the moment due to pipe reaction, the maximum force shall not exceed the allowable	Force in lb Moment in ft-lb	

Table 3.7-15 Sheet 2 of 2

STRESS SUMMARY FOR HPCI PUMP

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculation
Loads:		Normal and upset:	Normal and upset:
Normal and upset: Design pressure and temperature, deadweight, thermal expansion, and operating-basis earthquake	Total nozzle stress with these criteria does not exceed stress limits	Suction F = 21,000 - 1.83M	Suction Force = 1738 lb Moment = 3,284 ft-lb
		Discharge F = 23,000 - 3.17M, but not > 11,500 lb	Discharge Force = 947 lb Moment = 4,526 ft-lb
Loads:			
Emergency: Design pressure and temperature, deadweight, thermal expansion, and design-basis earthquake		Emergency: Suction F = 28,750 - 1.83M	Emergency: Suction Force = 2775 lb Moment = 4,813 ft-lb
Stress limit: ASME Code, Section VIII, for normal and upset loads; 1.5 allowable stress for emergency loads		Discharge F = 34,00 - 3.21M	Discharge Force = 1,077 lb Moment = 4,878 ft-lb

Table 3.7-16 Sheet 1 of 4

STRESS SUMMARY FOR MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
<u>Minimum body well thickness</u> <u>Loads</u>: Design pressure and temperature <u>Primary membrane stress limit</u> S = 7000 lb/in^2 per ASA B16.5	Minimum wall thickness in cylindrical portions of the valve shall be calculated using the following formula: $t = 1.5\left[\frac{Pd}{2S - 1.2P}\right] + C$ where: S = allowable stress of 7000 psi P = primary service pressure, 655 psi d = inside diameter of valve at section being considered, in. C = corrosion allowance of 0.12 in.	t = 1.468 in	t = 1.593 in at 18.155 in diameter
<u>Minimum cover thickness</u> <u>Loads</u> Design pressure and temperature, design bolting load, gasket load	$t = G\left[\frac{CP}{S} + \frac{1.78Wh_G}{SG^3}\right]^{1/2} + C_1$ where: t = minimum thickness, in. G = diameter or short span, in.	t = 4.37 in	t = 5.00 in. S_{allow} = 17500 psi

Table 3.7-16 Sheet 2 of 4

STRESS SUMMARY FOR MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Primary stress limit: Allowable working stress per ASME Code, Section VIII UG-32(c)(2) 1971 Edition	C = attachment factor S = allowable stress, psi W = total, bolt load, lb h_G = gaskets, moment arm, in. C_1 = corrosion allowance, in.		
Cover flange bolt area: Loads Design pressure and temperature, gasket load, stem operational load, seismic load (design-basis earthquake) Bolting stress limit: Allowable working stress per ASME Standard Code for Pumps and Valves for Nuclear Power	In accordance with paragraph F.105.2.6 of ANSI B31.7, the maximum value of bolt stress which results from preload operating pressure and differential thermal expansion shall not exceed twice the allowable value listed in Table A.1, except that the stem operational load and seismic loads shall be included in the total load carried by bolts. The horizontal and vertical seismic forces shall be applied at the mass center of the valve operator, assuming that the valve body is rigid and anchored.	S = 67,000 psi at 575°F	A_B = 16.735 in.^{2} S_b = 45,000 psi
Body flange thickness and stress Loads: Design pressure and temperature, gasket load, stem operational load, seismic load (design-basis earthquake) Flange stress limits (S_H, S_R, S_T): The allowable flange stresses are obtained from paragraph UA-52	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections," ASME Code, Section VIII, Appendix II, except that the stem operational load and seismic loads shall be included in the total load carried by the flange. The horizontal and vertical seismic force shall be applied at the mass center of the valve operator, assuming that the valve body is rigid and anchored.	S_H = 26,200 psi S_R = 17,500 psi S_T = 17,500 psi	t = 4.12 in. S_H = 20,300 psi S_R = 7,100 psi S_T = 8,400 psi

Table 3.7-16 Sheet 3 of 4

STRESS SUMMARY FOR MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Valve disk Thickness			
Loads: Design pressure and temperature Primary handling stress limit: ASME Section III 1986 subsection NG-3200		S_m = 18,200 psi 575°F $1.5S_m$ = 27,300 psi 575°F	Cylinder Flange Interface 1,250psi P_L = 6,500 psi $P_L + P_b$ = 11,800 psi Cylinder/Sphere Interface 1,250 psi P_L = 9,000 psi $P_L + P_b$ = 18,800 psi Stem Disk Seating Area 1,250 psi P_L = 7,000 psi $P_L + P_b$ = 27,000 psi Disk Piston Main Seat 1,250 psi P_L = 5,600 psi $P_L + P_b$ = 7,500 psi

Table 3.7-16 Sheet 4 of 4

STRESS SUMMARY FOR MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Valve operator supports:	The valve assembly shall be analyzed assuming that the valve body is an anchored, rigid mass and that the specified vertical and horizontal seismic forces are applied at the mass center of the operator assembly simultaneously with operating pressure plus deadweight plus operational loads		
Loads: Design pressure and temperature, stem operational load, equipment deadweight, seismic load (design-basis earthquake)		S = 90,000 psi	S = 49,500 psi (combined bending and tensile stress)
90% Tensile ASTM minimum yield			

Table 3.7-17 Sheet 1 of 5

STRESS SUMMARY FOR MAIN RELIEF VALVES[a]

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Minimum body wall thickness Loads: Design pressure and temperature Primary membrane stress limit: Allowable working stress as defined by USAS B16.5 (7000 psi at primary service pressure)	$t = 1.5\left[\frac{Pd}{2S - 1.2P}\right] + C$ where: T = minimum required thickness, in. S = allowable stress 7000 psi P = primary service pressure, 655 psi d = inside diameter of valve at section being considered, in. C = corrosion allowance of 0.12 in.	t = 1.253 in.	t = 1.5 in.
Top flange Loads: Design pressure and temperature, gasket load Primary stress limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	Discontinuity analysis based on ASME Code, Section VIII, Paragraph UA 99	S_m = 17,800 psi	σ_r = 7,335 psi σ_r = 727 psi

[a] This table applies to the original Dresser Industries, Inc., main steam relief valves which were replaced in 1977 with valves manufactured by the Target Rock Corporation.

Table 3.7-17 Sheet 2 of 5

STRESS SUMMARY FOR MAIN RELIEF VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Flange bolt area inlet flange, outlet flange, body to top flange Loads: Design pressure and temperature, gasket load, operational load, and design-basis earthquake Bolting stress limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	Total bolting loads and stresses shall be calculated in accordance with procedures of Paragraph 1-704.5.1, Flange Joints, of ANSI B31.7 Nuclear Piping Code A_b = total bolt area in.^{2}	Body to top flange A_b = 10.41 in.^{2} Inlet flange A_b = 9.88 in.^{2} Outlet flange A_b = 5.88 in.^{2}	Body to top flange A_b = 13.86 in.^{2} Inlet flange A_b = 13.86 in.^{2} Outlet flange A_b = 8.8 in.^{2}
Flange thickness - inlet, outlet Loads: Design pressure and temperature, gasket load, operational loads, and design-basis earthquake	Flange thickness and stress shall be calculated in accordance with procedures of Paragraph 1-704.5.1, Flanged Joints, of ANSI B31.7 Nuclear Piping Code		
Flange stress limits (S_H, S_R, S_T): 1.5S_m per ASME Standard Code for Pumps and Valves for Nuclear Power		Inlet flange S_H = 25,950 psi S_R = 25,950 psi S_T = 25,950 psi Outlet flange S_H = 25,950 psi S_R = 25,950 psi S_T = 25,950 psi S_m = 32,000 psi	Inlet flange S_H = 17,404 psi S_R = 6,217 psi S_T = 9,478 psi Outlet flange S_H = 15,128 psi S_R = 14,628 psi S_T = 4,671 psi S = 26,207 psi

Table 3.7-17

STRESS SUMMARY FOR MAIN RELIEF VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Valve disk thickness and stress Loads: Design pressure and temperature Primary stress limit: S_m per ASME Code, Section III, Appendix II	Maximum stress in disk shall be calculated by: $S_{max} = \frac{3P}{4t^2}\left[a^2 - 2b^2 + b^4(m-1) - 4b^4(m+1)\log\frac{a}{b} + \frac{a^2b^2(m+1)}{a^2(m-1)} + b^2(m+1)\right]$ where: t = disk thickness P = design pressure a = outer radius of disk b = radius of center guide m = reciprocal of Poisson's ratio		

Table 3.7-17 Sheet 4 of 5

STRESS SUMMARY FOR MAIN RELIEF VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Inlet nozzle diameter thickness and stress Loads: Design pressure and temperature, operational load, and design-basis earthquake Primary stress limit: 1.5 x allowable stress intensity, $1.5S_m$, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	$S = \frac{F_1 + F_2}{A} + \frac{M_1 + M_2}{2} + \frac{PR_i}{2t}$ where: S = combined bending and tensile stress, psi F_1 = vertical loads due to design pressure, lb F_2 = vertical component of reaction thrust, lb A = cross section areas of nozzle, in.^{2} M_1 = moment resulting from horizontal reaction, in.-lb M_2 = moment resulting from horizontal seismic force at mass center of valve, in.-lb P = design pressure, psi R_1 = inside radius of nozzle T = wall thickness, in.	S = 26,250 psi	S = 6,168 psi

Table 3.7-17 Sheet 5 of 5

STRESS SUMMARY FOR MAIN RELIEF VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Body stresses at neck below top flange	$S = \frac{F_1 + F_2}{A} + \frac{M}{z}$	S = 26,250 psi	S = 705 psi
Loads: Design pressure and temperature, operational load, and design-basis earthquake	where = S = combined bending and tensile strength, psi		
Primary stress limit: 1.5 x allowable stress intensity, $1.5S_m$, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	F_1 = axial load due to design pressure F_2 = axial load due to seismic acceleration on components attached to neck M = moment due to seismic acceleration on components attached to neck		

Table 3.7-18 Sheet 1 of 6

STRESS SUMMARY FOR MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Inlet nozzle wall thickness	$t = \frac{PR}{SE - 0.6P} + C$	t = 0.143 in.	t = 0.62 in.
Loads: 1.1 x design pressure at 600°F Primary membrane stress limit: Allowable stress intensity, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	where: t = minimum required thickness in. S = allowable stress, psi P = 1.1 x design pressure, psi R = internal radius, in. E = joint efficiency C = corrosion allowable, in.		
Valve disk thickness	$S_s = \frac{W}{A} = \frac{PA_1}{A}$	S_S = 20,190 psi	S = 14,351 psi
Loads: 1.1 x design pressure at 600°F Diagonal shear stress limit: 0.6 x allowable stress intensity, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	where: W = shear load, lb A = shear area, in.^{2} P = 1.1 x design pressure, psi A_1 = side area, in.^{2}		

Table 3.7-18 Sheet 2 of 6

STRESS SUMMARY FOR MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
	and: $A = \pi S (R+R^1)$ S = slope of frustrum of shear cone, in. R = radius at base of cone, in. R^1 = radius at top of cone, in.		
Inlet flange bolt area Loads: Design pressure and temperature, gasket load, operational load, and design-basis earthquake Bolting stress limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	Total bolting loads and stresses shall be calculated in accordance with procedures of Paragraph 1-704.5.1, Flange Joints, of ANSI B31.7 Nuclear Piping Code	S_b = 27,700 psi	S_b = 17,296 psi
Inlet flange thickness			
Loads: Design pressure and temperature, gasket load, and seismic load (design-basis earthquake)	Flange thickness and stress shall be calculated in accordance with procedures of Paragraph 1-704.5.1, Flanged Joints, of ANSI B31.7, Nuclear Piping Code	S_H = 27,300 psi S_R = 27,300 psi S_T = 27,300 psi	S_H = 21,363 psi S_R = 10,811 psi S_T = 4,584 psi

Table 3.7-18 Sheet 3 of 6

STRESS SUMMARY FOR MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Flange stress limits (S_H, S_R, S_T): $1.5S_m$ per ASME Standard Code for Pumps and Valves for Nuclear Power			
Valve spring - torsional stress	$S_{max} = \frac{8PD}{\pi d^3}\left[\frac{4C-1}{4C-4} + \frac{0.615}{C}\right]$	Setpoint S= 84,000 psi Maximum lift S = 112,500 psi	Setpoint S = 52,616 psi Maximum lift S = 91,747 psi
Loads: W_1 = setpoint load, lb W_2 = spring load at maximum lift, lb Torsional stress limit: 0.67 x torsional elastic limit when subject to a load of W_1; 0.90 x torsional elastic limit when subjected to a load of W_2	where: S_{max} = torsional stress, psi P = W_1 or W_2 = spring load, lb D = mean diameter of coil, in. d = diameter of wire, in. $C = \frac{D}{d}$ = correction factor		
Yoke rod area	$A = 2\frac{F}{S_m}$	A = 0.575 in.^{2}	A = 1.402 in.^{2}
Loads: Spring loads at maximum lift			

Table 3.7-18 Sheet 4 of 6

STRESS SUMMARY FOR MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Primary stress limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power	where: A = required area per rod, in.2 F = total spring load, lb S_m = allowable stress, psi		
Yoke bending and shear stresses	$S_b = \frac{M}{Z} S_s = \frac{V}{A}$	S_b = 17,800 psi S_s = 10,700 psi	S_b = 12,314 psi S_s = 2,998 psi
Loads: Spring load at maximum lift			
Bending and shear stress limits: Bending - allowable stress intensity, S_m, per ASME Standard Code for Pumps and Valves for Nuclear Power; shear - 0.6 x allowable stress intensity, 0.6 S_m, per ASME Standard Code for Pumps and Valves for Nuclear Power	where: S_b = bending stress, psi S_s = shear stress, psi M = bending moment, in.-lb Z = section modulus, in.^{3} V = vertical shear, lb A = shear area, in.^{2}		
Minimum body wall thickness	$t = 1.5\left[\frac{Pd}{2S - 1.2P}\right] + C$	t = 0.311 in	Body bowl t = 0.562 in.
Loads: Primary service pressure	where: t = required thickness, in.	t = 0.218 in	Inlet nozzle t = 1.228 in.

Table 3.7-18 Sheet 5 of 6

STRESS SUMMARY FOR MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Primary Stress limit: Allowable stress, 7000 psi, in accordance with ASA B16.5	S = allowable stress, 7000 psi P = primary service pressure, 150 psi d = inside diameter of valve at section being considered, in.	t = 0.250 in.	Outlet nozzle t = 0.5625 in.
Inlet nozzle combined stress	$S = \frac{F_1 + F_2}{A} + \frac{M_1 + M_2}{Z}$	S = 27,300 psi	S = 5,159 psi
Loads: Spring load at maximum lift operational load seismic load - DBE Combined stress limit: 1.5 x allowable stress intensity, $1.5S_m$, per ASME Standard Code for Pumps and Valves for Nuclear Power	where: S = combined bending and tensile stress, psi F_1 = maximum spring load, psi F_2 = vertical component of reaction thrust, lb A = cross section area of nozzle, in.^{2} M_1 = moment resulting from horizontal component of reaction, in.-lb M_2 = moment resulting from horizontal seismic force, in.-lb		

Table 3.7-18 Sheet 6 of 6

STRESS SUMMARY FOR MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
Spindle diameter	$F_c = \frac{\pi^2 EI}{L^2}$	Actual load F = 34,490 lb	Load limit ($0.2F_c$) F = 85,900 lb
Loads: Spring load at maximum lift	where: Fc = critical buckling load, lb E = modulus of elasticity, psi I = moment of inertia, $in.^4$ L = length of spindle in. compression, in.		
Spring washer shear area	$S_s = \frac{F}{A}$	S_s = 15,960 psi	S_s = 2,430 psi
Loads: Spring load at maximum lift			
Shear stress limit 0.6 x allowable stress intensity, $0.6S_m$, per ASME Standard Code for Pumps and Valves for Nuclear Power	where: S_s = shear stress, psi F = spring load, lb A = shear area, $in.^2$		

Table 3.7-19 Sheet 1 of 4

STRESS SUMMARY FOR RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
Minimum casing wall thickness Loads: Normal and upset condition, design pressure and temperature Primary membrane stress limit: Allowable working stress per ASME Code, Section III, Class C	$t = \left[\frac{PR}{SE - 0.6P}\right] + C$ where: t = minimum required thickness, in. P = design pressure, psig R = maximum internal radius, in. S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in.	s = 1,5075 psi at 575°F t = 2.12 in.	t = 2.500 in.
Minimum casing cover thickness Loads: Normal condition, design pressure and temperature Stress limit: Allowable working stress per ASME Code, Section III, Class C	$S_s = \frac{F}{A}$ where: S_s = shear stress F = total upward force, in.-lb A = area $S_s = \frac{S_t}{\sqrt{3}}$ S_t = allowable tensile strength	S_s (design) = 3531 psi	S_t = 15,075 psi at 575°F S_t = 8,750 psi at 575°F

Table 3.7-19 Sheet 2 of 4

STRESS SUMMARY FOR RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
Cover and seal flange bolt areas Loads: Normal and upset conditions, design pressure and temperature, design gasket load	Bolting loads, areas, and stresses shall be calculated in accordance with "Rules for Bolted Flanged Connections," ASME Code, Section VIII, Appendix II	Cover flange bolts W_{mi} = 1,431,000 lb A_{mi} = 71.6 in.^{2} S_b = 20,000 psi	 A_m = 84.2 in.^{2} S_b = 20,000 psi
Bolting stress limit: Allowable working stress per ASME Code, Section III, Class C		Seal flange bolts W_{mi} = 141,000 lb A_{mi} = 7.05 in.^{2} S_b = 20,000 psi	 A_m = 8.82 in.^{2} S_b = 20,000 psi
Cover clamp flange thickness Loads: Normal and upset conditions, design pressure and temperature, design gasket load, design bolting load	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections," ASME Code, Section VIII, Appendix II	Flange thickness and stress M_o = 6,240,750 in.-lb S_t = 17,500 psi t = 6.36 in.	t = 8-1/4 in. S = 17,500 psi
Tangential flange stress limit: Allowable working stress per ASME Code, Section III, Class C			

Table 3.7-19 Sheet 3 of 4

STRESS SUMMARY FOR RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
Pump nozzle membrane and bending stress	$SL = \frac{\pi D^2 P}{4A} + \frac{M}{Z} + \frac{F}{A}$	S_L = 7,002 psi S_c = 17,200 psi	S_m = 15,075 psi $1.5S_m$ = 22,112 psi
Loads: Normal and upset condition, design pressure, and temperature, piping reactions during normal operation	$S_c = \frac{PD}{2t}$ $S_s = \frac{tR_o}{J}$	S_s = 0.001 psi S = 12,100 psi	
Combined stress limits: $1.5S_m$, per ASME Standard Code for Pumps and Valves for Nuclear Power, Class 1	$S = \frac{S_L + S_C}{2} + \left[\left(\frac{S_L - S_C}{2}\right)^2 + S_s^2\right]^{1/2}$ where: S_L = longitudinal stress, psi S_C = circumferential stress, psi S_s = torsional stress, psi D = nozzle internal diameter, in. P = design pressure, psi A = nozzle cross-section metal areas, in.^{2}		

Table 3.7-19 Sheet 4 of 4

STRESS SUMMARY FOR RECIRCULATION PUMPS

Criteria	Method of Analysis	Analytical Results	Allowable Stress or Actual Thickness
Pump nozzle membrane and bending stress (continued)	M = maximum bending moment, in.-lb F = maximum longitudinal, force, lb t = nozzle wall thickness, in. J = polar moment of inertia, $in.^4$ R_o = nozzle outside radius, in Z = polar section modulus, $in.^3$		
Mounting bracket combined stress Loads: Flooded weight, DBE horizontal seismic force - 1.5g; DBE vertical seismic force - 0.14g Combined stress limit: Yield stress	Bracket vertical loads shall be determined by summing the equipment and fluid weights and vertical seismic forces. Bracket horizontal loads shall be determined by applying the specified seismic force at mass center of pump-motor assembly (flooded). Horizontal and vertical loads shall be applied simultaneously to determine tensile, shear, and bending stresses in the brackets. Tensile, shear, and bending stresses shall be combined to determine maximum combined stresses.	Maximum combined stress Shear stress (vertical) Lug 1 S = 2180 psi Lug 2 S = 2180 psi Lug 3 S = 4000 psi Shear stress (horizontal) Lug 1 S = 2960 psi Lug 2 S = 2960 psi Lug 3 S = 2960 psi	S_s = 8,650 psi

Table 3.7-20 Sheet 1 of 3

STRESS SUMMARY FOR RECIRCULATION VALVES

Criteria	Method of Analysis	Calculated Stress or Thickness	Allowable Stress or Minimum Wall Thickness
Minimum body wall thickness	$t = \dfrac{1.5Pd}{2S - 2P \times (1-y)} + 0.1$	4-in.valve t = .900 in.	4-in.valve t = 0.405 in.
Loads: Design pressure and temperature Primary membrane stress limit: Allowable working stress per ASME Code, Section VIII; USAS B31.1.0, 1967; manufacturer's standards MSS-Sp66	where: t = minimum wall thickness, in. P = design pressure, psig d = minimum diameter of flow passage, but not less than 90% of the inside diameter at welding end, in. S = allowable working stress, psi y = plastic stress distribution factor, 0.4	22x18x22 in. valve t = 1.630 in. (discharge and suction)	22x18x22 in. valve t = 1.520 in.
Body-to-bonnet bolt area			
Loads: Design pressure and temperature, gasket load, stem operational load, seismic load (DBE) Bolting stress limit: Allowable working stress per ASME Code, Section VIII, 1968	Total bolting loads and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections," ASME Code, Section VIII, load shall be applied at the mass center of the valve operator, assuming that the valve body is rigid and anchored	4-in. valve A_b = 9.29 in.^{2} S_b = 10,613 psi 22x18x22 in. valve (suction and discharge) A_b = 47.52 in.^{2} S_b = 15.777 psi	4-in.valve Actual bolt area = 10.24 in.^{2} S_{allow} = 20,000 psi 22x18x2 in. valve Actual bolt area S_{allow} = 20,000 psi

Table 3.7-20 Sheet 2 of 3

STRESS SUMMARY FOR RECIRCULATION VALVES

Criteria	Method of Analysis	Calculated Stress or Thickness	Allowable Stress or Minimum Wall Thickness
Flange thickness and stress Loads: Design pressure and temperature, gasket load , stem operational load, seismic load (DBE) Flange stress limit (S_H, S_R, S_T): 1.5S_m, per ASME Standard Code for Pumps and Valves for Nuclear Power, Class 1	Flange thickness and stress shall be calculated in accordance with “Rules for Bolted Flange Connections,” ASME Code, Section VIII, Appendix II, except that the stem operational load and seismic load shall be carried by the flange. The horizontal and vertical seismic forces shall be applied at the mass center of the valve operator, assuming that the valve body is rigid	22x18x22 in. valve (discharge and suction) S_H = 15,641 psi S_R = 10,997 psi S_T= 7,821 psi 4-in. discharge bypass S_H = 13,408 psi S_R = 6,303 psi S_T = 11,935 psi	 20,139 psi 13,426 psi 13,426 psi
Valve disk thickness Loads: Design pressure and temperature	$t_a 0.816R\sqrt{\frac{Pd}{S_m}}$ $S_a = \frac{P_d(0.816 \times R)^2}{t_A}$	22x18x22 in. valve (discharge) t_A = 2.018 in. S_A = 1,904 psi	22x18x22 in. valve t_a = 2.018 in. S_a = 1,904 psi
Primary bending stress limit: 1.5S_m, per ASME, Standard Code for Pumps and Valves for Nuclear Power, Class 1	where: t_a = calculated disk thickness S_a = design stress allowable t_A = actual disk thickness	22x18x22 in. valve (suction) t_A = 2.009 in. S_A = 14,031 psi 4-in. valve t_A = 0.550 in. S_A = 14,366 psi	22x18x22 in. valve t_A = 1.871 in. S_a = 15,904 psi 4-in. valve t_A = 0.523 in. S_a = 15,904 psi

Table 3.7-20 Sheet 3 of 3

STRESS SUMMARY FOR RECIRCULATION VALVES

Criteria	Method of Analysis	Calculated Stress or Thickness	Allowable Stress or Minimum Wall Thickness
Valve disk thickness (continued)	S_A = allowable stress R = Radius of disk P_d = design pressure		
Valve operator supports Loads: Design pressure and temperature, steam operational load, seismic load (DBE) Yoke and yoke bolt stress limits: Allowable working stress per ASME Code, Section VIII	The valve assembly is analyzed assuming that the valve body is an anchored, rigid mass and that specified vertical and horizontal seismic forces area applied at the mass center of the operator assembly simultaneously with operating pressure plus deadweight plus operational loads. These loads are used to determine stresses and deflections for the operator support components	Operator support bolt stress S_b = bolt stress (psi) 22-in. (suction and discharge) S_b = 20,602 psi 4-in. bypass S_b = 10,622 psi	S_b allowable = 20,000 psi

Table 3.7-21

STRESS SUMMARY FOR MAIN STEAM LOOP "A"

Criteria	Method of Analysis	Results of Analysis[a] (psi) Maximum Stress	Stress Limit
Stress for all normal and upset loading must not exceed the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure and deadweight must be less than hot allowable stress.	6,421	17,500
	The sum of the thermal expansion stress intensity range plus anchor displacement stresses caused by the OBE must be less than the allowable stress range for expansion stresses.	7,144	26,250
	The sum of longitudinal stresses due to pressure, deadweight, inertia effects of the OBE, and external loads must be less than 1.2 times the hot allowable stress.	16,183	21,000
	The sum of the longitudinal stresses due to pressure and deadweight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	11,524	43,750
Primary stress for all load combinations that have a very low probability of occurrences must not exceed 1.5 times the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0		
	The sum of the longitudinal stresses due to pressure, deadweight, inertia effects of the DBE, and external forces must be less than 1.8 times the hot allowable stress.	23,006	31,500
	The sum of the longitudinal stresses due to minimum pressure, deadweight, inertia effects of the OBE, and external forces must be less than 1.8 times the hot allowable stress.	16,546	31,500

[a] All limits were met. Highest stresses are given.

Table 3.7-22

STRESS SUMMARY FOR MAIN STEAM LOOP "B"

Criteria	Method of Analysis	Results of Analysis[a] (psi) Maximum Stress	Stress Limit
Stress for all normal and upset loadings must not exceed the limits of ANSI B31.1.0.	Effects from the following loading combination determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure and deadweight must be less than the hot allowable stress.	5,572	17,500
	The sum of the thermal expansion stress intensity range plus anchor displacement stresses caused by the OBE must be less than the allowable stress range for expansion stresses.	8,773	26,250
	The sum of the longitudinal stresses due to pressure, deadweight, inertia effects of the OBE, and external loads must be less than 1.2 times the not allowable stress.	15,382	21,000
	The sum of the longitudinal stresses due to pressure and deadweight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	10,581	43,750
Primary stress for all load combinations that have a very low probability of occurrences must not exceed 1.5 times the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure, deadweight, inertia effects of DBE, and external forces must be less than 1.8 times the hot allowable stress.	20,575	31,500
	The sum of the longitudinal stresses due to minimum pressure, deadweight, inertia effects of the OBE, and external forces must be less than 1.8 times the hot allowable stress.	15,745	31,500

[a] All limits were met. Highest stresses are given.

Table 3.7-23

STRESS SUMMARY FOR MAIN STEAM LOOP "C"

Criteria	Method of Analysis	Results of Analysis[a] (psi) Maximum Stress	Stress Limit
Stress for all normal and upset loadings must not exceed the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to the pressure and deadweight must be less than the hot allowable stress.	5,428	17,500
	The sum of the thermal expansion stress intensity range plus anchor displacement stresses caused by the OBE must be less than the allowable stress range for expansion stresses.	5,692	26,250
	The sum of the longitudinal stresses due to pressure, deadweight, inertia effects of the OBE, and external loads must be less than 1.2 times the hot allowable stress.	18,010	21,000
	The sum of the longitudinal stresses due to pressure and deadweight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	9,925	43,750
Primary stress for all load combinations that have a very low probability of occurrences must not exceed 1.5 times the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure, deadweight, inertia effects of DBE, and external forces must be less than 1.8 times the hot allowable stress.	24,661	31,500
	The sum of the longitudinal stresses due to minimum pressure, deadweight, inertia effects of the OBE, and external forces must be less than 1.8 times the hot allowable stress	18,373	31,500

[a] All limits were met. Highest stresses are given.

Table 3.7-24

STRESS SUMMARY FOR MAIN STEAM LOOP "D"

Criteria	Method of Analysis	Results of Analysis[a] (psi) Maximum Stress	Stress Limit
Stress for all normal and upset loadings must not exceed the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure and deadweight must be less than the hot allowable stress.	6,402	17,500
	The sum of the thermal expansion stress intensity range plus anchor displacement stresses caused by the ODE must be less than the allowable stress range for expansion stresses.	7,094	26,250
	The sum of the longitudinal stresses due to pressure, deadweight, inertia effects of the ODE, and external loads must be less than 1.2 times the hot allowable stress.	15,227	21,000
	The sum of the longitudinal stresses due to pressure and deadweight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	11,567	43,750
Primary stress for all load combinations that have a very low probability of occurrences must not exceed 1.5 times the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure, deadweight, inertia effects of the DBE, and external forces must be less than 1.8 times the hot allowable stress.	20,217	31,500
	The sum of the longitudinal stresses due to minimum pressure, deadweight, inertia effects of the OBE, and external forces must be less than 1.8 times the hot allowable stress.	15,590	13,500

[a] All limits were met. Highest stresses are given.

Table 3.7-25

STRESS SUMMARY FOR RECIRCULATION LOOP "A"

Criteria	Method of Analysis	Results of Analysis[a] (psi) Maximum Stress	Stress Limit
Stress for all normal and upset loadings must not exceed the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure and deadweight must be less than the hot allowable stress.	8,828	14,425
	The sum of the thermal expansion stress intensity range plus anchor displacement stresses caused by the OBE must be less than the allowable stress range for expansion stresses.	21,616	27,050
	The sum of the longitudinal stresses due to pressure, deadweight, and inertia effects of the OBE must be less than 1.2 times the hot allowable stress.	15,519	17,310
	The sum of the longitudinal stresses due to pressure and deadweight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	25,078	41,470
Primary stress for all load combinations that have a very low probability of occurrences must not exceed 1.5 times the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The summary of the longitudinal stresses due to pressure, deadweight, and inertia effects of the DBE must be less than 1.8 times the hot allowable stress.	22,828	25,965
	The sum of the longitudinal stresses due to minimum pressure, deadweight, and inertia effects of the OBE must be less than 1.8 times the hot allowable stress.	16,623	25,965

[a] All limits were met. Highest stresses are given.

Table 3.7-26

STRESS SUMMARY FOR RECIRCULATION LOOP "B"

Criteria	Method of Analysis	Results of Analysis[a] (psi) Maximum Stress	Stress Limit
Stress for all normal and upset loadings must not exceed the limits of ANSI B31.1.0.	Effects from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure and deadweight must be less than the hot allowable stress.	8,581	14,425
	The sum of the thermal expansion stress intensity range plus anchor displacement stresses caused by the OBE must be less than the allowable stress range for expansion stresses.	25,298	27,050
	The sum of the longitudinal stresses due to pressure, deadweight, and inertia effects of the OBE must be less than 1.2 times the hot allowable stress.	16,713	17,310
	The sum of longitudinal stresses due to pressure and deadweight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	28,673	41,470
Primary stress for all load combinations that have a very low probability of occurrences must not exceed 1.5 times the limits of ANSI B31.1.0.	Effect from the following loading combinations determined in accordance with rules of ANSI B31.1.0.		
	The sum of the longitudinal stresses due to pressure, deadweight, and inertia effects of the DBE must be less than 1.8 times the hot allowable stress.	24,418	25,965
	The sum of the longitudinal stresses due to minimum pressure, deadweight, and inertia effects of the OBE must be less than 1.8 times the hot allowable stress.	17,816	25,965

[a] All limits were met. Highest stresses are given.

Table 3.7-27

REACTION LOADS
RECIRCULATION INLET NOZZLE

	Forces (kips)			Moments (in.-kips)		
	Fx	Fy	Fz	Mx	My	Mz
Total thermal, weight, and seismic loads	5.1	13.8	3.9	543	265	292
External mechanical (weight and seismic) loads only	1.1	7.3	2.3	250	96	47
Hydraulic loads applied to thermal sleeve	0	3.8	14.3	-44	0	0

Notes:
1. Forces Fx and Fy are located 118.5 in. from vessel axis.
2. Nozzle loads may be in either the positive or negative direction.
3. All values are at design temperature and pressure.
4. These nozzle loads are also listed in APED-B11-232, APED-B11-235, and APED-B11-001<7>.

Table 3.7-28

SAMPLE CHART
FIRST MODE GREATER THAN SPECTRUM PEAK METHOD

Pipe Size, Span, and Load[a]	Insulation Class and Temperature (°F)				
	No Insulation Ambient	IV & V 125-150	III 251-350	II 351-500	I 501-750
1/2-in. pipe					
Span (ft-in.)	5-1	4-8	4-8	4-2	4-2
Load (lb)	7.0	8.5	8.5	11.5	11.5
3/4-in. pipe					
Span (ft-in.)	5-8	5-4	5-4	4-9	4-9
Load (lb)	12.0	14.0	14.0	17.0	17.0
1-in. pipe					
Span (ft-in.)	6-5	6-1	6-1	5-7	5-7
Load (lb)	20.0	22.0	22.0	26.0	26.0
1-1/2-in. pipe					
Span (ft-in.)	7-8	7-4	7-4	6-11	6-11
Load (lb)	42.0	45.0	45.0	51.0	51.0
2-in. pipe					
Span (ft- in.))	8-6	8-2	8-0	7-9	7-7
Load(lb)	72.0	75.0	78.0	81.0	85.0

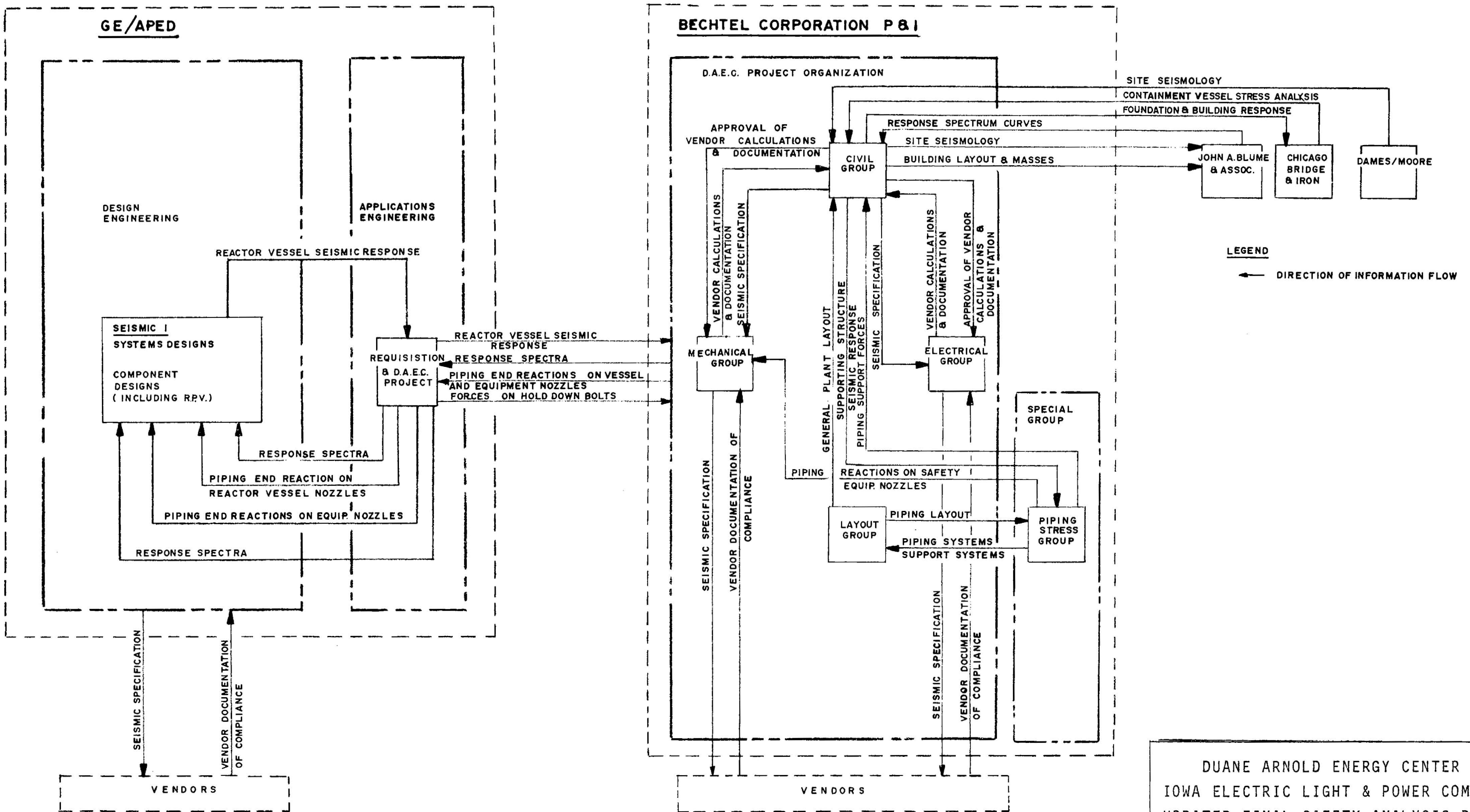
[a] Load is for each support.

