CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Summary Description

This section describes how the design of River Bend Station conforms with the NRC General Design Criteria for Nuclear Power Plants, Appendix A of 10CFR50. The General Design Criteria establish minimum requirements for the design of nuclear power plants.

The General Design Criteria were not written specifically for the boiling water reactor (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 to a particular criterion is not directly measurable. In these cases, the 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 USAR.

Based on the content herein, GSU concludes that the nuclear power plant known as River Bend Station fully satisfies and is in compliance with the General Design Criteria.

3.1.2 Criterion Conformance

3.1.2.1 Quality Standards and Records (Criterion 1)

<u>Criterion</u>

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 assure 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.

Design Conformance

Structures, systems, and components important to safety are listed in Table 3.2-1. The total quality assurance (QA) program is described in Chapter 17 and is applied to the items as indicated in this table. The intent of the QA program is to assure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures as well as the documentation of the foregoing by keeping appropriate records. The total QA program of GSU and its principal contractors is responsive to and satisfies the intent of the quality-related requirements of 10CFR50, including Appendix B.

Documents are maintained which demonstrate that all the requirements of the QA program are being satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are observed; specified materials are used; correct procedures are utilized; qualified personnel are provided; and that the finished parts and components meet the applicable specifications for safe and reliable operation. These records are available so that any desired items of information are retrievable for reference. These records are maintained during the life of the operating licenses.

The detailed QA program developed by GSU and its contractors satisfies the requirements of Criterion 1.

3.1.2.2 Design Basis for Protection Against Natural Phenomena (Criterion 2)

Criterion

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.

Design Conformance

All safety-related structures, systems, and components are protected from or designed to withstand the effects of natural phenomena. The structures, systems, and components are classified in Section 3.2 in accordance with the safety functions they perform. The procedures to determine the effect of natural phenomena on these structures, systems, and components are discussed in Sections 3.3, 3.4, 3.5, 3.6, and 3.7. Using historical data presented in Sections 2.3, 2.4, and 2.5, the natural phenomena and their magnitude are selected in accordance with their probability of occurrence at the site location. The appropriate combinations of the effects of these natural phenomena with plant operating and accident conditions are identified in Sections 3.8, 3.9, 3.10, and 3.11.

3.1.2.3 Fire Protection (Criterion 3)

Criterion

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. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Design Conformance

The power plant is designed to minimize the occurrence of fire by the use of noncombustible and heat-resistant materials wherever practicable. Plant arrangement allows for isolation of known fire hazards. Nonflammable materials are used to the greatest extent possible to hinder the creation and subsequent spread of fire. Automatic and manual fire protection systems are provided throughout the plant (Section 9.5.1).

National Fire Protection Association (NFPA) standards are used as guides for the development of all fire protection systems.

3.1.2.4 Environmental and Missile Design Bases (Criterion 4)

Criterion

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 (LOCA). 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.

Design Conformance

All safety-related structures, systems, and equipment are protected from, or designed to withstand, the effects of any conditions associated with normal operation, maintenance, testing, and postulated accidents, including the LOCA. Discussion of dynamic effects associated with the postulated rupture of piping is contained in Section 3.6. Missile protection is discussed in Section 3.5. Section 3.11 contains a discussion of design environmental conditions.

3.1.2.5 Sharing of Structures, Systems, and Components (Criterion 5)

Criterion

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Design Conformance

River Bend Station is a single unit plant and therefore does not share structures, systems, or components among units.

3.1.2.6 Criterion 6

This criterion has not been promulgated by the NRC.

3.1.2.7 Criterion 7

This criterion has not been promulgated by the NRC.

3.1.2.8 Criterion 8

This criterion has not been promulgated by the NRC.

3.1.2.9 Criterion 9

This criterion has not been promulgated by the NRC.

3.1.2.10 Reactor Design (Criterion 10)

Criterion

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

Design Conformance

The reactor core components consist of fuel assemblies, control rods, in-core ion chambers, neutron sources, and related items. The mechanical design is based on 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 core is sized with sufficient heat transfer area and coolant flow to ensure that fuel design limits are not exceeded under normal conditions or anticipated operational occurrence.

The reactor protection system is designed to monitor certain reactor parameters, to sense abnormalities, and to scram the reactor, thereby preventing fuel design limits from being exceeded when trip set points are exceeded. Scram trip set points are selected on operating experience and by the safety design basis to prevent the core from exceeding the thermal hydraulic safety limits. Power for the reactor protection system is supplied by two independent, uninterruptible ac power supplies. An alternate power source is available for each reactor protection system bus.

An analysis and evaluation has been made of the effects upon core fuel following adverse plant operating conditions. The results of abnormal operational transients are presented in Chapter 15 and show that the minimum critical power ratio (MCPR) does not fall below the transient MCPR limit, thereby satisfying the transient design basis.

The reactor core and associated coolant, control, and protection systems are designed to assure 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:

- 1.2.1 Principal Design Criteria
- 1.2.2 Station Description
- 4.2 Fuel System Design
- 4.3 Nuclear Design
- 4.4 Thermal and Hydraulic Design
- 4.5 Reactor Materials
- 5.4.1 Reactor Recirculation Pumps
- 5.4.6 Reactor Core Isolation Cooling System
- 5.4.7 Residual Heat Removal System
- 15 Accident Analysis

3.1.2.11 Reactor Inherent Protection (Criterion 11)

Criterion

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.

Design Conformance

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
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 use of coolant flow rate for load following.

In a BWR, the moderator void coefficient is of 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 void coefficient of reactivity, the BWR has a number of inherent advantages, such as:

1. Use of coolant flow as opposed to control rods for load following
2. Inherent self-flattening of the radial power distribution
3. Ease of control
4. 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. Typically, the power coefficient at full power is about -0.04 $\Delta k/k/\ \Delta P/P$ at the beginning of life and about -0.03 $\Delta k/k/\ \Delta P/P$ at 10,000 MWD/T. These values are well within the range required for adequate damping of power and spatial xenon disturbances.

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, see the following sections:

- 1.2.1 Principal Design Criteria
- 4.3 Nuclear Design
- 4.4 Thermal and Hydraulic Design

3.1.2.12 Suppression of Reactor Power Oscillations (Criterion 12)

Criterion

The reactor core and associated coolant, control, and protection systems shall be designed to assure 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.

Design Conformance

The reactor core is designed to ensure that no power oscillation causes 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 under-damped, unacceptable power distribution behavior could only be expected to occur with power coefficients more positive 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:

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 reactor coolant pressure boundary (RCPB) from excessive pressures which threaten 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 assure that Criterion 12 is met.

For further discussion, see the following sections:

1.2.1 Principal Design Criteria
4.3 Nuclear Design
4.4 Thermal and Hydraulic Design
4.6 Functional Design of Reactivity Control System
5.2.2 Overpressurization Protection
7.2 Reactor Protection System
7.7 Control Systems Not Required for Safety
15 Accident Analysis

3.1.2.13 Instrumentation and Control (Criterion 13)

Criterion

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 RCPB, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Design Conformance

The neutron flux in the reactor core is monitored by five subsystems. The source range monitor (SRM) subsystem measures the flux from startup through criticality. The intermediate range monitor (IRM) subsystem overlaps the SRM subsystem and extends well into the power range. The power range is monitored by many detectors which make up the local power range monitor (LPRM) subsystem. The output from these detectors is used in many ways. The output of selected, core-wide sets of detectors is averaged to provide a core average neutron flux. This output is called the average power range monitor (APRM) subsystem. The traversing incore probe (TIP) subsystem provides a means for calibrating the LPRM system. Both the IRM and APRM subsystems generate scram trips to the reactor trip system. All subsystems but the TIP subsystem generate rod-block trips. Additional information on the neutron monitoring system is given in Chapter 7.

The reactor protection system protects the fuel barriers and the RCPB by monitoring plant parameters and causing a reactor scram when predetermined set points are exceeded. Separation of the scram and normal rod control function prevents failures in the reactor manual control circuitry from affecting the scram circuitry.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and RCPB, the containment and reactor vessel isolation control system initiates automatic isolation of appropriate pipelines whenever monitored variables exceed preselected operational limits.

RCPB leakage limits are established so that appropriate action can be taken to ensure the integrity of the RCPB. RCPB leakage rates are classified as identified and unidentified, which correspond, respectively, to the flow to the equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based upon the makeup capabilities of various reactor component systems. Flow integrator and recorders are used to determine the leakage flow pumped from the drain sumps. High pump fill-up rate and pump-out rate are alarmed in the main control room. The unidentified leakage rate as established in Chapter 5 is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before integrity of the RCPB is threatened.

A 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 reduction, accounting, and logging functions supplement procedural requirements for control rod manipulation during reactor startup and shutdown.

As previously noted, adequate instrumentation has been provided to monitor system variables in the reactor core, RCPB, and 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 abnormal operational occurrence or accident. These instrumentation and controls meet the requirements of Criterion 13.

For further discussion, see the following sections:

- 1.2.1 Principal Design Criteria
- 4.6 Functional Design of Reactivity Control System
- 5.2 Integrity of Reactor Coolant Pressure Boundary
- 5.4.5 Main Steam Isolation System
- 6.2 Containment Systems
- 7.2 Reactor Protection System
- 7.3.1 Containment and Reactor Vessel Isolation Control System
- 7.6 All Other Instrumentation Systems Required for Safety
- 7.7 Control Systems Not Required for Safety

3.1.2.14 Reactor Coolant Pressure Boundary (Criterion 14)

<u>Criterion</u>

The RCPB 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.

<u>Design Conformance</u>

The piping and equipment pressure parts within the RCPB through the outer isolation valve(s) are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies systems and components within the RCPB as Safety Class 1. The design requirements and codes and standards applied to this safety class ensure a quality product in keeping with the safety functions to be performed.

In order to minimize the possibility of brittle fracture within the RCPB, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Section 5.2.3 describes the methods utilized 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 drop weight or Charpy V-notch methods in accordance with ASME Boiler and Pressure Vessel Code, Section III. Where RCPB piping penetrates the containment, the fracture toughness temperature requirements of the RCPB materials apply.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures are employed which produce welds of complete fusion and free of unacceptable discontinuities. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel 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.

Section 5.2 contains the detailed material and examination requirements for the piping and equipment of the RCPB prior to and after its assembly and erection. Leakage testing and surveillance is accomplished as described in the evaluation against General Design Criterion 30.

The design, fabrication, erection, and testing of the RCPB assure an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following sections:

- 1.2.1 Principal Design Criteria
- 3 Design of Structures, Components, Equipment, and Systems
- 5.2.2 Overpressurization Protection
- 5.3 Reactor Vessel
- 5.4 Component and Subsystem Design
- 15 Accident Analysis
- 17 Quality Assurance Program

3.1.2.15 Reactor Coolant System Design (Criterion 15)

Criterion

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

Design Conformance

The reactor coolant system consists of the reactor vessel and appurtenances, the reactor recirculation system, the nuclear system pressure relief system, the main steam lines, the reactor core isolation cooling (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 assure high integrity of the RCPB throughout the plant lifetime. The reactor coolant system is designed and fabricated to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III as indicated in Chapter 3.

The auxiliary, control, and protection systems associated with the reactor coolant system act to provide sufficient margin to assure that the design conditions of the RCPB 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 which provides sufficient margin to assure that the design conditions of the RCPB are not exceeded is the automatic initiation of the nuclear system pressure relief system upon receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided that can discharge steam from the nuclear system to the suppression pool. 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 operation of the low pressure emergency core cooling systems (ECCS) 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 RCPB 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 assure that the requirements of Criterion 15 are satisfied.

For further discussion, see the following sections:

1.2.1	Principal Design Criteria
3	Design of Structures, Components, Equipment, and Systems
5.2.2	Overpressurization Protection
5.2.5	Detection of Leakage Through the Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4	Component and Subsystem Design
7.6	All Other Instrumentation Systems Required for Safety
15	Accident Analysis

3.1.2.16 Containment Design (Criterion 16)

<u>Criterion</u>

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 assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

<u>Design Conformance</u>

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The primary containment system, which includes the drywell, suppression pool, and containment vessel, is designed, fabricated, and erected so as to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture or equivalent failure of any coolant pipe within the primary containment. For more details on the primary containment design, refer to Sections 3.8.2 and 3.8.3. A secondary containment consists of the shield building annular space and the auxiliary building. The fuel building has been removed from secondary containment. The fuel building integrity is ony required during movement of recently irradiated fuel. These containment systems and

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their associated safety systems are designed and maintained so that offsite doses, which could result from postulated design basis accidents (DBAs), remain below the guideline values stated in 10CFR50.67 when calculated by the methods of Regulatory Guide 1.183. Amendment 132 revised the design basis accident methodology from Regulatory Guide 1.3 to Regulatory Guide 1.183. Per Regulatory Guide 1.183 the acceptance criteria was also revised from 10CFR100 to 10CFR50.67. Sections 6.2 and 15.6 contain detailed information that demonstrates compliance with Criterion 16.

3.1.2.17 Electric Power Systems (Criterion 17)

Criterion

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 for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the RCPB 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 to 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 assure that specified acceptable fuel design limits and design conditions of the RCPB are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure 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.

Design Conformance

Two offsite transmission systems and three onsite standby diesel generators with their associated battery systems are provided. Either of the two offsite transmission power systems or any two of the three onsite standby diesel generator systems have sufficient capability to operate safety-related equipment for cooling the reactor in the event of postulated accidents.

Electric power to the onsite electric distribution system is supplied by the preceding two offsite power circuits from a 230-kV switchyard located approximately 4,000 ft southwest of the plant. These power circuits are physically independent and on the same right-of-way, but they are designed and located so as to minimize the possibility of their simultaneous failure under operating and postulated accident and environmental conditions.

Each offsite power source can supply a designated 4.16-kV bus through a 230-4.16 kV preferred transformer. Each offsite power source thus has access to all engineered safety feature buses of each unit (Fig. 8.1-6). Power from either of the two offsite power sources is available immediately to the safety-related buses following a LOCA. Loss of offsite power to a safety-related bus results in automatic starting and connection of the associated standby diesel generator within 10 sec.

The degree of reliability of the onsite power supplies, including the batteries and onsite electric distribution system, is considered very high due to independence, redundancy, and testability of these systems to perform their safety functions following a single failure.

For more information refer to Chapter 8.

3.1.2.18 Inspection and Testing of Electric Power Systems (Criterion 18)

Criterion

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 conditions 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.

Design Conformance

The onsite power systems, consisting of the standby diesel generators with their associated switchgear assemblies (supplying power to safety-related equipment) and the associated battery systems, are designed and arranged for periodic testing of each system independently. During refueling shutdowns, a test is conducted to prove the operability of the automatic starting and load sequencing capability of the standby diesel generators. The testing procedure simulates a loss of bus voltage to start each standby diesel generator and connect it to its bus. The normal loading sequence is carried out.

Full-load testing of each standby diesel generator can be performed at appropriate periodic intervals by manually starting each standby diesel generator, manually synchronizing to the normal power supply, and manually loading the unit by governor adjustment. These tests prove the operability of the electric power systems under conditions as close to design as practical to assess the continuity of these systems and condition of the components.

The transfer of power between the offsite power system, the normal power system, and the onsite power system can also be demonstrated during refueling outages.

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Inspection and testing of electric power systems, described in Chapter 8 and the Operating License Manual, conform with the preceding NRC General Design Criterion 18.

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3.1.2.19 Control Room (Criterion 19)

Criterion

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 LOCAs. 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.

Design Conformance

A main control room is provided and equipped to operate the unit safely under normal and accident conditions. River Bend has received approval, via Amendment 132, to use the revised source term, also known as alternate source term (AST); therefore, radiation protection of the control room is governed by 10CFR50.67. 10CFR50.67 section (iii) states "Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

Main control room shielding and ventilation are designed to permit continuous occupancy of the main control room for the duration of a DBA without the dose to personnel exceeding 5 TEDE.

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A remote shutdown panel located in the control building complete with equipment, controls, and instrumentation is provided to bring the reactor to hot standby or a cold shutdown in a safe manner. The remote shutdown panel and adjacent controls are located in an area which is physically isolated from the main control room so that any event which could cause the main control room to become inaccessible has no effect on the availability of the remote shutdown panel and adjacent controls. Also, equipment, controls, and instrumentation are located throughout the unit to provide capability for a subsequent cold shutdown through the use of suitable procedures. The main control room and the remote shutdown panel conform with the requirements of Criterion 19.

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For further discussion, see the following sections:

- 3.2.1 Seismic Classification
- 6.4 Habitability Systems
- 7.4 Systems Required for Safe Shutdown
- 9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems
- 12.3 Radiation Protection Design Features

3.1.2.20 Protection System Functions (Criterion 20)

Criterion

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure 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.

Design Conformance

The reactor protection systems are the aggregate of protection systems or safety systems, including the reactor trip system, which are provided to sense abnormal and accident conditions and automatically initiate reactor shutdown and the operation of the other systems and components important to safety. The reactor trip system is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Fuel damage is prevented by 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 trip system includes the high-inertia, motor-generator power system, sensors, transmitters, 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, main steam isolation valve closure, and reactor vessel low and high water level prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Additional scram trips are

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initiated by drywell high pressure and scram discharge instrument volume high water level. Response by the reactor trip system is prompt and the total scram time is short. Control rod scram motion starts in less than 180 msec after the sensor contacts actuate.
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In addition to the reactor trip system, which provides for automatic shutdown of the reactor to prevent fuel damage, other 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 ECCS are initiated automatically to limit the extent of fuel damage following a LOCA.

Other systems automatically isolate the reactor vessel or the containment to prevent the release of significant amounts of radioactive materials to the environment. The controls and instrumentation for the ECCS and the isolation systems are initiated automatically when monitored variables exceed preselected operational limits.

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

For further discussion, see the following sections:

- 1.2.1 Principal Design Criteria
- 4.6 Functional Design of Reactivity Control Systems
- 5.2.2 Overpressurization Protection
- 5.4.5 Main Steam Isolation System
- 6.3 Emergency Core Cooling System
- 7.2 Reactor Trip System
- 7.3 Engineered Safety Feature Systems
- 7.6 All Other Instrumentation Systems Required for Safety
- 15 Accident Analysis

3.1.2.21 Protection System Reliability and Testability (Criterion 21)

<u>Criterion</u>

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 assure 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.

Design Conformance

Reactor protection (trip) system design provides assurance that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass, maintenance operation, calibration operation, or test to verify operational availability impairs the ability of the system to perform its intended safety function. Additionally, the system design assures that when a scram trip point is exceeded there is a high scram probability. However, should a scram not occur, other monitored components scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between 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 (trip) system includes design features that permit inservice testing. This ensures the functional reliability of the system should the reactor variable exceed the corrective action set point.

The reactor protection (trip) system initiates an automatic reactor shutdown if the monitored plant variables exceed preestablished limits. This system is arranged as two separately powered trip systems. Each trip system has two trip channels. An automatic or manual trip in either or both trip channels constitutes a trip system trip. A scram occurs when both trip systems have tripped. This logic scheme is called a one-out-of-two twice arrangement. The reactor protection (trip) system can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls. Two manual scram controls are associated with each trip system, one in each trip channel. Operating one manual scram control tests one trip channel and one trip system. The total test verifies the ability to deenergize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

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Control rod drive operability can be tested during normal reactor operation. Drive position indicators and in-core neutron detectors are used to verify control rod movement. Each control rod can be inserted one notch and then withdrawn to the original position without significantly perturbing the nuclear system at most power levels. One control rod is tested at a time. Control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on main control room instrumentation. Also, the hydraulic control unit scram accumulator and the scram discharge instrument volume level are monitored in the main control room during this test.

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The main steam isolation valves may be tested during reactor operation. Individually, they can be closed to 85 percent of full-open position without affecting the reactor operation. If reactor power is reduced sufficiently, the isolation valves may be fully closed. During the refueling operation, valve leakage rates can be determined.

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RHR system testing can be performed during normal operation by taking suction from the suppression pool and discharging through test lines back to the suppression pool. System design also permits testing the discharge valves to the reactor recirculation loops. The low pressure coolant injection (LPCI) mode can be tested after reactor shutdown.

Each active component of the ECCS provided to operate in a 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:

1.2.1 Principal Design Criteria
4.6 Functional Design of Reactivity Control System
5.4.5 Main Steam Isolation Valves
5.4.7 Residual Heat Removal System
6.2 Containment Systems
6.3 Emergency Core Cooling Systems
7.2 Reactor Protection System

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7.3 Engineered Safety Feature Systems

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7.6 All Other Instrumentation Systems Required for Safety
15 Accident Analysis

3.1.2.22 Protection System Independence (Criterion 22)

Criterion

The protection system shall be designed to assure 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.

Design Conformance

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 do not interfere with the operation of that function. Wiring for the reactor protection system outside the main control room enclosures is run in rigid or flexible conduit. No other wiring is run in these conduits. 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 channel actuator logic circuit from each trip system may be run in the same conduit.

The reactor protection (trip) system is designed to permit maintenance and diagnostic work while the reactor is operating without restricting the plant operation or hindering the output of that safety function. The flexibility in design afforded the protection system allows operational system testing by 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. Maintenance operation, calibration operation, or test, unless manually bypassed, can result in a single channel trip and one trip system trip. This leaves 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 assures that a scram occurs 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:

1.2.1 Principal Design Criteria
4.6 Functional Design of Reactivity Control System
5.4.5 Main Steam Isolation System
5.4.7 Residual Heat Removal System
6.3 Emergency Core Cooling Systems
7.2 Reactor Protection System
7.3 Engineered Safety Feature Systems
7.6 All Other Instrumentation Systems Required for Safety
15 Accident Analysis

3.1.2.23 Protection System Failure Modes (Criterion 23)

Criterion

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.

Design Conformance

The reactor protection (trip) system is designed to fail into a safe state. Use of an independent trip channel actuator 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 causes a channel trip. Only one trip channel in each trip system must be actuated to initiate a scram. Maintenance operation, calibration operation, or test, unless manually bypassed, can result in a single channel trip and one trip system trip . A failure of any one reactor protection (trip) system input or subsystem component produces a trip in one of two channels and therefore in one trip system. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon another channel trip in the other trip system.

The environmental conditions in which the instrumentation and equipment of the reactor protection (trip) system must operate were considered in establishing the component specifications. Instrumentation specifications 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:

1.2.1 Principal Design Criteria
6.3 Emergency Core Cooling Systems
7.2 Reactor Protection System
7.3 Engineered Safety Feature Systems

3.1.2.24 Separation of Protection and Control Systems (Criterion 24)

Criterion

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 the protection system. Interconnection of the protection and control systems shall be limited to assure that safety is not significantly impaired.

Design Conformance

There is separation between the reactor protection system and the process control 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 failure of any portion of the protection (trip) system. High scram reliability is designed into the reactor protection (trip) system and hydraulic control unit for the control rod drive. The scram signal and mode of operation override all other signals.

The containment and reactor vessel isolation control system is designed so that any one failure, maintenance operation, calibration operation, or test to verify operational availability does not impair the functional ability of the isolation control system to respond to essential variables.

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Process radiation monitoring is provided on process liquid and gas lines that may serve as discharge routes for radioactive materials.
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The protection system is separated from control systems as required in Criterion 24.

For further discussion, see the following sections:

- 1.2.1 Principal Design Criteria
- 4.6 Functional Design of Reactivity Control System
- 6.3 Emergency Core Cooling System
- 7.2 Reactor Protection System
- 7.3 Engineered Safety Feature Systems
- 7.6 All Other Instrumentation Systems Required for Safety

3.1.2.25 Protection System Requirements for Reactivity Control Malfunctions (Criterion 25)

Criterion

The protection system shall be designed to assure 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.

Design Conformance

The reactor protection (trip) system provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Any monitored variable that exceeds the scram set point initiates an automatic scram and does not impair the remaining variables from being monitored, and if one channel fails the remaining portions of the reactor trip system function.

The rod control and information system is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the rod control and information 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 design of the protection system assures that specified acceptable fuel 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:

1.2.1 Principal Design Criteria
4.3 Nuclear Design
4.4 Thermal and Hydraulic Design
4.6 Functional Design of Reactivity Control System
7.2 Reactor Protection System
7.7 Control Systems Not Required for Safety
15 Accident Analysis

3.1.2.26 Reactivity Control System Redundancy and Capability (Criterion 26)

Criterion

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 assure 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 assure that acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Design Conformance

Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control employs control rod assemblies which contain boron carbide (B_4C) powder. Positive insertion of these control rods is provided by means of the control rod drive hydraulic system. The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup, and shutdown) via operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram does not adversely affect the capability to maintain the core within fuel design limits.

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The Hydraulic Control Unit 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. 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 event that they do occur. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance and, simultaneously, low individual rod worths. The operating procedures to accomplish such patterns are supplemented by the rod pattern control system (RPCS), which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. 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 occurrence of a limited number of stuck rods does not hinder the capability of the control rod system to render the core subcritical.

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The second independent reactivity control system is provided by the reactor coolant recirculation system. By varying reactor flow, it is possible to effect the type of reactivity changes necessary for planned, normal power changes (including xenon burnout). In the unlikely event that reactor flow is suddenly increased to its maximum value (pump runout), the core does not exceed fuel design limits because the power flow map defines the allowable initial operating states such that the pump runout does not violate these limits.

The control rod system is capable of holding the reactor core subcritical under cold conditions, even when the control rod of highest worth is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd-0-) to control the high reactivity of fresh fuel. In addition, the standby liquid control system (SLCS) is available to add soluble boron to the core and render it subcritical, as discussed in Section 9.3.5.

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:

- 1.2.1 Principal Design Criteria
- 4.6 Functional Design of Reactivity Control System
- 7.6 All Other Instrumentation Systems Required for Safety
- 7.7 Control Systems Not Required for Safety

3.1.2.27 Combined Reactivity Control Systems Capability (Criterion 27)

Criterion

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

Design Conformance

There is no credible event applicable to the BWR which requires 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. The BWR design is capable of maintaining the reactor core subcritical, including allowance for a stuck rod, without the addition of any poison to the reactor coolant. Abnormalities are sensed, and if protection system limits are reached, corrective action is initiated through an automatic insertion of control rods. High integrity of the protection system is achieved through the combination of logic arrangement, actuator redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by separation of scram and manual control circuitry, individual control units for each control rod, and fail-safe design features built into the rod drive system. Response by the reactor protection is prompt and the total scram time is short.

In the event that more than one control rod fails to insert and the core cannot be maintained in a subcritical condition by the control rods alone as the reactor is cooled down subsequent to initial shutdown, the SLCS is activated to inject soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical; hence, only decay heat is generated by the core which can be removed by the RHR, thereby ensuring that the core is always coolable.

The design of the reactivity control systems assures 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:

1.2.1 Principal Design Criteria
4.3 Nuclear Design
4.4 Thermal and Hydraulic Design
4.6 Reactivity Control System
7.2 Reactor Protection System
7.6 All Other Instrumentation Systems Required for Safety
7.7 Control Systems Not Required for Safety
15 Accident Analysis

3.1.2.28 Reactivity Limits (Criterion 28)

Criterion

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 RCPB 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.

Design Conformance

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 RPCS prevents withdrawal other than by the preselected rod withdrawal pattern. The RPCS assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level operations control rod procedures.

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The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod that prevents rapid rod ejection. 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 6-in increments and the rod withdrawal rate is controlled through the hydraulic valve to a nominal speed of 3 ips.

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The accident analysis (Chapter 15) evaluates the postulated reactivity accidents, as well as abnormal operational transients, 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 RCPB. 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 which 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:

1.2.1 Principal Design Criteria
3 Design of Structures, Components, Equipment, and Systems

4.2 Fuel System Design
4.3 Nuclear Design
4.5.1 Control Rod System Structural Materials
4.6 Functional Design of Reactivity Control System
5.2.2 Overpressurization Protection
5.3 Reactor Vessel
5.4.4 Main Steam Flow Restrictions
5.4.5 Main Steam Isolation System
7.6 All Other Instrumentation Systems Required for Safety
15 Accident Analysis

3.1.2.29 Protection Against Anticipated Operational Occurrences (Criterion 29)

Criterion

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

Design Conformance

The high functional reliability of the reactor protection (trip) system 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.

A high reliability of timely 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, RHR pumps, etc, are tested during 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 the failure probabilities of individual components and the reliability effects during individual component testing on the portion of the system not undergoing testing. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action set point.

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:

- 1.2.1 Principal Design Criteria
- 4.6 Functional Design of Reactivity Control System
- 5.4.5 Main Steam Isolation System
- 5.4.7 Residual Heat Removal System
- 6.2 Containment Systems
- 6.3 Emergency Core Cooling Systems
- 7.2 Reactor Protection System
- 7.3 Engineered Safety Feature Systems
- 7.6 All Other Instrumentation Systems Required for Safety
- 15 Accident Analysis
- 16 Technical Specifications

3.1.2.30 Quality of Reactor Coolant Pressure Boundary (Criterion 30)

Criterion

Components that are part of the RCPB 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.

Design Conformance

By utilizing conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components which compose the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Chapter 5. Further, product and process quality planning is provided as described in Chapter 17 to assure 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 RCPB, further discussion on this subject is treated in the response to Criterion 14, Reactor Coolant Pressure Boundary.

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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 RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, high-sensitivity sump level measurement, increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines and by 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 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 concurrent with a loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the leak detection system provides protection from small leaks, the ECCS provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

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The RCPB and the leak detection system are designed to meet the requirements of Criterion 30.

For further discussion, see the following sections:

1.2.1 Principal Design Criteria
3 Design of Structures, Components, Equipment, and Systems
5.2.2 Overpressurization Protection
5.2.5 Detection of Leakage Through the Reactor Coolant Pressure Boundary
5.3 Reactor Vessel
5.4.1 Reactor Recirculation Pumps
7.6 All Other Instrumentation Required for Safety
17 Quality Assurance

3.1.2.31 Fracture Prevention of Reactor Coolant Pressure Boundary (Criterion 31)

Criterion

The RCPB shall be designed with sufficient margin to assure 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 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.

Design Conformance

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against non-ductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel, the reactor pressure vessel is designed to meet the requirements of ASME Code, Section III, Appendix G which considers material properties, steady-state and transient stresses, and the size of flaws.

The nil-ductility transition temperature (NDTT) is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDTT increases as a function of neutron exposure at integrated neutron exposures greater than about 1 x 10-17 nvt with neutrons of energies in excess of 1 MeV.

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 upon the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. The effect of neutron radiation on the fracture toughness of the reactor pressure vessel material has been considered in the design, and plant operation is modified as necessary to accommodate the small change in the initial reference transition temperature (IRTT) that occurs.

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

For further discussion, see the following sections:

- 3 Design of Structures, Components, Equipment, and Systems
- 5.2 Integrity of the Reactor Coolant Pressure Boundary
- 5.3 Reactor Vessel

3.1.2.32 Inspection of Reactor Coolant Pressure Boundary (Criterion 32)

Criterion

Components that are part of the RCPB 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.

Design Conformance

The reactor pressure vessel design and engineering effort include provisions for inservice inspection. Removable plugs in the primary shield and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system safety relief valves, recirculation system, and on the main steam and feedwater systems extending out to and including the first isolation valve outside containment. Inspection of the RCPB is in accordance with the ASME Boiler and Pressure Vessel Code, Section XI. Section 5.2 defines the inservice inspection plan, access provisions, and areas of restricted access.

The reactor recirculation piping and main steam piping are hydrostatically tested with the reactor pressure vessel at a test pressure that is in accordance with Section III of the ASME Code.

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

The plant testing and inspection program ensure that the requirements of Criterion 32 are met.

For further discussion, see the following sections:

- 3 Design of Structures, Components, Equipment, and Systems
- 5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary
- 5.2.5 Detection of Leakage Through the Reactor Coolant Pressure Boundary
- 5.3 Reactor Vessel
- 5.4.1 Reactor Recirculation Pumps

3.1.2.33 Reactor Coolant Makeup (Criterion 33)

Criterion

A system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure 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.

Design Conformance

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 RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, high-sensitivity sump level measurement, increased frequency of sump pump operation, and by measuring fission product concentration. 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 capabiliity 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 RCIC system. While the leak detection system provides protection from small leaks, the ECCS 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 RCPB 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:

- 5.2.5 Detection of Leakage Through the Reactor Coolant Pressure Boundary
- 5.4.6 Reactor Core Isolation Cooling System
- 6.3 Emergency Core Cooling System
- 7.6 All Other Instrumentation Systems Required for Safety

3.1.2.34 Residual Heat Removal (Criterion 34)

Criterion

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 RCPB are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure 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.

Design Conformance

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.

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3. Remove heat from the containment in the long term following a LOCA.

The major equipment of the RHR system consists of heat exchangers and main system 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 containment cooling function, which is the mode requiring the maximum heat exchanger area.

Two loops, each consisting of a heat exchanger, main system pump, and associated piping, are located in separate protected areas of the auxiliary building. A third loop, made up of a pump and associated piping, is also located in a separate area of the auxiliary building to minimize the possibility of a single physical event causing the loss of the entire system. The loops of the RHR are connected so that any failure of one loop cannot cause failure of another.

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

1. Shutdown cooling
2. Suppression pool cooling
3. LPCI.

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Both normal ac power and auxiliary onsite power systems provide adequate 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 detection and isolation of system faults.

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

Two full-capacity standby diesel generators are provided to supply a source of electrical 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 system is adequate to remove residual heat from the reactor core to assure fuel and RCPB design limits are not exceeded. Redundant onsite electric power systems are provided. The design of the RHR system, including its power supply, meets the requirements of Criterion 34.

For further discussion, see the following sections:

- 5.4.7 Residual Heat Removal System
- 6.2 Containment Systems
- 6.3 Emergency Core Cooling Systems
- 7.3 Engineered Safety Feature Systems
- 8.3.1 AC Power Systems
- 9.2.1 Normal Service Water
- 9.2.7 Standby Service Water System
- 15 Accident Analysis

3.1.2.35 Emergency Core Cooling (Criterion 35)

Criterion

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 assure 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.

Design Conformance

The ECCS consists of the following:

1. High-pressure core spray (HPCS) system

2. Automatic depressurization system (ADS)

3. Low-pressure core spray (LPCS) system

4. Low-pressure core injection (LPCI) - an operating mode of the RHR system.

The ECCSs are designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the RCPB, including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel. The design basis break for the ECCS is the complete and sudden rupture of a recirculation system suction line which is not the largest pipe connected to the vessel, but produces the highest cladding temperature results.

The HPCS system consists of a single motor-driven pump, system piping, valves, controls, and instrumentation. The HPCS system is provided to assure that the reactor core is adequately cooled to prevent excessive fuel-clad temperatures for breaks in the nuclear system that do not result in rapid depressurization of the reactor vessel. The HPCS continues to operate when reactor vessel pressure is below the pressure at which LPCI operation or LPCS system operation maintain core cooling. A source of water is available from either the condensate storage tank or the suppression pool.

The ADS functions to reduce the reactor pressure so that flow from LPCI and the LPCS enters the reactor vessel in time to cool the core and prevent excessive fuel-clad temperature. The ADS uses seven of the nuclear system pressure relief valves to relieve the high-pressure steam to the suppression pool.

The LPCS system consists of: a centrifugal pump that can be powered by offsite power or the standby ac power system; a spray sparger in the reactor vessel above the core (separate from the HPCS sparger); piping and valves to convey water from the suppression pool to the sparger; and associated controls and instrumentation. In case of low water level in the reactor vessel or high pressure in the drywell, the LPCS 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 which initiate the LPCS system and operates independently to achieve the same objective by flooding the reactor vessel.

In case of low water level in the reactor or high pressure in the 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. Protection provided by LPCI extends to a small break where the ADS has operated to lower the reactor vessel pressure so LPCI and LPCS can start to provide core cooling.

Results of the performance of the ECCSs for the entire spectrum of liquid line breaks are discussed in Section 6.3. Peak cladding temperatures are well below the 2,200°F design basis.

Also provided in Section 6.3.3 is an analysis to show that the ECCSs conform to 10CFR50, Appendix K. This analysis shows complete compliance with the Final Acceptance Criteria with the following results:

1. Peak clad temperatures are well below the 2,200°F NRC acceptability limit.

2. The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1 percent acceptability limit.

3. The clad temperature transient is terminated while core geometry is still amenable to 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 onsite electrical power systems for the ECCS are represented in the evaluation against Criterion 34.

The ECCSs provided are adequate to prevent fuel and clad damage, which could interfere with effective core cooling and to limit clad metal-water reaction to a negligible amount.

The design of the ECCSs, including their power supply, meets the requirements of Criterion 35.

For further discussion, see the following sections:

- 5.4.7 Residual Heat Removal System
- 6.3 Emergency Core Cooling Systems
- 7.3 Engineered Safety Feature Systems
- 8.3.1 AC Power Systems
- 9.2.1 Normal Service Water
- 9.2.7 Standby Service Water Systems
- 15 Accident Analysis

3.1.2.36 Inspection of Emergency Core Cooling System (Criterion 36)

Criterion

The ECCS 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 assure the integrity and capability of the system.

Design Conformance

The ECCSs are as discussed in Criterion 35. The engineering and design effort for these systems include inservice inspection considerations. The spray spargers within the vessel are accessible for inspection during each refueling outage. Removable plugs in the primary shield and/or panels in the insulation provide access for examination of nozzles. Removable insulation is provided on the ECCS piping out to and including the first isolation valve outside containment. Inspection of the ECCS is in accordance with the intent of Section XI of the ASME Code. Section 5.2.4 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 drywell can be visually inspected at any time. Components inside the drywell 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 ECCS that are part of the RCPB are designed to specifications for inservice inspection to detect defects which might affect the cooling performance. Particular attention is 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 ensures that the requirements of Criterion 36 are met.

For further discussion, see the following sections:

- 4.2 Fuel System Design
- 5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary
- 5.3 Reactor Vessel
- 6.3 Emergency Core Cooling Systems

3.1.2.37 Testing of Emergency Core Cooling System (Criterion 37)

Criterion

The ECCS 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 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.

Design Conformance

The ECCS consists of the HPCS system, ADS, LPCI mode of the RHR system, and LPCS system. Each of these systems is provided with sufficient test connections and isolation valves to permit appropriate periodic pressure testing to assure the structural and leaktight integrity of its components.

The HPCS, LPCS, LPCI, and ADS are designed to permit periodic testing to assure the operability and performance of the active components of each system.

The pumps and valves of these systems are tested periodically to verify operability. Flow rate tests are conducted on LPCS, LPCI, and HPCS systems.

The ECCS is subjected to tests to verify the performance of the full operational sequence that brings each system into operation. The testing of the associated cooling water systems is discussed in the evaluation of Criterion 46. It is concluded that the requirements of Criterion 37 are met.

For further discussion, see the following sections:

5.2.2	Overpressurization Protection
6.3	Emergency Core Cooling Systems
7.3	Engineered Safety Feature Systems
8.3.1	AC Power Systems
14	Initial Test Program
16	Technical Specifications

3.1.2.38 Containment Heat Removal (Criterion 38)

Criterion

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 assure 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.

Design Conformance

In the event of a LOCA within the drywell, the pressure suppression system rapidly condenses the steam to prevent overpressurization. The pressure suppression concept employs a drywell that houses the nuclear system and a large volume of water outside the drywell called the suppression pool. Any increase in pressure in the drywell resulting from a leak in the nuclear system is relieved by venting to the suppression pool where any steam that was released or formed by flashing is condensed. Cooling systems remove heat from the reactor core, the drywell, and from the water in the suppression pool during accident conditions, and thus provide continuous cooling within the containment.

The ECCS 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 initiates the ECCS to prevent excessive fuel temperature. Sufficient water is provided in the suppression pool to passively accommodate the heat which can be released initially from the postulated pipe failure.

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

The pressure suppression system is capable of rapid drywell pressure and temperature reduction following a LOCA so that design limits are not exceeded. Redundant onsite electrical power systems provide assurance that system safety functions can be accomplished. The design of the containment heat removal system meets the requirements of Criterion 38.

For further discussion, see the following sections:

- 5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary
- 5.4.7 Residual Heat Removal System
- 6.2 Containment Systems
- 6.3 Emergency Core Cooling Systems
- 6.6 Inservice Inspection of ASME Code Class 2 and 3 Components
- 7.3 Engineered Safety Feature Systems
- 8.3.1 AC Power Systems
- 9.2.1 Normal Service Water
- 9.2.7 Standby Service Water Systems
- 15 Accident Analysis

3.1.2.39 Inspection of Containment Heat Removal System (Criterion 39)

Criterion

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 assure the integrity and capability of the system.

Design Conformance

Provisions are made to facilitate periodic inspections of active components and other important equipment of the containment heat removal systems. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the drywell can be visually inspected at any time and are inspected periodically. The testing frequencies of most components are correlated with the component inspection.

The pressure suppression pool is designed to permit appropriate periodic inspection. Space is provided outside the drywell for inspection and maintenance.

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

For further discussion, see the following sections:

- 5.4.7 Residual Heat Removal System
- 6.2 Containment Systems
- 6.3 Emergency Core Cooling Systems
- 7.3 Engineered Safety Feature Systems
- 9.2.1 Normal Service Water
- 9.2.7 Standby Service Water Systems

3.1.2.40 Testing of Containment Heat Removal System (Criterion 40)

Criterion

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.

Design Conformance

The containment heat removal function is accomplished by the suppression pool cooling mode of the RHR system.

The RHR system is provided with sufficient test connections and isolation valves to permit periodic pressure and flow rate testing.

The pumps and valves of the RHR are operated periodically to verify operability. The suppression pool cooling mode is not automatically initiated, but testing of the components is periodically verified. The testing of associated cooling water systems is discussed in the conformance to Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion, see the following sections:

- 5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary
- 5.4.7 Residual Heat Removal System
- 6.2 Containment Systems
- 7.3 Engineered Safety Feature Systems
- 8.3.1 AC Power Systems
- 16 Technical Specifications

3.1.2.41 Containment Atmosphere Cleanup (Criterion 41)

Criterion

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 quality 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 assure 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 assure 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) its safety function can be accomplished, assuming a single failure.

Design Conformance

Fission products, hydrogen, oxygen, and other substances released from the reactor are confined within the primary containment. Leakage from the containment during normal plant operation and following the DBA enters the annulus volume. This leakage is collected in the annulus and discharged from the plant through the plant exhaust duct during normal operation or diverted through the standby gas treatment system (Section 6.2.3) during accident conditions. A hydrogen mixing system is provided to mix the drywell and containment atmospheres (Section 6.2.5).

The DBA hydrogen recombiner system (Section 6.2.5) recirculates a portion of the containment atmosphere through a recombiner to maintain the hydrogen concentration below 4 volume percent. A hydrogen purge system is provided as backup for the hydrogen recombiner system.

These systems have sufficient redundancy to be able to withstand a single failure and are operable from either onsite or offsite power.

3.1.2.42 Inspection of Containment Atmosphere Cleanup Systems (Criterion 42)

<u>Criterion</u>

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

<u>Design Conformance</u>

The annulus pressure control system, the standby gas treatment system, and the hydrogen mixing, recombiner, and continuous containment purge systems are designed to permit appropriate periodic inspection of the important components (Sections 6.2.3 and 6.2.5, respectively).

3.1.2.43 Testing of Containment Atmosphere Cleanup Systems (Criterion 43)

<u>Criterion</u>

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 its components.

2. The operability and performance of the active components of the systems, such as fans, filters, dampers, pumps, and valves.

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 standby power sources, and the operation of the associated systems.

Design Conformance

The standby gas treatment, hydrogen mixing, hydrogen recombiner, and purge systems are designed to permit periodic pressure and functional testing of their components (Sections 6.2.3 and 6.2.5, respectively).

3.1.2.44 Cooling Water (Criterion 44)

Criterion

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 operating and accident conditions.

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

Design Conformance

The normal service water and reactor plant component cooling water systems provide cooling for removal of heat from structures, systems, and components important to safety during normal operation. The reactor plant component cooling system is an intermediate cooling system that transfers heat from heat exchangers containing reactor coolant or other radioactive liquids.

The standby service water system provides cooling water for removal of heat from the structures, systems, and components important to safety during all abnormal and accident conditions if normal cooling means are not available. The standby service water system supplies cooling water to the

RHR heat exchangers, standby diesel generators, containment unit coolers, main control room air conditioning chillers, auxiliary building unit coolers, control building unit coolers, fuel pool coolers, and penetration valve leakage control compressors. The standby service water system is designed to Safety Class 3 and Seismic Category I requirements. Redundant safety-related components served by the standby service water system are supplied through the redundant supply headers and returned through redundant discharge lines. Electric power for operation of redundant safety-related components of the standby service water system is supplied from separate redundant offsite and redundant onsite standby power sources. No single failure renders the standby service water system incapable of performing its safety function.

Referenced sections are as follows:

- 8.3.1 AC Power Systems
- 9.2.1 Normal Service Water System
- 9.2.2 Reactor Plant Component Cooling Water System
- 9.2.5 Ultimate Heat Sink
- 9.2.7 Standby Service Water System

3.1.2.45 Inspection of Cooling Water System (Criterion 45)

Criterion

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

Design Conformance

The standby service water, the normal service water, and the reactor plant component cooling water systems are designed to permit appropriate periodic inspection in order to assure the integrity and capability of the system.

Referenced sections are as follows:

- 9.2.1 Normal Service Water System
- 9.2.2 Reactor Plant Component Cooling Water System
- 9.2.7 Standby Service Water System

3.1.2.46 Testing of Cooling Water System (Criterion 46)

Criterion

The cooling water 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 design as practical, the performance of full operational sequence that brings the system into operation for reactor shutdown and for LOCA, including operation of applicable portions of the protection systems and the transfer between normal and emergency power sources.

Design Conformance

The normal service water and reactor plant component cooling water systems are in operation during normal plant operation and shutdown. Thus, component performance is continuously demonstrated.

The standby service water system, which is operated only during emergency conditions, is periodically tested to assure the structural and leaktight integrity of components, the operability of active components, and the operability of the total system. The design of the standby service water system and its components allows, to the extent practicable, demonstration of operability through periodic startup and operational testing of the system as required for operation during a LOCA or a loss of offsite power.

Referenced sections are as follows:

- 9.2.1 Normal Service Water System
- 9.2.2 Reactor Plant Component Cooling Water System
- 9.2.7 Standby Service Water System
- 16 Technical Specifications

3.1.2.47 Criterion 47

This criterion has not yet been promulgated by the NRC.

3.1.2.48 Criterion 48

This criterion has not yet been promulgated by the NRC.

3.1.2.49 Criterion 49

This criterion has not yet been promulgated by the NRC.

3.1.2.50 Containment Design Basis (Criterion 50)

Criterion

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.

Design Conformance

The primary containment structure, including access openings, penetrations, and the containment heat removal system, is designed to withstand the peak accident pressure and temperature that could occur during any postulated LOCA. Sections 3.8 and 6.2 have detailed information that demonstrates compliance with Criterion 50.

3.1.2.51 Fracture Prevention of Containment Pressure Boundary (Criterion 51)

Criterion

The reactor containment boundary shall be designed with sufficient margin to assure 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.

Design Conformance

The reactor containment boundary is designed to the load combinations shown in Section 3.8.2 that cover the operational, testing, and postulated accident conditions. Each condition results in a stress/strain level that is related to its corresponding temperature which is the basis for comparison with the allowable limits.

The ferritic steel to be used for the reactor containment boundary is specified so that the toughness of the material meets the preceding established conditions.

The ferritic steel to be used in the fabrication of the reactor containment boundary is tested to the requirements of ASME Section III, Subarticle NE 2300, and NB 2300 for RCPB piping which penetrates containment, to verify adequate toughness. This ensures nonbrittle behavior and minimizes the probability of a rapidly propagating fracture under the preceding established conditions.

The weld procedure qualification demonstrates that the toughness of the weld metal and heat-affected zones follow the same criteria as for the base metal.

3.1.2.52 Capability for Containment Leakage Rate Testing (Criterion 52)

Criterion

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

Design Conformance

The design of the reactor containment and any component that is part of the RCPB includes provisions for a periodic integrated leakage rate test in accordance with the requirements of Appendix J to 10CFR50.

Referenced sections are as follows:

5.2.5 Detection of Leakage through Reactor Coolant Pressure Boundary
6.2.6 Containment Leakage Testing

3.1.2.53 Provisions for Containment Testing and Inspection (Criterion 53)

Criterion

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.

Design Conformance

The reactor containment design includes provisions for periodic testing of the leaktightness of all penetrations and inserts in the RCPB. These provisions, in conjunction with the leakage monitoring system (Section 6.2.6), allow surveillance of the leaktightness conditions inside the containment.

The design of the penetrations includes provisions for periodic leakage rate tests in accordance with the requirements of Appendix J to 10CFR50.

3.1.2.54 Piping Systems Penetrating Containment (Criterion 54)

Criterion

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. Such piping 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.

Design Conformance

All piping systems penetrating the containment are provided with isolation valves. Isolation valves are discussed in Sections 7.3.1.1.2 and 6.2.4. Provisions, as described in Section 6.2.4, are made to permit leakage testing of the isolation valves.

By increased temperature, radiation, and/or drain sump flow, major leaks in the pipe are located. Isolation signals are discussed in Section 7.3.1.1.2.

3.1.2.55 Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55)

Criterion

Each line that is part of the RCPB 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 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.

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 assure 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.

Design Conformance

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The system-by-system conformance to the requirements of Criterion 55 is presented in Section 6.2.4.3.

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3.1.2.56 Primary Containment Isolation (Criterion 56)

Criterion

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.

Design Conformance

The system-by-system conformance to the requirements of Criterion 56 is presented in Section 6.2.4.3.

3.1.2.57 Closed-System Isolation Valves (Criterion 57)

Criterion

Each line that penetrates primary reactor containment and is neither part of the RCPB 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.

Design Conformance

The system-by-system conformance to the requirements of Criterion 57 is presented in Section 6.2.4.3.

3.1.2.58 Criterion 58

This criterion has not been promulgated by the NRC.

3.1.2.59 Criterion 59

This criterion has not been promulgated by the NRC.

3.1.2.60 Control of Releases of Radioactive Materials to the Environment (Criterion 60)

Criterion

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.

Design Conformance

In all cases, the design for radioactivity control is (1) on the basis of the requirements of 10CFR20, 10CFR50, and applicable regulations for normal operations and for any transient situation that might reasonably be anticipated to occur and (2) on the basis of 10CFR50.67 dosage level guidelines for potential accidents of exceedingly low probability of occurrence. All releases are expected to be reported consistent with Regulatory Guide 1.21. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

The activity level of waste-gas effluents is substantially reduced by differential holdup of noble gases from the off gas system in charcoal decay beds and subsequent release at the plant exhaust duct.

Control of liquid waste effluents is maintained by batch processing of all liquids, sampling before discharge, and controlled rate of release. Liquid effluents are monitored for radioactivity and rate of flow. Radioactive liquid waste system tankage and evaporator capacity is sufficient to handle any expected transient in the processing of liquid waste volume.

Solid wastes are prepared for offsite disposal by approved procedures. Shielded and reinforced containers which meet the applicable NRC and Department of Transportation requirements are used for the shipment of solid wastes when unshielded containers would exceed the NRC mandated dose criteria (Section 11.4).

The reference sections are as follows:

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11.2	Radioactive Liquid Waste Management Systems
11.3	Radioactive Gaseous Waste Management Systems
11.4	Radioactive Solid Waste Management System
11.5	Process and Effluent Radiological Monitoring and Sampling Systems
15	Accident Analysis
16	Technical Specifications

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3.1.2.61 Fuel Storage and Handling and Radioactivity Control (Criterion 61)

Criterion

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure 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 an RHR capability having reliability and testability that reflects the importance to safety of decay heat and other RHR, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Design Conformance

New Fuel Storage

New fuel is placed in dry storage in the new fuel storage vault which is located inside the fuel building. The storage vault within the fuel building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see evaluation against Criterion 62). The new fuel storage racks do not require any inspection and testing for nuclear safety purposes.

Spent-Fuel Handling and Storage

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Irradiated fuel is also stored in the fuel building. No spent (irradiated) fuel is stored inside the containment during plant operation. Fuel pool water is circulated through the fuel pool cooling and cleanup system to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental criticality (see evaluation against Criterion 62).

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No tests are required for nuclear safety purposes. At least one pump and heat exchanger 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 system operability.

Spent fuel is also stored in the HOLTEC HI-STORMs located on the ISFSI pad inside the protected area of the plant. The spent fuel inside the HI-STORMs is cooled through a passive natural air circulation system. Daily visual inspections by Operations are performed to assure the air inlets are not obstructed or blocked. No tests are required for nuclear safety purposes.

Radioactive Waste Systems

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

Liquid radwastes are classified, contained, and treated as high- or low-conductivity, chemical, sludges, or concentrated wastes. Processing includes filtration, ion exchange, analysis, and dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. Dry solid radwastes are packaged in shielded steel or fiber drums, cartons, or boxes. Gaseous radwastes are monitored, processed, recorded, and controlled so that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

Accessible portions of the fuel and radwaste buildings shall have sufficient shielding to maintain dose rates within the limits set forth in 10CFR20 and 10CFR50.

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

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

For further discussion, see the following sections:

- 5.4.7 Residual Heat Removal System
- 6.2 Containment Systems
- 9.1.1 New Fuel Storage
- 9.1.2 Spent Fuel Storage
- 9.1.3 Spent Fuel Pool Cooling and Cleanup System
- 9.4 Heating, Ventilating, and Air Conditioning System
- 11 Radioactive Waste Management
- 12 Radiation Protection
- 15 Accident Analysis

3.1.2.62 Prevention of Criticality in Fuel Storage and Handling (Criterion 62)

Criterion

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

Design Conformance

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new and spent fuel storage is prevented by the geometrically safe configuration of the storage rack. There is sufficient spacing between the assemblies to assure 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.

New fuel is 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 fuel 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 minimum center-to-center new fuel assembly spacing limits the effective multiplication factor (k_{eff}) of the array to not more than 0.95 for new dry fuel. k_{eff} does not exceed 0.95 if the new fuel is flooded.