Table 3.7-29

SAMPLE CHART
MODIFIED SPECTRUM METHOD

Pipe Size, Span, and Load[a]	Elevation (ft) Below 195	Below 165	Below 135
1/2-in. pipe			
Span (ft-in.)	4-6	4-9	5-0
Load (lb)	5.0	5.0	4.5
3/4-in. pipe			
Span (ft-in.)	5-0	5-6	5-6
Load (lb)	8.0	7.0	7.0
1-in. pipe			
Span (ft-in.)	6-0	6-6	6-9
Load (lb)	9.0	8.0	8.0
1-1/2-in. pipe			
Span (ft-in.)	7-0	7-3	7-6
Load (lb)	20.0	20.0	18.0
2-in. pipe			
Span (ft-in.)	9-0	9-6	10-0
Load (lb)	50.0	55.0	58.0

[a] Load is for each support.



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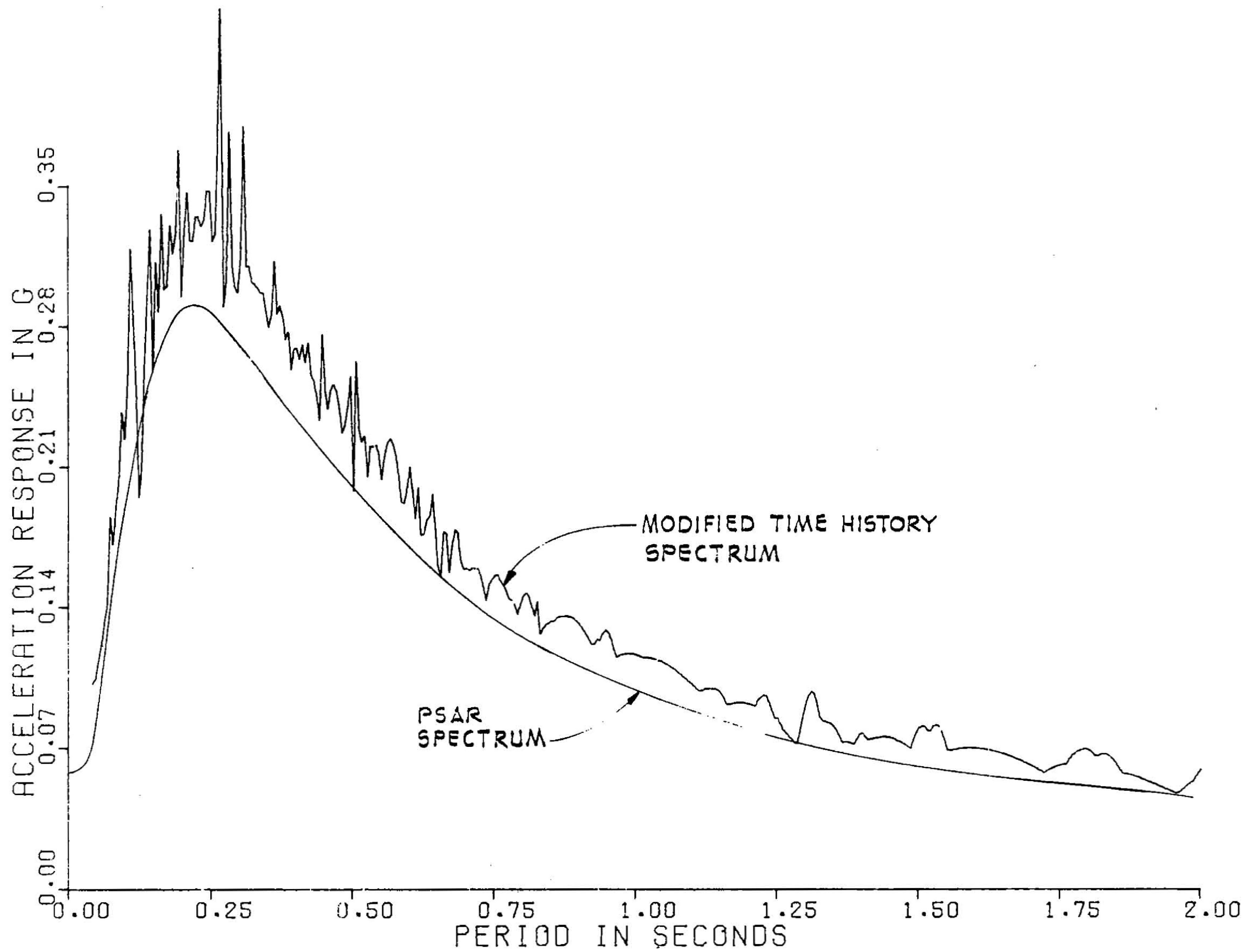
Seismic Design Interrelationships

Figure 3.7-1



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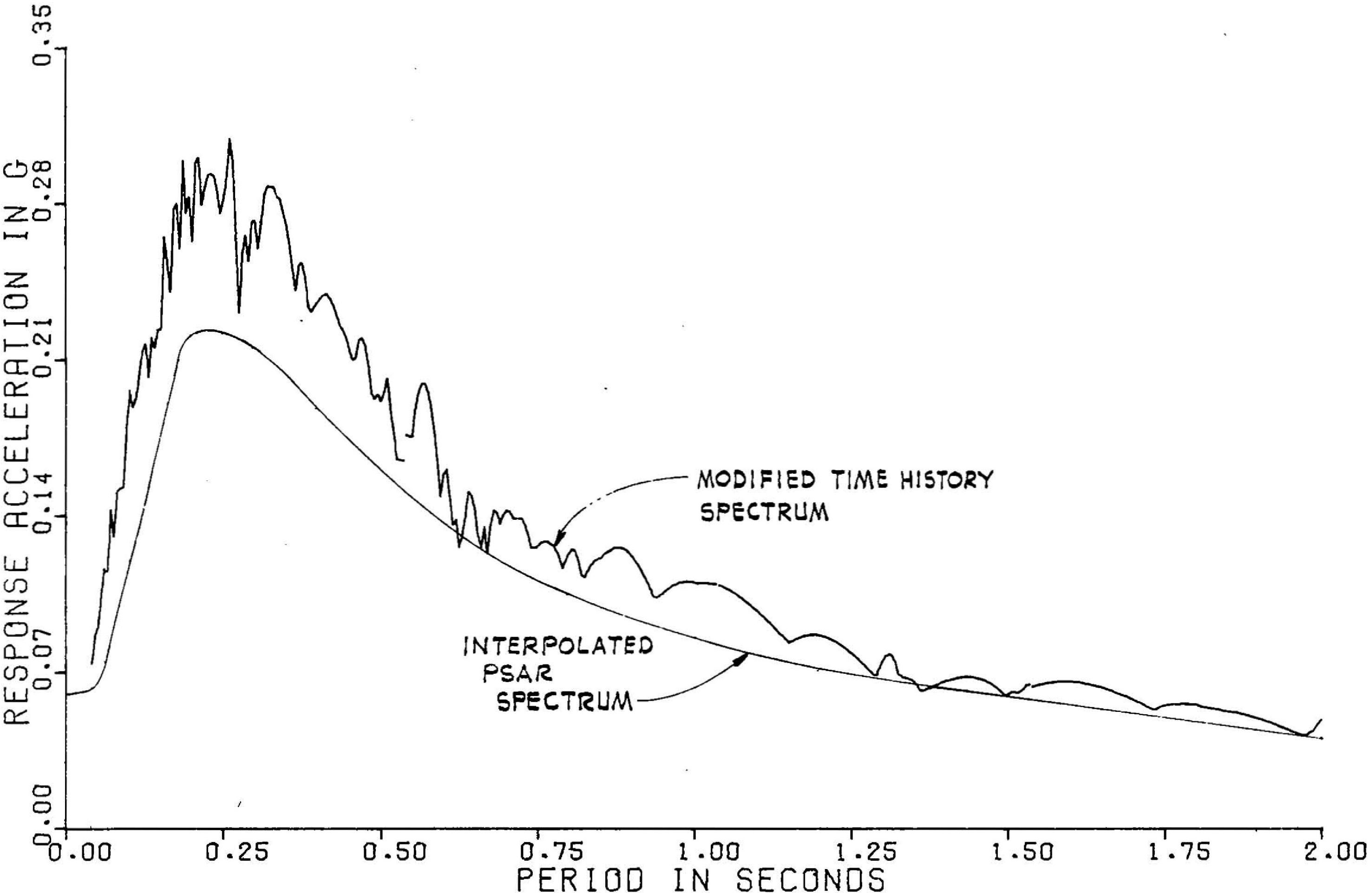
Ground Motion Equation Normalized to 0.06 G Response Spectra, Damping = 0.005

Figure 3.7-2



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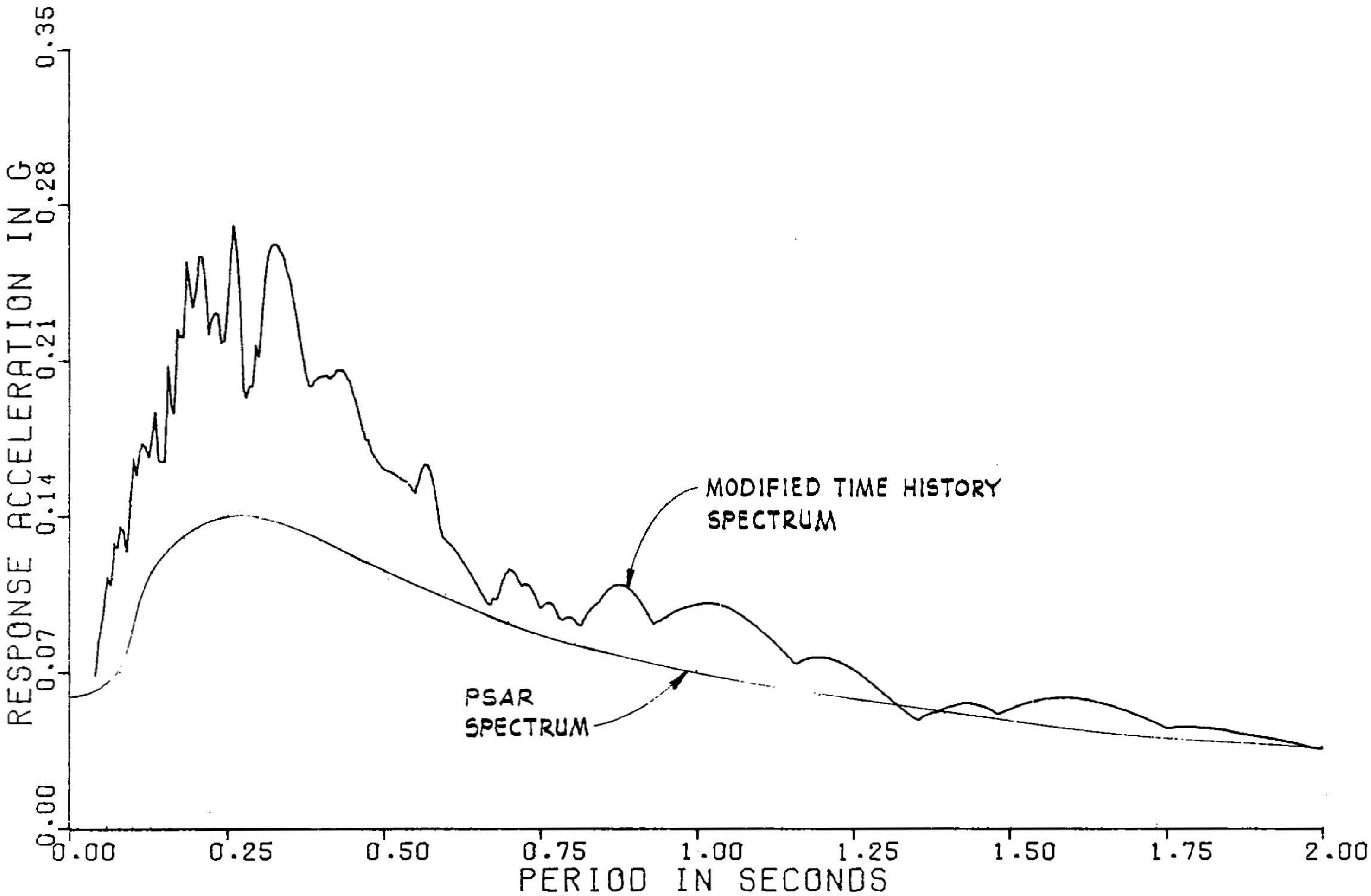
Ground Motion Equation Normalized to 0.06 G Response Spectra, Damping = 0.010

Figure 3.7-3



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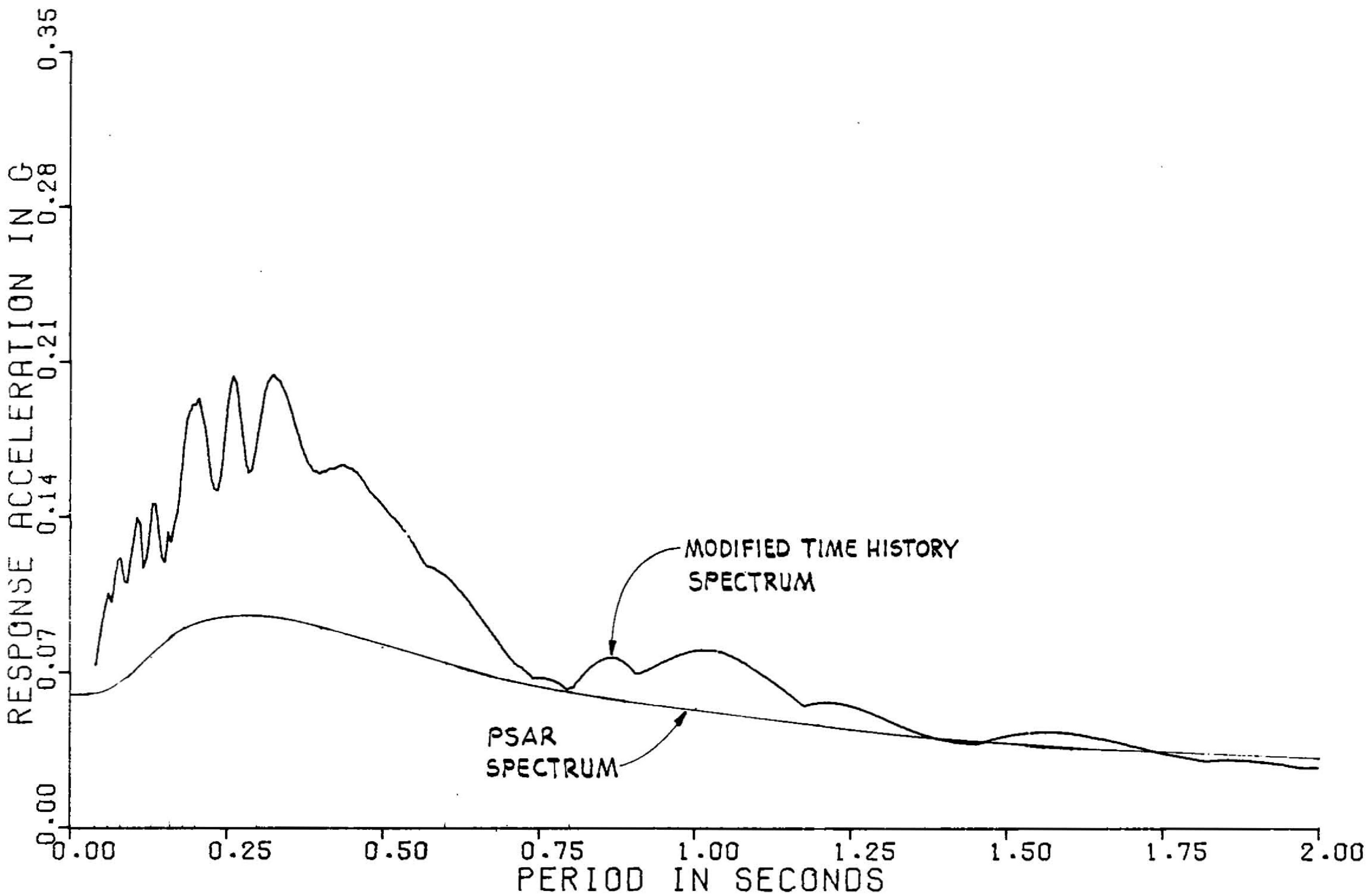
Ground Motion Equation Normalized to 0.06 G Response Spectra, Damping = 0.020

Figure 3.7-4



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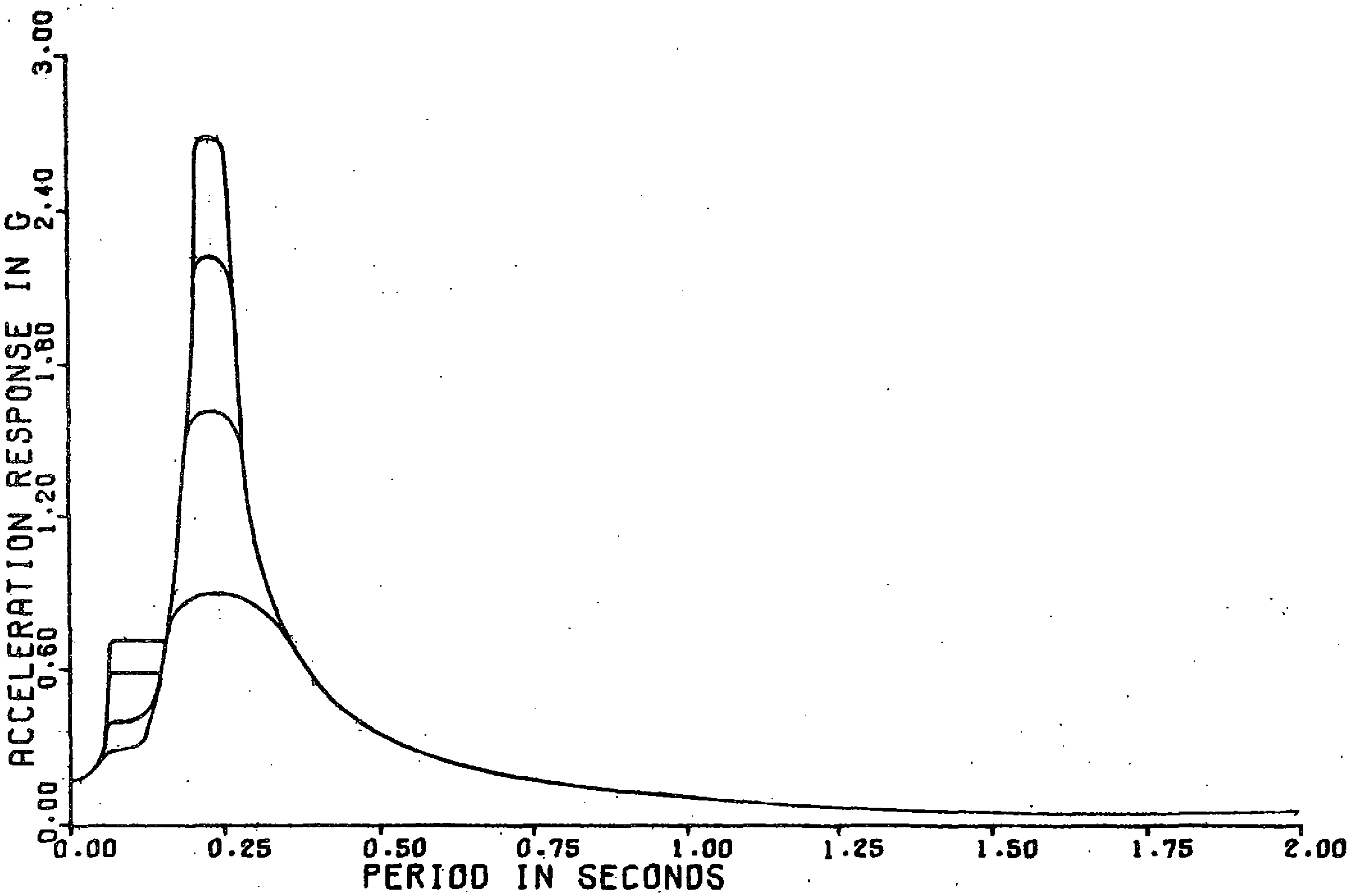
Ground Motion Equation Normalized to 0.06 G Response Spectra, Damping = 0.050

Figure 3.7-5

[redacted]



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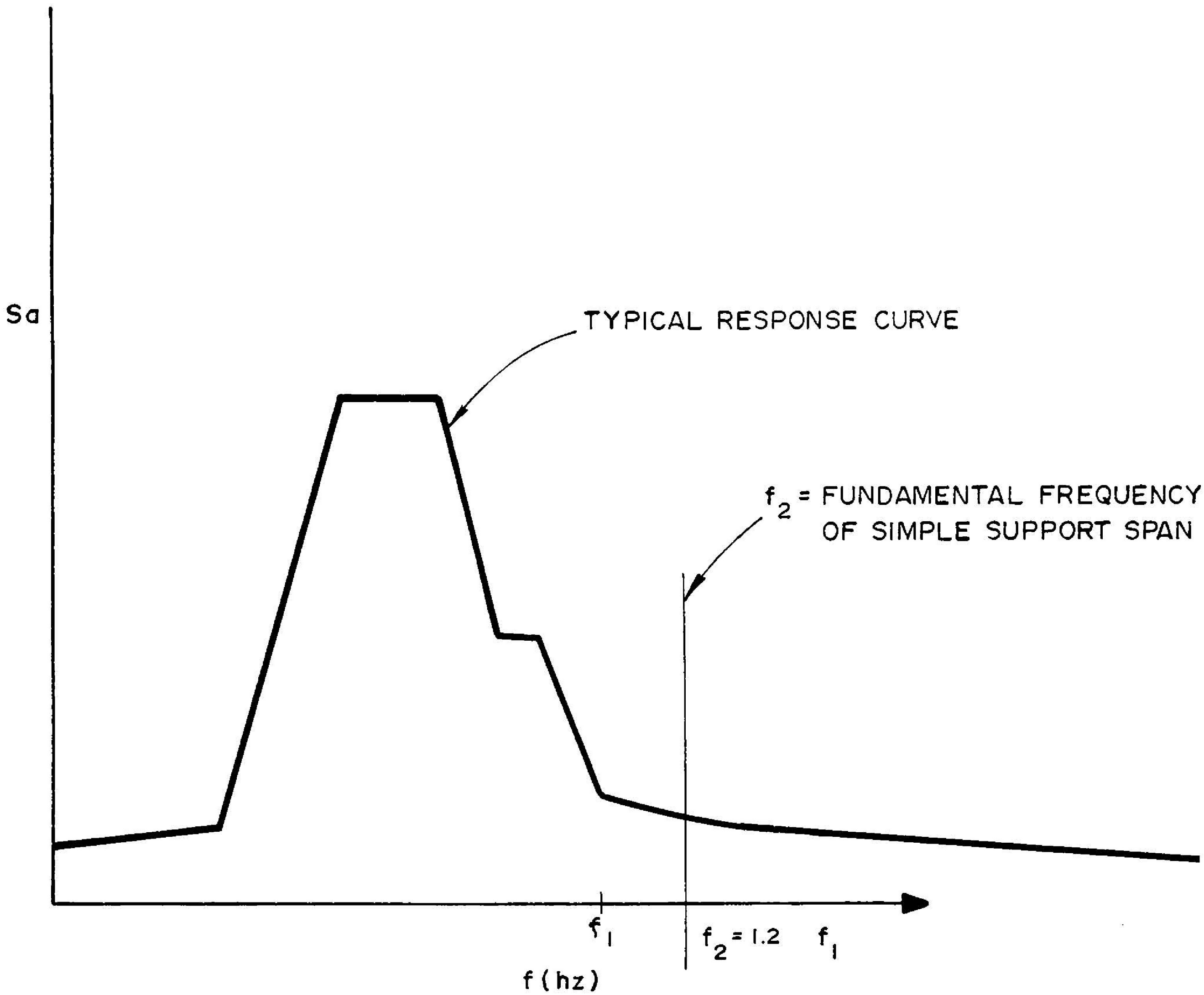
Example of Building Analysis
Horizontal OBE Response Spectra
Reactor Building, Elev. [redacted]
Damping – 0.005, 0.010, 0.020, 0.050

Figure 3.7-7

[redacted]

[redacted]



PEAK OF RESPONSE CURVE METHOD

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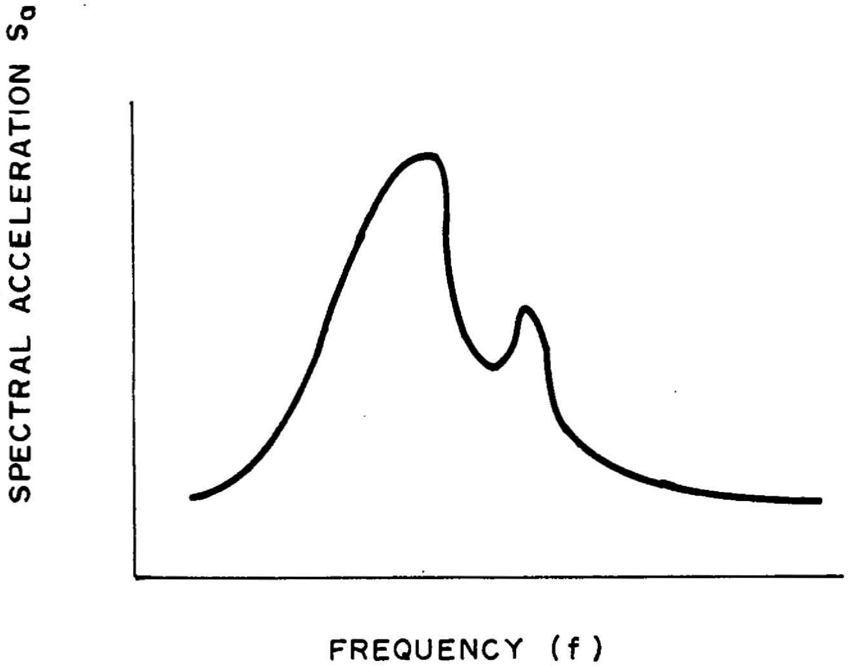
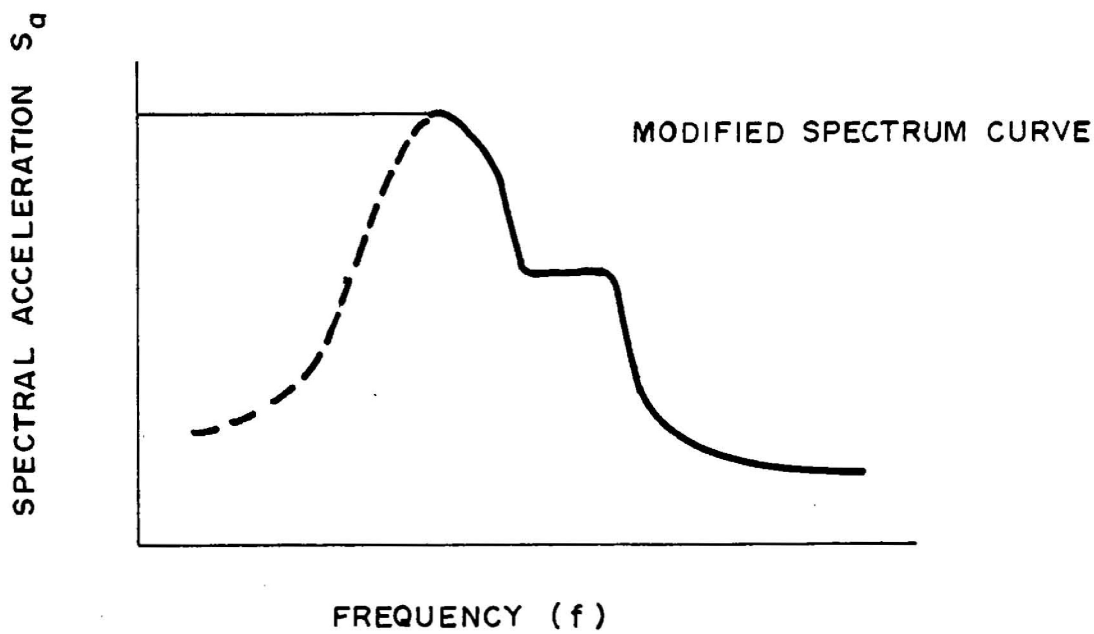
Peak of Response Curve Method

Figure 3.7-10



TYPICAL SPECTRUM CURVE



MODIFIED SPECTRUM CURVE

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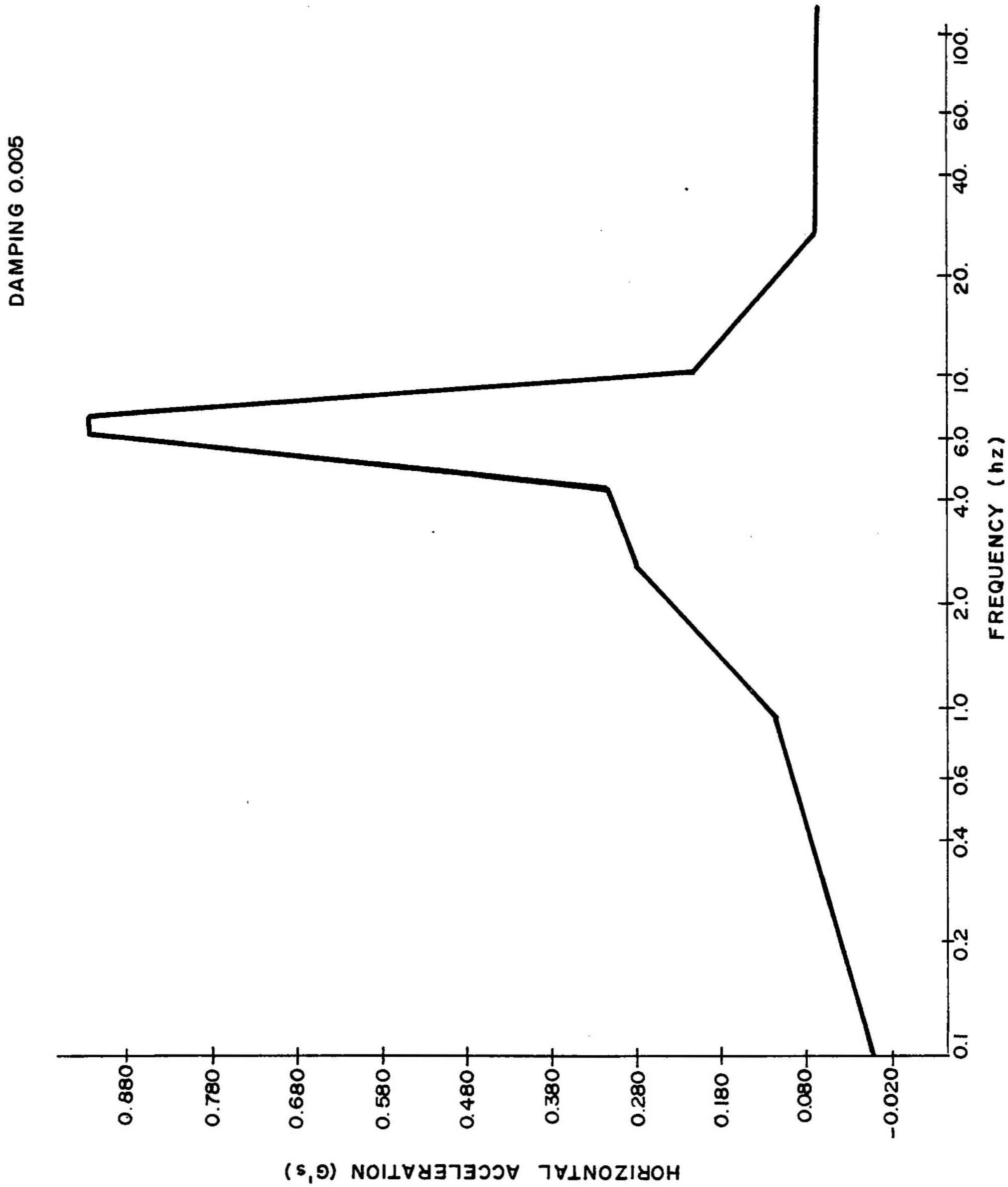
Modified Spectrum Curve