Spent fuel is stored under water in the spent fuel storage pool. The racks in which spent 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 0.95 under normal and 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.

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The fuel storage racks located in the upper containment building store fuel under water during normal and refueling operations. However, only new fuel may be stored in these fuel storage racks during normal operation. The racks are designed and arranged such that spent and new fuel is maintained at a subcritical multiplication factor K_{eff} of less than 0.95 under normal and abnormal conditions.

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Refueling interlocks include circuitry which senses conditions of the refueling equipment and the control rods. These interlocks supplement 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 and is designed to minimize the possibility of mishandling or maloperation.

The use of geometrically safe configurations for new and spent fuel storage and the design of fuel handling systems precludes accidental criticality in accord with Criterion 62.

For further discussion, see the following sections:

- 9.1.1 New Fuel Storage
- 9.1.2 Spent Fuel Storage
- 9.1.4 Fuel Handling System

3.1.2.63 Monitoring Fuel and Waste Storage (Criterion 63)

Criterion

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 RHR capability and excessive radiation levels, and (2) to initiate appropriate safety actions.

Design Conformance

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the fuel pool cooling and cleanup system which could result in loss of RHR capability and excessive radiation levels is alarmed in the main control room. Alarmed conditions include low fuel pool cooling water pump discharge pressure and high/low level in the fuel pool. System temperature is also monitored continuously and alarmed in the main control room.

Area radiation levels in the radwaste and fuel buildings are monitored, and excessive levels are alarmed in the main control room so that appropriate actions can be taken.

For further discussion, see the following sections:

- 9.1 Fuel Storage and Handling
- 11.2 Radioactive Liquid Waste Management Systems
- 11.3 Radioactive Gaseous Waste Management Systems
- 11.4 Radioactive Solid Waste Management System
- 11.5 Process and Effluent Radiological Monitoring and Sampling Systems

3.1.2.64 Monitoring Radioactivity Releases (Criterion 64)

Criterion

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.

Design Conformance

Means have been provided for monitoring radioactivity releases resulting from normal and anticipated operational occurrences. The following station releases are monitored:

1. Gaseous releases from the plant exhaust duct
2. Cooling tower blowdown line liquid discharge
3. Radwaste building ventilation exhaust
4. Liquid radwaste effluent
5. Fuel building ventilation exhaust.

The containment atmosphere is continuously monitored during normal and transient unit operations, using the drywell and containment continuous airborne radioactivity monitoring system (Section 7.6.1.10). In the event of an accident, the radiation levels of the containment atmosphere are monitored by high range area monitors located in the containment. The auxiliary building exhaust is monitored by ventilation air sample particulate and gas monitors for normal and accident conditions. Radioactivity levels in the normal plant effluent discharge paths and in the environs are continually monitored during normal and accident conditions by the various radiation monitoring systems (Section 7.6) and by the offsite radiological monitoring programs.

Operational reports are submitted according to NRC requirements. These reports include specific information on the quantities of the principal radionuclides released to the environs. This is done within 60 days after each successive 6-mo operating period.

For further discussion of the means and equipment used for monitoring radioactivity releases, see the following sections:

- 5.2.5 Detection of Leakage Through the Reactor Coolant Pressure Boundary
- 7.3 Engineered Safety Features Systems
- 11 Radioactive Waste Management
- 12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Certain structures, components, and systems of the plant are considered important to safety because they perform safety actions required to prevent or mitigate the consequences of abnormal operational transients or accidents. The purpose of this section is to classify structures, components, and systems according to the importance of the safety function they perform, and to define the design requirements applicable to each classification.

3.2.1 Seismic Classification

Seismic Category I structures, systems, and components are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary (RCPB), or

2. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposure of 10CFR50.67. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

Structures, systems, and components including their foundations and supports are designed to remain functional during a safe shutdown earthquake (SSE) and are designated Seismic Category I in accordance with Regulatory Guide 1.29. Stress limits in excess of yield are allowed provided safety functions are maintained. Seismic Category I structures, systems, and components are also designed to be within the elastic limit for a vibratory motion of an operating basis earthquake (OBE).

The SSE and the OBE are described in Section 2.5. The seismic design of Seismic Category I systems and components is described in Sections 3.7 and 3.10 and of structures in Sections 3.7 and 3.8. Seismic Category I systems, components, and structures are listed in Table 3.2-1.

3.2.2 System Quality Group Classifications

System safety classifications have been determined for each component of the following:

1. Those applicable fluid systems relied upon to prevent or mitigate the consequences of accidents

or malfunctions originating within the RCPB, or to permit shutdown of the reactor and maintenance in the safe shutdown condition.

2. Other associated safety-related systems.

These safety classifications meet the intent of Regulatory Guide 1.26. The safety class terminology of ANS 52.1-1978 is used instead of the quality group terminology of Regulatory Guide 1.26. Safety Classes 1, 2, and 3 are comparable to Regulatory Guide 1.26 Quality Groups A, B, and C, respectively. A classification of other structures, systems, and components (OSSC) in ANS 52.1-1978 is referred to as nonnuclear safety (NNS) for River Bend Station.

The designation NNS predates OSSC, from the ANSI N18.2 standard for PWRs, which formed a basis for the ANS 52.1 BWR standard, and is used as a convenience since it was the designation applied throughout the design effort. NNS is comparable to Regulatory Guide 1.26 Quality Group D classification.

Components of a system are individually classified, recognizing that they may be of varying safety significance. Component supports are in the same or higher class as the equipment they support. A discussion of each safety class is included in the following section. Section 3.2.2.2 discusses exceptions to Regulatory Guide 1.26.

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The safety class assigned to each system and component is shown in Table 3.2-1.

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3.2.2.1 Safety Classes

The safety classes are defined below with the accompanying design requirements for components of each class. Examples of broad application are given; however, these general definitions are subject to interpretation or exception by specific design conditions. Where possible, reference is made to accepted industry codes and standards which define design requirements 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 outside the RCPB are considered in Sections 2.5, and 3.3 through 3.8.

3.2.2.1.1 Safety Class 1

Definition

Safety Class 1 applies to components of the RCPB whose failure could cause a loss of reactor coolant.

Design Requirements

Table 3.2-4 lists industry code requirements for Safety Class 1 mechanical components and correlates these requirements with design condition categories.

3.2.2.1.2 Safety Class 2

Definition

Safety Class 2 applies to those structures, 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 operating and accident conditions

4. Providing emergency core cooling

5. Providing and maintaining containment

6. Removing residual heat from the reactor and reactor core.

Safety Class 2 includes the following:

1. Reactor protection system (RPS)

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

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

4. The assembly of components of the reactor core which maintain core geometry including the fuel

assemblies, core support structure, and core grid plate

5. 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

6. Emergency core cooling systems (ECCS)

7. Primary containment

8. Shield building and standby gas treatment system (SGTS)

9. Post-accident containment heat removal systems

10. Containment hydrogen control system

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

12. At least one of the systems which recirculates reactor coolant to remove decay heat when the reactor is pressurized

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

14. Pipes having a nominal pipe size of 3/4 in or smaller that are connected to the RCPB.

Design Requirements

In applying industry codes to Safety Class 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 are treated separately from electrical. Table 3.2-5 lists the code requirements for Safety Class 2 mechanical systems and structures within the RCPB, and correlates these requirements with design condition categories. Structures not within the RCPB are treated in Section 3.8. The requirements for instrument tubing are listed in Table 3.2-8.

Code requirements for design of Class 1E electrical and protection systems (as defined in IEEE-279 and IEEE-308) of Safety Class 2 are shown in Table 3.2-7.

3.2.2.1.3 Safety Class 3

Definition

Safety Class 3 applies to those structures, systems, and components that are not Safety Class 1 or 2, but which provide or support safety system functions.

Safety Class 3 includes the following:

1. Cooling water systems required for the purpose of:

    a. Removal of decay heat from the reactor

    b. Emergency core cooling

    c. Post-accident heat removal from the suppression pool

    d. Providing cooling water needed for the functioning of safety-related systems.

2. Fuel supply for the onsite emergency electrical system

3. Standby equipment area cooling

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

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

6. Spent fuel pool cooling

7. Spent fuel storage racks, pool, and liner.

Design Requirements

The design requirements for Safety Class 3 mechanical categories are listed in Table 3.2-6. This table correlates these requirements with design condition categories.

Code requirements for Safety Class 3 electrical equipment are shown in Table 3.2-7.

The requirements for instrument lines are listed in Table 3.2.8.

3.2.2.1.4 Nonnuclear Safety Class (NNS)

Definition

Structures, systems, and components in the power conversion or other portions of the facility which have no direct safety function but which are connected to or influenced by safety class items are designated as NNS.

Design Requirements

The design requirements for equipment classified as NNS are specified with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it operates. Design requirements are based on applicable industry codes and standards, except where it is impractical to do so, then accepted industry and engineering practices are used.

3.2.2.2 Exceptions to Regulatory Guide 1.26 Classification

3.2.2.2.1 Steam Systems

The main steam and feedwater system components are classified as shown in Table 3.2-1. This is in accordance with the intent of Regulatory Guides 1.26 and 1.29, as stated in the letter dated April 19, 1974, from J.M. Hendrie, Deputy Director for Technical Review, Directorate of Licensing, Atomic Energy Commission, to J.A. Hinds, Manager, Safety and Licensing, General Electric Company.

3.2.2.2.2 Fuel Pool Cooling and Cleanup Subsystem

The fuel pool cooling and cleanup system is designed as two separate subsystems: the fuel pool cooling subsystem and the fuel pool purification subsystem. The fuel pool purification subsystem and its components have no safety function and are classified NNS. The fuel pool cooling subsystem is Safety Class 3 and complies with Regulatory Guides 1.13, 1.26, and 1.29. All headers and connections common with the fuel pool cooling subsystem are Safety Class 3. The fuel pool cooling and cleanup system and interconnections with other systems are discussed in Section 9.1.3.

3.2.2.2.3 Exception to Regulatory Positions C.1.e and C.2.c

At interface boundaries between components of differing classes, the component forming the interface is of the higher safety class. In practice, one safety/relief valve designed and tested in accordance with ASME III, Division 1 (i.e., a code safety valve) is considered acceptable as the boundary between the RCPB and any lower safety class or NNS line.

3.2.2.2.4 Instrument Tubing

The boundary of jurisdiction of ASME Code Section III, Class 1, 2, or 3 process piping extends to and includes the root valve. The appropriate safety class extends from the root valve to the instrument. Seismic Category I supports are employed for Safety Class 2 and 3 instrument tubing.

3.2.3 Quality Assurance

Structures, systems, and components whose safety functions require conformance to the quality assurance requirement of 10CFR50, Appendix B, are summarized in Tables 3.2-1 and 3.2-2 under the heading, Quality Assurance Category.

3.2.4 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 is summarized in Tables 3.2-2 and 3.2-3.

3.2.5 Design Limits and Loading Combinations for Seismic Category I Fluid System Components

The design conditions and functional requirements of fluid system components important to safety are reflected in the application of appropriate design limits for the most adverse (limiting) combination of loadings to which components may be subjected in service. All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following sections are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in ASME Section III.

3.2.5.1 Plant Conditions

Normal

Normal conditions are any conditions in the course of system startup, operation in the design power range, hot standby, and system shutdown other than upset, emergency, or faulted.

Upset

Upset conditions are any deviations from normal conditions anticipated to occur often enough that design should include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power, occurring simultaneously with an OBE.

Emergency

Emergency conditions are those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the RCPB. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, transients caused by one of the following: a multiple safety/relief valve blowdown of the reactor vessel; loss of reactor coolant from a break or crack which does not depressurize the reactor system but which requires the safety functions of isolation of containment, emergency core cooling and reactor shutdown, improper assembly of the core during refueling, and seizure of one recirculation pump.

Faulted

Faulted conditions are those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated events are the most drastic that must be designed against and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: the control rod drop accident, a fuel-handling accident, a main steam line break, a recirculation loop break, an SSE, or the combination of any pipe break plus an SSE plus a loss of offsite power.

3.2.5.2 Loading Combinations

Loading combinations are defined as those loadings or combinations thereof that are associated with each plant condition or specified seismic event. These loadings result from the various transients or events that are included within each plant condition and the magnitude of the specified seismic events. The design loading combinations for plant conditions identified as normal, upset, emergency, and faulted are given in Section 3.8. A discussion of the particular transients or events evaluated for each plant condition is in Section 3.9.1.

Tables 3.2-4 through 3.2-7 provide correlation of the design plant conditions with the design codes for Safety Class 1, 2, and 3 components, systems, and structures.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

All Seismic Category I structures are designed to withstand 100 mph fastest mile of sustained wind 30 ft above ground, based upon a 100-yr period of recurrence.

3.3.1.1.1 Basis for Wind Velocity Selection

The wind velocity given in Section 3.3.1.1 is based on ASCE Paper No. 6038 by H. C. S. Thom[(1)]. The wind velocity used for the River Bend Station site was obtained from Fig. 5 of this paper, which is herein reproduced as Fig. 3.3-1. This paper is used by the ANSI A58.1-1972 Code for selecting basic wind speeds for any location in the United States[(2)].

A summary of the fastest mile winds in the Lake Charles, New Orleans, and Baton Rouge areas is presented in Table 2.3-1. During the observation period from 1881 through 1978 at New Orleans, the absolute peak wind speed of 98 mph was recorded in the month of September. The absolute peak wind speed of 58 mph was recorded at Ryan Airport near Baton Rouge during a period from 1963 to 1978. This velocity is far below that recorded at New Orleans, partially due to the intervening 70 mi of land which reduces storm wind speeds considerably.

The River Bend Station is located between the 80 and 90 mph contour lines as shown in Fig. 5 of ASCE paper No. 6038[(1)]. The 80 mph wind contour line intersects the Mississippi River at a point close to where the state of Louisiana extends east of the Mississippi River. The 90 mph wind contour line intersects the Mississippi River just north of New Orleans. The River Bend Station lies between these two contours about 20 mi south of the Mississippi-Louisiana state line. The 100 mph value was chosen for conservatism.

Additional meteorological data are presented in Section 2.3.2.2.

3.3.1.1.2 Vertical Velocity Distribution and Gust Factors

The vertical velocity distribution and effect of gust factors are in accordance with the American National Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, ANSI A58.1-1972. The values of GF and GP, which are used for design of the River Bend Station structures, are given in Table 3.3-1. The site is conservatively assumed to be flat, open country which results in the use of the 1/7 power law for vertical velocity distribution.

3.3.1.2 Determination of Applied Forces

The 100 mph basic wind speed in Section 3.3.1.1 is converted to velocity pressure in accordance with ANSI A58.1-1972 using the formula:

$$q_{30} = 0.00256V_{30}^2$$

where:

q_{30} = Basic wind pressure, psf

V_{30} = Basic wind speed, mph

The effective velocity pressure of wind for buildings, q_f, and for parts and portions, q_p, at various heights above the ground has been computed in accordance with the following ANSI A58.1-1972 formulas:

$$q_f = K_Z \ GF \ q_{30}$$

$$q_p = K_Z \ GP \ q_{30}$$

where:

q_f = Effective velocity pressure for ordinary buildings and structures, psf

q_p = Effective velocity pressure for parts and portions of buildings and structures, psf

K_Z = Velocity pressure coefficient which depends upon the type of exposure and height z above ground

GF and GP = Gust factors which depend upon the response characteristics of the structure or parts and portions thereof, respectively

Table 3.3-1 gives the effective velocity pressures for buildings and for portions of buildings at various heights above ground.

The design wind pressure for each structure or part thereof is obtained by the following procedure:

1. The external pressure load on the face of a structure is obtained by multiplying the appropriate effective velocity pressure (q_f or q_p) by the applicable external pressure coefficient C_{pe} from Tables 7, 8, or 9 of Reference 2.

2. The internal pressure load on the face of a structure is obtained by multiplying the internal pressure q_M from Table 12 (exposure c) of Reference 2 by the applicable internal pressure coefficient C from Table 11 of Reference 2.

3. The design wind pressure is then obtained by algebraic combination of values from 1 and 2 to yield the most critical positive or negative values.

It must be noted that the positive pressure on the windward side, the negative pressure on the leeward side, the negative pressure on the sides of the structure parallel to the direction of wind, and the suction on the roof of the structure are considered to act simultaneously.

The design wind pressures for cylindrical structures and dome are obtained from Table 4(f) and Fig. 9 of Reference 3, respectively. The open-framed steel structures are designed to withstand the wind pressures multiplied by the appropriate shape and drag factors in accordance with Reference 3.

3.3.2 Tornado Loadings

Systems and components that directly affect the ultimate safe shutdown of the station and which are important to safety are protected from tornado effects by being located either underground or within the protection of structures designed to retain their integrity without loss of function under the tornadic loadings. The structures, systems, and components requiring tornado protection are included in Table 3.2-1.

3.3.2.1 Applicable Design Parameters

The structures referenced in Table 3.2-1 are designed using the following tornado design parameters, taken from Regulatory Guide 1.76, Table 1, as follows:

1. A maximum rotational wind velocity of 290 mph
2. A maximum translational velocity of 70 mph
3. A minimum translational velocity of 5 mph
4. An external pressure drop of 3 psi at the vortex at a rate of 2 psi/sec
5. Radius at maximum rotational speed - 150 ft
6. Postulated tornado-generated missiles (described in Section 3.5.1).

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 Transformation of Tornadic Winds

Tornado-protected structures are designed to resist a maximum wind velocity associated with a tornado of 360 mph, which is obtained by adding the tornado rotational and translational velocities.

The tornado wind velocity is converted to an equivalent pressure using the formula from Reference 2:

$$P = 0.00256V^2$$

where:

P = Equivalent pressure, psf

V = Wind velocity, mph

Although the pressure value derived from the maximum resultant velocity of 360 mph occurs only in a localized area, conservatively this pressure is applied uniformly to the entire height of the structure. Pressure and shape factors are used in accordance with the methods described in ANSI A58.1 - 1972 and ASCE paper No. 3269, respectively. The gust factor is assumed to be 1.0, since the tornadic winds are of short duration.

Tornado wind pressures and differential pressure effects are considered static loading, since the natural period of components of structures exposed to tornadic loading is short compared to its period of application.

3.3.2.2.2 Venting of Structures

A rapid depressurization of the ambient air can occur if the low pressure within the funnel of a tornado engulfs a structure. This phenomenon would generate up to a maximum external pressure drop of 3 psi between the inside and outside of the structures. Although some structures are vented to reduce the effect of internal pressure to a value less than 3 psi, conservatively all the Seismic Category I structures are designed to withstand an internal pressure which varies from 0 to 3 psi at a rate of 2 psi/sec, remaining at 3 psi for 2 sec, and returning to 0 psi at a rate of 2 psi/sec. The differential pressure effects are not considered for overturning the structure because they are self-balancing within the structure.

3.3.2.2.3 Missile Impact Loads

The structures that are identified as being able to satisfy the missile criteria can withstand the missiles listed in Table 3.5-24 hitting the exposed walls or roof. The missile loads are considered impactive dynamic loads. The procedures used in designing the structures to withstand missile impact are outlined in Section 3.5.3.

The effects of tornadic wind pressure, missile load, and differential pressure are applied simultaneously with applicable dead and live loads, excluding earthquake forces, as shown in Section 3.3.2.2.4.

3.3.2.2.4 Tornado Load Combinations

Tornado-generated load combinations for all permanently enclosed structures that have to withstand the design basis tornado are listed as follows:

1. $W_t = W_W + 1/2\ W_p + W_m$

2. $W_t = W_w + W_m$

3. $W_t = W_p$

4. $W_t = W_w$

5. $W_t = W_m$

6. $W_t = W_w + 0.5\ W_p$

where:

W_t = Total tornado load

W_w = Tornado wind load

W_p = Tornado-generated differential pressure

W_m = Tornado-generated missile load

The most adverse of these combinations is used for designing each component of a structure (as applicable) in combination with other appropriate loads as specified in Sections 3.8.3, 3.8.4, and 3.8.5.

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

The arrangement of station structures is shown in Figure 1.2-2. Seismic Category I structures are arranged so that the failure of buildings not designed for tornado loads does not damage Seismic Category I structures.

Structures with functions that do not require tornado load design are either located so that structural failure does not affect the ability of structures designed for tornado loads to perform their intended design function, or they are designed not to collapse under tornado wind load.

Objects with a potential to become significant missiles, such as steel columns, beams, bracing, and purlins in the turbine building located within close proximity of Seismic Category I structures, are designed to withstand tornadic forces and remain in place. Objects such as metal siding, roofing, roof decks, and parapets may blow off during a tornadic event. These objects are not capable of producing significant missiles (i.e., missiles capable of impactive dynamic loads greater than the postulated tornado-generated missiles described in Section 3.5.1.4). Since the components of the Seismic Category I structures are designed to withstand the postulated tornado-generated missile loads and tornadic wind loads simultaneously, the failure of the components that may blow off is not considered to have a detrimental effect on these structures.

Other structures that are not designed for tornado loads are either lower than or located away from tornado-designed structures.

References - 3.3

1. Thom, H.C.S. New Distribution of Extreme Winds in the United States, ASCE Paper No. 6038, Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, Vol. 97, No. ST7, p 1787-1801, 1968.

2. Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, ANSI A58.1-1972.

3. Wind Forces on Structures, ASCE Paper No. 3269, Final Report of the Task Committee on Wind Forces, reprinted from transactions of the American Society of Civil Engineers, Vol 126, Part II, p 1124-1167, 1961.

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

This section discusses the flood protection measures provided for Seismic Category I structures, systems, and components.

3.4.1.1.1 Identification of Safety-Related Systems

Flood protection of safety-related systems and components as identified in Table 3.2-1 is provided for all postulated flood levels and conditions as described in Section 2.4.

3.4.1.1.2 Description of Structures

Structures which house the safety-related equipment and offer flood protection to this equipment are identified on Fig. 1.2-2.

A description of these structures is provided in Sections 3.8.2, 3.8.4, and 3.8.5. Exterior or access openings and penetrations that are below the postulated design flood level are identified in Table 3.4-1.

The groundwater level during normal plant operation and during the Design Basis Flood Level (DBFL) is tabulated for all Seismic Category I structures in Table 3.4-1. The exterior walls and the foundation mat of these structures are designed to withstand the hydrostatic and buoyant forces resulting from the flooding conditions.

3.4.1.1.3 Means of Providing Flood Protection

Internal and external flood protection is provided for the safety-related systems and components identified in Table 3.2-1, for all postulated flood levels and conditions described in Section 2.4, by one of the following methods:

1. Housing them in Seismic Category I structures designed to withstand the flood loads

2. Locating them above the maximum postulated flood level

3. Locating them in watertight cubicles (of a structure) designed to withstand external and/or internal flood loads.

When exposed to earth, the structural components of Seismic Category I structures are designed using:

1. Wall thicknesses below flood levels of not less than 2 ft

2. Waterstops at construction joints below flood level.

Waterproofing of foundations and exterior walls of Seismic Category I structures below grade is accomplished principally by the use of waterstops at expansion and construction joints. The waterstop is synthetic rubber. Typical details for waterstops are shown on Fig. 3.8-20.

All the penetrations through the exterior walls of the Seismic Category I structures below the DBFL are designed to withstand the hydrostatic head of water and are made watertight using air, water, and fire seals and waterstops around them as applicable.

The access openings to the structures housing safety-related components are either located above the DBFL or are required to be closed to prevent any adverse effect from flooding of the structures. If local seepage occurs through the walls, it is controlled by sumps and sump pumps. The operation of the plant, therefore, is not affected by flood conditions.

3.4.1.2 Permanent Dewatering System

No permanent plant dewatering system is provided.

3.4.2 Analytical and Test Procedures

The lateral hydrostatic pressure on the structures due to the Probable Maximum Flood water level, as well as groundwater and soil pressures, is determined as shown on Fig. 2.5-79.

The Seismic Category I structures are designed and analyzed for the maximum hydrostatic head and the buoyant forces due to the Probable Maximum Flood, in accordance with the loads and load combinations indicated in Section 3.8.4. A safety factor of 1.1 is used in designing these structures against flotation.

The most critical postulated flood condition results in standing water at about el 96.0 ft msl, which would be caused by an occurrence of the PMP in the immediate plant area prior to completion of excavation backfilling operations. The dynamic effect resulting from wave forces at this low level of ponding (1 to 1.5 ft at the plant buildings) is considered negligible. Therefore, the hydrodynamic loads due to floods are not considered in designing the Seismic Category I structures.

3.5 MISSILE PROTECTION

3.5.1 Missile Selection and Description

The following criteria have been adopted to assess the plant's integrated design to afford protection from generated missiles of the type postulated in this section:

1. No loss of containment function

2. No direct loss of reactor coolant

3. No loss of function to systems required to shut down the reactor and maintain it in a cold shutdown condition, or mitigate the consequences of the missile damage such that:

    a. No equipment is allowed to be damaged in one safety-related division, e.g., Division 1, from internally generated missiles originated from another safety-related division, e.g., Division 2.

    b. Missiles generated from nonsafety-related equipment do not damage any safe shutdown equipment.

    c. Offsite power is not available for shutdown of the plant.

    The systems required to be protected are:

    a. Reactor coolant pressure boundary (RCPB)

    b. Emergency core cooling systems (ECCS)

    c. Standby service water (SSW) system

    d. Ultimate heat sink (UHS)

    e. Core cooling systems

    f. Standby diesel generator system

    g. CRD hydraulic (scram section) system

    h. Fuel pool cooling system

    i. Remote shutdown panel

    j. Reactor protection system (RPS)

    k. All containment isolation valves

    l. HVAC systems required during operation of the previous items

    m. Electrical systems and control systems and instruments required for operation of safety-related equipment, components, and systems.

n. Safety-related portion of the reactor plant component cooling water system (RPCCW).

4. No offsite exposure exceeding the limits of 10CFR50.67. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

5. No loss of integrity of the spent fuel pool.

6. Although Class 1E sensors to the (RPS) are located on the turbine control valve (TCV) and stop valve, failure of these sensors does not prevent the reactor from being safely shut down since other RPS sensors (high-pressure scram or high-flux scram) located in safety-related buildings provide sufficient backup. Therefore, these sensors are not analyzed for missile hazards inside the turbine building.

Essential structures, systems, and components are protected from the effects of internal missiles by one or more of the following practices:

1. Locating the system or component in an individual missile-proof structure

2. Physically separating redundant systems or components of the system from the missile trajectory path

3. Providing localized protective shields or barriers for systems and components

4. Designing the particular structure or local protective shield/barrier to withstand the impact of the most damaging missile

5. Providing design features on the potential missile source to minimize the probability of missile generation

6. Orienting the potential missile source in such a manner as to prevent unacceptable consequences due to missile generation.

3.5.1.1 Internally Generated Missiles

Missile protection is provided within the plant structures that are important to safety inside and outside the containment for two general sources of postulated missiles:

1. Rotating component failure

2. Pressurized component failure.

The basic approach is to ensure design adequacy of the equipment components against the generation of missiles, rather than to allow missile formation and try to contain its effects.

3.5.1.1.1 Rotating Component Failure Missiles

Castastrophic failure of rotating equipment leading to the generation of missiles is not considered credible. Massive and rapid failure of these components is not credible because of the material characteristics, inspections, quality control during fabrication, erection and operation, conservative design, and prudent operation as applied to the particular component.

Various types of rotating equipment were analyzed, (e.g., pumps, fans, and turbines) as to their potential for becoming missile sources. The following was concluded:

1. The most substantial piece of NSSS rotating equipment is the recirculation pump and motor. This potential missile source is discussed in detail in Reference 5. It is concluded in Reference 5 that destructive pump overspeed cannot result in the generation of missiles.

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2. Large, massive rotating components, such as the various ECCS pumps and motors, do not have sufficient energy to move their masses through the housings in which they are contained. The RCIC turbine similarly is concluded not to generate missiles upon failure. An overspeed tripping device and automatic governing controls ensure that the RCIC turbine does not reach runaway speed where possible component failure could take place.

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3. Both axial and centrifugal fans were investigated to determine their potential as missile sources. The rotating components most likely to become missiles are the fan blades and the rotor impellers. A stress analysis was performed to determine the safety factors against the failure of these components. The results are shown in Table 3.5-26. For centrifugal fans, the safety factors were found to be about 14-16. For axial fans, the safety factors were found to range from 3 to 36. In addition, if a blade failure were to occur, the failed blade would be moving in a direction tangential to the housing. This would cause the blade to rotate upon impact due to the oblique angle and blade orientation, and then reimpact the housing. This would substantially reduce the energy to perforate the housing, with the blade being effectively contained. Due to the high factors of safety involved, and the mode of impact of the blade against the housing should a blade be thrown, it can be concluded that fans are not credible missile sources.

3.5.1.1.2 Pressurized Component Failure Missiles

The bases for the selection of missiles generated by postulated failures of pressurized components are:

1. Thermometers or other detectors installed on piping or in wells are evaluated. The analysis of the thermowell shows that thermowell ejection is very improbable because of its highly conservative design. The analysis shows that safety factors incorporated into the design range from 10 to 107 (Table 3.5-27). Consequently, thermowells are not considered as potential missile sources.

2. Valves of ANSI 900 psig rating and above, constructed in accordance with the ASME Code, Section III, are pressure seal bonnet-type valves. Valve bonnets for pressure seal bonnet-type valves were analyzed to evaluate the potential for the bonnets to become missiles. Bonnets could become missiles through failure of the bonnet retaining ring, failure of the valve body at the retaining ring interface, or failure of the bonnet critical thickness. All three of these items were investigated by analyzing a representative group of pressure seal bonnet-type valves and evaluating the safety factors against these types of failures (safety factors are based on the ultimate strengths of the materials). Results (Table 3.5-28) show the following:

    a. Safety factors against retaining ring failure range from 6 to 15 (for shear), and from 8 to 16 (for bearing).

    b. Safety factors against failure of the valve body at the retaining ring interface (for shear) ranged from 21 to 32.

    c. Safety factors against failure of the bonnet critical thickness (for shear) ranged from 10 to 22.

    Because of the high factors of safety involved against these types of failures, bonnets of pressure seal-type valves are not considered potential missile sources.

3. Most valves of ANSI 600 psig rating and below are valves with bolted bonnets. Valve bonnets for bolted bonnet-type valves were analyzed to evaluate the potential for the bonnets to become missiles. Bonnets could become missiles through failure of the bonnet bolts or failure of the bonnet critical thickness. Both of these items were investigated by analyzing a representative group of bolted bonnet-type valves and evaluating the safety factors against these types of failures. As before, safety factors are based on the ultimate strengths of the materials. Results (Table 3.5-28) show the following:

    a. Safety factors against bonnet bolt failure range from 4 to 9.

    b. Safety factors against failure of bonnet critical thickness range from 30 to 54.

    Due to the high factors of safety involved against these types of failures, and the low historical incidence of complete severance failure of the valve bonnets, bolted-type valve bonnets need not be considered as potential missile sources.

4. Valve stems were analyzed by assuming that a failure of the minimum stem thickness would allow the stem to become a missile. Analysis shows that safety factors (based on ultimate material strength) against this type of failure range from 3 to 11 (Table 3.5-28).

The analysis did not take into account backseats and stem threads, which would further prevent ejection, nor did it take into account valve operators (on air- and motor-operated valves), which would effectively restrain the valve stem. Based on this conservative approach, valve stems are not considered as potential missile sources.

5. Nuts, bolts, nut-and-bolt combinations, and nut-and-stud combinations which were part of bolted bonnet-type valves were considered to be the major concern for this type of missile. Analysis of bolted bonnet-type valves has already eliminated bolt missiles from this type of source. All other nut-and-bolt combinations have a minimal amount of stored energy and are not considered further as potential missile sources.

3.5.1.1.3 Gravitational Missiles

Seismic Category I systems, components, and structures are not potential gravitational missile sources.

Non-seismic items and systems in Seismic Category I buildings are classified as follows:

1. General

   All suspended nonsafety-related items such as piping, non-Class 1E conduit, instrument tubing structures, and HVAC ducting which could adversely affect safety-related equipment in the event of failure are supported to prevent collapse during an SSE.

2. Cable Tray

   All cable trays for both Class 1E and non-Class 1E circuits are seismically supported whether or not a hazard potential is evident.

3. Equipment for Maintenance

   All other equipment, such as hoists, which could adversely affect safety-related equipment in the event of failure that is required during maintenance is either removed during operation or is restrained to prevent it from becoming a missile.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Details for internally generated missiles inside the containment are given in Section 3.5.1.1.

3.5.1.3 Turbine Missiles

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3.5.1.3.1 Turbine Missile Analysis Statement

The maximum attainable speed for the River Bend Unit is 215-218% at full admission, and 218-222% at partial arc admission. The minimum speed capability of the monoblock rotors, assuming all buckets remain attached to the rotor, is 219-225%. This range is based on the minimum

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specification tensile strength value. Using a more typical tensile strength value the speed capability is increased to 230-235%. Therefore, based on a conservative evaluation, the speed capabiltiy of the monoblock rotor is considerably higher than the previous shrunk-on design and is in excess of the maximum speed capability of the River Bend Unit.

A complete failure of the control system is required to achieve the above overspeed. The annual probability of this complete control system failure is in the range of 10^{-8}. Since the stress levels are very low for the monoblock rotors when compared to the original shrunk-on design, and the keyway stress corrosion cracking mechanism is not present in the monoblock rotors, the probability of turbine missiles being generated is not present[17]. The annual probability of the digital TCPS failure to trip on overspeed is 5.41 x 10^{-9}. Thus this conclusion remains valid.

3.5.1.3.5 Turbine Overspeed Protection

A description of turbine overspeed protection systems is presented in Section 10.2. Component reliability and testing procedures are described in Section 10.2.

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3.5.1.3.7 Turbine Characteristics

The turbine characteristics are described in Sections 10.1 and 10.2. Steam environment parameters are discussed in Sections 1.3 and 10.1 and in Table 5.2-4. Operation of the turbine-generator unit under normal and transient conditions is described in Section 10.2.

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3.5.1.4 Missiles Generated by Natural Phenomena

It is assumed that a tornado could generate missiles as listed in Table 3.5-24. The minimum thickness of reinforced concrete barriers that are designed to provide protection against missiles generated by natural phenomena is 24 in. The strength of concrete used in the construction of these barriers is 3,000 psi at 28 days as a minimum. The corresponding curing time conforms to ACI 301, Chapter 12, as supplemented in Section 3.8.4.6.

3.5.1.5 Missiles Generated by Events Near the Site

No missiles of any significance are expected to be generated by events near the site, due to the distances from nearby transportation routes. The nearest transportation route is 2 mi for waterbound traffic, 1/2 mi for rail traffic and 5,000 ft for road traffic. Any explosion on one of these routes would not generate significant missiles at the plant site. For a detailed description, see Section 2.2.3.1.

3.5.1.6 Aircraft Hazards

There are no commercial airports located in the vicinity of the plant. The nearest such facility is Ryan Airport at Baton Rouge, Louisiana, about 19 mi southeast of the site[9]. A review of the flight patterns at this airport has shown that there are no published approaches near the station location[10]. Aircraft approaches to the airport are shown in Fig. 3.5-6. Air traffic at Ryan Airport does not represent a plant hazard. Fig. 3.5-6 also shows the locations of three low-altitude (below 18,000 ft) airways in the site vicinity[9]. The width of a low-altitude (Victor) airway is about 4 nautical mi (4.6 statute mi) either side of the airway centerliney[11]. The centerline of Victor airway V71 passes 2.5 mi east of the plant, while the centerlines of V222 and V114N pass 7 mi northwest of the site and 8.5 mi northeast of the site, respectively. The site is more than 2 mi beyond the edge of airways V222 and V114N, and it is unlikely that the use of these air routes would present any aircraft hazard to the plant[12]. A review of the accident probability on V71 revealed that air traffic would not represent a plant hazard.

An analysis of air traffic on V71 was performed and the plant hazard probability computed. The following formula from Reference 15 was applied:

$$P = C \times N \times A/w$$

where:

P = Probability per year of an aircraft crashing into the plant

C = In-flight crash rate per mile for aircraft using the airway

N = Number of flights per year along the airway

w = Width of the airway (plus twice the distance from the airway edge to the site when the site is outside the airway), mi

The formula variables were quantified in the following manner:

C = 3 x 10-9 [13]

A = 0.02 sq mi based on an estimate of 0.01 sq mi per unit effective plant area [13]

N = 6,935 flights per year [10]

w = 9.2 statute mi [11]

The resultant value of P is 4.5×10^{-8}/yr. A value of $P \leq 10^{-7}$/yr is considered reasonable from a safety standpoint and precludes the need for plant design consideration of aircraft hazards[13].

There are two high-altitude (above 18,000 ft) jet routes near the site, as shown in Fig. 3.5-6[15]. Route J22 passes about 7 mi northwest of the site, and Route J58 passes about 13.5 mi southwest of the site. Due to the distance from the site of these air corridors, the aircraft hazard at the plant is insignificant[13].

There are no airfields within 5 mi of the site. There are two small airfields within 10 mi of the plant. The Jackson Airport is 8.1 mi northeast of the site, and the False River Air Park at New Roads is about 10 mi west-southwest of the site (Fig. 3.5-6). The movements at these locations are about 1,000/yr and 4,000/yr, respectively, which do not present a significant aircraft hazard at the plant site[12,16]. There are no military installations or any military airspace usage that might present a hazard to the site.

In summary, it has been determined that there is no significant aircraft hazard at the River Bend Station site.

3.5.2 Structures, Systems, and Components to be Protected from Missiles

3.5.2.1 General

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The systems and components that are required for a safe shutdown of the reactor and maintenance of a safe shutdown condition are identified in Section 3.5.1. The missiles to be considered in this section are the tornado-generated missiles. All other equipment generated missiles have been rendered noncredible as explained in Section 3.5.1.

3.5.2.2 Missile Barriers

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The exterior walls and roofs of the Seismic Category I structures act also as protective barriers to withstand the effects of missiles generated by natural phenomena. The thickness of these protective barriers meets or exceeds the minimum thickness requirements of NRC-SRP, NUREG-0800, Section 3.5.3, Table 1, Revision 1, dated July 1981. These structures are listed in Table 3.2-1 and are shown in Fig. 1.2-2.

3.5.3 Barrier Design Procedures

Missile barriers are designed to defeat the missiles described in Section 3.5.1. Defeat of the missile is achieved if the missile is stopped with no generation of secondary missiles and structural collapse of the barrier is precluded.

Local response of steel barriers is evaluated by using the Ballistic Research Laboratory Formula in Gwaltney(8). The thickness of steel barriers to prevent perforation is obtained by multiplying 1.25 by the thickness for threshhold perforation (P) as determined by the Ballistic Research Laboratory Formula. The procedure used to evaluate the local response of concrete barriers to missile impact with no scabbing is based on Appendix B of SWECO 7703(7). The thickness of concrete barriers provided conform to the minimum acceptable barrier thickness requirements of Table 1 of the Standard Review Plan (SRP) 3.5.3, Revision 1, dated July 1981.

The overall structural response of the concrete barriers to missile impact is evaluated using methods presented in Appendix C of SWECO 7703(7). Using these methods, the structural design of the barrier is controlled by the ductility factor as described herein.

If the barrier is required to carry loads during and after missile impact, the maximum allowable ductility is limited to a factor of 10. In particular:

1. For beam-column members where the compressive load is equal to or less than one-third of that which would produce balanced conditions (i.e., P_b or $0.1f_c A_g$ whichever is smaller), the allowable ductility is 10.

where:

P_b = axial load capacity at simultaneous assumed ultimate strain of concrete and yielding of tension steel

f_c = specified compressive strength of concrete, psi.

A_g = gross area of section, sq in

2. For beam-column members where the design is controlled by compression, the allowable ductility is 1.3.

3. For members which are between the cases of items 1 and 2, the ductility ratio should be taken as decreasing linearly from 10 to 1.3.

4. Where shear controls the design, the permissible ductility ratios are as follows:

   a. When shear is carried by concrete along, the allowable ductility is ≤ 1.0.

   b. When shear is carried by a combination of concrete and stirrups (or bent bars), the allowable ductility is ≤ 1.3.

The overall structural response of the steel barriers to missile impact is evaluated in accordance with the following:

1. When flexural compression or shear governs, the allowable ductility is ≤ 10.

2. For columns with slenderness ratio (1/r):

   a. Equal to or less than 20, the allowable ductility is ≤ 1.3.

   b. Greater than 20, the allowable ductility is ≤ 1.0.

   where:

   l = Effective length of the member.

   r = Least radius of gyration.

3. When the members are subjected to tension, the ductility ratio (u) is given by:

$$u = 0.5\frac{\varepsilon_u}{\varepsilon_y}$$

where:

ε_u = Ultimate strain

ε_y = Yield strain.

The above ductility factors are in accordance with the allowable ductility ratio and criteria outlined in Appendix C, ACI 349, as modified by NRC Regulatory Guide 1.142, Revision 1, dated October 1981.

If a concrete barrier is not required to carry other loads during and after impact, the maximum allowable ductility is limited to correspond to a rebar elongation of 5 percent. Similarly, for steel barriers not required to carry other loads, the maximum allowable ductility is also limited to correspond to an elongation of 5 percent. There are no openings in the walls or roofs of Seismic Category I structures which could allow a tornado missile to pass through and hit any safety-related targets.

References - 3.5

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1. Deleted

2. Deleted

3. Deleted

4. Deleted

5. General Electric Report, Analysis of Recirculation Pump under Accident Conditions. Submitted to the NRC (Attention: Mr. D. R. Vassallo, Assistant Director for Light Water Reactors) by GE vide their Letter No. MFN-104-79 dated March 30, 1979.

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7. Stone & Webster Engineering Corporation Topical Report SWECO-7703, Missile - Barrier Interaction, September 1977.

8. Gwaltney, R.C. Missile Generation and Protection in Light Water-Cooled Power Reactor Plants. ORNL-NSIC-22, September 1968.

9. Federal Aviation Administration. Houston Sectional Aeronautical Chart, 25th edition. U.S. Department of Commerce, Washington, DC, February 21, 1980.

10. Letter from Mr. Clair Billington, Federal Aviation Administration, Air Space Analysis Branch, Fort Worth, TX, to Mr. F. P. Maiuri, Lead Environmental Engineer, Stone & Webster Engineering Corporation, November 1979.

11. 14CFR71, Designation of Federal Airways, Area Low Routes, Controlled Airspace, and Reporting Points. Federal Register, January 1977.

12. Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Regulatory Guide 1.70, Section 3.5.1.6. Nuclear Regulatory Commission, Washington, DC, November 1978.

13. Standard Review Plan, NUREG-75/087, Section 3.5.1.6 Nuclear Regulatory Commission, Washington, DC, November 1975.

14. Telephone conversation between Mr. Jacques Read, Nuclear Regulatory Commission, Accident Analysis Branch, Washington, DC, and Mr. D.R. Jaquette, Stone & Webster Engineering Corporation, Boston, MA, October 20, 1977.

15. Enroute High Altitude - U.S. Chart. U.S. Department of Commerce, Washington, DC, November 2, 1978.

16. Telephone conversation between Mr. Charles David, State Aviation Department, Ryan Airport, Baton Rouge, LA, and Mr. E.T. Blake, Stone & Webster Engineering Corporation, September 26, 1979.

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17. Letter from S.J. Coluccio (GE) to C.P. McNemar (GSU) dated February 18, 1994 transmitting River Bend Turbine S/N 170X662 Missile Analysis Statement. RBC-45208

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

Two inputs to Section 3.6 are provided. Section 3.6A is applicable to the SWEC scope of supply. Section 3.6B is applicable to the GE scope of supply.

With regard to design for protection against dynamic effects associated with the postulated rupture of piping, the respective GE and SWEC responsibilities are as follows:

1. GE's responsibility includes the reactor recirculation piping only. For the recirculation piping GE determines the postulated break locations and the blowdown reactions resulting from each postulated break, and provides the restraints to restrict pipe whip in the event that a postulated break occurs.

2. SWEC's responsibilities include the balance of piping inside and outside containment. For all piping, except the recirculation piping, SWEC determines the break locations and the resulting blowdown reactions, and provides the required pipe whip restraints and guard pipes. In addition, for all piping <u>including</u> the recirculation piping SWEC analyzes the jet impingement effects resulting from each postulated break.

This section describes the design for protection against postulated piping failures both inside and outside containment including all high and moderate energy piping systems. This section includes or references plant layout drawings, system piping and arrangement drawings, and a description of how the plant structure systems and components conform to related design criteria and bases and demonstrates the ability to perform a safe shutdown after a postulated piping failure of a high or moderate energy system. Breaks and cracks are extremely unlikely due to the conservative design required by codes, standards, and other regulatory criteria as part of the defense-in-depth approach to nuclear safety.

3.6A PROTECTION AGAINST EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING (SWEC SCOPE OF SUPPLY)

3.6.1A Postulated Piping Failures in Fluid Systems Inside and Outside the Containment

3.6.1.1A Design Bases

1. Criteria

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The pipe failure protection conforms to Appendix A of 10CFR50, General Design Criterion 4, Environmental and Missile Design Bases. The overall design for this protection is in compliance with NRC Regulatory Guide 1.46 and NRC Branch Technical Positions (BTP) APCSB 3-1 and MEB 3-1, the implementation of which is discussed herein.

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2. Objectives

Protection against pipe failure effects is provided to fulfill the following objectives:

a. To assure that the reactor can be shut down safely and can be maintained in a safe cold shutdown condition or that the consequences of a loss-of-coolant accident (LOCA) can be mitigated.

b. To assure that containment integrity is maintained.

c. To assure that a pipe break which is not a loss of reactor coolant does not cause loss of reactor coolant.

d. To assure that the radiological doses resulting from a postulated piping failure remain below the limits of 10CFR50.67. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

e. To assure that the consequences of the postulated piping failure can be mitigated considering any single active component failure except as noted below. The single active failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure such as unit trip and loss of offsite power.

Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual purpose, moderate-energy essential system (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single failures of components in the other train or trains of that system are not assumed since:

(1) The system is designed to Seismic Category I standards.

(2) Power is provided from both offsite and onsite sources.

(3) Construction, operation, and inspection are done in accordance with quality assurance, testing, and in-service requirements apppropriate for nuclear safety systems.

Examples of systems that qualify as moderate-energy, dual-purpose, essential systems are standby service water (SSW) systems and residual heat removal (RHR) systems.

3. Assumptions

The following assumptions are used to determine the protection requirements:

a. Pipe breaks or cracks are postulated to occur during normal plant operation (i.e., reactor startup, operation at power, hot standby, or reactor cooldown to a cold shutdown).

b. Only high-energy piping as defined in Section 3.6.2.1.1A is capable of producing breaks. Moderate-energy piping as defined in Section 3.6.2.1.2A is capable of producing only cracks.

c. Pipe breaks are evaluated for the effects of pipe whip, jet impingement, flooding, room pressurization, and other environmental effects such as temperature.

d. Pipe cracks are evaluated for wetting from spray, flooding, and other environmental effects.

e. Each longitudinal or circumferential break in high-energy fluid system piping, or leakage crack in moderate-energy fluid system piping, is considered separately as a single postulated initial event occurring during normal plant conditions.

f. Pipe failures (breaks or cracks) inside the containment are not postulated concurrently with pipe failures outside the containment.

g. Offsite power is assumed to be unavailable when a trip of the turbine generator system or reactor protection system (RPS) is a direct consequence of the postulated piping failure, unless it is more conservative to assume that offsite power is available (e.g., a feedwater line break with offsite power available leads to a larger inventory of water for flooding considerations).

h. All available systems, including those initiated by operator actions, are employed to mitigate the consequences of a postulated piping failure to the extent below:

   The postulated failure and its direct consequences are taken into account when judging the availability of systems. The feasibility of carrying out operator actions is judged on the basis of ample time and adequate access to equipment for the proposed actions.

i. Neither a whipping pipe nor the jet discharging from it is considered capable of affecting the functional capability of a piping system by impacting the system, when both the nominal pipe diameter and wall thickness of the impacted piping system are greater than or equal to those of the ruptured pipe.

j. Pipe whip is assumed to occur in the plane defined by the piping geometry and cause movement in the direction of the jet reaction, unless shown to be otherwise by analysis.

k. The fluid internal energy associated with the pipe break reaction takes into account any line restrictions (e.g., flow limiter) between the pressure source and break location and absence of energy reservoirs, as applicable.

l. Protection of the reactor pressure vessel (RPV) from the surface impact effects of pipe rupture is not considered due to its relative thickness and location relative to piping systems.

m. Initial pipe break events are not assumed to occur in pump and valve bodies because of their greater wall thicknesses.

4. Approach

To comply with the previously defined objectives, systems, components, and equipment required to safely shut down the plant and mitigate the consequences of postulated piping failures (hereinafter called essential systems components and equipment) are reviewed to determine their susceptibility to the pipe failure effects.

Piping system break and crack locations are determined in accordance with Section 3.6.2A. Fig. 3.6A-12 through 3.6A-33 show the high-energy pipe break locations, safety-related equipment, and system locations.

Those essential items which are subject to the consequences of pipe failures are summarized in Tables 3.6A-25 through 3.6A-51. The type of hazard (i.e., whipping, jet impingement, spraying, and flooding) and the type of protection provided are also shown. This summary was based on the detailed failure mode analysis discussed in Section 3.6.1.3A and Appendix 3C.

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Postulation of arbitrary intermediate breaks is no loner required. Relaxation in arbitrary intermediate breaks is provided in NRC Generic Letter 87-11. Previously postulated in arbitrary intermediate breaks and their effects may be deleted.

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3.6.1.2A Description of Piping Failures

A list of essential systems, components, and equipment, or portions thereof are provided in Tables 3.6A-23 and 3.6A-24. A list of high energy lines as discussed in

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Section 3.6.2.1.1A are given in Tables 3.6A-21 and 3.6A-22 for inside and outside the containment respectively.

Moderate-energy piping defined in Section 3.6.2.2A is not listed.

Composite drawings, Fig. 3.6A-34 through 3.6A-49, show the routing of high-energy piping in relation to compartments inside the containment and the auxiliary building. Nearby essential items are discussed in Appendix 3C.

Pressure response analyses are performed for the subcompartments containing high-energy piping. For a detailed discussion of the line breaks selected, vent paths, room volumes, analytical methods, pressure results, etc, refer to Section 6.2.1.2 for containment subcompartments and Appendix 3B for subcompartments located outside the containment.

The effects of pipe whip, jet impingement, spraying, and flooding on essential systems, components, and equipment are discussed in Appendix 3C.

There are no high-energy lines anywhere near the main control room. As such, there are no effects upon the habitability of the main control room by pipe break either from pipe whip, jet impingement, or transport of steam. Further discussion on main control room habitability systems is provided in Section 6.4.

3.6.1.3A Safety Evaluation

3.6.1.3.1A General

An analysis of pipe failures postulated in the design is performed to identify those safety-related systems, components, and equipment that provide protective actions required to mitigate the consequences of the postulated accidents including postulated pipe failure.

By means of design features such as separation, barriers, and pipe whip restraints, all of which are discussed hereafter, the effects of breaks and cracks do not damage essential items to an extent that would impair the integrity or operability of essential systems and components.

Specific design features used for protecting the essential systems, components, and equipment and the ability of specific safety-related systems to withstand a single active failure concurrent with a postulated event are discussed in Appendix 3C.

When the pipe layout and plant arrangement drawings show that the effects of postulated breaks/cracks are isolated, physically remote, or restrained by plant design features from essential systems or components, no further evaluation is performed.

3.6.1.3.2A Protection Methods

1. General

   The effects associated with a particular break/crack must be mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the specific measures for protection against actual pipe movement and other associated consequences of postulated failures.

   a. Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, jet impingement shields, barriers, compartments, and physical separation of piping, equipment, and instrumentation.

   b. The specific method chosen depends on physical limitations such as accessibility, maintenance, and proximity to other essential systems, components, and equipment.

   c. Protective measures utilized to meet these requirements consider access requirements for conducting the inservice examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI, Inservice Inspection of Nuclear Power Plant Components.

2. Separation and Enclosure

   Separation is achieved to the extent practicable by plant physical layouts that provide sufficient distances so that essential systems and components are separated from other fluid systems.

   Fluid systems which are not physically separated from essential systems and components are enclosed, when practical, within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or compartments designed to withstand the effects of postulated piping failures in nearby fluid systems.

3. Barriers and Shields

   In many cases protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations. Where adequate protection did not already exist due to separation, additional barriers, deflectors, or shields are provided as necessary.

   If required, jet impingement shields may consist of two types:

   a. Target Shield

      The target shield is a flat plate at an angle to the jet or a pair of plates assembled in a wedge shape (Fig. 3.6A-50). This shape provides a shape factor which reduces the effective jet intensity on the shield. The shield geometry, size, and location are such that the target does not directly intercept any of the postulated jets.

   b. Source Shield

      The source shield is a removable casing which forms an annulus about the process pipe in the region of a postulated longitudinal pipe break location (Fig. 3.6A-51). When the pipe ruptures, the source shield redirects the escaping fluid jet parallel to the piping axis.

4. Piping Restraint Protection

   Measures for protection against pipe whipping as a result of high-energy pipe breaks are not provided if:

   a. The piping is physically separated (or isolated) from any essential safety-related structure, system, or component required to place the plant in a safe shutdown condition, or is restrained from whipping by plant design features such as concrete encasement.

b. Following a single break, unrestrained pipe movement of either end of the rupture pipe could not damage, to an unacceptable level, any structure, system, or component required to place the plant in a safe shutdown condition.

c. The energy associated with the whipping pipe is demonstrated to be insufficient to impair, to an unacceptable level, the safety function of any structure, system, or component required to place the plant in a safe shutdown condition.

In cases where the above criteria are not met pipe restraints are provided. The design criteria for restraints are given in Section 3.6.2.3.1A.

3.6.1.3.3A Specific Protection Measures

1. Nonessential systems and system components are not required for the safe shutdown of the reactor, nor are they required for the limitation of the offsite release in the event of a pipe rupture. However, while none of this equipment is needed during or following a pipe break event, pipe whip protection is considered where a nonessential system or component failure could initiate or escalate a pipe break event in an essential system or component, or another nonessential system whose failure could affect an essential system.

2. The pressure, water level, and flow sensor instrumentation for those essential systems which are required to function during or after accident conditions are protected from pipe rupture effects.

3. High-energy fluid system piping restraints and protective measures are designed in such a way that a postulated break in one pipe could not, in turn, lead to rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break.

4. For any postulated pipe rupture, the structural and leaktight integrity of the containment structure is maintained.

5. High-energy piping which penetrates both the drywell and the containment is provided with guard pipes designed in accordance with Section 3.8.2.

6. To maintain the ability to insert the control rods in the event of a pipe break, the control rod drive (CRD) withdraw lines are protected from the dynamic effects so that no more than one in any nine-rod array is allowed to be completely crimped (totally blocked). Complete severance of withdraw lines will not affect the control rod insert function. Protection for the CRD insert lines is not required since a reactor pressure of 450 psig or higher (CRD insert lines principal backup) could adequately insert the control rods even with a complete loss of insert lines. Routing of high-energy lines in the vicinity of the CRD withdraw lines is strictly controlled.

7. The escape of steam, water, combustible, or corrosive fluids, gases, and heat in the event of a pipe rupture does not preclude:

    a. Habitability of the main control room

    b. The ability of essential instrumentation, electric power supplies, components, and controls to perform their safety function.

3.6.2A Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2.1A Criteria Used to Define Break and Crack Location and Configuration

3.6.2.1.1A Definition of High-Energy Fluid System

High-energy fluid systems are defined as those systems or portions of systems that during normal plant conditions are either in operation or are maintained pressurized under conditions where either or both of the following are met:

Maximum temperature exceeds 200°F, or

Maximum pressure exceeds 275 psig.

Normal plant conditions are defined as the plant operating conditions during reactor startup, power plant operation, and reactor cold shutdown, but excluding test modes.

3.6.2.1.2A Definition of Moderate-Energy Fluid System

Moderate-energy fluid systems are defined to be those systems, or portions of systems, that during normal plant conditions are either in operation or are maintained pressurized under conditions where both of the following are met:

Maximum temperature is 200°F or less, and

Maximum pressure is 275 psig or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system functions, but for the major operational period qualify as moderate-energy fluid systems. An operational period is considered "short" if the total fraction of time that the system operates within the pressure-temperature conditions specified for the high-energy fluid system is less than two percent of the total operating time for which the system is designed.

3.6.2.1.3A Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden, gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or as the development of a sudden longitudinal, uncontrolled crack (longitudinal split), and is postulated for the high-energy fluid system only. For moderate-energy fluid systems, pipe breaks are confined to the postulation of controlled cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only, and do not result in whipping of the cracked pipe.

Portions of piping systems that are isolated from the source of the high-energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This would include portions of piping systems beyond a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness. Internal missiles that might be generated from failures of these components are evaluated as discussed in Section 3.5.1.

A high-energy piping system break is not postulated simultaneously with a moderate-energy piping system crack.

The evaluations of pipe breaks and cracks are in accordance with Revision 0 of SRP 3.6.2 and Branch Technical Position MEB 3-1, November 24, 1975, the documents applicable at the time the evaluations were done. Therefore, high energy leakage cracks were not considered.

High energy line leakage cracks would not be postulated in fluid system piping located in containment penetration areas, since the design stress and fatigue limits specified in BTP MEB 3-1, Section B.1.b, are met for high energy piping in these areas (see Section 3.6.2.1.5.2A).

The effects of high energy line leakage cracks in other areas would generally be bounded by the analyses performed for high energy line breaks, moderate energy line cracks, and inadvertent fire suppression system actuation.

3.6.2.1.4A Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip need not be provided if any one of the following exists:

1. Piping which is classified as moderate-energy piping.

2. Following a single postulated pipe break, piping for which the unrestrained movement of either end of the ruptured pipe in any feasible direction cannot impact any structure, system, or component important to safety.

3. Piping for which the internal energy level associated with whipping is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. Any line restrictions (e.g., flow limiters) between the pressure source and break location, and the effects of either a single-ended or double-ended flow condition are accounted for in the determination of the internal fluid energy level associated with the postulated pipe break reaction. The energy level in a whipping pipe is considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

3.6.2.1.5A Postulated Pipe Break Locations

3.6.2.1.5.1A Criteria for Inside the Containment

For ASME Section III, Class 1 piping systems within the containment, design basis piping break locations are selected using the following criteria:

1. At the terminal ends including:

   a. Piping, pressure vessel, or equipment nozzle intersections

   b. High energy-moderate energy boundary

   c. A branch connection to a main run unless all the following are met:

      (1) The branch and main runs are of comparable size and fixity (i.e., the nominal size of the branch run is at least one-half that of the main run);

      (2) The intersection is not rigidly constrained to the building structure; and

      (3) The branch and main runs are modeled as a common piping system during the pipe stress analysis.

2. At the intermediate locations between the terminal ends selected by either of the following criteria:

   a. At each fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve

   b. At locations where the maximum stress range for the normal and upset plant conditions and for an operating basis earthquake (OBE) exceeds 2.4 S_m, calculated either by Equations (12) or (13) in ASME Code Section III, Subsection NB-3653; and at locations where the cumulative usage factor, U, derived from the piping fatigue analysis, under the loadings associated with OBE and operational plant conditions exceeds 0.1. S_m is the allowable stress intensity as specified in ASME Code Section III, Subsection

NB-3213.1
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For ASME Code Section III, Class 2 and 3 piping systems, break locations are postulated by the following criteria:

1. At the terminal ends

2. At the intermediate locations between the terminal ends selected by either of the following criteria:

    a. At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve

•→12

    b. At each location where the stress associated with normal and upset plant conditions and an OBE event calculated by Equations (9) plus (10) in Paragraph NC-3652 of the ASME Code, Section III, exceeds 0.8 (1.2 S_h+S_A).

12←•

* Cumulative usage factor (cuf) is specified in ASME Code Section III, Subsection NB3222.4.

S_h = Allowable stress at the elevated temperature calculated according to ASME Code Section III, Subsections NC-3600 and ND-3600 for Class 2 and 3 components, respectively.

S_A = Allowable stress range for the expansion stress calculated according to ASME Code Section III NC-3600 and ANSI B31.1.

3. If a fatigue analysis is performed at any intermediate locations between the terminal ends where the cumulative usage factor under the loading associated with OBE and operational plant conditions exceeds 0.1.

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3.6.2.1.5.2A Criteria for Outside the Containment

3.6.2.1.5.2.1A High-Energy Fluid Systems

The following criteria are used to define break and crack locations in high-energy fluid systems outside the containment:

1. Fluid Systems Separated from Essential Structures, Systems, and Components

   Breaks are not postulated in high energy piping at locations that are isolated or physically remote from essential equipment, structures, and the containment. For locations that are marginal, breaks are assumed for the purpose of establishing separation.

2. Fluid System Piping in Containment Penetration Areas

   Breaks are not postulated in the portions of high energy piping between the containment isolation valves, outside and inside containment. Breaks are not postulated in the portions of high energy piping between the isolation valve and the first restraint or groups of restraints designed to protect these portions of piping from breaks outboard of this area both inside and outside containment.

These pipe whip restraints are capable of resisting bending and torsional moments produced by a postulated piping failure outboard of the first restraint or group of restraints beyond the containment isolation valves.

The restraints are designed to withstand the loadings resulting from a postulated piping failure beyond these portions of piping, so that neither the isolation valve operability nor the leaktight integrity of the associated containment penetration will be impaired. These portions of piping are designed to meet the requirements of ASME Code, Section III, Subarticle NE-1120, and the following additional design requirements, which are in conformance with Revision 1 (July, 1981) of SRP 3.6.2 and BTP MEB 3-1, the documents applicable at the time the analysis was performed:

a. The following design stress and fatigue limits are not exceeded for Class 1 piping:

    1) The maximum stress range between any two load sets (including the zero load set), calculated by Equation (10) in Paragraph NB-3653, ASME Code, Section III, for those loads and conditions thereof, for which level A and level B stress limits have been specified in the design specification, including an operating basis earthquake (OBE) event transient, do not exceed 2.4 S_m. If the calculated maximum stress range of Equation (10) exceeds 2.4 S_m, the stress ranges calculated by both Equation (12) and Equation (13) in Paragraph NB-3653 should meet the limit of 2.4 S_m.

    (2) The cumulative usage factor is less than 0.1.

    (3) The maximum stress, as calculated by Equation (9) in Paragraph NB-3652 under the loading resulting from a postulated

piping failure beyond these portions of piping, does not exceed 2.25 S_m, except that following a failure outside containment, the pipe between the outboard isolation valve may be permitted higher stresses provided a plastic hinge is not formed.

b. The following design stress limits are not exceeded for Class 2 piping:

(1) The maximum stress ranges do not exceed 0.8 $(1.2S_h+S_A)$, as calculated by Equations (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event.

(2) The maximum stresses do not exceed $1.8S_h$, as calculated by Equation (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping.

c. Welded attachments for pipe supports or other purposes, to these portions of piping are avoided, except where detailed stress analysis demonstrates compliance with the limits previously discussed in Section 3.6.2.1.5.2A, Items 2a and 2b.

d. The number of circumferential and longitudinal piping welds and branch connections is minimized.

e. The length of these portions of piping is reduced to the minimum length practicable.

f. The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) does not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings are used), except where such welds are capable of 100 percent volumetric inservice inspection. This criterion is also applicable to the portion of piping between the containment and the inside containment isolation valves.

g. For these portions of high-energy fluid system piping, preservice and subsequent inservice examinations are performed in accordance with the requirements specified in ASME Code Section XI. In addition, during each inspection interval, volumetric examination of nonexempt ASME Code Section XI, Class 1 and 2 circumferential butt welds greater than 4-in NPS will be performed on those portions of high-energy fluid system piping between the first moment-limiting restraint outside containment and the first moment-limiting restraint inside containment in accordance with a risk-informed methodology. This area of examination interest is referred to as the break exclusion region. Details of containment penetration, identification of pipe welds, access for inservice inspection points of fixity and discontinuity are provided in Section 3.8.2.

h. Regardless of the fact that all the conditions above have been met, a crack in the main steam or feedwater piping in this region is postulated. The crack in the pipe, equal in area to a single-ended pipe rupture, is considered a singular event. Pipe whip and jet impingement are not considered and a single-active failure is not taken as a concurrent event.

3. Balance of Piping Outside the Containment

a. Breaks in ASME Code, Section III, Class 2 and 3 piping and in nonnuclear class piping seismically analyzed and supported are postulated at the following locations in each piping and branch run (except those portions of fluid system piping identified in Section 3.6.2.1.5.2A, items 1 and 2):

(1) At terminal ends of the pressurized portions of the runs.

(2) At intermediate locations selected by either of the following criteria:

(a) At each pipe fitting (e.g., elbow, tee, cross, and nonstandard fitting), welded attachment, and valve, or, if the run contains no fittings, at one location at each extreme of the run within the protective structure (a terminal end, if located within a protective structure, may substitute for one intermediate break).

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(b) At each location where the stresses associated with normal and upset plant conditions and an OBE event exceed 0.8 $(1.2S_h+S_A)$, as calculated by Equations (9) and (10), Paragraph NC-3652 of the ASME Code, Section III, for Class 3 piping.

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b. Breaks in nonnuclear safety class piping not seismically qualified are postulated at the following locations in each piping or branch run:

(1) At terminal ends of the pressurized portions of the runs

(2) At each pipe fitting, welded attachment, and valve.

These breaks are sufficient to establish the worst pipe break effects since either:

(1) The piping is physically remote from essential equipment and structures,

(2) There are a large number of pipe breaks postulated on the same line in the same area due to a large number of pipe fittings or attachments on the pipe, or

(3) For non-seismic piping in close proximity to safety related systems, components, or structures, either the piping is seismically analyzed and supported or other protection is provided.

3.6.2.1.5.2.2A Moderate-Energy Fluid Systems

The following criteria are used to define crack locations in moderate-energy fluid systems outside the containment:

1. For the purpose of satisfying the separation provisions of plant arrangement, a review of the piping layout and plant arrangement drawings is conducted. The effects of through-wall leakage cracks are isolated or physically remote from safe shutdown systems, to the extent this is practical.

2. Leakage cracks are not postulated in those portions of piping between the isolation valve and the containment, provided they meet the requirements of ASME Code, Section III, Subarticle NE-1120, and are designed so that the maximum stress range associated with normal and upset plant conditions and an OBE event does not exceed 0.4 $(1.2S_h+S_A)$ (as calculated by Equations (9) and (10), Paragraph NC-3652 of the ASME Code, Section III, Class 2 piping).

3. Through-wall leakage cracks are postulated in fluid system piping, except where exempted by Section 3.6.2.1.5.2A, Items 1, 2, and 4, or where the maximum stress range, associated with normal and upset plant conditions and an OBE event, in these portions of ASME Code Section III, Class 2 or 3 piping and nonnuclear piping is less than 0.4 $(1.2S_h+S_A)$ (as calculated by Equations (9) and (10), Paragraph NC-3652 of the ASME Code, Section III). The cracks are postulated to occur individually at locations that result in the maximum effects from fluid spray and flooding. Only environmental effects that develop from these cracks are considered.

4. Cracks are not postulated in moderate-energy fluid system piping located in an area in which a break in high-energy piping occurs. Where a postulated leakage crack in the moderate-energy fluid system piping results in more limiting environmental conditions than the break in proximate high-energy fluid system piping, the provisions identified in Section 3.6.2.1.5.2A, Item 3, are applied.

5. Through-wall leakage cracks, instead of breaks, are postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods, but qualify as moderate-energy fluid systems for the major operational period. An operational period is considered short if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2 percent of the time that the system operates as a moderate-energy fluid system (e.g., systems such as the reactor RHR system qualify as moderate-energy fluid systems).

3.6.2.1.6A Design Basis Break/Crack Types and Orientation

3.6.2.1.6.1A Circumferential Pipe Breaks

The following circumferential breaks are postulated in high-energy fluid system piping at the locations specified in Section 3.6.2.1.5A:

1. Circumferential breaks are postulated in fluid system piping runs and branches exceeding a nominal pipe size of 1 in. When the maximum stress range or usage factor exceeds the limits specified for break postulation, and if it is determined by detailed stress analysis that the maximum stress range in the circumferential direction is at least 1.5 times that in the axial direction, then only longitudinal breaks are postulated.

2. Where break locations are selected at pipe fittings without the benefit of stress calculations, breaks are postulated at the piping weld to each fitting, valve, or welded attachment. If detailed stress analyses or tests are performed, the maximum stressed location in the fitting may be selected instead of the pipe-to-fitting weld.

3. Circumferential breaks are assumed to result in pipe severance and separation amounting to a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic analysis.

4. The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs will be taken into account, as applicable, in the reduction of jet discharge.

5. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and is assumed to cause pipe movement in the direction of the jet reaction, unless shown to be otherwise by analysis.

3.6.2.1.6.2A Longitudinal Pipe Breaks

The following longitudinal breaks are postulated in high-energy fluid system piping at the locations of each circumferential break specified in Section 3.6.2.1.6.1A, except as noted:

1. Longitudinal breaks in fluid system piping and branch runs are postulated in nominal pipe sizes 4 in and larger. However, when the maximum stress range or usage factor exceeds the limits specified for break postulation, and it is determined by detailed stress analysis that the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, then only a circumferential break is postulated.

2. Longitudinal breaks are not postulated at:

    a. Terminal ends

    b. Intermediate locations where the criterion for a minimum number of break locations must be satisfied.

3. Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are located (but not concurrently) at two diametrically opposed points on the piping circumference in such a way that a jet reaction causing out-of-plane bending of the piping configuration results. Alternately, a single split may be assumed at the section of highest stress as determined by detailed stress analysis.

4. The dynamic force of the fluid jet discharge is based on a circular break area equal to the effective cross-sectional flow area of the pipe at the break location, and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.

5. Pipe movement is assumed to occur in the directions defined by the stiffness of the piping configuration and jet reaction forces, unless limited by structural members or piping restraints.

3.6.2.1.6.3A Through-Wall Leakage Cracks (Outside the Containment Only)

Through-wall leakage cracks are postulated in main steam or feedwater piping systems in containment penetration areas as stated in Section 3.6.2.1.5.2.1A Item 2h. The following through-wall leakage cracks are postulated in moderate-energy fluid system piping at the locations specified in Section 3.6.2.1.5.2.2A:

1. Cracks are postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 in.

2. Fluid flow from a crack is based on a circular opening of area equal to that of a rectangle one-half the nominal pipe diameter in length and one-half the pipe wall thickness in width.

3. The flow from the crack is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments.

Flooding effects are determined on the basis of a conservatively estimated time period required to effect corrective actions.

3.6.2.1.7A Conformance With Regulatory Guide 1.46, May 1973

1. Protection against pipe whip inside the containment is in conformance with Regulatory Guide 1.46, with the following modifications or exceptions,

   Paragraph C.1.b

   "Any intermediate locations where the maximum stress ranges ... exceed 2.4S calculated by either EQ(12) or EQ(13) in Paragraph NB-3653 of the ASME Code Section III;..."

   In lieu of

   "Any intermediate locations ... conditions exceed $2.0S_m$ for ferritic steel and $2.4S_m$ for austenitic steel."

   Paragraph C.2.b

   "Any intermediate locations ... conditions exceed 0.8 $(1.2S_h+S_A)$."

   In lieu of

   "Any intermediate locations ... conditions exceed 0.8 (S_h+S_A)."

   Section 3.6.2A allows for a single intermediate break for a straight run of pipe under specified conditions.

   Section 3.6.2A states that the junction of a branch run, if included in the same structural model as the main run, is not a terminal end, whereas Regulatory Guide 1.46 states that a branch connection is a terminal end.

   Paragraph C.3.a,b

   The following criteria in Section 3.6.2A are not stated in the Regulatory Guide:

   Longitudinal breaks are not required at terminal points nor at locations where the criterion for a minimum number of break locations must be satisfied.

Circumferential breaks are not postulated where detailed stress analysis shows that circumferential stress is a least 1.5 times that in the axial direction. If the axial stress is at least 1.5 times the circumferential stress, longitudinal breaks are not postulated.

Paragraph C.3.a, Footnote (10)

Longitudinal breaks are assumed to result in axial split without pipe severance. Splits are oriented (but not concurrently) at two diametrically opposed points on the piping circumference, in such a way that the jet reactions cause out-of-plane bending of the piping configuration. Alternatively, a single split is assumed at the section of highest stress as determined by detailed stress analysis (e.g., finite element analysis).

The dynamic force of the fluid discharge is based on a circular or elliptical (2D X 1/2D) break area equal to the effective cross sectional flow area of the pipe at the break location.

In lieu of

Footnote 10 of Regulatory Guide 1.46.

Paragraph C.3.b, Footnote (11)

"Pipe whipping is assumed to occur in the plane defined by the piping configuration and geometry and to cause pipe movement in the direction of the jet reaction," unless shown to be otherwise by analysis.

In lieu of

"Dynamic forces resulting ... cause whipping in any direction normal to the pipe axis." (In Footnote 11 of Regulatory Guide 1.46)

Paragraph C.4.d

Regulatory Guide 1.46 excludes piping from break postulation if the design temperature is 200°F or less and pressure is 275 psig or less.

Section 3.6.2A specifies temperature and pressure during normal plant operation.

2. Certain provisions of Regulatory Guide 1.46 are impractical. While NRC is in the process of revising Regulatory Guide 1.46, the NRC Standard Review Plan 3.6.2, paragraph II.1 states: "If the criteria specified in Regulatory Guide 1.46 are impractical to implement for a specific application, the criteria of Branch Technical Position BTP MEB 3-1 will be considered," and "BTP MEB 3-1 may be used for all applications, in lieu of References 12 and 13, at the option of the applicants." The modification or exceptions on the position of Regulatory Guide 1.46 are based upon the contents of BTP MEB 3-1.

3.6.2.2A Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1A Introduction

Pipe rupture analyses consist of calculations to determine the fluid forces generated by the blowdown of pressurized lines, complemented by dynamic or energy-balance analyses to determine pipe motion and impact effects (Fig. 3.6A-1). Restraints for lines 6 in and less in diameter are usually qualified on a generic basis using an energy balance. However, restraints for larger lines are engineered individually for each system, usually using standard design concepts and worst case dynamic analysis to qualify several similar restraints in different locations. The response of unrestrained lines is analyzed by either inelastic dynamic analysis or energy balance analysis.

Criteria for the response analyses are as follows:

1. An analysis of the pipe run or branch is performed for each postulated longitudinal and circumferential rupture or, alternatively, for a worst case. Worst cases are selected on the basis of gap, fluid force, and piping system stiffness.

2. The loading condition of a pipe run or branch prior to postulated rupture in terms of internal pressure, temperature, and stress state is that condition associated with reactor operation at 100-percent power.

3. For a circumferential rupture, pipe whip dynamic analyses are only performed for that end (or ends) of the pipe or branch that is (are) connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream.

4. Dynamic analytical methods, used for calculating the piping or piping/restraint system response to the jet thrust developed after a postulated rupture, adequately account for the effects of the following:

   a. Mass inertia and stiffness properties of the system

   b. Impact and rebound (if any) as permitted by gaps between piping and restraint

   c. Elastic and inelastic deformation of piping and/or restraint

   d. Support boundary conditions.

5. An allowable design strain limit of 0.5 ultimate uniform strain of the restraints is used for tensile energy-absorbing components. For compressive energy-absorbing components, a design limit of 80-percent of energy-absorbing capacity is used.

6. A 10-percent increase of minimum specified yield strength (S_y) may be used to account for strain rate effects in inelastic nonlinear analyses. Alternatively, experimental data may be used to determine the strain rate parameters for use in nonlinear codes which monitor strain rate.

3.6.2.2.2A Time-Dependent Blowdown Force

The blowdown force calculations, which are similar to those of Moody, are based on the transient pressures, velocities, and other thermodynamic properties of the fluid[1]. To provide the time history of pressure, velocity, etc, the method of characteristics is used to solve the continuity and momentum equations simultaneously. A general description of the method can be found in most gas dynamics textbooks[2,3,4,5]. For these one-dimensional fluid mechanics analyses, the pipe is regarded as straight, despite numerous bends. The calculated momentum and pressure forces are applied at changes in direction or cross section of the piping to provide time-dependent loads for pipe dynamic analysis.

The transient forces result from wave propagation and fluid momentum. It is assumed that pipe bends and elbows neither attenuate the traveling pressure waves nor cause reflections. Immediately following the rupture of a pipe, a decompression wave travels from the break at the speed of sound relative to the fluid. The fluids ahead of and behind the wave are at different states. This initial blowdown condition will last until a return signal from a pressure reservoir reaches the break. Repeated wave reflections between the reservoir and break prevail until a steady-state flow condition is established. Boundary conditions that govern the flow at the break end and at the inlet from the vessel to the pipe are applied.

The time histories of transient pressure, mass flow rate, and other thermodynamic properties of the fluid are based on the following equation, which includes static and dynamic effects, to calculate the blowdown force:

$$F = \left[P_e - P_a + \frac{RU_e^2}{144g} \right] A$$

where:

F = Blowdown force, lb_f

P_e = Pressure at exit plane, psia

P_a = Ambient pressure, psia

U_e = Velocity of fluid at exit plane, fps

R = Density of fluid, lb_m/ft^3

A = Pipe break area, sq in

g = Gravitational constant, lb_m-ft/$lb_f sec^2$

The effects of line friction are included in the evaluation of steady-state blowdown. For the calculation of the transient fluid response, however, friction may or may not be considered.

3.6.2.2.2.1A Subcooled Nonflashing Waterline Blowdown

Transient Flow

Immediately following the rupture, a flow disturbance propagates from the break at a speed of sound relative to the fluid, leaving the fluid behind the wave at a thermodynamic state of U and P = P-. The governing equation across the wave is:

$$\Delta P = \pm RC/g \ \Delta U$$

where:

ΔP = Differential pressure across wave, psia

ΔU = Differential velocity across wave, fps

C = Speed of sound in fluid, fps.

When the disturbance reaches a pressure reservoir, it is reflected and travels toward the break end. The boundary conditions that govern the flow at the break location and at the inlet to the pipe (from the reservoir) are:

$$P_e = P_a$$

$$P_i = P_o - \frac{RU_i^2}{2g}$$

where:

P_i = Pressure at pipe inlet, psia

U_i = Velocity of fluid at pipe inlet, fps

P_e, P_a = Pressure at the break location, psia

P_o = Reservoir pressure, psia.

The initial blowdown flow remains constant until the disturbance, which is reflected from the vessel, reaches the break end. Then it is reflected again, and that brings a change of blowdown flow. These repeated wave transmissions and reflections continue until the steady-state flow is established.

Steady-State Flow

For steady-state flow, the blowdown forcing function calculations become:

$$F = \left[\frac{2(P_o - P_a)}{P_o} \frac{1}{1 + \frac{fL_e}{D}} \right] P_o A$$

which is derived by applying Bernoulli's equation across the pipe and by using the expression for the forcing function calculation,

where:

L_e = Total equivalent length of pipe friction, ft

f = Friction factor (Reynolds number and pipe surface roughness dependent)

D = Pipe inside diameter, ft

3.6.2.2.2.2A Steamline Blowdown

Transient Flow

Steam is treated as an ideal, single-phase gas with a constant specific heat ratio, k, of 1.3. Except for the case of steady-state blowdown flow, the flow is assumed to be isentropic with negligible pipe friction. The characteristic method, which is a finite difference approximation using the principle of characteristics, is used as a basis for the numerical solution of the continuity and momentum equations[6,7]. The transient pressure, mass flow rate, and other thermodynamic properties are then used to calculate the transient-state forcing function.

Immediately following the break, a decompression wave travels into the pipe toward the pressure reservoir. The fluid in front of the wave is at a state

$U_i = 0$

$C_i = C_o$

where:

U_i = Velocity of fluid, fps

C_o = Speed of sound in fluid, fps

The fluid state at the exit is at the sonic condition, because the initial pressure was sufficiently high[8]:

$$\frac{U_e}{C_o} = \frac{C_e}{C_o} = \frac{2}{k+1} = 0.8695 \text{ for } k=1.3$$

The blowdown force can be calculated as

$$F = \left[\frac{P_e}{P_o} + \frac{R_e C_e^2}{gP_o}\right] P_o A = \left[\frac{P_e}{P_o} + \frac{R_e}{R_o}\left(\frac{C_e}{C_o}\right)^2 \frac{R_o C_o^2}{gP_o}\right] P_o A$$

The pressure ratio across the wave is

$$\frac{P_e}{P_o} = \left(\frac{T_e}{T_o}\right)^{\frac{k}{k-1}} = \left(\frac{C_e}{C_o}\right)^{\frac{2k}{k-1}} = \left(\frac{2}{k+1}\right)^{\frac{2k}{k-1}} = 0.298$$

where:

T = Temperature, and the density ratio is

$$\frac{R_e}{R_o} = \left(\frac{P_e}{P_o}\right)^{\frac{1}{k}} = \left(\frac{C_e}{C_o}\right)^{\frac{2k}{k-1}\left(\frac{1}{k}\right)} = \left(\frac{2}{k+1}\right)^{\frac{2}{k-1}}$$

Therefore, the blowdown force can be reformulated as

$$F = (1+k)\left(\frac{2}{k+1}\right)^{\frac{2k}{k-1}} P_o A = 0.685\ P_o A$$

The blowdown force is constant until a return signal from the pressure source reaches the break.

When the wave reaches the reservoir, it is reflected as a compression wave. The boundary condition at the pressure lies on the steady-state ellipse,

$$\left(\frac{C_i}{C_o}\right)^2 + \frac{k-1}{2}\left(\frac{U_i}{C_o}\right)^2 = 1$$

which is the energy equation applying across the vessel-pipe inlet.

The boundary condition for this case is

$$T_o = T_i + U_i^2/2C_p$$

where:

C_p = Constant pressure specific heat of a fluid, Btu/Slug °F

i = State at the inlet to the pipe.

Steady-State Flow

If the steady state is reached, the flow in the pipe is uniform and, if the pressure in the pressure vessel remains high, then the boundary condition at the break always lies on the sonic line, that is,

$$U^*/C_o = C^*/C_o$$

Then from the critical flow condition,

$$U^*/C_o = C^*/C_o = \sqrt{(2/k+1)} = 0.9325$$

where:

* = Critical flow condition.

Then, the steady-state blowdown force is

$$F = \left(\frac{P^*}{P_o} + \frac{R^*(U^*)^2}{g}\right) P_oA = (1+k)\left(\frac{2}{k+1}\right)^{\frac{k}{k-1}} P_oA = 1.255\ P_oA$$

For steady-state flow with friction losses, the analysis is based on the theory of compressible flow with friction[8]. The pipe friction is the chief factor bringing about the change of fluid properties in the flow. A curve which describes the variation of steady state steam blowdown force versus friction parameter fL/D is shown in Fig. 3.6A-2.

3.6.2.2.3A Simplified Blowdown Analysis

A conservative steady-state forcing function may be used for calculations based on the energy balance method. The function has a magnitude of:

$$T = KPA$$

where:

P = System pressure prior to pipe break, psia

A = Pipe break area, sq in

K = Thrust coefficient (theoretical maximum)

K values are as follows:

1. 1.26 for saturated steam, water, and steam/water mixture

2. 2.00 for nonflashing subcooled water.

An amplification factor between 1.1 and 1.2, to account for rebound, is applied to the above force. Alternatively, the maximum fluid force during the energy input phase as determined by the detailed methods of Section 3.6.2.2.2A may be used. In determining this maximum, a brief initial force of 1 PA may be ignored since the initial pipe velocity is low and the resulting work input is inconsequential. The above amplification factor for rebound is also included.

3.6.2.2.4A Lumped-Parameter Dynamic Analysis

The piping system is modeled mathematically as a series of beam elements connected at nodes. The geometry of the model matches that of the pipe. The distributed mass of the pipe and contained fluid is modeled as lumped masses located at the nodes. The beam elements have the stiffness properties of the pipe in the elastic range and approximate the plastic behavior after yield.

Before a rupture, the pipe is stressed by internal pressure, but remains in static equilibrium. When initial conditions have a significant effect on the parameters being calculated, such as stresses in break exclusion regions or loads on attached components, this effect is considered.

As a circumferential break propagates, the load-carrying metal area of the pipe decreases so that a force unbalance results. The force initially transmitted across the break is assumed to drop linearly to zero in 1 millisecond. After the break, the forces exerted on the pipe by the fluid are determined by the time-dependent blowdown force derived in Section 3.6.2.2.2A. Similarly, for a longitudinal split, the crack propagation speed limits the rate at which the split opens, so a 1-millisecond force rise time is assumed. Other break opening times may be used if justified.

Subsequent to a postulated rupture, the inelastic system response is analyzed by the use of an elastic-plastic lumped-mass beam element computer code such as DINASAW or LIMITA (Appendix Sections 3B.2.1 through 3B.2.3). The analysis considers the free motion of the pipe through a gap, if one exists, using the appropriate initial conditions and the fluid blowdown forces as calculated in Section 3.6.2.2.2A. The mathematical model includes the restraint or barrier, and sometimes a member simulating the local crush resistance of the pipe. Rebound effects are considered by automatically connecting and disconnecting that member for impact and rebound, respectively.

Sample Dynamic Analysis

Pipe rupture restraint MSS-PRR-905 outside the containment in the auxiliary building limits the motion of the main steam line following a circumferential break at the elbow. The restraint is a limit stop-type restraint (Fig. 3.6A-11) with a 1.23-in gap between the hot pipe and the restraint.

The analysis of the pipe-restraint interaction used the LIMITA2 computer code. The finite-element model is shown in Fig. 3.6A-3. The fluid forces depicted in Fig. 3.6A-4 were applied to the pipe elbow as shown in Fig. 3.6A-3.

The restraint reaction load is shown in Fig. 3.6A-5. The maximum restraint load is 485 kips and the maximum deformation of the honeycomb panel is 0.7 in. The peaks of Fig. 3.6A-5 are flattened at 485 kips due to the nature of the honeycomb load/deflection curve.

3.6.2.2.5A Energy Balance Analysis

The energy balance technique for analyzing pipe impact equates the work done by the escaping fluid to the energy absorbed in deforming the ruptured pipe and the impacted target. A steady-state blowdown force is used for the energy balance analysis. The magnitude of the force is described in Section 3.6.2.2.3A.

The input energy of the system is determined by multiplying the pipe displacement at the break end by the component of the fluid blowdown force in the direction of the displacement.

The input energy is

$$E = F \times D$$

where:

F = Component of blowdown force in direction of pipe displacement, lb

D = Displacement of break end of pipe, in.

The strain energy absorbed during pipe whip and impact consists of the energy absorbed by pipe bending, E_{pb}, the energy absorbed by pipe crush during impact, E_{pc}, and the energy absorbed by deformation of the target, E_t.

To determine post-impact target deformation and the peak reaction force, the input energy is equated to the strain energy absorbed by the pipe and target. The energy absorption characteristics of the pipe crush and target deformation are calculated on the basis of the displacement integral of the appropriate force-deformation curves.

<u>Sample Energy Balance Analysis</u>

The same main steam restraint MSS-PRR-905 (Fig. 3.6A-6) is analyzed here by the energy balance technique. The energy input from the fluid blowdown force is:

$$E_{in} = F_b(g+d)(L_h/L_h-L)$$

where:

F_b = Fluid blowdown force, lb

g = Acceleration gap of restraint, in

d = Restraint deflection, in

L_h = Length from break to plastic hinge, in

L = Length from break to restraint, in

The ratio $L_h/(L_h-L)$ represents the increase pipe displacement at the break, compared to displacement at the restraint, due to the assumed pipe rotation about a plastic hinge.

The plastic hinge length L_h is derived by an iteration technique based upon the following:

Based on the pipe model and considering the dynamic equilibrium of the portion of pipe before the plastic hinge, the following equation can be derived.

$$\ell_p = \frac{3M_p}{2F}\left(1 + \sqrt{1 + \frac{8LF}{3M_p}}\right)$$

where:

ℓ_p = Length of the dynamic plastic hinge

M_p = Plastic bending moment

F = Blowdown force

L = Length of pipe between break and elbow

Then a simplified estimate of ℓ_p is found by conservatively assuming L = 0 and thus:

$$\ell_p = \frac{3M_p}{F}$$

Comparison of the minimum ultimate moment of the pipe to the dynamic analysis and deflection data assures that the whipping pipe is not severed at the restraint to become a missile.

The fluid force is calculated:

$F_b = k_r\ k_f\ P_o\ A = 320$ kips

where:

k_r = Rebound factor (1.2)

k_f = Thrust coefficient (0.7)

•→15

P_o = Initial pressure (1050 psi) * Based on pre-Uprate pressure. Blow down forces were evaluated to a revised dome Pressure of 1074 psia due to Power Uprate.

15←•

A = Pipe flow area (365.1 sq in)

In this sample energy balance analysis, the force history had been previously calculated. It corresponds to a thrust coefficient of 0.7. Further, this sample energy balance represents a conceptual determination of the reaction force (embedment load) at the restraint which is later verified by dynamic analysis. The elastic portion of the load and deflection of the honeycomb panel exists at initial impact only, and the elastic strain energy is negligible. Also, the honeycomb panels are precrushed thereby eliminating the peak of the deflection.

This energy may be absorbed in plastic bending of the pipe and in crush of the restraint. The energy absorbed by bending at the plastic hinge is:

$$E_b = M_{p\theta} = M_p\ (g+d)/(L_h - L)$$

where:

M_p = Plastic moment of the pipe

$= \sigma_y\ (D_o^3 - D_i^3)/6$

$= 27{,}100\ (24^3 - 21.564^3)/6$

$= 1.72 \times 10^7$ in-lb

$L_h = 3\ M_p / F_b = 160.$ in

L = 56 in

g = 1.23 in

d = Allowable crushing of the honeycomb panel

= 0.8 (total crushable depth of honeycomb panel)

= 0.8 x 0.7 x 5.5 = 3.1 in

The energy absorbed in the honeycomb panel is

$$E_a = F \times d$$

By equating E_{in} to (E_a+E_b), the restraint reaction force F is found to be 460 kips, and the honeycomb panel is crushed 3.1 in.

3.6.2.2.6A Local Pipe Indentation

The local shell indentation stiffness of the pipe is usually considered where other energy-absorbing mechanisms are not available at the point of impact. Examples include impacts into rigid displacement-limiting bumpers, concrete walls, and the omnidirectional restraint weldment (the latter interposes a significant mass between the impacting pipe and the energy absorbers).

Two methods have been used to determine the shell indentation stiffness. The earlier was analytical and tended to overpredict conservatively the indentation stiffness. The other was a series of pseudostatic pipe crush tests covering several crush geometries and a sufficient range of pipe thicknesses and diameters to develop parametric scaling laws[9,10]. This was augmented by analyses to determine the sensitivity to material strength, dynamics, and variations in loading geometry.

3.6.2.2.7A Concrete Barrier Impact

In a pipe whip impact, the force on the barrier is a complex function of time depending primarily on the sudden deceleration of the pipe wall at the impact point (slug impact), the shell indentation of the pipe as it locally crushes against the wall, and the force transmitted to the impact point by the more gradual deceleration of the adjacent run of pipe. After impact, the pipe also transmits a more enduring force resulting from the continuing fluid blowdown. The concrete is affected in the same way as in any other missile impact event, the only significant difference being the long term fluid force. To evaluate this postulated event, the pipe is transformed into an equivalent missile and the concrete is analyzed for scabbing and structural response using the procedure described in Section 3.5.3. The analysis for structural response includes the impulse of the initial impact as well as the subsequent fluid blowdown force and other concurrent loads.

Four basic parameters must be determined to define the equivalent missile: the kinetic energy (or impulse), the impact velocity, the pipe crush stiffness, and the bearing area. The kinetic energy and velocity can be found by either of two methods:

1. Simplified Method - Use the total input energy (fluid blowdown force times distance of pipe travel) less the energy absorbed in pipe bending prior to impact. Compute the velocity using approximate formulae.

2. Lumped-Parameter Dynamic Analysis (Section 3.6.2.2.4A) - This method is especially suited for evaluating the impact of piping systems with complex geometries and can even consider multiple impact points. As an alternative to the kinetic energy, the impact force history (impulse) can be computed.

Regardless of which analysis method is used, the crush resistance of the equivalent missile and the bearing area are derived from the experimental data described in Section 3.6.2.2.6A. These data are modified to account for the effect of dynamics and internal pressure.

3.6.2.3A Dynamic Analysis Methods to Verify Integrity and Operability

Pipe rupture loads to determine the integrity of mechanical components are determined using the analytical methods described in Section 3.6.2.2.2A. The applicable load combinations for the components and for break exclusion regions are presented in Sections 3.9 and 3.6.2.1.5.2A, Item 2, respectively. Criteria for rupture restraints are presented in Section 3.6.2.3.1A.

3.6.2.3.1A Jet Impingement Analysis

Jet impingement loadings are determined as follows:

1. The jet force at the exit plane of a pipe break is calculated as discussed in Section 3.6.2.2.2A. This jet force is dependent on the fluid condition in the system, which varies with time. For jet impingement analysis, only the peak force is used unless a complete jet time history is required to reduce conservatism. When pipe friction is negligible and there are no upstream flow restrictions, the peak jet load is 1.26 P_oA for saturated steam and saturated water, and 2.0 P_oA for nonflashing subcooled water.

2. The jet expands as it travels along its path. The jet shape is assumed to be conical at a 10-degree half-angle expansion for subcooled water. Moody's asymptotic expansion model[1] is adopted for saturated water and saturated steam (Figure 3.6A-5(a)).

3. The jet is assumed to proceed in a straight path. The directions of the jet paths are based on the type of break, direction of pipe whip, and pipe restraint configuration.

    a. For circumferential breaks where the two separated pipe ends are not physically restrained or restrained by pipe whip restraints with large restraint gap, the pipe ends move clear of each other so no interference with the jet issuing from each severed end occurs, and the centerline of each jet at the break is coincident with the pipe centerline (Figure 3.6A-52(a)).

    b. For circumferential breaks in piping physically restrained from significant separation (axial pipe movement equal to or less than 1/2 pipe diameter and lateral pipe movement less than pipe wall thickness) following the break, the jet centerline is assumed normal to the pipe centerline and extends 360° around the circumference of the pipe (Figure 3.6A-52(b)).

    c. For longitudinal breaks, circular jet shapes identical to circumferential breaks are assumed, and the jet issues from the break opening with its centerline normal to the opening areas and the pipe centerline (Figure 3.6A-52(c)).

4. The total jet force on any cross-section is assumed distance-invariant, with a total magnitude equivalent to the jet thrust force defined in item 1. The jet pressure is uniformly distributed across the cross-section of the jet.

    The proportion of the total jet force acting on the target is determined from the fraction of the jet intercepted and by the shape factor of the target. For a target with a flat surface area normal to the center of the jet stream, the impingement load is the product of the pressure and the intercepted jet area. For cases where the target area is such that the intercepted jet stream is deflected rather than totally stopped, a shape factor which is a function of the target geometry is used in calculating the total jet impingement load. Shape factors are calculated according to the method described in ANSI/ANS-58.2-1980[14].

5. All potential targets in the jet paths are identified and examined for jet impingement effects. Targets are classified in five categories:

    a. Structural targets

    b. Fluid piping targets

    c. Control system and instrumentation targets

    d. Electrical system targets

    e. Equipment targets

    All essential targets are identified for further evaluation. Nonessential targets in the jet paths are excluded from further consideration unless jet damage to them could initiate or escalate failure of an essential target.

6. Jet impingement loadings on affected targets are evaluated to determine whether:

    a. Structural integrity or operability, if required, can be demonstrated.

    b. Loss of function is acceptable, considering all target damages due to each jet in conjunction with the loss of offsite power and the postulated worst single active failure.

    When jet impingement loads on a target lead to an unacceptable consequence, protective measures described in Section 3.6.1.3.2A are instituted.

    Since the jet impingement force is a dynamically applied load, the target is analyzed either by static methods using an appropriate dynamic load factor, or dynamically using elastic or inelastic structural response codes (Appendix 3A). The load combinations and design allowables are given in Sections 3.8 and 3.9A. The effects of jet impingement are discussed in detail in Appendix 3C.

3.6.2.3.2A Pipe Rupture Restraints

Two basic restraint types are used, elastic and energy-absorbing. The elastic restraints are generally used where displacements subsequent to a postulated pipe rupture must be minimized to either restrict the break opening area or limit loads in the broken piping run. Energy-absorbing restraints are used where the primary objective is to dissipate the energy of a ruptured pipe.

Pipe rupture restraints which also support piping are designed to meet the requirements of ASME III criteria for pipe supports (Section 3.9).

Elastic portions of pipe rupture restraints and intermediate structures (auxiliary steel) not governed by ASME III, Subsection NF, are designed in accordance with the loads, loading combinations, and stress limits outlined in Section 3.8.4.3.3 for steel structures, with the exception that for the abnormal/extreme environmental condition (combinations 21 and 21.1), the allowable stress is 1.7S.

Only a portion of any pipe rupture restraint is permitted to strain beyond the elastic limit (i.e., energy-absorbing restraints, which are described in Section 3.6.2.3.2.2A). Design of the pipe rupture restraints and supporting structures as described above includes the forces induced by SSE. Therefore, failure can not occur during a seismic event.

Postulation of pipe ruptures and consequences such as the whipping of an unrestrained pipe, dictate the design criteria used to determine the location of pipe rupture restraints. Pipe rupture restraints are located sufficiently close to the break to prevent the ruptured process pipe from being hinged about the restraint. For circumferential breaks at elbows, the maximum distance between the break and the restraint is limited to $\ell = 0.8\ Mu/F_b$ where Mu is the ultimate moment capacity of the pipe and F_b is the peak blowdown force. Other values may be used if dynamic analysis indicates that uncontrolled motion does not occur.

3.6.2.3.2.1A Elastic Restraints

Since elastic restraints are used to minimize displacements of the broken pipe, they are close-gapped. For some applications, this requires that they contact the pipe during conditions other than a postulated rupture, in which case they are designed as a pipe support. If an elastic restraint only contacts the pipe following a rupture, it is designed according to the criteria for structural steel (Section 3.8.3).

3.6.2.3.2.2A Energy-Absorbing Restraints

Several approaches are used for energy absorption in pipe rupture restraints. In tension, stainless steel studs or straps are used, with a design limit of 50 percent of uniform ultimate strain. In compression, honeycomb panels or pipe are used. Compressive components are designed to 80 percent or less of their energy absorption capacity. Other energy-absorbing devices that may be used are designed to these limits.

One or more of the above energy-absorbing mechanisms are utilized in each of the typical restraints described below. When a single energy-absorbing mechanism is utilized, the design limits are met for the design range of loading directions.

Elastic components of energy-absorbing restraints are designed to the criteria for structural steel (Section 3.8).

Pipe Crush Bumper

The pipe crush bumper is a unidirectional restraint which absorbs impact energy in a direction toward the supporting structure. The energy absorber is a length of pipe placed normal to the axis of the process pipe. Subsequent to a rupture, the bumper pipe is crushed between its support structure and the moving process pipe. This absorbs energy and forms a retaining recess in the bumper pipe. The retaining recess is not intended to restrain lateral movement of the process pipe. The bumper pipe is attached to its support by welding, bolting, etc (Fig. 3.6A-7 and 3.6A-8).

Laminated Strap Restraint

The laminated strap restraint is capable of absorbing impact loads in the outward direction from the supporting structure (Fig. 3.6A-9). The energy-absorbing component is a U-shaped strap which consists of multiple strips (depending on energy to be absorbed) of highly ductile material (Type 304 stainless steel).

This laminated design results in great flexibility. If the process pipe contacts the sides of the restraint during an event other than pipe rupture, only negligible loads are transmitted. The design also minimizes bending strains, permitting the strap to act mainly as a membrane during the rupture event.

Omnidirectional Restraint

The omnidirectional restraint is capable of absorbing impact loads applied in any direction in the plane of the restraint (Fig. 3.6A-10). This restraint consists of a base weldment, an arch, ductile holddown studs on each side of the base weldment, and a honeycomb panel. The primary function of the studs is to absorb energy from impact loads acting outward from the support structure. The honeycomb panel absorbs energy from impact loads acting in an inward direction. Side load impacts are absorbed by the combined action of the studs and honeycomb. A limit stop (Fig. 3.6A-11) is a restraint whose design is a special case of the omnidirectional restraint. This restraint is designed to absorb energy from the impact load in the inward impact direction only.

The methods employed for the design of the omnidirectional restraint are as follows:

1. Approximate dimensions for the initial layout of the standard concept, bolt, and honeycomb are determined by dynamic analysis using the LIMITA computer program.

2. Restraint arch sizing is performed by elastic-plastic static analysis using the LIMITA program.

3. Final dimensions of the restraint are verified by a stress analysis design calculation, including shear blocks.

The LIMITA program is described in Appendix 3A.

Combinations of pipe crush bumpers and laminated straps may also be used to achieve energy absorption over a range of impact directions up to a full 360 deg.

3.6.2.4A Guard Pipe Assembly Design Criteria

Piping which normally carries high-energy fluid and passes through both the drywell and the containment are provided with guard pipes in accordance with NRC Branch Technical Positions MEB 3-1, Paragraph B.1.b(6).

Although these lines are designed to comply with the "break exclusion requirements" described in Section 3.6.2.1.5A, through-wall leakage cracks are postulated (Section 3.6.2.1.6.3A). The crack requirement resulted from meetings with the ACRS and NRC (AEC at that time) prior to issuance of the construction permit.

The guard pipes protect the containment from being overpressurized by such a crack, and vent the fluid discharge back into the drywell.

Restraints were provided outside the break exclusion region as required by pipe break analysis. The detail and design criteria of the guard pipe assemblies are given in Section 3.8.2 and Appendix 3D.

3.6.2.5A Material to be Submitted for the Operating License Review

Pipe break and crack locations are obtained in accordance with the criteria of Section 3.6.2.1A. High-energy piping with break locations identified are provided in isometric drawings, Fig. 3.6A-12 through 3.6A-33. High-energy piping composites, Fig. 3.6A-34 through 3.6A-49, have been provided to show graphically the pipes in relation to the rooms, equipment, and other piping. The stress results which are utilized to determine the break types and locations are given in Tables 3.6A-1 through 3.6A-20. Break types are also shown (i.e., circumferential or longitudinal). If there are changes in the pipe routing, restraint locations, or stress analysis as a result of modification, the figures and tables will be updated only when those changes significantly affect the pipe break evaluation. Cumulative usage factors are limited to less than 1.0 in these analyses.

The augmented inservice inspection plan is discussed in Section 6.6. Pipe whip restraints are designed as discussed in Section 3.6.2.3.1.2A. The restraint locations and orientation are shown in Fig. 3.6A-12 through 3.6A-33. (These same figures depict the break locations.) Jet thrust and impingement forces are determined in accordance with Sections 3.6.2.3A.

The effects of breaks and cracks are discussed in detail in Appendix 3C. The results of this Appendix are based on the protection evaluation criteria of Section 3.6.1A. Any protective measures to assure a safe shutdown (i.e., barriers, separation, and restraints) are also discussed.

The guard pipe design details are given in Section 3.8.2 and Appendix 3D.

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13. Letter from J. F. O'Leary, July 12, 1973, and attachment entitled, Criteria for Determination of Postulated Break and Leakage Locations in High and Moderate Energy Fluid Piping Systems Outside of Containment Structures, Appendix C to BTP APCSB 3-1, attached to Standard Review Plan 3.6.1, November 1975.

14. ANSI/ANS-58.2-1980, Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Ruptures.

3.6B PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH POSTULATED RUPTURE OF PIPING (GE Scope of Supply)

See Section 3.6 for explanation of GE/SWEC scope of supply.

3.6.1B Postulated Piping Failures in Fluid Systems Outside of Containment

See Section 3.6.lA.

3.6.2B Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Information concerning break and crack location criteria and methods of analysis is presented in this portion of the Safety Analysis Report. The location criteria and methods of analysis are needed to evaluate the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping inside and outside of primary containment. This information confirms that the requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe break have been met.

3.6.2.1B Criteria Used to Define Break and Crack Location and Configuration

The following section establishes the criteria for the location and configuration of postulated breaks and cracks.

3.6.2.1.LB Criteria for Main Steam Piping System - Inside Containment

See Section 3.6.2.1.5.lA.

3.6.2.1.2B Criteria for Recirculation Piping System - Inside Containment

3.6.2.1.2.lB Definition of High-Energy Fluid System

High-energy fluid systems are defined to be those systems, or portions of systems, that during normal plant conditions*

*Normal plant conditions are defined as the plant operating conditions during reactor startup, power plant operation, reactor cold shutdown, but excluding test modes.

are either in operation or are maintained pressurized under conditions where either or both of the following are met:

Maximum operating temperature exceeds 200'F, or

Maximum operating pressure exceeds 275 psig.

3.6.2.1.2.2B Definition of Moderate-Energy Fluid System

Moderate-energy fluid systems are defined to be those systems, or portions of systems, that during normal plant conditions are either in operation or are maintained pressurizedunder conditions where either or both of the following are met:

Maximum operating temperature is 200'F or less, and

Maximum operating pressure is 275 psig or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function, but for the major operational period qualify as moderate-energy fluid systems. An operational period is considered "short" if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid system is less than 2 percent of total operating time the system is designed for. (There is no moderate-energy pipe in the GE scope of supply.)

3.6.2.1.2.3B Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden, gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or as development of a sudden longitudinal, uncontrolled crack (longitudinal split) and is postulated for high-energy fluid system only. For moderate-energy fluid systems, pipe breaks are confined to postulation of controlled cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe.

The following high-energy piping systems (or portions of systems) are considered as potential initiators of a postulated pipe break during normal plant conditions, and are analyzed for potential damage dynamic effects:

1. All piping which is part of the reactor coolant pressure boundary and subject to reactor pressure continuously during station operation;

2. All piping which is beyond the second isolation valve but which is subject to reactor pressure continuously during station operation;

3. In addition to Piping under 1 and 2, all other piping systems or portions of piping systems considered high-energy systems.

Portions of piping systems that are isolated from the source of the high-energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This would include portions of piping systems beyond a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

A high-energy piping system break is not postulated simultaneous with a moderate-energy piping system crack nor is any pipe break or crack outside containment postulated concurrently with a postulated pipe break inside containment.

The evaluations of pipe breaks and cracks are in accordance with Revision 0 of SRP 3.6.2 and Branch Technical Position MEB 3-1, November 24, 1975, the documents applicable at the time the evaluations were done. Therefore, high energy leakage cracks were not considered.

High energy line leakage cracks would not be postulated in fluid system piping located in containment penetration areas, since the design stress and fatigue limits specified in BTP MEB 3-1, Section B.l.b, are met for high energy piping in these areas (see Section 3.6.2.1.5.2A).

The effects of high energy line leakage cracks in other areas would generally be bounded by the analyses performed for high energy line breaks, moderate energy line cracks, and inadvertent fire suppression system actuation.

3.6.2.1.2.4B Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip need not be provided if any one of the following conditions exist:

1. Piping which is classified as moderate-energy piping.

2. Following a single postulated pipe break, piping for which the unrestrained movement of either end of the ruptured pipe in any feasible direction about a plastic hinge, formed within the piping, cannot impact any structure, system, or component important to safety.

3. Piping for which the internal energy level associated with whipping is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. Any line restrictions (e.g., flow limiters) between the pressure source and break location, and the effects of either a single-ended or double-ended flow condition are accounted for, in the determination of the internal fluid energy level associated with the postulated pipe break reaction. The energy level in a whipping pipe is considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

3.6.2.1.2.2B Location for Postulated Pipe Breaks

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Postulated pipe break locations are selected in accordance with the intent of Regulatory Guide 1.46, the NRC Branch Technical Position APCSB 3-1, Appendix B and as expanded in NRC Branch Technical Position MEB 3-1. Postulation of arbitrary intermediate line breaks is no longer re~quired. Relaxation in arbitrary intermediate breaks is provided in NRC Generic Letter 87-11. Previously postulated arbitrary intermediate breaks and their affects may be deleted. For ASME Section III, Class 1 piping systems which are classified as high energy, the postulated break locations are:

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1. The terminal ends* of the pressurized portions of the run.