Figure 3.7-11



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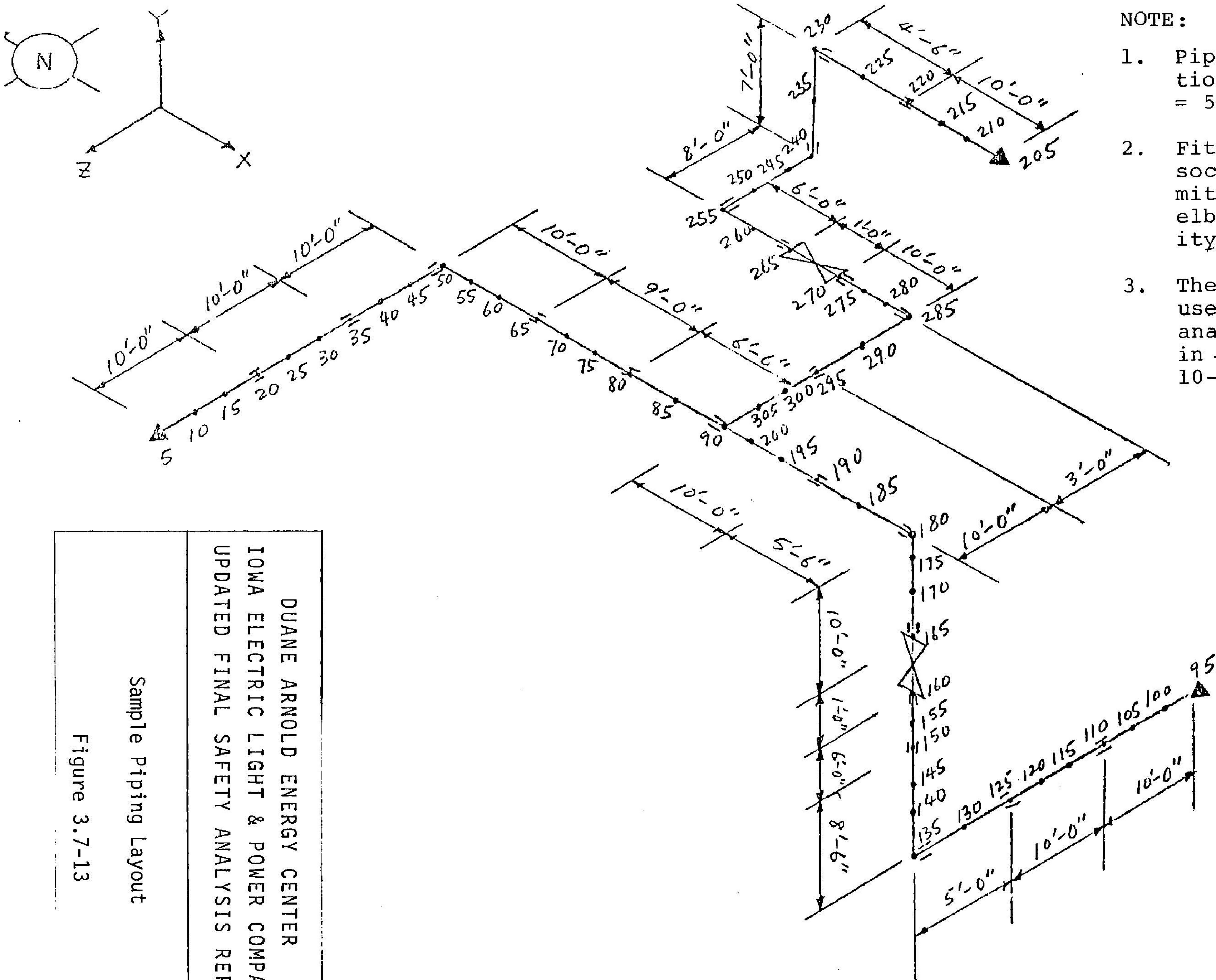
Acceleration Spectrum vs. Frequency

Figure 3.7-12



NOTE:

1. Pipe weight (insulation included) = 5.086 lbs/ft.

2. Fittings are either socket welded or mitre bend (for elbow). Flexibility factor is one.

3. The spectrum curve used for dynamic analysis is shown in Figure 10-C1.16-4.

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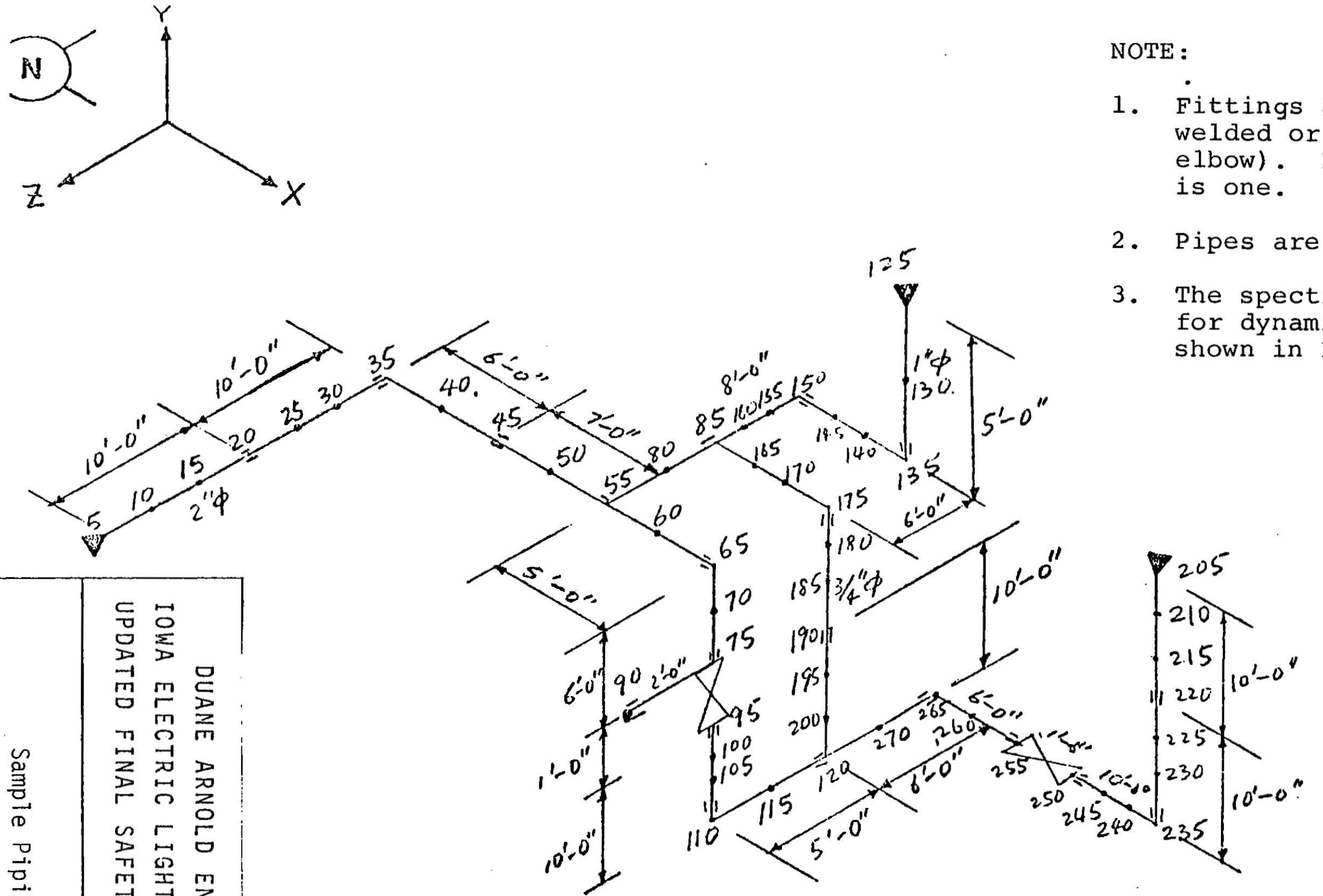
Sample Piping Layout

Figure 3.7-13



NOTE:

1. Fittings are either socket welded or mitre bend (for elbow). Flexibility factor is one.

2. Pipes are not insulated.

3. The spectrum curve used for dynamic analysis is shown in Figure 10-Cl.16-6

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Sample Piping Layout

Figure 3.7-14

3.8 DESIGN OF SEISMIC CATEGORY I STRUCTURES

All structures and equipment are designed in accordance with applicable codes for dead loads, live loads, seismic loads, and wind loads. The loading conditions and combinations are determined by the function of the structure and its importance in meeting the plant safety and power generation objectives.

To protect vital equipment and systems, certain critical plant structures must remain functional both during and following the most severe natural phenomena that could affect the site. These conditions are considered in the design and are investigated and defined in Chapter 2. Combinations of structural loads resulting from environmental events, normal operation, and design accidents are given in this Section.

Dead load includes self-weight of structures and permanent equipment or any other permanent loads contributing stress, such as soil or hydrostatic loads. The live loads that have been used in the design of structures are given in Table 3.8-1.

The following notations are used in this section:

U = required ultimate load capacity

D = dead load of structure, equipment, and other loads contributing permanent stress

L = live load as indicated in Table 3.8-1

R = force on structure from the rupture of any one pipe

T_o = thermal loads due to temperature gradient through wall under operating conditions

P = design-basis accident pressure load

H_o = force on structure from the thermal expansion of pipes under operating conditions

T_A = thermal loads due to temperature gradient through wall under accident conditions

H_A = force on structure from the thermal expansion of pipes under accident conditions

E = OBE resulting from ground surface accelerations listed in Section 3.7

E' = DBE resulting from ground surface accelerations listed in Section 3.7

A = hydrostatic load due to high water level at elevation 767 ft

W = wind load

W' = tornado

∅ = capacity reduction factor (defined in ACI 318-63, Section 1504)

f_s = allowable stress for structural steel

F_y = yield strength for steel

3.8.1 CONCRETE CONTAINMENT

Because the DAEC uses the Mark I steel containment (light bulb-torus), this section is not applicable.

3.8.2 STEEL CONTAINMENT

3.8.2.1 Description of the Containment

The Mark I suppression type containment consists basically of two steel pressure vessels, the drywell and the torus, joined by large vent pipes.

3.8.2.1.1 Drywell

The drywell is a steel pressure vessel with a spherical lower portion, 63 ft in diameter, and a cylindrical upper portion 32 ft in diameter. The overall height is approximately 108 ft 9 in. The design, fabrication, inspection, and testing of the drywell vessel complies with requirements of the ASME B&PV Code, Section III, Subsection B, "Requirements for Class B Vessels," Summer 1968 Addenda, and ASME Code Cases 1177, 1330, and 1413 which pertain to containment vessels for nuclear power stations. The primary containment is primarily fabricated of SA-516 GR 70 plate.

The drywell was designed for an internal pressure of 56 psig coincident with a temperature of 281°F with applicable dead, live, and seismic loads imposed on the shell. Thus, in accordance with the ASME Code, Section III, the maximum internal drywell pressure is 62 psig. Design external pressure is 2 psig at 281°F. Thermal stresses in the steel shell due to temperature gradients were taken into account in the design. Containment stresses are within allowable stresses permitted by the ASME Code. Detailed investigation was performed in areas where local buckling could occur due to the effects of concentrated loads, thermal loads, and non-axisymmetric distribution loads. Where stress intensities in these areas were not covered by the ASME Code, allowable stresses were determined using recognized buckling formulae such as those used by the American Institute of Steel Construction (AISC) and recognized, reputable authors (Roark, Timoshenko, and Grintner).

Special precautions not required by codes were taken in the fabrication of the steel drywell shell. Charpy V -notch speciments were used for impact testing of plate and forging material to give assurance of correct material properties. Plates, forgings, and pipe associated with the drywell had an initial NDT temperature of 0°F when tested in accordance with the appropriate code for the materials. It is not intended that the drywell will be pressurized or subjected to substantial stress at temperatures below 30°F.

The drywell is enclosed in a reinforced concrete structure for shielding purposes. In areas where it backs up the drywell shell, this reinforced concrete provides additional resistance to deformation and buckling of the shell. Above the transition zone, and below the flange, the drywell is separated from the reinforced concrete by a gap of approximately 2 in. Shielding over the top of the drywell is provided by removable, segmented, reinforced concrete shield plugs.

In addition to the drywell head, one combination double-door air lock/equipment lock, one bolted equipment hatch, and one bolted personnel access hatch are provided for access into the drywell.

3.8.2.1.2 Pressure Suppression Chamber

The pressure suppression chamber is a steel pressure vessel in the shape of a torus located below and encircling the drywell, with a major diameter of 98 ft 8 in. and a cross-sectional diameter of 25 ft 8 in. The pressure suppression chamber contains the suppression pool and the gas space above the pool. The suppression chamber will transmit seismic loading to the reinforced concrete foundation slab of the reactor building. Space is provided outside the chamber for inspection.

The toroidal suppression chamber was designed to the same material and code requirements as the steel drywell vessel. The material has an NDT temperature of 0°F.

3.8.2.1.3 Vent System

Large vent pipes connect the drywell and the pressure suppression chamber. A total of eight circular vent pipes are provided, each having a diameter of 4 ft 9 in. The vent pipes were designed for the same pressure and temperature conditions as the drywell and suppression chamber. Jet deflectors in the drywell at the entrance of each vent pipe prevent possible damage to the vent pipes from jet forces which might accompany a pipe break in the drywell. The vent pipes are fabricated of SA-516 GR 70 steel, and comply with requirements of the ASME B&PV Code, Section III, Subsection B. The vent pipes are provided with two-ply expansion bellows to accommodate differential motion between the drywell and suppression chamber. The vent pipe bellows are designed and fabricated to the same criteria as the containment vessels (ASME Section III, Class B, Summer 1968) with complete radiograph, dye penetrant, or magnetic particle inspection as required by the code.

The drywell vents are connected to a 3 ft 6 in. diameter vent header in the form of a torus which is contained within the airspace of the suppression chamber. Projecting downward from the header are 48 downcomer pipes, 24 in. in diameter and terminating not less than 3 ft below the water surface of the pool. The vent header has the same temperature and pressure design requirements as the vent pipes.

3.8.2.2 Applicable Codes, Standard and Specifications

The design of all structures and facilities conforms to the applicable general codes and specifications listed below, except where specifically stated otherwise:

1. Uniform Building Code (UBC) 1970

   Portions that apply to seismic design of Nonseismic structures only.

2. American Institute of Steel Construction (AISC)

   Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 1963 and 1970.

3. American Concrete Institute (ACI)

   Building Code Requirements for Reinforced Concrete (ACI 318-63) and Requirements for Reinforced Concrete Chimneys (ACI 307-69).

4. American Welding Society (AWS)

   Standard Code for Arc and Gas Welding in Building Construction (AWS D.1.0-66 and AWS D.2.0-66).

5. API Specification 650 for Welded Steel Storage Tanks.

6. ASME B&PV Code, Section III, Class B, governs the design and fabrication of the drywell and suppression chamber.

7. Official Linn County, Iowa,. Building Code.

8. American Society of Civil Engineers, Paper 3269, for wind design requirements.

9. American Iron and Steel Institute Specification for the Design of Light Gauge Cold-Formed Steel Structural Members, 1960.

Specifications and codes set down minimum requirements for the design and construction of structural elements of any structure.

Although some special structures involving unique problems are not explicitly covered by the codes, many provisions concerning the quality of materials and construction and design principles are considered to be generally applicable. Sound engineering knowledge, experience, and judgment, which are so essential in engineering, rather than blind adherence to codes, must be used to adapt the various codes to structures that may not appear to be within the scope of the codes.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 Basis for Loading Combination

The load combination basis for Seismic Category I structures is summarized as follows:

Load Combination	Minimum Requirements for Seismic Category I Structural Components
Normal loads + operating-basis earthquake	Within code allowable stresses
Normal loads + maximum probable flood	No functional failure
Normal loads + design-basis earthquake	No functional failure
Normal loads + tornado loads	No functional failure
Normal loads + design-basis LOCA	No functional failure

In addition to the above, the primary containment (including penetrations) and the reactor vessel support pedestal will be designed for normal loads = design-basis LOCA loads + earthquake loads for no functional failure.

3.8.2.3.2 Loading Using Normal Limits

Reinforced Concrete

Reinforced-concrete structures are designed for ductile behavior whenever possible, that is, with steel stresses controlling.

Concrete structures are designed to satisfy the most severe loading combinations, based on the load factors shown below:

U = $1.5\ D + 1.8\ L + 1.0\ T_o + 1.25\ H_o$

U = $1.25(D + L + H_o + E) + 1.0\ T_o$

U = $1.25(D + L + H_o + W) + 1.0\ T_o$

U = $0.9\ D + 1.25(H_o + E) + 1.0\ T_o$

U = $0.9\ D + 1.25(H_o + W) + 1.0\ T_o$

In addition, for ductile moment-resisting concrete space frames and for shear walls,

$U = 1.4(D + L + E) + 1.0\ T_o + 1.25\ H_o$

$U = 0.9\ D + 1.25\ E + 1.0\ T_o + 1.25\ H_o$

For structural elements such as equipment supports carrying mainly earthquake forces,

U = $1.0\ D + 1.0\ L + 1.8\ E + 1.0\ T_o + 1.25\ H_o$

<u>Structural Steel</u>

Steel structures are designed to satisfy the following loading combinations without exceeding the specified stresses:

$D + L + T_o + H_o$ - stress limit = f_s

$D + L + T_o + H_o + E$ - stress limit = $1.25\ f_s$

$D + L + T_o + H_o + W$ - stress limit = $1.33\ f_s$

In addition, for structural elements such as struts and bracings carrying mainly earthquake forces,

$D + L + T_o + H_o + E$ - stress limit = f_s

3.8.2.3.3 Loading Using Higher Limits

The Seismic Category I structures are in general proportioned to maintain elastic behavior when subjected to various combinations of dead loads, thermal loads, seismic loads, and accident loads. The upper limit of elastic behavior is considered to be the yield strength of the effective load-carrying structural materials. The yield strength, F_y, for steel (including reinforcing steel) is considered to be the guaranteed minimum given in appropriate ASTM specifications. The yield strength for reinforced-concrete structures is considered to be the ultimate resisting capacity as calculated from the ultimate strength design portion of ACI 318-63.

<u>Concrete</u>

Concrete structures are designed to satisfy the most severe of the following loading combinations:

2013-018 U = $1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ T_A + 1.25\ H_A + 1.0\ R$

2013-018 U = $1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ T_o + 1.0\ H_o + 1.0\ R$

U = $1.0\ D + 1.0\ L + 1.0\ A + 1.0\ T_o + 1.25\ H_o$

U = $1.0\ D + 1.0\ L + 1.0\ W' + 1.0\ T_o + 1.25\ H_o$

U = $0.95\ D + 1.25\ E + 1.0\ T_A + 1.0\ H_A + 1.0\ R$

U = $1.05\ D + 1.05\ L + 1.25\ E + 1.0\ T_A + 1.0\ H_A + 1.0\ R$

<u>Structural Steel</u>

Steel structures are designed to satisfy the most severe of the following loading combinations without exceeding the specified stresses:

$D + L + R + T_o + H_o + E'$ - stress limit* = $1.5\ f_s$

$D + L + R + T_A + H_A + E'$ - stress limit* = $1.5\ f_s$

$D + L + A + T_o + H_o$ -stress limit* = $1.5\ f_s$

$D + L + T_o + H_o + W'$ - stress limi*t = $1.5\ f_s$

*Maximum allowable stress in bending and tension is $0.9\ F_y$. Maximum allowable stress in shear is $0.5F_y$

Concrete structures are designed using the ultimate strength design method and allowable stresses in accordance with ACI 318-63. In no case did the actual design stresses for the DAEC exceed the ACI allowable stresses. The only modification to the ACI provision is in the assignment of load factors as indicated in Section 3.8.2.4. The conservative choice of loading conditions justifies the use of this modification.

Concentrated loads were provided for by the addition of special restraining systems.

3.8.2.3.4 Pipe Jet Effects

The primary containment system is designed to withstand forces imposed by an earthquake that occurs simultaneously with a LOCA. In addition to the pressure and the thermal loading condition described in Section 6.2.1, the primary containment is designed to withstand the jet forces shown below at the locations indicated from any direction within the drywell:

Location	Jet Force (maximum)	Interior Area Subjected to Jet Forces
Spherical part of drywell	393,000 lb	2.19 ft^2
Cylinder and sphere to cylinder transition	325, 000 lb	1.80 ft^2
Suppression chamber	21,000 lb	Each pipe

These forces are described in Section 3.6.2

The capabilities of the primary containment with respect to stress levels and load combinations for the postulated events are noted below:

Primary Containment (including penetrations)

(1) $D + L + T_A + H_A + P + E$	ASME, Section III, Class B, without the usual increase for seismic loading.
(2) $D + L + H_A + P + R + T_A + E$	Same as (1) above, except local yielding is permitted in the area of the jet force where the shell is backed up by concrete. In areas not backed up by concrete, primary local membrane stresses at the jet force do not exceed 0.90 times the yield point of the material at 300°F.
(3) $D + L + H_A + P + R + T_A + E$	Primary membrane stressed in general do not exceed the yield point of the material. The same criteria as in (2) above are applied to the effect of jet forces for this loading condition.

The jet forces consist of steam and/or water at 300°F maximum. The jet forces do not occur simultaneously. However, a jet force is considered to occur coincident with design internal pressure and a temperature of 150°F.

3.8.2.3.5 Flooded Containment

For the postaccident internal flood condition, the containment is analyzed for the dead load of the vessel, all equipment and appurtenances, OBE, and hydrostatic water pressure to the refueling floor level [redacted]

The maximum possible postaccident flood condition temperature is 212°F; the containment is designed for a temperature of 281°F. Thus, temperature stresses are not governing criteria. The allowable stresses for this condition are as follows:

ASME Section III B material		- code permitted yield
AISC material	bending	- 1.5 x yield
	bearing	- 1.2 x yield
	shear	- 0.8 x yield
	compression	- 1.0 x yield
All welds		- 0.8 x yield
Concrete bearing		- 0.8 f 'c

3.8.2.3.6 Forces Applied by Piping

Piping systems that impose forces on structures from thermal expansion are supported so as to reduce the effect both in the piping and on the structures. When properly supported, the loads on the structures do not substantially change from the normal operation to the postaccident condition.

3.8.2.4 Design and Analysis Procedures

The drywell concrete reinforcement design was based on a combination of theories of shell design, chimney design, and ring design.

A finite-element analysis was used to evaluate local stresses caused by end moments of deep beams at the drywell interface.

Structural design and construction were performed in such a way as to prevent cracking by mix design, pour limitation, and curing precautions.

The effect of shear was evaluated, and the stresses are within the allowable limits for an unreinforced web. Thus, no shear reinforcing was required.

All steel structures are designed by the elastic analysis method.

For all structures, minimum factors of safety of 1.5 against overturning and 1.2 against uplift (buoyancy) have been maintained.

3.8.2.4.1 Limiting Stress and Strain for Concrete

At ultimate strength of concrete structural members, concrete stress is not proportional to strain although strains vary linearly with distance from the vertical axis. The actual geometric shape of concrete compression stress distribution varies considerably. To determine the ultimate capacity of the structural member, it is not necessary to know the exact shape of compression stress distribution; only the magnitude of the resultant of the compression stresses and its location need be known. The actual complex stress distribution may be replaced by a fictitious one of some simple geometric shape that results in the same total compression force applied at the same location in the member when it is on the point of failure.

ACI 318 has adopted a rectangular stress distribution with an average compressive stress intensity of 0.85 f 'c corresponding to a strain of 0.003 in the extreme concrete fiber. The average compressive stress of 0.85 f 'c is thought of as acting over part of the section with a depth of 0.85 c, where c is the distance to the actual neutral axis from the compression edge. It has been experimentally verified that ultimate strength of members calculated by means of this assumed equivalent stress block agrees with the actual ultimate strength.

Columns with or without moments about one or both major axes may be thought of as uni-, bi- or tri-axial stress distributions. The ultimate load or carrying capacity of such members is calculated using the general propositions of ultimate strength design based on rectangular equivalent stress block and ultimate concrete strain of 0.003. Structural members with bi- and tri-axial stresses were designed such that the sums of the strains did not exceed the limiting strain of 0.003.

The important point is that the maximum concrete strain should not exceed 0.003 under any state of stress because this figure conservatively represents the strain in compression at which concrete will fail by crushing.

3.8.2.4.2 Use of Ultimate Strength Method

Ultimate strength design method is used to determine the ultimate carrying capacities of all reinforced-concrete structural members. The ultimate carrying capacities of structural members are calculated using f_y for reinforcement and 0.85 f 'c for concrete.

3.8.2.4.3 Load Factors

Design loads are increased by overload factors to obtain the ultimate loads to be used for design. Overload factors are selected such that these ultimate loads have an acceptably small probability of ever being exceeded. The theoretical member capacities are lowered by the reduction factor ∅ to allow for variations in quality of materials and construction and the approximations and assumptions made in theoretical analysis. Considering the high standards of quality control enforced during the construction of nuclear power plants, the need for the factor ∅ in the design of these structures is debatable.

Load factors of 1.0 are used only for combinations of loading conditions that are extremely improbable (e.g., a combination of dead load + live load + DBE + thermal loads + pipe rupture). It should be noted that the DBE itself is a hypothetical maximum that may never occur during the lifetime of the structure.

In view of the extreme improbability of the occurrence of the loading combinations for which load factors of 1.0 are used and in view of the availability of some reserve in the ultimate capacity of a structural member corresponding to the factor ∅, it is felt that the load factors of 1.0 are reasonable and justified.

3.8.2.4.4 Design of Openings

As a minimum requirement, all openings in walls and floors were provided with additional reinforcements at sides and corners. Openings in shear walls were included in the plane stress finite-element analysis. Reinforcement was provided in accordance with the stress concentration around the openings without exceeding the allowable stresses.

When openings were provided within an area of high-stress concentrations, the openings were analyzed by accepted engineering principles and completely framed as required to carry the loads.

3.8.2.4.5 Piping Near Containment Penetration

The following design criteria were applied to piping systems in the design of the containment penetrations:

1. Nuclear Class I - Large-Diameter Piping

   a. The flued head is anchored to the reactor building concrete structure. The flued heads and associated structures are capable of sustaining all the postulated loads from the process piping. The stresses resulting from the application of these loads will remain below yield unless appropriate analyses can demonstrate that no gross loss of structural integrity is suffered. A metallic bellows-type expansion joint is provided between the flued head fitting and the containment penetration, thereby allowing essentially free thermal expansion of the drywell shell.

   b. The containment penetration is designed for the bellows spring rate forces resulting from thermal displacement of the containment during normal operation and postaccident conditions.

   c. The stresses in the structural restraints for loadings resulting from a postulated break in the piping, postaccident conditions, or seismic-induced loads are limited to:

Steel	0.9 yield stress
Concrete	0.8 f 'c

   d. With the exception of the main steam system, the stresses in the piping and flued head fittings will be within the allowable stresses permitted in ANSI B31.7, including Code Case 70. The stresses in the main steam piping and flued head fitting are within the allowable stresses permitted in ANSI B31.1.0.

2. Nuclear Class I - Small-Diameter Piping

    a. If the piping is connected directly to the containment nozzle at the flued head, all piping reactions will be transmitted to the containment. In general, however, the penetration is much larger than the piping, and therefore the piping has little influence on it.

    b. The flued head is designed to take the full moment carrying capability of the process pipe.

    c. The stresses in the piping and flued head fitting will be within the allowable stresses permitted in ANSI B31.7, including Code Case 70.

    d. The stresses in the containment vessel from piping loads caused by pipe rupture, postaccident, and seismic conditions will be within the allowable stresses permitted by the 1968 Edition of ASME Code, Section III.

3. Nuclear Class II Piping

    a. The piping is connected directly to the containment nozzle at the flued head, and all piping reactions will be transmitted to the containment.

    b. The flued head is designed to take the full moment carrying capability of the process pipe.

    c. The stresses in the flued head fittings and piping from pipe rupture, postaccident, and seismic loads will be within the allowable stresses permitted in ANSI B31.1.0.

The piping systems that could impose significant reactions to the containment penetrations are the Nuclear Class I, large-diameter, high-pressure piping systems. As described above, these piping systems are restrained and anchored in such a way as to preclude damage to the containment penetration. These piping systems are main steam, feedwater, core spray, residual heat removal, high-pressure coolant injection, and reactor water cleanup.

There are several small-diameter piping systems that contain high pressure. Although these are not expected to possess the energy to cause any significant damage, they were investigated and restraints or anchors were placed where required to keep containment penetration stresses to an acceptable limit. Among these are the main steam drain, recirculation loop sample, and the control rod drive return systems.

3.8.2.5 Structural Acceptance Criteria

The acceptance criteria, generally as prescribed by appropriate codes, are stated in Section 3.8.2.3 along with the load combinations to which each criterion is applicable.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.l General

Detailed specifications and working drawings for the installation of all construction materials were of such scope as to ensure that the quality of work was commensurate with that necessary to preserve the integrity of the plant structures. Noncombustible and fire-resistant materials were used wherever necessary throughout the facility, particularly in areas containing systems or components that affect unit safety.

3.8.2.6.2 Concrete

All concrete work was in accordance with ACI 318-63 and technical specifications incorporating the latest engineering knowledge in quality construction. Admixtures were added to improve the quality and workability of the plastic concrete during placement and to retard the set of the concrete. Maximum practical size aggregate, water-reducing additives, and a low slump were used to minimize shrinkage and creep. Aggregates conformed to "Standard Specifications for Concrete Aggregate," ASTM Designation C-33.

3.8.2.6.2.1 Portland Cement. Portland cement conformed to "Specifications for Portland Cement," ASTM C-150.

3.8.2.6.2.2 Aggregates. The acceptability of aggregates was based on the ASTM tests listed in Table 3.8-2. These tests were performed by a qualified commercial testing laboratory.

3.8.2.6.2.3 Water. Water used in mixing concrete was clean and free from injurious amounts of oils, acids, alkalis, salts, organic materials, or other substances that could be deleterious to cement, aggregate, or steel.

3.8.2.6.2.4 Mixing and Placing. Concrete mixes were designed in accordance with ACI 613, using materials qualified and accepted for this work. Trial mixes were tested in accordance with applicable ASTM Codes as indicated in Table 3.8-3.

Slump, air content, and temperature measurements were taken before cylinders were cast from the sample. Slump tests were performed in accordance with ASTM C-143, "Standard Method of Test for Slump of Portland Cement Concrete." Air content tests are performed in accordance with ASTM C-231, "Standard Method of Test for Air Content of Freshly Mixed Concrete by the Pressure method." Compressive strength tests were made in accordance with ASTM C-39, "Standard Method of Test for Compressive Strength of Molded Concrete Cylinders."

The evaluation of compressive tests is in accordance with ACI 214-65.

3.8.2.6.2.5 Crack Control. Structural design and construction were performed in such a way as to prevent cracking by mix design, pour limitation, and curing precautions. The stress limits should result in very limited cracking on the order of a few hundredths of an inch. Such cracking would not significantly affect the leak resistance of the structure.

3.8.2.6.3 Reinforcing Steel

Reinforcing steel for concrete consists of deformed bars meeting the requirements of ASTM A615-68 and appropriate ASTM specifications for the various grades and strengths of bars employed. Placing and splicing of bars are in accordance with the requirements of ACI 318-70.

Mill test results were obtained from the reinforcing steel supplier for each heat of steel to substantiate the required composition, strength, and ductility.

ASTM-A-615 states that the deformations shall be measured on one bar of each 10 tons for No. 3 to No. 11 bars and one bar of each 25 tons for No. 14 and No. 18 bars. Random testing by the supplier in this fashion verified that the deformations on the bars met the requirements of Safety Guide 15.

In addition to this check by the supplier, a Bechtel inspector performed periodic random inspections at the fabricator's plant to verify the compliance of the rebar to ASTM A-615, detail drawings, and requirement specifications.

3.8.2.6.4 Structural Steel

Structural steel was in conformance with one of the following specifications:

Structural Steel, ASTM A36-70

High-Strength Structural Steel, ASTM A440-66

High-Strength Low-Alloy Structural Manganese Vanadium Steel, ASTM A441-66a

Carbon Steel Plates with Improved Transition Properties, ASTM A442-69

High-Strength Steel Bolts for Structural Joints, ASTM A325

Certified mill test reports or certified reports of tests made by the fabricator or a testing laboratory in accordance with ASTM A6-69a and the governing specification constituted evidence of conformity with one of the above ASTM specifications. In addition, the fabricator, when appropriate, provided an affidavit stating that the structural steel furnished met the requirements of the grade specified (ASTM C-150).

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL CONTAINMENTS

3.8.3.1 General

The design of all steel floor framing systems including connections to vertical members is in accordance with AISC Manual of Steel Construction 1963 and 1970 editions.

3.8.3.2 Reactor Concrete Pedestal

The reactor vessel pedestal is a [redacted] cylinder with [redacted] concrete walls. The outside diameter of the pedestal is [redacted] and the inside diameter was formed by the [redacted] outside diameter reactor vessel erection skirt. A 3-in. layer of shotcrete was applied to the inside face of the erection skirt to act as thermal insulation. Reactor vessel loads are transferred from the skirt to the pedestal concrete by means of shear rings welded circumferentially around the skirt.

The cylindrical pedestal cantilevers from a spherical-shaped base formed by the inside of the drywell. The shears and moments are transferred to the drywell foundation concrete by the erection skirt acting as a shear ring and by friction between the drywell and the concrete.

The overall design was based on very conservative assumptions to allow for the complex interactions of the various loads. The loading combinations and allowable stresses are given in Table 3.8-4. The design stresses are within the allowable stresses resulting in a high safety factor.

As shown in Table 3.8-4, the accident + DBE + temperature was considered. Since, for the accident condition combined with the OBE, the OBE load factor for ultimate strength design is 1.25, this condition is not as severe as the accident condition + DBE; thus, the accident + OBE condition need not be considered.

The reactor vessel supports were designed to withstand (1) forces imposed by operating loads, (2) seismic loads, and (3) jet forces resulting from the complete instantaneous severance of any one of the largest connecting pipes. The magnitudes of the seismic forces imposed on the vessel are determined by dynamic analysis as described in Sections 3.7.2 and 3.10.1. Both the OBE and DBE were considered.

The jet forces considered in the design of the reactor pressure vessel (RPV) supports are those resulting from a complete severance of one of the main steam lines or one of the recirculation outlet lines. Jet forces considered in the design, calculated as described in Section 3.6.2.2.4.1, were as follows:

RPV Nozzle	Force (kips)
Main Steam	325
Reciruclation outlet	393

The transient and steady-state thermal gradients in the reactor pedestal were determined using a finite-element heat flow computer program based on the work of Wilson and Mitchell[2]. The calculated stresses in the pedestal are within the allowances permitted by the ACI ultimate strength design.

Since the reactor vessel for the DAEC was field fabricated, the reactor support skirt extends through the drywell vessel foundation, and a bolted support system is not used.

The concrete shear stresses at the RPV pedestal base for the condition of DL + LL + DBE + Jet are 33.5 psi with the drywell empty and 35.1 psi with the drywell flooded. The shear stresses in the steel shear ring connecting the spherically shaped concrete base to the drywell for the condition of DL + LL + DBE + Jet are 3100 psi with the drywell empty and 3200 psi with the drywell flooded. This assumes no friction between the concrete and steel.

The factor of safety for shear stress at the connection between the bottom of the drywell and the concrete foundation for the DL + LL + DBE + Jet condition is 6.9 with the drywell empty and 4.3 with the drywell flooded. These safety factors are based on the yield strength of the material and neglect friction between the drywell steel plates and the concrete.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

The criteria described here apply to Seismic Category I structures and are intended to supplement applicable industry design codes where necessary to provide design safety margins for rare events like postulated LOCAs, DBEs, tornados, and missiles.

3.8.4.1 Description of the Structures

This section describes the structures, the loads and load combinations, and the static and dynamic analyses used in the structural design of Seismic Category I structures and briefly discusses typical structural elements of the reactor building.

Design procedures used for the reactor building were also used for the other Seismic Category I structures, such as the control building, offgas stack, turbine building, intake structure, and pump house.

3.8.4.1.1 Reactor Building Floor System

The reactor building has more than one type of floor system. Basically, the floor system is of composite construction that consists of cast-in-place reinforced concrete so interconnected with steel beams or precast concrete members that the component elements act together as a unit.

The floor system is designed to support dead load, equipment loads, laydown loads, piping loads, live loads, and vertical seismic loads based on dead load plus permanent equipment loads.

The floor slabs are generally designed as one-way continuous slabs in accordance with ACI 318-63.

Most of the steel beams and girders are designed as composite sections in accordance with AISC using 0.75-in.-diameter headed shear studs.

The refueling floor at [redacted] is selected to illustrate the implementation of the design criteria because of its critical function and particularly heavy loading. In addition to the 1000-lb/ft^2 floor live load, the floor slab area supporting the fuel cask and the reactor head is designed to sustain a load of 100 tons distributed over an area of 5 ft 0 in. in diameter or a load of 110 tons distributed over an area of 6 ft 7 in. in diameter.

Table 3.8-5 lists the allowable stresses and loading combinations as they apply to the particular component described. The design stresses are within the allowable stresses, and the system is structurally adequate.

3.8.4.1.2 Reactor Building Concrete Wall

The west wall of the reactor building is selected to illustrate the implementation of the design criteria.

The wall is constructed in the following way:

Substructure - [redacted]

Superstructure -

This wall is designed to resist several loading combinations. It acts as a shear wall for seismic forces in the north-south direction. The substructure wall experiences soils and hydrostatic loads, and the superstructure wall is designed to withstand normal wind loads (W) as well as tornado loads (W'). The wall was also investigated for the effect of tornado missiles, and the wall thickness was determined to be adequate.

The governing design conditions and allowable stresses are given in Table 3.8.6.

3.8.4.1.3 Reactor Building Steel Superstructure

The reactor building steel superstructure houses the 100-ton crane runway. It is a steel structure and consists of [redacted] rigid portal frames [redacted]. Each frame consists of [redacted] flange columns and rafters.

[redacted] Horizontal bracing has been provided in the plane of the roof to transmit lateral forces to vertical bracing in walls.

[redacted]

All bracing has been designed to resist wind or earthquake loading. The rigid frames have been designed for dead and live loads, wind loads, or earthquake loads.

The superstructure has also been designed for tornado loads (W1) on the assumptions that all or part of the metal siding is blown away by the tornado and the basic superstructure frame is subjected to full tornado winds of 300 mph. Under these conditions, the frame will withstand the loading without failure. The stresses may exceed normal allowable stresses, but will not exceed 90% of the yield stress of structural steel.

Table 3.8-7 summarizes the governing loading combinations, method of analysis and allowable stresses. The design stresses are well within the limits of the allowable stresses.

3.8.4.1.4 Precast T-beams

[redacted]
The T-beams act as supports for the poured-in-place concrete, and thus form a composite concrete floor system.

The concrete elements constructed in separate placements are so interconnected that the elements respond to loads as a unit.

The grade floor is designed to act as a diaphragm for the basement walls and does not experience bending moments and the reversal of stresses at the supports from horizontal seismic loads. However, vertical seismic loadings that are based on dead load and permanent equipment loads will cause tension in the top fibers of the slab system at the supports. Rebars have been added to account for these additional stresses.

Details of the design connections are shown in Figure 3.8-1.

3.8.4.1.5 Precast Concrete Panels

[redacted]

The connections to the structural columns were designed to withstand a construction wind pressure of 40 psf or suction of 19 psf. They were also checked for the DBE seismic accelerations at the reactor building operating floor level that would be the worst-case seismic condition.

The mode of failure for the connections under seismic loadings would be pull-out of the concrete inserts in the panel.

The factor of safety against the failure of all these inserts and the individual panel coming loose is 18.7.

Typical connection details for the reactor building precast panels are shown in Figure 3.8-2.

3.8.4.1.6 Reactor Building Crane

The crane runway is an integral part of the reactor building superstructure. The anchorage of the runway rails to the crane runway is designed to resist the horizontal and vertical forces transmitted by the crane during an earthquake. Safety clamps secure the bridge end trucks to the runway rails.

Safety clamps also secure the trolley to the bridge rails. The crane is designed for the earthquake accelerations at the level of the crane runway. The anchorage clamps and the crane structure are designed for the most severe of the following:

	Allowable Stress Increase
D + L + impact and crane forces (per AISC specification)	None
D + L + OBE	None
D + L + DBE	0.9 F_y bending, 0.5 F_y shear
D + tornado	0.9 F_y bending, 0.5 F_y shear

3.8.4.1.7 Masonry Block Walls

Masonry walls perform the functions of partitions, partition bearing, shielding, shielding bearing, missile shielding, missile protection, shielding blockout, and partition/fire walls. Table 3.8-8 lists the masonry walls and their functions.

The NRC IE Bulletin 80-11 required the DAEC to identify and reevaluate the design adequacy of all masonry walls which are in the proximity or have attachments from safety-related piping or equipment such that wall failure could affect a safety-related system. A review of all known masonry block walls that are in buildings containing safety-related components was completed to determine which are near to, or have attachments to, safety-related piping or equipment. There were 445 masonry walls reviewed, and of these, 150 were neither in the proximity of nor had attachments from safety-related piping or equipment. The remainder were considered to be in proximity of or have attachments from safety-related piping or equipment. Table 3.8-8 lists the walls and identifies the safety-related system or the components attached to the wall. Column 2 of the table provides the priority assigned to the wall for reevaluation. The order of review was as follows:

Priority 1	Masonry walls which support safety-related piping 2.5 in. or greater.
Priority 2	Masonry walls which support any safety-related item weighing 100 lb or more.
Priority 3	Masonry walls which support any nonsafety-related item weighing 100 lb or more, but are in the proximity to or have attachments from safety-related items.
Priority 4	Masonry walls which support loads less than 100 lb, but are in the proximity to or have attachments from safety-related items.
Priority 5	Masonry walls which will not have a detailed survey or be reanalyzed because the wall is neither in proximity to nor attached to safety-related items.

The criteria for the reevaluation of the masonry walls can be found in Attachment 3 of Reference 3. The walls were reevaluated. Reference 3 is the DAEC response to IE Bulletin 80-11 concerning masonry wall design. Reference 4 transmitted DAEC's response to the NRC requests for additional information relative to masonry wall design.

The NRC accepted all but five of the walls as meeting the requirements of IE Bulletin 80-11 (Reference 5). Three of these walls were reevaluated and found to be acceptable using elastic methods. This was reported to the NRC by Reference 6. The final two were reevaluated and found to be acceptable by using the results of new seismic analyses for the reactor and turbine building. These new analyses incorporated radiation damping associated with soil-structure interaction and provide more realistic time histories and input acceleration. The input accelerations for the two walls in question [redacted] were thereby reduced significantly. This was reported to the NRC by Reference 7.

3.8.4.2 Applicable Codes, Standards, and Specifications

The documents referenced in Section 3.8.2.2 are also applicable to all Seismic Category I structures.

3.8.4.3 Loads and Loading Combinations

The loading criteria of Section 3.8.2.3 are also applicable for all Seismic Category I structures. In addition, the effects of missiles and of the failure of Nonseismic structures is considered.

3.8.4.3.1 Turbine Missiles

All safety-related equipment was evaluated with respect to its shutdown capability in the unlikely event that a turbine- generator failure would result in a postulated turbine missile.

[redacted] For a further discussion of Turbine Missiles see Section 3.5.1.3.

3.8.4.3.2 Tornado-Generated Missiles

Tornado-generated missiles are discussed in Section 3.5.1. To prevent a loss of function due to tornado-generated missiles, both structural stability and penetration have been investigated. [redacted] This concept is used throughout
2012-012 the plant wherever tornado-generated missiles could damage Seismic Category I equipment, unless other provisions are provided in the UFSAR, i.e. missile protection for the EDGs (see Section 8.3.1.3).

As stated in Section 8.3.1.3, separation of the EDGs meet single failure criterion and components of the EDGs are located so as to minimize the possibility of damage due to explosions or missiles.

3.8.4.3.3 Interaction Between Seismic Category I and Nonseismic Structures

All Seismic Category I structures are protected from damage by Nonseismic structures during an OBE or DBE earthquake by physically separating Nonseismic structures from Seismic Category I structures, with the exception of the emergency diesel compartment of the turbine building and the emergency pump house compartment.

[redacted] A complete dynamic analysis has been conducted for the turbine building to ensure the integrity of Seismic Category I equipment within the building and Seismic Category I equipment and structures adjacent to it.

For structures defined as partially Seismic Category I and partially Nonseismic, those portions of Nonseismic structures housing Seismic Category I equipment are designed in accordance with Seismic Category I design criteria. In addition, the structure as a whole was investigated to ensure that damage to the Nonseismic part would not endanger the area housing Seismic Category I equipment. This generally involves a complete dynamic analysis of the entire building. The structures falling within this category are the turbine building and the pump house.

3.8.4.4 Design and Analysis Procedures

In general, procedures for all Seismic Category I structures are as given in Section 3.8.2.4. However, a few additional considerations apply.

3.8.4.4.1 Criteria

The Seismic Category I concrete and steel structures are designed considering three interrelated primary functions for the design-loading combinations described in Section 3.8.2.3. The first consideration is to provide structural strength equal to or greater than that required to sustain the combination of design loads and provide protection to other Seismic Category I structures and components. The second consideration is to maintain structural deformations within such limits that Seismic Category I components and/or systems will not experience a loss of function. The third consideration is to limit excessive containment leakage by preventing excessive deformation and cracking where containment integrity is required.

In general, structures are analyzed by elastic methods, and structural design is performed using the ultimate strength method for reinforced concrete and working stress method for structural steel as defined in ACI 318-63, and in AISC Manual of Steel Construction, 1970. Finite-element stress analysis and other techniques are also used where applicable or necessary.

3.8.4.4.2 Two- or Three-Dimensional Stresses

Structural steel members of Seismic Category I structures subjected to two- or three-dimensional stress conditions are typically beams and beam columns. Beams subjected to bending about both axes were designed so that the sum of the bending stresses about each axis did not exceed the stress limits recommended by the AISC. Beam columns subjected to bending about both axes and axial load were designed using the interaction formulas presented in the AISC specifications. The interaction formulas contain the allowable stress limits for bending and axial loads as parameters. The allowable stress limits used have appropriate margins of safety built into them.

3.8.4.4.3 Shear and Bond Stresses

The allowable ultimate diagonal tension shear stress permissible on an unreinforced web is $\sqrt{f'c}$. The biaxial state of shear stress may be thought of as that existing in a beam subjected to shear parallel to both major axes. In such cases, the allowable shear stress in each direction may be taken as half of the above value. A beam subjected to shear in both directions and torsional moments could be thought of as being in a state of triaxial shear stress. The allowable ultimate torsional shear stress is taken as 2.4 f 'c when torsion acts alone. The design for cases involving shear in both principal directions and torsion is based on interaction formulas as recommended by Mattock.[8] The ultimate bond stress allowed is 6.7 f 'c/D for top bars and 9.5 f 'c/D for other bars. The main requirement for safety against bond failure is that the length of the bar from any point of steel stress (f_s or at most f_v) to its nearby force end must be at least equal to its development length. If this requirement is satisfied, the magnitude of the nominal flexural bond stress along the beam is of only secondary importance. The integrity of the members is ensured even in the face of possible minor local bond failures. If the actual available length is inadequate for full development, special anchorage, such as hooks, must be provided.

3.8.4.5 Structural Acceptance Criteria

To prevent a loss of function for all structures required for safe shutdown, limiting values were established to control the maximum structural deformations to within defined limits and to provide strength equal to or greater than that required to sustain the loads and limit deformations. The limiting values for deformation and strength are normally set by the need to maintain structural integrity, the need to prevent structural deformation from displacing the equipment to the extent that the equipment suffers a loss of function, and the need to prevent deformation that would inhibit the ability of the structures to control leakage. In the design of Seismic Category I structures, structural deformations were never the controlling criteria.

Limit design of concrete and plastic design of steel were never used. All steel structures were designed so that for any combination of loads the stresses did not exceed 0.9 F_y in bending and tension and 0.5 F_y in shear.

The yield strength for reinforced-concrete structures is considered to be the ultimate resisting capacity as calculated from the ultimate strength design provision of the ACI 318-63.

3.8.5 FOUNDATIONS

[redacted]

[redacted]

[redacted]

The [redacted] overturning moments were calculated using estimated lateral acceleration coefficients exceeding those determined by the reactor building seismic analysis. These overturning moments were included in the finite-element analysis of the foundation mat and rock substructure. This analysis yielded vertical bearing values at the foundation mat and substructure interface.

The safety factor against the overturning of the reactor building is 2.3 for the DBE combined horizontal and vertical accelerations.

REFERENCES FOR SECTION 3.8

1. Broms, B. B., "Crack Width and Crack Spacing in Reinforced Concrete Members," Journal of the ACI, Vol. 62, No. 10, 1965.

2. Wilson, E. L., and Mitchell, R. E., Journal of Nuclear Engineering and Design, Vol. 4, 1966.

3. Letter from L. D. Root, Iowa Electric, to J. G. Keppler, NRC, Region III, Subject: Response to IE Bulletin 80-11 Concerning Masonry Wall Design, dated November 10, 1980 (With Attachment) (LDR-80-335).

4. Letter from L. D. Root, Iowa Electric, to H. R. Denton, NRC Subject: DAEC, IE Bulletin 80-11; Masonry Wall Design, dated October 6, 1982 (LDR 82-264).

5. Letter from D. Vassallo, NRC, to L. Liu, Subject: Masonry Wall Design, IE Bulletin 80-11, dated August 22, 1985.

6. Letter from R. W. McGaughy, Iowa Electric, to H. R. Denton, NRC, Subject: DAEC Masonry Wall Design, IE Bulletin 80-11, dated November 4, 1985.

7. Letter from R. W. McGaughy, Iowa Electric, to H. R. Denton, NRC, Subject: DAEC Masonry Wall Design, IE Bulletin 80-11, dated April 30, 1986.

8. Mattock, "How to Design for Torsion," ACI SP-18.

LIVE LOADS ON STRUCTURES

General	Loads
[redacted]	40 psf
[redacted]	100 psf
[redacted]	100 psf
[redacted]	100 psf
[redacted]	250 psf (or 5 kips wheel load)
[redacted]	100 psf
[redacted]	Cooper E-72
Turbine Building	
[redacted]	1000 psf
[redacted]	400 psf
[redacted]	300 psf
[redacted]	350 psf (or 5 kips wheel load)
[redacted]	1000 psf
Reactor building	
[redacted]	[redacted] 100 tons (5 ft in diameter) or 110 tons (6 ft 7 in. in diameter)
[redacted]	1000 psf
[redacted]	1500 psf
[redacted]	[redacted] 2400 psf
[redacted]	[redacted] 400 psf
[redacted]	[redacted] 200 kips
Drywell interior	
[redacted]	150 psf
[redacted]	150 psf
[redacted]	150 psf
[redacted]	150 psf

LIVE LOADS ON STRUCTURES

General	Loads
Control building	
[redacted]	200 psf 400 psf 100 psf
Intake structure	
[redacted]	250 psf 250 psf
Administration building	
[redacted]	100 psf 200 psf
Pump house	
[redacted]	100 psf
2014-007 [redacted]	250 psf 100 psf 200 psf
Radwaste building	
[redacted]	250 psf
Low-level radwaste processing and storage facililty	
[redacted]	250 psf 100 psf

Crane and elevator loads

Crane and elevator loads are considered as live loads. Impact allowance for traveling crane supports and their connections is 25%. A 100% impact is used for elevator supports.

In addition to live loads given above, beams over equipment areas are designed for an additional 5 kip load applied anywhere on the beam.

Table 3.8-2

AGGREGATE TESTS

Test	ASTM
L. A. abrasion	C-131
Friable particles in aggregate	C-142
Material finer No. 200 sieve	C-117
Mortar-making properties	C-87
Organic impurities in sands for concrete	C-40
Potential reactivity (chemical)	C-289
Potential reactivity (mortar bar)	C-227 (if necessary after performance of C-289)
Sieve analysis	C-136
Soundness	C-88
Specific gravity and absorption for coarse aggregate	C-127
Specific gravity and absorption for fine aggregate	C-128
Petrographic	C-295

Table 3.8-3

TRIAL MIX TESTS

Test	ASTM
Making and curing cylinder in laboratory	C-192
Air content	C-231
Slump	C-143
Bleeding	C-232
Weight cubic foot	C-138
Compressive strength Tests	C-39

Table 3.8-4

REACTOR CONCRETE PEDESTAL

Element Location/ Description	Criteria	Method of Analysis	Load Combinations	Allowable Stress (ksi)
Reactor Pedestal [redacted]	Concrete ACI 318-63 fc' = 5000 psi at 28 days	Ultimate Strength Analysis ACI 505 and finite-element heat flow computer program	Accident Condition U = 1/∅ (1.0D + 1.0L + 1.0T+1.0E' + 1.0R) U = 1/∅ (1.0D + 1.0L + 1.0T_A + 1.0E' + 1.0R)	fc' = 5.0 ft = 54.0 fv = 0.120
	Reinforcing ASTM A615 Grade 60 Fy = 60 ksi			
Reactor Pedestal [redacted]	Concrete Same as above	Working Stress Design ACI 505 and finite-element heat flow computer program	Normal Operating Condition D+L+T_o	fc = 2.25 ft = 24.0 fv = 0.078
	Reinforcing Same as above			

Table 3.8-5

REACTOR BUILDING FLOOR SYSTEM

Element Location/ Description	Criteria	Method of Analysis	Load Combinations	Allowable Stress (ksi)
Floor Beams [redacted]	AISC Manual of Steel Construction 1963, 1970 Section 1.10 and 1.11, ASTM A36 Steel	Elastic Analysis. Simple span composite steel and concrete beam.	D+L	Fb = 22.0 Fv = 14.5 fc = 1.8
			D+L+E'	Fb = 32.4 Fv = 18.0 fc = 4.0
Floor Slab [redacted]	Concrete ACI 318-63 fc' = 4000 psi at 28 days	Ultimate Strength Analysis. Continuous one way slab.	U = 1/∅ (1.5D + 1.8L)	fc = fc' = 4.0 $fv = 2\varnothing\sqrt{fc'} = 0.108$ fy = 60.0
	Reinforcing ASTM A615 Grade 60 Fy = 60 ksi	Same as above	U = 1/∅ (1.0D + 1.8L+ 1.0E')	fc = fc' = 4.0 $fv = 2\varnothing\sqrt{fc'} = 0.108$

Table 3.8-6

REACTOR BUILDING CONCRETE WALLS

Element Location/ Description	Criteria	Method of Analysis	Load Combinations	Allowable Stress (ksi)
Substructure [redacted]	Concrete ACI 318-63 fc' = 5000 psi at 28 days	Ultimate Strength Analysis. Continuous two-way vertical slab with support at base slab and diagonal walls. No support assumed at top during construction.	U = 1/∅ (1.5D + 1.8L)	fc = fc' = 5.0 fy = 60.0 $fv = 4\varnothing\sqrt{fc'} = 0.240$
	Reinforcing ASTMA 615 Grade 60 Fy = 60 ksi		U = 1/∅ (1.0D + 1.0L +1.0E') U = 1/∅ (1.0D + 1.0L + 1.0A)	Same as above Same as above
Superstructure [redacted]	Concrete ACI 318-63 fc' = 4000 psi at 28 days	Ultimate Strength Analysis. Continuous one-way slab. Shear Wall Analysis: Plane Stress Finite Element Method.	U = 1/∅ (1.0D + 1.0L + 1.0W')	fc = fc' = 4.0 fy = 60.0 $fv = 2\varnothing\sqrt{fc'} = 0.108$
	Reinforcing ASTM A615 Grade 60 Fy = 60 ksi		U = 1/∅ (1.50D + 1.8L) U = 1/∅ [1.4(D + L + E)] U = 1/∅ (1.0D + 1.0L + 1.0E')	Same as above fc (comp) = 1.9 x 0.225 fc' = 1.54 fv (shear wall) = 0.158

Table 3.8-7

REACTOR BUILDING STEEL SUPERSTRUCTURE

Element Location/ Description	Criteria	Method of Analysis	Load Combinations	Allowable Stress (ksi)
Main Columns	Material Structural Steel ASTM Designation: A36-6% Design per AISC Manual and Specification 1963.	1. Structure analyzed as a space frame by means of computer program using stiffness method. 2. Forces due to earthquake determined by dynamic analysis. 3. Members designed by elastic methods.	D + L	Fa = -16.6 Fb = 22.0
[redacted]	Same as above	Same as above	D + L	Fb = 22.0
Wind Columns In End Walls	Same as above	Same as above	D + W'	Fa = -19.8 Fb = 27.0
[redacted]	Same as above	Same as above	D + E'	Fa = -11.0
[redacted]	Same as above	Same as above	D + E'	Fa = -11.9
[redacted]	Same as above	Same as above	D + E'	Fa = -11.2

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
			Intake Structure
[redacted]	[redacted]	[redacted]	[redacted]
2015-006			[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
			[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
2015-006 [redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

Table 3.8-8

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

Table 3.8-8

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

Table 3.8-8

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
			[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	

Table 3.8-8

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
[redacted]			
[redacted]	[redacted]	[redacted]	[redacted]

Table 3.8-8

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
			[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
[redacted]			
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]

2013-011

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
		[redacted]	
2015-006 [redacted]	[redacted]	[redacted]	[redacted]
2016-003 [redacted]	[redacted]	[redacted]	[redacted]

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
[redacted]			
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]			
[redacted]	[redacted]	[redacted]	[redacted]

Table 3.8-8

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
			[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

MASONRY WALL SURVEY SUMMARY SHEET

Wall I.D. / Building	Priority	Function	Attached Safety-Related Systems
			[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
			[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

[redacted]

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 General

The introduction to Section 3.7 discusses the organizations responsible for the seismic design of mechanical components for the DAEC. Section 3.7.3 describes analysis methods, procedures, computer codes and criteria used in the seismic analysis of nuclear steam supply system and balance-of-plant mechanical systems and components. The dynamic testing and analysis of piping systems and mechanical components are discussed in Section 3.9.2. Loading conditions, design criteria, and loading combinations for structures and equipment are discussed in Section 3.9.3. Design loads, stress limits, and allowable deformations of the CRD system are discussed in Section 3.9.4. The response of the reactor vessel internals to loads imposed during normal, upset, emergency, and faulted conditions is discussed in Section 3.9.5. The reactor vessel design is discussed in Section 5.3.3.

3.9.1.2 Reactor Internals Design Analysis

Both elastic and inelastic stress analysis techniques may be used in the design of the reactor vessel core support and reactor internal structures to show that specified stress limits are not exceeded. When an inelastic stress analysis is performed on these components, the elastic (linear) system analysis is checked to see if the analysis requires modification. The procedure is to perform a linear analysis with the stiffness of the inelastic component reduced to the stiffness value corresponding to the inelastic displacement value. A nonlinear dynamic analysis is performed in lieu of a linear analysis if the natural frequencies of the system with reduced stiffness deviate significantly from those of the unreduced system. Tables 3.9-1 through 3.9-4 summarize the stress results obtained from certain key reactor vessel, core support structure, and reactor internal structure components.

In order to ensure that the reactor and internals are adequately designed to resist the oscillatory nature of blowdown forces, a dynamic analysis was made of the reactor internal components being acted upon by the applied forces. The component natural periods were determined from a comprehensive dynamic model of the reactor pressure vessel (RPV) and internals, including hydrodynamic mass effect of the water inside the reactor pressure vessel.

The dynamic analysis was performed by determining the natural frequencies and mode shapes of the internal components under consideration. The oscillatory forces were then applied to these components to determine the dynamic load response. Negligible amplification of the applied forces has resulted.

In addition, extensive dynamic analyses of the reactor and internals were performed before the initiation of the preoperational test program as DAEC is considered to be the prototype plant for BWR/4-183 inch reactors. The results of these analyses were used to generate the allowable vibration levels during the preoperational test. The vibration data obtained during the preoperational test were analyzed in detail, and the results of the data analysis, vibration amplitudes, natural frequencies, and mode shapes were then compared to those obtained from the theoretical analysis (Reference 3). Such comparisons, in addition to data from forced oscillation tests, provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained is used in the generation of the dynamic models for vibration, seismic, and LOCA analysis.

The components that were identified as being susceptible to flow- induced vibrations (FIV) in the original design basis for the BWR/4-183 size plants, and the components which have encountered problems in actual operation (such as steam dryers and jet pump sensing lines) were re-evaluated for Extended Power Uprate (EPU). Although operation at EPU conditions reduces the margin to FIV with respect to allowable stress, the resulting maximum stresses remain within the acceptance criteria.

2011-014 As part of the Steam Dryer Tie Bar Replacement project that occurred in 2010, a comparative analysis was used to evaluate the design change. This type of analysis is described in EPRI report BWRVIP-181-A. BWRVIP-181-A has been reviewed by the NRC and approved through a Safety Evaluation Report. This analysis compared the welded connections and natural frequencies of the new design, which was machined from a solid piece, to the original design, which was an L-bracket.

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1 General Requirements

The dynamic analysis of the steam and recirculation piping and restraints with forcing functions other than earthquake is not routinely performed. The vibration of the recirculation system caused by flow and/or the recirculation pump is measured in the field, and the actual displacements are compared with allowable displacements. Allowable displacements are defined as those displacements that produce stresses whose amplitudes are less than one-half the endurance limit, as defined in the ASME Code, Section III. The dynamic loads acting on the main steam line include those associated with turbine stop valve closure, relief/safety valve lifting, and steam flow. Turbine stop valve closure has been evaluated using plants with a similar geometry and operating conditions as DAEC and has shown to be a small contributor of overall stress (~2%). The effects of relief and safety valve lifting have been evaluated by an analysis that properly and conservatively accounts for the dynamic loads. Flow-induced vibration in the main steam lines has been shown to be insignificant by actual measurements with remote instrumentation (Section 14.2.14).

Dynamic testing is not required in ASME Code, Section III, for DAEC Class 1 mechanical equipment. However, the seismic stress (obtained from criteria specified in

Section 5.2.4) of the Class 1 equipment is added to those from other loading conditions to ensure that the equipment will function as required.

Seismic Category I piping is designed to the maximum relative differential movement that could occur at the support points. Since the stresses resulting from this maximum relative differential movement are not likely to occur in phase with the maximum stresses due to dynamic response of the pipe, if any, the two were combined on a root-mean-square basis.

The seismic stresses are combined in accordance with the design rules of Section 5.2.

The dynamic analysis of the recirculation piping is discussed in Section 3.7.3.8.3.1

3.9.2.2 Piping and Pipe Restraint Structural Integrity Acceptance Criteria

The acceptance criteria that were implemented to confirm the structural integrity of the piping and pipe restraints in the event of vibratory responses were vibration measurements taken of the final installation of the recirculation piping system during preoperational tests. The actual displacements were compared with allowable displacements. Allowable displacements were defined as those displacements that produce stresses whose amplitudes are less than half of the endurance limit, as defined in the ASME Code, Section III. The ASME Code, Section III, page 99, defines the endurance limit as "...two times the S_a value at 10^6 cycles in the applicable fatigue curve...."

A portable Bentley-Nevada Vibration Meter was used for measuring the vibration amplitudes on the recirculation piping system.

During the testing of main steam turbine stop valve closure and relief valve opening, instrumentation with remote readout capability was installed to provide actual data that could be compared with calculations to ensure that displacements were within allowable limits.

As part of the preoperational test specification for the recirculation piping system, the measurement of the vibration frequencies and amplitudes was required. Frequencies and amplitudes were taken in the horizontal direction (X and Z directions) at the five locations listed below. When the total deflections (peak to peak) due to vibration were less than those shown in the right-hand column, stresses were less than half of the endurance limit, and therefore acceptable. When deflections exceeded those in the table below, a detailed analysis was made from the actual measured deflections to determine their acceptability.

Measurement Location	Maximum Deflection (peak to peak)
On suction piping near pump	0.994 in.
Midspan on suction piping	0.043 in.
Midspan on discharge between pump and circular header	0.041 in.
Circular header (any convenient location)	0.096 in.
Midspan of jet pump riser	0.018 in.

During the preoperational and startup testing program at the DAEC, all other Seismic Category I piping was observed for vibratory responses. When significant vibratory responses were observed, measurements were taken to determine actual displacements, and those measurements were compared with allowable displacements to ensure that excessive vibration did not occur.

Dynamic testing was also used to assist in the design of Class 1 equipment. The dynamic testing employed was one of the two types described below:

1. Free Vibration Test

   This test was performed on equipment whose response is dominated by the fundamental mode. The critical damping ratio and fundamental frequency were determined from this test and were used to verify or supplement calculated values used in the dynamic analysis of this equipment. This test was not used alone to demonstrate dynamic capability.

   In this test, an initial displacement or initial velocity is imparted to the equipment. The initial displacement is introduced by forcibly displacing the equipment and then suddenly releasing the force. The initial velocity is obtained by applying an impulse. Accelerometers or strain gauges are mounted on the equipment. After first ensuring that the equipment is vibrating in its primary mode, the critical damping ratio is calculated from the logarithmic decrement.

2. Forced Vibration Tests

   The equipment is mounted on a shake table or driven by an eccentric shaker. The critical damping ratios, resonant frequencies, and the equipment's functional capability are determined.

The critical damping ratio of the equipment is determined by applying a sinusoidal acceleration and measuring the forced-response curve (amplitude versus forcing frequency). The critical damping ratio is then calculated by using the half-power method, fitting a theoretical forced-response curve through the data points, or direct reading of the resonant amplification. The vibratory motion used is such that the vibratory loads equal or exceed the seismic loads represented by the applicable floor spectra. When testing is the only method used to demonstrate functional capability of equipment, the mounting conditions are simulated and the equipment is operating during and after the tests.

When the seismic testing is supplemented by analysis, the seismic stresses are added to those from normal and accident conditions in the appropriate loading combinations as described in Section 3.7 in order to ensure that the equipment will perform its required safety functions.

As an example, the tests and analysis performed on the HPCI turbine are summarized below.

The major structure of the turbine was qualified by dynamic analysis. The turbine-control-unit components were qualified by dynamic testing on a shake table with electrical and hydraulic systems functional. The-actual mounting brackets were simulated in the test mounting. Vibration in all three perpendicular axes (two horizontal and one vertical) was accomplished by orienting the equipment in three directions on a horizontal shake table. A resonant search was made from 1 to 200 Hz, and the components with substantial resonances below 33 Hz were modified before performing the functional qualification test. These modifications were applied to the standard design. This equipment was then tested with a sinusoidal input of 1.5g and then 3.0g for at least 30 sec at each of the arbitrary frequencies of 10, 15, and 23 Hz in each of the three perpendicular directions, with all systems operational. Since there were no functional failures, the equipment was deemed qualified for up to 3.0g horizontal or vertical maximum floor acceleration for all frequencies 33 Hz and below.

All tests conducted used methods and procedures delineated in the example above. In addition, the amplitudes supplied at the support brackets were equal to or greater than the levels predicted by system dynamic analyses.

All Seismic Category I equipment has been either tested or analyzed to ensure its ability to withstand the design loading requirements.

3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

3.9.3.1.1 Loading Conditions

The loading conditions, which are to be considered in addition to loads from normal conditions, are divided into three categories: upset conditions, emergency conditions, and faulted conditions. The conditions are defined as follows:

1. Normal Conditions

   Any condition in the course of operation of the plant under planned, anticipated conditions, in the absence of upset, emergency, or faulted conditions.

2. Upset Conditions

   Any deviations from normal conditions anticipated to occur often enough that the design should include a capability to withstand the conditions without operational impairment. The upset conditions include abnormal operational transients caused by a fault in a system component requiring its isolation from the system, transients due to the loss of load or power, and any system upset not resulting in a forced outage. The upset conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status.

3. Emergency Conditions

   Any deviations from normal conditions that require shutdown for the correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a result of any damage.

4. Faulted Conditions

   Those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the nuclear system may be impaired to the extent where considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities. Among the faulted conditions may be a specified earthquake for which safe shutdown is required.

3.9.3.1.2 Safety Margins

In addition to the definitions above, the meaning of these terms is expanded in quantitative probabilistic language. The purpose of this expansion is to clarify the classification of any hypothesized accident or sequence of loading events so that the appropriate structural safety margins are applied. Knowledge of the event probability is necessary to establish meaningful and adequate safety factors for structural design. The following summary illustrates the quantitative event classifications:

Generic Definition	P_{40}, 40-Year Interval Event Encounter Probability
Upset (likely)	$1.0 > P_{40} \geq 10^{-1}$
Emergency (low probability)	$10^{-1} > P_{40} \geq 10^{-3}$
Faulted (extremely low probability)	$10^{-3} > P_{40} \geq 10^{-6}$

The event probabilities currently in use as governing accident or faulted conditions are the following:

P_{40} (N and U and A_o) $= 10^{-1}$ to 10^{-2}

P_{40} (N and A_d) $= 10^{-3}$

P_{40} (N and R) $=10^{-3}$

P_{40} (N and A_d and R) $\Leftrightarrow 1.5 \times 10^{-6}$

where

N= normal loads

U= upset loads excluding earthquake

A_o = OBE

A_d = DBE

R= any pipe rupture

The minimum safety factor decreases as the event probability diminishes; and if the event is too improbable (incredible: $P_{40} \Leftrightarrow 10^{-6}$), then no safety factor is appropriate or required. The minimum safety factor used in design is that associated with an event probability of 10^{-5}.

The term SF_{min}, which appears in Table 3.9-5, is identical with the classical definition of a minimum safety factor on load or deflection. SF_{min} is related to the event probability by the following equation:

$$SF_{min} = \frac{9}{3 - \log_{10} P_{40}}$$

where $10^{-1} > P_{40} \geq 10^{-6}$

These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SF_{min} values corresponding to the current governing accident event probabilities are summarized as follows:

P_{40}	SF_{min}
10^{-1}	2.25
10^{-2}	1.80
10^{-3}	1.50
10^{-5}	1.125

3.9.3.1.3 Governing Loading Conditions and Criteria

The governing loading conditions are summarized as follows:

1. N and U and A_o

2. N and A_d

3. N and R

4. N and A_d and R

The loading criteria are covered by the ASME Code, Section III, other industrial codes, or special criteria where no applicable standards exist. These special criteria are classified into four categories: deformation limits, primary stress limits, buckling stability limits, and fatigue limits.

Table 3.9-6 summarizes the loading conditions and criteria. Table 3.9-5 lists the special loading criteria.

3.9.3.1.4 Design Criteria

These general design criteria are intended to apply to those ductile metallic structures or components that are normally designed using rational stress analysis techniques, such as pressure vessels and core support structures. The criteria may also be applied to those components or structures whose ultimate loading capability is determined by tests. These criteria are intended to supplement applicable industry design codes where necessary. Compliance with these criteria provides design safety margins that are appropriate to extremely reliable structural components when account is taken of rare event potentialities such as might be associated with a DBE or primary pressure boundary coolant pipe rupture, or a combination of events.

There are many important Seismic Category I components or pieces of equipment that are not normally designed or sized directly by stress analysis techniques. Simple stress analyses are sometimes used to augment the design of these components, but the primary design does not depend on detailed stress analysis. These components are usually designed by tests and empirical experience. Complete detailed stress analysis is currently neither meaningful nor practical for these components. Examples of such components are valves, pumps, electrical equipment, and mechanisms. Field experience and testing are used to support the design. Where the structural or mechanical integrity of components is essential to safety, the components referred to in these criteria must be designed to accommodate the DBE, the OBE, a design-basis pipe rupture, or an appropriate combination. The reliability requirements of such components cannot be quantitatively described in a general criterion because of the varied nature of each component and its specific function in the system.

3.9.3.1.5 Justification of Square Root of the Sum of the Squares (SRSS) Load Combination

Nuclear power plant structures and equipment are designed to accommodate many operational and transient-tape loads and load combinations. Included are such postulated events as the loss-of-coolant accident (LOCA) and high-intensity earthquakes. In most cases for dynamic loadings, the peak structural dynamic response is calculated using linear elastic multidegree of freedom system analysis. When two or more structural dynamic responses are considered, the peak response from each dynamic loading event is combined using the square root of the sum of the squares (SRSS) rule. Subsequently, this combined peak dynamic response is added absolutely to the calculated static or slowly varying response. The use of the SRSS method for combining dynamic responses has been restricted to separate physical events that are postulated to occur together but for which the precise timing (phasing) is unknown. Furthermore, the use of the SRSS combinations of peak responses has also been limited to physical loading events and structural systems where the dynamic responses have rapidly varying amplitudes and short duration responses. The application of SRSS combination is limited to combining earthquake, LOCA, and safety relief valve (SRV) actuation dynamic responses.