2. At intermediate locations-between the terminal ends where the maximum stress range between any two load sets (including zero load set) according to Subarticle NB-3600 ASME Code Section III for upset

*Terminal ends are extremities of piping runs that connect to structures, equipment, or pipe anchors that are assumed to act as rigid constraints to free thermal expansion of piping. A branch connection to a main piping run is a terminal end for a branch run, except when the branch and main run is modeled as a common piping system during the piping stress analysis.

plant conditions and an independent OBE event transient, exceeds the following:

a. If the stress range calculated using Equation (10) of the Code exceeds 2.4 S_m but is not greater than 3 S_m, no breaks are postulated unless the cumulative usage factor exceeds 0.1.

b. The stress ranges, as calculated by Equations (12) or (13) of the Code, exceed 2.4 S_m or if the cumulative usage factor exceeds 0.1 when equation (10) exceeds 3 S_m .

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For ASME Section III Class 1 piping systems which contain moderate-energy fluids, through wall leakage cracks are postulated at locations that:

1. Demonstrate the adequacy of separation or other means of protection, from required structures, systems, and components.

2. Through wall leakage cracks are postulated in moderate-energy fluid system piping located within structures and compartments containing required systems and components. The cracks are postulated to occur individually at locations appropriate to form the basis for providing required protection from the hazards of fluid spraying, flooding, pressurization, and other environmental conditions.

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3. Moderate-energy fluid system piping or portions thereof that are located within a compartment or confined area containing a postulated break in high-energy fluid system piping are acceptable without postulation of throughwall leakage cracks except where a postulated leakage crack in the moderate-energy fluid system piping results in more severe environmental conditions than the break in the proximate high-energy fluid piping system, in which case the provisions of Paragraph 2. above will be applied.

Criteria for break locations in ASME Section III Class 1 piping systems in the area of the containment isolation valves is provided in Section 3.6.2.1.3B.

GE-supplied NSSS analysis, design, and/or equipment utilized in this facility is in compliance with the intent of Regulatory Guide 1.46 through the incorporation of the following alternate approach.

Regulatory Guide 1.46 describes an acceptable basis for selecting the design locations and orientations of postulated breaks in fluid systems piping within the reactor containment and for determining the measure that should be taken for restraint against pipe whipping that may result from such breaks.

The design of the containments structure, component arrangement, Class 1 pipe runs, pipe whip restraints, and compartmentalization was done in consonance with the acknowledgement of protection against dynamic effects associated with postulated rupture of piping. Analytically sized and positioned pipe whip restraints were engineered to preclude damage based on the pipe break evaluation. Pipe whip requirements for fluid systems piping within the primary containment that, under normal operation, has service temperatures higher than 200°F, or pressures higher than 275 psig, complied with ANS-N176, Design Basis for Protection Against Pipe Whip, and Regulatory Guide 1.46 except as delineated in the following criteria for no breaks in Class 1 piping:

1. If Equation 10 of NB-3653-1, ASME Code III results in $S \leq 5\ 2.4\ S_m$ for ferrite or austenitic steels, no other requirements need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and an OBE event transient.

2. If Equation 10 results in 2.4 $S_m < S \leq 3.0 S_m$ for ferrite or austenitic steels, the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6, must be <0.1.

3. If Equation 10 results in $S > 3.0 S_m$ for ferrite or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653.6 must be less than 2.4 S_m and the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6 must be $\leq$ 0.1.

3.6.2.1.2.6B Types of Breaks to be Postulated in Fluid Systems Piping

The following types of breaks are postulated in high-energy fluid system piping:

1. No breaks need he postulated in piping having a nominal diameter less than or equal to 1 in.

2. Circumferential breaks are postulated only in piping exceeding a 1 in nominal pipe diameter.

3. Longitudinal splits are postulated only in piping having a nominal diameter, equal to or greater than 4 in.

4. Circumferential breaks are to be assumed at all terminal ends and at intermediate locations identified by the criteria in Section 3.6.2.1.2.5B for Class 1 piping systems. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Section 3.6.2.1.2.5B for Class 1 piping systems, either a circumferential or a longitudinal break, or both, are postulated per the following:

   a. Circumferential breaks are postulated at fitting joints, and

   b. Longitudinal breaks are postulated in the center of the fitting at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping and produces out-of-plane bending.

c. Consideration is given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location may be used to identify the most probable type of break.

d. At intermediate locations chosen to satisfy the minimum break location criteria, only circumferential breaks are postulated.

5. For design purposes, a longitudinal break area is assumed to be the equivalent of one circumferential pipe area.

6. For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibilities, pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and outof-plane for longitudinal breaks, and to cause pipe movement in the direction of the jet reaction.

7. For a circumferential break, the dynamic force of the jet discharge at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Justifiable line restrictions, flow limiters, and the absence of energy reservoirs are used, as applicable, in the reduction of the jet discharge.

3.6.2.1.3B Criteria for Main Steam Piping System in Area of Containment Isolation Valves

See Section 3.6.2.1.5.2A.

3.6.2.1.4B Guard Pipe Design

See Section 3.6.2.4A.

3.6.2.2B Analytical Methods to Define Blowdown Forcing Functions and Response Models

3.6.2.2.1B Analytical Methods to Define Blowdown Forcing Functions

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces which can dynamically excite the piping system~. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for various piping systems are presented in the following sections.

3.6.2.2.1.LB Main Steam Piping System - Inside Containment

See Section 3.6.2.2A.

3.6.2.2.1.2B Recirculation Piping System - Inside Containment

The criteria that should be used for calculation of fluid blowdown forcing functions includes:

1. Circumferential breaks should be assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by the inelastic pipe whip analysis (Section 3.6.2.2.2B).

2. The dynamic force of the jet discharge at the break location should be based on the effective crosssectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

3. All breaks are assumed to attain full area instantaneously. A rise time not exceeding one millisecond is used for the initial pulse.

Blowdown forcing functions are determined by either of two methods given in 1 and 2 below:

1. The predicted blowdown forces on pipes fed by a pressure vessel can be described by transient and steady-state forcing functions. The forcing functions used are based on methods described in Reference 1. These may be simply described as follows:

a. The transient forcing functions at points along the pipe result from the propagation of waves (wave thrust) along the pipe, and from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

b. The waves cause various sections of the pipe to be loaded with time-dependent forces. it is assumed that the pipe is one-dimensional, in that there is no attenuation or reflection of the pressure waves at bends, elbows, and the like. Following the rupture, a decompression wave is assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections occur at the break end, changes in direction of piping, and the pressure vessel until a steady flow condition is established. Vessel and free space conditions are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the pipe break.

c. The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (P_o) times the break area (A). After the initial decompression period (i.e., the time it takes for a wave to reach the first change in direction), the force is assumed to drop off to the value of the blowdown thrust (i.e., 0.7 P_oA).

d. Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the blowdown force on the pipe using the following equation:

$$F = \left[(P - P_a) + \frac{\rho u^2}{g_c} \right] A$$

where:

F = Blowdown force

P = Pressure at exit plane

P_a = Ambient pressure

u = Velocity at exit plane

❐ = Density at exit plane

A = Area of break

g_c = Newton's constant

e. Following the transient period a steady-state period is assumed to exist. Steady-state blowdown forces are calculated including frictional effects. ANS-58.2 is the base document used for determining steady state thrust coefficients (C_T) in evaluating the dynamic force ($C_T P_O A$) due to jet discharge[9]. For frictionless flow, the theoretical maximum values of thrust coefficients are 1.26 and 2.0 for saturated steam and subcooled water, respectively. The justification is discussed below.

(1) <u>Saturated and Superheated Steam</u>

Saturated or superheated steam is treated as an ideal gas with a ratio of specific heat equal to 1.3. Considering the flow to be isentropic, the thrust coefficient for frictionless flow is given by[6]:

$$C_T = 1.26 - P_a/P_O$$

where:

P_a = ambient pressure around pipe

P_o = pressure in the pipe

C_T = thrust coefficient

Since $P_a << P_o$, $C_T \cong - 1.26$

(2) <u>Subcooled Water</u>

The thrust coefficient for frictionless flow of subcooled water based on the Henrey-Fauske model is given by('):

$$C_T = 3.0 - 0.861h^{*2}; \; 0 \leq h^* \leq 0.75,$$

$$CT = 3.22 - 3.0h^* + 0.97h^{*2};\ 0.75 < h^* \leq 1.0$$

where:

$h^* = (h_o - 180)/(h_{saturated} - 180)$

ho = stagnation enthalpy (Btu/lbm)

$h_{saturated}$ = saturated water enthalpy at the stagnation pressure (Btu/lbm)

This model was confirmed by the experimental comparison work of Hanson[8]. For all values of h*, C_T is no greater than 2.0. For conservatism, C_T = 2.0 is used for the recirculation line break throughout the enthalpy range.

If C_T <1.26 or C_T <2.0 is ever used for Case (1) or (2), respectively, detailed evaluation of the above equations will be provided. .

2. The following is an alternate method for calculating blowdown forcing functions.

The computer codes RELAP 3 and RELAP 4 are used to obtain exit plane thermodynamic states for postulated ruptures[2]. Specifically, RELAP 3 supplies exit pressure, specific volume, and mass rate. From these data the blowdown reaction load is calculated using the following relation:

$$\frac{T}{A_E} = P_E - P_\infty + \frac{G^2{}_E \overline{V}_E}{g_c}$$

$$R = -\frac{T}{A_E} \times A_E$$

where:

T/A	=	Thrust per unit break area (lbf/ft^2)
P_E	=	Exit pressure (lbf/ft^2)
$P_{\circledcirc}$	=	Receiver pressure (lbf/ft^2)
G	=	Exit mass flux ($lbm/sec\text{-}ft^2$)
V	=	Exit specific volume (ft^3/lb)
g	=	Newton's constant (32.174 $ft\text{-}lb/lbf\text{-}sec^2$)
R	=	Reactor force on the pipe (lbf)

3.6.2.2.2B Pipe Whip Dynamic Response Analyses

The prediction of time-dependent and steady-thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from a ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads is given in Section 3.6.2.2A for main steam piping and 3.6.2.2.1.2B for the recirculation piping. A detailed discussion of analytical methods used to account for this loading is discussed below.

3.6.2.2.2.lB Main Steam Piping System - Inside Containment

See Sections 3.6.2.2.4A, 3.6.2.2.5A, 3.6.2.2.6A, and 3.6.2.2.7A.

3.6.2.2.2.2B Recirculation Piping System - Inside Containment

The criteria used for performing the pipe whip dynamic response analyses includes:

1. A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other break locations.

2. The analysis includes the dynamic response of the pipe in question, and the pipe whip restraints which transmit loading to the structures.

3. The analytical model adequately represents the mass/inertia and stiffness properties of the system.

4. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

5. Piping within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain are imposed which are similar to strain levels allowed in restraint plastic members. Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences does not result in direct damage to any essential system or component.

6. Components such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or whose failure would not further escalate the consequences of the accident are not designed to meet ASME Code imposed limits for essential components under faulted loading. However, if these components are required for safe shutdown, or serve a safety function to protect the structural integrity of an an essential component, limits to meet the Code requirements for faulted conditions and limits to ensure operability, if required, are met.

The pipe whip analysis was performed using the PDA computer program[3]. PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-dependent stress-strain relations are used for the pipe and the restraint. Similar to the popular plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these location is obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using an energy consideration and the equations are numerically integrated in small time steps to yield time-history information of the deformed pipe.

A comprehensive verification program has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods utilized. Part of this verification program included an independent analysis by Nuclear Services Corporation (NSC), under contract to the General Electric Company, of the recirculation piping system for the 1969 Standard Plant Design. The recirculation piping system was chosen for study due to its complex piping arrangement and assorted pipe sizes. The NSC analysis included elastic-plastic pipe properties, elastic-plastic restraint properties, and gaps between the restraint and pipe and is documented in Reference 4. The piping/restraint system geometry and properties and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by NSC for the restraint load-deflection curve supplied by GE. This approximation is demonstrated in Fig 3.6B-1. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint to structure loads than the GE analysis. The deflection limit used by NSC is the design deflection at one-half of the ultimate uniform strain for the GE restraint design. The restraint properties used for both analyses are provided in Table 3.6B-1.

A comparison of the NSC analysis with the PDA analysis, as presented in Table 3.6B-2, shows that PDA predicts higher loads in 15 of the 18 restraints analyzed. This is due to the NSC model including energy absorbing effects in secondary pipe elements and structural members. However, PDA predicts higher restraint deflections in 50 percent of the restraints. The higher deflections predicted by NSC for the lower loads are caused by the linear approximation used for the force-deflection curve rather than by differences in computer techniques. This comparison demonstrates that the simplified modeling system used in PDA is adequate for pipe rupture loading, restraint performance, and pipe movement predictions within the meaningful design requirements for these low probability postulated accidents.

A comprehensive test program was conducted to develop restraint properties such as the load-deflection relationship shown in Table 3.6B-1. A series of static and dynamic deformation tests of model restraints was performed in which the model restraints were scaled down from the restraints suitable for 26-in pipes. The material properties were obtained from tensile tests of bar specimens. Test results were analyzed for use in the development of an analytical model that predicts the restraint behavior when loaded by a moving piping. Tests were also performed on some full scale restraints; the results showed that restraints can adequately perform designated functions.

3.6.2.3B Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1B Jet Impingement Analyses and Effects on Safety-Related Components

The methods used to evaluate the jet effects resulting from the postulated breaks of high-energy piping are presented in Section 3.6.2.3A.

3.6.2.3.2B Pipe Whip Effects on Safety-Related Components

This section of the FSAR provides the criteria and methods used to evaluate the effects of pipe displacements on safety-related structures, system, and components following a postulated pipe rupture.

3.6.2.3.2.1B Pipe Whip Effects Following a Postulated Rupture of the Main Steam Piping - Inside Containment

See Sections 3.6.1A and 3.6.2A.

3.6.2.3.2.2B Pipe Whip Effects Following a Postulated Rupture of the Recirculation Piping System - Inside Containment

Pipe whip (displacement) effects on safety-related structures, system, and components can be placed in two categories: (1) pipe displacement effects on components (nozzles, valves, tees, etc) which are in the same piping run that the break occurred in and (2) pipe whip or controlled displacements onto external components such as building structure, other piping systems, cable trays and conduits, etc.

1. Pipe Displacement Effects on Components in Same Piping Run

    a. The criteria which are used for determining the effects of pipe displacements on the in-line components are as follows:

(1) Components such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or whose failure would not further escalate the consequences of the accident, need not be designed to meet ASME Code Section III imposed limits for essential components under faulted loading.

(2) If these components are required for safe shutdown, or serve a safety function to protect the structural integrity of an essential component, limits to meet the Code requirements for faulted conditions and limits to ensure operability, if required, are met.

b. The methods used to calculate the pipe whip loads on piping components in the same run as the postulated break are described in Section 3.6.2.2.2.lB.

3.6.2.3.3.B Loading Combinations and Design Criteria for Pipe Whip Restraints

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry load for an extremely low probability gross failure in a piping system carrying high-energy fluid. The piping integrity does not usually depend on the pipe whip restraints for any loading combination. When the piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices which serve only to control the movement of a ruptured pipe following gross failure) are subjected to once-in-a-lifetime loading. For the purpose of design, the pipe break event is considered to be a faulted plant condition and the pipe, its restraints, and structure to which the restraint is attached are analyzed and designed accordingly.

3.6.2.3.3.lB Main Steam Pipe Whip Restraints

See Section 3.6.2.3.lA.

3.6.2.3.3.2B Recirculation Piping System Pipe Whip Restraints

The pipe whip restraints designed, tested, and fabricated by GE for the recirculation loop piping utilize energy-absorbing U-rods to attenuate the kinetic energy of a ruptured pipe. A typical pipe whip restraint is shown in Fig. 3.6B-3. A principle feature of these restraints is that they are installed with several inches of annular clearance between them and the process pipe. This allows for installation of normal piping insulation and unrestricted pipe thermal movements. Select critical locations inside primary containment are also monitored during hot functional testing to provide verification of adequate clearances prior to plant operation.

The specific design objectives for the restraints are:

1. The restraints shall in no way increase the reactor coolant pressure boundary stresses by their presence during any normal mode of reactor operation or condition.

2. The restraint system shall function to stop the movement of a pipe failure (gross loss of piping integrity) without allowing damage to critical components or missile development.

3. The restraints should provide minimum hindrance to inservice inspection of the process piping.

For the purposes of design, the pipe whip restraints are designed for the following dynamic loads:

1. Blowdown thrust of the pipe section that impacts the restraint;

2. Dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust and subsequent impact on the restraint;

3. Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Section 3.6.2.2.2.2B;

4. Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

The recirculation loop pipe whip restraints are composed of several components, each of which perform a different function. These components are categorized as Types I, II, III, and IV, as described below:

Type I - Restraint Energy Absorption Members - Members that, under the influence of impacting pipes (pipe whip), absorb energy by significant plastic deformation (e.g., U-rods).

Type II - Restraint Connecting Members - Those components which form a direct link between the restraint plastic members and the structure (e.g., clevises, brackets, pins).

Type III - Restraint Connecting Member Structural Attachments - Those fasteners which provide the method of securing the restraint connecting members to the structure (e.g., weld attachments, bolts).

Type IV - Structural and Civil Components - Steel and concrete structures which ultimately must carry the restraint load (e.g., sacrificial shield, trusses).

Each of these components is typically constructed of a different material, with a different design objective in order to perform the overall design function. Therefore, the material and inspection requirements and design limits for each are somewhat different. These requirements for each component are as given below:

1. Type I Restraint Material (e.g., U-rods)

   a. Materials. All materials used to absorb energy through significant plastic deformation conform to:

      (1) ASME - Section III, Subsection NB, Boiler and Pressure Vessel Code for Class I Components; or

      (2) ASTM Specifications with consideration for brittle fracture control; or

      (3) ASME - Section III, subsection NF, Boiler and Pressure Vessel Code if applicable.

      (4) GE Material Specifications

b. Inspection. Inspection and identification of material conform to:

(1) ASME - Section III, Subsection NB, Boiler and Pressure Vessel Code for Class I components (Section NonDestructive Examination Methods); or

(2) ASTM Specifications procedures including volumetric and surface inspection; or

(3) ASME - Section III, Subsection NF, Boiler and Pressure Vessel Code if applicable.

(4) GE Methods and Acceptance Standards

c. Design Limits.

(1) Design local strain. The permanent strain in metallic ductile materials is limited to:

(a) 50 percent of the minimum actu~,al ultimate uniform strain (at the maximum stress on an engineering stress-strain curve) based on restraint material tests, or

(b) 1/2 of minimum percent elongation as specified in the applicable ASME Section III Boiler and Pressure Vessel Code or ASTM Specifications, when demonstrated to be as or more conservative than the above.

(2) Design steady-state load. The maximumrestraint load is limited to:

(a) 80 percent of the minimum calculated static ultimate restraint strength at the drywell design temperature. This strain is less than

(b) 50 percent of the ultimate uniform strain for all materials which is used for Type I components,

(3) Dynamic material mechanical properties. The material selected must exhibit tensile and impact properties which are not less than:

(a) 70 percent of the static percentelongation; or

(b) 80 percent of the statically determined minimum total energy absorption.

2. Type II Restraint Material (e.g., clevises, brackets, pins)

a. Materials. Material selection conforms to:

(1) ASTM Specifications including consideration for brittle fracture control; or

(2) ASME - Section III, Subsection NF, Boiler and Pressure Vessel Code if applicable.

(3) GE Material Specifications

b. Inspection. Inspection conforms to:

(1) ASME/ASTM requirements or process qualification and finished part surface inspection per ASTM methods; or

(2) ASME - Section III, Subsection NF, Boiler and Pressure Vessel Code, if applicable.

(3) GE Methods and Acceptance Standards

c. Design Limits. Design limits are based on the following stress limits:

(1) Primary stresses (in accordance with definitions in ASME Section III) are limited to the higher of:

(a) 70 percent of S where S = minimum ultimate strength by tests or ASTM specification.

(b) S + 1/3 (S - S) where S = minimum yield strength by test or ASTM specification; or

(2) Recommended stress limits per ASME Section III, Subsection NF for faulted conditions, if applicable.

3. Type III Restraint Material (fasteners)

   a. Materials. Fastener material conforms to ASTM, ASME, or MIL requirements.

   b. Inspection. All fasteners are inspected or certified per applicable ASTM, ASME, or MIL specifications.

   c. Design limits. Same as Type II.

4. Type III Restraint Material (welds)

   a. Materials. Weld materials for attachment to carbon steel structures are limited to low hydrogen type.

   b. Inspection. Liquid penetrant surface inspection is performed in accordance with:

      (1) ASTM Specification E165; or

      (2) AWS Structural Welding Codes, AWS~-Dl.l

   c. Design limits. Design limits are based on the following stress limits:

      (1) The maximum primary weld stress intensity (two times maximum shear stress) is limited to three times AWS or AISC building allowable weld shear stress.

   d. Procedures. Procedures and welders are qualified in accordance with the latest AWS Code for welding in building structures.

5. Type IV Restraint Material (structural and civil components)

   Material, inspection, and design requirements for the structural and civil components are provided by industry standards such as AISC, ACI, and ASME Section III Division II, along with appropriate requirements imposed for similar loading events. These components are also designed for other operational and accident loadings, seismic loadings, wind loadings, and tornado loadings.

The design basis approach of categorizing components is consistent in allowing less stringent inspection requirements for those components subject-to lower stresses. Considerable strength margins exist in Type II through IV components event to limit of load capacity (fracture) of a Type I component. Impact properties in all components are considered since brittle type failures could reduce the restraint system effectiveness.

In addition to the design considerations discussed above, strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties utilized in the design have included one or more of the following methods:

1. Code minimum or specification yield and ultimate strength values for the effected components and structures are used for both the dynamic and steady-state events, or

2. Not more than a 10 percent increase in code or specification values is used when designing components or structures for the dynamic event. Code minimum or specification yield and ultimate strength values are used for the steady-state loads, or

3. Representative or actual test data values are used in the design of components and structures, including justifiably elevated strain rate affected stress limits in excess of 10 percent, or

4. Representative or actual test data are used for any affected components(s) and the minimum code or specification values for the structures for the dynamic and the steady-state events.

3.6.2.4B Material To Be Submitted For the Operating License Review

3.6.2.4.lB Implementation of Criteria for Pipe Break and Crack Location and Orientation

3.6.2.4.1.lB Postulated Pipe Breaks in Main Steam Piping System (Including RCIC Piping) - Inside Containment

The criteria for selection of postulated pipe breaks in the main steam piping system, inside containment, is provided in Section 3.6.2.1.lA. The postulated pipe break locations and types selected in accordance with this criteria for main steam lines A-D are shown in Fig. 3.6A-12, 3.6A-13, and 3.6A-14. Conformance with this criteria is demonstrated in Tables 3.6A-1, 3.6A-2, 3.6A-3, and 3.6A-4.

For each line, no breaks have been postulated in that portion of the main steam piping between the containment isolation valves in accordance with the criteria of Section 3.6.2.1.5.2.1A, Item 2.

3.6.2.4.1.2B Postulated Pipe Breaks in Recirculation Piping System Inside Containment

The criteria for selection of postulated pipe breaks in the recirculation piping system, inside containment, is provided in Section 3.6.2.1.2B. The postulated pipe break locations and types selected in accordance with this criteria are shown in Fig. 3.6B-4. Conformance with this criteria is demonstrated in Table 3.6B-3.

3.6.2.4.2B Implementation of Special Protection Criteria

3.6.2.4.2.1B Pipe Whip Restraints for Main Steam Piping System (Including RCIC Piping) - Inside Containment

See Section 3.6.2.5A.

3.6.2.4.2.2B Pipe Whip Restraints for Recirculation Piping System - Inside Containment

The pipe whip restraints provided for this recirculation piping system are also shown in Fig. 3.6B-4. This system of restraints has also been found to prevent unrestrained pipe whip resulting from a postulated rupture at any of the identified break locations.

References - 3.6B

1. GE Spec. No. 22A2625. System Criteria and Applications for Protection Against the Dynamic Effects of Pipe Break.

2. Relap 3 - A Computer Program for Reactor Blowdown Analysis IN-1321. Reactor Technology TID-4500, June 1970.

3. GE Report NEDE-10813A. PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement (Proprietary Filing).

4. Nuclear Services Corporation Report No. GEN-02-02, Final Report Pipe Rupture Analysis of Recirculation System for 1969 Standard Plant Design.

5. Moody, F.J. Fluid Reactor and Impingement Loads. ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities. Vol 1, December 1973, p 219-262.

6. Shapiro, A. H. The Dynamics and Thermodynamics of Compressible Fluid Flow, Vol. 1, Ronald Press, NY, 1965.

7. Webb, S. W. Evaluation of Subcooled Water Thrust Forces, Nuclear Technology, Vol. 31, October 1976.

8. Hanson, G. H. Subcooled-Blowdown Forces on Reactor System Components: Calculation Method and Experimental Confirmation, Idaho Nuclear Corporation Report IN-1354, June 1970.

9. ANS-58.2 (ANSI N176), Proposed American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture.

3.7 SEISMIC DESIGN

Two inputs to Section 3.7 are provided. Section 3.7A is applicable to the seismic design applied to structures, systems, and components within the SWEC scope of supply. Section 3.7B is applicable to the seismic design of structures, systems, and components within the GE scope of supply.

3.7A SEISMIC DESIGN (SWEC SCOPE OF SUPPLY)

3.7.1A Seismic Input

3.7.1.1A Design Response Spectra

The design response spectra for horizontal ground motion for the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) are shown in Fig. 3.7A-1 and 3.7A-2, respectively, and for vertical ground motion for the SSE and OBE are shown in Fig. 3.7A-3 and 3.7A-4, respectively. These curves are in accordance with Regulatory Guide 1.60 and studies by N. M. Newmark, et al[1].

Maximum ground acceleration for both horizontal and vertical motion for SSE is 0.10g, and for OBE is 0.05g in accordance with Section 2.5.2.

3.7.1.2A Design Time History

The synthesized acceleration time histories for the SSE case are shown in Fig. 3.7A-5 through 3.7A-7. The synthesized time history accelerograms for the two orthogonal horizontal directions are shown in Fig. 3.7A-5 and 3.7A-6. The synthesized time history accelerogram for the vertical direction is shown in Fig. 3.7A-7. These accelerograms are normalized to 0.10g for the SSE case and 0.05g for the OBE case. These statistically independent artificial motions are generated by matching the design ground response spectra described in Section 3.7.1.1A for several specified percentages of critical damping at 250 periods distributed logarithmically from 0.02 sec (50 Hz) to 5.0 sec (0.2 Hz). The duration of the synthesized time history is 15.03 sec with a uniform time digitization interval of 0.01 sec. Fig. 3.7A-8 through 3.7A-19 show the horizontal spectra for several values of percentage of critical damping derived from the synthesized horizontal time history. They are plotted against the corresponding smooth design response spectra for horizontal SSE. Fig. 3.7A-20 through 3.7A-25 show the vertical spectra derived from the synthesized vertical time history. They are plotted against the corresponding smooth design response spectra for vertical SSE.

To demonstrate the adequacy of the frequency interval used to calculate the spectra from the design time histories, the spectrum for one case (i.e., ARS-East-West SSE for 0.5 percent damping) was recalculated using three times the number of frequencies, thus decreasing the frequency interval by a factor of three. The results are shown in Figure 3.7A-33. It can be seen from this figure that the difference is small compared to the response spectra shown in Figure 3.7A-9.

3.7.1.3A Critical Damping Values

The percentages of critical damping values assigned for various structural elements are presented in Table 3.7A-1. The subgrade component damping ratios are taken as 10 percent of critical damping for translation and rotation for both the OBE and SSE. The damping ratio in any mode, however, is limited to a maximum value of 10 percent. The damping values assigned to Seismic Category I subsystems and components are given in Section 3.7.3.15A.

3.7.1.4A Supporting Media for Seismic Category I Structures

As described in Section 2.5.4.5, all Seismic Category I structures are founded on dense, compacted, granular fill overlying dense, buried channel sands and gravelly sands and hard tertiary clays. The sedimentary deposits overlie bedrock which is at a depth of approximately 27,000 ft (Section 2.5.1.2). Profiles showing the soil stratigraphy for the site are presented in Fig. 2.5-25 through 2.5-30. The density of the plant backfill is discussed in Section 2.5.4.5. The shear wave velocities and the shear moduli for the supporting soils are presented in Sections 2.5.4.4 and 2.5.4.7. The founding elevations and dimensions of the Seismic Category I structures are shown in Table 2.5-17.

3.7.2A Seismic System Analysis

This section applies to the design of Seismic Category I structures as well as the radwaste building and turbine building, which are discussed in Sections 3.7.2.16A and 3.7.2.17A, respectively. Seismic Category I subsystems are described in Section 3.7.3A.

3.7.2.1A Seismic Analysis Methods

3.7.2.1.1A Seismic Analysis of Structures

3.7.2.1.1.1A Method of Analysis

The structural responses of the reactor building and other Seismic Category I structures to the application of horizontal and vertical earthquake ground motions are determined by the response spectra modal analysis method. Seismic responses for all Seismic Category I structures are determined from an application of two orthogonal horizontal and one vertical earthquake ground motions, assumed to be acting simultaneously. The earthquake ground motions are established in the form of response spectra for the SSE and OBE as described in Section 3.7.1A. The combination of design loading conditions with seismic loading and the allowable stress levels are given in Section 3.8.

3.7.2.1.1.2A Criteria Used in Modeling Structures

The dynamic models of Seismic Category I structures consist of systems of generalized lumped masses, each with six degrees of freedom, connected by massless, linearly elastic springs. The system is connected to the subgrade by springs derived from the soil properties. Horizontal, vertical, rocking, and torsional spring constants are included to represent the subgrade. The number and location of the lumped masses in the analytical model are chosen so as to obtain a satisfactory representation of the dynamic behavior of the actual structure. In general, the lumped masses consist of the masses of the floors, walls, columns, equipment, and piping concentrated in the vicinity of the lumped mass location. The locations of these lumped masses are generally at points where there is a concentration of mass (e.g. floor elevations), or at points where there is a special interest in the response. For structures which have a continuous mass distribution, such as the containment shell in the reactor building, a sufficient number of points are chosen to adequately represent the dynamic behavior. This is determined by first representing the structure by a number of lumped masses and springs. The number of mass points is then increased until additional mass points do not appreciably change the dynamic characteristics of the model.

To demonstrate less than 10 percent increase in response for additional mass points, the shield building wall, representing a continuous mass distribution, was evaluated using an eight and then a nine lumped mass model. The study indicated less than a 2 percent increase in the shield building wall response for the nine versus eight lumped mass model.

The analytical approach described herein yields conservative results with respect to modeling the soil as a finite element mesh. Modeling the soil by constant impedence parameters adequately represents soil-structure interaction effects. Stiffness of the springs used is equivalent to that in Reference 8. The damping values in Section 3.7.2.15A and Table 3.7A-1 are much lower than those in Reference 8 which leads to conservative results.

In the modeling of structures, the following criteria are used to determine whether separate models for equipment, components, or systems are to be included in the structural dynamic model:

1. If $R_m<0.01$, decoupling can be done for any R_f

2. If $0.01 \leq R_m \leq 0.1$, decoupling can be done when $0.8 \geq R_f \geq 1.25$

3. If $R_m > 0.1$, an approximate model of the subsystem is included in the primary system model.

where:

$$R_m = \frac{\text{Total mass of the supported subsystem}}{\text{Mass that supports the subsystem}}$$

$$R_f = \frac{\text{Fundamental frequency of the supported subsystem}}{\text{Frequency of the dominant support motion}}$$

3.7.2.1.1.3A Description of Mathematical Models for Structures

The dynamic model of the reactor building (Fig. 3.7A-26) consists of a system of spring-connected lumped masses coupled to the subgrade by springs which represent the stiffness of the soil. This multiple-degree-of-freedom model is used to establish the free undamped vibrational characteristics of the reactor building.

Fig. 3.7A-26 depicts the dynamic model of the reactor building with 33 lumped mass points and 33 equivalent springs. Masses M3 through M9 represent the shield building; M11 through M16 represent the steel containment; M17 and M18 represent the reactor pedestal; M19 through M21 represent the primary shield wall; M22 through M27 represent the drywell structure; M29 through M33 represent the reactor pressure vessel (RPV); M10 and M34 represent the combined steel containment, shield building, and fill concrete; M1 and M28 represent the base mat. The mat is modeled with two points, one at the bottom and one at the top, connected by rigid member K35. Members K2 through K26 and K36 represent the stiffness of the walls between the two elevations. These spring stiffnesses are determined from beam theory, which takes into account axial deformation, torsion, flexure, and shear. Members K28 through K32 represent the stiffness of the reactor pressure vessel and internals and member K27 represents the stiffness of the refueling bellows seal. Translational rocking and torsional soil springs are connected to the bottom of the mat (M28) to simulate the subgrade. These springs are evaluated as follows:

Translational	$= \dfrac{32(1-U)GR}{7-8U}$	Bycroft, 1956 (in Whitman)[2]
Rocking	$= \dfrac{8GR^3}{3(1-U)}$	Borowicka, 1943 (in Whitman)[2]
Vertical	$= \dfrac{4GR}{1-U}$	Timoshenko and Goodier, 1951[3]
Torsion	$= \dfrac{16GR^3}{3}$	Reissner and Sagoci, 1944 (in Whitman)[2]

where:

G = Shear modulus of subgrade

R = Radius of foundation mat

U = Poisson's ratio of the subgrade

A discussion of the detailed spring-connected lumped mass model of the RPV (Fig. 3.7B-2) can be found in Section 3.7B. A simplified dynamic model of the RPV is combined with the dynamic model of the reactor building to form a dynamic model (Fig. 3.7A-26) which exhibits soil-structure-reactor interaction.

Other Seismic Category I structures (e.g., fuel building, auxiliary building) listed in Section 3.8 are modeled in a similar fashion to the reactor building; that is, the models consist of systems of generalized spring-connected lumped masses coupled to the subgrade by springs derived from the soil stiffness. These models are shown on Fig. 3.7A-28 to 3.7A-32.

The seismic motion of all Seismic Category I structures is determined by applying the earthquake ground motions at the base of the appropriate dynamic model. In general, interaction between Seismic Category I and non-Seismic Category I structures is eliminated by providing separate foundations for the structures. Also, rattlespace between abutting buildings is provided so that seismic motion between buildings is unimpeded. In general, the periphery of this rattlespace between buildings is sealed off with compressible material to prevent extraneous material from entering this space.

A tabulation of the rattlespaces surrounding Seismic Category I structures is shown in Table 3.7A-1a. To determine the relative deflection between structures the following equation is used:

$$\Delta_R = \sqrt{|Xa + Xb|^2 + Xo^2}$$

where:

Δ_R = Relative deflection between structures

X_a = Deflection of structure a relative to base

X_b = Deflection of structure b relative to base

X_o = Orbital motion between points a and b

A tabulation of relative deflections for an SSE event is incorporated in Table 3.7A-1a. See Fig. 1.2-2 for the arrangement of plant structures. As can be seen from the tabulation, the cumulative deflection (displacement) under an SSE event does not exceed the rattlespace provided in each case.

Where non-Seismic Category I structures are attached to, or influence, Seismic Category I structures, the effects are analyzed by including the influence of the non-Seismic Category I structure in the seismic model of the Seismic Category I structure.

3.7.2.1.1.4A Analysis of Mathematical Models for Structures

To determine the free vibrational characteristics of the dynamic models, the modal equation for a multi-degree lumped-mass system may be written in matrix notation [4]:

$$[K]_{nxn} - \omega^2{}_i[M]_{nxn} \; \{\phi_i\}_{nx1} = 0$$

where:

$[K]$ = System stiffness matrix

$[M]$ = System diagonal mass matrix

ϕ_i = Mode or characteristic shape for i^{th} mode

n = Number of dynamic degrees of freedom

ω_i = Circular natural frequency of i mode

This set of equations has as eigenvalues, the squares of the circular natural frequencies, ω^2. Associated with each frequency is a mode shape
$\{\phi i\}_{nx1}$, which may be arranged as one of the columns of the matrix $[\phi]_{nxn}$.

The modal participation factors are given by:

$$[\Gamma]_{nx1} = \frac{[\phi]^T \quad [M] \; \{D\}}{[\phi]^T \; [M] \; [\phi]}$$

where:

$[\Gamma]_{nx1}$ = Modal participation factor associated with the direction of excitation

$\{D\}$ = Direction vector for base excitation

The acceleration response of the system in one mode $\{A_i\}$ is given by:

$$\{A_i\} = \begin{Bmatrix} a_i^1 \\ a_i^2 \\ \vdots \\ a_i^n \end{Bmatrix} = \Gamma_i\{\phi_i\}R_i$$

where:

$\{A_i$ = Maximum response vector for i^{th} mode

R_i = Maximum response of a single-degree-of-freedom system of period T_i and damping ratio, β_i, from the ground response spectrum forthat direction of excitation.

At any mass coordinate in the system, the total response, A_k, is given by:

$$A^K = \sqrt{\sum_{i=1}^{n} \left(A_i^K\right)^2}$$

where:
$A^K{}_i$ = Maximum response at coordinate K in the i^{th} mode

Closely spaced modes are discussed in Section 3.7.2.7A.

3.7.2.2A Natural Frequencies and Response Loads

The first few significant fixed base natural frequencies for all Seismic Category I structures are presented in Table 3.7A-2.

Response loads (for Seismic Category I structures) which were determined by seismic analyses are shown in Tables 3.7A-3 through 3.7A-6.

Amplified response spectra (ARS) are generated for all Seismic Category I structures to define the seismic environment for the subsystem analyses. The procedure is described in Section 3.7.2.5A.

3.7.2.3A Procedure Used for Modeling

The procedure used for modeling systems is discussed in Section 3.7.2.1.1.2A.

3.7.2.4A Soil-Structure Interaction

Because of the great depth of soil beneath the site (Section 3.7.1.4A) and the relatively shallow embedment of foundation structures, the foundation mat-subgrade conditions closely approximate the case of a rigid plate on an elastic half-space.

Accordingly, structure-foundation interactions are taken into account by coupling the structural model with the supporting medium by the use of soil springs (Section 3.7.2.1.1.3A). The stiffness of the soil springs used to model the flexible supports of the foundation mats of Seismic Category I structures are based on the theory of elasticity[2]. Unit displacements applied over a region of the surface of a semi-infinite elastic half-space cause resultant forces and moments on that region equal to the stiffness of the equivalent springs used in the lumped mass model.

Properties used to arrive at the value of shear modulus and Poisson's ratio which is used to calculate stiffness values for the soil springs are derived from actual properties of the in situ soil and backfill as described in Section 2.5. Any variation in this value is accounted for by using a range of shear moduli (12, 18 and 24 ksi) for structural design and peak spreading ARS curves as described in Section 3.7.2.9A.

3.7.2.5A Development of Floor Response Spectra

ARS are defined as plots of the maximum response of a family of idealized linear single-degree-of-freedom damped oscillators as a function of period (or natural frequency) at various locations in the structure subjected to a specified acceleration time history at their support. In the analysis of subsystems which meet the requirements for decoupling (Section 3.7.2.1.1.2A), the response of the structure is independent of the properties and dynamic behavior of the subsystems. The problem can then be solved in two parts: the response of the structure due to the ground acceleration can be determined; then that response is applied as support accelerations to the subsystems. In such cases, the use of ARS methods is an acceptable approach to the problem of determining the dynamic loads on subsystems.

The time history method of analysis is used to generate the ARS for design of Seismic Category I piping and equipment. The equations of motion can be written in matrix notation:

$$[M]_{nxn}\{\ddot{U}(t)\}_{nx1} + [C]_{nxn}\{\dot{U}(t)\}_{nx1} + [K]_{nxn}\{U(t)\}_{nx1} \qquad (1)$$

$$= -[M]_{nxn}\ddot{U}_g(t)\{D\}$$

where:

M = Mass matrix for the structural system with n degrees of freedom

C = Damping matrix for the structural system with n degrees of freedom

K = Stiffness matrix for the structural system with n degrees of freedom

$U_g(t)$ = Ground acceleration in one of the three global directions

$\{D\}$ = Excitation vector consisting of zeros and ones. The zeros are associated with the degrees of freedom that are not parallel to the direction of excitation.

For an undamped system (C=O) the equation of free vibration reduces to:

$$[M]_{nxn} \{\ddot{U}(t)\}_{nx1} + [K]_{nxn}\{\dot{U}(t)\}_{nx1} = \{0\}$$

The above equation yields the square of the circular natural frequencies, ω^2, as eigenvalues and the mode shapes as eigenvectors expressed in the form of a matrix $[\phi]_{nxn}$.

Using the substitution

$$\{U(t)\}_{nx1} = [\phi]_{nxn} \{X(t)\}_{nx1}$$

and premultiplying by both sides by $[\phi]^T$, Equation 1 becomes

$$[\phi]^T [M] [\phi]\{\ddot{X}(t)\} + [\phi]^T[C] [\phi]\{\dot{X}(t)\} + [\phi]^T[K][\phi]\{X(t)\} = -[\phi]^T[M] \ddot{U}_g(t)\{D\}$$

β = Modal damping ratio

ω = Circular frequency of the mode

Normalizing $[\phi]$ so that

$[\phi]^T\ [C]\ [\phi] = [I]$ = Identity matrix

then

$$[\phi]^T\ [C]\ [\phi] = [\omega]^2$$

and equation 2 can be rewritten

$$\{\ddot{X}(t)\} + [2\beta\omega]\ \{\dot{X}(t)\} + [\omega^2]\{X(t)\} = -\ [\Gamma]\ddot{U}_g(t) \qquad (3)$$

where:

$$[\Gamma] = [\phi]^T\ [M]\ \{D$$

It is evident that Equation 3 is a set of decoupled equations of motion which can be solved for X(t) numerically. The solution for structural response is then:

$$\{U(t) = [\phi]\ \{X(t)$$

$$\{\ddot{U}(t) = [\phi]\ \{\dot{X}(t)$$

in which $\{\ddot{U}(t)$ are the time histories of acceleration for the coordinates of the dynamic structural system. Structural acceleration, $\{\ddot{U}(t)$,at a particular time step, is the algebraic sum of the response to the three directions of ground motion applied simultaneously. Having the structural accelerations $\{U(t)$, they may be applied to the supports of damped, single-degree-of-freedom systems, and the time histories of the responses determined. The maximum values of the responses produce the amplified response spectra at various locations in the stuctural system.

The ground acceleration, $\ddot{U}_g(t)$, in Equation 3 is an artificial time history as described in Section 3.7.1.2A.

For the calculation of acceleration time histories, all modes of vibration which contribute more than 1 percent of the total response are included.

The method used to calculate equivalent modal damping is as follows:[5]

$$B_{eq}^{j} = \frac{\sum_{i=1}^{N_H} D_i E_i^j}{\sum_{i=1}^{N_H} E_i^j} + \frac{\sum_{k=1}^{N_V} \frac{\omega^j}{\omega_k} B_k E_k^j}{\sum_{k=1}^{N_V} E_k^j} \qquad (4)$$

where:

B_{eq}^{j} = Equivalent viscous damping ratio (fraction of critical) for the j^{th} mode

N_H, N_V = Number of hysteretically or viscously damped elements respectively

D_i = Hysteretic damping ratio for element i

E_i^j, E_k^j = The i^{th} or k^{th} element potential energy when deformed in the j^{th} mode shape.

B_k= Viscous damping ratio (% critical) at frequency

ω^j = j^{th} mode frequency

ω_k = k^{th} element natural stuctural frequency.

The first half of the right hand side of equation (4) represents the hysteretic (Bigg's damping) portion of the equivalent damping term and the latter half represents the viscous term. The damping is calculated two ways: one by assuming all damping to be hysteretic type (Biggs), and the other by assuming soil springs (transverse and vertical) damping to be viscous while the balance to be hysteretic type[5]. Conservatively the lower value of modal damping from the two methods above is used in seismic analysis.

3.7.2.6A Three Components of Earthquake Motion

When using the response spectrum method, the effects of the three directional (two horizontal and one vertical) components of earthquake motion are considered. In order to properly account for the responses of systems subjected to a multidirectional excitation, the manner in which their effects are combined must be considered since phasing of the responses has not been realistically simulated. Consequently, a statistical combination is used to obtain the net response according to the square root of the sum of the squares (SRSS) criterion, which accounts for the randomness of magnitude and direction of earthquake motion. This SRSS criterion, considering the three components of ground motion, is used for final structural design. This procedure conforms to the guidelines of Regulatory Guide 1.92 as discussed in Section 1.8. In the case of time history analysis method, the procedure used to combine the three-dimensional earthquake response is described in Section 3.7.2.5A.

3.7.2.7A Combination of Modal Responses

When the modes of vibration for a structure are not closely spaced, the representative maximum response value of interest is obtained by the SRSS criterion. Two consecutive modes are defined as closely spaced if their frequencies differ from each other by 10 percent or less of the lower frequency. If the modes are closely spaced, the responses of these modes are combined in an absolute manner; the resulting total is treated as a pseudo-mode and then combined with the remainder of the modal responses in an SRSS fashion. This procedure conforms to the guidelines of Regulatory Guide 1.92.

Closely spaced modes are divided into groups that include all modes having frequencies lying between the lowest frequency in the group and a frequency 10 percent higher. Groups are formed starting with the lowest frequency and working toward successively higher frequencies. No one frequency is in more than one group.

The equation is:

$$R = \left[\sum_{k=1}^{N} R_k^2 + \sum_{q=1}^{P} \sum_{l=i}^{j} \sum_{m=i}^{j} \left| R_{lq} R_{mq} \right| \right]^{1/2} \qquad l \neq m$$

R = The representative maximum value of a particular response of a given element to a given component of an earthquake

R_k = The peak value of the response of the element in the k^{th} mode

N = The number of significant modes considered in the modal response combination

R_{1q}, R_{mq} = Modal responses R_1 and R_m within the qth group respectively.

P = Number of groups of closely spaced modes, excluding individual separated modes

i = the number of the mode where a group starts

j = The number of the mode where a group ends

This method is identical to the grouping method in Regulatory Guide 1.92.

River Bend Station seismic analyses utilize all modes for all Seismic Category I structures except for the reactor building. Of all reactor building modes, only the first 60 modes are selected for obtaining responses. The building is designed using all modes to generate the member forces.

The 60 modes considered for the reactor building range in frequency from about 1 Hz to 42 Hz. A ground acceleration time history and corresponding fast fourier transformer (FFT) of the time history, as shown in Figures 3.7A-34 and 3.7A-35, show little detectable fourier amplitudes in frequencies above 30 Hz. It can be seen from these figures that the contribution to the response from modes over 30 Hz is not significant.

3.7.2.8A Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

The manner in which the seismic analysis of non-Seismic Category I structures is treated depends upon their location with adjacent Seismic Category I structures. Any isolated non-Seismic Category I structure whose failure due to a seismic event would not endanger a Seismic Category I structure is designed in accordance with codes and standards applicable to non-Seismic Category I structures. When a non-Seismic Category I structure is adjacent to a Seismic Category I structure, either the consequences of failure of the non-Seismic Category I structure are considered in the design of the Seismic Category I structure or the non-Seismic Category I structure is designed so as not to collapse onto the Seismic Category I structure.

In this regard, the possibility of the failure of framing members in areas adjacent to Seismic Category I structures has been evaluated to ensure that the Seismic Category I structures are not adversely affected. Additionally, the design of the radwaste building has been verified to demonstrate that the structure does not collapse under seismic and tornadic loading conditions and impair the integrity of any adjacent Seismic Category I structure.

3.7.2.9A Effects of Parameter Variations on Floor Response Spectra

The effects of expected variations of structural properties, damping values, soil properties, and soil structure interaction on floor response spectra and on time histories are taken into account.

3.7.2.9.1A Piping

For SWEC-supplied Seismic Category I piping systems, the following methods are applied.

<u>Response Spectrum Method</u>

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The floor response spectra are peak-spread in the acceleration versus period plot in the following manner: All peaks are broadened a minimum of -20 and +18 percent. The slopes of the broadened peaks are maintained parallel to the original slopes. All peak-spreading is performed with the computer program PSPECTRA (Appendix 3A).

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This peak broadening and side sloping procedure conforms to the requirements of Regulatory Guide 1.122.

<u>Time History Method</u>

Time histories of floor motions may be used in a few cases as excitations to the subsystems. To account for the effect of possible frequency variation of the structure, the same time history data are used with three different time intervals:

$$\Delta t, \text{ and } (1 \pm \Delta fj/fj)\ \Delta t$$

where:

f_j = Dominant structural frequency in the response range of the piping system

Δf_j = Parameter defining the frequency variation due to uncertainties. $\Delta fj/fj = 0.15$ is used.

This variation of the time interval has an effect similar to widening the spectral peak. The maximum system responses to the three time histories are enveloped, and this envelope is used in the same way as the response in the spectral method.

3.7.2.9.2A Equipment

The response spectra developed in Section 3.7.2.5A form the basis for equipment seismic qualification. The floor response spectra are peak-spread with parallel slopes, by the use of PSPECTRA program (Appendix 3A), in the following manner: The peak broadening and side sloping criteria conform to the requirements of Regulatory Guide 1.122. For seismic loads, peak spreading of +25 and -20 percent is applied to all peaks.

3.7.2.9.3A Damping Considerations

Damping values are assigned to structural materials as outlined in Section 3.7.1.3A. Realistic estimates of damping, which affect the results of the dynamic analysis, are provided by the use of the concept of modal damping as discussed in Section 3.7.2.5A.

3.7.2.10A Use of Constant Vertical Static Factors

Since the seismic analysis of Seismic Category I structures considers vertical degrees of freedom, use is not made in any of the seismic analyses for Seismic Category I structures of constant vertical load factors to take into account vertical response to earthquakes.

3.7.2.11A Method Used to Account for Torsional Effects

Seismic Category I structures may have natural torsional modes of vibration due to eccentricities between the centers of rigidity and centers of mass of the structural elements. The presence of eccentricities generates coupling between translational directions of motion resulting in torsion. Therefore, a general three-dimensional model is set up, followed by a complete dynamic analysis as described previously in Section 3.7.2.1.1A. Since the three-dimensional model accounts for the torsional effects, including the effects of eccentricities between the centers of rigidity and centers of mass of the structural components, an additional eccentricity of 5 percent of the maximum building dimension is not considered in the analyses. The results of these analyses, therefore, include torsional modes.

3.7.2.12A Comparison of Responses

Since both modal response spectrum and modal time history methods are applied, the responses obtained from both methods at selected points are tabulated in Table 3.7A-7 for a check and comparison to demonstrate the approximate equivalency between the two methods.

3.7.2.13A Methods for Seismic Analysis of Dams

There are no Seismic Category I dams which directly impact River Bend Station.

3.7.2.14A Determination of Seismic Category I Structures Overturning Moments

The overturning moments induced by seismic excitation are computed by applying the inertia forces determined from the seismic analysis (Section 3.7.2.1A) with vertical inertia forces taken upward, reducing the effective weight of the structures. Tensile soil reactions are not allowed.

3.7.2.15A Analysis Procedure for Damping

Structural damping is energy loss due to internal friction within the structural material and at connections. The damping force is a function of the intensity of motion and the stress levels induced in the system. Damping is also highly dependent upon the type of structural system and the energy absorption mechanisms within the system. Considerable energy is also absorbed at cracked surfaces when the elements on each side of the crack can move relative to one another.

Seismic analysis is performed using total system damping characterized by modal damping. The modal damping value is calculated as a ratio of the sum of the energy dissipated in each component element (based upon the assigned damping ratio of each element) to the total available modal energy.

In determining the modal damping ratios, component damping values consistent with the stress intensities given in Regulatory Guide 1.61 are normally used. For example, damping for welded structures is assigned a value of 2 percent for OBE and 4 percent for SSE. The subgrade component damping values are taken as 10 percent of critical for translation and rotation in either OBE or SSE. The damping ratio in any mode, however, is limited to a maximum value of 10 percent.

•→3
For piping systems, damping values of 1 or 2 percent for OBE and 2 or 3 percent for SSE are used as specified in Regulatory Guide 1.61. However, as an alternative, the following damping characteristics developed by the Pressure Vessel Research Council Technical Committee on Piping Design may be applied to piping systems: 5 percent of critical damping up to 10 Hz natural frequency, with a linear decrease to 2 percent at 20 Hz, remaining constant above that frequency. This damping characteristic is applicable to all pipe sizes and to both OBE, SSE, and other dynamic loads for which the response spectra is generated after being filtered through the building structure. The alternate damping is based on Code Case N-411-1. (Table 3.7A-1).
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The soil damping ratio of 10 percent of critical was chosen to conservatively model the effects of soil structure interaction. Damping values for a constant (frequency-independent) impedence function to represent the soil-structure interaction effects can be estimated by equations in Reference 8. The damping values calculated by these methods are much higher than 10 percent. For example, the values which have been calculated for the standby service water cooling tower are provided below:
14←•

	Calculated[8] (%)	River Bend Station (%)
Horizontal	44	10
Rocking	19	10
Vertical	70	10
Torsional	15	10

The term "subgrade components" refers to the constant springs used to model the soil as described in Section 3.6.2.1.1.3A.

3.7.2.16A Seismic Analysis of Radwaste Building

The radwaste building is analyzed for earthquake loads in the same manner as Seismic Category I structures, using the design response spectra (as directed by Regulatory Guide 1.60) normalized to OBE level (0.05g) ground acceleration.

The structural design of the radwaste building is discussed in Section 3.8.4.4.8.

3.7.2.17A Seismic Analysis of Condensate Demineralizer, Regeneration, and Off Gas Building and Turbine Building

The turbine building complex, including the condensate demineralizer, regeneration, and off gas building, is analyzed for earthquake loads in the same manner as Seismic Category I structures, using the design response spectra (directed by Regulatory Guide 1.60) normalized to OBE level (0.05g) ground acceleration.

The structural design of the turbine building complex is discussed in Section 3.8.4.4.9.

3.7.3A Seismic Subsystem Analysis

The design of Seismic Category I subsystems (i.e. components, equipment, piping, supports) includes OBE and SSE seismic loading conditions. For the OBE loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment are required to operae within design limits. The seismic design for the SSE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is necessary to ensure that required critical systems and components do not lose their capability to perform their safety-related function. This is referred to as the no-loss-of-function criterion.

Not all critical components have the same functional requirements for safety. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practice, general elastic behavior of this structure under the SSE loading condition must be ensured. On the other hand, some systems and components may experience local permanent deformation without loss of function. Piping and vessels are examples of the latter, where the principal requirement is that they retain their contents and allow fluid flow.

System seismic classification is given in Table 3.2-1.

3.7.3.1A Seismic Analysis Methods

3.7.3.1.1A Seismic Qualification of Components

This section provides the qualification methods for equipment affected by seismic loads. The methods for the qualification of equipment affected by the suppression pool induced dynamic loads are provided in Appendix 6A, subsection 6A.17.

All Seismic Category I equipment is qualified for seismic adequacy. Depending upon equipment location, the basic source of seismic design data is either the ground response spectra or the amplified response spectra, derived through a dynamic analysis of the structure.

The four principal methods of documenting adequacy for Seismic Category I components are static analysis, dynamic analysis, dynamic testing, and static deflection testing. These methods are used singly or in combination to qualify this equipment.

3.7.3.1.1.1A Static Analysis

Static analysis is used for equipment that can be modeled as relatively simple structures. This type of analysis involves the multiplication of the component weights by the specified seismic accelerations (direction dependent loadings), to produce forces that are applied at the centers of gravity in the horizontal and vertical directions. A stress analysis of critical components, such as feet, holddown bolts, and other structural members, is performed to determine their adequacy. The deflections of critical components are also calculated and compared with specified tolerances.

In the specification of equipment for static analysis, two ranges of acceleration data are provided: a resonant range, distinguished by lower frequencies with amplified response accelerations; and a rigid range characterized by higher frequencies and essentially nonamplified response. The division between the two ranges is termed the cutoff frequency.

Selection of the appropriate range depends upon the fundamental natural frequency of the equipment. If this value is beyond the resonant range, i.e., higher than the cutoff frequency, the equipment is analyzed to rigid range response accelerations.

Equipment having a fundamental frequency in the resonant range of the ARS is analysed by using the peak resonant acceleration, increased by a static coefficient of 1.3. This factor accounts for potential multi-mode response (Section 3.7.3.5A).

Each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) are evaluated separately. The calculated results of the three analyses are superimposed using the SRSS criterion. The particular response values (e.g., acceleration, force, stress) to be combined are optional, but the option selected remains consistent throughout, following the recommendations of Regulatory Guide 1.92.

3.7.3.1.1.2A Dynamic Analysis

A detailed dynamic analysis is performed when component complexity or dynamic interaction precludes static analysis, or when static analysis is too conservative.

To describe fully the behavior of a component subjected to dynamic loads, an infinite number of coordinates would be required. Since calculation at every point of a complex model is impractical, the analysis is simplified by the selection of a limited number of mass points. The lumped mass approach is employed in which the main structure is represented in a model with masses interconnected by flexible elements. The nature of the component and the stiffness properties of the corresponding modeling elements determine the minimum spacing of the mass points and the degrees of freedom associated with each point.

In cases where some dynamic degrees of freedom do not contribute to the total response, static or kinematic condensation is employed in the analysis.

The normal mode approach is employed for dynamic analysis of components. Natural frequencies, eigenvectors, participation factors, and the required component dynamic responses such as modal member-end forces, and moments of the undamped structure are calculated. The basis for combination of modal responses is discussed in Section 3.7.2.7A.

The mathematical models used for dynamic analysis use a sufficient number of modes to assure participation of all significant modes. The criterion employed is that a sufficient number of degrees of freedom is taken equal to twice the number of modes with frequencies less than 10 cps. The cutoff frequency of 10 cps is used because there are no amplification effects above 10 cps for River Bend Station.

Vendors are requested to use documented computer programs in the public domain for performing dynamic analysis. However, if proprietary computer programs are used, qualification of the program is required.

Each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) are evaluated separately. The calculated results of the three analyses are superimposed on a SRSS basis. The particular response values (e.g.,acceleration, force, stress) to be combined are optional, but the option selected remains consistent throughout, following the recommendation of Regulatory Guide 1.92.

Maximum relative displacement among supports has been considered in the analyses. In regions where high relative displacement exists, mechanical joint releases have been employed where possible. In designs where restraint of free end displacement is necessary, stress analysis has been performed within the guidelines of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF.

When the effects from piping interactions, externally applied structural restraints, hydrodynamic loads and non-linear responses are determined to be significant, these effects are included in a dynamic model. Piping interactions are considered in the dynamic model of the diesel generator skid assembly. Hydrodynamic and stiffness effects on the spent fuel racks, which are submerged in water, are also addressed in a dynamic model. For equipment such as the polar crane and those mentioned above, stresses are kept well below the yield strength, thereby avoiding material nonlinearity.

For the polar crane, where relative motion is permissible with respect to the support, the effect of sliding, if present, is accounted for in the following manner:

> Equipment is analyzed as though fully supported, thereby allowing transfer of all base excitation energy into the equipment. This approach is conservative in determining the maximum dynamic response in that no allowance is made for energy absorbed during sliding. However, when sliding is present, an impact analysis is performed.

3.7.3.1.1.3A Testing

Equipment that is too complex to analyze or whose operability cannot be adequately demonstrated by analysis is qualified by dynamic testing. Testing methods conform to IEEE 344-1975[6], as supplemented by Regulatory Guide 1.100.

The minimum acceptance criteria for equipment adequacy are:

1. No loss of function, or ability to function, before, during, or after the proposed test
2. No structural/electrical failure (i.e., connections and anchorages) which would compromise component integrity
3. No adverse or maloperation before, during, or after the test that could result in an improper safety action.

Equipment vendors and suppliers are required to formulate programs for qualifying the equipment in accordance with the specified seismic requirements.

The base motions used to simulate the seismic loadings consist of either a single frequency or multiple frequencies and are applied either along one axis or along horizontal and vertical axes simultaneously. The choice of the input motion, i.e., frequency and axis, depends on the dynamic characteristics of the equipment and on the frequency content of the seismic loading. The criteria for selecting these specific input test motions are in accordance with IEEE 344-1975 and Regulatory Guide 1.100, Rev. 1.

Exploratory tests are run to determine the response characteristics of the equipment and to aid in selecting the method of testing. The exploratory test consists of a low level sinusoidal sweep over the frequency range of seismic loading (1 to 33 Hz). The sweep rate is 2 octaves/min or lower, to excite all the resonances. If equipment is shown to be nonresonant in the frequency range of seismic loading, it is considered a rigid body and tested accordingly. If equipment exhibits multiple resonant response, further testing programs, based on multifrequency input, are more appropriate.

3.7.3.1.1.3.1A Multifrequency Testing

Multifrequency input, applied biaxially, is the preferred method of qualification. Other methods are used as justified. Input motion for testing is applied to the vertical and one of the two principal horizontal axes simultaneously, unless it is demonstrated that the equipment response along the vertical direction is not sensitive (coupled) to the vibratory motion along the horizontal direction and vice versa. Phase incoherent (statistically independent) inputs in the vertical and horizontal directions are used to avoid purely rectilinear motion. When the test facility limitations do not allow the use of independent inputs, two tests are performed: 1) vertical and horizontal inputs in-phase, and 2) vertical and horizontal inputs 180 deg out-of-phase.

The above test is repeated with the equipment rotated 90 deg in the horizontal plane. The test setup simulates as closely as possible the actual in-service installation. Equipment is tested in the mode (i.e., energized or de-energized) that reflects its design safety function. Equipment operability is verified during and after the dynamic tests.

The basic objective of qualification or proof testing is to produce a test response spectrum (TRS) which envelops the required response spectrum (RRS). ARS, when properly broadened to account for variations in the soil and structural properties, become the RRS for qualification.

For the multifrequency input applied, the testing machine input must, as a minimum, equal the maximum floor acceleration of the RRS. The TRS is adjusted in successive test runs so that it envelops the RRS over the required frequency range. Curves for identical damping are used in comparing TRS and RRS information. Five OBE-level tests are performed prior to SSE qualification testing, following the recommendations of IEEE 344-1975.

Multifrequency testing provides broad band test input motion which produces simultaneous response from all the modes of the equipment. Multifrequency motions are derived using any of the following techniques:

Time History

This is an acceleration motion in the time domain, at the equipment mounting location, obtained from dynamic analysis of the structure.

Random Motion

This is an electrically generated random noise signal which is selectively amplified, or attenuated, in one-third or smaller frequency bandwidths. The motion resulting from this modified signal is arranged so that it envelops the TRS. This is the most commonly used input motion for multifrequency testing. The peak acceleration amplitude of this motion equals or exceeds the zero-period acceleration

(ZPA) of the RRS. The random motion signal is applied for a minimum duration of 15 sec.

Complex Wave

A complex wave is a sum of a group of decaying sinusoidal signals spaced at one-third octave or narrower frequency intervals over the frequency range of the RRS.

3.7.3.1.1.3.2A Single Frequency Testing

Following the recommendations in IEEE 344-1975, single frequency input for testing is applicable, provided one of the following conditions is met:

1. The characteristics of the required input motion indicate that the motion is dominated by one frequency i.e., by structural filtering effects).

2. The anticipated response of the equipment is adequately represented by one mode.

The objective is to produce a TRS acceleration at the test frequency which is at least equal to that given by the RRS. The test table input equals or exceeds the maximum floor acceleration of the RRS.

The single frequency test consists of an exploratory test and a dwell test. In the exploratory test, the table input motion equals or exceeds the maximum floor acceleration, i.e., the ZPA of the RRS.

Dwell testing is performed at the natural frequency identified during the exploratory test. The dwell test consists of applying a continuous sinusoidal input motion at the maximum floor acceleration for a minimum duration of 20 sec.

Dwell testing is also performed using a sine beat input instead of a continuous sine input. A sine beat consists of a continuous sinusoid at the test frequency, amplitude modulated by a sinusoid of a lower frequency.

The duration and peak amplitude of the beat for each particular test frequency are chosen to generate a magnitude of equipment response which is at least equal to that imposed by the RRS at justifiable damping levels. As a minimum, the peak amplitude of the beat should equal the rigid range acceleration of the RRS. Ten cycles per beat are used, following the recommendations of IEEE 344-1975.

3.7.3.1.1.4A Static Deflection Testing

A static deflection test consists of applying a sustained static load on critical sections of the component in such a way that the deflection caused by this load duplicates or exceeds the calculated SSE deflection. Concurrently, the component is operated in the required manner, and all applicable design loads are superimposed during the test.

3.7.3.1.2A Seismic Analysis of Piping

This is described in Section 3.7.3.8A.

3.7.3.2A Determination of Number of Earthquake Cycles

The following criteria are applied to all Seismic Category I subsystems:

1. A total of five OBE and one SSE are considered.

2. The ASME code requires no fatigue analysis for the faulted condition; therefore, stress cycling does not apply to the SSE.

3. For subsystems, except piping, 20 cycles (full sign reversals) per OBE, i.e., a total of 100 cycles, are considered.

4. For all piping systems, 10 stress cycles per OBE, i.e., a total of 50 cycles, are postulated.

5. Where time history analysis is performed, a minimum duration of 10 sec is assumed.

3.7.3.3A Procedure Used for Modeling

The procedure described in the following subsections is specifically written for piping systems. Other subsystems are discussed in Section 3.7.3.1.1A.

3.7.3.3.1A Summary

Portions of piping systems which are bounded by anchors or equipment are statically and dynamically independent from the remainder of piping. Generally, a piping system consists of several such subsystems. The analytical model and its geometric boundaries are described in detail in the following sections.

3.7.3.3.2A Geometrical Boundaries of Analytical Models

For the purpose of analysis, the piping systems are broken down into smaller units (in the context of analysis called problems) which are bounded by structural anchors (six-degree-of-freedom constraints) or by other virtually rigid points such as equipment, penetrations, and piping of much larger diameter.

A branch line with a moment of inertia of 1/10 or less of the run pipe may be ignored in the model. However, if the branch line needs to be analyzed, its model includes the effect of the run pipe.

Where Seismic Category I piping is connected to nonseismic piping, a portion of the nonseismic piping is included in the analytical model up to the first anchor (Section 3.7.3.13A).

3.7.3.3.3A Model

The basic method of analysis used in NUPIPE (Appendix 3A) is the finite element stiffness method. In accordance with this method, the continuous piping is mathematically idealized as an assembly of elastic structural members connecting discrete nodal points. Nodal points are placed in such a manner as to isolate particular types of piping elements such as straight runs of pipe, elbows, valves, etc, for which force-deformation characteristics can be categorized. Nodal points are also placed at all discontinuities such as piping supports, concentrated weights, branch lines, and changes in cross section. System loads such as weights, equivalent thermal forces, and earthquake inertia forces are applied at the nodal points. Stiffness characteristics of the interconnecting members are related to the effective shear area and moment of inertia of the pipe. The stiffness of piping elbows and certain branch connectors is modified to account for local deformation effects by the flexibility factors suggested in ASME Section III, 1974, Articles NB-3600 (Class 1 Piping Analysis) and NC-3600 (Class 2 Piping Analysis). The increased stiffness of valve bodies is taken into consideration.

The rules governing the design of branch connections to sustain internal and external pressure are contained in paragraph NX-3643 of the 1974 edition of ASME Section III. The code further states that reinforcement of a branch connection need not be provided if the fitting is manufactured in accordance with one of the standards listed in Table NX-3691-1 and used within the limits of the pressure-temperature ratings specified in such standards.

Since the piping shop fabrication specification for River Bend Station allows the use of various types of branch connections, including pipe-to-pipe, the pipe system stress analysis is performed using an unreinforced pipe-to-pipe connection. No further action is required if the allowable stresses are met. If the allowable stresses are not met, then the piping stress calculation identifies the reinforcement of the branch connection required to meet the stress allowables.

For cases where the branch line is decoupled from the run piping, the proper intensification factor is used in the analysis of both the branch line and the main run piping. If reinforcement is required, it is so identified in the piping stress calculation and drawings.

After completion of all piping fabrication and installation, those connections which are not considered to be reinforced will be indicated on as-built drawings. These drawings will then be reviewed against the piping stress calculation drawings, and reinforcement will be added where required.

Pressure reinforcement calculations are on file with the piping fabricator.

3.7.3.3.4A Selection of Mass Points

The lumped masses are located to adequately represent the dynamic properties of the piping system. Mass points are generally selected in accordance with the following guidelines.

1. At each node where a concentrated weight is placed (valves, flanges, or other in-line piping components).

2. At each intersection where three or more piping elements are connected (branch connections, tees, and y-fittings).

3. At the end of elbows and turns of direction.

4. At nodes subjected to input of dynamic force excitation.

5. At each terminal (node where only one element is connected such as end caps, valve operators).

6. At least one mass point between two restraints acting in the same direction.

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7. Lumped mass points are not be placed at, or in close proximity to, dynamic restraints, unless such placement is dictated by the characteristics of the system being modeled.

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The increased distributed mass of fittings is taken into consideration. When these guidelines are used the number of degrees of freedom in the dynamic model is greater than twice the number of modes with frequencies less than 33 Hz.

Mass points are treated as having 3 translational degrees of freedom. In special cases, rotational degrees of freedom are added.

3.7.3.4A Basis for Selection of Frequencies

3.7.3.4.1A Components

Amplified response spectra (floor) developed for the two orthogonal and vertical direction earthquakes are the basic source of seismic design accelerations. Seismic accelerations are selected from the amplified response spectra (ARS) based on the natural frequency calculations of the components with proper consideration of the frequency characteristics of the component supports. Appropriate amplification factors are included in the seismic loads to ensure the adequacy of the design of the components.

3.7.3.4.2A Piping

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Nearly all seismic floor response spectra of the Seismic Category I buildings, after peak spreading, have a broadened peak acceleration below a frequency of 5.5 Hz. Therefore, piping systems are supported in such a way that the lowest natural frequency of every analytical subsystem (piping bounded by components or anchors) is above the applicable spectrum peak range, unless other factors, such as seismic anchor movements, or stress problems affected by snubber reduction, dictate larger spans.

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For small-bore Seismic Category I piping (Section 3.7.3.8A) the same design approach is used. Based on envelope spectra for certain regions in the Seismic Category I buildings, maximum support spans for various configurations are determined so that the ASME Section III stress criteria are satisfied.

3.7.3.5A Use of Equivalent Static Load Method of Analysis

Those components which are considered relatively simple or rigid are designed, by virtue of natural frequency calculations, to withstand the effects of amplified seismic acceleration values dependent upon frequency and amplitude ranges associated with the relevant amplified response spectrum. Analysis of components to the peak value of resonant response is considered conservative, since fundamental natural frequencies do not generally coincide with the frequency at resonance of the relevant response curve. Components having fundamental natural frequencies less than the cutoff frequency (Section 3.7.3.1.1.1A) are designed to peak acceleration values, increased by a factor of 1.3, or as justified, to account for the contribution of all significant dynamic modes under a resonant condition. Justification for the use of 1.3 as a static coefficient can be found in Reference 7.

The factor of 1.3 is not applied to equipment whose natural frequency is 10 Hz or above. In the frequency range of the response spectrum greater than 10 Hz, no significant amplification effect is encountered and the equipment reacts to only the maximum floor response acceleration. The region of the response spectrum greater than 10 Hz is termed the rigid range. Equipment whose natural frequency is 10 Hz or above is analyzed to the rigid range response acceleration.

3.7.3.6A Three Components of Earthquake Motion

The maximum structural responses (displacements, acceleration, forces, and moments) due to each of the three components of earthquake motion, are combined by taking the SRSS of the maximum codirectional responses, caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model. This is in conformance with Regulatory Guide 1.92.

3.7.3.7A Combination of Modal Responses

The basis for computing combined response for use in subsystem analysis is presented in Section 3.7.2.7A.

3.7.3.8A Analytical Procedures for Piping

3.7.3.8.1A Introduction

Piping classified as Seismic Category I is designed to withstand levels of loading imposed by the OBE and the SSE. The piping systems are classified as:

1. Those governed by the ASME Code as Class 1, 2, or 3 piping

2. Those governed by the ANSI B31.1 code and requiring seismic analysis

3. Buried piping (Section 3.7.3.12A).

The seismic response of piping systems is analyzed by the response spectrum method or the time history method. The response spectrum method requires that seismic loading be combined from the dynamic response of the system based on an amplified response spectrum, and from the response to a quasi-static differential support movement, also called seismic anchor movement, which represents the out-of-phase movement of portions of the structure to which the system is attached. Computer analysis considers all vibration modes up to at least the mode beyond which the contribution to the overall seismic dynamic response is insignificant.

All safety-related piping systems that have been seismically analyzed are reviewed to verify that engineering input information and as-installed configurations are consistent with the latest design requirements as required by IE Bulletin 79-14. The process that governs this is a part of the design verification program for all Category I piping systems. The review consists of two parts; one that examines design inputs such as amplified response spectra and anchor motion, and one that compares as-built drawings against the as-analyzed calculations of record. The as-built drawings are the installation control drawing for piping and pipe supports which have been marked up to show the as-installed configuration in accordance with the specification requirements. The drawings are sent to the groups responsible for the final analysis where as-built as-analyzed comparisons are performed. Differences in configuration or input information are either justified on a case-by-case basis or the necessary changes are issued to the field. The marked-up drawings are reissued to incorporate the as-built information and become the drawings of record for the code qualification of piping and pipe supports.

The design attributes that are reviewed and the source documents that provide these attributes are provided in Table 3.7A-11. A list of applicable safety-related piping systems is provided in Table 3.2-1.

The final documentation of this program occurs at the time of N-5 signoff when a review is conducted to ensure that all input information is still valid and that any revisions that have taken place do not change the basis for the final analysis of record.

The structural damping is the same for all modes of the piping system and varies only with pipe size (Section 3.7.3.15A). A response spectrum curve contains a certain damping value implicitly. In time history analysis, the damping value is an input parameter to the analysis.

The number of earthquake cycles needed for fatigue analysis is given in Section 3.7.3.2A.

Pipe stress analysis classifications are given in Table 3.9A-4.

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An as-built analysis is performed on all Seismic Category I piping in conformance with NRC IE Bulletin 79-14.

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3.7.3.8.2A Analytical Techniques

3.7.3.8.2.1A General Criteria

In most cases, the maximum peak of the seismic response spectrum lies below a frequency of 5.5 Hz. Piping systems are rigidly supported, where possible, to assure a first mode natural frequency above the peak frequency after peak spreading.

3.7.3.8.2.2A Qualification of Small Bore Piping

The scope of small bore piping is limited to:

1. ASME Code Class 1 piping of 1-in NPS and under
2. ASME Code Class 2 and 3 piping of 2-in NPS and under, and
3. ANSI B31.1 piping (Class 4 piping) of 2-in NPS and under.

ASME Section III, Subarticle NB-3630 permits the use of Class 2 rules to qualify ASME Code Class 1 small bore piping.

In general, seismic qualification of small bore piping systems is performed by means of a simplified seismic analysis without computer application. As an example, Fig. 3.7A-27 illustrates a basic support concent for small bore piping. The seismic spans are given in Table 3.7A-8, the restraint design loads in Table 3.7A-9, and anchor design loads in Table 3.7A-10. Special cases outside the range of these tables are qualified by individual analysis.

3.7.3.8.3A Dynamic Analysis

3.7.3.8.3.1A Model

The procedure used for modeling is described in Section 3.7.3.3A.

3.7.3.8.3.2A Response Spectrum Method

When a piping system is analyzed by means of the response spectrum method, NUPIPE is used to calculate the modal response at each node point in the piping system due to the amplified response spectra excitation applied to the system. Generation or selection of the appropriate set of amplified response spectra for a subsystem supported at different elevations, and consideration of the effect of seismic differential displacements between restraints are discussed in Section 3.7.3.9A. The damping values for piping depend on pipe size and are given in Table 3.7A-1.

The equations of motion and their solution are the same as for other subsystems.

3.7.3.8.3.3A Time History Method

The applicable base motion time history is the structural response at a representative mass point of the structure to the ground motion time history.

The equations of motion and their solution are the same as in Section 3.7.2.5A, but the scalar acceleration term in the excitation function is now the amplitude of the acceleration of the base of the subsystem (points of attachment), not of the ground.

The effect of parameter variations on the time history are accounted for (Section 3.7.2.9A).

3.7.3.8.3.4A Dynamic Analysis Formulation

The basic equations of motion and their solutions are the same as for structures (Section 3.7.2.1.1.4A).

Absolute accelerations at points on the piping system are sometimes needed for qualification of equipment. With the response spectrum method, the maximum absolute accelerations at the mass points in mode i are obtained from Newton's law by dividing the effective inertia force by the mass of the mode:

$$\{a\}_i = [M]^{-1}\{Q\}_i$$

where:

$[M]$ = Diagonal mass matrix of the system

$\{Q\}_i$ = Effective inertia forces in mode i

With the time history method, the absolute accelerations are obtained by adding the base acceleration to the relative accelerations of the mass points.

3.7.3.8.3.5A Seismic Differential Displacements

3.7.3.8.3.5.1A Description of Input

The seismic differential displacements are also called seismic anchor movements. This effect is analyzed in a separate static load case for OBE anchor movements. The anchor movements are obtained from the seismic differential displacements of the structural nodes.

The displacements are obtained in the following form, one set for each mass point, N, of the building model:

DISPLACEMENTS

Mass Node	Earthquake Direction X	Y	Z
1	D_{1x}	D_{1y}	D_{1z}
2	D_{2x}	D_{2y}	D_{2z}
3	.	.	.
	.	.	.
N	D_{Nx}	D_{Ny}	D_{Nz}

where:

DN_x, N_y, N_z = The SRSS of the three directional displacement components at node N due to excitation in the north-south, east-west, and vertical directions

These are the movements of points on the walls relative to the foundation of the building.

For the purpose of calculating the relative movement between two points on the same wall or on two different walls of the same building the rigid body motion of the building is subtracted from its absolute motion.

3.7.3.8.3.5.2A Orbital Motion

Orbital motion is the relative movement between a point in the subsoil below one building and the corresponding point below another building. As a simplification, it is assumed that the relative seismic displacement between any two points of two buildings due to orbital motion response to an earthquake can be reduced to a horizontal movement applicable for either of the two horizontal earthquake components, and a vertical movement. Both of these are proportional to the horizontal distance between the centroids of the two buildings.

The orbital motion must be considered only when a subsystem is connected to two buildings with separate basemats. If the reference for the orbital motion is one of the buildings to which the piping is attached, all the support points in the other building have the effects of the orbital motion and of the free-body rotational motion of both buildings relative to each other superimposed on their movement relative to their mats.

3.7.3.8.3.5.3A Combination of Anchor Movement Loads

The seismic anchor movement load case is the root sum square of three static support displacement cases, one each for the X, Y, and Z components of the OBE. This operation is not built into the NUPIPE program. It is performed by first combining the X- and the Y-quake responses and then combining this case with the Z-quake response.

3.7.3.8.3.6A Combined Seismic Response

The system response to the response spectrum excitation (i.e., displacements, internal forces and moments, stresses, and support reactions) is obtained by first combining the modal contributions for each earthquake component by the SRSS method, taking into account the effect of closely spaced modes by the procedure described in Section 3.7.2.7A. The contributions of each of the three components are then combined by the SRSS methods.

When the response spectrum method is used, response to the differential support motion is considered. In Class 1 piping analysis, this motion is combined with the inertial response; then the result is combined with other load cases. In the analysis of other piping classes, the seismic anchor movement is combined with secondary loads. Seismic load cases are combined with other load cases (thermal, weight, pressure, other occasional loads) in accordance with ASME Section III, 1974. The load combinations are given in Section 3.9.3.1A.

3.7.3.8.3.7A Fatigue Considerations

For ASME Code Class 1 piping, a fatigue analysis must be performed when the stress limit is exceeded for Equation 10 of NB-3652.1 of ASME Section III, 1974, for the combined load cases. The number of stress cycles of all OBEs combined must be input. The number of earthquake cycles is discussed in Section 3.7.3.2A. (Fatigue analysis is performed in accordance with ASME Section III, 1974.)

3.7.3.8.3.8A Computer Programs Used for Seismic Analysis

All analyses are performed with NUPIPE. This program handles response spectrum and support motion time history analyses.

PSPECTRA is used for peak spreading, as well as for envelope generation, to obtain the amplified response spectra required as NUPIPE input to all spectral analyses.

All programs used in piping analysis are listed with others in Appendix 3A.

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3.7.3.9A Multiple Supported Equipment Components with Distinct Inputs

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When a subsystem is attached to different parts of a structure, such as separate elevations on one wall or

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several walls, the response spectra of all structural nodes for which response spectra exist and which lie nearest to the support elevation at the subsystem, both below and above the support elevation, are enveloped, and this envelope spectrum is applied to the subsystem. Alternately, multiple support excitation analysis may be used where different response spectra are applied to individual piping system attachment points. The use of multiple support excitation analysis is also discussed in Section 3.7.2.1.5.1B. The multiple support excitation analysis does not concurrently utilize the alternative damping suggested by the Pressure Vessel Research Council Technical Committee on Piping Design discussed in Section 3.7.2.15A and Table 3.7A-1.
1←• 14←•
In cases where a subsystem runs between two different buildings, a single ARS enveloping the spectra associated with all support points is used.
•→3 •→1 •→14
Independent support excitation was used to qualify the drywell portion of the RHR shut down cooling line connected to Recirculation Loop B. Independent support exitation was also used to qualify the RCIC Head Spray Line for the faulted condition due to Annulus Pressurization Steam Break.
1←• 3←• 14←•
In conjunction with the response spectrum loading, the loading from differential support displacements is calculated, and the two load cases are combined as described in Section 3.9.3.1A. Components and equipment generally have localized supports, and the effect can be ignored. The application to piping is discussed in detail in Section 3.7.3.8.3.5A.

3.7.3.10A Use of Constant Vertical Static Factors

Constant vertical static factors are not used.

3.7.3.11A Torsional Effects of Eccentric Masses

For Seismic Category I piping systems, if the torsional effect of the valve operator or other eccentric masses is likely to have a significant effect on the results of analysis described in Section 3.7.3.8A, the eccentric mass and its moment arm are included in the mathematical model described in Section 3.7.3.3A. However, if the pipe stress due to the torsional effect is expected to be less than 500 psi, the offset moment due to the eccentric mass is neglected.

3.7.3.12A Buried Seismic Category I Piping Systems and Tunnels

3.7.3.12.1A Piping

The only applicable piping in this power plant is two segments of about 100 ft of ASME Class 3 piping in the remote area intake section of the control building ventilation system.

Responses of buried Seismic Category I piping to differential ground motion, due to particle motions caused by seismic wave propagations, are calculated by a method reported in Sections 10.6 and 16.5 of Reference 1. The Rayleigh surface waves are the only waves considered in the analysis as they induce the highest axial strain. The net

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relative displacement between soil and pipe, caused by the seismic waves, is used to find the bending moments and shear forces by formulas for a beam on an elastic foundation.

Reactions and bending moments of buried Seismic Category I piping due to differential motion at structural penetrations are calculated by considering the buried pipe as a semi-infinite beam on an elastic soil foundation with full restraint at structural penetrations. Building seismic displacements are used as boundary condition input to the analysis. The transverse soil reaction is represented by restraints with linear spring constants. Static axial forces are imposed on straight sections of pipe to represent the soil friction. When a buried piping system is within the influence of two buildings, resultant movements of the buildings are conservatively assumed to be out of phase. The maximum expected seismic displacements at the structural penetration and the maximum modulus of the soil foundation are used to calculate the stress. The results are superimposed with axial tension and compression stress to meet the requirements defined in ASME Section III, 1974, Subarticles NC-3600 and ND-3600.

3.7.3.12.2A Tunnels and Other Structures

The Seismic Category I tunnels and other structures (e.g., electrical ducts, pipelines, etc) partially or completely buried are shown in Fig. 3.7A-36 and 2.5-107. These electrical ducts, pipelines, and tunnels are either covered with backfill or provided with a reinforced concrete roof slab at grade level. The seismic analyses of these structures are performed using the methods described in Section 3.7.2A. These structures are designed for the forces resulting from accelerations due to an OBE (or SSE) event (Section 3.7.2A) and static and dynamic soil pressures (in addition to other concurrent loading, e.g., dead load, live load, etc) as described in Section 3.8.4.3. The allowable stresses are described in Sections 3.8.4.3 and 3.8.4.5.

3.7.3.13A Interaction of Other Piping with Seismic Category I Piping

Where possible, non-seismic piping systems are designed to be isolated from any Seismic Category I piping system by a constraint or barrier, by differences in diameters, or by routing away from the Seismic Category I piping system. If it is not possible to isolate the Seismic Category I piping system from the non-seismic piping system, the adjacent non-seismic piping is seismically designed according to the same criteria as the Seismic Category I piping system.

For the non-seismic piping systems attached to Seismic Category I piping systems, the dynamic effects of the non-seismic piping are simulated in the analytical model of the Seismic Category I piping. The attached non-seismic piping is also supported in such a manner that its failure (e.g., a circumferential break) does not cause a failure of the Seismic Category I piping.

In addition to the loads from seismically designed piping, the anchors that separate seismically designed piping and nonseismic Category I piping are evaluated to accommodate the maximum loads the nonseismic side can transmit. This ensures the integrity of the anchor so that failure, if it occurs, is on the nonseismic piping, not the anchor.

3.7.3.14A Seismic Analysis for Reactor Internals

See Section 3.7.3.14B.

3.7.3.15A Analysis Procedure for Damping

The percentages of critical damping values assigned to Seismic Category I systems and components are in accordance with Regulatory Guide 1.61 and are presented in Table 3.7A-1, with the following clarification: for some of the equipment the use of lower damping values than those specified in Table 1 of the Regulatory Guide is evaluated to determine that the increased amplitude of vibration and dynamic stress are not adversely affecting the performance and safety of the equipment. Where increased amplitude and dynamic stresses are expected to affect the performance and safety of the equipment, a detailed analysis is performed to comply with the requirements of Paragraph C.3 of the Regulatory Guide.

3.7.4A Seismic Instrumentation

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The surveillance requirements for these instruments are listed in and controlled by the RBS Technical Requirements Manual.

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3.7.4.1A Comparison with Regulatory Guide 1.12

To monitor and record input motion and behavior of the station in the event of an earthquake, a seismic instrumentation program is implemented. This instrumentation program complies with the requirement of Regulatory Guide 1.12.

3.7.4.2A Location and Description of Instrumentation

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Strong motion triaxial accelerographs are installed in four different locations. Three sensor packages are installed in the reactor building. One is located on the reactor mat in the auxiliary building. The second is located vertically over the first, mounted on the inside of the shield building. The third is located on the drywell. These locations are separated from each other by a vertical distance which is a significant fraction of the reactor building height. The fourth strong motion triaxial accelerograph is located in the free field to obtain a more detailed knowledge of soil-structure interaction. All sensor packages are located in areas where they can be serviced.

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The strong motion triaxial accelerographs to be installed have the following physical characteristics:

1. Accelerometers are the transducer-type with the capability of recording a maximum of 1.0 g at full scale.

2. Accelerometers are sensitive to frequencies in the range of 0.1 to 50 Hz.

3. The seismic instrumentation and recording system is in a quiescent state until activated by seismic triggers which are set at 0.01 g. These seismic triggers (both horizontal and vertical) activate the recording system in less than 100 ms. Recording continues until the level of motion drops below 0.01 g.

4. The recording system is powered by internal batteries with trickle charge from 110 V ac.

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5. Each sensor package contains three mutually orthogonal accelerometers. All four sensor packages are oriented to the same azimuths.

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6. Recording of the electrical signals from the accelerometers is by magnetic tape with the acceleration signal and the time signal occupying separate tracks on the tape.

Also, in accordance with Regulatory Guide 1.12, three peak recording accelerographs are installed on selected Seismic Category I piping and equipment to verify the seismic response determined analytically from the traces recorded by the strong motion accelerographs. The locations of peak recording accelerographs are the standby liquid control storage tank in the reactor building, RHR injection piping in the reactor building, and service water piping in the auxiliary building. The peak recording accelerographs are installed to detect and record peak amplitude of low frequency accelerations. These seismic instruments have the following physical characteristics:

1. Accelerometers have a sensitivity of 10 g full scale.

2. The accelerograph records by scribing excursions of diamond stylus on replaceable metal plate, one for each of the three orthogonal axes.

3. No power is required to operate the instruments.

4. Air damping is used to greater than 0.5 critical.

5. Operating temperature is -40~F to 185~F.

A seismic switch, located on the reactor building mat, is used to trip an audio and visual alarm system in the event of an earthquake of 0.05 g or larger. The seismic switch has the following physical characteristics:

1. Frequency range: 0.1 to 33 Hz

2. The trigger levels are set to 0.082 g horizontally and 0.083 g vertically.

Seismic instrumentation that provides a spectrum of measured responses is installed in four different locations: one on the reactor building mat; the second and third on the floors at el 70 ft 0 in and 141 ft 0 in of the auxiliary building; and the fourth in the reactor building at the floor el 141 ft 0 in. Floors of the reactor and auxiliary buildings are selected because nuclear safety-related equipment is supported on the floors. This instrument is called a triaxial response spectrum recorder and senses, as well as permanently records, the information defining a response spectrum for each of the three mutually perpendicular directions.

The response spectrum recorder located on the reactor building mat is an active type and is connected to the main control room audio and visual annunciator. The other three recorders are a passive type, which record the data on metal plates. The recorders cover the range of 1 to 32 Hz in 1/3 octave increments. Sixteen mechanical accelerometers, whose damping is 2 percent of critical, use diamond-tipped styli to inscribe permanent records of displacement on record plates. Recorded displacements can be converted into a plot of acceleration versus frequency.

3.7.4.3A Main Control Room Operator Notification

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Plant seismic instrumentation provide the following control room signals to the operator:

a. A visual and audible annunciator is actuated in the control room when the triaxial seismic switch, located on the Reactor Building mat, signals that the OBE acceleration has been exceeded in either one of the horizontal directions or in the vertical direction.

b. A visual and audible annunciator is actuated in the control room when any of the 16 accelerometers of each triaxial component of the "active" triaxial response spectrum recorder, located on the Reactor Building mat, exceeds their frequency setpoint.

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Each of the accelerometers have two setpoints. Exceeding the first setpoint illuminates an amber light which indicates that accelerations are exceeding 100% OBE. If the second setpoint is exceeded, a red light is illuminated which indicates that the accelerations are exceeding 100% SSE. This light display indication is the Response Spectrum Annunciator and is located in the main control room.

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c. An annunciator alarm (visual and audible) is actuated when the seismic trigger, located in the "free field", detects accelerations greater than 0.01g in either one of the horizontal directions or in the vertical direction. Exceedence of the trigger setpoint activates the strong motion triaxial accelerometers located throughout the plant and the time-history accelerograph recorders in the main control room.

   Upon activation, the strong motion triaxial accelerometers transmit signals to the Magnetic Tape Recorders. When processed through the Magnetic Tape Playback system, an acceleration time-history strip chart is produced.

In the event of an earthquake, the control room operator determines whether or not the OBE acceleration level has

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been exceeded. This is accomplished by inspection of the indications and alarms described above. If the OBE spectrum has been exceeded, the plant is shutdown and an evaluation of the impact of the event is initiated.
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3.7.4.4A Comparison of Measured and Predicted Responses

In order to make detailed comparisons between measured seismic responses of Seismic Category I structures and equipment with calculated accelerations determined from dynamic analysis, the following procedure is implemented:

1. The magnetic tape records are digitized and corrected for time signal variations and baseline deviations.

2. The time history records from the triaxial sensors located on the shield building and the base mat of the reactor building are used directly to calculate

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amplified response spectra at the appropriate damping ratio.

3. The time history records from the free field triaxial sensor are used as input ground motion for the reactor building dynamic model. Amplified response spectra are then calculated at the locations of the other two sensors in the reactor building for comparison and correlation with the response spectra determined as in 2. Reasonable correlation between the spectra is accomplished on an iterative basis by varying the physical properties of the models (stiffnesses and damping characteristics) to calibrate the dynamic model. Once the dynamic model has been calibrated, additional verification of its correctness is made by use of the acceleration readings from the peak recording accelerograph.

4. Structural responses and amplified response spectra are calculated using the free field time history records with the calibrated dynamic model for comparison with the original plant design parameters. This comparison permits evaluation of seismic effects on structures and equipment and forms the basis for detailed analyses and physical inspection.

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3.7B SEISMIC DESIGN

This section deals with systems, components, and equipment of the NSSS scope of supply; Section 3.7A is referred to for structural aspects. The NSSS systems, components, and equipment are categorized as Seismic Category I or non-Seismic Category I. The requirements for Seismic Category I qualification are given in Section 3.2 along with a list of systems, components, and equipment which are so categorized.

All systems, components, and equipment related to plant safety are designed to withstand potential earthquakes defined herein.

The "safe shutdown earthquake" is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which Seismic Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary.

2. The capability to shut down the reactor and maintain it in a safe shutdown condition.

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines exposures of 10CFR50.67. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

The "operating basis earthquake" is that earthquake which, considering the regional and local geology, seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

3.7.1B Seismic Input

3.7.1.1B Design Response Spectra

See Section 3.7.1.1A.

3.7.1.2B Design Time History

See Section 3.7.1.2A.

3.7.1.3B Critical Damping Values

The damping factors indicated in Table 3.7B-1 were used in the response analysis of various systems, components, and equipment and in preparation of floor response spectra used as forcing inputs for piping and equipment analysis or testing.

GE-supplied NSSS analysis, design and/or equipment utilized in this facility is in compliance with Regulatory Guide 1.61. This guide delineates damping values that should be applied to modal dynamic seismic analysis of Seismic Category I structures and components.

The damping values listed in Table 3.7B-1 are consistent with the values given by the Regulatory Guide. Paragraph C.3 of the Regulatory Guide requires that lower damping values be used if the maximum combined stresses are significantly lower than yield or one-half yield stresses for the SEE and OBE, respectively, in any structure or component.

3.7.1.4B Supporting Media for Seismic Category I Structures

See Section 3.7.1.4A.

3.7.2B Seismic System Analysis

3.7.2.1B Seismic Analysis Methods

Analysis of Seismic Category I GE-supplied systems and components is accomplished, where applicable, using the response spectrum or time-history approach. Either approach utilizes the natural period, mode shapes, and appropriate damping factors of the particular system. Certain pieces of equipment having very high natural frequencies may be analyzed statically. In some cases, dynamic testing of equipment may be used for seismic qualification.

The time history analyses involve the solution of the equations of the dynamic equilibrium (Section 3.7.2.1.1B) by means of the methods discussed in Section 3.7.2.1.2B. In this case, the duration of motion is of sufficient length to ensure that the maximum values of response have been obtained.

A response spectrum analysis involves the solution of the equations of motion (Section 3.7.2.1.1B) by the method discussed in Section 3.7.2.1.3B.

3.7.2.1.1B The Equations of Dynamic Equilibrium

Assuming velocity proportional damping, the Dynamic Equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as:

$$[M]\ \{\ddot{u}(t)\} + [C]\ \{\hat{u}(t)\} + [K]\ \{u(t)\} = \{P(t)\} \qquad (3.7B\text{-}1)$$

where:

$\{u(t)\}$ = Time dependent displacement vector of non-support points relative to the supports

$\{\hat{u}(t)\}$ = Time dependent velocity vector of non-support points relative to the supports

$\{\ddot{u}(t)\}$ = Time dependent acceleration vector of non-support points relative to the supports

$[M]$ = Diagonal matrix of lumped masses

$[C]$ = Damping matrix

$[K]$ = Stiffness matrix

$\{P(t)\}$ = Time dependent inertial force vector acting at non-support points.

The manner in which a distributed mass, distributed stiffness system is idealized into a lumped mass distributed stiffness system representation of the NSSS component is shown in Fig. 3.7B-1, along with a schematic representation of relative acceleration. The total acceleration on which the inertia force is based is the sum of the base acceleration and the relative acceleration.

3.7.2.1.2B Solution of the Equations of Motion by Mode Superposition

The first technique used for the solution of the equations of motion is the method of mode superposition. The set of homogeneous equations represented by the undamped free vibration of the system is:

$$[M]\ \{\ddot{u}(t)\} + [K]\ \{u(t)\} = \{0\} \quad (3.7B\text{-}2)$$

Since the free oscillations are assumed to be harmonic, the displacements can be written as:

$$\{u(t)\} = \{\phi\}e^{i\omega t} \qquad (3.7B\text{-}3)$$

where:

$\{\phi\}$ = Column matrix of the amplitude of displacements [u]

ω = Circular frequency of oscillation

t = Time.

Substituting Equation 3.7B-3 and its derivatives in Equation 3.7B-2 and noting that $e^{i\omega t}$ is not necessarily zero for all values of ωt yields:

$$[-\omega^2\ [M] + [K]\ ]\ \{\phi\} = \{0\} \qquad (3.7B\text{-}4)$$

Equation 3.7B-4 is the characteristic equation for the classical algebraic eigenvalue problem wherein the eigenvalues are the frequencies of vibration ω_i and the eigenvectors are the mode shapes, $[\phi]_i$.

For each frequency ω_i there is a corresponding solution vector $[\phi]_i$. It can be shown that the mode shape vectors are orthogonal with respect to the weighted stiffness matrix [K] in the n-dimensional vector space.

The mode shape vectors are also orthogonal with respect to the weighted mass matrix [M].

The orthogonality of the mode shapes is used to effect a coordinate transformation of the displacements, velocities and accelerations such that the response in each mode is independent of the response of the system in any other mode. Thus, the problem becomes one of solving n independent differential equations rather than n simultaneous differential equations, and since the system is linear, the principle of superposition holds and the total response of the system oscillating simultaneously in n modes is determined by direct algebraic addition of the responses in the individual modes.

3.7.2.1.3B Analysis by Response Spectrum Method

The response spectrum method is based on the fact that the modal responses can be expressed as a set of convolution integrals which satisfy the governing differential equations. The advantage of this form of solution is that for a given ground motion the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor it is possible to construct a curve which gives a maximum value of the integral as a function of frequency. This curve is called a response spectrum for the particular input motion and the specified damping factor. The integral has units of velocity; consequently the maximum of the integral is called the spectral velocity.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Section 3.7.3.7B.

3.7.2.1.4B Support Displacement in Multi-Supported Systems, Components, and Equipment

The preceding sections have discussed analysis procedures for forces and displacements induced by time dependent support accelerations. In a multi-supported system there are, in addition, time dependent support displacements which produce additional displacements at non-support points and pseudo-static forces at both support and non-support points.

The governing equation of motion of a system, component, or equipment which is supported at more than one point and has different excitations applied at each may be expressed in the following concise matrix form:

$$\begin{bmatrix} M_a & O \\ o & M_s \end{bmatrix} \begin{Bmatrix} \ddot{U}_a \\ \ddot{\bar{U}}_s \end{Bmatrix} + \begin{bmatrix} C_{aa} & C_{as} \\ C_{as} & C_{ss} \end{bmatrix} \begin{Bmatrix} \dot{U}_a \\ \dot{\bar{U}}_s \end{Bmatrix}$$

$$+ \begin{bmatrix} K_{aa} & K_{as} \\ K_{as} & K_{ss} \end{bmatrix} \begin{Bmatrix} U_a \\ \bar{U}_s \end{Bmatrix} = \begin{Bmatrix} \bar{F}_a \\ F_s \end{Bmatrix} \qquad (3.7B\text{-}5)$$

Where U_a represents the displacement of the active (unsupported) degrees of freedom, U_s denotes the specified displacements of support points, M_a and M_s are the lumped diagonal mass matrices associated with the active degrees of freedom and the support points, C_{aa} and K_{aa} are the damping matrix and elastic stiffness matrix respectively expressing the forces developed in the active degrees of freedom due to the motion of the active degrees of freedom, C_{ss} and K_{ss} are the support forces due to unit velocities and displacement of the supports, C_{as} and K_{as} are the damping and stiffness matrices denoting the coupling forces developed in the active degrees of freedom by the motion of the supports, and vice versa, $\bar{F}_a$ is the prescribed external time-dependent forces applied on the active degrees of freedom, and F_s is the reaction force at the system support points. Total differentiation with respect to time is denoted by ($^\bullet$). Also, the contributions of the fixed degrees of freedom have been removed in the above equation. The procedure utilized to construct the damping matrix is discussed in Section 3.7.2.15B. The mass matrix and elastic stiffness matrix are formulated by using standard procedures.

Equation 3.7B-5 can be separated into two sets of equations. The first set of equations can be written as:

$$(3.7B\text{-}6a)$$

$$[M_s]\{\ddot{\bar{U}}_s\} + [C_{ss}]\{\dot{\bar{U}}_s\} + [K_{ss}]\{\bar{U}_s\} + [C_{as}]\{\dot{U}_a\}$$
$$+ [K_{as}]\{U_a\} = \{F_s\}$$

and the second set as

$$[M_a]\{U_a\} + [C_{aa}]\{U_a\} + [K_{aa}]\{U_a\} + [C_{as}]\{\dot{\bar{U}}_s\}$$
$$+ [K_{as}]\{\bar{U}_s\} = \{\bar{F}_a\}$$

(3.7B-6b)

The timewise solution of Equation 3.7B-6a can be obtained easily by using the standard normal mode solution technique. After obtaining the displacement response of the active degrees of freedom, U_a, Equation 3.7B-6b then can be used to solve the support point reaction forces, F_s.

Modal superposition is used to determine the solutions of the uncoupled form of Equation 3.7B-6a. The procedure is identical to that described in Section 3.7.2.15B. C_{aa} and K_{aa} are the damping matrix and elastic stiffness matrix, respectively, expressing the forces developed in the active degrees of freedom.

3.7.2.1.5B Dynamic Analysis of Seismic Category I Systems, Components, and Equipment

Time-history and response spectrum techniques are used as applicable for the dynamic analysis of Seismic Category I systems, components, and equipment which are sensitive to dynamic seismic events.

3.7.2.1.5.1B Dynamic Analysis of Piping Systems

Each pipe line is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the piping system is anchored and supported at points with different excitations the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternately multiple support excitation analyses may be used where different acceleration time histories or response spectra are applied to individual piping system attachment points.

The relative displacement between support points is determined from the dynamic analysis of the structures. The results of the relative anchor point displacement are used for a static analysis to determine the additional stresses due to relative anchor point displacements.

3.7.2.1.5.2B Dynamic Analysis of Equipment

Equipment is idealized as a mathematical model consisting of lumped masses connected by elastic members or springs.

When the equipment is supported at two or more locations, an upper bound envelope of all the individual response spectra is used to calculate maximum inertial responses of multiple supported items. Alternately, the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically to all the floors, provided there is no significant shift in frequencies of the spectra peaks.

In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the subsystems. Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiple supported systems.

When the equipment is supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternately the multiple support excitation analysis methods may be used where accumulation time histories or response spectra are applied at all the equipment attachment points.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the additional stresses due to support displacements. Further details are given in Section 3.7.2.1.5.2.1B.

3.7.2.1.5.2.1B Differential Seismic Movement of Interconnected Components

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitation is as follows:

The relative displacements between the supporting point induces additional stresses in the equipment supported at these points. These stresses can be evaluated by performing a static analysis where each of the supporting points is displaced a prescribed amount. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. These displacements are used to calculate stresses by determining the peak nodal responses.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the modal displacement is used. Therefore, the mathematical model of the equipment is subjected to a maximum displacement at its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the structure. The total stresses due to relative displacements are obtained by combining the modal results using the SRSS (square root of sum of the squares) method. Since the maximum displacements for different modes do not occur at the same time, the SRSS method is a realistic and practical method.

When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

3.7.2.1.6B Seismic Qualification by Testing

For certain Seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results reflect the following:

1. Performance data of equipment which, under the specified conditions, has been subjected to dynamic loads equal to or greater than those to be experienced under the specified seismic conditions.

2. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to dynamic loads equal to or greater than those specified.

3. Actual testing of equipment in accordance with one of the methods described in Sections 3.9B and 3.10B.

3.7.2.2B Natural Frequencies and Response Loads

See Section 3.7.2.2A.

3.7.2.3B Procedure Used for Modeling

3.7.2.3.1B Modeling Techniques for Seismic Category I Systems, Components, and Equipment

The techniques currently being used for modeling represent the Seismic Category I systems, components, or equipment by lumped masses and a set of spring dashpots idealizing both the inertial and stiffness properties of the system. The details of the mathematical models are determined by the complexity of the actual system and the information required from the analysis.

3.7.2.3.2B Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the reactor pressure vessel (RPV) and internals are based on a dynamic analysis of an entire RPV-building complex with the appropriate forcing function supplied at ground level. For this analysis, the models shown in Fig. 3.7B-2 and the mathematical model of the building are coupled together.

The RPV internals mathematical model consists of lumped masses connected by linear elastic beam element members. Using the elastic properties of the structural components, the stiffness properties of the model are determined and the effects of both bending and shear are included. Mass points are located at points of critical interest such as anchors, supports, etc. In addition, mass points are chosen such that the mass distribution in various zones is as uniform as practicable and the full range of frequency of response of interest is adequately represented. Further, in order to facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel), are selected at the same elevation. The various lengths of control rod drive housings are grouped into the two representative lengths shown in Fig. 3.7B-2. These lengths represent the longest and shortest housings in order to adequately represent the full range of frequency response of the housings.

The high fundamental natural frequency of the CRD housings results in very small seismic loads. Furthermore, the small frequency differences between the various housings due to the length differences result in negligible differences in dynamic response. Hence, the modeling of intermediate length members becomes unnecessary. Not included in the mathematical model are the stiffness of light components such as jet pumps, in-core guide tubes and housings, sparger, and their supply headers. This is done to reduce the complexity of the dynamic model. For the seismic responses of these components, floor response spectra generated from the analysis are used.

The presence of a fluid and other 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 introduction of a hydrodynamic mass matrix, which will serve to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in Reference 1. The seismic model of the RPV and internals has two horizontal coordinates for each mass point considered in the analysis. The remaining translational coordinate (vertical) is excluded because the horizontal and vertical degrees of freedom of RPV and internals are elastically decoupled. Furthermore, all support structures, building and containment walls have a common centerline, and hence, the coupling effects are negligible. A separate vertical linear analysis is performed. Dynamic loads due to vertical motion are added to or subtracted from the static weight of components, whichever is more conservative. The rotary inertia are assumed negligible.

The shroud support plate is loaded in its own plane during a seismic event and hence is extremely stiff and modeled as a rigid link in the translational direction. The shroud support legs and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

3.7.2.4B Soil-Structure Interaction

See Section 3.7.2.4A.

3.7.2.5B Development of Floor Response Spectra

See Sections 3.7.2.5A and 3.7.3.1B

3.7.2.6B Three Components of Earthquake Motion (NSSS)

Details are the same as given in Section 3.7.3.6B.

3.7.2.7B Combination of Modal Responses (NSSS)

All the modal responses are combined as described in Section 3.7.3.7B.

3.7.2.8B Interaction of Non-Category I Structures with Seismic Category I Structures

See Section 3.7.2.8A.

3.7.2.9B Effects of Parameter Variations on Floor Response Spectra

To account for potential frequency variations, the recommendations of Regulatory Guide 1.122 are followed.

3.7.2.10B Use of Constant Vertical Static Factors (NSSS)

Constant vertical static factors are not used for systems such as the RPV, internals, and large piping. See Section 3.7.3.10B for subsystems and components.

3.7.2.11B Method Used to Account for Torsional Effects (NSSS)

The reactor pressure vessel (RPV) is an axisymmetric model with no built-in eccentricity. Hence, the torsional effects on the RPV are only those associated with the reactor building model. The torsional effects are accounted for in the reactor building model as described in Section 3.7.2.11A.