To date, technical justification for the use of the SRSS rule has centered around the fact that

1. The maximum peaks of individual responses are highly unlikely to coincide in time.

2. The large conservatism in the total design process ensures sufficient structural design margin to protect against failure.

The dynamic design margin inherent in nuclear plant structures designed to meet ASME Code (or equivalent) stress limits is related to the energy absorption capability of the component. For structures exhibiting even moderately ductile behavior, this margin is significantly greater than the ratio of static failure stress to code allowable, and therefore ensures enough additional margin to protect against structural failure even in the unlikely event that the peak combined dynamic response exceeds the SRSS value.

The use of the SRSS method is technically justifiable and represents a prudent and practical engineering approach for the combination of dynamic responses originating from earthquake, LOCA transients, and SRV discharge loadings.

Because the phasing between simultaneous dynamic responses such as earthquake and LOCA, is unknown, the SRSS rule provides a reasonable representation of resultant responses assuming their simultaneous occurrence. Combining the individual responses by the direct addition of the individual response peaks by the so-called absolute sum (ABS) method is unnecessarily conservative because of the low probability of the peaks being coincident. Hence, this "probable" response combination, accounting for the low probability of coincidence, is appropriate and thus has been the most common practice. Such a probable response combination is provided by the SRSS rule.

The SRSS rule has been used in the following three distinct engineering applications:

1. Combination of seismic modal responses.

2. Combination of three responses along each of the three earthquake directions.

3. Combination of two or more dynamic responses in a given direction.

The justification for the use of the SRSS method in these applications is based on random vibrations and probability theory, and the observation of the following:

1. Amplitude variations in time and uncertain phasing of the responses.

2. Rapid variation and the short duration of the peak responses.

Thus, the maximum peaks of individual responses are unlikely to coincide. Furthermore, the probability of the actual response combination significantly exceeding the SRSS value is exceedingly small.

In addition to the low probability of significantly exceeding the SRSS value, the consequences of exceedance are negligible because of the following:

1. The existence of margins at all steps of the design process, for example, dynamic load definition, structural models, damping parameters, and allowable stress values in industry codes.

2. The additional inherent structural dynamic reserve margins due to the short duration of the peak responses and energy absorption capability of the structure.

3.9.3.2 Pipe Support Attachment Review

In response to IE Bulletin 79-02, the design and installation of pipe support structural baseplates, the concrete expansion bolts attaching the baseplates to the concrete structure, and the masonry block walls were reviewed. By a combination of analysis and test, it was found that a small percentage of seismic supports had a smaller safety margin than had been intended. Appropriate modifications were made. A complete report of the review was submitted to the NRC by Iowa Electric letter LDR-80-283, dated October 17, 1980.

3.9.3.3 Critical Elements of the Station Piping

Critical elements of the station piping, including the connections of piping to the drywell and pressure suppression chamber of the primary containment, are designed to withstand, without overstress, the maximum forces resulting from the DBE. This has been accomplished by appropriate analyses of the important piping in systems critical to reactor safety or to safe shutdown of the station. The stresses resulting from these earthquake forces have been calculated to be within the acceptable limits for the piping materials and other associated components involved, according to appropriate ANSI and ASME Codes.

3.9.3.4 Review of Design Stress Calculations

An independent review of design stress calculations as well as assumptions used in the calculations has been performed within Bechtel and GE. In addition, the displacement of recirculation, steam, and feedwater system piping was observed in the hot condition during startup testing of the DAEC to confirm design calculations.

3.9.4 CONTROL ROD DRIVE SYSTEM

Safety Objective

The safety objective of the control rod drive (CRD) mechanical design is to insert the control rods with sufficient speed to limit fuel barrier damage.

Power Generation Objective

The power generation objective of the CRD mechanical design is to position the control rod within the core to control power generation.

Safety Design Bases

The reactivity control mechanical design meets the following safety design bases:

1. Design provides for a sufficiently rapid control rod insertion so that no fuel damage results from any abnormal operating transient.

2. Design includes positioning devices, each of which individually supports and positions a control rod.

3. Each positioning device

    a. Prevents its control rod from withdrawing as a result of a single malfunction.

    b. Is designed so that no single failure of a component will prevent its control rod from being inserted.

    c. Is individually operated so that a failure in one positioning device does not affect the operation of any other positioning device.

    d. Is individually energized when rapid control rod insertion (scram) is signaled so that a failure of power sources external to the positioning device does not prevent other positioning devices' control rods from being inserted.

    e. Is locked to its control rod to prevent undesirable separation.

Power Generation Design Basis

The reactivity control mechanical design provides for positioning the control rods to control power generation in the core.

3.9.4.1 Descriptive Information of Control Rod Drive System

The CRD mechanisms are part of the CRD system, which hydraulically operates the CRD mechanisms using processed condensate water as hydraulic fluid. The CRD mechanisms operate manually to position the control rods but act automatically to rapidly insert the control rods during abnormal conditions requiring rapid shutdown (scram).

The control rods, CRD mechanisms, and that part of the CRD hydraulic system necessary for scram are designed as Seismic Category I equipment. The piping and valves in the CRD system that are required to effect a scram and serve as part of the reactor coolant pressure boundary are designed and fabricated to high-quality levels as defined in Section 17.1.

3.9.4.1.1 Control Rod Drive Mechanisms

The CRD mechanism (drive) used for positioning the control rod in the reactor core is a double-acting, mechanically latched, hydraulic cylinder using water from the condensate storage tank as its operating fluid (see Figures 3.9-1, 3.9-2, 3.9-3, and 3.9-4). The individual drives [redacted] The drives do not interfere with refueling and are operative even when the head is removed from the reactor vessel. The drives are also readily accessible for inspection and servicing. The [redacted] location makes maximum use of the water in the reactor as a neutron shield and gives the least possible neutron exposure to the drive components. Using reactor water from the condensate storage tank as the operating fluid eliminates the need for special hydraulic fluid. Drives can use simple piston seals whose leakage does not contaminate the reactor water and does cool the drive mechanisms and their seals.

The drives can insert or withdraw a control rod at a slow, controlled rate, as well as provide rapid insertion when required. A mechanism on the drive locks the control rod in 6-in. increments of stroke over the 12-ft length of the core.

A coupling spud at the top end of the drive index tube (piston rod) engages and locks into a mating socket at the base of the control rod. The weight of the control rod is sufficient to engage and lock this coupling. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before components can be separated.

The drive holds its control rod in distinct latch positions until the hydraulic system actuates movement to a new position. An alarm annunciates if the withdraw overtravel limit on the drive is reached. Normally, the seating of the control rod at the lower end of its stroke prevents reaching the drive withdraw overtravel limit. If the drive can reach the withdrawal overtravel limit, this means the control rod is uncoupled from its drive.

The individual rod indicators, grouped in one control panel display, correspond to relative rod locations in the core. A separate, four-rod display is located just below the large display on the vertical part of the benchboard. This display presents the positions of the control rod selected for movement and of the other rods in the affected rod group. For display purposes, the control rods are considered in groups of four adjacent rods centered around a common core volume. Each group is monitored by four LPRM strings (see Section 7.6). Rod groups at the periphery of the core may have less than four rods. The four-rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A white light indicates which of the four rods is the one selected for movement.

Drive Components. Figure 3.9-2 illustrates the operating principle of a drive, and Figures 3.9-3 and 3.9-4 illustrate the drive in more detail. The main components of the drive and their functions are described below.

The drive piston is mounted at the lower end of the index tube. This tube functions as a piston rod. The drive piston and index tube make up the main moving assembly in the drive. The drive piston operates between positive end stops, with a hydraulic cushion provided at the upper end only. The piston has both inside and outside seal rings and operates in an annular space between an inner cylinder (fixed piston tube) and an outer cylinder (drive cylinder). Because the type of inner seal used is effective in only one direction, the lower sets of seal rings are mounted with one set sealing in each direction.

A pair of nonmetallic bushings prevents metal-to-metal contact between the piston assembly and the inner cylinder surface. The outer piston rings are segmented step-cut seals with expander springs holding the segments against the cylinder wall. A pair of split bushings on the outside of the piston prevents contact with the cylinder wall. The effective piston area for downtravel, or withdrawal, is approximately 1.2 $in.^2$ versus 4.0 $in.^2$ for uptravel, or insertion. This difference in driving area tends to balance the control rod weight and ensures a higher force for insertion than for withdrawal.

The index tube is a long hollow shaft made of nitrided Type 304 stainless steel. Circumferential locking grooves, spaced every 6 in. along the outer surface, transmit the weight of the control rod to the collet assembly.

The collet assembly serves as the index tube locking mechanism. It is located in the upper part of the drive unit. This assembly prevents the index tube from accidentally moving downward. The assembly consists of the collet fingers, a return spring, a guide cap, a collet housing (part of the cylinder, tube, and flange), and the collet piston seals.

Locking is accomplished by six fingers mounted on the collet piston at the top of the drive cylinder. In the locked or latched position, the fingers engage a locking groove in the index tube.

The collet piston is normally held in the latched position by a force of approximately 150 lb supplied by a spring. Metal piston rings are used to seal the collet piston from reactor vessel pressure. The collet assembly will not unlatch until the collet fingers are unloaded by a short, automatically sequenced, drive-in signal. A pressure, approximately 180 psi above reactor vessel pressure, must then be applied to the collet piston to overcome spring force, slide the collet up against the conical surface in the guide cap, and spread the fingers out so they do not engage a locking groove. The collet piston is nitrided to minimize wear.

A guide cap is fixed in the upper end of the drive assembly. This member provides the unlocking cam surface for the collet fingers and serves as the upper bushing for the index tube. It is nitrided to provide a compatible bearing surface for the index tube.

The center tube of the drive mechanism forms a well to contain the position indicator probe. This probe is an aluminum extrusion attached to a cast aluminum housing. Mounted on the extrusion are hermetically sealed, magnetically operated, position indicator switches. Each switch is sheathed in a braided glass sleeve, and the entire probe assembly is protected by a thin-walled stainless steel tube. The switches are actuated by a ring magnet located at the bottom of the drive piston. The drive piston, piston tube, and indicator tube are all of nonmagnetic stainless steel, allowing the individual switches to be operated by the magnet as the piston passes. One switch is located at each position corresponding to an index tube groove, thus allowing indication at each latching point. An additional switch is located at each midpoint between latching points to indicate the intermediate positions during drive motion. Thus, indication is provided for each 3 in. of travel. Duplicate switches are provided for the full-in and full-out positions. One additional switch (an overtravel switch) is located at a position below the normal full-out position. Because the limit of downtravel is normally provided by the control rod itself as it reaches the backseat position, the drive can pass this position and actuate the overtravel switch only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at any time during plant operation.

A flange and cylinder assembly is made up of a heavy flange welded to the drive cylinder. A sealing surface on the upper face of this flange makes the seal to the drive housing flange. Teflon-coated, stainless steel rings are used for these seals. In addition to the reactor vessel seal, the two hydraulic control lines to the drive are sealed at this face. A drive can thus be replaced without removing the control lines, which are permanently welded into the housing flange. The drive flange contains the integral ball, or two-way, check (ball-shuttle) valve. This valve directs either the reactor vessel pressure or the driving pressure, whichever is higher, to the underside of the drive piston. Reactor vessel pressure is admitted to the valve from the annular space between the drive and drive housing through passages in the flange. A strainer is provided to intercept foreign material at this point.

Water used to operate the collet piston passes between the outer tube and the cylinder tube. The inside of the cylinder tube is honed to provide the surface required for the drive piston seals.

Both the cylinder tube and outer tube are welded to the drive flange. The tops of these tubes have a sliding fit to allow for differential expansion.

The upper end of the index tube is threaded to receive a coupling spud. The coupling (see Figure 3.9-1) accommodates a small amount of angular misalignment between the drive and the control rod. Six spring fingers allow the coupling spud to enter the mating socket on the control rod. A plug then enters the spud and prevents uncoupling.

Two means of uncoupling are provided. With the reactor vessel head removed, the lock plug can be raised against the spring force of approximately 50 lb by a rod extending up through the center of the control rod to an unlocking handle located above the control rod velocity limiter. The control rod, with the lock plug raised, can then be lifted from the drive.

The lock plug can also be pushed up from below, if it is desired to uncouple a drive without removing the reactor pressure vessel head for access. In this case, the central portion of the drive mechanism is pushed up against the uncoupling rod assembly, which raises the lock plug and allows the coupling spud to disengage the socket as the drive piston and index tube are driven down.

The control rod is heavy enough to force the spud fingers to enter the socket and push the lock plug up, allowing the spud to enter the socket completely and at the plug to snap back into place. Therefore, the drive can be coupled to the control rod using only the weight of the control rod. However, with the lock plug in place, a force in excess of 50,000 lb is required to pull the coupling apart.

Materials of Construction. Factors that determine the choice of construction materials are discussed below.

The index tube must withstand the locking and unlocking action of the collet fingers. A compatible bearing combination must be provided that is able to withstand moderate misalignment forces. The reactor environment limits the choice of materials suitable for corrosion resistance. The column and tensile loads can be satisfied by an annealed 300 series stainless steel. The wear and bearing requirements are provided by Malcomizing the completed tube. To obtain suitable corrosion resistance, a carefully controlled process of surface preparation is used.

The coupling spud is made of Inconel 750 that is aged for maximum physical strength and the required corrosion resistance. Because misalignment tends to cause chafing in the semispherical contact area, the entire part is protected by a thin vapor-deposited chromium plating (electrolized). This plating also prevents the galling of the threads attaching the coupling spud to the index tube.

Inconel 750 is used for the collet fingers, which must function as leaf springs when cammed open to the unlocked position. Colmonoy 6 hard facing provides a long-wearing surface, adequate for design life, to the area contacting the index tube and unlocking cam surface of the guide cap.

Graphitar 14 is selected for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water lubricated.

Because a loss of seal strength is experienced at higher temperatures, the drive is supplied with cooling water to hold the temperature below 250°F. The Graphitar is relatively soft, which is advantageous when an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until worked smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel.

All drive components exposed to reactor vessel water are made of AISI 300 series stainless steel except for the following:

1. Seals and bushings on the drive piston and stop piston are Graphitar 14.

2. All springs and members requiring spring action (collet fingers, coupling spud, and spring washers) are made of Inconel 750.

3. The ball check valve is a Haynes Stellite cobalt-base alloy.

4. Elastomeric O-ring seals are ethylene propylene.

5. Collet piston rings are Haynes 25 alloy.

6. Certain wear surfaces are hard faced with Colmonoy 6.

7. Nitriding by a proprietary new Malcomizing process, electrolizing (a vapor deposition of chromium), and chromium plating are used in certain areas where resistance to abrasion is necessary.

8. The drive piston head is made of Armco 17-4PH.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of the ASME Code, Section III.

3.9.4.1.2 Control Rod Drive Hydraulic System

The CRD hydraulic system (Figure 3.9-5) supplies and controls the pressure and flow to and from the drives. One supply subsystem supplies water to all hydraulic control units (HCU) at the correct flow. Each hydraulic control unit controls the flow to and from one drive. The water discharged from the drives during a scram flows through the hydraulic control units to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation returns to the reactor vessel through a reverse flow path involving the insert exhaust directional control valves of nonactuated CRD hydraulic control units.

Originally, the water discharged from a drive during normal rod positioning operation flowed back to the reactor vessel through a control rod drive return line.

However, in order to eliminate the potential for thermal fatigue cracking of the return line nozzle at the reactor vessel, a blind (spectacle) flange was installed on the return line in an area of the reactor building that is accessible under most accident conditions. If emergency conditions render it desirable to provide water to the reactor vessel via the return line, the blind flange can be removed.

CRD Supply and Discharge Subsystems. The CRD hydraulic supply and discharge subsystems (Figures 3.9-5, 3.9-6 and 3.9-7) control the pressure and flow required to operate the CRD mechanisms. These hydraulic requirements, identified by the function they perform, are as follows:

1. An accumulator charging pressure of approximately 1400 to 1500 psig is required. Flow to the accumulators is required only during scram reset or system startup.

2. Drive pressure of approximately 260 psi above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required.

3. Cooling water to the drives is supplied at approximately 20 psi above reactor vessel pressure. (Cooling water can be interrupted for short periods without damaging the drive.)

4. The scram discharge header pipe is sized to receive and contain all the water discharged by the drives during a scram. A scram discharge volume of at least 3.34 gal per drive is required. The scram discharge volume normally contains air at atmospheric pressure, except during scram when it is filled with water and air and until the scram signal is cleared. The scram discharge volume will reach reactor pressure following a scram.

5. The scram valve pilot air header supplies compressed air to the hydraulic control units for operation of the scram inlet valves and scram exhaust valves through the scram pilot valve. In the event of a scram, the header is depressurized by action of [redacted] which are operated from the reactor protection system trip system and provide a backup to the scram pilot valves. In the event of an ATWS-ARI actuation, the header would also be depressurized by action of [redacted] which are operated from the ATWS-ARI system as described in section 7.2.3.

The CRD hydraulic supply and discharge systems provide the required functions with the pumps, filters, valves, instrumentation, and piping shown in Figure 3.9-5 and described in the following paragraphs.

Duplicate components are included where necessary, to ensure continuous system operation if an inservice component requires maintenance.

One supply pump pressurizes the system with deaerated, low conductivity water from the condensate reject line.

One parallel spare pump is provided for standby. Each pump is installed with a suction strainer. A discharge stop-check valve prevents bypassing flow through the nonoperating pump. Flow is continuously diverted through a minimum-flow bypass connection into the condensate storage tank. Consequently, pump overheating is avoided if the pump discharge is inadvertently closed.

Two parallel filters remove foreign material larger than 50 μm absolute (25 μm nominal) from the hydraulic supply subsystem water. Only one filter is in operation at any given time. The filter installation allows the addition of 2 ft minimum thickness of temporary external radiation shielding for personnel protection. A local differential-pressure indicator and control room alarm monitor the filter element as it collects foreign material. A strainer in each filter discharge line protects the hydraulic system in the event of filter-element failure.

Accumulator charging pressure is established by the discharge pressure of the system supply pump and is maintained by the system flow control valve during normal operation. During scram, the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to “run out" (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow sensing system upstream of the accumulator charging header detects high flow and closes the flow control valve. This action maintains increased flow through the charging water header.

Pressure in the charging header is monitored in the control room with a pressure indicator and high pressure alarm. Individual accumulators have low-pressure alarms.

During normal operation, the flow control valve maintains a constant system flow rate. This flow is used for drive flow, drive cooling, and system stability.

Drive water pressure required in the drive header is maintained by the drive pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 6 gpm (the sum of the flow rates required to insert and withdraw a control rod) normally passes from the drive water pressure stage through two solenoid-operated stabilizing valves (arranged in parallel) and then goes into the return line downstream of the cooling pressure control valve. The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by closing the appropriate stabilizing valve. Thus, flow through the drive pressure control valve is always constant.

Flow indicators in the drive water header and in the line downstream of the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary.

Differential pressure between the reactor vessel and the drive pressure stage is indicated in the control room.

A small amount (nominally .012 gpm) of CRD drive water is diverted to each of the two ambient reference columns for the reactor water level instrumentation to purge the columns of non-condensible gases (see Section 7.6.4.6).

The cooling water header is located upstream from the cooling pressure control valve. Water not required for drive cooling passes through the cooling pressure control valve to the reactor vessel. The cooling pressure control valve is manually adjusted from the control room to produce the required cooling water pressure.

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive pressure control valve and the cooling pressure control valve can maintain their required pressure independent of reactor pressure. Changes in the setting of the pressure control valves are required only to adjust for the changes in the cooling requirements of the drives, as their seal characteristics change with time. A flow indicator in the control room monitors cooling water flow. A differential-pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long-term exposure to reactor temperatures. The temperature of each drive is recorded in the control room, and excessive temperatures are annunciated.

The scram discharge volume is used to limit the loss of reactor water discharged from all the drives during a scram. It is also used to contain the reactor water that leaks past the drives following a scram. This volume is provided in the scram discharge header.

During normal plant operation, each scram discharge header is empty, and the drain and vent valves are open. Position indicator switches on the drain and vent valves actuate valve lights in the control room.

During a scram, the scram discharge volume partly fills with water discharged from above the drive pistons. While scrammed, the CRD seal leakage from the reactor continues to flow into the scram discharge volume until the discharge volume pressure equals the reactor vessel pressure. A check valve in each hydraulic control unit prevents reverse flow from the scram discharge header volume to the drive.

A test pilot valve allows the discharge volume vent and drain valves to be tested without disturbing the reactor protection system or the ATWS-ARI system. Closing the discharge volume valves allows the outlet scram valve seats to be leak tested by timing the accumulation of leakage inside the scram discharge instrument volume.

The initial design of the scram discharge volume included six level-measuring magnetrol float switches to prevent operating the reactor without sufficient free volume present to accommodate scram discharge. In response to IE Bulletin 80-17 (Reference 1), the scram discharge volume was modified by adding a redundant and diverse set of six thermally-actuated liquid level switches and a redundant set of vent and drain valves. The modifications were implemented to ensure that there will be no accumulation of water in the scram discharge volume and that there will be no blowdown if a single vent or drain valve does not close on a scram signal.

Both sets of level switches are set at the same setpoint levels and provide identical functions (scram, rod withdrawal block, and alarm). At the first (lowest) level, one level switch initiates an alarm. At the second level, one level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, the four level switches (two for each reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is still present in the scram discharge volume to receive the scram discharge water. After a scram and before reactor operation can be resumed, the level-measuring switches must be cleared by draining the scram discharge volume. When the initial scram signal is cleared from the reactor protection system, the scram discharge instrument volume signal is overridden with a key-lock override switch, and the scram discharge volume is drained.

Additional control room information regarding the sequence of events during and after a scram is obtained by computer logging of changes in the status of the scram discharge volume vent and drain valves (two vent valves and two drain valves) and 10 of the 12 scram discharge volume level measuring switches (the two "alarm" switches are not monitored).

Piping and equipment pressure parts in the CRD hydraulic supply and discharge subsystems are designed and fabricated in conformance with the guidance as described in Sections 3.1 and 3.2.

<u>Hydraulic Control Units</u>. Each hydraulic control unit furnishes pressurized water, on signal, to a drive unit. The drive then positions its control rod as required. The operation of the electrical system that supplies scram and normal control rod positioning signals to the hydraulic control unit is described in Section 7.7.3.

The basic components in each hydraulic control unit are manual, pneumatic, and electric valves; an accumulator; related piping; electric connections; filter; and instrumentation (see Figures 3.9-5, 3.9-6, and 3.9-8). These components and their functions are described below.

The <u>insert drive valve</u> is solenoid operated and opens on an insert signal. The valve supplies drive water to the bottom side of the main drive piston.

The <u>insert exhaust valve</u> also opens by solenoid on an insert signal. The valve discharges water from above the drive piston to the exhaust water header.

The withdraw drive valve is solenoid-operated and opens on a withdraw signal. The valve supplies drive water to the top of the drive piston.

The solenoid-operated withdraw exhaust valve opens on a withdraw signal and discharges water from below the main drive piston to the exhaust header. It also serves as the settle valve. The valve opens following any normal drive movement (insert or withdraw) to allow the control rod and its drive to settle back more quickly into the nearest latch position.

The speed control valves regulate the control rod insertion and withdrawal rates during normal operation. They are manually adjustable flow control valves used to regulate the water flow to and from the volume beneath the main drive piston. A correctly adjusted valve does not require readjustment except to compensate for changes in piston seal leakage.

The scram pilot valve is operated from the reactor protection system trip system. A single scram pilot valve controls both the scram inlet valve and the scram exhaust valve. The scram pilot valve is a three-way, dual-solenoid-operated, in-line, normally energized valve. On a loss of electric signal to the solenoid coils, such as the loss of external ac power, both coils deenergize and the exhaust port opens on the pilot valve. The pilot valve (Figures 3.9-5 and 3.9-6) is designed so that the trip system signal must be removed from both solenoids before air pressure can be discharged from the scram valve operators. This prevents the inadvertent scram of a single drive in the event of a failure of one of the solenoid pilot valves.

The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. This quick-opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the control room as soon as the valve starts to open.

The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston. The exhaust valve opens faster than the inlet valve because of a large spring in the valve operator. Otherwise the valves are similar.

The scram accumulator stores sufficient energy to fully insert a control rod independent of any other source of energy. The accumulator consists of a water volume pressurized by nitrogen. A piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents a loss of water pressure in the event supply pressure is lost.

During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. The loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the control room.

To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A float-type level switch actuates an alarm if water leaks past the piston barrier and collects in the accumulator instrumentation block.

3.9.4.1.3 Control Rod Drive System Operation

The CRD system performs rod insertion, rod withdrawal, and scram. These operational functions are described below.

Rod insertion is initiated by a signal from the operator to the insert valve solenoids. This signal causes both insert valves to open. The insert drive valve applies reactor pressure plus approximately 100 psi to the bottom of the drive piston. The insert exhaust valve allows water from above the drive piston to discharge to the exhaust header.

As is illustrated in Figure 3.9-2, the locking mechanism is a ratchet-type device and does not interfere with rod insertion. The speed at which the drive moves is determined by the pressure drop through the insert speed control valve, which is set for approximately 4 gpm for a shim speed (nonscram operation) of 3 in./sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through, and pressure drop across, the insert speed control valve will decrease; the full differential pressure (260 psi) will then be available to cause continued insertion. With 260-psi differential pressure acting on the drive piston, the piston exerts an upward force of 1040 lb.

Rod withdrawal is, by design, more involved than insertion. The collet finger (latch) must be raised to reach the unlocked position (see Figure 3.9-2). The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding paragraph) for approximately 1 sec. The withdraw valves are then opened, applying driving pressure above the drive piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the piston raises, the collet fingers are cammed outward, away from the index tube, by the guide cap.

The pressure required to release the latch is set and maintained at a level high enough to overcome the force of the latch return spring plus the force of reactor pressure opposing movement of the collet piston. When this occurs, the index tube is unlatched and free to move in the withdraw direction. Water displaced by the drive piston flows out through the withdraw speed control valve, which is set to give the control rod a shim speed of 3 in./sec. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdraw switch.

During a scram, the scram pilot valve and scram valves are operated as previously described. With the scram valves open, accumulator pressure is admitted under the drive piston, and the area over the drive piston is vented to the scram discharge volume.

The large differential pressure (initially approximately 1500 psi and always several hundred psi, depending on reactor vessel pressure) produces a large upward force on the index tube and control rod. This force gives the rod a high initial acceleration and provides a large margin of' force to overcome any possible friction. After the initial acceleration is achieved, the drive continues at a nearly constant velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke, the piston seals close off the large passage (buffer orifices) in the stop piston tube, and the drive slows.

Each drive requires approximately 2.5 gal of water during the scram stroke. The water capacity in each drive accumulator is adequate to complete a scram in the required time at low reactor vessel pressure. At higher reactor vessel pressures, the accumulator is assisted on the upper end of the stroke by reactor vessel pressure acting on the drive via the ball check (shuttle) valve. As water is forced from the accumulator, the accumulator discharge pressure falls below reactor vessel pressure. This causes a check valve, located in the drive flange, to shift its position to admit reactor pressure under the drive piston. Thus, reactor vessel pressure furnishes the force needed to complete the scram stroke at higher reactor vessel pressures. When the reactor vessel reaches full operating pressure, the accumulator is actually not needed to meet scram time requirements. With the reactor at 1000 psig and the scram discharge volume at atmospheric pressure, the scram force without an accumulator exceeds 1000 lb.

Allowable scram insertion times are given in the Technical Specifications.

3.9.4.1.4 Control Rod Drive Housing Supports

The CRD housing supports (Figure 3.9-9) protect against additional damage to the nuclear system process barrier or damage to the fuel barrier by preventing any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

Safety Design Bases

1. Control rod downward motion is limited, following a postulated CRD housing failure, so that any resulting nuclear transient could not be sufficient to cause fuel damage.

2. Clearance is provided between the housings and the supports to prevent vertical contact stresses due to their respective thermal expansion during plant operation.

Description

The control rod housing supports are illustrated in Figure 3.9-9. [redacted]

Hanger rods, about 10 ft long by 1.75 in. in diameter, are supported from the beams on stacks of disk springs that compress about 2 in. under the design load.

The support bars are bolted between the bottom ends of the hanger rods. The spring pivots at the top and the beveled loose fitting ends on the support bars prevent substantial bending movement in the hanger rods.

Individual grids rest on the support bars between adjacent beams. Because a single-piece grid would be difficult to handle in the limited work space and because it is necessary that control rod drives, position indicators, and incore instrumentation components be accessible for inspection and maintenance, each grid is designed to be assembled or disassembled in place. Each grid assembly is made from two grid plates, a clamp, and a bolt. The top part of the clamp acts as a guide to ensure that each grid is correctly positioned directly below the respective CRD housing that it would support in the postulated accident.

When the support bars and grids are installed, a gap of about 1 in. at room temperature (approximately 70ºF) is provided between the grid and the bottom contact surface of the CRD flange. During system heatup, this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, the gap is approximately 0.25 in.

In the postulated CRD housing failure, the CRD housing supports are loaded when the lower contact surface of the CRD flange contacts the grid. The resulting load is then carried by two grid plates, two support bars, four hanger rods, their disk springs, and two adjacent beams.

To provide a housing support structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses were taken as 90% of yield, and the shear stress as 60% of yield. These are 1.5 times the corresponding AISC allowable stresses of 60% and 40% of yield. This stress criterion is considered desirable for this application and adequate for the "once in a lifetime" loading condition.

For mechanical design purposes, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with an internal pressure of 1250 psig (reactor vessel design pressure) acting on the area of the separated housing. The weight of the separated housing, control rod drive, and blade, plus the pressure force, is approximately 35,000 lb. This force is multiplied by a factor of three for impact, conservatively assuming the housing travels through a 1-

in. gap before contacting the supports. The total force (10^5 lb) is then treated as a static load in design formulas.

Safety Evaluation

The downward travel of CRD housing and its control rod following the postulated housing failure is the sum of the compression of the disk springs under dynamic loading and the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the approximate 2-in. spring compression plus approximately a 1-in. gap. If the reactor were hot and pressurized, the gap would be approximately 0.25 in. and the spring compression slightly less than in the cold condition. In either case, the control rod movement following a housing failure is limited substantially below one drive "notch" movement (6 in.). The nuclear transient from sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not result in a transient sufficient to cause damage to any radioactive material barrier.

The CRD housing supports are in place any time the reactor is to be operated. The housing supports may be removed when the reactor is in the shutdown condition even when the reactor is pressurized, because all control rods are then inserted. Even if a control rod is ejected under the shutdown condition, the reactor remains subcritical, because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

At plant operating temperature, a gap of approximately 0.25 in. is maintained between the CRD housing and the supports; at lower temperatures the gap is greater. Because the supports do not come in contact with any of the CRD housing, except during the postulated accident condition, vertical contact stresses are prevented.

Inspection and Testing

When the reactor is in the shutdown mode, the CRD housing supports may be removed to permit inspection and maintenance of the control rod drives. When the support structure is reinstalled, it is inspected for proper assembly, particular attention being given to ensure that the correct gap between the CRD flange lower contact surface and the grid is maintained.

3.9.4.2 Applicable Control Rod Drive System Design Specifications

As discussed in Section 3.9.5.1.3, the guide tubes are designed as lateral guides for the control rods and as vertical support for a fuel support casting and four fuel assemblies. The 89 guide tubes are made of Type 304 stainless steel. The guide tubes are straight cylindrical tubes whose nominal dimensions are as follows:

Inside diameter, 10.420 in.
Wall thickness, 0.165 in.

Length, 159 in.

Significant limits are as follows:

Minimum wall thickness, 0.144 in.
Circular within, 0.030 in.

The guide tube can be subjected to any or all of the following loads:

1. Inward load due to pressure differential.
2. Lateral loads due to flow across the guide tube.
3. Deadweight.
4. Seismic.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9.4.3.1 Pressure Differential Loading

The pressure differential loading on the guide tube is evaluated under normal, emergency, upset and faulted conditions, such as recirculation and steamline break accidents. See Section 3.9.5.2 for a more detailed discussion of this evaluation.

3.9.4.3.2 Lateral Loading

During normal operation, the 16 jet pumps spaced around the circumference of the shroud support ring will be discharging the recirculation flow into the lower plenum. This flow will pass through the forest of guide tubes and enter the core. The lateral loads on the guide tubes will be very small. Even if it is assumed that all the recirculation flow passes through the outer ring of the guide tubes over only a 6 ft vertical height (the guide tubes are 159 in. long), and if it is further assumed that all the velocity head of the water flowing between the tubes is lost, the lateral load would be less than 1 psi. This is negligible in terms of guide tube bending stress (approximately 230 psi).

This analytical conclusion is supported by observations in the field. With the reactor head off but with rated core volumetric flow, the flow distribution in the [redacted] core was measured by placing a flow measuring device on top of selected fuel assemblies[2]. No significant maldistribution of flow was observed. From these observations, it can be concluded that no significant radical pressure distribution exists in the lower plenum since any such distribution would have resulted in higher flow in the fuel assemblies at the core periphery than those in the central regions of the core. Since there was full-rated volumetric flow during the tests, it can be concluded that during reactor power operation there will not be a significant pressure drop across the guide tubes.

The lateral loading that could occur during LOCAs has been examined for the entire spectrum of credible accidents, and it is concluded that there is no accident that can generate significant lateral guide tube loads.

1. Steam-Line Breaks

   Figure 3.9-14 shows the flow patterns in the lower plenum during the course of a steam-line break. Early in the transient, the recirculation system will continue to provide flow into the lower plenum but at a rate less than normal flow. The lateral load on the guide tubes will thus be less than that occurring during normal operation.

   When the vessel pressure has dropped below the saturation value of the fluid in the lower plenum, steam voids will start to form and fluid will leave the region via the core and the jet pump diffusers. (Cavitation effects will have caused the recirculation system drive pumps to become inoperative, thus allowing reverse flow in the jet pump diffusers.) The flow in the lower plenum is essentially axial with respect to the guide tubes, and no significant lateral loads will be generated.

2. Recirculation-Line Break

   The recirculation line of a jet-pump-equipped BWR does not connect directly to the lower plenum of the reactor, and it is because of this geometry that the rupture of a recirculation line does not produce significant lateral loads on the guide tubes. Figure 3.9-15 shows the flow patterns in the-lower plenum during a recirculation-line break.

   Immediately following the accident, the flow in the jet pump diffusers associated with the broken loop will reverse. The flow from the jet pumps of the unbroken loop will increase to ≅ 120% of rated flow; this flow will go partly to providing the reverse flow in the 8 jet pumps associated with the broken loop and partly to core flow. The maximum lateral loadings on the guide tubes will occur as a result of the 120% recirculation flow from the one pump in the unbroken side. As discussed above, no significant lateral loads exist during normal operation when 100% of rated flow * is being generated by the recirculation system; thus, it can be concluded that, during the early phases of a recirculation-line break when 120% of rated flow from one pump will be crossing some guide tubes, there will be no significant lateral loads.

   When all the liquid outside the core shroud has been discharged through the break, the blowdown flow will change to steam. There will be a rapid increase in the vessel depressurization rate at this time, which will cause steam voids to be generated throughout the lower plenum inventory. This void creation will force flow out of the lower plenum via both the core and the jet pump diffusers. This situation is very similar to the conditions in the lower plenum during the latter stages of the steam-line break. As discussed above, the flow in the lower plenum is essentially axial with respect to the guide tubes, and no significant lateral loads will be generated. Thus, the recirculation-line break will not result in large lateral guide tube loads.

* The impact of Increased Core Flow (105% of rated) is judged to be minimal also.

3.9.4.3.3 Deadweight Loading

The column load on the guide tube changes only slightly with time. During normal operating conditions, the column load is the deadweight of the fuel and fuel support casting minus the force due to the core pressure drop acting on the fuel bundles. The column load for a main steam line break decreases further due to the increase in pressure differential across the fuel bundles. The pressure differential for a recirculation-line break remains essentially the same as normal operating conditions. To simplify the analysis, it was assumed that the column load is 2762 lb and is never decreased by any pressure differential operating on this area. The column load versus time curve is then a straight line at 2762 lb.

The 2762 lb is derived as follows:

Fuel bundles = 675 lb (4) =	2700 lb
Fuel casting = 62 lb (1) =	<u>62 lb</u>
Total	2762 lb

3.9.4.3.4 Control Rod Displacement

As mentioned in Chapter 15, the reactor is shut down by void formation in the core in the event of a recirculation-line break. However, the control rod insertion versus time will meet the Technical Specifications limits.

For the recirculation-line break, the drywell pressure would reach the 2-psig scram setpoint in less than 0.1 sec; thus, control rod movement would start at this time.

3.9.4.3.5 Summation of Maximum Applied Loads

There are two primary modes of failure to be considered in the guide tube analysis: (1) excessive stress and (2) excessive elastic deformation. First, it will be shown that the allowable stress limit will not be exceeded and, second, that the elastic deformations of the guide tube never are great enough to cause the free movement of the control rod to be jeopardized.

The first mode of failure is evaluated for the faulted condition.

The evaluation of the second mode of failure is based on clearance reduction between the guide tube and the control rod. The minimum allowable clearance is 0.099 inches. This assumes maximum ovality and minimum diameter of the guide tube, and the maximum control rod dimension. Referring to Figure 3.9-18, it can be seen that, the clearance between the control rod and the guide tube would only be reduced by approximately 0.016 inches for a 40 psi pressure differential (which is greater than the pressure differential under faulted conditions). Since the calculated maximum displacement does not exceed the minimum allowable clearance, failure due to excessive elastic deformation will not occur.

Two types of instability also were considered in the analysis of guide tube design. The first was the classic instability associated with vertically loaded columns (i.e. buckling). The second was the diametral collapse of a circular tube under external to internal differential pressure.

These evaluations concluded that there is significant margin between the calculated loads due to either buckling or differential pressure and the allowable stresses. Thus, the guide tube is not an unstable column.

To demonstrate the adequacy of the guide tube design, a series of sensitivity studies have been conducted. Parameters evaluated are guide tube ovality and wall thickness and their effect on yield pressure and radial deflection. The results of the sensitivity study are shown in Figures 3.9-16, 3.9-17, and 3.9-18.

These figures demonstrate the sensitivity of a BWR guide tube to changes in the manufacturing tolerances. Figure 3.9-17 shows the sensitivity of the yield pressure calculation to initial ovality of the guide tube assuming minimum wall thickness. Figure 3.9-16 shows the sensitivity of the yield pressure calculation to guide tube wall thickness assuming the maximum initial ovality. Figure 3.9-18 shows the radial deflection for a range of differential pressures for two given initial ovalities and minimum wall thickness.

3.9.4.4 <u>Control Rod Drive Performance Assurance Program</u>

3.9.4.4.1 Development and Design Conformation Tests

The development drive (one prototype) testing to date included more than 5000 scrams and approximately 100,000 latching cycles. One prototype was exposed to simulated operating conditions for 5000 hr. These tests demonstrated the following:

1. The drive easily withstands the forces, pressures, and temperatures imposed.

2. Wear, abrasion, and corrosion of the nitrided Type 304 stainless parts are negligible. Mechanical performance of the nitrided surface is superior to that of materials used in earlier operating reactors.

3. The basic scram speed of the drive has a satisfactory margin above

minimum plant requirements at any reactor vessel pressure.

4. Usable seal lifetimes in excess of 1000 scram cycles can be expected. Operating experience on BWR plants to date confirms the above development test findings.

3.9.4.4.2 Factory Quality Control Tests

Quality control of welding, heat treatment, dimensional tolerances, material verification, and similar factors is maintained throughout the manufacturing process to ensure reliable performance of the mechanical reactivity control components. Some of the quality control tests performed on the control rods, CRD mechanisms, and hydraulic control units are listed below:

1. Control rod absorber tube tests:

    a. Material integrity of the tubing and end plug is verified by ultrasonic inspection.

    b. The boron-10 fraction of the boron content of each lot of boron carbide is verified.

    c. Weld integrity of the finished absorber tubes is verified by helium leak testing.

2. CRD mechanism tests:

    a. Pressure welds on the drives are hydrostatically tested in accordance with the ASME Code, Section III, Class A vessels.

    b. Electric components are checked for electrical continuity and resistance to ground.

    c. Drive parts that cannot be visually inspected for dirt are flushed with filtered water at high velocity. No significant foreign material is permitted in effluent water.

    d. Seals are tested for leakage to demonstrate correct seal operation.

    e. Each drive is tested for shim motion, latching, and control rod position indication.

    f. Each drive is subjected to cold scram tests at various reactor pressures to verify correct scram performance.

3. Hydraulic control unit tests:

a. Hydraulic systems are hydrostatically tested in accordance with Section 5.2.

b. Electric components and systems are tested for electrical continuity and resistance to ground.

c. Correct operation of the accumulator pressure and level switches is verified.

d. The unit's ability to perform its part of a scram is demonstrated.

e. Correct operation and adjustment of the insert and withdrawal valves are demonstrated.

3.9.4.4.3 Operational Tests

After installation, all rods, hydraulic control units, and drive mechanisms are tested through their full range for operability.

Each time a control rod is withdrawn a notch during normal operation, the operator can observe incore monitor indications to verify that the control rod is following the drive mechanism. All control rods that are partially withdrawn from the core can be tested for rod-following by inserting or withdrawing the rod one notch and returning it to its original position, while the operator observes incore monitor indications.

To make a positive test of the control-rod-to-control-rod-drive coupling integrity, the operator can withdraw a control rod to the end of its travel and then attempt to withdraw the drive to the overtravel position. The failure of the drive to reach the overtravel position demonstrates rod-to-drive coupling integrity.

Hydraulic supply subsystem pressures can be observed from instrumentation in the control room. Scram accumulator pressures can be observed on the nitrogen pressure gauges. Surveillance requirements for the control rod drive system are included in the Technical Specifications.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

3.9.5.1 Design Arrangements

The reactor vessel internals include the following components:

Startup neutron sources
Core shroud

Shroud head and steam separator assembly
Core support (core plate)
Top guide
Fuel support pieces
Control rod guide tubes
Jet pump assemblies
Steam dryers
Feedwater spargers
Differential pressure and liquid control line
Incore flux monitor guide tubes
Surveillance sample holders

The overall arrangement of the internals within the reactor vessel is shown in Figures 3.9-19 and 3.9-20.

Although it was not mandatory, the manufacturer of the reactor vessel internals used weld procedures and welders qualified in accordance with the intent of ASME Code, Section IX. This means that welding techniques, procedures, methods, qualifications, and tests were used that ensured that the design integrity and design requirements were maintained on the equipment items to which each level of welding control was applied. All facets of reactor internals that form a primary pressure boundary were welded per ASME Code, Sections III and IX. Much of the non-pressure boundary reactor internals were welded under ASME Code, Section IX, requirements also. The jet pump instrumentation lines were welded per Mil-Specification MIL-T-5021C, which meets or exceeds ASME Code, Section IX, requirements. In other cases, proprietary welding documents containing requirements excerpted from ASME Code, Section IX; AWS Standards; and GE processes and procedures were used. Examples of ASME Code, Section IX, variables that are excepted by some GE welding documentation are for procedures described in ASME Code, Section IX, paragraphs V-2b-1, V-2b-2, V-2b-3, V-2d-1, V-2d-2, V-2e-1, V-3, V-7a, and V-7b, where some of these are not applicable due to the process that GE used, or they were not required to ensure the design intent of the components to which these weld documents were applied. Also excepted from ASME Code, Section IX, were some welder qualification paragraphs such as those that called for weld specimens to be radiographed or cross sectioned, whereas GE required cross sectioning only. The quality control aspects of all welding specifications used on reactor internals are available for quality assurance audit as required.

3.9.5.1.1 Core Structure

The core structure surrounds the active core of the reactor and consists of the core shroud, shroud head and steam separator assembly, core support, and top guide. This structure is used to form partitions within the reactor vessel, to sustain pressure differentials across the partitions, to direct the flow of the coolant water, and to locate laterally and support the fuel assemblies, control rod guide tubes, and steam separators. Figure 3.9-20 shows the reactor vessel internal flow paths.

Core Shroud

The core shroud is a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus, thus providing a floodable region following a recirculation-line break. The volume enclosed by the core shroud is characterized by three regions, each with a different shroud diameter. The upper shroud has the largest diameter and surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section has an intermediate diameter and is bounded at the bottom by the core support. The lower shroud, surrounding part of the lower plenum, has the smallest diameter and, at the bottom, is welded to the reactor vessel shroud support ring (see Section 5.3).

Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the upper shroud to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators, shown in Figure 3.9-21, are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes over vanes that impart spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the recirculation flow downcomer annulus.

Core Support

The core support (core plate) consists of a circular stainless steel plate stiffened with a rim and beam structure. Perforations in the plate provide lateral support and guidance for the control rod guide tubes, incore flux monitor guide tubes, peripheral fuel support pieces, and startup neutron sources. The last two items are also supported vertically.

The entire assembly is bolted to a support ledge between the central and lower portions of the core shroud. Alignment pins that bear against the shroud are used to correctly position the assembly before it is secured.

Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings. Each opening provides lateral support and guidance for four fuel assemblies. Holes are provided in the bottom of the beams to anchor the incore flux monitor guide tubes and startup neutron sources. The top guide is positioned with alignment pins that bear against the shroud.

3.9.5.1.2 Fuel Support Pieces

The fuel support pieces, shown in Figure 3.9-22, are of two basic types, namely peripheral and four-lobed. The peripheral fuel support pieces, which are welded to the core support assembly, are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support piece will support one fuel assembly and contains an orifice assembly designed to ensure proper coolant flow to the fuel assembly. Each four-lobed fuel support piece will support four fuel assemblies and is provided with orifice plates to ensure proper coolant flow distribution to each fuel assembly. The four-lobed fuel support pieces rest in the top of-the control rod guide tubes and are supported laterally by the core support. The control rods pass through slots in the center of the four-lobed fuel support pieces. A control rod and the four adjacent fuel assemblies represent a core cell (see Section 4.2).

3.9.5.1.3 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the CRD housings up through holes in the core support. Each tube is designed as the lateral guide for a control rod and as the vertical support for a four-lobed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing (see Section 5.3) which in turn transmits the weight of the guide tube, fuel support piece, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the CRD housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the CRD housing to hold the thermal sleeve in position.

3.9.5.1.4 Jet Pump Assemblies

The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the reactor vessel wall. Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser (see Figure 3.9-23). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High-pressure water from the recirculation pumps (see Section 5.4) is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A horseshoe-shaped riser brace, [redacted] is welded to the jet pump riser at the inside "U" portion of the brace, and the cantilever portions of the brace are welded to pads extending from the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

To monitor its flow, each jet pump has a sensing line which is welded to the diffuser at two points. To ensure against fatigue failure of the sensing lines, beam-and-clamp assemblies have been installed on certain jet pumps. This modification reinforces the welds and changes the natural frequency of the sensing lines to avoid resonance at excitation frequencies which were measured during plant startup. The beam-and-clamp assemblies are described in reference 5.

3.9.5.1.5 Steam Dryers

The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture drips down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus (see-Figure 3.9-24). A skirt extends from the top of the steam dryers to the steam separator standpipe, below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly is restricted by steam dryer holddown brackets attached to the reactor vessel top head.

3.9.5.1.6 Feedwater Spargers

The feedwater spargers are perforated stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle and is shaped to conform to the curve of the vessel wall. Sparger end brackets are attached to vessel brackets to support the weight of the spargers and position the spargers away from the vessel wall. Feedwater flow enters the center of the spargers and is discharged radially inward and downward to mix the cooler feedwater with the downcomer flow from the steam separators and dryers before it contacts the vessel wall. The feedwater also serves to collapse the steam voids in the mixing plenum and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.7 Core Spray Lines

Two 100%-capacity core spray lines enter the reactor vessel through the two core spray nozzles (see Section 5.3). The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular header, which is routed halfway around the inside of the upper shroud. The ends of the two headers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the shroud and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the headers are at a slightly different elevation in the shroud. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the headers.

3.9.5.1.8 Differential Pressure and Standby Liquid Control Line

The differential pressure and standby liquid control line serves a dual function within the reactor vessel--to inject liquid control solution into the coolant stream (see Section 9.3.4) and to sense the differential pressure across the core support assembly (described in Section 5.3). This line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support assembly. It is used to sense the pressure below the core support during normal operation and to inject liquid control solution when required. This location facilitates good mixing and dispersion. The inner pipe also reduces thermal shock to the vessel nozzle should the standby liquid control system be actuated. The outer pipe terminates immediately above the core support and senses the pressure in the region outside the fuel assembly channels.

3.9.5.1.9 Incore Flux Monitor Guide Tubes

The incore flux monitor guide tubes extend from the top of the incore flux monitor housings (see Section 5.3) in the lower plenum to the top guide. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring and intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes. The guide tubes are held in place below the top guide by spring tension. A latticework of clamps, tie bars, and spacers gives lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.9.5.1.10 Initial Startup Neutron Sources

Each initial startup source consists of two irradiated antimony rods within a single beryllium cylinder. The antimony-beryllium cylinder assemblies are further encased in stainless steel tubes. These tubes have fitted nosepieces on one end and axial spring-loaded detent pins on the other end. The nosepieces and detent pins mate, respectively, with notches in the top of the core support plate and the bottom of the top guide to position the startup sources securely in the vertical position. The design provides a sufficient source of neutrons in the core to ensure that the core neutron flux monitors are operating and that significant changes in core reactivity can be readily detected by the installed neutron flux instrumentation (see Section 7.6.1).

3.9.5.1.11 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (see Section 5.3). The baskets hang from brackets that are attached to the inside wall of the reactor vessel and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while avoiding jet pump removal interference or damage.

3.9.5.2 Loading Conditions

3.9.5.2.1 Evaluation Methods

To determine that the safety design bases are satisfied, responses of the reactor vessel internals to loads imposed during normal, upset, emergency, and faulted conditions were examined. The effects on the ability to insert control rods, cool the core, and flood the inner volume of the reactor vessel were determined.

The ASME Code, Section III, for Class A vessels was used as a guide to determine limiting stress intensities and cyclic loadings for the reactor vessel internals. When buckling was not a possible failure mode and stresses were within those stated in the ASME Code, either the elastic stability of the structure or the resulting deformation was examined to determine whether the safety design bases were satisfied.

Events To Be Evaluated

The examination of the spectrum of conditions for which the safety design bases must be satisfied reveals three significant events:

1. LOCA: A break in a recirculation line. The accident results in pressure differentials, across some of the reactor vessel internals, that exceed normal loads.

2. Steam-line break accident: A break in one main steam line between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across some of the reactor vessel internals.

3. Earthquake: This condition subjects the reactor vessel internals to significant forces as a result of ground motion.

The analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the reactor vessel internals are less severe than the design-basis postulated events.

3.9.5.2.2 Recirculation-Line and Steam-Line Break

Accident Definition

The recirculation-line break is the same as the design-basis LOCA described in Section 6.3 and Chapter 15. A sudden, complete circumferential break is assumed to occur in one recirculation loop.

The analysis of the steam-line break assumes a sudden, complete circumferential break of one main steam line between the reactor vessel and the main steam line restrictor. A steam-line break upstream of the flow restrictors produces a larger blowdown area than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor assembly internal structures.

Both the recirculation-line break and steam-line break have been examined for a spectrum of initial reactor operating conditions. These studies show that a recirculation-line break at any initial reactor condition would be very mild with regard to resultant pressure differentials.

The steam-line break accident produces significantly higher pressure differentials across the reactor assembly internal structures than does the recirculation-line break. This results from the higher reactor depressurization rate associated with the steam-line break. The depressurization rate is less for mixed flow than for steam flow. Therefore, the steamline break is the design-basis accident for internal pressure differentials.

For a steam-line break accident, a low initial reactor power level results in a more severe pressure transient across some components than would be the case at maximum power. This is because the difference between energy removal through the break and energy addition to the reactor vessel inventory increases as the reactor power decreases. Thus, the depressurization rate following a steam-line break would increase with decreasing initial reactor power level. Consequently, both a high power and a low power case are examined.

The maximum differential pressures across the reactor assembly internals resulting from the postulated accidents are discussed in Section 15.3.5. Figure 15.3-1 shows the differential pressures for various internals.

Response of Reactor Internals to Pressure Differences

The maximum differential pressures are used, in combination with other structural loads, to determine the total loading on the various reactor vessel internals. The internals are then evaluated to assess the extent of deformation and collapse, if any. Of particular interest are (1) the responses of the guide tubes and the metal channels around the fuel bundles and (2) the potential leakage around the jet pump joints.

The guide tube was evaluated for collapse caused by externally applied pressure, as discussed in Section 3.9.4.3.5.

Channel Response with Respect to Structural Integrity

The channel wall ΔP at which a DAEC channel assembly will fail due to yielding is 16.0 psi. The calculated maximum ΔP for a main steam line break is 14.8 psi (Section 15.3.5.4). Since this is below the yield limit, the channel would maintain its integrity due to the increased ΔP experienced during a main steam line break.

Channel Response with Respect to Channel/Control Blade Interaction

The clearance to the control rod is affected by the channel wall ΔP due to the added elastic channel wall bulge. This effectively reduces the gap between channels which may lead to interaction with the control blade roller. The nominal gap between fuel channels is 700 mils. If the components of the gap are assumed to be a normal distribution with a standard deviation equal to 1/3 the tolerance range, the minimum gap is 632 mils. The channel bulge due to the maximum ΔP is 30 mils. Subtracting this from the gap standard deviation produces a minimum gap between channels at the roller location of 602 mils. Assuming the control blade is centered in the channel gap and the control rod roller diameter is at its maximum of 525 mils, then a net one-sided clearance to the control rod roller of 39 mils exists. Therefore, if the main steam line break were to occur near the beginning of life, no channel/control blade interaction would be expected.

Channel Response with Respect to Channel/Control Blade Interactions Towards the End of Life

The channel wall will experience irradiation induced creep causing more bulge throughout its life as a function of exposure. For conservatism, an exposure of 50 GWD/MTU is assumed to quantify the irradiation induced creep permanent deflection. Where there is interference, the channel and control rod will interact. If the control rod is assumed to be rigid, then it must displace the channel equal to the amount of the interference. This displacement will produce a normal force at the roller channel contact. Since the roller is essentially pinched between the channel gap, it will slide as opposed to rolling and the normal force produced by the displacement will contribute to a friction force that the control rod drives must overcome to insert the control rod. The calculated frictional force is 330 lbs. This calculated force is very conservative since the actual friction loading could not exceed the weight of the control rod or a “no settle” condition would have occurred and the rod would have been fully inserted. The maximum normal hydraulic drag as specified by the control rod is 230 lbs. Adding the maximum weight of a control rod (250 lbs) to the frictional force and the hydraulic drag results in a total resistive force of 810 lbs. With an available driving force of 1040 lbs, there remains a margin of 230 lbs to insert the control rod. It is concluded that the main steam line break accident can pose no significant interference to the movement of control rods.

Additional analysis indicates that no fuel pins will come in contact with the fuel element channels as the result of the DBE concurrent with rapid depressurization of the reactor core.

Jet Pump Joints: An analysis was originally performed to evaluate the potential leakage from within the floodable inner volume of the reactor vessel during the recirculation-line break and subsequent LPCI reflooding. The two possible sources of leakage are the following:

1. Jet pump throat to diffuser joint.

2. Jet pump nozzle to riser joint.

The jet pump to shroud support joint is welded and therefore is not a potential source of leakage. The slip joints for all jet pumps leak no more than a total of 225 gpm. The jet pump bolted joint, by analysis, is shown to leak no more than 542 gpm for the pumps through which the vessel is being flooded.

The summary of maximum leakage is as follows:

Total leakage through all throat to diffuser joints	225 gpm*
Total leakage through all nozzle to riser joints	542 gpm
Total maximum rate	767 gpm

The original sizing calculation procedure for LPCI capacity included a total leakage of 3000-gpm from the core shroud. The ECCS performance analysis in Section 15.2 accounts for all known leakage paths in the core shroud and ECCS flow paths. It is concluded that the reactor vessel internals retain sufficient integrity during the recirculation-line break accident to allow reflooding of the inner volume of the reactor vessel.

3.9.5.2.3 Seismic Analysis of the RPV and Internals

The seismic loads on the reactor pressure vessel (RPV) and internals are based on a dynamic analysis of the RPV and internals shown in Figure 3.9-28. The dynamic model of the RPV and internals is briefly described below.

The presence of a fluid and structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by the introduction of a hydrodynamic mass matrix.

The seismic model of the RPV and internals analyzed has one horizontal translation coordinate for each node point considered in the analysis. Due to the approximate coincidence of the mass center and elastic axis of the building, and the symmetry of the RPV and internals, one horizontal coordinate was excluded. The remaining translational coordinate for the various node points was the vertical coordinate. This coordinate (vertical) was excluded because the frequency content of earthquakes is such that the vertical frequencies of the RPV and internals are well above those of earthquakes. Dynamic loads due to vertical motion were added to or subtracted from the

* Subsequent evaluations determined that this leakage path has negligible impact. The current analysis (Section 15.2) uses a bounding generic value of 600 gpm.

static loads of the components, whichever was the more conservative. The two rotational coordinates about each node point were excluded because the moment contribution of rotary inertia is negligible. The remaining rotational coordinate is neglected since the building, and hence the RPV, has negligible torsional motion.

Seismic analysis was performed by coupling the lumped mass model of the RPV and internals with the building model to determine the system natural frequencies and mode shapes. The load response of the RPV and internals was then determined by the response spectrum method. The spectral accelerations, velocities, and displacements for the OBE and DBE are taken from figures in Section 2.5.4 for the modes of interest. The root mean square of these individual modal responses was then used for design calculation.

The natural frequencies of the reactor internals, reactor vessel, and pedestal system in the vertical direction have been found to be 19 Hz or higher. The examination of the response spectra shows no significant amplification at this frequency. Hence, omitting the vertical motion from seismic analysis to reduce the analytical complexities is acceptable. The effects of vertical excitations are accounted for by increasing or decreasing (whichever causes higher stress) the weight of the various components by a percentage equal to the vertical acceleration expressed in percent “g”.

The basis for the derivation of the LOCA excitation input design is contained in the response to Question A.9 of Amendment 6 to the Browns Ferry Nuclear Power Station Units 1 and 2 Design and Analysis Report. The peak values of dynamic pressure differences calculated for the LOCA were then used in the design of the reactor internals.

To ensure that no significant dynamic amplification of loading occurs as a result of the oscillatory nature of the blowdown forces, a comparison was made of the periods of the applied forces and the natural periods of the reactor internal components being acted upon by the applied forces. These periods were determined from a dynamic model of the RPV and internals shown in Figure 3.9-28.

Besides the real masses of the RPV and internals, the hydrodynamic mass effects of the water inside the RPV were also accounted for.

The natural frequencies of the first five modes for the DAEC RPV and internals are tabulated below:

Mode	Frequency Hz
1	18.8 (shroud)
2	25.5 (fuel-guide tube)
3	33.3 (RPV head)
4	46.6 (RPV head)
5	51.2 (RPV head)

All other reactor internal components have natural frequencies higher than those shown above. The smallest period of the applied force (approximately 0.7 sec), , is more than 10 times the largest period of the component upon which the force acted (i.e., natural frequency of component is more than 10 times greater than the frequency of the applied load). It is evident that this conclusion would apply for the higher modes, since they would have shorter periods. It is a well-known fact that for a damped single degree of freedom system subjected to a sinusoidal forcing function, the amplification factor is essentially equal to unity. Therefore, it was concluded that no significant load amplification occurred because of the "slowly" changing nature of the applied load and that a statically applied load equal in value to the peak transient load could be used for design purposes.

Rather than measuring input forcing functions during normal operation, the vibration response of the RPV internals was actually measured.

3.9.5.2.4 Conclusions

Response analyses of the reactor vessel internals show that deformations are sufficiently limited to allow both adequate control rod insertion and proper operation of the core standby cooling systems. Sufficient integrity of the internals is retained during accident conditions to allow successful reflooding of the reactor vessel inner volume. The analyses considered various loading combinations, including loads imposed by external forces.

3.9.5.2.5 Inspection and Testing

Quality control methods were used during the fabrication and assembly of reactor vessel internals to ensure that the design specifications were met.

The reactor coolant system, which includes the reactor vessel internals, was thoroughly cleaned and flushed before fuel was loaded initially.

During the preoperational test program, operational readiness tests were performed on various systems. In the course of these tests, such reactor vessel internals as the feedwater spargers, the core spray lines, and the standby liquid control system line were functionally tested.

Before the startup of the unit, steam separator-dryer performance tests were conducted on several plants using the DAEC separator-dryer design concept to determine the carryunder and carryover characteristics. It was not planned, therefore, to test the DAEC steam separators and dryers to determine these characteristics since they would have been already demonstrated on other plants. Moisture carryover is determined from sodium-24 activity in samples from main steam lines and samples from primary containment during normal radiochemical quality surveillance tests.

Vibration analysis of reactor vessel internals was included in the design to eliminate failures caused by vibration.

The nature of the tests and the components tested were dictated largely by the results of vibrations testing conducted earlier on the other plants. The DAEC vibration test plan was completed during the first quarter of 1972.

The vibration analyses and tests were designed to determine any potential hydraulically induced equipment vibrations and to verify that the structures do not fail because of fatigue. The structures were analyzed for natural frequencies, node shapes, and vibrational magnitudes that could lead to fatigue at these frequencies. With this analysis as a guide, the reactor internals were instrumented and tested to ascertain that there are no gross instabilities. The cyclic loadings were evaluated using, as a guide, the cyclic stress criteria of the ASME Code, Section III. Field test data were correlated with the analyses to ensure the validity of the analytical techniques on a continuing basis.

For Extended Power Uprate (EPU), the original test data was reviewed to determine which internal components were likely to experience significant vibration at the new operating conditions. The actual measured frequencies and vibration amplitudes were linearly extrapolated to the uprated conditions. The effects of forced vibration and resonance on the extrapolation methodology were considered and found to be inconsequential. The extrapolated vibration amplitude response at EPU conditions was compared with the acceptance criteria to obtain the percent criteria for each mode. The percent criteria for all modes were absolute summed. This total percent criteria was shown to be less than 100%.

In addition, vessel internal components that operating experience have shown to be susceptible to flow-induced vibration problems (such as steam dryers and jet pump sensing lines) were included in the EPU evaluation. While no new vulnerabilities were identified, it was recommended that continued inspections, per BWRVIP-06, for dryer drain channel cracking be performed.

The critical reactor internals were tested up to 51 Mlbm/hr core flow and at 50%, 75%, and 100% load line during the original startup tests (Reference 3). This data was used in the evaluation of Increased Core Flow (ICF) (105% of rated). The expected vibration levels for ICF were estimated by extrapolating the vibration data recorded during startup. The extrapolation was from 51 Mlbm/hr to 51.5 Mlbm/hr (105% of rated core flow). The vibration was extrapolated by using the square relationship: vibration varies as the square of flow. The design configuration and basis used for rated flow is also used for increased core flow.

The results of the vibration of evaluation (Reference 7) show that continuous operation at 1,912 MWt and 105% of the rated core flow (51.5 Mlbm/hr) does not result in any detrimental effects on the reactor internal components. There is no concern for vane passing frequency (VPF) resonance up to the maximum design pump speed VPF of 142.5 Hz. The maximum design recirculation pump speed at DAEC is 1,710 rpm. This corresponds to a VPF of 142.5 Hz. The calculations for the ICF operating condition indicate that the vibration of the components evaluated are within the acceptance criteria.

The acceptance criterion of 10,000 psi peak stress intensity is less than the ASME Code criteria of 13,600 psi.

The reactor vessel and internals were designed to ensure adequate working space and access for the inspection of selected components and locations. Criteria for selecting the components and locations to be inspected were based on the probability of a defect occurring or enlarging at a given location and include areas of known stress concentrations and locations where cyclic strain or thermal stress might occur.

3.9.5.3 Design Bases

3.9.5.3.1 Power Generation Objectives

Reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are provided to achieve the following power generation objectives:

1. Maintain partitions between regions within the reactor vessel to provide correct coolant distribution, thereby allowing power operation without fuel damage.

2. Provide positioning and support for the fuel assemblies, control rods, incore flux monitors, and other vessel internals to ensure that control rod movement is not impaired.

3.9.5.3.2 Safety Design Bases

The reactor vessel internals meet the following safety design bases:

1. The internals are arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

2. The deformation of internals is limited to ensure that the control rods and emergency core cooling systems can perform their safety functions.

3. The mechanical design of applicable internals ensures that safety design bases 1 and 2 are satisfied so that the safe shutdown of the plant and the removal of decay heat are not impaired.

3.9.5.3.3 Power Generation Design Bases

The reactor vessel internals are designed to meet the following power generation design bases:

1. They provide the proper coolant distribution during all anticipated normal operating conditions to allow power operation of the core without fuel damage.

2. They are arranged to facilitate refueling operations.

3. They are designed to facilitate inspection.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

An inservice testing program for pumps and valves has been prepared. This program is revised as required for each 120-month inspection interval to incorporate the latest applicable addenda to ASME Code, Section XI.

Inservice testing of pumps and valves complies with the requirements of Subsections IWP and IWV of ASME Code, Section XI, respectively.

3.9.6.1 Relief Requests

When compliance with ASME Code, Section XI, is impractical for specific items, relief is requested from the NRC in compliance with 10CFR50.55a(g)(5).

3.9.6.2 Inservice Testing Program

The inservice testing program for the DAEC fourth 10-year Inservice Testing interval commenced February 1, 2006. The current revised program has been prepared and implemented according to the 2001 Edition of Section XI of the ASME Code, through the 2003 Addenda.

In response to NRC Generic Letter 87-06, a list of all pressure isolation valves along with information on periodic tests was submitted in Reference 4.

REFERENCES FOR SECTION 3.9

1. U.S. Nuclear Regulatory Commission, Failure of Control Rods to Insert During a Scram at a BWR, IE Bulletin 80-17 (series), 1980-1981.

2. H. T. Kim, Core Flow Distribution in a Modern BWR as Measured at Monticello, NEDO-10299, 1971.

3. General Electric Co., "Duane Arnold Reactor Internals Vibration Measurements", NEDE-23736, Oct. 1977.

4. Letter from R. McGaughy (Iowa Electric) to T. Murley (NRC), Subject: Periodic Testing of Leak Tight Integrity of Pressure Isolation Valves (Generic Letter 87-06), dated June 11, 1987 (NG-87-1881).

5. J. G. Erbes, Safety Evaluation of the Jet Pump Sensing Line Clamp and Beam Assembly Duane Arnold, RDE #04-387, General Electric Company, March 1987.

6. General Electric Report, Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate, NEDC-32980P, May 2001.

7. GE Hitachi Nuclear Energy, Safety Analysis Report for Duane Arnold Energy Center Increased Core Flow, NEDC-33439P, Revision 3, August, 2009.

Table 3.9-1

STRESS SUMMARY FOR SHROUD SUPPORT LEGS

Criteria	Loading	Primary Stress Type	Stress (psi) Allowable	Stress (psi) Calculated
ASME Code, Section III Primary Stress Limit for SB-168				
For normal and upset condition	Normal and upset condition loads	General membrane	23,300	([a])
S_m = 23,300 psi	Dead weight			
	Operating-basis earthquake			
For emergency condition	Emergency condition loads	General membrane	34,950	16,748
$1.5S_m$ = 34,950 psi	Dead weight			
	Design-basis earthquake			
For faulted condition	Faulted condition loads	General membrane	46,600	42,300
$2.0S_m$ = 46,600 psi	Dead weight			
	Design-basis earthquake			
	Jet reaction forces			
	Pressure drop across shroud support and shroud during faulted condition			

[a] Since the calculated stress for the emergency condition is less than the allowable stress for this loading, this stress has not been listed.

Note: The shroud support legs are stiff enough to prevent buckling.

Table 3.9-2

STRESS SUMMARY FOR VESSEL SUPPORT SKIRT

Criteria	Loading	Primary Stress Type	Stress (psi) Allowable	Stress (psi) Calculated
ASME Code, Section III Primary Stress Limit for SA-516 Grade 70				
For normal and upset condition	Normal and upset condition loads	General membrane	19,600	([a])
S_m = 19,600 psi	Dead weight			
	Operating-basis earthquake			
For emergency condition	Emergency condition loads	General membrane	29,400	5,400
$1.5S_m$ = 29,400 psi	Dead weight			
	Design-basis earthquake			
For faulted condition	Faulted condition loads	General membrane	39,200	7,700
$2.0S_m$ = 39,200 psi	Dead weight			
	Design-basis earthquake			
	Jet reaction forces			

[a] Since the calculated stress for the emergency condition is less than the allowable stress for this loading, this stress has not been listed.

Table 3.9-3

STRESS SUMMARY FOR STABILIZER BRACKET - ADJACENT SHELL

Criteria	Loading	Primary Stress Type	Stress (psi) Allowable	Stress (psi) Calculated
ASME Code, Section III Primary Stress Limit for SA-533 Grade B, Class I				
For normal and upset condition	Normal and upset condition loads	Pure shear	16,000	([a])
0.6 x 26,700 = 16,000 psi	Operating-basis earthquake			
For emergency condition	Emergency condition loads	Pure shear	24,000	([a])
1.5 x 16,000 = 24,000 psi	Design-basis earthquake			
For faulted condition	Faulted condition loads	Pure shear	32,000	13,000
2.0 x 16,000 = 32,000 psi	Design-basis earthquake Jet reaction forces			

[a] Since the calculated stress for the faulted condition less than the allowable stress for this loading, this stress has not been listed.

STRESS SUMMARY FOR REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Shroud			
Primary Stress Limit The allowable primary membrane and membrane plus bending stresses are based on ASME Code, Section III, for Type 304 stainless steel plate.				
For normal and upset condition Stress intensity $S_A = 1.0\ S_m$ S_m = 16,950 psi	Normal and upset condition loads Upset condition pressure drop Operating-basis earthquake Weight of structure	General membrane	16,950 psi	4,259 psi*
For Faulted condition $S_{limit} = 2.0\ S_m = 2.0 \times 16{,}950$ = 33,900 psi	Faulted condition loads Faulted condition pressure drop Design-basis earthquake Weight of structure	General membrane	33,900 psi	8,782 psi
For faulted condition $S_{limit} = 2.0\ S_m = 2 \times 16,950$ = 33,900 psi	Faulted condition loads Pressure drop after main steamline rupture Acoustic Weight of structure Design Basis Earthquake	General membrane plus bending	33,900 psi	22,686 psi

Note: 2 S_m is conservatively used as the Faulted condition allowable stress.