3.7.2.12B Comparison of Responses (NSSS)

Either the time history method or the response spectra approach may be used for the seismic analysis of NSSS components. Generally, the responses computed by both methods are comparable in magnitude, with the loads determined by the response spectrum method being somewhat more conservative. As both of these approaches are acceptable, additional comparison of results was deemed unnecessary.

3.7.2.13B Methods for Seismic Analysis of Dams

See Section 3.7.2.13A.

3.7.2.14B Determination of Seismic Category I Structure Overturning Moments

See Section 3.7.2.14A.

3.7.2.15B Analysis Procedure for Damping

In a linear dynamic analysis, the procedure to be utilized to properly account for damping in different elements of a coupled system model is as follows:

1. The structural percent critical damping of the various structural elements to the model are first specified. Each value is referred to as the damping ratio (C_j) of a particular component which contributes to the complete stiffness of the system.

2. An eigenvalue analysis of the linear system model is performed. This results in the eigenvector matrices (ϕ_i), which are normalized and satisfy the orthogonality conditions:

   $$\phi_i^T K \phi_i = \omega_i^2 \text{, and } \phi_i^T K \phi_j = 0 \text{ for } i \neq j$$

   where:

   K = Stiffness matrix

   ω_i = Circular natural frequency associated with mode i

   ϕ_i^T = Transpose of i^{th} mode eigenvector ϕ_i

   Matrix ϕ contains all translational and rotational coordinates.

3. Using the strain energy of the individual components as a weighting function, the following equation is derived to obtain a suitable damping ratio (β_i) for mode i.

$$\beta_i = \frac{1}{\omega_i^2} \sum_{j=1}^{N} \left[C_j \left(\phi_i^T K \phi_i \right)_j \right]$$

where:

β_i = Modal damping coefficient for ith mode

N = Total number of structural elements

ϕ_i = Components of i^{th} mode eigenvector corresponding to j^{th} beam element

ϕ_i^T = Transpose of ϕ_i defined above

C_j = Percent critical damping associated with element

K = Stiffness matrix of element j

ω_i = Circular natural frequency of mode i

3.7.3B Seismic Subsystem Analysis

3.7.3.1B Seismic Analysis Methods

The seismic system analysis methods described in Section 3.7.2.1B are applicable to the subsystems, components, and equipment. The following is a description of the methods by which Seismic Category I subsystems and components are qualified to ensure the functional integrity of the specific operating requirements which characterize their Seismic Category I designation.

In general, one of the following five methods of seismically qualifying the equipment is chosen based upon the characteristics and complexities of the subsystem:

1. Dynamic analysis
2. Testing procedures
3. Equivalent static load method of analysis
4. A combination of Items 1 and 2
5. A combination of Items 2 and 3.

Equivalent static load method of subsystem analysis is described in Section 3.7.3.5B.

Appropriate design response spectra (OBE and SSE) are furnished to the manufacturer of the equipment for seismic qualification purposes. Additional information such as input time history is also supplied only when necessary.

When analysis is used to qualify Seismic Category I subsystems and components, the analytical techniques must conservatively account for the dynamic nature of the subsystems or components. Both the SSE and OBE, with their different damping values, are considered when the dynamic analysis is performed.

The general approach employed in the dynamic analysis of Seismic Category I equipment and component design is based on the response spectrum technique. The time-history technique described in Section 3.7.2.1.1B generates time histories at various support elevations for use in the analysis of subsystems and equipment. The structural response spectra curves are subsequently generated from the time-history accelerations.

At each level of the structure where vital components are located, three orthogonal components of floor response spectra (two horizontal and one vertical) are developed. The floor response spectrum is smoothed and envelopes all calculated response spectra from different site soil conditions. To account for frequency uncertainties, the response spectra are peak broadened plus or minus 15 percent. When components are supported at two or more elevations, the response spectra of each elevation are superimposed and the resulting spectrum is the upper bound envelope of all the individual spectrum curves considered. Alternatively, multiple support excitation analyses may be used where different response spectra are applied to the individual supports.

For vibrating systems and their supports, multi-degree-of-freedom models are used in accordance with the lumped-parameter modeling techniques and normal mode theory described in Section 3.7.2.1.1B and the references listed in Section 3.7B. Piping analysis is described in Sections 3.7.3.3.1B and 3.7.2.1.5.1B.

When testing is used to qualify Seismic Category I subsystems and components, all the loads normally acting on the equipment are simulated during the test. The actual mounting of the equipment is also simulated or duplicated. Tests are performed by supplying input accelerations to the shake table to such an extent that generated test response spectra (TRS) envelope the required response spectra.

For certain Seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results reflect the following:

1. Performance data of equipment which has been subjected to dynamic loads equal to or greater than those experienced under the specified seismic conditions.

2. Test data from previously tested comparable equipment which has been subjected under similar conditions to dynamic loads equal to or greater than those specified.

3. Actual testing of equipment in accordance with one of the methods described in Sections 3.9.2.2B and 3.10B.

3.7.3.2B Determination of Number of Earthquake Cycles

3.7.3.2.1B Piping

Fifty peak OBE cycles are postulated for fatigue evaluation.

3.7.3.2.2B Other Equipment and Components

To evaluate the number of cycles engendered by a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories - May 18, 1940, El Centro NS component 29.4 sec; 1952, Taft N 69° W component, 30 sec; and March 1957, Golden Gate S80E component, 13.2 sec. The modal response was truncated such that the response of three different frequency bandwidths could be studied, 0-10 Hz, 10-20 Hz, and 20-50 Hz. This was done to give a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior as given in Table 3.7B-4 was formed.

Independent of earthquake or component frequency, 99.5 percent of the stress reversals occur below 75 percent of the maximum stress level, and 95 percent of the reversals lie below 50 percent of the maximum stress level.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

1. The fundamental frequency and peak seismic loads are found by a standard seismic analysis (i.e., from eigen extraction and forced response analysis).

2. The number of cycles which the component experiences is found from Table 3.7B-4 according to the frequency range within which the fundamental frequency lies.

3. For fatigue evaluation, one-half percent (0.005) of these cycles are conservatively assumed to be at the peak load, and 4.5 percent (0.045) at the three-quarter peak. The remainder of the cycles have negligible contribution to fatigue usage.

The safe shutdown earthquake has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40-yr life of a plant. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Section III.

The OBE is an upset condition and therefore, must be included in fatigue evaluations according to ASME Section III. Investigation of seismic histories for many plants show that during a 40-yr life that five earthquakes with intensities one-tenth of the SSE intensity, and one earthquake approximately 20 percent of the proposed SSE intensity, are probable. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, ten peak OBE stress cycles are postulated for fatigue evaluation.

3.7.3.3B Procedure Used for Modeling

3.7.3.3.1B Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of three dimensional straight or curved pipe elements. The mass of each pipe element is lumped at the nodes connected by weightless elastic member, representing the physical properties of each segment. The pipe lengths between mass points is no greater than the length which would have a natural frequency of 33 Hz when calculated as a simply supported beam. In addition, mass points are located at all points on the piping system such as main valves, relief valves, pumps, and motors are modeled as lumped masses. The torsional effects of the valve operators and other equipment with an offset center of gravity with respect to center line of the pipe is included in the analytical model. When the torsional effect is found to cause pipe stresses less than 500 psi, this effect is neglected.

The procedure employed for decoupling the main steam and recirculation piping systems when establishing the analytical models to perform seismic analysis is given below:

1. The small branch lines (6-in diameter and less) are decoupled from the main steam and recirculation piping systems and analyzed separately.

2. The stiffness of all the anchors and its supporting steel is large enough to effectively decouple the piping on either side of the anchor for analytical and code jurisdictional boundary purposes. The RPV is very stiff compared to the piping system and thus during normal operating conditions the RPV is also assumed to act as an anchor. Penetration assemblies (head fittings) are also very stiff compared to the piping system and are assumed to act as an anchor. The stiffness matrix at the attachment location of the process pipe (i.e., main steam, RCIC, RHR supply or RHR return) head fitting is sufficiently high to decouple the penetration assembly from the process pipe. Analysis indicates that a satisfactory minimum stiffness for this attachment point is equal to the stiffness in bending and torsion of a cantilevered pipe section of the same size as the process pipe and equal in length to three times the process pipe outer diameter.

3.7.3.3.2B Modeling of Equipment

For dynamic analysis, Seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

1. The number of modes of a dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are

considered significant if the corresponding natural frequencies are less than 33 Hz and the stress calculated from these modes is greater than 10 percent of the total stresses obtained from lower modes.

2. Mass is lumped at any point where a significant concentrated weight is located. Examples are: The motor in the analysis of pump motor stand, the impeller in the analysis of pump shaft, etc.

3. If the equipment has a free-end overhang span with flexibility significant compared to the center span, a mass is lumped at the overhang span.

4. When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to lower the natural frequencies of the equipment. Similarly, in the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen so as to yield the lowest frequency content for the system. This is to ensure conservative dynamic loads since the equipment frequencies are always higher than the frequencies at which the spectral peaks occur. If such is not the case, the model is adjusted to give more conservative results.

3.7.3.3.3B Field Location of Supports and Restraints

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the responsible design engineer. An additional examination for these as-built supports and restraining devices is made to assure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems. The final analyses of the as-built systems are performed as necessary, and the final certified as-built design report is issued.

3.7.3.4B Basis of Selection of Frequencies

All frequencies in the range of 0.25 to 33 Hz are considered in the analysis and testing of systems, components, and equipment. These frequencies are excited under the seismic excitation.

If the fundamental frequency of a component is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered as they represent very flexible structures and are not encountered in this plant.

The frequency range of between 0.25 Hz and 33 Hz covers the range of the broad band response spectrum used in the design.

3.7.3.5B Use of Equivalent Static Load Method of Analysis

When the natural frequencies of a system, component, or equipment are unknown, they may be analyzed by applying an equivalent static coefficient analysis. This procedure allows a simpler technique in return for added conservatism. The static acceleration of a component is conservatively assumed to be the peak spectral acceleration of the required response spectrum (RRS) which envelops the multisupport input spectra. The oscillator damping associated with the enveloping RRS must be representative of the actual component damping.

The equivalent static acceleration is then obtained by multiplying the static acceleration by a static coefficient that takes into account the effects of both multifrequency excitation and multimode response. For verifying the structural integrity of frame-type components physically similar to beams and columns, the static coefficient is taken as 1.5. For equipment having other than a frame-type configuration, justification is provided for the static coefficient used.

The equivalent static forces on each subcomponent of the equipment are obtained by multiplying the subcomponent masses by the equivalent static acceleration. The resulting static load vector is distributed over the equipment in a manner proportional to its mass distribution. The static stress analysis is then performed in a normal manner.

3.7.3.6B Three Components of Earthquake Motion

The total seismic response is obtained by combining the colinear responses corresponding to each of the three orthogonal components (two horizontal and one vertical) of seismic excitation. The total response can be generated by either the time history or response spectrum method of analysis. This is in compliance with Regulatory Guide 1.92 as discussed in Table 1.8-1.

when the time history method is utilized the colinear time history responses from each of the three seismic excitation components are independently obtained and then algebraically combined at each time step. Alternatively, the total response at each step is calculated directly from the simultaneous application of all three components of excitation. The maximum total response is then the peak value of the simultaneous time history solution. When this method is used, the three orthogonal components of excitation must be statistically independent.

The time history method can also be used to calculate peak dynamic responses due to the independent application of each of the orthogonal components of excitation. When this procedure is followed the total dynamic response is obtained as an SRSS combination of the corresponding colinear responses due to each of the three components of excitation. The procedure is identical to that for the response spectrum method described next.

When the response spectrum method is employed the representative maximum values of corresponding colinear responses, due to each component of excitation, are combined by SRSS. The justification for this procedure is based on the following three basic criteria:

1. The peak responses of different modes to the same earthquake excitation do not occur at the same time.

2. The peak responses of a particular mode to earthquake excitations in different directions do not occur at the same time.

3. The peak stress due to different modes and excitations may not occur at the same location or in the same direction.

Mathematically, the seismic responses are combined in the following manner:

$$R_i = \left[\sum_{j=1}^{j=3} \left(R_{ij}\right)^2 \right]^{0.5}$$

where:

R_{ij} = Maximum, colinear seismic response of interest (e.g., strain, displacement, stress, moment, shear) corresponding to degree of freedom, i, due to earthquake excitation in direction, j.

R_i = Seismic response of interest for design (e.g., strain, displacement, stress, moment, shear) obtained by the square root of the sum of squares rule to account for the non-simultaneous occurrence of the peak value of R_{ij}'s.

The above process is used for all structures whether axisymmetric or not.

3.7.3.7B Combination of Modal Responses

All piping and equipment analyzed or supplied by GE are evaluated to the methodology of Regulatory Guide 1.92.

In the response spectrum method of modal analysis, if the modes are not closely spaced (i.e., the difference between any two natural frequencies is equal to or less than 10 percent), the modal responses are combined by the square root of the sum of the squares (SRSS) as described in Section 3.7.3.7.1B. Closely spaced modes are combined by the double sum method described in Section 3.7.3.7.2B.

In the time history method of dynamic analysis, the vector sum of every step is used to calculate the combined response. The use of the time history method precludes the need to consider closely spaced modes.

3.7.3.7.1B Square Root of the Sum of the Squares Method

The square root of the sum of the squares (SRSS) combination of modal responses is defined mathematically as:

$$R = \left[\sum_{i=1}^{n} (R_i)^2 \right]^{0.5}$$

where:

R = Combined response

R_i = Response to the i^{th} mode

n = Number of modes considered in the analysis

3.7.3.7.2B Double Sum Method

The double sum method is used to combine the responses of closely spaced modes when the response spectrum method of modal dynamic analysis is used. This method is defined mathematically as:

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| E_{ks} \right]^{0.5}$$

where R is the representative maximum value of a particular response of a given element to a given component of excitation, R_k is the peak value of the response of the element due to the k^{th} mode, and N is the number of significant modes considered in the modal response combination. In addition, R_s is the peak value of the response of the element attributed to s^{th} mode. Also,

$$E_{ks} = \left[1 + \left[\frac{\omega'_k - \omega'_s}{\beta'_k \omega_k + \beta'_s \omega_s} \right]^2 \right]^{-1}$$

in which

$$\omega'_k = \omega_k \left[1 - \beta_k^2 \right]^{0.5} \quad \text{and} \quad \beta'_k = \beta_k + \frac{2}{t_d \omega_k}$$

where ω_k and β_k are the modal frequency and the damping ratio in the k^{th} mode, respectively, and t_d is the duration of the earthquake.

3.7.3.8B Analytical Procedure for Piping

The analytical procedures for piping analysis are described in Section 3.7.2.1.5.1B. Methods to include differential piping support movements at different support points are described in Section 3.7.2.1.4B.

3.7.3.9B Multiple Supported Equipment Components With Distinct Inputs

The procedure and criteria for analysis is described in Section 3.7.2.1.5.3B.

3.7.3.10B Use of Constant Vertical Static Factors

Constant vertical static factors in the analysis of subsystems and components are used as described in Section 3.7.3.5B.

3.7.3.11B Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are discussed in Section 3.7.3.3.1B.

3.7.3.12B Buried Seismic Category I Piping Systems and Tunnels

None in NSSS scope of supply.

3.7.3.13B Interaction of Other Piping with Seismic Category I Piping

When other (non-Seismic Category I) piping is attached to Seismic Category I piping, the other piping is analytically coupled sufficiently so as not to significantly degrade the accuracy of the analysis of the Seismic Category I piping. Furthermore, the other piping is designed to withstand the SSE sufficiently to prevent the Seismic Category I piping failure.

3.7.3.14B Seismic Analysis for Reactor Internals

The modeling of RPV internals has been discussed in Section 3.7.2.3.2B and shown on Fig. 3.7B-2. The damping values are given in Table 3.7B-1. Table 3.9B-2 includes a summary of the loading conditions, evaluation criteria, calculated maximum stresses in the selected locations, and the allowable stresses.

3.7.3.15B Analysis Procedures for Damping

Analysis procedures for damping are discussed in Section 3.7.2.15B.

3.7.4B Seismic Instrumentation

See Section 3.7.4A.

References - 3.7B

1. Liu, L. K. Seismic Analysis of the Boiling Water Reactor. Symposium on Seismic Analysis of Pressure Vessel and Piping Components, First National Congress on Pressure Vessel and Piping, San Francisco, California, May 1971.

2. Singh, A. K., et al. Influence of Closely Spaced Modes in Response Spectra Method of Analysis. Published in ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, December 1973.

3.8 DESIGN OF CATEGORY I STRUCTURES

Seismic Category I structures are the reactor building, auxiliary building, fuel building, control building, standby service water cooling tower and pump house, diesel generator building, and electrical and piping tunnels. The reactor building is comprised of the shield building, containment, and internal structures including the drywell, weir wall reactor pedestal, and the foundation mat. The containment structure is comprised of a steel containment vessel backed by hoop and vertical stiffeners in the lower 20 ft and structural concrete fill in the lower 24 ft 8 in above the mat. The concrete portion of the containment structure is described in Section 3.8.1, while the containment vessel is addressed in Section 3.8.2. The internal structures of the steel containment are covered in Section 3.8.3. The shield building and the remaining Seismic Category I structures are addressed in Section 3.8.4. Finally, all of the foundations of the Seismic Category I structures are described in Section 3.8.5.

The major structures in the reactor building are shown in Figure 3.8-1.

3.8.1 Concrete Containment

The steel containment vessel, described in Section 3.8.2, is backed by structural concrete fill up to elevation 94 ft 8 in (Figure 3.8-1a) in the annulus area between the shield building and the containment vessel. This concrete portion of the containment is structurally anchored to the steel containment vessel and the shield building to form a composite section. Other design details of the structural concrete fill in the annulus up to elevation 94 ft 8 in are covered in Sections 3.8.2 and 3.8.4.

3.8.2 Steel Containment

3.8.2.1 Description of Containment

The steel containment system consists of a steel containment vessel, penetrations, and access openings described in Sections 3.8.2.1.1, 3.8.2.1.2, and 3.8.2.1.3, respectively (Figure 3.8-1).

3.8.2.1.1 Containment Vessel

The containment vessel consists of a continuous, essentially leaktight steel membrane which includes the cylindrical portion, the torispherical dome, and the floor liner plate with embedments. In addition, certain mechanical elements (crane supports, beam seats, weld pads, etc) are supported by the vessel.

Cylindrical Portion and Torispherical Dome

The steel containment vessel is in the form of a vertical cylinder anchored at its base to the concrete mat and closed at the top by a torispherical dome. The bottom portion of the cylindrical shell is backed by structural concrete placed to elevation 94 ft 8 in the entire annulus between the containment and shield building. Above this elevation, the steel shell is free standing and is designed to act as an independent structural component within the reactor building. The steel vessel is provided with two personnel air locks that permit access during normal plant operation and shutdown. The vessel measures approximately 186 ft in height and 120 ft in diameter. The thickness of the cylindrical portion of the vessel and the torispherical dome is determined by analysis and complies with Article NE-3000 of ASME Section III, July 1, 1974 edition.

The lower part of the cylindrical portion of the steel containment vessel also serves as the outer boundary of the volume of water to be stored in that area (suppression pool). The inner water boundary is a weir wall located inside the drywell chamber.

A transition section is used at the junction of the steel containment cylinder and the floor steel liner plate (Fig. 3.8-2) to provide for the continuity of the steel membrane. The lower cylindrical portion of the containment vessel is stiffened both circumferentially and vertically by steel stiffeners and reinforced concrete fill placed in the annulus between the containment and the shield building for the lower 24 ft 8 in.

The interface between the containment cylinder and dome is shown in Fig. 3.8-2a.

Floor Liner Plate

The bottom of the containment vessel consists of steel liner plates welded together and anchored to the top of the mat concrete.

The steel floor plates are 3/8 in thick, except at areas where the transfer of loads requires a reinforced thickness; in those areas it is 1 1/2 in (nominal) thick.

All welded floor steel liner plate seams are covered with continuously welded test channels (Fig. 3.8-2).

These channels are zoned into test areas by dams welded to the ends of the channel sections. The channels are used to check leaktightness of welds during vessel fabrication. The nondestructive examination of the floor liner plate seam welds meets the requirements of Regulatory Guide 1.19, with the following exception: when radiographic, ultrasonic, and/or magnetic particle testing of austenitic stainless steel liner welds is not feasible, liquid penetrant examination is used.

The leaktightness of the leak chase system channel is tested using the pressure drop test method or the vacuum box test method.

Embedments

To maintain the leaktightness of the steel membrane in transferring loads across the floor liner plate to the concrete mat, three different types of embedments are used (Fig. 3.8-2):

1. Corner transition section
2. Bridging bars
3. Mat embedment plates.

A corner transition section is used at the junction of the cylindrical portion of the containment vessel and the floor liner plates. This corner transition section provides for continuity of the steel membrane and is designed to provide adequate anchorage of the vessel to the reinforced concrete mat.

Bridging bars are used to continuously weld the steel floor liner plates to each other and to provide, at the periphery of the plate, anchorage to the concrete mat.

In those areas within the containment vessel where there are interior reinforce concrete walls to be anchored to the concrete mat, mat embedment plates with cadweld sleeves are used to transfer loads to the mat. The cadweld sleeves are welded on both sides of the mat embedment plates.

Crane Supports

A continuous girder support welded to the containment vessel, as shown in Fig. 3.8-9, serves to support the rail on which the polar crane operates. It provides lateral and vertical crane support under all load conditions.

Beam Seats

The transfer of floor loads to the containment vessel is accomplished through beam seats (Fig. 3.8-3).

Weld Pads

The transfer of loads from pipe supports or miscellaneous equipment loads (non-ASME components) to the steel containment vessel is accomplished through weld pads or A500 Grade B tube steel with wall thickness 1/4 in or less attached to the containment.

3.8.2.1.2 Penetrations

Penetrations are used to carry piping, mechanical systems, and electrical wiring through the drywell, containment vessel, and shield building walls.

These penetrations (Fig. 3.8-4 and 3.8-8) can be classified as follows:

1. Piping system penetrations, unsleeved and sleeved

2. Mechanical system penetrations, fuel transfer tube enclosures and CRD removal tube enclosures

3. Electrical penetrations

4. Instrument penetrations.

3.8.2.1.2.1 Piping System Penetrations

Two basic types of penetrations are used for piping systems, unsleeved and sleeved.

Unsleeved Penetrations

These piping penetrations consist of piping installed through any of the three walls: shield building, containment vessel, or drywell. Unsleeved penetrations are used for low-temperature piping systems (temperature of the fluid in the piping is less than 200°F) when only one pipe passes through the penetration.

Unsleeved process piping penetrations are anchored to the reinforced concrete drywell wall when passing through it, or are welded to the containment vessel when passing through the vessel wall. A flexible seal element is used to seal the annular space between the shield building wall and the process piping.

Sleeved Penetrations

These piping penetrations have a sleeve or guard pipe around the process piping. Sleeved penetrations are used for all moderate and high temperature (temperature of the fluid in the piping is more than 200°F) piping systems, carrying both single and multiple piping. The sleeve or guard pipe is attached directly to the containment vessel or through the expansion bellows which are welded to the containment vessel reinforcement plate. The expansion bellows allow differential radial, torsional, and vertical motion between the guard pipe and containment vessel. A flexible seal element is used to seal the annular space between the shield building and the guard pipe. The sleeve or guard pipe is welded to the process piping through the pipe flued head forging (Fig. 3.8-4) located outside the shield building for containment vessel penetrations, and outside the drywell for drywell penetration. Potential leakage through and around containment penetrations is included in the containment leak rate testing, discussed in Section 6.2.6.

Thermally hot piping is insulated to limit the radial heat flow from thermally hot pipe penetrations and to prevent the temperature of the drywell concrete wall adjacent to the sleeve or guard pipe from exceeding 200°F during normal plant operation. Subarticle CC-3440 of ASME Section III, 1977 Edition, Division 2 was used as a guide in establishing the 200°F temperature limit.

When insulation alone is inadequate to maintain the concrete temperature below 200°F, a cooling fin is welded to the sleeve between the flued head and the concrete drywell wall. Fig. 3.8-4 shows a typical penetration with cooling fin attached.

The cooling fins limit the axial heat flow to the concrete resulting from heat conduction through the flued head and the sleeve.

Penetration Forms

Steel forms are used to line shield building openings for all pipe penetrations. These forms are for the attachment of restraints and sealing bellows. Additional forms are provided in the drywell wall to anchor the end of the guard pipe for penetrations shown on Fig. 3.8-4.

Restraints

Lateral restraints are provided at the shield building wall for some sleeved or unsleeved penetrations, to restrict the lateral movement of the guard pipe or process pipe due to dynamic loads and also to limit the stresses and deformations in the process, sleeve, or guard pipes. Wherever required, mid-guard restraints are provided to the guard pipe to limit the stresses and deformations in the process pipe. These restraints (Fig. 3.8-4) are located between the drywell wall and the steel containment.

3.8.2.1.2.2 Mechanical System Penetrations

Fuel Transfer Tube Enclosures

The function of the fuel transfer tube enclosures is to provide adequate protection for the fuel transfer tube as it passes through the four walls of the structures (Fig. 3.8-8). The enclosures are welded to the liners in the fuel pools to prevent leakage of water from the refueling pool and the spent fuel storage pool. The annular space between the ends of the fuel transfer tube and the enclosures are sealed to prevent water from the fuel pools from escaping.

The fuel transfer tube enclosures (Fig. 3.8-8) consist of four separate pipe sleeve sections:

1. Fuel transfer pool section
2. Containment vessel section
3. Shield building section
4. Fuel transfer canal section in the fuel building

The fuel transfer pool section is anchored in the concrete slab and seal welded to the pool liner. A flexible leaktight connection is provided between the enclosure and the upper end of the fuel transfer tube by means of a flanged bellows expansion joint. Slip-on and welding neck flanges are welded to the transfer tube and the enclosure to provide for the attachment of the flanged bellows expansion joint.

The bellows expansion joint provides flexibility to allow for movement of the fuel transfer tube within the enclosures due to differential movements of the buildings. The lower end of the fuel pool enclosure is sealed to the transfer tube by an expansion type sealing element.

A slip-on flange is welded to the outside diameter of the enclosure where the expansion sealing element is installed to provide additional rigidity to the enclosure against the radial forces created by the expansion seal.

The containment vessel section of the fuel transfer tube is welded to a reinforcement plate which in turn is welded to the containment vessel. The upper end of this enclosure section has a welding neck flange welded to it. The flange provides a mounting surface for the blankoff cap (Fig. 3.8-8). The cap is used to seal off the containment during normal operations. During refueling operations, the blankoff cap is removed and a connection is installed on the fuel transfer tube to provide continuity of the tube.

The shield building section is anchored in the concrete wall. The annular space between the fuel transfer tube and the enclosure is sealed off at both ends by an expansion-type sealing element. Slip-on flanges are welded to the outside diameter of the enclosure where the expansion sealing elements are installed to provide additional rigidity to the enclosure against the radial forces created by the expansion seal.

The fuel transfer canal section in the fuel building is anchored in the concrete wall and seal welded to the pool liner. A flexible leaktight connection is provided between the enclosure and the fuel transfer tube at the lower end by means of a flanged bellows expansion joint. Flanges are welded to the transfer tube and the enclosure to provide for the attachment of the flanged bellows expansion joint. In addition to the bolts used for attaching the expansion joint to the flanges, four through bolts are installed with spacer sleeves to limit the downward movement of the transfer tube during the fuel transfer operation. The through bolts limit the upward displacement of the transfer tube when the load in the transfer tube is decreased.

The bellows expansion joint provides flexibility to allow for movement of the fuel transfer tube within the enclosure due to differential movement of the buildings. The upper end of the fuel transfer canal enclosure is sealed to the transfer tube by an expansion-type seal element. A slip-on flange is welded to the outside diameter of the enclosure where the expansion sealing element is installed to provide additional rigidity to the enclosure against the radial forces created by the expansion seal.

CRD Removal Tube Enclosures

The CRD removal tube enclosures consist of two sections: the containment vessel and shield building section, and the drywell wall section.

The first section consists of a sleeve welded to the containment vessel, projecting through the shield building wall, and passing through a larger diameter sleeve anchored in the shield building concrete wall. Expansion-type seals are installed between the CRD enclosure and the shield building sleeve.

The second section is anchored to the drywell wall and projects inside the drywell.

3.8.2.1.2.3 Electrical Penetrations

Electrical conductors penetrating the containment vessel pass through steel pipe sleeves (Fig. 3.8-4). The sleeves are welded into the containment vessel reinforcement plate.

The factory-sealed canister is mounted to the pipe sleeve by a bolted-flange connection. Each containment vessel electrical penetration has provisions for periodic testing for leaktightness.

3.8.2.1.2.4 Penetrations in the Concrete Fill Area

The details of penetrations in the concrete fill area below El 94 ft 8 in between the containment vessel and the shield building wall are shown in Figure 3.8-4a.

3.8.2.1.3 Access Openings

Access openings are used in the drywell, containment vessel, and shield building.

3.8.2.1.3.1 Drywell

The drywell access openings consist of one combination equipment hatch and personnel door assembly, one personnel air lock, and one drywell head (Fig. 3.8-3 and 3.8-5).

Combination Equipment Hatch and Personnel Door Assembly

The drywell combination equipment hatch and personnel door assembly is located at el 100 ft 8 in, azimuth 225 deg. There are four main parts: the personnel door, the hatch flanged head, the body ring and flange, and the monorail. The body ring flange provides for an 11-ft 6-in diameter clear opening. The personnel door has a 3-ft 6-in x 6-ft 8-in clear opening.

Separate double inflatable seals are used on the door. The space between the double seals is capable of being pressurized to door design pressure without the use of test clamps, and both seals are designed to be leaktight under this condition. The door is opened and closed manually. The sealing mechanism is pneumatically actuated and the mechanical latching mechanism is manually operated. The door design permits operation of the door from either side. Loss of the main air supply does not in any way jeopardize the integrity of the seal barriers or of the locking mechanism. A reserve air supply system is designed for 30 days of sealing and locking. In the event of a power supply failure, it is possible to manually operate the door.

The removable equipment hatch flanged head is secured by bolting to the mating body ring flange in the drywell wall. The body ring flange has double O-ring seals with a pressure test connection between the seals. When unbolted, the hatch hangs from a complete monorail system, which allows the hatch to slide sideways along the inner drywell wall, providing clear access through the hatch. There is a floor provided in the body ring 4 ft 11 in below the centerline of the hatch. The body ring is welded to the barrel, which in turn is embedded in the drywell concrete.

Personnel Air Lock

The personnel air lock (Fig. 3.8-5) is located in the drywell wall at el 134 ft 8 in, azimuth 163 deg 30 sec. It consists of three main components: doors, bulkheads, and a rectangular barrel with reinforcing plates at each end of the barrel. Both the barrel and the reinforcing plates are anchored to the drywell wall. The air lock has two doors, which swing away from the lock at each end of the barrel.

The doors have a clear opening of 3 ft 6 in x 6 ft. These doors are electrically coupled and mechanically interlocked, so that one door cannot be opened unless the other is closed and sealed. However, provision is made for deliberate override of the interlock during plant maintenance periods when permitted by administrative procedures and the technical specifications.

The enclosed space between the double seals on each door can be pressurized to the lock design pressure without the use of test clamps, and both seals must be leaktight under this condition.

Drywell Head

The drywell head assembly is part of the drywell structure. It is located directly above the reactor pressure vessel (RPV). Removal of the drywell head provides access to the reactor vessel for inspection and refueling. The head assembly consists of two parts, the removable head and the chimney section, which is anchored to the drywell roof slab. The closure joint between the head and the chimney section is a finger-pin closure, which consists of a meridional tongue and groove arrangement and radial locking pins. The drywell head flange is equipped with double O-ring seals with a leak test tap between the O-rings. The enclosed space between the O-rings can be pressurized to test the leaktight integrity of the head assembly without pressurizing the entire drywell.

3.8.2.1.3.2 Containment Vessel

The containment vessel access openings consist of one equipment hatch, two personnel air locks, and one dome ventilation opening (Fig. 3.8-6).

Equipment Hatch

The containment vessel equipment hatch is located at el 103 ft 9 in, azimuth 225 deg, to provide access for large pieces of equipment being moved from outside the reactor building into the containment vessel. It is welded to the vessel and is equipped with one hatch cover bolted on the outside of the vessel.

The hatch cover is double gasketed with a leakage test tap between the O-rings. The enclosed space between the O-rings will be pressurized to containment design pressure to test for leakage through the seal when the cover is bolted in place.

The equipment hatch cover is provided with a hoist with two-point suspension and a sliding rail for storage. A positive locking device prevents circular swing.

Personnel Air Locks

The containment vessel personnel air locks are located at el 118 ft 0 in, azimuth 135 deg and el 175 ft 0 in, azimuth 315 deg. These air locks are welded to the vessel with a portion of the barrel extending beyond the shield building wall. A flexible link-seal element is used to seal the annular space between the shield building wall and the barrel of the personnel air locks.

•→7

Both personnel air locks are double-closure penetrations. Each closure head is hinged and has two separate inflatable seals mounted on the door. Each seal has a separate air system isolated from the main air supply. Each air system which penetrates the pressure boundary includes a valve qualified to IEEE 323. The enclosed space between the inflatable seals has provisions for pressurizing to the containment design pressure to test the seals for leakage when the door is locked. This leakage test can be accomplished without the use of test clamps. In addition, the interior volume of the personnel air locks can be pressurized to containment design pressure for leak testing the complete air lock.

The personnel airlocks are designed to operate in the manual mode. Both doors are mechanically latched and manually swung, after the latch is released. The doors are also mechanically interlocked so in the event one door is opened, the other door cannot be actuated. In conjunction with the mechanical interlock, a spring loaded mechanical handwheel locking device ensures the mechanical interlock cannot be defeated with the door being operated out of the door frame. Both the mechanical interlock and handwheel locking device ensure that both doors are not opened simultaneously inadvertently.

Both doors are furnished with pressure equalizing valves and a differential pressure (Δ P) personnel protection system. The Δ P system locks the handwheel, on the door being operated, with an electro-pneumatic cylinder using Instrument Air in the "EQUALIZE" position until the Δ P across the door is .5 PSID or less. When the Δ P across the door has equalized, the electro-pneumatic cylinder is de-energized and the handwheel is unlocked. The Δ P system is controlled through a Programmable Logic Controller (PLC). With the electro-pneumatic cylinder de-energized, the operation of the door continues. If Instrument Air is not available, the electro-pneumatic cylinder fails unlocked.

Category I power is connected to the airlock as an alternate power source. This power is to ensure a reliable power source is available to the airlocks. Electric power to the airlock is not required to operate the airlocks. In the event of a power failure, it is possible to operate the airlock.

•→12

The airlock design is such that a failure of two devices/systems (double failure) is required to place the doors in a condition where both doors could be opened simultaneously. So, the single failure criteria is satisfied.

7←• 12←•

Dome Ventilation Opening

A containment vessel dome ventilation opening is installed at the apex of the containment vessel.

The dome ventilation opening is used for ventilation during construction only and is closed permanently afterwards. The closure of the dome ventilation opening is accomplished by welding a 32-in diameter flanged and dished head to the apex of the containment vessel.

3.8.2.1.3.3 Shield Building

The shield building access openings (Fig. 3.8-7) consist of one equipment hatch, two personnel doors, and one construction dome ventilation opening.

Equipment Hatch

The shield building equipment hatch is located at azimuth 225 deg (northeast) and provides access for equipment installation and removal. The access opening is 21 ft 0 inches in diameter and is enclosed by two 10 ft 10 3/4 in wide x 21 ft 11 1/2 in high rectangular steel panels. In addition to resisting tornadic or seismic forces (whichever are more critical), in combination with other gravity forces including the effects of natural phenomena, the steel door panels are also designed to prevent perforation by the postulated tornado-generated missiles identified in Section 3.5. The panels are mounted on the outside of the shield building wall using swing-open arrangement. The door hinges connect the panels to the stiffened section of the shield building concrete wall through the embedded metal frame. The door panels are furnished with perimeter gaskets at head, jambs, and sill to provide leaktightness.

Personnel Doors

The shield building personnel doors are located at el 118 ft 1 in, azimuth 125 deg and at el 99 ft 10 in, azimuth 305 deg. The door frames are anchored in the shield building wall. Each door is secured by four manually operated latches and is fitted with a compression seal in order to make it leaktight.

Dome Ventilation Opening

A shield building dome ventilation opening is located at the apex of the shield building during construction. After ventilation equipment has been installed in the shield building, the opening is permanently closed with concrete.

3.8.2.2 Applicable Codes, Standards, and Specifications

3.8.2.2.1 Containment

3.8.2.2.1.1 Steel Vessel

ASME Codes

The following sections of the ASME Boiler and Pressure Vessel Code, July 1, 1974, edition are used:

1. Section II, Material Specifications, Parts A and C - All steel materials used in the containment vessel conform to the requirements of this section.

2. Section III, Nuclear Power Plant Components - Subsection NE of Section III is used for the design, fabrication, examination, and testing for the containment vessel cylindrical section and dome (Section 3.8.2.3) with the exception of referenced NA requirements such as Certificates of Authorization, Authorized Nuclear Inspection, Code Data Reports, and Code N Stamping. The containment vessel is not a code-stamped vessel because of the membrane floor.

AISC Construction Manual

AISC Code, Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, dated February 12, 1969, Supplement No. 1, November 1970, and Supplement No. 2, December 1971, is used in the design of nonpressure-retaining components such as beam seats, crane supports, and other structural steel components.

Code of Federal Regulations (CFR)

Containment vessel leakage rate tests are performed in accordance with the requirements of Appendix J, 10CFR50. Details of the type A test performed are covered in Section 6.2.6.

3.8.2.2.1.2 Concrete Portion of the Containment Structure

ASME Codes

Section III, Division 2, Concrete Reactor Vessels and Containments, Subsection CC-3000 of the ASME Boiler and Pressure Vessel Code, July 1, 1977 Edition is used for the analysis and design of the structural concrete fill in the composite section of the containment.

ACI Codes

Chapter 17 of the ACI Code 318-77 is used for the design of the interface between the shield building and the concrete fill in the annulus.

3.8.2.2.2 Penetrations

ASME Codes

The following sections of the ASME Boiler and Pressure Vessel Code, July 1, 1974 edition are used:

1. Section II, Material Specifications, Parts A and C - Materials used on the penetrations conform to the requirements of this section.

2. Section III, Nuclear Power Plant Components - Subsections NB (for Class 1 piping systems), NC (for Class 2 piping), and NE (for Class MC components) are used for the design, fabrication, examination, inspection, testing, and specification of materials for the piping penetrations. The bellows used for the piping penetrations are designed in accordance with ASME Code Section III, Article NE-3000. The containment portions of the electrical penetrations are designed and code stamped to ASME Section III Class MC requirements. Drywell penetrations associated with the reactor cavity drain lines and Neutron Monitoring Systems' Traversing Incore Probe (TIP) drives are not designed to ASME Section III.

Code of Federal Regulations (CFR)

The penetration leak detection and leak rate testing are performed in accordance with the requirements of Appendix J 10CFR50. Details of the type B test performed are covered in Section 6.2.6.

IEEE 317

IEEE 317 - 1976 and ASME Section III Class MC are used for the design, construction, testing, and installation of electrical penetrations.

3.8.2.2.3 Access Openings

ASME Codes

The following sections of the ASME Boiler and Pressure Vessel Code, July 1, 1974 edition are used:

1. Section II, Material Specifications, Parts A and C - Materials used on the structural portion of the access openings conform to the requirements of this section.

2. Section III, Nuclear Power Plant Components - Subsection NE of Section III is used for the design, fabrication, examination, inspection, testing, and specification of materials for the structural portion of the access openings as described in Section 3.8.2.3, with the exception that the containment vessel equipment hatch, drywell combination equipment hatch and personnel door assembly, drywell head, and flange will not be code stamped.

To preclude distortions on an insert plate in a localized area on the containment vessel equipment hatch, postweld heat treatment following a weld repair as required by ASME III, paragraph NE-4642, was not performed. Since the actual stresses in the repaired area of the containment vessel equipment hatch are approximately 60 percent of the allowable stresses under the worst design conditions, this exclusion is justified.

The following sections of the ASME Boiler and Pressure Vessel Code, July 1, 1977 edition and all addenda through Winter 1978 are used:

1. Section II, Material Specifications, Part A - Materials used on the air system of the access openings conform to the requirements of this section.

2. Section III, Nuclear Power Plant Components - Subsection NC of Section III is used for the design, fabrication, examination, inspection, testing, and specification of materials for the air systems of the access openings as described in Section 3.8.2.3, with the exception that the air system on the drywell combination equipment hatch and personnel door assembly and drywell personnel airlock will not be code stamped.

•→2

3. Section III, Code Case N-192, Revisions and Additional Requirements in Accordance with Regulatory Guide 1.84. (Use of Flexible Hose for Section III, Division 1, Class 1, 2, and 3 Construction) is used for the construction of the flexible hoses of the access opening air systems.

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AISC Construction Manual

AISC Code, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, dated February 12, 1969, Supplement No. 1, November 1970, Supplement No. 2, December 1971, and Supplement No. 3, June 12, 1974 (for shield building equipment hatch only) is used for the design of nonpressure-retaining components.

Codes of Federal Regulations (CFR)

The access opening seal leak detection and leak rate testing are performed in accordance with the requirements of Type B testing in Appendix J, 10CFR50. Details of the Type B tests performed are covered in Section 6.2.6.

ASME Code Stamping

The containment vessel personnel air locks are Code stamped Class MC.

IEEE 323

IEEE 323-1974 is used to qualify isolation valves for the main air supply valves.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 Containment Vessel

Cylindrical Portion and Dome

The free-standing portion of the containment vessel (above elevation 94 ft 8 in) is subjected to the loads and loading combinations described in Table 3.8-1. The design is in accordance with ASME Section III, Subsection NE-3000. The lower cylindrical portion (below elevation 94 ft 8 in) is included as a part of the composite section (with the shield building and concrete annulus fill). The steel portion of the composite section is analyzed for the same loads and load combinations as the upper portion of the steel shell.

The containment is protected from pipe rupture jet and reaction forces (Section 3.6) and from concentrated missile loads (Section 3.5). Flooding of the containment to a level of 6 ft 10 in above the core (el 138 ft 5 in) may be necessary for post-accident recovery. Design load combinations for this event are listed in Table 3.8-1, Design III. Design III (Flooded + OBE) stress limits, which are higher than those outlined in Regulatory Guide 1.57, are justified as follows:

1. There is an extremely low probability of required flooding.

2. The limits used are comparable to those for other extreme loading conditions with respect to safety margin against rupture of the shell, thereby against loss of containment function.

3. The limits are consistent with, or lower than, past practice for the flooded condition.

Containment pressure is not included in the Design III load combinations (Table 3.8-1) because there is no potential for pressurization of the containment following flooding. The containment atmosphere, prior to and following containment flooding, is cleaned to permit personnel entry for recovery operations. The systems used for containment cleanup are also used to maintain the containment pressure equalized with atmospheric pressure, resulting in no pressure loads on the containment.

Floor Liner Plate

This part of the containment vessel is subjected to the loads and loading combinations described in Table 3.8-2. ASME Section III, Division 2, is used as a design guideline.

Embedments

The corner transition section is subjected to the loads and loading combinations described in Table 3.8-1. Its design is in accordance with ASME Section III, Article NE-3000.

The bridging bars in the suppression pool area are subjected to loads exerted by the mat liner. The loads on the other bridging bars and the embedment plates are negligible.

Crane Supports

These parts of the containment vessel are subjected to the loads and loading combinations described in Table 3.8-3.

Since they are not pressure-retaining parts, their design utilizes the AISC specification. The part of the support that is attached to the containment vessel is designed using Table 3.8-1 and ASME Section III, Article NE-3000 and paragraph NE-4430.

Beam Seats

Same as for crane supports.

Weld Pads

Same as for crane support except that materials for nonpressure parts or pads which are permanently attached to the containment vessel by welding shall meet the requirements of NE-4431 without the following requirement:

> Nonpressure material shall extend the lesser of 16 times the thickness of the attachment or 4 inches from the weld joint before material which is neither impact tested nor postweld heat treated is welded to it.

3.8.2.3.2 Penetrations

3.8.2.3.2.1 Piping System Penetrations

Unsleeved

Pressure-retaining parts of the unsleeved piping system penetrations are subjected to the loads and loading combinations described in Table 3.8-4. These parts are designed in accordance with ASME Section III. For the pipe, which is Code Class 2 (all Code Class 1 piping is sleeved), the design methods prescribed by Article NC 3000 are used. In the case of more than one code class within the penetration, the highest code class is selected for the interface boundary in the determination of the applicable ASME subsection.

Sleeved (Other than Guard Pipes)

Pressure-retaining parts of the sleeved piping system penetrations are subjected to the loads and loading combinations described in Tables 3.8-4 through 3.8-6. These parts are designed in accordance with ASME Section III. For the pipe, including its flued fittings, either Article NB-3000 or NC-3000 is used, depending on the code class of the pipe. In the case of more than one code class of pipe within a penetration, the highest code class is selected for the interface boundary in determination of the applicable ASME subsection. The bellows and the remaining pressure-retaining parts, including the guard pipes, are designed in accordance with ASME Section III Article NE-3000. Additionally, the bellows are designed for the loading combinations and maximum displacements described in Table 3.8-8.

Guard Pipes

Guard pipe design criteria and loading combinations are presented in Appendix 3D.

3.8.2.3.2.2 Mechanical System Penetrations

Fuel Transfer Tube Enclosure

The only parts of the fuel transfer tube enclosure that function as parts of the containment boundary are the sleeve passing through the containment vessel and the blankoff cap, which is bolted to the end of the sleeve inside the containment during power operation. These parts are subjected to the loads and loading combinations described in Table 3.8-6. Their design is in accordance with ASME Section III, Article NE-3000.

The remaining parts of the fuel transfer tube enclosure are designed to withstand the hydrostatic loads of water applied to them, thermal effects, and differential building motions associated with the earthquake loadings (Section 3.7).

Control Rod Drive Tube Enclosure

For the CRD removal tube enclosure, two sections perform containment functions:

1. The first section includes the sleeve that passes through the containment vessel and the blankoff cap, which is bolted to the end of the sleeve inside the containment during power operation. These parts are subjected to the loads and loading combinations described in Table 3.8-6. Their design is in accordance with ASME Section III, Article NE 3000.

2. The second section includes the sleeve that extends beyond the drywell wall and the blankoff cap, which is bolted to the sleeve inside the drywell. These parts are subjected to the loads and loading combination described in Table 3.8-6. Their design

is in accordance with ASME Section III, Article NE-3000.

3.8.2.3.2.3 Electrical Penetrations

The mechanical components of electrical penetrations through the containment are subjected to the mechanical loads and loading combinations described in Table 3.8-6. As required by Subarticle NE-3720 of ASME Section III and IEEE 317, the design of the electrical penetrations is in accordance with Article-NE 3000 of ASME Section III.

3.8.2.3.3 Access Openings

Drywell

The combination equipment hatch and personnel door assembly, personnel air lock, drywell head, and flange are subjected to the loads and loading combinations described in Table 3.8-1.
•→2
The structural portion of the combination equipment hatch and personnel door assembly, personnel airlock, drywell head, and flange are designed in accordance with ASME Section III, Article NE-3000. The air system of the combination equipment hatch and personnel door assembly and personnel airlock is designed in accordance with ASME Section III, Article NC-3000, and ASME Section III Code Case N-192, Revisions and Additional Requirements in Accordance with Regulatory Guide 1.84.
2←•
The following information is required by Regulatory Guide 1.84.

Data to demonstrate compliance to NC-3649.4(e) is contained in References 4 and 5. The design pressure, maximum operating pressure, and pressure-temperature rating for the flexible hoses is 150 psig, 120 psig, and 150 psig at 330°F respectively.

Containment Vessel

The equipment hatch and personnel air locks are subjected to the loads and loading combinations described in Table 3.8-1.
•→2
The equipment hatch and structural portion of the airlocks are designed in accordance with ASME Section III, Article NE-3000. The air system of the airlocks is designed in accordance with ASME Section III, Article NC-3000, and ASME Section III Code Case N-192, Revisions and Additional Requirements in Accordance with Regulatory Guide 1.84.
2←•

The following information is required by Regulatory Guide 1.84.

Data to demonstrate compliance to NC-3649.4(e) is contained in References 4 and 5. The design pressure, maximum operating pressure, and pressure-temperature rating for the flexible hoses is 150 psig, 120 psig, and 150 psig at 330°F respectively.

Shield Building

Shield building equipment hatch and personnel doors are designed in accordance with AISC specifications. The applicable loads and loading combinations from Section 3.8.4.3 are used.

3.8.2.4 Design and Analysis Procedures

The computer programs discussed in the following sections are explained in more detail in Appendix 3A. Certain portions of these computer programs are considered proprietary to SWEC. The seismic analyses follow the procedures described in Section 3.7.

3.8.2.4.1 Containment Vessel

Cylindrical Portion and Dome

The cylindrical portion and dome are analyzed using computer code SHELL 1 for thin shells of revolution asymmetrically as well as axisymmetrically loaded. Included in the analysis of the cylindrical portion and dome are the effects of dead and live loads, internal and external pressures, temperatures, earthquake loads, and the hydrostatic load of water in the suppression pool. The containment is analyzed for hydrodynamic loads as described in Appendix 6A.

For combining responses due to dynamic loads applied to the containment, the following procedure is used. Loads due to the three directions of earthquake excitation are combined by the SRSS method, as described in Regulatory Guide 1.92. These seismic loads are combined with other dynamic loads by the absolute sum method.

The boundary conditions at the base of the cylinder are taken from mat analysis (Section 3.8.5). The lower portion of the cylinder is considered a part of the composite section acting integrally with reinforced concrete. This section is analyzed as a composite shell with orthotropic stiffness properties for all applied loads. The effect of discontinuties at elevations 70 ft and 94 ft 8 in for the containment vessel are considered in the analyses. Temperature effects due to both operating and accident conditions are considered. In addition, thermal growth of surface steel plates is restrained both radially and vertically by the concrete fill. These loads on the composite section are included in the design. Nonlinear temperature distributions through the composite section are converted to an equivalent linear distribution to simulate the effects caused by the nonlinear temperature distribution. Properties of the section are dependent on the state of stresses in the wall, i.e., amount of cracking. Therefore, the procedure used in the analyses of the composite section is an iterative technique.

The effects from penetrations, access openings, beam seats, and the crane support are local in nature and are not considered to affect the overall analysis. These localized effects are analyzed individually as described in the following sections. The containment vessel is reinforced around penetrations and access openings to reduce any stress concentration effects due to localized loads. The effects of significant nonaxisymmetric and transient loads are considered in analyses. The stresses in the containment shell remain within their allowable limits under all localized, nonaxisymmetric, and transient loads due to localized loads. The effects of significant nonaxisymmetric and transient loads are considered in analyses. The stresses in the containment shell remain within their allowable limits under all localized, nonaxisymmetric, and transient loads.

Fatigue analysis requirements for the steel containment cylinder and dome are evaluated in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section III, Division I, Subsection NE.

The design of the steel containment cylinder and dome is in compliance with the NRC Staff's interim criteria which states:

Under normal operating condition, the steel containment should maintain a minimum of 3.0 safety factor for all loading combinations. The safety factor (S.F.) is defined as follows:

$$\text{S.F.} = \frac{\text{Buckling strength of the containment shell}}{\text{Buckling load imposed on the shell}}$$

When design basis accident loads are considered, the safety factor should be a minimum of 2.0.

The safety factors against buckling are calculated based on ASME Code Case N-284 dated August 25, 1980.

The potential for instability (buckling) of the containment vessel is analyzed using a linear elastic finite element or finite difference computer code for the loading combinations specified in Table 3.8-1.

Floor Liner Plate

The floor liner is analyzed by plate theory for the loading combinations specified in Table 3.8-2. The design is based on strain criteria. The effect of concrete displacements is considered in calculating maximum strains.

Fatigue analysis requirements for the floor liner plate are evaluated in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Division 2.

Embedments

The corner transition section is analyzed using finite element analysis techniques. Included in the model loading are the restraining effects of the concrete surrounding the steel plates that make up the corner transition section, as well as the effects of dead and live loads, internal and external pressures, temperatures, earthquake loads, and the hydrostatic load of water in the suppression pool (Table 3.8-1). Boundary conditions for the corner transition section are taken from the model of the cylindrical portion and dome.

The plate to which the containment vessel is anchored at the corner transition section is subjected to tensile loads across its thickness. Limiting these stresses to those given in Table 3.8-1 ensures that no laminations occur in the plate.

Crane Supports

The analysis of the crane supports uses hand calculations. The analysis of the local region of the containment vessel to which the crane supports are attached is performed using finite element analysis techniques. Boundary conditions for this section are taken from the model of the cylindrical portion and dome.

Beam Seats

The analysis of beam seats uses hand calculations and is based on strength of material methods.

Weld Pads

Same as for beam seats.

3.8.2.4.2 Penetrations

Piping System Penetrations

Unsleeved

Unsleeved piping system penetrations are analyzed using computer codes such as SHELL 1 and ASAAS. Both of these programs, SHELL 1 for thin shells of revolution and ASAAS for arbitrary axisymmetric solids, have the capability to include the effects of asymmetric loads. This is essential to calculate the effects of the pipe loads. Other effects include those of dead weight, pressure, and temperature. The process pipe is assumed to be fixed in the drywell.

Sleeved

Sleeved piping system penetrations are analyzed in much the same manner as the unsleeved. Exceptions include the fact that thermal conditions are analyzed using a computer code such as TAC2D (thermal analysis code-2 dimensional) to solve for the temperature distributions in critical regions of the penetration assemblies such as around the flued heads. A temperature distribution study of the drywell wall, around the sleeve, was made to ensure that the temperature of concrete in the region does not exceed 200°F.

Mechanical System Penetrations

The fuel transfer tube enclosure and the CRD removal tube enclosure are analyzed using computer codes such as SHELL 1 and ASAAS. The codes enable the user to include the effects of the concrete and associated reinforcement bars. The effects of dead load, pressure, temperature, and earthquake are also included.

Electrical Penetrations

Same as for unsleeved piping penetrations.

3.8.2.4.3 Access Openings

Drywell

The combination equipment hatch and personnel door assembly, personnel air lock, and drywell head are analyzed and designed in accordance with ASME Code Section III, Subsection NE. Nonpressure-retaining structures, such as the monorail assembly, are designed to the requirements of AISC specifications. Classical strength of material methods and finite element techniques using computer code ANSYS are used for static and dynamic analyses.

Containment Vessel and Shield Buildings

The access openings in the containment vessel and shield building except the shield building equipment hatch are analyzed in the same manner as those in the drywell. The shield building personnel door and equipment hatch are analyzed using plate theory and other classical strength of material techniques. The door panels of the shield building equipment hatch are also analyzed for the missiles generated during tornadic events, using the methods described in Section 3.5. The design of these panels, hinges, and embedment (anchors) uses the applicable loads and loading combinations described in Sections 3.3.2 and 3.8.4.3.

3.8.2.5 Structural Acceptance Criteria

3.8.2.5.1 Containment Vessel

Cylindrical Portion and Dome

See Table 3.8-1 and Section 3.8.2.3.1.

Floor Liner Plate

See Table 3.8-2 and Section 3.8.2.3.1.

Embedments

See Table 3.8-1 and Section 3.8.2.3.1.

Crane Supports

See Table 3.8-3 and Section 3.8.2.3.1.

Beam Seats

See Table 3.8-3 and Section 3.8.2.3.1.

Weld Pads

See Table 3.8-3 and Section 3.8.2.3.1.

3.8.2.5.2 Penetrations

Piping System Penetrations

Unsleeved

See Table 3.8-4 and Section 3.8.2.3.2.1.

Sleeved

See Tables 3.8-4 through 3.8-6 and Section 3.8.2.3.2.1.

Mechanical System Penetrations

Fuel Transfer Tube Enclosure

See Table 3.8-6 and Section 3.8.2.3.2.2.

CRD Removal Tube Enclosure

See Table 3.8-6 and Section 3.8.2.3.2.2.

Electrical Penetrations

See Table 3.8-6 and Section 3.8.2.3.2.3.

3.8.2.5.3 Access Openings

Drywell

See Table 3.8-1 and Section 3.8.2.3.3.

Containment Vessel

See Table 3.8-1 and Section 3.8.2.3.3.

Shield Building

See Section 3.8.2.3.3.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Containment Vessel

3.8.2.6.1.1 Materials and Quality Control

Material for the containment vessel (i.e., cylindrical shell, vertical and circumferential stiffeners, dome and corner junction) is SA-516, Grade 70, fine grain, normalized, and fully killed. Additionally, corner junction embedment plate material and embedments having cadweld sleeves attached are Lukens Lectrefine steel.

Material for the steel floor liner plates (between the weir wall base plate and the centerline of the reactor building) and bridging plates is SA-516, Grade 60. These plates are ordered to conform with standard mill practice with regard to thickness tolerances.

Toughness tests (Charpy V-notch) are performed for all ferritic materials greater than 5/8 in thick, which form part of the containment vessel, floor liner, and mat embedments. The tests conform to the acceptance criteria of NE-2300.

All ferritic steel plates for the preceding components less than 5/8 in thick, except for backing strips and gas test channels, are impact-tested in accordance with ASTM A-20, and the test temperatures were not more than 0°F. ASME Section III Class 1 and 2 penetration with process piping thickness greater than 5/8 in were impact tested at 40-F or lower in accordance with acceptance criteria of NB-2300.

The plate to which the containment shell is anchored at the corner transition section and all cadweld mat plates (Fig. 3.8-2) are ultrasonically tested in accordance with ASTM A578, Acceptance Level 1 for 100 percent of the plate.

The lower 25 ft of the containment vessel plate is fabricated from SA-516, Grade 70 carbon steel clad with SA-240, Type 304L stainless steel. The floor liner plate between the containment/mat corner junction and weir wall base plate, and the outer drywell suppression pool plate are fabricated from SA-240, Type 304L stainless steel. The clad/stainless steel or stainless plate was selected to minimize suppression pool water corrosion product buildup and to eliminate the need for a protective coating system in the submerged portions of the suppression pool boundaries. This stainless steel pool lining also facilitates decontamination work, if necessary, and provide a maintenance-free pressure boundary at both the drywell and containment walls. The cladding has a 30,000 psi shear strength which serves to preclude delamination of cladding. The provision of stainless steel plates (liner) on the drywell walls and the weir wall ensure that no concrete will be in direct contact with suppression pool water. All clad plates were ultrasonically tested after cladding in accordance with ASTM A578, Supplement 56, performed on 9-in centers grid and accepted in accordance with Level 1.

All materials meet the requirements of ASME Section III, Article NB-2000, NC-2000, or NE-2000 except that the requirements of NE-2000 do not apply to nonpressure-retaining materials such as shafts, stems, bushings, bearings, springs, wear plates seals, packing, gaskets, and cotter pins. When NC-2000 is used, the impact test requirements of NE-2300 also apply.

The fabrication and construction of the steel containment is performed under a quality control program that ensures compliance with the requirements of the 1974 edition (no addenda) of ASME Section III, Division 1, Subsection NE. Compliance with the requirements of Article NE-5000 is ensured by the following measures:

1. Detail drawings used during fabrication and construction of the steel containment specify the nondestructive examinations to be performed on welded joints and are reviewed and approved by the Architect/Engineer for compliance with NE-5000.

2. Welding procedures used during the fabrication and construction of the steel containment are specified on the detail drawings and are reviewed and approved by the Architect/Engineer for compliance with Subsection NE of the 1974 ASME Code.

Copies of the above documents, with the Architect/Engineer's approval shown, are furnished to the quality control inspectors prior to shipment from the shop and prior to final acceptance at the construction site.

The fabrication and construction of the steel containment is performed in accordance with the tolerances given in Subarticle NE-4200 of ASME Section III, Division 1 (1974 edition) except that the requirements of Subparagraph NE-4221.2 cannot be applied as the limits of applicability given therein do not define tolerances for a containment shell of the dimensions used for this containment vessel.

In lieu of the tolerances given in NE-4221.2, the following tolerances for the containment shell are applied:

1. Overall plumbness - ±2 3/4 in, measured at the mid-height of the plates, 30 deg apart.

2. Diameter at each circumferential seam shall be 120 ft 0 in, ±3 3/4 in measured not closer than 12 in from seams, 30 deg apart.

3.8.2.6.1.2 Special Construction Techniques

Containment vessel erection started after the completion of the concrete mat. The cylinder to mat liner corner junction bridging plates were placed with the concrete mat pour.

3.8.2.6.2 Penetrations

3.8.2.6.2.1 Materials and Quality Control

Materials for the different components of the penetrations are listed below.

For each component, codes (shown in parentheses) are used: U for unsleeved penetrations, S for sleeved penetrations, and E for electrical penetrations.

These materials, except ethyl-propylene-diene-monomer (EPDM), meets the requirements of ASME III, Articles NB-2000, NC-2000, and NE-2000. The plates are ordered to conform with standard mill practice with regard to thickness tolerances.

Piping System Penetrations

1. Piping Penetrations

Process Piping

Pipes

24 in or less nominal dia	SA-333, GR 6, seamless, or SA-376, Type 304, or SA-312, Type 304
Greater than 24 in nominal dia	SA-155, Gr C55, C1. 1

Flued head (attachment to guard pipe)	SA-508, Class 1 (Code Case 1332-6; carbon limited to 0.30% max), or SA-182, GR F304
<u>Control Rod Drive System Piping</u>	
Piping (seamless)	SA-312, Type 304, or SA-376, Type 304
Sleeve	SA-334-6
Adaptor	SA-182, Type F304
<u>Instrument Piping</u>	
Piping (seamless)	SA-312, Type 304
Sleeve	SA-334-6
Adaptor	SA-182, Type F304
Sleeves	
42 in dia, 1 1/2 in thick	SA-516, Gr 70
All others	SA-333, Gr 6, or SA-106, Gr B, or SA-516, Gr 70
Anchoring rings (S,U)	SA-516, Gr 70
Shear lugs (S,U)	SA-516, Gr 60
Bellows	SA-240, Type 304L or 316L
Cooling fins	SA-516, Gr 60, or SA-516, Gr 70

2. Electrical Penetrations

Sleeve	SA-106, Gr B, or SA-333, Gr 6 (seamless)
Anchoring rings	SA-516, Gr 70

Flanges	SA-105 or SA-350 LF2

Mechanical System Penetrations

1. Fuel Transfer Tube Enclosures

Fuel Transfer Pool and Fuel Transfer Canal Sections

Pipe sleeve	SA-312, Types 304L or 316L
Welding neck flange	SA-234 made from SA-182, Type F304
Bellows expansion joint	SA-240, Types 304L or 316L, or SA-240, Type 321
Bolts	SA-320-B8 or SA-193-B8
Nuts	SA-194-8
Expansion seal	EPDM

Containment Vessel and Shield Building Sections

Pipe sleeve reinforcement plate	SA-516, Gr 70
Blankoff cap	SA-234, Type A181
Welding neck flange	SA-234, Type A181
Bolts	SA-193-B7
•→3	
Nuts	ASTM A194-2H or A194 GR7
3←•	
Expansion seal	EPDM

2. CRD Removal Tube Enclosures

Shield Building Containment Vessel Section

Pipe sleeve	SA-333, Gr 6

Reinforcement plate	SA-516, Gr 70
Expansion seal	EPDM
Drywell Section	
Pipe sleeve	SA-333, Gr 6
Expansion seal	EPDM

3.8.2.6.2.2 Special Construction Techniques

There are no special construction techniques employed.

3.8.2.6.3 Access Openings

3.8.2.6.3.1 Materials and Quality Control

Following is a listing of the materials for the different components of the access openings for the drywell and containment vessels.

All materials meet the requirements of ASME Section III, Articles NB-2000, NC-2000, and NE-2000, as applicable, except that the requirements of NE-2000 do not apply to shafts, stems, bushings, bearings, springs, wear plates, seals, packing, gaskets, and cotter pins.

The plates are ordered to conform with standard mill practice with regard to thickness tolerances.

Toughness tests (charpy V-notch) are performed for all ferritic materials greater than 5/8 in thick, which form part of the containment vessel equipment hatch and personnel air locks, drywell combination equipment hatch and personnel door, drywell personnel air lock, and attachments to these components. The tests conform to ASME Section III, Class MC, NE-2300 requirements.

All ferritic steel plates for the preceding components less than 5/8 in thick, except for backing strips and gas test channels, are impact-tested in accordance with ASTM A-20.

Drywell

Combination Equipment Hatch and Personnel Door Assembly and Personnel Air Lock

Cylinder	SA-516, Gr 70

Anchoring rings	SA-516, Gr 70
Hatch cover	SA-516, Gr 70
Hatch cover flange	SA-516, Gr 70
Cylinder mating flange	SA-516, Gr 70
Bolts	SA-193-B7
Nuts	SA-194-2H
Gaskets	EPDM or equivalent
Drywell head:	
Cylinder form embedded in concrete slab	SA-240, Type 304
Dome	SA-240, TYPE 304
Cylindrical form (part of head assembly)	SA-516 Gr 60
Flange	SA-240, YPE 304
Gaskets	EPDM or equivalent

The materials listed apply to the combination equipment hatch and personnel door assembly, personnel air lock, and drywell head.

<u>Containment Vessel</u>

Cylinder	SA-516, Gr 70
Reinforcement ring	SA-516, Gr 70
Hatch cover flange	SA-516, Gr 70
Hatch cover	SA-516, Gr 70
Bolts	SA-193-B7
●→3	
Nuts	SA-194-2H or SA-194 GR7
3←●	
Gaskets	EPDM or equivalent

The materials listed apply to the equipment hatch, personnel air locks, and dome ventilation opening.

The materials for different components of the shield building equipment hatch meet the requirements of the AISC Manual of Steel Construction. The door panels and the door hinges are made of ASTM A-36 and ASTM A108 steel, respectively, with ASTM A325 bolts as required.

3.8.2.6.3.2 Special Construction Techniques

Drywell Building

The personnel airlock barrel was placed into position before pouring the concrete. The combination equipment hatch cover is stored inside the drywell. Before pouring concrete, the hatch barrel with ring flange and attachment collar were installed in position and the attachment collar was welded to the drywell steel plate. The drywell head and mating flange were fabricated in the shop; the head and flange were secured together and placed in position prior to pouring the drywell top slab concrete. After the concrete was poured and the mating flange securely anchored, the drywell head was dismounted to proceed with the construction sequence.

Containment Vessel

The equipment hatch cover is stored in the annular space between the containment vessel and shield building. The construction and installation of the dome ventilation opening did not require special construction techniques.

Shield Building

The construction of the equipment hatch, personnel doors, and dome ventilation opening does not require any special construction techniques.

3.8.2.7 Testing and In-service Inspection Requirements

Two types of tests are performed on the primary containment structure: acceptance test and leakage rate test. The acceptance test (Section 3.8.2.7.1.1) is conducted to verify the structural adequacy of the primary containment.

3.8.2.7.1 Containment Vessel

3.8.2.7.1.1 Testing

Structural Acceptance Test

The containment vessel is subjected to a pressure test in accordance with the requirements of Subsection NE-6300 of ASME III, July 1, 1974 edition. The concrete fill in the containment/shield building annulus is not subjected to a structural acceptance test because the stresses and strains in the concrete are calculated to be insignificant under test pressure load.

The containment vessel is internally pressurized in levels up to a test pressure of 1.15 times the design pressure and then depressurized to the design pressure. At the design pressure, the containment vessel is examined for leakage as described below.

Prior to performing the pressure test, the following requirements are required to be completed to ensure structural integrity:

1. The containment vessel floor liner plate seam welds are pressure tested to 20 psig using the leak chase channel system.

2. The containment vessel seam welds are volumetrically examined by either radiography or ultrasonic methods in accordance with ASME III, Subsection NE-5000.

3. The containment vessel seam welds in the concrete fill area are vacuum box tested prior to being embedded in concrete.

The acceptance criteria for the initial structural acceptance test are as described in ASME III, Subsections NE-6300 and NE-6315. The test requires examination of all joints, connections, and high-stress regions after the application of pressure, equal to the greater of the design pressure or three-fourths of the test pressure determined in accordance with NE-6320, for a period of at least 10 min. Any detected leaks are repaired, and the system is retested in accordance with the same requirements.

The containment vessel design incorporates a steel shell attached to a concrete floor. Concrete structures are not covered in ASME III, Division 1; therefore, the vessel is not code stamped.

Leakage Rate Test

The leakage rate test is described in Section 6.2.6.

3.8.2.7.1.2 Inservice Inspection Requirements

Inservice surveillance requirements have not yet been defined by ASME Section XI Code, but they are now under development.

3.8.2.7.2 Penetrations

3.8.2.7.2.1 Testing

Piping System Penetrations

Unsleeved and sleeved piping system penetrations are tested in conjunction with the containment vessel acceptance test. Process pipe is capped off if necessary for the containment vessel structural acceptance testing.

Mechanical System Penetrations

Fuel Transfer Tube Enclosures

Welded joints between the enclosures and pool liners are enclosed with cover plates. These cover plates are pressurized to check the leaktightness of the welds. Flanged joints at the bellows expansion joint are provided with a double O-ring seal. The annular space between these seals is pressurized to test the leaktightness of the flanged joints.

The fuel transfer tube enclosure (Fig. 3.8-8) installed in the containment vessel is sealed off with a blankoff cap. This section is pressure tested during the containment vessel acceptance test.

CRD Removal Tube Enclosure

Pipe sleeve sections that have an expansion-type sealing element are provided with a pressure tap connection. The annular space between the pipe sleeve and the CRD removal tube is pressurized to test the leaktightness of the seal.

The CRD removal tube installed in the containment vessel is sealed off with a blind flange.

Electrical Penetrations

Containment electrical penetrations are tested by pressurizing the space between the flange seals.

3.8.2.7.2.2 Inservice Inspection Requirements

●→12

Inservice surveillance requirements are defined by ASME Section XI Code, subsections IWE and IWL.

12←●

3.8.2.7.3 Access Openings

3.8.2.7.3.1 Testing

Drywell

The personnel air lock is pressurized with air for structural integrity in accordance with the ASME III, NE-6300. The structural overpressure test is conducted at a test pressure of 1.15 times design pressure. In addition, during the structural acceptance test of the drywell, the personnel air lock is subjected to the drywell test pressure of 1.00 x 25 psig on the drywell side.

The combination equipment hatch and personnel door assembly is subjected to the internal test pressure of 25 psig during the drywell structural acceptance test.

Containment Vessel

After complete shop assembly and prior to shipment, the personnel air locks are pressurized with air for structural integrity in accordance with the ASME Section III, NE-6300. The structural overpressure test is conducted at a test pressure of 1.15 times design pressure.

In addition, during the containment vessel acceptance test, the personnel air locks and equipment hatch are subjected to a vessel internal pressure of 1.15 x 15 psig at a minimum test temperature of 70°F.

Shield Building

No formal testing is required for the leaktightness of compression seals provided in the personnel doors. The shield building equipment hatch is not subjected to any specific testing requirement.

3.8.2.7.3.2 Inservice Inspection Requirements

Inservice surveillance requirements have not been defined by the ASME Code Section XI. As such, in-service surveillance is not planned.

3.8.3 Containment Internal Structures

3.8.3.1 Description of the Internal Structures

The containment internal structures (Fig. 1.2-12) are Seismic Category I. The containment internal structures are heavily reinforced concrete walls and slabs, except for the primary shield wall and steel framing members. They are designed to support the principal nuclear steam supply equipment, the refueling pools, and the several floor levels within the containment. The structures are also designed for DBA conditions and are provided for radiation shielding (Chapter 12). The level of radiation does not adversely affect these structures. The containment internal structures include the major components described in the following sections.

3.8.3.1.1 Drywell

The drywell is a 69-ft ID right vertical cylinder 92 ft 3 in high. It is supported by the reactor building foundation mat and is concentric with the reactor pressure vessel's (RPV) vertical centerline. The wall and top slab are 5-ft thick reinforced concrete. The reactor building foundation mat forms the bottom of the drywell.

The lower portion of the cylindrical drywell wall has 129 horizontal vents directed radially through the wall. These vents are arranged in three circumferential rows of 43 equally spaced openings, located vertically over each other, with a nominal diameter of 27 1/2 in. The centerline elevations of the rows of vents are 2 ft 3 in, 7 ft 3 in, and 12 ft 3 in above the base of the drywell wall.

Access to the drywell interior is provided through one sealed personnel air lock (3 ft 6 in x 6 ft) and one sealed combination equipment/personnel hatch with a diameter of 11 ft 6 in. Piping and electrical services pass through the drywell wall in leaktight penetrations. The personnel access lock, combination equipment/personnel hatch, and penetrations are described in Section 3.8.2.

The lower portion below el 93 ft 0 in msl of the drywell wall is lined on both sides with 3/8-in thick stainless steel plates to prevent contact of the suppression pool water with the concrete. The inside face of the upper portion is lined with 3/8-in thick carbon steel plate. It acts as a form during the concrete pour and provides a continuous membrane to inhibit leakage during an accident. The steel plate is anchored to the concrete wall by welded anchors embedded in the concrete. Some of the anchors for the steel plate on the inside face of the wall are welded on threaded inserts. In addition to anchoring the steel plate to concrete, the threaded inserts also serve as support points inside the drywell for pipes, conduits, etc.

The drywell top slab is stiffened by the refueling pool walls which are structurally monolithic with the slab. The slab has a 30-ft 2-in dia opening at its center, which is sealed with a steel cylindrical-torispherical dome drywell head (Fig. 3.8-3).

3.8.3.1.2 Weir Wall

The weir wall is located within the drywell. Its function is to prevent the suppression pool water from entering the interior of the drywell. It is a 2-ft 1-in thick reinforced concrete wall, 21 ft 3 in high, 59 ft 10 in ID right vertical cylinder located concentric with the vertical centerline of the RPV and supported on the reactor building foundation mat. The outside face of this wall in contact with the suppression pool water is lined with 3/8-in stainless steel plates.

3.8.3.1.3 Primary Shield Wall

The primary shield wall surrounds the major portion of the RPV. Its primary functions are to provide radiation shielding and accommodate pipe restraint loads. This wall is 2 ft thick by 25 ft 10 in ID by 46 ft 10 in high. It is located concentric with the RPV centerline and is supported on the reactor vessel pedestal. The wall surfaces are constructed of structural steel plates interconnected by horizontal and vertical stiffeners. The spaces bounded by wall surfaces and stiffeners are filled with nonstructural concrete (Fig. 3.8-10).

3.8.3.1.4 Reactor Pressure Vessel Pedestal

The reactor vessel pedestal is a reinforced concrete right circular cylindrical structure. The wall extends from el 70 ft 0 in (top of mat) to el 100 ft 8 in, at which point it supports the primary shield wall and the reactor pressure vessel skirt. The lower portion of the pedestal (between el

70 ft 0 in and 91 ft 8 in) has an inside diameter of 19 ft 5 1/2 in and a thickness of 4 ft 8 1/4 in. The upper portion (between el 93 ft 8 in and 100 ft 8 in) has an inside diameter of 16 ft 2 in and a thickness of 6 ft 10 in. Between el 91 ft 8 in and 93 ft 8 in, the wall thickness tapers from 4 ft 8 1/4 in to 6 ft 10 in. The pedestal is located concentric with the RPV centerline and is supported on the reactor building foundation mat. The pedestal is anchored to the reactor building mat by anchoring the vertical reinforcing bars through the mat embedment plate. The vertical bars are anchored through the embedment plate by welding two concentric cadweld sleeves to the embedment plate, one to the top and the other to the underside of the plate (Fig. 3.8-11).

The major openings in the pedestal (Fig. 3.8-12) are one CRD removal opening and four CRD piping openings. The CRD removal opening is 2 ft 6 in wide and extends from el 82 ft 6 in to 89 ft 6 in. The four rectangular CRD piping openings are 5 ft 6 in wide and extend from el 94 ft 1 in to 97 ft 2 1/2 in.

3.8.3.1.5 Upper Containment Pool

The upper containment pool (Fig. 3.8-13) is located above, and supported by, the drywell. It is divided into four sections: the fuel transfer and storage area, the separator storage pool, the dryer storage pool, and the reactor cavity. The rectangular dimensions of the upper containment pool are 94 ft 3 in by 35 ft 10 in inside, by 24 ft deep. The walls are constructed of approximately 4-ft thick reinforced concrete with a stainless steel inner liner. The pool is connected to the fuel building by the fuel transfer tube.

3.8.3.1.6 Main Steam Tunnel

The main steam tunnel provides radiation protection from the four main steam lines that are contained within it. In addition, feedwater, RCIC, RHR suction, and RWCU suction lines are contained within the tunnel. The portion of the tunnel located within the containment has a rectangular cross section of 34 ft inside width by 19 ft inside height and extends horizontally from the drywell wall toward the steel containment wall. The tunnel is supported by the drywell wall and is separated from the steel containment by a 3-in rattle space. The bottom of the tunnel is located 40 ft above the top of the reactor building foundation mat. The side walls, top, and bottom of the tunnel are constructed of 4-ft thick reinforced concrete.

3.8.3.1.7 Floors

Floors are located within the containment to provide support for and access to equipment (Fig. 1.2-12). The floors are generally constructed of steel framing with steel grating or checkered plate decks. Some areas, such as the decontamination area at the refueling level, have concrete decks supported on steel framing. Also, some floor areas, where radiation protection is required or where maintenance requires floors other than grating or checkered plate, are constructed of concrete.

3.8.3.1.8 Supports for Reactor Coolant System

Steel linear supports for the reactor coolant system are designed in accordance with Subsection NF of the ASME Code, Section III, Division 1.

3.8.3.2 Applicable Codes, Standards, and Specifications

With the exception of the drywell wall and top slab, the design codes, standards, specifications, and regulations that are used for the design and construction of the containment internal structures are the same as those used for the design of all Seismic Category I structures. They are listed in Section 3.8.4.2.

The concrete pressure-resisting portions of the drywell wall and top slab are designed in accordance with Article CC-3000 of the ASME Code, Section III, Division 2, (1977 Edition) for Concrete Reactor Vessels and Containments, and the steel pressure-resisting portions of the drywell (i.e., drywell head, drywell combination equipment hatch and personnel door assembly, and drywell personnel air lock) that are not backed by structural concrete are designed in accordance with Subsection NE of the ASME Section III, Division 1 Code (July 1, 1974 Edition), as described in Section 3.8.2.

3.8.3.3 Loads and Loading Combinations

With the exception of the drywell wall and top slab, the loads and loading combinations used to design the internal structures are the same as those for other Category I structures. They are given in Section 3.8.4.3.

3.8.3.3.1 Drywell

The loads imposed upon the drywell wall and top slab are in accordance with Table CC-3230-1 of the ASME Code, Section III, Division 2, 1977 Edition, with the addition of appropriate hydrodynamic loadings (Appendix 6A) as follows:

SERVICE LOADS

Test Condition

(1) $D + L + P_t + T_t$

Construction Condition

(2) $D + L + T_o + W$

Normal Operating Condition

(3) $D + L + T_o + R_o + P_v + SRV_1$

Severe Environmental Condition

(4) $D + L + T_o + OBE + R_o + P_v + SRV_1$

(5) $D + L + T_o + W + R_o + P + SRV_1$

FACTORED LOADS

Extreme Environmental Condition

(6) $D + L + To + SSE + Ro + Pv + SRV_1$

(7) $D + L + T_o + W_t + R_o + P_v + SRV_1$

Abnormal Condition

(8.1) $1.0 (D + L + T_{a1} + R_a) + 1.5 (P_{a1} + LOCA) + 1.25\ SRV_2$

(8.2) $1.0 (D + L + T_{a2} + R_a) + 1.5 (P_{a2} + LOCA) + 1.25\ SRV_3$

(9.1) $1.0 (D + L + P_{a1} + T_{a1} + LOCA + SRV_2) + 1.25\ R_a$

(9.2) $1.0 (D + L + P_{a2} + T_{a2} + LOCA + SRV_3) + 1.25\ R_a$

Abnormal/Severe Environmental Condition

(10.1) $1.0 (D + L + Ta_1 + R_a + LOCA + SRV_2) + 1.25 (P_{a1} + OBE)$

(10.2) $1.0 (D + L + T_{a2} + R_a + LOCA + SRV_3) + 1.25(P_{a2} + OBE)$

(11.1) $1.0 (D + L + T_{a1} + R_a + LOCA + SRV_2) + 1.25 (P_{a1} + W)$

(11.2) $1.0 (D + L + T_o + OBE + W + H_a)$

(12) $1.0 (D + L + T_{a2} + R_a + LOCA + SRV_3) + 1.25 (P_{a2} + W)$

Abnormal/Extreme Environmental Condition

(13.1) $D + L + P_{a1} + T_{a1} + SSE + R_a + R_m + R_j + R_r + LOCA + SRV_2$

(13.2) $D + L + P_{a2} + T_{a2} + SSE + R_a + R_m + R_j + R_r + LOCA + SRV_3$

where:

Pt = Pressure during the structural integrity and leak rate tests

Tt = Thermal effects and loads during the test

D = Dead loads, including hydrostatic and permanent equipment loads

L = Live loads, including any moveable equipment loads and other loads which vary with intensity and occurrence, such as soil pressures

W = Loads generated by the design wind

Wt = Tornado loading, including the effects of missile impact

To = Thermal effects and loads during startup, normal operating or shutdown conditions, based on the most critical transient or steady state condition

Ro = Pipe reactions during startup, normal operating or shutdown conditions, based on the most critical transient or steady state condition

Pv = External pressure loads resulting, from pressure variation either inside or outside the drywell.

Safety Relief Valve Discharge Loads

SRV_1 - Safety relief valve discharge loads resulting from any of the following events: one valve (first actuation), one valve (subsequent actuation), 2, 7, 9, or 16 valve blowdown events

SRV_2 - Safety relief valve discharge loads resulting from any of the following events: one valve (first actuation), 2, 7, or 9 valve blowdown events

SRV_3 - Safety relief valve discharge loads resulting from one valve (first actuation) blowdown.

Notes:

1. Safety relief valve discharge loads for multiple valves are for first actuation only

2. Thermal loads due to SRV discharge are treated as T_a for normal operation and T_o for accident conditions.

Design Basis High Energy Pipe Break

The design basis pipe break accident (DBA) is defined here as a large circumferential instantaneous double-ended pipe break of the main steam or recirculation suction line.

The intermediate pipe break accident (IBA) is defined as a break that is less than the DBA but of sufficient size to automatically depressurize the primary system due to a loss of fluid and/or the automatic depressurization system (ADS). The lowest break size of the IBA range is approximately 0.1 sq ft.

The small pipe break accident (SBA) is defined as a break that is not large enough to automatically depressurize the primary system. Accident termination is dependent upon operator action to manually depressurize the reactor.

P_{a1} - Pressure load from IBA or SBA pressure transient including design margin

T_{a1} - Direct temperature loads due to IBA or SBA temperature transient

R_a - Piping loads due to temperature resulting from the high energy pipe breaks

P_{a2} - Pressure loads from DBA pressure transient including design margin

T_{a2} - Temperature load due to DBA temperature transient

LOCA- The worst case dynamic loads resulting from structural vibrations due to the following loads resulting from individually postulated DBA, SBA, or IBA:

1. Condensation oscillation phenomenon
2. Chugging loads (symmetric and asymmetric)
3. Asymmetric annulus pressurization loading on RPV and primary shield wall
4. Pool swell loads (froth impingement and/or drag loads), as described in Appendix 6A
5. Vent clearing loads.

Note: LOCA loads and accident pressure are combined in accordance with their time histories of occurrence.

OBE = Loads generated by the operating basis earthquake

SSE = Loads generated by the safe shutdown earthquake

Ha = Load on the drywell resulting from post-LOCA flooding of containment

Rj = Jet impingement effects on the structure generated by a ruptured high energy pipe

Rm = Missile impact effects on the structure generated by rupture of a high energy pipe

Rr = Reaction loads on the structure generated by a ruptured high energy pipe. (Effects due to rupture of high energy lines are described in Section 3.6.)

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Drywell Structure

The drywell is analyzed and designed for the loading combinations as outlined in Section 3.8.3.3. For axisymmetric and non-axisymmetric load cases, except for the hydrodynamic loads, the drywell wall is modeled together with top slab and is analyzed using SHELL 1, a finite-difference computer program described in Appendix 3A. The discontinuity moments and shears, obtained from the mat analysis (Section 3.8.5.4), are used as boundary conditions for the base of the drywell wall.

The seismic analysis of the reactor building (Section 3.7A) provides accelerations to which the drywell structure may be subjected. These accelerations are applied conservatively as static loading to the drywell shell. The drywell is analyzed for hydrodynamic loads, as described in Appendix 6A.

The discontinuity forces and moments at the base of the drywell wall are calculated assuming that the wall is uncracked vertically and cracked horizontally to the neutral axis of the transformed section. This is a reasonable assumption, because hoop compressive forces exist in the lower portion of the drywell wall, and the large meridional bending moments cause considerable cracking of a vertical cross section. The drywell wall, therefore, has the hoop stiffness of the uncracked concrete section, and the meridional bending stiffness of the transformed section. Some iteration is involved, since the magnitude of the moment at the base influences the bending stiffness, and vice versa. The 129 vent holes are modeled into the shell by reducing the hoop stiffness and meridional bending stiffness.

Arrangements of reinforcing steel at these discontinuities can be found in Fig. 3.8-14 and 3.8-15.

The shell properties are dependent on the state of stress in the wall. For example, in the membrane region where hoop tension exists due to accident pressure which causes the concrete to be cracked, the hoop stiffness is taken to be that of the hoop reinforcing steel only.

Temperature effects due to both operating and accident conditions are considered. In addition, the thermal growth of surface steel plates is restrained both radially and vertically by the drywell wall. These loads on the drywell wall are included in the design. These steel plates are not assumed to assist in resisting loads. Nonlinear temperature distributions through the wall are converted to equivalent linear distributions so that the equivalent linear distribution produces the same uncracked moment about the centerline of the section as does the nonlinear temperature distribution.

Design and analysis for tangential shear is done in accordance with ASME Code (1977 Edition), Section III, Division 2, Article CC-3000, except that the maximum allowable tangential shear stress carried by concrete, V, does not exceed 40 psi for abnormal/severe environmental conditions and 60 psi for abnormal/extreme environmental conditions.

Because the vent holes produce an unusual geometry at the base of the drywell, the design of this region is verified with conservative values of moments and shears. A parametric study was performed which varied the stiffness properties in the meridional and circumferential directions. The study shows that with the most conservative stiffness properties, the stresses in the drywell wall are within ASME III, Division 2, Article CC-3000 stress allowables.

The top portion of the drywell is capped with a flat annular slab which is stiffened by the fuel pool walls and various other appendages. Analysis based on the theory of elasticity for plates and shells is used. Finite element analysis is used to verify the results.

Arrangements of reinforcing steel in the membrane region of the drywell are shown in Fig. 3.8-16. Arrangements of reinforcing steel in the top slab of the drywell are shown in Fig. 3.8-17.

Penetrations through the drywell structures are divided into one of the following three categories:

1. Penetrations 12-in dia or less: No special concrete reinforcement is provided for penetrations 12 in or less in diameter. Penetrations in this category are located to avoid interference with the reinforcing steel wherever possible.

2. Penetrations greater than 12-in dia excluding the access hatches: For penetrations greater than 12-in dia excluding the access hatches, supplementary reinforcement is provided. Reinforcing steel interrupted by the opening is terminated at each side of the opening. Supplementary reinforcement is placed parallel to the bars which are interrupted. Horizontal, diagonal, and vertical bars are used to transfer forces around the opening. The total area of additional reinforcement provided in any direction is not less than twice the area of steel which is interrupted by the opening, with one-half of this supplementary reinforcement placed on each side of the opening, unless a more detailed analysis is performed.

3. The drywell has one personnel access hatch and one combination personnel/equipment hatch. The combination personnel/equipment hatch opening in the drywell is analyzed by means of the three-dimensional finite-element capability of the computer program STRUDL (Appendix 3A). The hatch model consists of a section of wall extending from the center of the opening, vertically and circumferentially, a distance of about 2.5 times the diameter of the opening. Three solid elements are modeled through the thickness of the wall. Applicable loads, as given in Section 3.8.3.3, are applied to the model. Boundary conditions are obtained from drywell shell analysis.

   The personnel access hatch opening has a similar method of analysis except that a two-dimensional finite element model is used. Elements with plane stress and plate bending capability are used. The rectangular configuration of the opening simplifies the reinforcing details because the pattern of meridional and circumferential reinforcement, used in the typical drywell wall, can be easily maintained around the hatch. Additional reinforcing bars as indicated by the analysis are provided.

The structural characteristics of the wall (cracked or uncracked) are modelled using the STRUDL computer program. The steel plate in the suppression pool zone, however, is assumed not to contribute to the structural capacity of the wall for any loading condition other than for the containment structural acceptance test. During an accident, the steel plate is normally in a state of compression due to the sharp temperature rise inside the containment structure and, therefore, adds load to the concrete wall. These loads are included in the design.

To obtain a realistic assessment of the strains, displacements, and stresses in the areas of the hatches, the cylindrical wall is assumed to be fully cracked and to have the extensional stiffness of only the reinforcing bars.

The STRUDL program provides the discontinuity effects on the wall and the pattern of the membrane forces in the region of the hatch openings. Additional reinforcement (ring, circumferential, meridional, and diagonal) is provided in such regions where a significant increase over the typical membrane forces (meridional and circumferential) occurs.

The criteria for steel pressure-resisting portions of the drywell structure are described in Sections 3.8.2.2.2, 3.8.2.3.2, 3.8.2.2.3, and 3.8.2.3.3.

Typical arrangements of the reinforcing bars provided around these hatch openings are shown in Fig. 3.8-18.

3.8.3.4.2 Pedestal

The reactor pressure vessel pedestal is analyzed and designed by the use of computer code SHELL 1, a finite difference computer program described in Appendix 3A, for the loading combinations, as outlined in Section 3.8.4.3.

The discontinuity moments and shears at the mat-pedestal junction are obtained from the mat analysis (Section 3.8.5.4) and are used as bottom boundary condition in pedestal analysis. Forces at the base of the primary shield wall and reactor pressure vessel, resulting from the loads outlined in Section 3.8.4.3, are used as the top boundary condition. Inertial loadings from earthquake are obtained from the dynamic analyses of the reactor building as outlined in Section 3.7.2A. Stress resultants throughout the pedestal are then calculated, for these and other loads (Section 3.8.4.3), using shell equations.

All pipe rupture forces on the reactor vessel are resisted entirely by the reactor pedestal with no consideration given to the stabilizing effect of the refueling bellows seal.

Arrangements of reinforcement at the mat-pedestal junction and the pedestal wall are shown in Fig. 3.8-11 and 3.8-19. The pedestal has been additionally reinforced at CRD openings to ensure continuity and strength. The analysis and design of the additional reinforcement around these openings is performed using curved beam and column theory for loads defined in Section 3.8.4.3.

3.8.3.4.3 Primary Shield Wall

The primary shield wall is analyzed and designed for the loading combinations described in Section 3.8.4.3. Analysis of the wall is performed using the two-dimensional finite elements capability of the STRUDL computer code. Both plane stress and plate bending elements were used. As the structure is sufficiently symmetrical about 0 deg-180 deg axis, only one-half of the structure need be modeled. The use of a 180 deg model allows for the analysis of asymmetric loadings by applying half of the load to the model with symmetric boundary conditions and half with asymmetric boundary conditions. The results of the two analyses are then superimposed for the net results. In addition, classical beam theory and plate and shell theory are used for the analysis and design of local areas. When analyzing for the effects of a LOCA, the following were considered:

1. Jet impingement forces on the primary shield wall

2. Impact loads transmitted to the primary shield wall by any attached pipe rupture restraints

3. Pressurization of the annulus between the primary shield wall and reactor vessel

4. Thermal effects.

Each of the above results in loads which were considered in the design. The loads were combined as appropriate, taking account of the postulated failure locations and types. It has been concluded, because of the dynamic characteristics of the primary shield wall, that peak restraint impact loads are local, impulsive loads on the primary shield wall. These impact loads occur in the first milliseconds after rupture. The shield wall was allowed to yield locally at regions of impact loads provided that:

1. Overall capability of the shield wall to resist elastically to the other forces listed was not affected.

2. Local yielding did not produce effects which jeopardize the safety of other components.

3.8.3.4.4 Weir Wall

The weir wall is designed to resist the loading conditions, as defined in Section 3.8.4.3. The loading conditions include normal operating conditions, the short-term and long-term basis accidents resulting from pipe rupture, safety relief valve discharge loads, and the effects from earthquake. Water and steam pressures, stresses induced from temperature gradients, and base moments and shears due to mat rotations and displacements are considered. The moments, shears, and deflections from the above conditions are calculated by the use of the SHELL 1 program discussed in Appendix 3A. These moments and shears form the basis for the design of reinforcement in the weir wall. The effects of possible jet impingement are also considered. The resistance of the weir wall to concentrated jet impingement forces is evaluated on three levels. The localized surface effects, such as cracking and/or crushing of the concrete, the regional shear stresses due to punching action, and the overall response of the weir wall are considered. These analyses are used to ensure that deflections resulting from jet impingement loads do not impair the function of the pressure-suppression system. The outside surface steel plate is not considered to assist in resisting the loads.

3.8.3.4.5 Upper Containment Pool

The upper containment pool is designed to resist the loading conditions, as defined in Section 3.8.4.3. The analysis uses the finite element capabilities of the STRUDL program in addition to beam and plate theory.

3.8.3.4.6 Main Steam Tunnel

The main steam tunnel is designed to resist the loading conditions, as defined in Section 3.8.4.3. The analysis uses the finite element capabilities of the STRUDL program.

3.8.3.4.7 Floors

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Containment floor framings are vertically supported at the drywell and the steel containment. At the drywell end, the floor beams have pinned connections, whereas at the containment end the beams have sliding supports. This arrangement provides for the minimum amount of load and constraint from floors on the containment. Specific platform beams at elevation 95'-9" are pinned at both ends and are provided with slip connections.

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Drywell floor framings are vertically supported by the drywell and the primary shield wall. At the drywell end, beams have sliding support, whereas at the primary shield wall the beams have pinned connections.

The amplified floor response spectra method is used in analysis of dynamic loadings, such as SRV, seismic, and LOCA. The STRUDL computer program is used to aid the analysis of floors and platforms. The STRUDL computer program is described in Appendix 3A.

3.8.3.5 Structural Acceptance Criteria

Design of interior concrete structures, except the drywell wall and top slab, follows ACI 318-71, using strength design. The basic criterion for concrete strength design is expressed as:

Required Strength ≤ Calculated Strength

All members and all sections of members are proportioned to meet this criterion. The required strength is expressed in terms of design loads, or their related internal moments and forces. Design loads are defined as loads which are multiplied by their appropriate load factors. Calculated strength is that computed by the provisions of ACI 318-71, including the appropriate capacity reduction factors. Capacity reduction factors are taken as given in Section 9.2 of ACI 318-71.

The drywell wall is designed such that the stresses in concrete and reinforcing steel are within the limits specified by ASME Code, Section III, Division 2, Article CC-3000.

The structural steel framing for the floors within the containment and drywell are designed using the loads and loading combinations listed in Section 3.8.4.3. The stresses in the steel members, and the factors of safety, are in accordance with the requirements of AISC steel construction manual, as described in Section 3.8.4.5.

Section 3.7.2.8A describes the variations incorporated into the seismic analysis structural model to account for variations and uncertainties in soil shear modulus and spring constants. Design of the internal structures is based upon the most conservative values resulting from these variations in assumptions and design parameters. Section 3.7.3.6A discusses differential seismic movement relating to interconnected components, systems, and equipment.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The construction materials used for the containment internal structures are the same as those used for other Seismic Category I structures, as described in Section 3.8.4.6.

The material and quality control requirements for the steel linear supports of the reactor coolant system comply with Subsection NF of the ASME Code, Section III, Division 1.

3.8.3.7 Testing Requirements

The drywell is subjected to a structural proof test at design pressure. The drywell is pressurized in approximately four equal increments from atmospheric pressure up to the drywell design pressure and depressurized in a similar manner. At each pressure increment radial deflections are measured at three points along three meridians of the drywell, at one point near the base, one near midheight, and one near the top. In addition, strains are measured on three equally spaced meridians near the bottom of the wall and at midheight. These measurements are compared to predicted values. In addition, at design pressure, air flow into the drywell which is required to maintain pressure is measured along with other thermodynamic variables of interest to determine the effective bypass leakage area [$A/\sqrt{K}$] of the drywell. (See Section 6.2.1 for a description of bypass leakage.)

The strain measurements are taken by dual type strain gauges mounted on "sister" reinforcing bars which are placed in the drywell wall adjacent to actual reinforcing bars.

Displacement measurements are made using direct current displacement transducers (DCDT).

If the pressure drops during the test, the drywell is depressurized to atmospheric pressure and the cause of the pressure drop is corrected. The test is then restarted.

The theoretical displacements and strains are calculated and compared with measured values. A close correlation between measured and calculated values is required for acceptance.

The drywell leak test acceptance criterion is that the measured leakage must be less than 10 percent of the leakage corresponding to an equivalent bypass leakage ($A/\sqrt{K}$) of 1.0 sq ft at design pressure. A test report containing the procedure, description of the instruments and monitoring equipment, location of instruments, and test results is prepared after the test.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

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All other Seismic Category I structures, e.g., auxiliary building, control building, etc, which contain or support safety-related systems and/or equipment are designed to withstand both the SSE and tornado loads, including tornado-generated missiles. Seismic loads are not considered to act simultaneously with tornado loads. Table 3.2-1 identifies Seismic Category I equipment and structures which are tornado protected. Aircraft traffic does not represent a plant hazard at River Bend Station.

In general, Seismic Category I structures are completely independent of adjacent structures. In a few instances, when Seismic Category I structures are integrally connected to the other structures, the Seismic Category I structures are analyzed and designed considering the effect of interconnection and modeling them as a unit. When not connected to the adjacent structures, adequate shake space (i.e., rattle space) is provided between structures to retain their independent functional characteristics, and to allow for rotation, translation, and deformation under seismic loading. The rattle spaces are provided with flexible seals (Fig. 3.8-20). The compressible material in the shake spaces between the reactor building and surrounding adjacent was removed at all levels, except below the top of foundation mat level, prior to plant operation.

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The compressible material in the shake spaces between other Seismic Category I structures was removed, except at the following locations:
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1. Shake spaces between the piping (or electrical) tunnels and the adjacent areas of the auxiliary and radwaste buildings.

2. Shake spaces between the control and diesel generator buildings at all levels.

3. Shake spaces between Seismic Category I structures below the foundation mat level.

The loads, resulting from the compression of the compressible material (that remains permanently in the shake spaces) during an environmental event are considered in designing the walls adjoining the compressible material. In addition to this, the Seismic Category I structures below the design basis flood level (DBFL) are provided with waterstops.

There are no unique materials or new features used in the design or construction of the structures described in this section.

The relative locations of the Seismic Category I structures are shown in Fig. 1.2-1 (Site Plan) and Fig. 1.2-2 (Station Arrangement). The layouts of the Seismic Category I buildings are illustrated in the following figures:

General Arrangement, Reactor Building, Fig. 1.2-9 through 1.2-12

General Arrangement, Auxiliary Building, Fig. 1.2-13 through 1.2-19

General Arrangement, Fuel Building, Fig. 1.2-20 through 1.2-23

General Arrangement, Control Building, Fig. 1.2-24 through 1.2-27

General Arrangement, Standby Service Water Pumphouse, Fig. 1.2-44

General Arrangement, Standby Diesel Generator Building, Fig. 1.2-28.

The major buildings which are not Seismic Category I are illustrated in the following figures:

General Arrangement, Turbine Building, Fig. 1.2-33 through 1.2-37

General Arrangement, Radwaste Building, Fig. 1.2-29 through 1.2-32

General Arrangement, Water Treatment Building, Fig. 1.2-38

General Arrangement, Condensate Demineralizer Regenerative and Off-Gas Building, Fig. 1.2-39 and 1.2-40

General Arrangement, Operating Personnel Access between Buildings, Fig. 1.2-3 through 1.2-8

General Arrangement, Circulating Water Pump Structure, Fig.1.2-41 and 1.2-42

General Arrangement, Makeup Water Pumphouse, Fig. 1.2-45 through 1.2-47

General Arrangement, Circulating Water Cooling Towers, Fig. 1.2-43.

The Seismic Category I structures other than the containment and its internal structural components are described below.

3.8.4.1.1 Auxiliary Building

The auxiliary building houses safety-related equipment including residual heat removal heat exchangers and pumps, core spray pumps, and standby gas treatment equipment. The main steam tunnel passes through and is an integral part of this building. Other piping and electrical cables pass through this building in separate tunnels and connect with adjacent buildings.

The auxiliary building is a Seismic Category I reinforced concrete structure supported on a soil bearing reinforced concrete mat foundation. The exterior walls and roof are a minimum of 2 ft thick and are designed to provide tornado-missile protection. The reinforced concrete steam tunnel walls, floor, and roof protect the equipment outside the tunnel from the effects of a postulated steam line break within the tunnel. The effects of postulated steam line breaks are described in Section 3.6A.

The auxiliary building is located immediately adjacent to and south of the reactor building. The turbine building adjoins the auxiliary building immediately to its south. Seismic rattle spaces are provided at the interface between this building and both the reactor building and the turbine building. Access to the building is provided from outdoors at station grade, the turbine building, the electrical tunnel, and the reactor building.

3.8.4.1.2 Fuel Building

The fuel building houses the new fuel, spent fuel, spent fuel storage or shipping casks, associated handling equipment, fuel pool cooling and cleanup system equipment, and fuel building air filtration equipment.

The fuel building is a seismic Category I reinforced concrete structure founded on a soil bearing reinforced concrete mat foundation. The spent fuel storage pool has approximately 4-ft to 11-ft thick concrete walls lined with stainless steel plates. The structural components of the fuel building are designed for tornado protection.

The spent fuel cask pool is separated from the spent fuel pool by a reinforced concrete wall. The spent fuel storage or shipping cask is not moved over the spent fuel because the crane which lifts the cask is prevented by location from traveling over the spent fuel. This prevents damage to stored spent fuel or to the boundaries of the spent fuel pool due to a spent fuel cask drop. Thus, water retention ability of the spent fuel pool is not impaired by a dropped fuel cask.

The fuel building is located immediately adjacent to and north of the reactor building. A seismic rattle space is provided between the building and the reactor building. The fuel transfer tube allows direct transfer of spent fuel from the reactor containment into the fuel building spent fuel pool, while keeping the fuel building structurally independent of the reactor building. Personnel access to this building is from outdoors at station grade.

The spent fuel storage pool, the spent fuel cask pool, and the fuel transfer canal are provided with a stainless steel liner. The liner is further described in Section 9.1.2.2.

3.8.4.1.3 Control Building

The control building is a Seismic Category I structure. The lowest level is located below grade and is connected to the electric tunnel west of the control building. This level is used to house air handling units and is also used as a cable spreading area. The ground level houses Div. I, Div. II standby switchgear, and air-conditioning equipment. The third level houses heating, ventilating, and air conditioning equipment, batteries, battery chargers, and Div. III switchgear. The fourth level houses the main control room and its associated equipment. Vertical cable chases within the building are enclosed within protective walls.

The exterior walls and roof are constructed of a minimum of 2 ft thick reinforced concrete and are designed to provide tornado-missile protection. The interior floors have concrete decks supported on steel framing. The building is supported on soil bearing reinforced concrete mat foundation.

This building is located east of and adjacent to the auxiliary building. However, it is structurally independent. Personnel access to and from the control building is provided via an east-west passageway at el 123 ft 6 in leading to the services building, turbine building, and auxiliary control building. Access at ground level is from the diesel generator building and outdoors.

3.8.4.1.4 Standby Service Water Pumphouse

The ultimate heat sink consists of a Seismic Category I cooling tower and a Seismic Category I water storage and pump facility (described as standby cooling tower). This tower has four separate cells and has the capacity to dissipate the maximum heat released in an orderly shutdown or an accident condition. The ultimate heat sink is described in Section 9.2.5.

The standby cooling tower is located northwest of the fuel building.

The standby cooling tower and its basin is a reinforced concrete structure supported on a soil bearing mat. The tower uses a multicell design fill of dense vitreous clay.

The exterior walls of the standby cooling tower's pumphouse and roof are a minimum of 2 ft thick reinforced concrete and are designed to prevent damage to the safety-related components during tornadic events.

3.8.4.1.5 Diesel Generator Building

The diesel generator building is a seismic Category I, structure enclosing the three diesel generators and their associated equipment. The building is divided into three separate rooms constructed of reinforced concrete walls and roof. Each room houses one diesel generator. The divider walls are 2 ft thick and have a 3-hr fire rating. There are no major openings in these walls. These walls are designed as Seismic Category I, load bearing shear walls in accordance with Chapter 14 and Appendix A.8 of ACI 318. The loads and loading combinations (including seismic and tornadic events) used in the analysis and design of these walls, are the same as those listed in Section 3.8.4.3 for reinforced concrete structures. Three fuel oil storage tanks are located in the lower level of the building, covered with sand, with their fuel oil pumps housed in the individual diesel generator rooms.

The diesel generator building is a reinforced concrete structure founded on soil bearing mat. The exterior walls and roof are a minimum of 2 ft thick and are designed to provide tornado-missile protection. All ventilation intakes are arranged to preclude penetration from tornado-generated missiles. The diesel generator exhaust, including the muffler, is also arranged to provide protection from tornado missiles. The building is located north of the main control building. It is separated from the control building by a seismic rattle space. Personnel access to the building is provided from the outdoors at station grade.

3.8.4.1.6 Shield Building

The shield building protects the steel containment from tornado winds and missiles and other environmental effects. It also provides biological shielding in the event of a loss-of-coolant accident. This structure, or building, completely encloses the containment. It is a right vertical cylinder capped with a spherical segment dome and supported on the reactor building mat. The cylindrical portion of the structure and dome are 2 ft 6 in and 2 ft thick respectively. The inside radius of the cylindrical portion is 65 ft and the dome has an inside radius of 85 ft 3 in. It is separated from adjacent structures with a seismic rattle space.

The annulus between the shield building and steel containment is filled with reinforced concrete to elevation 94 ft 8 in (Figure 3.8-1a). To ensure that the shield building concrete and the concrete fill act compositely, the shield building inside surface is roughened and a positive mechanical anchoring system across the shield building/concrete fill interface is provided as shown in Figure 3.8-1b. In order for the concrete fill to transfer horizontal forces, a shear key is cut into the mat.

The shield building is a Seismic Category I reinforced concrete structure. Steel framing is used to support the dome during its construction; however, the dome concrete is designed to withstand all loading conditions without the aid of the steel framing when it is completely constructed. The steel framing is designed to withstand all loading and to remain permanently in place.

Personnel access to the shield building is through two personnel doors leading to the auxiliary building and fuel building (Section 3.8.2). The shield building has one equipment hatch, which provides access for large pieces of equipment being moved into or out of the reactor building.

3.8.4.1.7 Electrical Tunnels and Piping Tunnels

Seismic Category I electrical tunnels and piping tunnels (Fig. 1.2-2) contain seismic Category I systems and are constructed of reinforced concrete. The tunnel walls and roof are either of sufficient thickness to resist penetration by tornadic missiles, or the tunnels are buried underground as required for missile protection.

Tunnels are isolated from adjoining structures by a seismic rattle space except that they are integrally connected to the adjacent structures when required to prevent sliding overturning and/or flotation.

Seismic Category I electrical and piping tunnels are protected from external flooding by:

1. Sealing the shake-space between the tunnels and the adjoining structures using waterstops and flexible seals, as shown in Fig. 3.8-20 and

2. Providing all penetrations below grade using air and water seals, as applicable.

3.8.4.1.8 Radwaste Building

The radwaste building contains storage facilities and equipment for the treatment of radioactive gas, liquid, and solid waste material.