* Bending stress was conservatively considered as the primary membrane stress

STRESS SUMMARY FOR REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type or Location	Stress	
	Top Guide - Longest Beam			
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Code, Section III, for Type 304 stainless steel plate.				
For normal and upset condition Stress intensity $S_A = 1.5$ $S_m = 1.5 \times 16{,}925$ psi = 25,388 psi	Normal and upset condition loads Operating-basis earthquake Weight of structure	General membrane plus bending	25,388 psi	15,258 psi
For emergency condition $S_{limit} = 1.5S_A = 1.5 \times 25{,}388$ = 38,081 psi	Emergency condition loads Design-basis earthquake Weight of structure	General membrane plus bending	38,081 psi	26,900 psi
For faulted condition $S_{limit} = 2S_A = 2 \times 25{,}388$ = 50,775 psi	Faulted condition loads (same as emergency condition)	General membrane plus bending	50,775 psi	26,900 psi
	Top Guide Beam End Connections			
Primary Stress Limit ASME Code, Section III, defines material stress limit for Type 304 stainless steel.				
For normal and upset condition Stress intensity $S_A = 0.6$ $S_m = 0.6 \times 16{,}925$ psi = 10,155 psi	Normal and upset condition loads Operating-basis earthquake Weight of structure	Pure shear	10,155 psi	6,509 psi
For emergency condition $S_{limit} = 1.5$ $S_A = 1.5 \times 10{,}155$ psi = 15,232 psi	Emergency condition loads Design-basis earthquake Weight of structure	Pure shear	15,232 psi	11,800 psi

Table 3.9-4

STRESS SUMMARY FOR REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT (Continued)

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Top Guide Beam End Connections (Continued)			
For faulted condition $S_{limit} = 2S_A = 2 \times 10,155$ psi $= 20,310$ psi	Faulted condition loads (same as emergency condition)	Pure shear	20,310 psi	11,800 psi
	Top Guide Seismic Restraint Blocks			
Primary Stress Limit ASME Code, Section III, defines material stress limit for Type 304 stainless steel plate.				
For normal and upset condition Stress intensity $S_A = 0.6$ $S_m = 0.6 \times 16,925 = 10,155$ psi	Normal and upset condition loads Operating-basis earthquake Weight of structure	Pure shear	10,155 psi	7,600 psi
For emergency condition $S_{limit} = 1.5$ $S_A = 1.5 \times 10,155 = 15,232$ psi	Emergency condition loads Design-basis earthquake Weight of structure	Pure shear	15,232 psi	15,100 psi
For faulted condition $S_{limit} = 2S_A = 2 \times 10,155$ $= 20,310$ psi	Faulted condition loads (same as emergency condition)	Pure shear	20,310 psi	15,100 psi
	Core Support			
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Code, Section III, for Type 304 stainless steel plate.				
	Normal and upset condition loads Normal operation pressure drop Operating-basis earthquake	General membrane plus bending	25,388 psi	16,982 psi

Table 3.9-4
STRESS SUMMARY FOR REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT (Continued)

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Core Support (continued)			
For allowable stresses, see Top Guide - Longest Beam, above.	Emergency condition loads Normal operation pressure drop Design-basis earthquake	General membrane plus bending	38,081 psi	26,196 psi
	Faulted condition loads Pressure drop after recirculation line rupture Design-basis earthquake	General membrane plus bending	50,775 psi	29,889 psi
	Core Support Aligners			
Primary Stress Limit ASME Code, Section III, defines material stress limit for Type 304 stainless steel.				
	Normal and upset condition load Operating-basis earthquake	Pure shear	10,155 psi	0[a]
For allowable shear stresses, see Top Guide Beam and Connections, above.	Emergency condition load Design-basis earthquake	Pure shear	15,232 psi	0[a]
	Faulted condition load Design-basis earthquake	Pure shear	20,310 psi	0[a]
	Reactor Pressure Vessel Stabilizer			
Primary Stress Limit AISC Specification for the instruction, fabrication, and erection of structural steel for buildings.				
For normal and upset condition AISC allowable stresses, but without the usual increase for earthquake loads	Upset condition loads Spring preload Operating-basis earthquake	Rod Bracket	63,000 psi 22,000 psi 14, 000 psi	f_t = 62,500 psi[b] f_b = 12,200 psi f_v = 3,900 psi

[a] The friction force between core support and core support flange due to the preload of 9,000 lb per stud is greater than the shear load on the core support induced by design-basis earthquake. Therefore, all aligners will not see any shear load during a design-basis earthquake. These studs have the capability, without exceeding allowable stresses, of being preloaded to even higher levels per the newest produced ASME Section III, Subsection NG, for reactor internals criteria.

[b] The ratio maximum stress/stress limit is highest for upset loading conditions.

Table 3.9-4
STRESS SUMMARY FOR REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT (Continued)

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Reactor Pressure Vessel Stabilizer (Continued)			
For emergency conditions 1.5 x AISC allowable stresses	Emergency condition loads Spring preload Design-basis earthquake	Bracket	33,000 psi 21,000 psi	f_b = 16,100 psi f_v = 5,100 psi
For faulted conditions	Faulted condition loads Spring preload Design-basis earthquake Jet reaction load	Bracket	36,000 psi 21,500 psi	f_b = 12,200 psi f_v = 5,600 psi
	Control Rod Drive Housing Support			
Primary Stress Limit AISC Specification for the design, fabrication, and erection of structural steel for buildings.				
For normal and upset condition F_a = 0.60 Fy (tension) F_b = 0.60 Fy (bending) F_v = 0.40 Fy (shear)				
For faulted conditions F_a limit = 1.5F_a (tension) F_b limit = 1.5f_b (bending) F_v limit = 1.5F_v (shear) F_y limit = material yield strength	Faulted condition loads Dead weight Impact force from failure of a control rod drive housing (dead weights and earthquake loads are very small as compared to jet force)	Beams (top cord) Beams (bottom cord) Grid structure	33,000 psi 33,000 psi 33,000 psi 33,000 psi 41,500 psi 27,500 psi	f_a = 11,800 psi f_b = 19,800 psi f_a = 9,900 psi f_b = 13,800 psi f_a = 40,000 psi f_b = 11,100 psi
	Control Rod Drive Housing			
Primary Stress Limit The allowable primary membrane stress is based on the ASME Code, Section III, for Class A vessels for Type 304 stainless steel.				
For normal and upset conditions S_m = 15,800 psi at 575°F	Normal and upset condition loads Design pressure Stuck rod scram loads Operating-basis earthquake	Maximum membrane stress intensity occurs at the tube-to-tube weld near the center of the housing for normal, upset, and emergency conditions	15,800 psi	14,480 psi

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Control Rod Drive Housing (Continued)			
For emergency conditions $S_{limit} = 1.5$ $S_m = 1.5 \times 15{,}800 = 23{,}700$ psi	Emergency condition loads Design pressure Stuck rod scram loads Design-basis earthquake		23,700 psi	22,030 psi
	Control Rod Drive			
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Code, Section III, for SA-212 TP 316 tubing.				
For normal and upset condition $S_A = 1.5$ $S_m = 1.5 \times 17{,}375 = 26{,}060$ psi	Normal and upset condition loads[c] Maximum hydraulic pressure from the control rod drive supply pump	Maximum stress intensity occurs at a point on the Y-Y axis of the indicator tube	26,060 psi	20,790 psi
	Control Rod Guide Tube			
Primary Stress Limit[d] The allowable primary membrane stresses plus bending stress is based on the ASME Code, Section III, for Type 304 stainless steel tubing.				
For normal and upset condition[e] $S_m = 15{,}800$			15,800 psi	5,049 psi
For faulted condition $S_{limit} = 2.0$ $S_A = 2.0 \times 23{,}700 = 47{,}400$ psi	Faulted condition loads Dead weight Pressure drop across guide tube due to failure of main steamline Design-basis earthquake	The maximum bending stress under faulted loading conditions occurs at the center of the guide tube	15,800 psi	9,175 psi

Table 3.9-4

[c] Accident conditions do not increase this loading. Earthquake loads are negligible.
[d] The Guide Tube was also evaluated and qualified for its ability to resist buckling under the external pressure differentials and the axial compressive loads for the Normal, Upset, Emergency, and Faulted conditions (See Reference 3.9 – 7).
[e] The Normal/Upset condition allowable stress is conservatively used for Faulted condition allowable stress.

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Orificed Fuel Support			
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Code, Section III, for Type 304 stainless steel.				
For normal and upset condition Stress intensity S_A = 15,580 psi	Normal and upset condition loads Upset condition pressure drop Operating-basis earthquake Weight of structure	General membrane plus bending	15,580 psi	12,657 psi
For faulted condition S_{limit} =35,440 psi Note: Quality factor (0.65) was considered in $P_m + P_b$ allowable stress.	Faulted condition loads Pressure drop after main steamline rupture Design-basis earthquake Weight of structure	General membrane Plus bending	35,440 psi	23,413 psi
	Incore Housing			
Primary Stress Limit The allowable primary membrane stress is based on ASME Code, Section III, for Class A vessels, for Type 304 stainless steel.				
For normal and upset condition[e] S_m = 15,800 psi at 575ºF			15,800 psi	15,290 psi
For emergency condition (H+E′)	Emergency condition loads Design pressure Design-basis earthquake	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration	15,800 psi	15,290 psi

[e] The calculated stresses are based on the Emergency and Faulted loads. This stress is conservatively used for the Normal and Upset conditions also since it is less than the membrane stress allowable of Sm.

STRESS SUMMARY FOR REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT (Continued)

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Hydraulic Control Unit Piping			
Primary Stress Limit For ANSI B31.1.0 for power pressure piping				
For normal condition S_h = 15,000 psi	Normal condition load Maximum normal hydraulic system pump pressure	3/4 in. drive withdraw piping	15,000 psi	14,596 psi
For upset and emergency condition When upset or emergency condition exists for less than 1% of the time, the code allows 20% increase in stress. S_a = 1.2 S_b = 18,000 psi	Upset condition load Shut off pump pressure Operating-basis earthquake	3/4 in. drive withdraw piping	18,000 psi	16,950 psi
	Emergency condition Shut off pump pressure Design-basis earthquake	3/4 in. drive withdraw piping	18,000 psi	16,950 psi

STRESS SUMMARY FOR REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT (Continued)

Criteria	Loading	Primary Stress Type or Location	Stress Allowable	Stress Calculated
	Fuel Channels			
Primary Stress Limit Allowable stress S_m for zircaloy determined according to methods recommended by ASME Code, Section III.				
	Normal and upset condition loads Operating-basis earthquake Normal pressure load	Membrane and bending	20,793in-lb[f]	6,850 in-lb[g]
Emergency limit load 1.5 x normal limit load calculated, using 1.5 S_m = yield	Emergency condition load Design-basis earthquake Normal pressure load	Membrane and bending		
	Faulted condition load Design-basis earthquake Loss-of-coolant accident pressure	Membrane and bending	31,190 in-1b[f]	13,700 in-lb[g]
	Recirculating Pipe and Pump Restraints			
Primary Stress Limit Structural steel: AISC specification for the design, fabrication, and erection of structural steel for buildings.				
For normal or upset condition $F_a = 0.60\ F_y$ (tension)				
For faulted condition F_a limit = 1.5 F_a (tension) F_y = yield strength cable (wire rope)	Faulted condition loads Jet force from a complete circumferential failure (break) of recirculation line	Brackets on 22-in. pipe Cable on pump restraints	33,000 psi 99,000 psi	29,300 psi 61,200 psi
For faulted condition $F_a = 0.80\ F_u$ (tension) F_u = ultimate strength				

[f] Maximum limit accounting for pressure loads.
[g] Maximum moment.

SPECIAL LOADING CRITERIA[a]

1. Criteria F - Deformation Limit

Any one of			General Limit
a.	$\dfrac{\text{(permissible deformation)}}{\text{(analyzed deformation causing loss of function)}^{1}}$	$\leq$	$\dfrac{0.9}{SF_{min}}$
b.	$\dfrac{\text{(permissible deformation)}}{\text{(experimental deformation causing loss of function)}}$	$\leq$	$\dfrac{1.0}{SF_{min}}$

2. Criteria F - Primary Stress Limit

Any one of			
a.	$\dfrac{\text{(elastic evaluated primary stresses)}}{\text{(ASME III normal event permissible primary stresses)}}$	$\leq$	$\dfrac{2.25}{SF_{min}}$
b.	$\dfrac{\text{(permissible load)}}{\text{(largest lower bound limit load with } \measuredangle\text{y.p. = 150\% } S_m \text{ ASME III)}^{2}}$	$\leq$	$\dfrac{1.5}{SF_{min}}$
c.	$\dfrac{\text{(permissible load)}}{\text{(elastic-plastic}^{3}\text{ calculated load causing loss of function)}}$	$\leq$	$\dfrac{0.9}{SF_{min}}$

[a] Superscript numbers are keyed to the notes at the end of this table.

SPECIAL LOADING CRITERIA[a]

2. Criteria F - Primary Stress Limit (Continued) — General Limit

d.. $\dfrac{\text{(elastic evaluated nominal primary stress)}^{4}}{\text{(conventional ultimate strength at temperature)}} \leq \dfrac{0.75}{SF_{min}}$

e. $\dfrac{\text{(elastic-plastic evaluated nominal primary stress)}^{5}}{\text{(conventional ultimate strength at temperature)}} \leq \dfrac{0.9}{SF_{min}}$

f. $\dfrac{\text{(permissible load)}}{\text{(plastic instability load)}^{6}} \leq \dfrac{0.9}{SF_{min}}$

g. $\dfrac{\text{(permissible load)}}{\text{(ultimate load from fracture analysis)}^{7}} \leq \dfrac{0.9}{SF_{min}}$

h. $\dfrac{\text{(permissible load)}}{\text{(ultimate load or loss of function load from test)}} \leq \dfrac{1.0}{SF_{min}}$

[a] Superscript numbers are keyed to the notes at the end of this table.

SPECIAL LOADING CRITERIA[a]

3. Criteria F - Buckling Stability Limit — General Limit

Any one of

a. $\dfrac{\text{(permissible load)}}{\text{(ASME III normal event permissible load)}} \leq \dfrac{2.25}{SF_{min}}$

b. $\dfrac{\text{(permissible load)}}{\text{(stability analysis load)}^{8}} \leq \dfrac{0.9}{SF_{min}}$

c. $\dfrac{\text{(permissible load)}}{\text{(instability load from test)}} \leq \dfrac{1.0}{SF_{min}}$

4. Criteria F - Fatigue Limit

Summation of fatigue damage usage with operation loads following Miner hypotheses, either one

a.	Mean fatigue cycle usage from analysis [9]	≤	0.05
b.	Mean fatigue cycle usage from test[9]	≤	0.33
c.	Design fatigue cycle usage from analysis per ASME III or ANSI B31.7	≤	1.0

[a] Superscript numbers are keyed to the notes at the end of this table.

NOTES

1. "Loss of function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest where deformation limits can affect the function of Seismic Category I structures, they will be specifically delineated. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, core support deformation causing fuel disarrangement, excess leakage of any component, or required pumps or valves failing to operate.

2. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

3. It is permissible to take credit for material strain hardening in computing the load. A shear or strain energy of deformation theory may be used with a monotonic stress curve at the temperature of load application. Any approximation to the actual stress strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used.

4. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear, or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

5. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress strain curve which everywhere has a lower stress for the same strain as the actual montonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

6. The "plastic instability load" is defined here as the load at which any load bearing section begins to diminish its cross-sectional area at a faster rate that the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

7. For components which involve sharp discontinuities (local theoretical stress concentration >3) the use of a "fracture mechanics" analysis, where applicable, utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis my be used in the fracture analysis where its use is supported by experimental evidence. Examples where fracture mechanics may be applied are for fillet welds or end of fatigue life crack propagation.

NOTES (Continued)

8. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity of column members.

9. Fatigue failure is defined here as a 25% area reduction for a load carrying member which is required to function or excess leakage causing loss of function, whichever is more limiting. In the fatigue evaluation the methods of linear elastic stress analysis may be used when the $3S_m$ range limit of ASME III has been met. If 35_m is not met, account will be taken of (a) increases in local strain concentration, (b) strain ratcheting, (c) redistribution of strain due to elastic-plastic effects. ANSI B31.7 piping code may be used where applicable or detailed elastic-plastic methods, strain hardening my be used, not to exceed in stress for the same strain, the steady-state cyclic strain hardening measured in a smooth low cycle fatigue specimen at the average temperature of interest.

Table 3.9-6

SUMMARY OF LOADING CONDITIONS AND CRITERIA

	Loading Conditions	Criteria
Reactor pressure vessel	1	C_1, F
	2	C_1, F
	3	C_1, F
	4	C_1, F
Internals	1	F
	2	F
	3	F
	4	F
Piping	1	C_2, F
	2	F
	3	F
	4	F
Equipment and valves	1	C_2, F
	2	F
	3	F
	4	F
Supports and restraints	1	F
	2	F
	3	F
	4	F

where the criteria are

C_1 = ASME Code, Section III

C_2 = Industry codes

F = Special loading criteria (see Table 3.9-5)

Table 3.9-7

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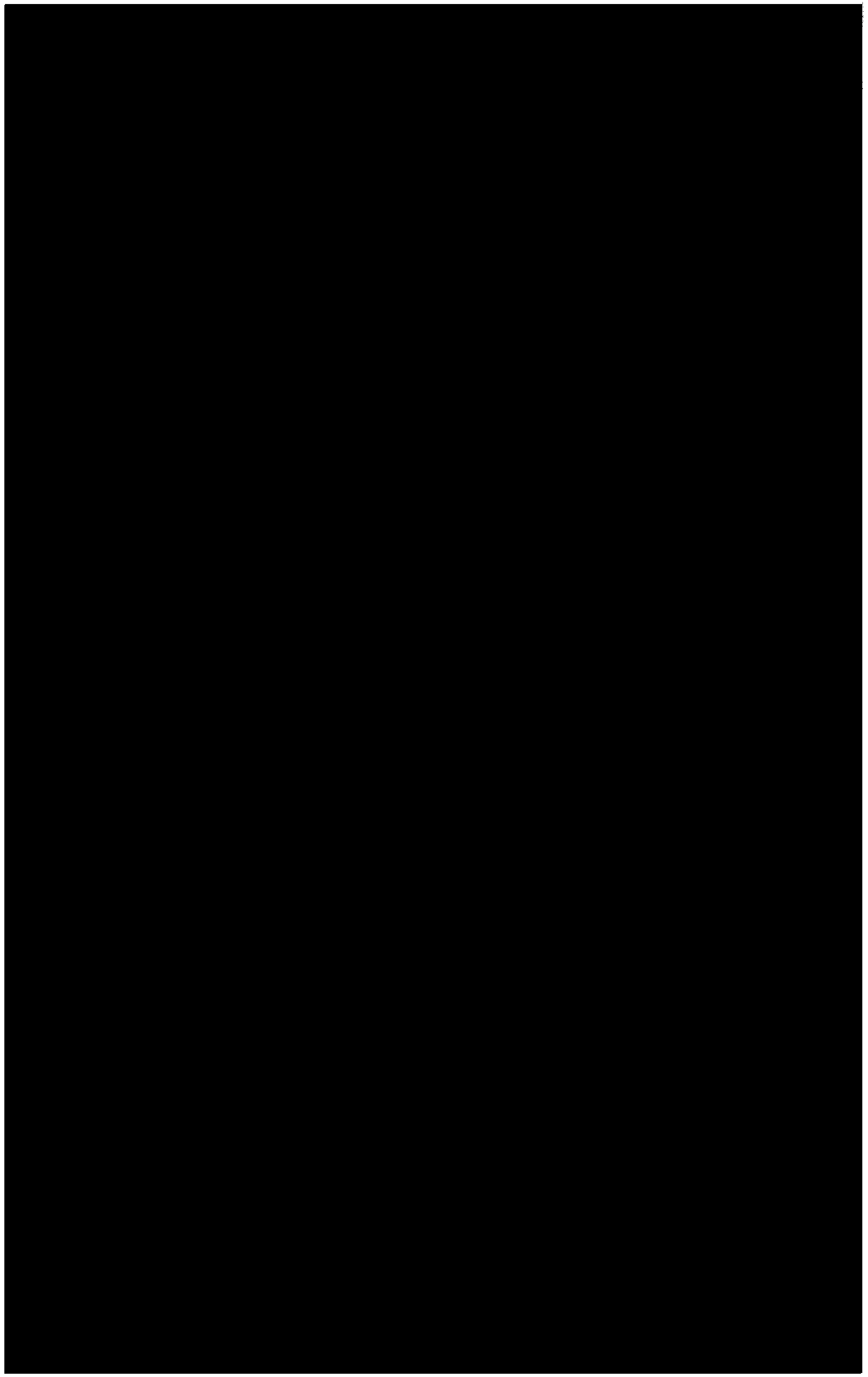
[redacted]



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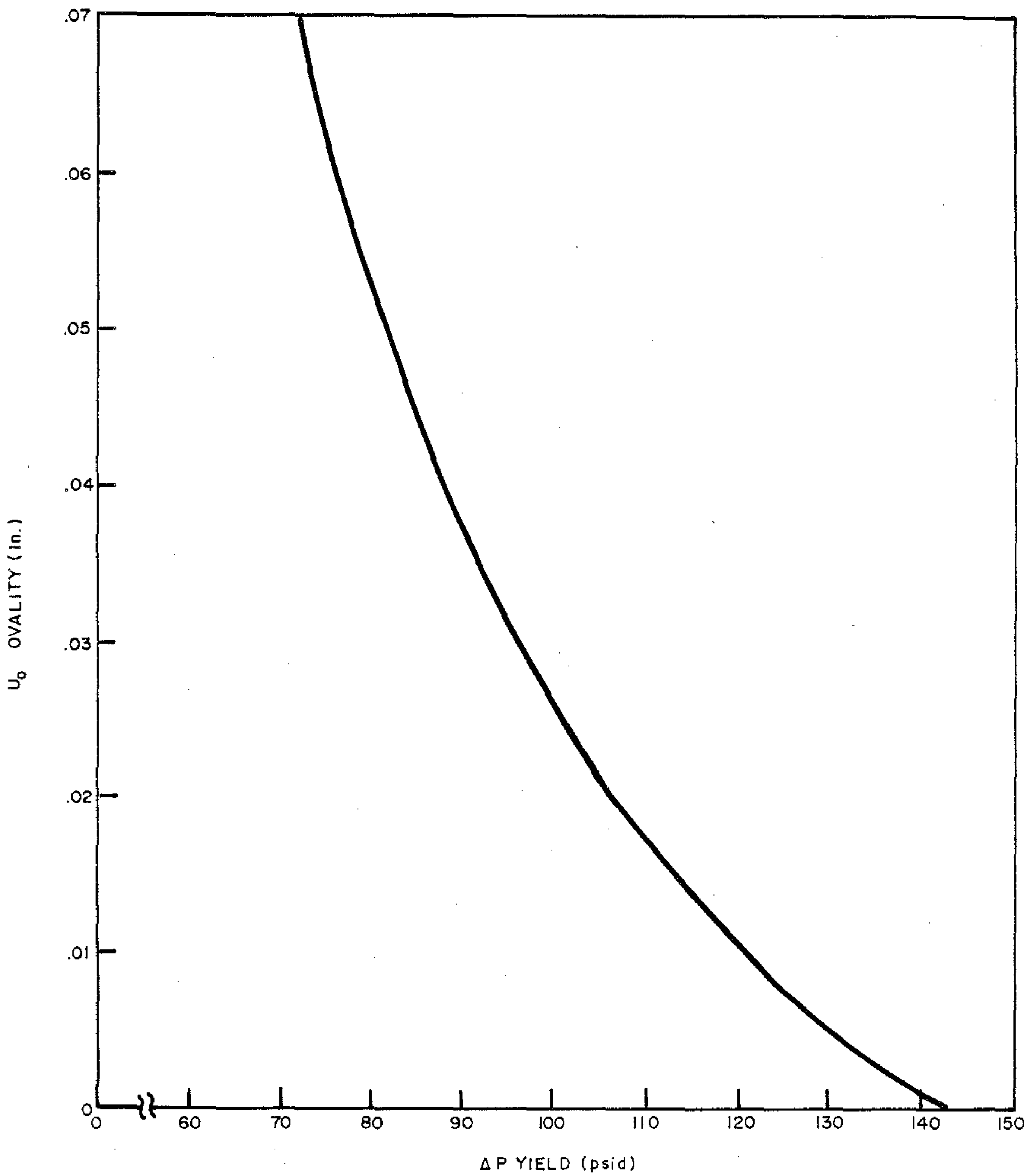
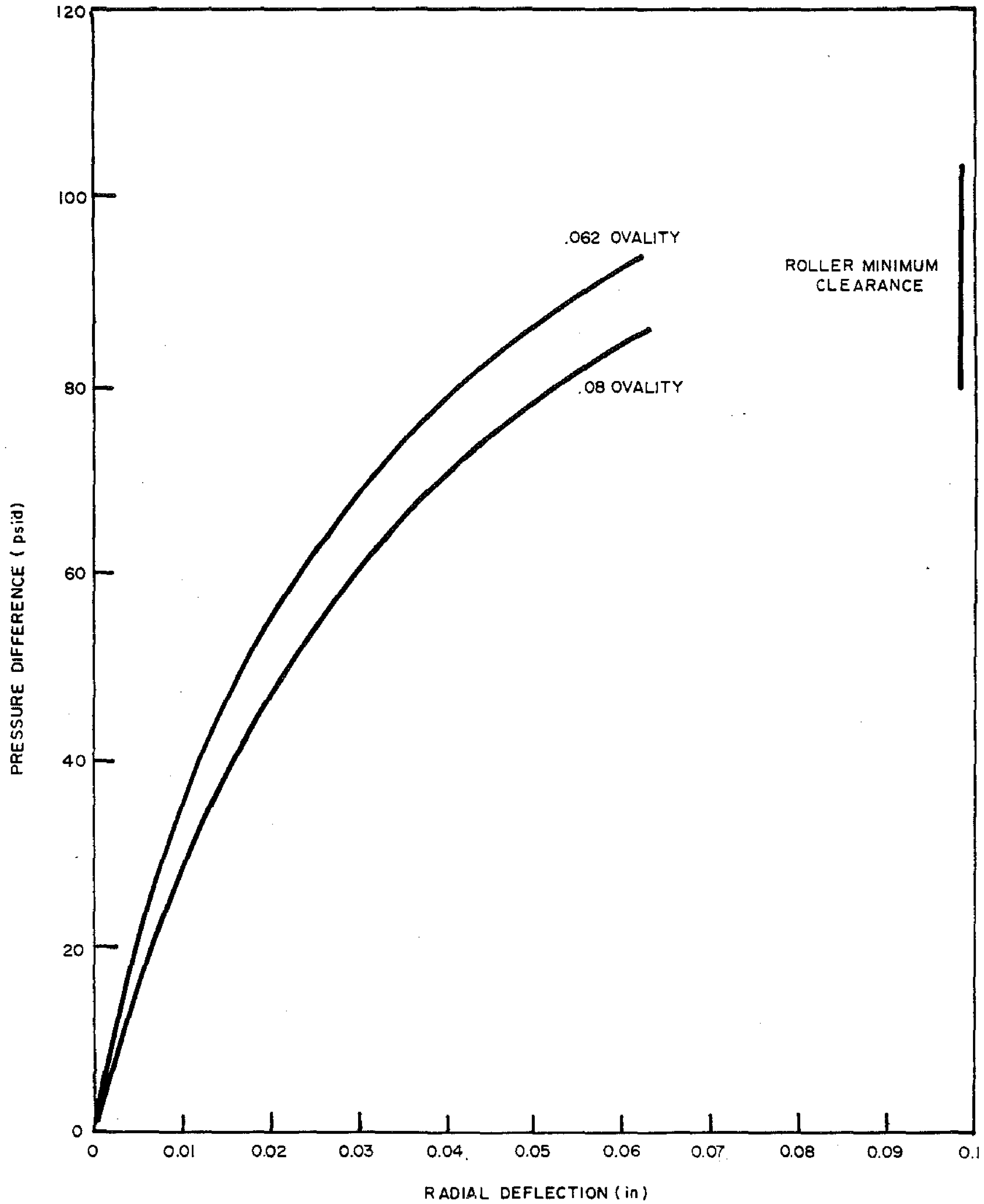


DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Guide Tube Ovality

Figure 3.9-17



DUANE ARNOLD ENERGY CENTER

IES UTILITIES, INC.

UPDATED FINAL SAFETY ANALYSIS REPORT

Guide Tube Deflection

Figure 3.9-18

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3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC QUALIFICATION CRITERIA

3.10.1.1 General Electric-Supplied Instrumentation and Control Equipment

All instrumentation required for nuclear safety is capable of performing all functions important to safety during normal reactor operation, design-basis accidents and postaccident operation while being subjected to accelerations that are in excess of those calculated for the DBE at the point of attachment of the instrument (or module) to the building structure. Qualification is achieved by test and/or analysis at acceleration values of 1.5g horizontal and 0.5g vertical over a frequency range of 0.25 to 33 Hz. The seismic analysis of balance-of-plant (BOP) instrumentation is discussed in Sections 3.10.1.2 and 3.7.2, and Table 3.10-1 lists the criteria used to determine seismic classification of BOP instrumentation.

Acceleration at the point of attachment for a specific instrument is related to the floor acceleration by the transmissibility of the supporting structure (panel or rack). The racks and panels are designed to have low amplification (close to 1 at frequencies below 10 Hz and not to exceed 2.5 at frequencies above 10 Hz up to 33 Hz). The amplification characteristics of each general type of rack or panel design are demonstrated by a vibration test supplemented by analysis of the low end of the frequency spectrum (outside the capability of the test equipment). Instrumentation device types are individually qualified by vibration test for 1.5g (or more) horizontal and 0.5g (or more) vertical.

A panel or rack assembly is thus conservatively qualified for use where the actual floor acceleration does not exceed the value obtained by dividing the lowest instrument qualification value by 2.5. However, where the response spectrum (acceleration versus frequency) of the floor at the location of the panel is know and the amplification spectrum (amplification versus frequency) of the panel (or rack) is known, a more accurate qualification limit may be established.

Seismic qualification at 1.5g horizontal and 0.5g vertical at point of attachment is sufficient to ensure operability of instrumentation in a worst-case loading situation at any location of essential instrumentation in the plant.

The small incremental loading contributed by the connecting wiring (given appropriate cable support) is considered to be adequately provided for by the margin contained in the general seismic qualification requirement.

The effect of electric conduit connections to instruments and of instrument piping connections to instrument racks (again given appropriate conduit and pipe support) is generally to increase the stiffness of the instrument or rack support system and thus reduce rather than increase the maximum loading on individual instruments.

Attachment systems (bolts, clamps, etc.) are demonstrated to be capable of supporting operating instrumentation that they are designed to support during seismic testing without the benefit of additional support normally offered by connections to cables, conduits, and instrument piping.

Condensing chambers, temperature reference columns, and SRM/IRM dry tubes are designed and fabricated in accordance with the ADME Code, Section III (Class 1 equipment), and are required to be inspected by a third party and appropriately code stamped as certification of their compliance. They are also required to be dynamically analyzed with seismic forces superimposed on normal operating loads from system pressure and temperature for purposes of qualification.

Table 3.10-2 lists maximum usable g levels for which the various types of instrumentation devices or module have been seismically qualified by actual vibration testing.

3.10.1.2 Bechtel-Supplied Instrumentation and Control Equipment

The purchase specifications for the instrumentation and control equipment supplied by Bechtel required that each type of Class 1 device by individually qualified for Seismic Category I service by vibration test or suitable analysis.

The methods of test or analysis used for seismic qualification of Class 1 electrical instrumentation and control equipment met the general requirements of IEEE Standard 344-1971.

The maximum acceleration levels (vertical and horizontal) that each device can endure without failure while performing its function were determined by the test and/or analysis.

The device was accepted for application in DAEC only if the failure threshold acceleration level was greater than the calculated DAEC DBE acceleration at the point of its attachment to the supporting structures (appropriate panel, rack, or individual mounting), under all modes of operation including the design-basis accident and postaccident operation.

The supporting structure was analyzed to show that, during a DBE, none of the instruments or electrical devices mounted thereon would be subjected to acceleration levels equal to or greater than their failure threshold acceleration.

Table 3.10-3 lists the essential (Class 1) generic types of instrumentation and control devices supplied by Bechtel for the DAEC.

3.10.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

All types of instrumentation and control devices used in reactor protection system and safeguard systems were tested in an operational condition. The equipment, for instance, was supplied with appropriate input signals and/or trip inputs and monitored with the trips set within 2% (upscale and downscale) of the input signal value. Relays were monitored in energized and deenergized conditions for both normally open and normally closed contacts. Pressure, level, and flow switches were vibrated while provided with simulated input signals that approached setpoints within 2% of setting and switch contacts monitored for false closure or opening (spurious trips). The instrument or device was mounted the way it is mounted in its actual application in the plant.

During seismic scans, the devices were monitored for resonant frequencies using either accelerometers, strobe lights or both. The accelerometers were connected to charge amplifiers which were used to drive a recorder for permanent recording of data. A meter output strobe light aided in detection of the resonant frequencies and response modes of the devices. The detection and exploration of the resonant frequencies were first made over the criteria frequency range to detect possible weak points that could result in failure during subsequent endurance and higher acceleration runs.

Vibration endurance and maximum acceleration scans over the frequency range were then conducted (after the resonant search) to subject the hardware to the maximum specified accelerations of 1.5g horizontal in two perpendicular axes and 0.5g vertical. Also each instrument or device was tested at 33 Hz at increasing amplitudes until the maximum acceleration without malfunction was determined.

Table 3.10-2 lists maximum usable g levels for which the various types of instrumentation devices or modules have been seismically qualified by actual vibration testing.

3.10.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING AND SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

In general, the method selected to satisfy seismic design criteria for cable tray supports, battery and instrument racks, and control consoles consists of the following:

1. Calculation of the fundamental frequency.

2. Determination of the acceleration response from the building floor spectrum curves.

3. Stress calculations to verify structural adequacy of cabinets and support bolts and brackets.

Some units, like battery chargers, were tested by subjecting them to vibrations simulating the seismic disturbance.

The means used to verify the adequacy of the seismic design are discussed in Section 3.7.2.

REFERENCES FOR SECTION 3.10

1. General Electric Service Advice Letter (SAL) 721-PSM-174.1, “PVD and HGA Relay Seismic Data,” dated May 12, 1983.

Table 3.10-1

CRITERIA FOR SEISMIC CLASSIFICATION OF INSTRUMENTATION

Seismic Requirements of Process Line to Which Instrument is Attached	Function of Instrument: Safety-Related[a]	Function of Instrument: Not Safety Related
Seismic Category 1	Instrument process lines: Seismic Category 1 criteria up to the instrument itself Instrument itself: Must remain functional during and after the seismic event	Instrument process lines: Seismic Category 1 criteria up to instrument itself; if an auto/remote isolation valve (EFCV) exists, Seismic Category 1 criteria need only extend through the isolation valve.
No seismic requirements	None should exist	Instrument process lines: No requirement except that instrument lines that penetrate the drywell[b] will have EFCVs and will meet Seismic Category 1 criteria from the drywell penetration through the EFCV. Instrument itself: No requirements

[a]Instruments are classified as safety related if they

- Initiate reactor shutdown (scram).
- Initiate reactor vessel-drywell isolation; this includes instruments that will isolate individual process lines upon indication of rupture or leakage.
- Initiate reactor building isolation
- Activate emergency core cooling systems and their support systems for initial and long-term core cooling.

Two additional classifications are implied by the above and are listed for completeness.

- Monitor/indicate possible radioactive process line leakage.
- Monitor for possible radioactive releases.

[b]Reactor coolant pressure boundary process lines will have EFCVs.

Table 3.10-2 Sheet 1 of 3

SUMMARY OF SEISMIC QUALIFICATION OF INSTRUMENTATION

Equipment Description	Maximum Usable Level (g) Horizontal	Vertical	Remarks
1. Voltage preamplifier	8.5	8.5	
2. TIP ball valve	25	25	
3. IRM detector	>1.5	>1.5	(maximum not determined)
4. Reactor level switch (Yarway) snap acting	10	10	
5. Temperature control switch	12	12	
6. Contractor (GECr 105)	12	12	
7. Indicator trip unit (GE/MAC)	15	15	
8. LPRM fixed incore detectors	>1.5	>1.5	(maximum not determined)
9. TIP shear valve assembly	10	10	
10. Timer	9	9	
11. Temperature switch	4	4	
12. Pressure transmitter	10	12	
13. Flow switch	4	10	HPCI and RHR minimum flow bypass
14. Pressure switch	11	11	
15. Flow switch (standby liquid flow)	15	15	
16. Flow converter	15	15	
17. Flow auxiliary unit	11	11	
18. Source-range monitor	3	15	
19. Intermediate-range monitor (dc)	1.5	0.5	
20. Power supply (20 vdc)	1.5	0.5	

Table 3.10-2 Sheet 2 of 3

SUMMARY OF SEISMIC QUALIFICATION OF INSTRUMENTATION

Equipment Description	Maximum Usable Level (g) Horizontal	Vertical	Remarks
21. Intermediate-range monitor	3	15	
22. Sensor converter	15	15	
23. Pressure switches	15	15	Scram and low pressure permissive
24. Temperature element	15	15	
25. Pressure switch (drywell)	15	15	Drywell pressure scram
26. Pressure switch	15	15	
27. Pressure switch	10	10	
28. Pressure switch (drywell)	15	15	
29. Pressure switch	2	2	
30. Relay (CR120A)	12	12	
31. Relay (HFA)	4	10	
32. Relay (HGA)[a]	1.1	5	
33. Relay (CR2820)	25	25	Time delay
34. Relay (CR120K)	25	25	
35. Relay (CR120KT)	12	12	Time delay
36. Switch (SBM)	25	25	
37. IRM range switch	8.5	8.5	
38. T/C selection switch	25	25	
39. Switch oil-tight (CR2940)	20	20	

[a]Application of this relay, where opening of a normally closed contact on a deengergized HGA relay can defeat a safety action, is unacceptable (see Reference 1)

Table 3.10-2 Sheet 3 of 3

SUMMARY OF SEISMIC QUALIFICATION OF INSTRUMENTATION

Equipment Description	Maximum Usable Level (g) Horizontal	Vertical	Remarks
40. IRM trip auxiliary	12	12	
41. Scram solenoid fuse panel	10	10	
42. Fuse	15	15	
43. Gamma chamber	1.5	0.5	
44. Controller	5	5	
45. Manual loading station	2	2	
46. Millivolt converter	3	3	
47. Pressure transmitter	2	2	
48. Flow transmitter	2	2	
49. Pressure transmitter	12	12	
50. Dual alarm unit	5	5	
51. Proportional amplifier (flow summer)	3	3	
52. Square-root converter	11	11	
53. GE/MAC power supply	11	11	
54. LPRM “page”	1.5	0.5	
55. APRM “page”	1.5	0.5	
56. ICPS “page”	1.5	0.5	
57. RBM “page”	1.5	0.5	
58. PRM system	1.5	0.5	
59. Agastat TR relay (GE No. 164C5257)	4.6	4.6	

Table 3.10-3

BECHTEL GENERIC INSTRUMENT LIST

Switch, pressure, bourdon tube
Switch, pressure, diaphragm
Switch, differential pressure
Switch, level, electronic
Switch, hand
Indicator, ammeter
Transmitter, pressure, force balance
Transmitter, differential pressure, force balance
Timer, electric, pneumatic rc
Controller, electronic input and output
Converter, signal, square root
Recorder, electronic
Power supply, solid state, dc, low voltage
Valve, control, air-operated
Valve, control, motor-operated
Valve, excess flow check
Valve, solenoid
Analyzer, hydrogen*
Analyzer, oxygen *
Electrical control relays

* Original design classification. In September 2003, NRC revised 10 CFR 50.44, "Standards for Combustible Gas Control for Nuclear Power Reactors," to downgrade these instruments to non-safety grade.

3.11 ENVIRONMENTAL DESIGN OF ELECTRICAL EQUIPMENT

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

3.11.1.1 Equipment Identification

Electrical equipment required to function under postulated harsh environment accident conditions is identified in the DAEC Equipment Data Base. The NRC Safety Evaluation Report (Reference 1) concludes that the DAEC Equipment Qualification Program is in compliance with the requirements of 10 CFR 50.49.

3.11.1.2 Environmental Service Conditions

3.11.1.2.1 Harsh and Mild Service Conditions

Environmental service conditions are categorized as harsh or mild. These categories describe the type of postaccident environment that the equipment is subjected to before or during the time that the equipment performs its safety function. To be classified as a harsh environment, a location must meet either of the following two criteria:

1. The total integrated radiation dose over the installed life of the component plus the required operating post accident time exceeds 1 x 10^5 rads for all equipment in the equipment environmental qualification program except for certain electronic devices identified in DAEC Design Control Procedures. The total integrated radiation dose limit over the required operating time for those electronic devices is 1 x 10^3 rads. DAEC Design Control Procedures address the radiation sensitivity of electronic devices such as silicon-controlled rectifiers, fiber optic cables, and metal oxide silicon field effect transistors and require design evaluations for their placement or relocation.

2. The nonradiation parameters as a result of a high energy line break (HELB) (temperature, pressure, humidity, etc.) would be significantly more severe than the non-radiation parameters that would occur during normal plant operation, including anticipated operational transients.

All other equipment areas are classified as mild environments. The environmental service conditions for the site are described in DAEC controlled document QUAL-SC101 (Reference 9) which references calculations and analyses to support the values assigned to each area. The environmental service conditions analysis QUAL-SC100 (Reference 2) contains more detailed "working" guidance on harsh environment criteria.

In assessing radiological conditions for accident doses to equipment, DAEC methods are those based on RG 1.89. No change was made to adopt Alternative Source Term methodology (NUREG-1465) to the EQ Program as allowed by RG 1.183.

3.11.1.2.2 Environmental Conditions Inside the Drywell

The environmental conditions inside the drywell on the basis of postulated LOCA and HELB conditions are described in Reference 2 and summarized in Table 3.11-1. The drywell pressure and temperature response following the postulated LOCA are provided in Chapter 15. Relative humidity is taken as 100% based on practical knowledge of a steam-water environment. Containment spray is limited to suppression pool water. The maximum flood level in the drywell has been calculated to be at [redacted]

3.11.1.2.3 Environmental Conditions Outside the Drywell, Subject to HELB

HELB is discussed in detail in Section 3.6. The environmental conditions (temperature, pressure, radiation and humidity) due to a pipe break in the steam tunnel, HPCI room, RCIC room, RWCU Heat Exchanger room, TIP room, TIP room mezzanine, torus room, and turbine building are listed in Reference 9. Analysis of submergence of safety-related electrical equipment outside the containment is not required but has been performed and all safety-related electrical equipment that is required to function to mitigate the consequences of the HELB in an area is located above the maximum flood elevation. Chemical and demineralized water sprays other than fire protection systems do not exist outside the drywell.

All electrical and control panels are located outside the drywell. Several panels are located in areas subject to HELB. Each of these panels has been reviewed to determine what components are located on each panel and the function of each component. It has been determined that no component on an electrical panel or instrument rack located in a HELB environment is required to function to mitigate the consequences of the HELB in that area. Therefore, HELB environmental conditions are not applicable for electrical and control panels or for components mounted on these panels or racks.

3.11.1.2.4 Environmental Conditions Outside the Drywell Where the Recirculation of Post-LOCA Fluid Would Occur

The 30-day dose levels outside the primary containment are listed in Reference 9. Detailed radiation dose calculations have been performed for plant areas in which electrical panels and instrumentation are located. Those instruments located in radiation harsh environments (per the definition in Reference 2) and which meet the qualification requirements in 10CFR50.49 or the Division of Operating Reactor (DOR) Guidelines, are evaluated for inclusion in the DAEC EQ Program based on the guidelines given in controlled administrative procedures and instructions. Those instruments which meet the program requirements for inclusion in the EQ Program are listed on the EQ Program master list (Reference 11). A list of equipment excluded from the EQ Program is contained in Reference 12. None of the areas in which electrical panels are located are identified as radiation harsh.

3.11.1.2.5 Mild Environments

The plant areas that are not covered under Sections 3.11.1.2.2, 3.11.1.2.3, and 3.11.1.2.4 are considered mild environmental areas with respect to HELB and post-LOCA radiation. Equipment located in these areas is protected and maintained in a suitable environmental condition by the heating and ventilation systems. Heating and ventilation systems employing redundant components and powered by essential power buses are provided for the following areas:

1. Control building, including control room, cable spreading room, battery rooms, and essential switchgear rooms.

2. Standby diesel-generator rooms.

3. Intake structure.

4. Emergency service water/RHR service water pump rooms.

5. Reactor building via standby gas treatment system and engineered safeguards area HVAC.

3.11.1.2.6 Evaluation of Service Conditions Inside Containment for a LOCA

As described in Reference 2, a combination of accidents establish the environmental conditions inside the drywell. The following discussion of service conditions is based on these accidents.

1. Temperature Conditions

   The drywell temperature conditions are established by an intermediate (1.0 sq. ft to 0.25 sq. ft) or a small (0.1 sq. ft to 0.01 sq. ft) pipe break accident.

   The drywell gas temperature reaches a maximum of 331°F which is above the containment design temperature of 281°F; however, the containment shell temperature does not reach that temperature.

   DAEC has established the qualification temperature for the drywell as 340°F for equipment which has been qualified by the DOR Guidelines as shown in the BWR Environmental Design Specification (Reference 10). New and replacement equipment is qualified to the current requirements of 10 CFR 50.49 using the temperature from the current analysis of record.

2. Pressure Conditions

The drywell pressure conditions are established by a recirculation line break. The maximum calculated DAEC drywell pressure is 45.7 psig.

3. Radiation

A 60-yr normal integrated dose plus a 30-day postaccident dose have been used to evaluate the adequacy of equipment qualification inside the drywell. Reference 9 gives the integrated dose for inside the drywell 30 days following a LOCA and the normal 60-year integrated dose. This plant-specific analysis establishes the maximum total integrated radiation doses inside the drywell.

Only gamma radiation has been considered if the component is enclosed in a nonorganic material (e.g., valve operators, splices, and terminal boards in junction boxes). For components in organic material (e.g., cable), it has been shown that 70 mils of jacket insulation reduces the beta dose to less than 10% of the total gamma dose. Therefore, beta radiation is only considered in the total integrated dose for unjacketed cable.

4. Submergence

The maximum possible flood level in the drywell is approximately [redacted] which corresponds to the entrance to the vent pipes from the drywell to the suppression pool. When the water level in the drywell reaches this elevation, it will flow into the torus via the vent pipes. All electrical equipment required to function under harsh environment accident conditions that is inside the primary containment is located above [redacted] and therefore submergence is not considered in the evaluation of environmental qualification adequacy.

The maximum possible flood level [redacted] is based on the following fluid volumes being discharged to the primary containment:

a. Reactor vessel.

b. Limited condensate storage tank discharge (limited by automatic transfer of ECCS pumps to the torus on high torus level or by manual operator action in accordance with emergency operating procedures).

c. Recirculation system piping.

d. Feedwater system piping.

e. Main steam piping.

These volumes, combined with the maximum normal volume of water in the suppression pool, is less than the total volume of the suppression pool up to the suppression chamber/drywell vacuum breaker. Therefore, the maximum water level in the drywell corresponds to the entrance to the vent pipes from the drywell to the suppression pool [redacted] The drywell can be flooded with river water from the RHR service water system. While this mode of operation is possible, it would be deliberate and the consequences of flooding the electrical equipment in the drywell would have to be considered before flooding. For the purpose of environmental qualification, the maximum flood elevation is calculated to be [redacted]

5. Containment Spray

The use of a chemical spray at the DAEC is limited to the potential use of the low-pressure injection system to inject suppression pool water into the drywell via spray headers. This suppression pool water is expected to maintain a relatively neutral pH even following the postulated LOCA. Hence, the corrosive reactions with suppression pool water are expected to be negligible because of its neutral condition.

3.11.2 QUALIFICATION TESTS AND ANALYSES

3.11.2.1 Qualification Test Requirements

Equipment and components requiring environmental qualification have been tested or analyzed in accordance with 10 CFR 50.49 or the NRC Division of Operating Reactors (DOR) guidelines. A record of the demonstrated level of qualification is contained in the DAEC controlled documents (Qual series) for each equipment and component item.

3.11.2.2 Qualification Test Results

Qualification test results are recorded in the DAEC Equipment Data Base and Environmental Qualification files for each piece of equipment in the Environmental Qualification Program.

3.11.2.3 Qualification Methods

Equipment qualification methods are consistent with 10 CFR 50.49 requirements or DOR guidelines, as applicable.

3.11.3 LOSS OF VENTILATION

The criteria governing the design of the air conditioning and ventilation for the control room and other safety-related equipment rooms require that maximum temperatures for normal operating conditions not be exceeded assuming the failure of any one active component.

Reference 9 compares the maximum temperatures for normal equipment operation and the calculated maximum room temperature under accident conditions. The calculated maximum room temperatures are based on outside ambient temperatures of 90°F dry bulb and 76°F wet bulb. Local outside ambient is expected to exceed these values approximately 2.5% of the time.

If abnormal outside ambient temperatures occur, equipment operation will not be impaired. As an example, outside ambient temperature of 105°F maintained for 24 hr would cause the control room temperature to increase to 95°F if only one cooling system were in operation. This maximum room temperature is well below the maximum operating temperature and will not make the room uninhabitable.

To achieve the design objectives, the listed areas are cooled and/or ventilated by systems that provide for 100% redundancy. The calculated temperatures are based on only one of these units operating. Furthermore, the units are housed in Seismic Category I structures or, in the case of the diesel-generators, in a structure whose failure will not cause the failure of the contained equipment. Where cooling water is required for chillers or cooling units, it is supplied by the emergency service water system, which is also designed using safety system criteria. The cooling and/or ventilating units are powered from redundant portions of the standby power system.

Based on this design, there is no postulated instance wherein a complete loss of air conditioning or ventilation can occur. The redundancy ensures that one unit will provide the designed cooling or ventilation, thereby ensuring the operability of safety-related control and electrical equipment.

The inspection and testing of the heating, ventilation and air conditioning (HVAC) systems is discussed in Section 9.4.

3.11.4 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

3.11.4.1 Chemical Environment

The chemical environment for design-basis accidents is described in Section 3.11.1.2.

3.11.4.2 Radiation Environment

The radiation environment for design-basis accidents is described in Section 3.11.1.2.

An evaluation of ESF systems is described in the following sections.

In addition to the analysis presented in Chapter 15, an evaluation was made of the adequacy of the containment and engineered safety features using the assumptions described in Section 3.11.4.2.1.

3.11.4.2.1 Source Terms Assumptions

For the purposes of calculating the dose, heat loading, airborne, or waterborne activity, the following assumptions were made:

1. The halogen and noble gas initial sources were taken from data on fission product loading for BWR fuel. This data contains a more extensive list of isotopes, and results in a larger (conservative) radiological source term than would be obtained using only the isotopes listed in Table IV, External Gamma Dose Rates of TID-14844.[6]

2. The core particulate activity was taken from the ANS standard afterheat curve.[7] The activity at any time was obtained by dividing the afterheat curve at that particular time by an average energy of 0.7 MeV.

3. The charcoal adsorber iodine loading includes iodine-129 and iodine-127. The amount of each of these isotopes in the core was determined from Blomeke and Todd.[8]

4. The activity in the suppression pool was assumed to be 50% of the core halogen inventory and 1% of the core particulate activity, which are instantaneously released to the suppression pool.

5. The airborne activity in the primary containment was assumed to consist of 100% of the core noble gas activity, 25% of the core halogen activity, and 1% of the core particulate activity, which are instantaneously released to the primary containment.

6. The airborne activity noted in item 5 is released at a constant leak rate of 2.0% per day to the secondary containment, uniformly mixed in the secondary containment and released to the standby gas treatment system at the rate of 1.0 air changes per day.

7. For the determination of the activity and heat loading on the charcoal adsorbers and the HEPA filters in the standby gas treatment system, the primary containment activity noted in item 5 above was assumed to be released at a constant leak rate of 2.0% per day directly to the standby gas treatment system where the filter and adsorber efficiency was assumed to be 100%.

A historic table of the activities in the various systems at various times after the TID-14844 release accident is shown in Table 3.11-3. The values in Table 3.11-3 are based on a primary containment leak rate of 2.0% per day. Reference 9 contains references to current analyses and source terms. These source terms contain a larger set of fission products and result in conservative dose rates compared to TID-14844.

3.11.4.2.2 Standby Gas Treatment System

The standby gas treatment system can contribute as a significant postaccident radiation source. The primary contribution is from the accumulation of radioactive iodine in the system's filter units. This source term is evaluated in calculations referred to in Reference 2.

The charcoal adsorber in each train contains approximately 1224 lb of net effective charcoal. Charcoal radioiodine loading calculations for the DAEC were originally derived using the TID-14844 Source Term methodology. The total iodine loading at the end of 30 days of a design basis LOCA is 1900g. This is predominantly iodine 127 and iodine-129. The resultant specific loading is 3.4 mg of iodine per gram of charcoal. Work performed at Oak Ridge National Laboratory has shown that removal efficiencies over 99% for both elemental and organic iodine can be achieved with charcoal loadings as high as 4.4 mg/gm.

DAEC analyses of dose consequences have been revised to use the Alternative Source Term of NUREG-1465. Most fission product iodine is now assumed to be released as a particulate form that will collect on the HEPA filters, reducing specific iodine loading 30 days after a LOCA to 0.003 mg of iodine per gram of carbon. This value is well below the 2.5 mg/gm iodine loading stated in Regulatory Guide 1.52. The larger values for iodine filter loading are still conservatively used for charcoal filter radiation heating.

3.11.4.2.3 Emergency Core Cooling System Components

Postaccident recirculation of radioactive contaminated fluid is a potentially significant source term. Such source terms are evaluated in calculations referred to in Reference 9.

The emergency core cooling system components are located in compartments shielded from the torus and the drywell. All ECCS components that handle torus water after the LOCA are located inside the secondary containment. There are no ECCS components located inside the primary containment that would be required to function following the postulated LOCA and that could suffer significant radiation damage. The ECCS components have been examined for materials that are subject to radiation damage.

The principal dose to the ECCS components located in the shielded compartments is from the radioactive torus water being circulated through the ECCS components. The residual heat removal (RHR) pump suction is 14 in.; the discharge is 12 in. The dose rate on the surface of a 14-in. standard weight pipe is listed in Table 3.11-2.

The integrated dose is for various times after the accident based on the source terms listed in Table 3.11-3. The doses to various ECCS components in the compartments are evaluated relative to the reference dose where the reference dose is equated to the 30-day dose on the surface of the 14-in. standard weight pipe. This reference dose is 5.6×10^5 rad.

The RHR and core spray pumps must operate after the LOCA. These pumps contain seals made of a carbon washer moving relative to a metal seat. Both the washer and seat materials have thresholds for damage on the order of 10^{11} rad (see Reference 13). This threshold for damage is orders of magnitude above the reference dose of 5.6×10^5 rad. The seals also contain elastomer O-rings made of buna-N and a special elastomer material. One of the O-rings has a Teflon retainer ring. All of the O-rings and seals are stationary. All of these materials suffer 25% damage from doses on the order of 4×10^6 rad, which is less than the reference dose. The radiation dose to the Teflon would be an even higher percentage of damage. However, if the O-rings and the Teflon retainer failed, there would be only a slight increase in leakage and the pump would continue to run and provide the essential ECCS cooling as all of the O-rings are stationary seals.

The RHR system and core spray pump motors contain reservoirs of lubricating oil. The lubricating oil will stand a dose on the order of 10^8 rad, which is well above the reference dose of 5.6×10^5 rad. The RHR and core spray pump motor insulation is a proprietary material with a radiation damage tolerance for 25% damage to the dielectric properties, which is essentially the same as the reference dose. However, it should be noted that the motors are located several feet from the pipes and therefore should receive less than the reference dose.

3.11.4.2.4 <u>Materials Within Primary Containment</u>

1. The Company is not committed to Regulatory Guide 1.54, June 1973. See Chapter 1, Section 1.8.30 for details.

2. Drywell Coating - The doses at the interior surface of the drywell are listed in Table 3.11-2. These doses are based on the source strength listed in Table 3.11-3.
2011-021 Service Level I coating materials on the interior surface of the drywell will be subjected to a 30-day dose of 2.3×10^7 rad at the surface of the drywell, which is
2011-021 within their radiation capability.

3. Electrical Penetrations - The primary containment electrical penetrations contain double seals. One seal is inside the primary shield and is subjected to a 30-day integrated dose of 2.3×10^7 rad. The other seal is outside the primary shield and would be subjected to a dose that is orders of magnitude less than at the interior surface of the drywell. Both seals are made of a vacuum-cast epoxy resin. Some low voltage power and control penetrations have been replaced. Seals on each end of these replacement penetrations are manufactured of polysulfone. The conductors passing through the replacement low voltage power and control penetrations are insulated with polyimide (Kapton).

3.11.4.2.5 <u>Control Room</u>

See Section 6.4.4

REFERENCES FOR SECTION 3.11

1. Letter from D. B. Vassallo, NRC, to L. Liu, Iowa Electric, Subject: Safety Evaluation Report, Environmental Qualification of Electric Equipment Important to Safety, Duane Arnold Energy Center, dated January 10, 1985.

2. QUAL-SC100 “Duane Arnold Energy Center Environmental Service Conditions Analysis”.

3. Deleted.

4. Letter from T. A. Ippolito, NRC, to Duane Arnold, Iowa Electric, Subject: Safety Evaluation Report for the Environmental Qualification of Safety-Related Electrical Equipment at the Duane Arnold Energy Center, dated June 3, 1981.

5. Letter from L. D. Root, Iowa Electric, to H. R. Denton, NRC, Subject: Response to NRC Staff's Safety Evaluation Report on Environmental Qualification of Safety-Related Electrical Equipment at the Duane Arnold Energy Center, dated September 8, 1981 (LDR-81-257).

6. J. J. DiNunno, et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 23, 1962.

7. "Proposed Standard - Energy Release Following Shutdown of Uranium Fueled, Thermal Reactors," American Nuclear Society (approved by Subcommittee ANS-5 on June 11, 1968).

8. J. O. Blomeke and M. F. Todd, "Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time," ORNL-2127, November 12, 1958.

9. QUAL-SC101, Duane Arnold Energy Center Environmental and Seismic Service Conditions.

10. GE Specification 22A3018, BWR Environmental Requirements.

11. QUAL-E001, “Environmental Qualification Master List”.

12. DBD-A64-001, “Environmental Qualification Topical Design Bases Document”.

13. Rockwell, III, Theodore, TID-7004 - Reactor Shielding Design Manual, March 1956.

Table 3.11-1

MAXIMUM ENVIRONMENTAL CONDITIONS INSIDE THE DRYWELL
FOLLOWING THE POSTULATED LOCA/HELB

2012-001

Temperature	Pressure[a]	Relative Humidity[b]	Containment Spray[c]	Gamma Radiation[d,e,g]	Submergence Elevation[f]
340°F	45.7 psig	100%	Suppression pool water	30-day 2.0×10^7 rad 60-year 3.2×10^7 rad 60-year plus 30-day 5.2×10^7 rad	[redacted]

[a]Section 6.2

[b]Postulated.

[c]Section 1.8.

[d]GE Specification 22A3018, BWR Environmental Requirements.

[e]The beta radiation dose mentioned in paragraph 3, of section 3.11.1.2.6 is not included in the 30-day accident dose shown in the table. The beta dose requirement is shown in reference 9.

[f]By evaluation.

[g]Reference 9 to Section 3.11.

Table 3.11-2

APPROXIMATE DOSE RATES AND INTEGRATED DOSES FOR VARIOUS EQUIPMENT OR LOCATIONS

2012-001

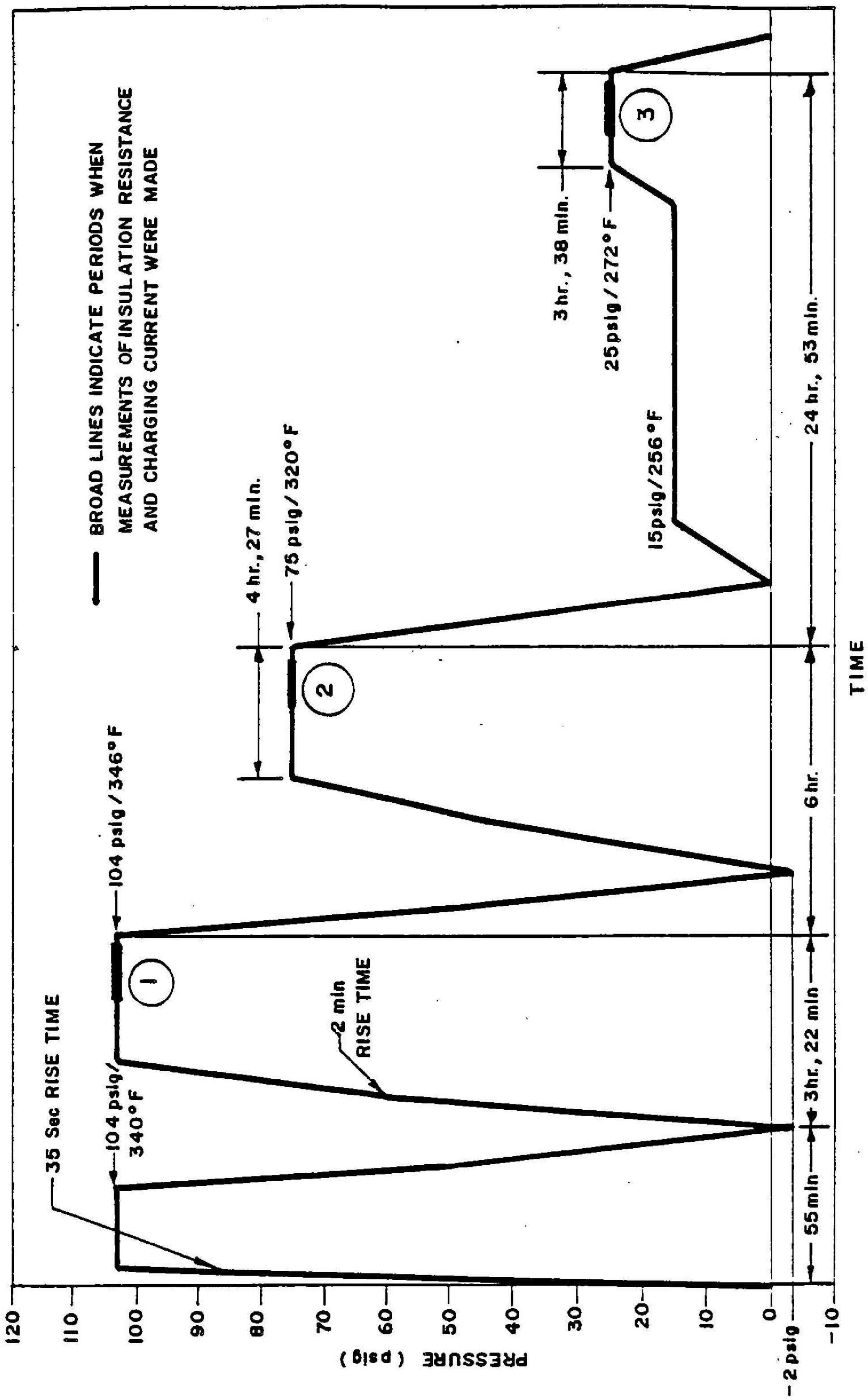
Location or Equipment	Maximum Dose Rate (rad/hr)	Loss-of-Coolant Fission Product Release Based on TID-14844 Assumptions, Integrated Dose (rad)				60-Year Normal Integrated Doses (rad) (100% load factor at rated power)
		12 hr	2 days	30 days	180 days	
Surface 14 in. standard weight pipe	1.4×10^{4}	7.5×10^{4}	2.5×10^{5}	5.6×10^{5}	7.9×10^{5}	0.0
Interior surface drywell (core region)	1.0×10^{6}	4.8×10^{6}	1.2×10^{7}	2.3×10^{7}	3.3×10^{7}	3.0×10^{7}
Floor or corner compartment containing core spray pump seals	3.3×10^{1}	1.2×10^{2}	1.3×10^{3}	3.8×10^{3}	9.1×10^{3}	7.5×10^{3}
Pump seals (ECCS)	1.4×10^{4}	7.5×10^{4}	2.5×10^{5}	5.6×10^{5}	7.9×10^{5}	0.0
Secondary containment ground floor area	1.3×10^{2}	5.3×10^{2}	4.8×10^{3}	1.4×10^{4}	3.3×10^{4}	4.5×10^{2}
Refueling floor	5.3×10^{2}	2.2×10^{3}	2.0×10^{4}	5.8×10^{4}	1.4×10^{5}	4.5×10^{2}

Table 3.11-3

ACTIVITY, MASS LOADING, AND HEAT LOADING AT VARIOUS LOCATIONS FOR TID-14844 RELEASE ASSUMPTIONS, PRIMARY CONTAINMENT LEAK RATE OF 2.0% PER DAY

Parameter	1 Hr	8 Hr	1 Day	10 Days	30 Days	Peak Value and Time
Activity in SP, Ci	2.1×10^8	9.6×10^7	5.6×10^7	1.8×10^7	8.0×10^6	2.1×10^8 at T = 0 hr
Heat load in SP, kW	2.4×10^3	6.1×10^2	2.9×10^2	7.5×10^1	4.3×10^1	2.4×10^3 at T = 0 hr
Activity airborne in PC, Ci	3.0×10^8	1.8×10^8	1.2×10^8	3.8×10^7	8.8×10^6	3.0×10^8 at T = 0 hr
Heat load in PC, kW	1.6×10^3	4.7×10^2	2.4×10^2	6.6×10^1	2.3×10^1	1.6×10^3 at T = 0 hr
Activity air in SC, Ci	2.0×10^5	8.4×10^5	1.4×10^6	6.6×10^5	1.1×10^5	1.6×10^6 at T = 50 hr
Heat load in SC, W	1.4×10^3	2.8×10^3	2.9×10^3	1.1×10^3	2.5×10^2	2.9×10^3 at T = 24 hr
Activity on HEPA, Ci	1.3×10^4	7.2×10^4	1.8×10^5	9.0×10^5	1.8×10^6	1.8×10^6 at T = 30 days
Heat load of HEPA, W	4.5×10^1	1.1×10^2	3.9×10^2	1.7×10^3	3.0×10^3	3.0×10^3 at T = 30 days
Activity on CF, Ci	5.7×10^4	2.3×10^5	4.5×10^5	9.0×10^5	4.8×10^5	9.0×10^5 at T = 10 days
Heat of CF, W	2.6×10^2	8.4×10^2	1.2×10^3	1.8×10^3	9.6×10^2	1.8×10^3 at T = 10 days
Iodine load on CF, g	1.8×10^0	2.1×10^1	6.6×10^1	6.6×10^2	1.9×10^3	1.9×10^3 at T = 30 days

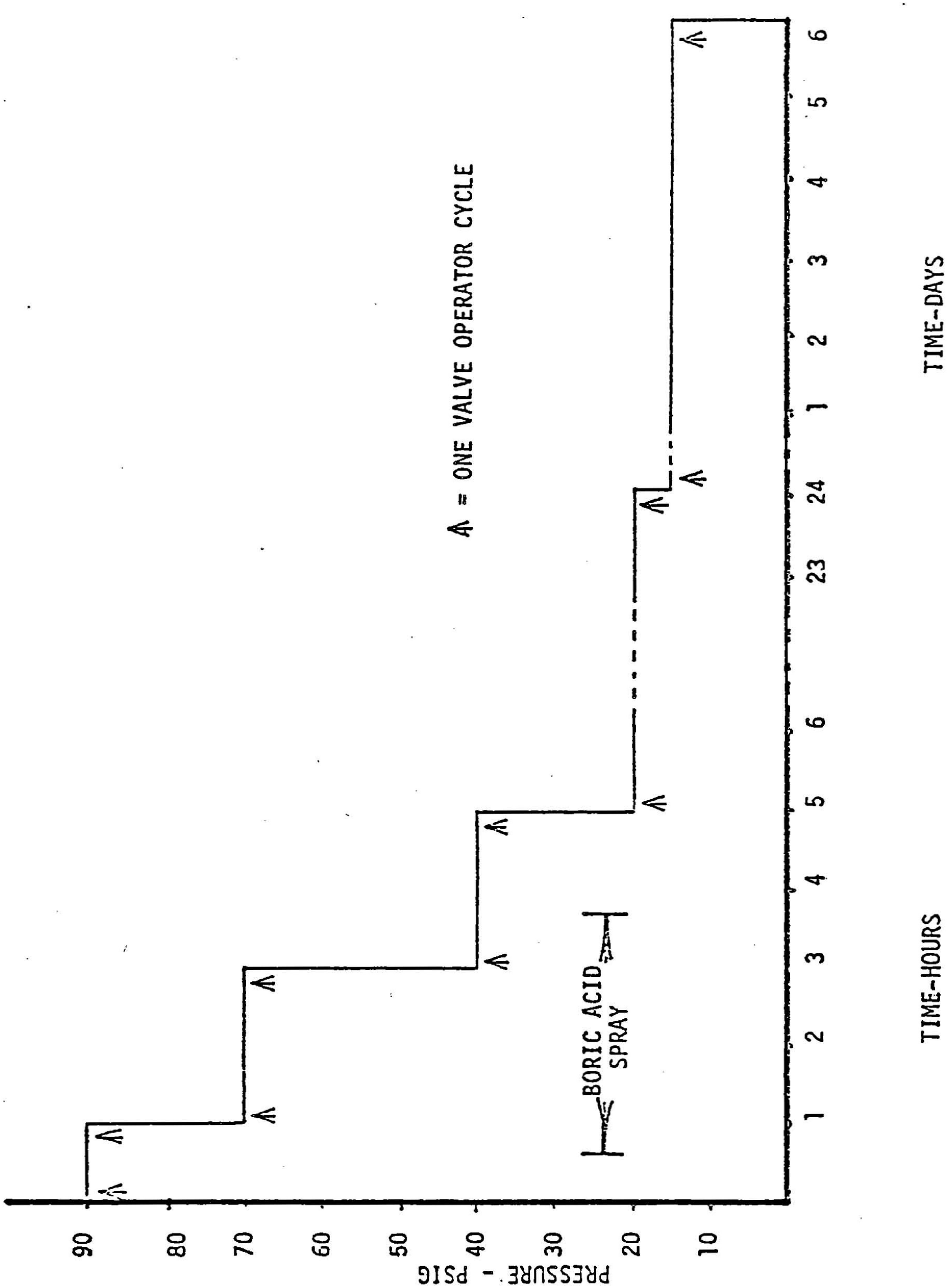
Key: SP = suppression pool
PC = primary containment
SC = secondary containment
HEPA = high-efficiency particulate air filter
CF = charcoal filter



DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Test Profile

Figure 3.11- 1



DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Schedule of Test Environments
Saturated Steam Pressure Cycle
and Limitorque Operation Cycle

Figure 3.11-2