The radwaste building is constructed mostly of reinforced concrete as required for shielding and foundations. A seismic analysis is performed on the radwaste building (see Section 3.7.2.16A); however, the building is not classified as a seismic Category I structure.

The radwaste building is located west of the reactor building and separated from the reactor building by a driveway. Personnel access is off the station's main east-west passageway and the stair tower at the north end of the building.

3.8.4.1.9 Turbine Building

The turbine building complex includes turbine building, heater bays, main steam tunnel, and condensate demineralizer regenerative and off-gas area. The complex houses the turbine generator, condenser, moisture separator, etc, in the turbine building areas, heaters, and related pumps and accessories in heater bay areas and off-gas system equipment and tanks in off-gas areas.

The turbine building is located immediately adjacent to and south of the auxiliary building, with the main steam tunnel passing through north-south and terminating at the turbine generator. Heater bays are located west of the turbine building. The condensate demineralizer and off-gas areas are located immediately adjacent to and south of the heater bays.

The turbine building complex is founded on select granular fill using spread footings for walls and columns. Although the structure is not classified as a seismic Category I structure, the portions of the structure housing off-gas systems are designed to withstand a seismic event using the seismic analysis, as described in Section 3.7.2.17A.

The structure is generally constructed of structural steel and metal roof decking and exterior siding above the operating floor at el 123 ft 6 in and of reinforced concrete below el 123 ft 6 in. The behavior of the steel superstructure during a tornadic event is described in Section 3.3.2. The off-gas area is constructed of concrete walls and floors. A seismic rattle space is provided between the turbine building complex and the adjacent structures, such as the auxiliary building and the auxiliary control building. Horizontal and vertical waterstops are provided at construction joints below grade to provide watertightness.

3.8.4.1.10 Fuel Building Cask Handling Area

Fuel Building Cask Handling Area (FBCHA) structure is a steel framing that extends out from the Fuel Building double doors in the north direction and is approximately 100 feet long by 27 feet wide by 70 feet high. The structure supports the northern end of the Cask Handling Crane rails. FBCHA structure is founded on select granular fill using spread footings for columns. This steel framing was originally designed and constructed as a non-safety related structure. The quality classification of the FBCHA structure has been upgraded to Quality Assurance Program Applicable requirements and designed as a Seismic Cat I structure. The QAPA classification requires that future design changes and/or alterations be performed under 10CFR50 Appendix B program (as if the structure is a safety related item). A seismic shake space is provided between the Fuel Building and the FBCHA structure.

3.8.4.2 Applicable Codes, Standards, and Specifications

Codes, specifications, standards, and NRC regulatory guides that are used in establishing design methods, analytical techniques, and material properties for seismic Category I structures are listed herein. The criteria for the design of seismic Category I structures are developed using the NRC regulatory guides and Code of Federal Regulations as follows:

Regulatory Guide 1.10	Mechanical (Caldweld Splices in Reinforcing Bars of Category I Concrete Structures (Rev. 1, 1/2/73)
Regulatory Guide 1.12	Instrumentation for Earthquakes (Rev. 1, 4/74)
Regulatory Guide 1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Rev. 1, 12/28/72)
Regulatory Guide 1.54	Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants (6/73)
Regulatory Guide 1.55	Concrete Placement in Cate-gory I Structures (6/73)
Regulatory Guide 1.69	Concrete Radiation Shields for Nuclear Power Plants (12/73)
Regulatory Guide 1.76	Design Basis Tornado for Nuclear Power Plants (4/74)
Regulatory Guide 1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete or Structural Steel during the Construction Phase of Nuclear Power Plants (Rev. 1, 4/76)
Regulatory Guide 1.117	Tornado Design Classification (Rev. 1, 4/78)

Appendix A of 10CFR50, Criteria 1, 2, 4, and 5 of General Design Criteria for Nuclear Power Plants.

Appendix B of 10CFR50, Quality Assurance Criteria for Nuclear Power Plants.

The degree of compliance to these documents is discussed in Sections 1.8 and 3.1, respectively.

The codes and standards used in the structural design of concrete and steel components of the seismic Category I structures are listed below:

ACI 211.1-1970	American Concrete Institute - Recommended Practice for Selecting Proportions for Concrete
ACI 214-1965	American Concrete Institute - Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ACI 301-1972	American Concrete Institute - Specification for Structural Concrete for Buildings. (The exceptions to this code are listed in Section 3.8.4.6.)
ACI 304-1973	American Concrete Institute - Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ACI 305-1972	American Concrete Institute - Recommended Practice for Hot Weather Concreting
ACI 306-1966	American Concrete Institute - Recommended Practice for Cold Weather Concreting
ACI 315-1974	American Concrete Institute - Manual of Standard Practice for Detailing Reinforced Concrete Structures
ACI 318-1971	American Concrete Institute Building Code Requirements for Reinforced Concrete (including 1974 supplement)
ACI 347-1968	American Concrete Institute - Recommended Practice for Concrete Formwork

AISC 1969	American Institute of Steel Construction - Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, dated February 12, 1969, including Supplements 1, 2, and 3 dated Nov. 1, 1970, Dec. 8, 1971, and June 12, 1974
AISC 1978	American Institute of Steel Construction - Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, dated November 1, 1978, Section 1.5.2
AISC 1972	Code of Standard Practice for Buildings and Bridges - AISC Manual
AWS D1.1-1975	American Welding Society - Structural Welding Code (Exceptions to this code are listed in Section 3.8.4.6.)
NCIG-01 Rev. 2	Nuclear Construction Issues Group, Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, May 7, 1985
AWS D12.1-75	American Welding Society Recommended Practice for Welding Reinforcing Steel Metal Inserts, and Connections in Reinforced Concrete Construction

U.S. Department of Labor, Occupational Safety and Health Administration. Occupational Safety and Health Standards, October 18, 1972.

Southern Standard Building Code, 1969 (including revisions up to 1972).

Louisiana Building Codes, as required.

The following is a summary of the principal plant structural specification that is prepared for procurement, fabrication, installation, and placement of components and materials for seismic Category I structures. This summary also includes the reference to the appropriate American Society for Testing and Materials specification used in procurement and testing of materials for seismic Category I structures, as applicable, and exceptions and deviations from the codes and standards, if any. The current editions of the ASTM standards adopted by the vendors' fabricating facilities at the time of procurement, fabrication, and testing of the structural materials are utilized.

Furnish Reinforcing Steel

Reinforcing Bars - ASTM A615, Grade 40 and Supplement S-1

Reinforcing Bars - ASTM A615, Special Chemistry Steel, (Section 3.8.4.6.) and Supplement S-1

Welded Wire Fabric - ASTM A185

Detailing and Fabrication - ACI-315, ACI-318, CRSI Manual of Standard Practice

Preparation of Ends of Rebars for Welding - AWS D12.1 Shop Detail Drawings - Reg. Guide 1.55

Quality Control-Testing, Inspection, and Documentation-ANSI N45.2.2, ANSI N45.2.5, and 10CFR50 Appendix B.

Furnish Radial Shear Bar Assemblies

Inclined Flat Shear Bars - ASTM A572 Gr. 50 or ASTM A588 Gr. A or B

Reinforcing Bars - N14 and N18 Special Chemistry, (Section 3.8.4.6)

Filler Metal for Welding - AWS A5.1 or AWS A5.5, E70xx Series, low hydrogen

Cadweld sleeves - (Section 3.8.4.6)

Fabrication - AWS D12.1

Quality Control - Testing, Inspection, and Documentation - ANSI N45.2.2, ANSI N45.2.5, and 10CFR50 Appendix B.

Mixing and Delivering Concrete

Cement - ASTM C150, Type II; ASTM C191, ASTM C266.

Air-entraining agent - ACI 301, ASTM C260 and ASTM C233

Aggregates - ASTM C33, ASTM C227, CRD-C119, ASTM C295, ASTM C586, ASTM C289

Aggregate storage - ANSI N45.2.2

Entrained air - ACI 301

Water - ASTM C109, ASTM C151, and ASTM C191

Proportioning - ACI 211.1, ACI 301

Slump requirement - ACI 301

Other admixtures - ASTM C494

Heavy aggregate concrete - ASTM C637, CRD-C119, ASTM C567

Batching - ACI 304

Truck mixers - ASTM C94

Mixing time - ASTM C94

Delivery - ASTM C94

Cold weather requirements - ACI 306

Hot weather requirements - ACI 305

Quality Control - Testing, Inspection, and Documentation - ACI 301, ACI 214, ANSI N45.2.2, ANSI N45.2.5, and 10CFR50 Appendix B.

Concrete Testing Services

Inspection of testing agency - ASTM E329

Gather aggregate samples - ASTM D75

Test water and ice - ASTM C109, ASTM C151, ASTM C191

Test fine aggregate - ASTM C33, ASTM C637 (for heavy aggregate only)

Sieve analysis - ASTM C136 or manufacturer's recommendations

Unit weight - ASTM C29

Organic impurities (chemical) - ASTM C40

Material finer than 200 sieve - ASTM C117

Light weight pieces - ASTM C123

Potential alkali reactivity - ASTM C227, ASTM C289, ASTM C295

Clay lumps and friable particles - ASTM C142

Specific gravity and absorption - ASTM C128

Soundness - ASTM C88 (magnesium sulfate - five cycles)

Freezing and thawing - ASTM C666

Petrographic examination - ASTM C295 to ascertain conformance with ASTM C33 by visual examination

Organic impurities (strength) - ASTM C87

Test coarse aggregates - ASTM C33, ASTM C637 (for heavy aggregates only)

Sieve analysis - ASTM C136 or manufacturer's recommendations

Unit weight - ASTM C29

Los Angeles abrasion - ASTM C131 (100 and 500 revolutions)

Compressive strength - ASTM C535, ASTM C39

Soundness - ASTM C88 (magnesium sulfate) 5 cycles

Specific gravity and absorption - ASTM C127

Freezing and thawing - ASTM C666

Potential alkali reactivity - ASTM C227, ASTM C289, ASTM C 295, ASTM C586

Clay lumps and friable particles - ASTM C142

Scratch hardness - ASTM C235

Material finer than 200 sieve - ASTM C117

Light weight pieces - ASTM 123

Flat and elongated pieces - CRD - C119

Petrographic examination - ASTM C295 to ascertain conformance with ASTM C33 by visual examination

Test admixtures, water-reducing admixtures - ASTM C494

Air entraining admixtures - ASTM C260

Test cement - ASTM C150 excluding ASTM C186, C115, C266, and C452 and ASTM C451 and ASTM C109

Design concrete mixes - ACI 211.1

Testing design mixes - ACI 301, Section 3.8.2.1, method 1

Aggregate moisture control - ASTM C566

Curing cylinders - ASTM C192

Test concrete (Section 3.8.4.6)

Compressive strength - ACI 301 - Sections 16.3.4.1 to 16.3.4.3 inclusive, ASTM C39

Slump test - ASTM C143

Air dry weight - ASTM C567, ASTM C642

Unit weight, yield, and air content - ASTM C138, ASTM C231

Hardened concrete - ASTM C42 (4 in)

Compressive strength of hydraulic cement mortars - ASTM C109

Evaluation of concrete strength - ACI 214 (Section 3.8.4.6) ACI 301, Chapter 17

Quality Control - ANSI N45.2.2, ANSI N45.2.5 and 10CFR50, Appendix B

Prepare concrete cylinders - ASTM C31, ASTM C39

Obtain hardened concrete cores - ASTM C42, ACI 301, Section 17.3.2

Concrete air content tests - ASTM C231.

Placing Concrete and Reinforcing Steel

Cadweld splices (Section 3.8.4.6) - Reg. Guide 1.10

Cadweld sleeve steel - ASTM A519, 85 ksi min. yield

Welded splices (Section 3.8.4.6) - AWS D12.1.

Welding cadweld sleeves to plates (sister splices) -AWS D1.1 or AWS D12.1

Form work - ACI 347 and ACI 301 (except where steel plate (liner) is used as formwork - see Section 3.8.4.6)

Reinforcing Steel - ASTM A615 (Refer to "Reinforcing Steel")

Placing reinforcement - ACI 301 and ACI 318

Concrete protection for reinforcement - ACI 318, ACI 301, and ACI 315

Concrete construction, expansion, and control joints -ACI 301, ASTM D1752, ASTM D994

Water stops - CRD-C-513

Anchor bolts and miscellaneous steel - ASTM A307

Inserts, sleeves, and pipes

Concrete placing - ACI 301, ACI 304, ACI 305, ACI 306, ACI 318

Cold weather requirements - ACI 306

Hot weather requirements - ACI 305

Vibration of concrete - ACI 301

Finishing of concrete lift surfaces - ACI 301

Concrete in blockouts - ACI 301

Depositing underwater - ACI 301

Watertight concrete - ACI 301

Strength tests of concrete (Section 3.8.4.6) and ACI 301

Repair of surface defects - ACI 301

Finishing of formed and flat surfaces - ACI 301

Curing and protection - ASTM C171, ASTM C309, ACI 301 (with the exceptions noted in Section 3.8.4.6)

Grouting - ACI 301

Quality Control - Testing, Inspection, and Documentation - ANSI N45.2.2, ANSI N45.2.5, and 10CFR50 Appendix B

Structural Steel

Shop detail drawings - Reg. Guide 1.55

Inspection and tests - ANSI N45.2.5

Structural Steel - ASTM A36, ASTM A440, ASTM A441, and ASTM A242

Bolts - ASTM A325, ASTM A307, and ASTM A490

End connections - AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings

Welding - AWS D1.1

Painting - SSPC-PA1, SSPC-PA2, SSPC-SP10, SSPC-Vis 1, Reg. Guide 1.54

Galvanizing - ASTM A123, ASTM A153, ASTM A384, AZI/AHDGA Manual

Metalizing - AWS C2.2, SSPC-SP10

Erection - AISC Manual

Quality Control - Testing, Inspection, and Documentation - ANSI N45.2.2,ANSI N45.2.5, and 10CFR50 Appendix B.

Applicable codes, standards, and specifications relating to the steel containment structure are identified in Section 3.8.2.

3.8.4.3 Loads and Loading Combinations

The design loading combinations imposed on Seismic Category I structures, including the weir wall, shield building, primary shield wall and pedestal, are identified in this section.

3.8.4.3.1 Notations Used in Loading Combinations

The following are the loads and notations used in the loading combination equations for the design of Seismic Category I structures.

Dead Load

D = Dead load of structure including the weight of all permanent construction such as walls, floors, roofs, partitions, stairways, fixed equipment, pipe, cable trays, and ducts. Forces resulting from hydrostatic pressure due to internal water are also included.

Live Loads

L = Live load superimposed by the use and occupancy of the structure but not including the wind load, earthquake load, or dead load. Where applicable, reduction in live load is in accordance with the American National Standard Building Code requirements for minimum design loads in buildings and other structures. Crane and elevator loads including their impact effects are included in live loads. In combination with earthquake, pipe rupture, and tornado loads, live loads present with these loads are used.

Wind Loads

W = Wind loads (identified in Section 3.3.1)

Tornado Loads

W_t = Tornado loads (identified in Section 3.3.2)

Earthquake Loads

SSE = Loads due to safe shutdown earthquake. Development of earthquake loads for various structures is described in Section 3.7. The earthquake loadings include forces due to two horizontal and vertical accelerations and are combined using square root of sum of squares (SRSS) method to produce the maximum stress resultants.

OBE = Loads due to operating basis earthquake. Development of earthquake loads for various structures is described in Section 3.7. The earthquake loadings include forces due to two horizontal and vertical accelerations and are combined using the SRSS method to produce the maximum stress resultants.

Earth Pressure

H = Load due to lateral earth pressure including the effects of surcharge.

Water Pressure

F = Force resulting from hydrostatic pressure due to external water or normal groundwater. The vertical pressure due to water is considered as dead load. Groundwater level is identified in Section 2.4.13.

F' = Force due to maximum postulated flood. Design flood level is identified in Section 3.4.

Design Basis High Energy Pipe Break

The design basis pipe break accident (DBA) is defined here as a large circumferential instantaneous double-ended pipe break, of the main steam or recirculation suction line.

The intermediate pipe break accident (IBA) is defined as a break that is less than the DBA but of sufficient size to automatically depressurize the primary system due to a loss of fluid and/or the automatic depressurization system (ADS). The lowest break size of the IBA range is approximately 0.1 sq ft.

The small pipe break accident (SBA) is defined as a break that is not large enough to automatically depressurize the primary system. Accident termination is dependent upon operator action to manually depressurize the reactor.

P_{a1} = Pressure load from IBA or SBA pressure transient, including design margin

T_{a1} = Direct temperature loads due to IBA or SBA temperature transient

R_a = Piping loads due to temperature resulting from the high energy pipe breaks

P_{a2} = Pressure loads from DBA pressure transient including design margin

T_{a2} = Temperature load due to DBA temperature transient

LOCA = The dynamic loads resulting from structural vibrations due to any of the following loads resulting from DBA, SBA, or IBA:

1. Condensation oscillation phenomenon
2. Chugging loads (symmetric and asymmetric)
3. Asymmetric annulus pressurization loading on RPV and primary shield wall
4. Pool swell loads (froth impingement and/or drag loads)
5. Vent clearing loads.

●→12

Note: LOCA loads and accident pressure are combined in accordance with their actual time histories of occurrence.

12←●

Pipe Break Loads

R_j = Jet impingement effects on a structure due to a ruptured high energy pipe.

R_m = Missile impact effects on a structure due to a rupture of high energy pipe.

R_r = Reaction on a structure due to the ruptured high energy pipe.

Operating Loads

P_v = Load due to differential pressure between internal and external areas of enclosed structures.

R_o = Piping loads during operating conditions.

T_o = Loads due to temperature gradient through concrete or steel elements plus loads exerted by liners due to the temperature associated with operating conditions.

Safety Relief Valve Discharge Loads

SRV_1 = Safety relief valve discharge loads resulting from any of the following events: one valve (first actuation), one valve (subsequent actuation), 2,7,9, or 16 valve blowdown events

SRV_2 = Safety relief valve discharge loads resulting from any of the following events: one valve (first actuation), 2, 7, or 9 valve blowdown events

SRV_3 = Safety relief valve discharge loads resulting from one valve (first actuation) blowdown.

Notes:

1. Safety relief valve discharge loads for multiple valves are for first actuation only.

2. Thermal loads due to SRV discharge are treated as T_o for normal operation and T_a for accident conditions.

3.8.4.3.2 Loading Combinations for Concrete Structures

With the exception of the basin wall and foundation mat of standby cooling towers and pumphouse structures, all seismic Category I concrete structures are designed so that the ultimate load capacity U, as modified by the standard provisions of ACI 318, Section 9.2, which requires the application of capacity reduction factors, will not be less than required by the following loading equations. The loading combinations for the basin wall and foundation mat of standby cooling towers and pumphouse structures are described in Section 3.8.4.3.4. The loading combinations for the shield building, weir wall, and pedestal are included in the following list. Loading combinations for the drywell and reactor building mat are identified in Sections 3.8.3.3 and 3.8.5.3, respectively.

Loading combinations for the steel primary containment are identified in Section 3.8.2.3.

The terms used in the following equations are as defined in Section 3.8.4.3.1.

Normal Operating Conditions

1. $U = 1.4D + 1.7 (L + F + H + P_v + SRV_1) + 1.3 (T_o + R_o)$

Severe Environmental Condition

2. U = $1.4D + 1.7 (L + F + H + P_v + W + SRV_1) + 1.3 (T_o + R_o)$

2.1 U = $1.2D + 1.7W$

3. U = $1.4D + 1.7 (L + F + H + Pv + SRV_1) + 1.3 (T_o + R_o) + 1.9\ OBE$

3.1 U = $1.2D + 1.90BE$

4. U = $0.9D + 1.4F + 1.7H + 1.3W$

5. U = $0.9D + 1.4F + 1.7H + 1.45\ OBE$

Note: Equations 4 and 5 are primarily to check against overturning.

Extreme Environmental Condition

6. U = $D + L + T_o + F + Pv + H + R_o + W_t + SRV_1$

7. U = $D + L + T_o + F + P_v + H + R_o + SSE + SRV_2$

8. U = $D + L + T_o + P_v + H + R_o + F' + SRV_1$

Abnormal/Severe Environmental Condition

9. U = $1.0 (D + L + F + H + T_{a1} + R_a + R_m + R_j + R_r + SRV_2 + LOCA) + 1.25 (P_{a1} + OBE)$

9.1 U = $1.0 (D + L + F + H + T_{a2} + R_a + R_m + R_j + R_r + SRV_3 + LOCA) + 1.25 (P_{a2} + OBE)$

Abnormal/Extreme Environmental Condition

10. U = $D + L + F + H + P_{a1} + T_{a1} + R_a + R_m + R_j + R_r + SSE + LOCA + SRV_2$

10.1 U = $D + L + F + H + P_{a2} + T_{a2} + R_a + R_m + R_j + R_r + SSE + LOCA + SRV_3$

Abnormal Loading Condition

11. U = $1.0 (D + L + F + H + T_{a1} + R_a) + 1.25\ SRV_2 + 1.5 (Pa_1 + LOCA)$

11.1 U = $1.0 (D + L + F + H + T_{a2} + R_a) + 1.25\ SRV_3 + 1.5 (P_{a2} + LOCA)$

Notes:

1. Loads resulting from thermal stratification, if applicable, will be included wherever temperature loads are considered.

2. Both cases in which L has its full value or is completely absent will be checked.

3. In combinations 9, 9.1, 10, 10.1, 11, and 11.1, the maximum values of P_a, T_a, R_a, R_j, R_m, and R_r, including a dynamic load factor, are used unless a time-history analysis is performed to justify otherwise.

4. Combinations 6, 9, 9.1, 10, and 10.1 are satisfied first without the tornado missile load in 6 and without R_r, R_j, and R_m in 9, 9.1, 10, and 10.1. When considering these concentrated loads, local section strength capacities may be exceeded, provided there is no loss of function of any safety-related system.

3.8.4.3.3 Loading Combinations for Steel Structures

The Seismic Category I structures are designed in accordance with the AISC steel construction manual (7th Edition). The notation S used in the following equations represents the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC code, Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, dated February 12, 1969, with its supplements and Section 1.5.2 of AISC specifications dated November 1, 1978. The 33 1/3 percent increase in the allowable stresses S permitted in the AISC code, when earthquake and wind loads are present in the loading combinations, is not used. Loads resulting from thermal stratification, if applicable, are included wherever temperature loads are considered. The terms used in the following equations are defined in Section 3.8.4.3.1. The loading combinations for the primary shield wall are included in the following list.

<u>Normal Operating Condition</u>

12. $S = D + F + L + H + P_v + SRV_1$

13. $1.5S = D + F + L + H + P_v + T_o + R_o + SRV_1$

Severe Environmental Condition

14. $S = D + F + L + H + P_v + OBE + SRV_1$

15. $S = D + F + L + H + P_v + W + SRV_1$

16. $1.5S = D + F + L + H + P_v + T_o + R_o + OBE + SRV_1$

17. $1.5S = D + F + L + H + P_v + T_o + R_o + W + SRV_1$

Extreme Environmental Condition

18. $1.6S = D + F + L + H + P_v + T_o + R_o + SSE + SRV_1$

19. $1.6S = D + F + L + H + P_v + T_o + R_o + W_t + SRV_1$

Abnormal/Severe Environmental Condition

20. $1.6S = D + F + L + H + P_{a1} + T_{a1} + R_a + R_r + R_j + R_m + OBE + SRV_2 + LOCA$

20.1 $1.6S = D + F + L + H + P_{a2} + T_{a2} + R_a + R_r + R_j + R_m + OBE + SRV_3 + LOCA$

Abnormal/Extreme Environmental Condition

21. $1.6S = D + F + L + H + P_{a1} + T_{a1} + R_a + R_r + R_j + R_m + SSE + SRV_2 + LOCA$

21.1 $1.6S = D + F + L + H + P_{a2} + T_{a2} + R_a + R_r + R_j + R_m + SSE + SRV_3 + LOCA$

Abnormal Condition

22. $1.6S = D + F + L + H + P_{a1} + R_a + T_{a1} + SRV_2 + LOCA$

22.1 $1.6S = D + F + L + H + P_{a2} + R_a + T_{a2} + SRV_3 + LOCA$

Notes:

1. Values of P_a, T_a, R_a, R_r, R_j, and R_m loads are based on time load characteristics in all the above load combinations. Wherever the time load characteristics are not used, the maximum value of the load, including a dynamic load factor, is used.

2. Combinations 19, 20, 20.1, 21, and 21.1 are satisfied first without the tornado missile load in 19 and without R_r, R_j, and R_m in 20, 20.1, 21, and 21.1. When considering these concentrated loads,

local section strengths may be exceeded provided there is no loss of function of any safety-related system. Furthermore, in combinations 20, 20.1, 21, and 21.1, in computing the required section strength S, the plastic section modulus of steel shapes may be used.

3. Both cases in which L has its full value or is completely absent are checked.

4. In combinations 18 through 22.1, thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

5. It should be noted that many loads presented in the above equations either may not be applicable or may not be a limiting factor for a particular design condition.

3.8.4.3.4 Loading Combinations for Basin Wall and Foundation Mat of Standby Cooling Tower and Pumphouse

The structural design of these water-retaining components is based on the working stress design (WSD) method, as recommended in ACI 350R, Concrete Sanitary Engineering Structures. For service load conditions represented by normal operating conditions and severe environmental conditions, the loading combinations for these components are:

<u>Normal Operating Condition</u>

1. $S = D + L + F + H$

2. $S = D + L + F + H + T_o + R_o$

<u>Severe Environmental Condition</u>

3. $S = D + L + F + H + OBE$

4. $S = D + L + F + H + W$

5. $S = D + L + F + H + T_o + R_o + OBE$

6. $S = D + L + F + H + T_o + R_o + W$

For extreme and abnormal environmental conditions, the structural design of these components is based on the ultimate strength design (USD) method, using the following loading combinations:

Extreme Environmental Condition

7. $U = D + L + F + H + T_o + R_o + SSE$

8. $U = D + L + F + H + T_o + R_o + W_t$

9. $U = D + L + F + H + T_o + R_o + F'$

Abnormal/Severe Environmental Condition

10. $U = 1.0(D + L + F + H) + 1.25\ OBE$

Notes:

1. Loads resulting from thermal stratification, if applicable, are included wherever temperature loads are considered.

2. Both cases in which L has its full-value or is completely absent are checked.

3. Since no pipe rupture loading is present, abnormal/extreme environmental conditions do not govern the design and therefore are not included here.

4. Loading combination 8 is satisfied first without the tornado missile load. When considering the concentrated missile load, local section strength capacities may be exceeded, provided there is no loss of function of any safety-related system.

5. Notations used in the above equations are as defined in Section 3.8.4.3.1 except as follows:

   U = The ultimate load capacity, as modified by the standard provisions of ACI 318, Section 9.2, which requires the application of capacity reduction factors.

   S = (For concrete structures) The required section strength based on the WSD method and the allowable stresses defined in Section 2.6 of ACI 350R.

3.8.4.3.5 Loading Combinations for Radwaste and Turbine Buildings

The load factors, loading combinations, required strength, and allowable stresses to be used in structural design of the radwaste and turbine buildings are based on ACI 318 code and AISC Manual of Steel Construction for reinforced concrete and structural steel, respectively, except that while evaluating the radwaste building for seismic and tornadic events, the loading combinations and allowable stresses used are identical to those for Seismic Category I structures. The seismic design of the radwaste building meets or exceeds the requirements of Regulatory Guide 1.143.

3.8.4.4 Design and Analysis Procedures

All other seismic Category I structures are analyzed and designed, as described herein. The structures are analyzed and designed for the loading combinations, as outlined in Section 3.8.4.3.

Seismic Category I structures are supported on reinforced concrete mat foundations. The design of reinforced concrete components of the structures follows ACI 318, whereas the structural steel components of the structures are designed using the AISC Manual of steel construction.

The exterior walls and roof of the structures are of minimum 2 ft-thick reinforced concrete and are designed to withstand the most critical loading, as applicable, including the tornado generated missile impact loads. The exterior walls below grade are designed for earth pressure, hydrostatic pressure, and surcharge loads, as applicable, including the dynamic effect of these loading during OBE or SSE events.

The roof and floors of the structures, except shield building and standby service water tower and pumphouse structures, are supported on steel framing. These components serve as shear diaphragm to transfer lateral loads to the exterior and interior concrete walls acting as shear walls. These walls are designed to withstand gravity loads, in addition to acting as a shear wall, and transmit all loads to the foundation. These walls are designed for in-plane shear forces in accordance with the requirements of Section 11.16 of ACI 318-71, except that the shield building wall and the cylindrical walls of the standby service water towers are designed in such a way that the vertical and hoop reinforcing bars carry in-plane shear forces, as described in Sections 3.8.4.4.6 and 3.8.4.4.4, respectively. Partition walls such as concrete block walls and drywalls are anchored using metal studs and/or gypsum wall boards, but are not considered to act as shear walls. However, these walls are designed so as not to damage any safety-related structure, system, or component.

Seismic Category I structures are essentially considered non-vented structures. Exterior walls, roof, and exterior doors are conservatively designed to withstand a maximum of 3 psi pressure drop and other tornadic loads, in addition to other applicable gravity loads. The exterior doors are either protected from postulated tornado generated missile impingement by providing protective structures around them, or designed to withstand the tornado-generated missiles.

Concrete masonry walls are utilized in Seismic Category I structures, specifically in the auxiliary, fuel, and control buildings. In general, masonry walls are provided as removable, nonload-bearing type plugs for equipment removal. They consist of stacked concrete blocks, as required for shielding, and are contained by steel framing attached to the walls. Steel framing provides confinement of the concrete blocks from in-plane and out-of-plane displacements, and is designed to withstand all forces other than masonry dead load. Since no failure of masonry walls, thus designed, can be postulated, damage to any safety-related component or system is not anticipated.

3.8.4.4.1 Auxiliary Building

The auxiliary building is designed as a reinforced concrete structure supported on a mat foundation. The foundation mat is analyzed and designed using the finite element capability of computer program STRUDL (Appendix 3A).

The safety-related pumps, (such as RHR, HPCS, and LPCS pumps) are supported at building mat level. Each pump is located in a separate watertight cubicle. Each cubicle is designed to withstand internal or external flooding due to DBA conditions, so that the flooding of one cubicle does not affect the operability of the pumps in the other adjacent cubicles. The lower section of each pump is housed in a pump shaft structure below the foundation mat of the building. These cylindrical shaft structures are designed to withstand SSE events and all other applicable loading conditions, described in Section 3.8.4.3. The main steam tunnel floor and walls are 4-ft thick reinforced concrete sections, designed to withstand pipe rupture loads due to high energy line breaks.

3.8.4.4.2 Fuel Building

The fuel building is designed as a reinforced concrete structure supported on a mat foundation. The pool walls and the foundation mat are analyzed and designed using the finite element capability of computer program STRUDL.

The structural components above the spent fuel pool area are designed to preclude the possibility of a drop of any component that could damage the stored spent fuel. The support structure for the cask handling crane within the fuel building is designed to withstand OBE and SSE events, considering the most critical live load condition. The spent fuel pool walls are also designed for the effects of temperature differential across the thickness of the pool walls for all postulated DBA conditions.

3.8.4.4.3 Control Building

The control building is designed as a reinforced concrete structure supported on a mat foundation. The foundation mat is analyzed and designed using the finite element capability of computer program STRUDL.

The control building is designed for all postulated events and applicable loading combinations outlined in Section 3.8.4.3, using conventional design procedures.

3.8.4.4.4 Standby Service Water Cooling Tower and Pumphouse

This structure is a reinforced concrete structure supported on a mat. The foundation mat is analyzed and designed using the finite element capability of computer program STRUDL.

The cooling tower basin shell is analyzed and designed using the finite element capability of computer program STRUDL. The WSD method is used in the reinforced concrete design of the basin wall and foundation mat. The balance of the structure is designed using the USD method. The structure is designed to prevent damage to any safety-related component or system due to missile impact. Precast beam arrangement is used to support the tile fill. The basin is designed for the loading, considering whether it is filled with water or empty, whichever is more critical, in conjunction with concurrent postulated loading conditions.

3.8.4.4.5 Diesel Generator Building

The diesel generator building is designed as a reinforced concrete structure supported on a mat foundation. The foundation mat is analyzed and designed using the finite element capability of computer program STRUDL.

The diesel generator building is designed for all postulated events and applicable loading combinations outlined in Section 3.8.4.3, using conventional design procedures.

Diesel generator fuel oil storage tanks are housed in the basement of the building and are covered with sandfill all around for fire protection. The interior and exterior walls are designed considering the effect of sandfill under postulated accident conditions.

3.8.4.4.6 Shield Building

The shield building above elevation 94 ft 8 in is analyzed and designed for the loading combinations outlined in Section 3.8.4.3. The lower portion of the structure up to elevation 94 ft 8 in is designed to resist loading combinations described in Section 3.8.3.3.1.

Discontinuities exist at the junction of the mat and composite section of the shell, on the top of composite section, and also at the junction of the shell and the dome. Arrangements of reinforcing steel at these discontinuities are shown on Figures 3.8-21 and 3.8-22.

During the long-term DBA, the shield building wall and dome are subjected to a temperature increase. Resultant stresses in the wall and the dome from the temperature gradient are determined using the computer code SHELL 1, a finite difference computer program described in Appendix 3A. The properties of the section are dependent on the state of stresses in the wall, i.e., amount of cracking. Therefore, the procedure used in the analyses is an iterative technique.

The shield building wall above elevation 94 ft 8 in is not subjected to the direct pressure loads resulting from a DBA, but discontinuity moments and shears at elevation 94 ft 8 in occur as an effect resulting from deformation of the mat and the lower composite portion of the wall. The discontinuity moments and forces at elevation 70 ft are obtained from the mat analysis (Section 3.8.5) and are applied as boundary conditions at the base of the shield building. The discontinuity effect at elevation 94 ft 8 in has been investigated by calculating stress resultants using the SHELL 1 program.

The cylindrical shell of the shield building is also investigated for both seismic shear and lateral earth pressure loads which are generated during an earthquake by backfill against the shield building wall. Determination of forces, moments, and shears in the walls of the shield building are calculated by the use of a finite-difference computer program SHELL 1 (Appendix 3A).

Wind pressure is assumed to be distributed over the shield building dome[1,2]. For the cylindrical part of the shell, wind pressure distribution is assumed to be in accordance with the methods given in ASCE Paper No. 3269[2].

The equivalent wind pressure from tornado conditions is presented in Section 3.3.2. The shield building is also analyzed for a pressure drop of 3 psi due to tornado conditions (Section 3.3.2) and for the impact of tornado borne missiles using methods described in Section 3.5.3.

Design and analysis of the tangential and radial shear for composite section in the bottom 24 ft 8 in of the wall has been performed in accordance with the ASME Code, 1977 Edition, Section III, Division 2, Article CC-3000, except that the maximum allowable tangential shear stress carried by concrete does not exceed 40 psi for abnormal/extreme environmental conditions. The section of the shield building at elevation 94 ft 8 in and above is analyzed and designed for tangential shear in accordance with the ACI 318-77 Code, Chapter 11.

To ensure that the shield building concrete and concrete fill act compositely, a positive mechanical anchoring mechanism across the shield building concrete fill interface is provided, in accordance with the ACI 318-77 Code, Chapter 17, Section 17.4 and 17.5 (Figure 3.8-1b) to adequately transfer the flexural shear between the two structures.

Arrangement of reinforcing steel for the shield building wall and structural concrete fill are presented in Figures 3.8-23, 3.8-23a, and 3.8-23b.

The shield building equipment hatch, which is not subjected to DBA pressure loading, is analyzed by means of the two-dimensional finite element capability of the computer program STRUDL (Appendix 3A).

3.8.4.4.7 Electrical Tunnels and Piping Tunnels

Electrical tunnels and piping tunnels housing safety-related systems are constructed of reinforced concrete.

The tunnel is designed for gravity loads in addition to other applicable loads, such as hydrostatic loads, seismic loads, earth pressure, etc, in accordance with the loading combinations described in Section 3.8.4.3. In addition to these loadings, the tunnel roof and walls are designed to carry the applicable crane loads resulting from the movement of cranes during and after construction. In several instances, the tunnels were integrally connected to the adjacent structures, so as to prevent sliding, overturning, and/or floatation. The tunnel roof is designed for surcharge loading, seismic or tornadic forces, whichever is critical, in addition to other applicable loadings in accordance with the loading combinations outlined in Section 3.8.4.3.

3.8.4.4.8 Radwaste Building

The radwaste building is designed as a reinforced concrete structure supported on a mat foundation. The foundation mat is analyzed and designed using the finite element capability of STRUDL program.

The structural steel and reinforced concrete components of this building were originally designed using a one-third increase in allowable stresses for wind and seismic loads in the loading combinations of the AISC Manual of Steel Construction and ACI 318, respectively. Seismic loads are derived using load profiles developed by seismic analysis described in Section 3.7.2.16A. The building is subsequently evaluated using loading combinations and allowable stresses similar to Seismic Category I structures (including tornadic and SSE seismic events) to verify that the structure does not collapse on adjacent Seismic Category I structures.

The tanks containing radioactive waste are protected from rupturing by housing them in the enclosures designed to withstand postulated tornado-generated missiles. In the event of a postulated tank rupture, the base mat and exterior walls are designed to retain the spillage within the building. Watertight concrete mixes are used for construction of the foundation mat and exterior walls up to 5 ft above the mat level to retain the spillage within the building.

3.8.4.4.9 Turbine Building

The turbine building complex is constructed partially on spread footings and partially on a mat foundation analyzed and designed using the finite element capability of STRUDL computer program.

Structural steel and reinforced concrete components of this building are designed using the loading combinations in accordance with the AISC Manual of Steel Construction, and ACI 318, respectively. A one-third increase in allowable stresses is used while designing the structure for wind and seismic loads. The seismic loads are derived using a modified seismic analysis approach, as described in Section 3.7.2.17A. Only one orthogonal horizontal seismic value at a time is considered in designing the structure. The greater of the two orthogonal horizontal seismic values producing the most critical stresses is used in designing the structural components.

This building complex is constructed of reinforced concrete floors and walls up to the operating floor level, with the exception of the off gas area. The structure above the operating floor level is constructed of a structural steel rigid bent system braced by vertical and horizontal bracing systems up to roof level, enclosed by metal siding. A steel roof deck with roofing is provided at the top of the structure. To prevent collapse of the steel superstructure on the auxiliary building and off gas area, the rigid bent system above the operating level is analyzed and designed, using the space frame capability of the STRUDL computer program, for the tornado or seismic loads, whichever is critical, in addition to other gravity loads. The metal siding, roof decking, girts, etc, are assumed to blow away during a tornadic event; however, the main structural steel members, such as columns, beams, bracing members, etc, are designed to remain in place. The off gas area is constructed of reinforced concrete walls and floors and is designed using modified seismic analysis, as described above.

3.8.4.5 Structural Acceptance Criteria

For the steel structures, the allowable stresses and factors of safety are in accordance with the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, with the following exceptions:

Safety-related structures as identified in Table 3.2-1 are capable of withstanding the SSE loads in combination with applicable dead and live loads. The allowable steel stresses under this condition are described in Section 3.8.4.3.

These safety-related structures are also checked using OBE loads in combination with applicable dead and live loads. For this loading condition, allowable stresses are the normal working stresses as allowed by AISC specification.

Concrete structures are designed by the strength design method of ACI 318. Load factors are as given in Section 3.8.4.3.

The basic criterion for strength design is expressed as:

Required strength ≤ Calculated strength.

All members and all sections of members are proportioned to meet this criterion. The required strength is expressed in terms of design loads, or their related internal moments and forces. Design loads are defined as loads which are multiplied by their appropriate load factor (safety factors). Calculated strength is that computed by the provisions of ACI 318, including the appropriate capacity reduction factors. Capacity reduction factors are taken as given in Section 9.2 of ACI 318.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

The materials for construction of seismic Category I structures are procured, fabricated, and delivered to the site in accordance with the codes, standards, and specifications described in Section 3.8.4.2. The shipping, storage, and handling of materials during construction follow the requirements of ANSI N45.2.2. The major materials for construction of seismic Category I structures are described herein.

Concrete

ACI 301, Specification for Structural Concrete for Buildings, together with ACI 347, Recommended Practice for Concrete Formwork and ACI 318, Building Code Requirements for Reinforced Concrete, forms the general basis for the concrete specifications.

ACI 301 is supplemented as necessary with mandatory requirements relating to types and strengths of concrete, including minimum concrete densities, proportioning of ingredients, reinforcing steel requirements, joint treatments and testing requirements.

Admixtures, types of cement, bonding of joints, embedded items, concrete curing, additional test specimens, additional testing services, cement and reinforcing steel mill test report requirements, and additional concrete test requirements are specified in detail.

All cement conforms to the Specification for Portland Cement, ASTM Designation C 150, Type II, except that precast concrete sections such as concrete blocks for rebar supports, concrete masonry blocks, and concrete sills, lintels, and copings, may be manufactured using other than Type II cement. If aggregates are determined to be potentially reactive by any of tests ASTM C295, C289, or the inservice inspection of structures, low alkali cement (Na_2O + $0.658K_2O$ < 0.6 percent) will be used. The ASTM C227 test need not be completed prior to aggregate usage when using low alkali cement. If the results of ASTM C227 demonstrate that higher alkali cement can be used, a decision will be made regarding alkali content for project use. Certified copies of mill test, showing that the cement meets or exceeds the ASTM requirements for portland cement, are furnished by the manufacturer. An independent testing laboratory may be retained to perform tests on the cement for compliance with the specifications.

An air-entraining agent is used in the concrete in an amount sufficient to satisfy ACI 301, Section 3.4.1 (Durability). This agent conforms to the requirements of the Standard Specification for Air-entraining Admixtures for Concrete, ASTM C 260. The air-entraining agent is added separately to the batch in solution in a portion of the mixing water or to sand when using water exceeding 150°F during cold weather conditions. The solution is batched by means of a mechanical dispenser capable of accurate measurement and in such a manner as to assure uniform distribution of the agent throughout the batch during the specified mixing period. Air-entrained cement is not used.

Other admixtures to control the rate of set, reduce the water content, or improve the workability and cohesiveness of concrete may be used in specific instances. Such admixtures are used only after tests are made in combination with the cement and aggregates being used and specifically approved by the Structural Engineer. Calcium chloride is not used under any circumstances.

Mixing water is clean and free from injurious amounts of oils, acids, alkalies, salts, organic materials, or other substances deleterious to concrete or steel. The mixing water is periodically checked and tested for suitability by ASTM C 109, ASTM C151, and ASTM C191.

Fine and coarse aggregates for normal weight concrete conform to the requirements of the Standard Specifications for Concrete Aggregates, ASTM C33, except that minor deviations from ASTM C33 requirements are accepted, contingent upon engineering evaluation of compressive strength test results and workability of concrete during placement. In addition to this, averaging of gradation test results as recommended in Section 2.1.5 of ACI 304 is used as necessary, for acceptance of the aggregates. Aggregates for normal weight concrete are evaluated for potential chemical alkali reactivity. Aggregates are free from any materials that would be deleteriously reactive in any amount sufficient to cause excessive expansion of mortar or concrete. All aggregates for normal weight concrete are tested for compliance with the above requirements.

The fine aggregates for nonstructural high density fill material conform to ASTM C-637 for gradation and specific gravity requirements. The fine aggregates also conform to ASTM C-33 for soundness and clay lumps and friable particle requirements.

Proportioning of structural concrete conforms to ACI 301, Chapter 3. In general, concrete mixes have a 28-day specified strength of 3,000 psi.

Concrete or grout used for biological shielding purposes in floors, walls, roofs, and foundations of the buildings has a weight not less than 135 lb/cu ft, when air-dried in accordance with ACI 301, Section 3.3. Reference to lightweight concrete in Section 3.3 is construed as applicable to regular structural concrete for the purpose of these requirements. In cases where space is not available, it may be necessary to use high density fill material having a density of 200 lb/cu ft or greater to provide biological shielding.

Proportions of ingredients are determined and tests conducted in accordance with the method detailed in ACI 301 and ACI 211.1 for combinations of materials to be established by trial mixes.

Concrete protection for reinforcement, preparation, cleaning of construction joints, concrete mixing, delivering, placing, and curing, is equal to or exceeds the requirements of ACI 301, with the following exceptions:

1. Section 3.5 - The allowable slump for tremie concrete shall be between 5 and 7 1/2 in. The allowable slump of concrete, when used for caisson and steel piles, shall be between 4 and 6 in. In congested areas, the allowable slump will be 5 in to permit concrete placement in the heavily reinforced structures. Section 14.4.1 - The allowable slump for massive concrete shall be 4 in except that in congested areas the allowable slump shall be 5 in to permit concrete placement in heavily reinforced structures.

2. When concrete is placed by pumping, sampling for tests of air content, slump, compressive strength, and temperature may be permitted at the delivery point rather than at the placement point if correlation testing is in effect.

   When concrete is placed by means other than pumping, sampling for in-process tests of air content, slump, compressive strength, and temperature is permitted at the delivery point, i.e., the truck mixer discharge, rather than the placement point, provided these points can be considered coincident. If 5 min or less are used in the transit of concrete from the delivery point to the placement point, they can be considered coincident.

3. Anticipated slump at the point of mixing would range between 1/2 in to 2 in more than the slump at the point of placement.

4. Section 14.5.1 - The minimum curing period will be one week.
5. In lieu of Section 14.5.4, use Section 12.3.3.

6. Maximum placing temperature of the concrete when deposited conforms to the requirements of ACI 301 and ACI 305, Recommended Practice for Hot Weather Concreting, except for the placing of mass concrete. The placing temperature of mass concrete does not exceed 80°F. ACI 301 indicates placement of mass concrete sections to 70°F. This limit is based on concrete using standard, or common, Type I cement. Type II cement, which is used for this

project, generates 80 to 85 percent of the heat of hydration of Type I cement. ACI 207 states that the heat-generating characteristic of Type II cement corresponds closely to that of Type I cement at 10°F lower placing temperatures.

7. All mass concrete placed at a temperature above 75°F is water cured in accordance with ACI 301, Chapter 12.

8. Section 4.3 of ACI 301 and Section 2.4 of ACI 301 form the basis for establishing formwork tolerances except that when a steel plate (liner) is used for formwork, the liner tolerances will govern. Also, when the side of a wall opposite the steel liner is to be formed using something other than a liner the theoretical form line will be established by measuring the thickness of the wall from the steel liner on the opposite face. Once the theoretical form line is established the variation in thickness will be governed by the tolerances given in ACI 301 and ACI 347, as applicable.

9. Section 12.3.1 - The temperature of the concrete is maintained above 50°F during cold weather curing.

10. Concrete maturity meters may be used as an additional means of estimating the in-place concrete strength in addition to the methods stated in Section 4.7 of ACI 301. The maturity method is a refinement of the minimal time method of Section 4.7.2 of ACI 301 since both the length of curing time and the curing temperature are used to determine the concrete strength. Concrete strength-maturity curves are constructed for laboratory cured cylinders based upon the following relationship:

$$M = \sum_{0}^{t} (C + 10)\ \Delta t$$

where:

M = Maturity number, deg C-hour

C = concrete temperature, deg-celsius

Δt = curing time interval, hours

Concrete maturity meters measure the maturity number of the in-place concrete. The maturity curves are used to determine the corresponding concrete strength.

11. Section 4.4.1 - When Q-decking is used as a permanent left-in-place form, small amounts of scattered construction debris may be left in place provided that debris does not reduce the minimum concrete cover, the minimum amount of structural concrete to be provided, or the depth of concrete required for radiation shielding.

Batching and mixing conform to ACI 301, Chapter 7. Concrete ingredients are batched in a batch plant and transferred to transit mix trucks for mixing, agitating, and delivering to the point of placement, or are batched and mixed in a controlled mixer and transferred to a truck for delivery.

Placing of concrete is by bottom dump buckets, chuting, concrete pump, or conveyor belt. The rate of placing concrete is controlled so that concrete may be effectively placed and compacted by vibrating with particular attention given around embedded items and near the forms.

Vertical drops greater than 5 ft for any concrete are not permitted, except where suitable equipment is provided to prevent segregation.

After the initial concrete set has occurred, but before the concrete has reached its final set, the surfaces of all construction joints are thoroughly cleaned using an air-water jet to remove all laitance and to expose clean, sound aggregate. After cutting, the surface is washed and rinsed. All excess water which is not absorbed by the concrete is removed.

Where, in the opinion of the field engineer, the use of an air-water jet is not advisable, then that surface is bush hammered or sand blasted, or other satisfactory means are used to produce the requisite clean surface. If hand tools are used to roughen concrete while it is in the plastic state, aggregate exposure is not required if a requisite clean surface of sufficient roughness to meet the required design strength across the joint is provided. Horizontal construction joints are covered by a 1/2-in thick layer of sand/cement grout, which has a compressive strength that is equal to or exceeds that of the concrete, and new concrete is then placed immediately against the fresh grout.

Curing and protection of freshly deposited concrete conforms to ACI 301, Chapter 12, with the following supplementary provision:

1. Concrete to be cured with water is kept wet by covering with an approved water saturated material or by a system of perforated pipes or mechanical sprinklers, or by any other approved methods which keeps surfaces continuously wet. Water used for curing is generally clean and free from any elements which might cause objectionable effects.

Concrete Testing

Compressive strength tests of concrete placed in seismic Category I structures are performed in accordance with ACI 301, Chapter 16, Section 16.3.4 for every 100 cu yd of concrete or a minimum of one set per 8-hr shift, whichever is greater.

The test specimens for compressive strength are 6-in diameter by 12-in long cylinders. Each set consists of at least three specimens. At least one is tested after 7 days and two after 28 days or 60 days age, as applicable.

Concrete strength tests are evaluated in accordance with ACI 214, Recommended Practice for Evaluation of Compression Test Results of Field Concrete, and ACI 301, Chapter 17.

The strength of concrete is considered satisfactory, if the averages of all sets of three consecutive strength test results of the laboratory-cured specimens at the specified age is equal to or greater than the specified compressive strength, f'_c, of the concrete and if no individual strength test result falls below the specified strength, f'_c, by more than 500 psi.

The field tests for slump of portland cement concrete are in accordance with ASTM C 143. Any batch not meeting specified requirements is rejected. Slump tests are made frequently during concrete placement and each time concrete test specimens are taken.

If cylinders should fail to meet the concrete strength requirements at the specified age, strength development and design strength requirements are reviewed. If this evaluation deems necessary, core tests are conducted in accordance with ASTM C42, Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete.

Should core tests be inconclusive or impracticable to obtain and structural analysis does not confirm the safety of the structure, load tests are performed. Concrete work judged inadequate by structural analysis or by load tests is reinforced with additional construction, if so directed by the engineers, or is removed and rebuilt.

Statistical quality control of the concrete is maintained by a computer program based on an article in ACI Publication SP-16, Computer Applications in Concrete Design and Technology. This program analyzes compression test results by the testing laboratory in accordance with methods established by ACI 214, Recommended Practice for Evaluation of Compression Test Results of Concrete.

Reinforcing Steel

Except for the special chemistry N14 and N18 reinforcing bars, all reinforcing conforms to Grade 40 or Grade 60 of the Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement, ASTM A615 and Supplement S-1. In addition to this standard, deformation is checked twice as a minimum for each heat for each reinforcing size. This check is made for the mill physical test report and for the required confirmatory physical test reports. The check measurements are recorded.

Mill test reports showing actual chemical and physical properties, including bend tests, are furnished for each heat of steel furnished.

Special chemistry reinforcing bars N14 and N18 are steel of 50,000 psi minimum yield point, conforming to the Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement, ASTM A615, as modified to meet the following chemical and physical requirements:

Carbon	0.35 percent maximum
Manganese	1.25 percent maximum
Silicon	0.15 to 0.25 percent
Phosphorous	0.05 percent maximum
Sulfur	0.05 percent maximum
Minimum yield strength	50,000 psi
Elongation	13 percent minimum in an 8-in test sample
Tensile strength	70,000 psi minimum

For these special chemistry bars, all ingots are identified and all billets are stamped with identifying heat numbers.

All bundles of bars are tagged with a heat number as they come off the rolling mill. A special mark is rolled into all bars conforming to this special chemistry to identify them as possessing the chemical and mechanical qualities specified. The chemical variations allowed for special chemistry bars are in accordance with ASTM A29.

Placing of reinforcing steel conforms to the requirements of Chapter 5 of ACI 301, Structural Concrete for Buildings, and Chapter 7 of ACI 318, Building Code Requirements for Reinforced Concrete.

Tack welding of designed reinforcing steel that does not become an integral part of the weldment is not permitted.

Reinforcing Steel Inspection and Testing

The engineers' inspectors witness, on a random basis, the pouring of the heats and the physical and chemical tests performed by the manufacturer for the special chemistry reinforcing steel.

Bars containing injurious defects or failing to conform to required chemistry and physical requirements are rejected.

Mill test reports showing actual chemical ladle analysis, physical properties, bend test, variations in weight, and conformance of deformations will be obtained from the manufacturer for each heat. In addition, confirmatory tensile tests for each 50 tons of every heat of steel for every bar size are made to determine physical properties. Further, for the special chemistry bars an additional chemical analysis of each heat will be made to confirm the chemical content.

Full-size test specimens of all rebars are tested on a testing machine using an 8-in gage length. Speed of testing is as specified in ASTM A370. The acceptance standards are in accordance with ASTM A615.

Reinforcing Steel Splices

Cadweld reinforcing steel splices, manufactured by Erico Products, Inc., Cleveland, Ohio, are used to splice N14 and N18 reinforcing bars; reinforcing bars are also butt-welded in a manner conforming to the requirements of AWS D12.1.

All cadweld splices are made in accordance with the instructions for their use issued by the manufacturer, Erico Products, Inc.

Reinforcing bars No. 11 and smaller are generally lap spliced. Where lap splicing is impractical, splicing is accomplished by:

1. Cadweld as manufactured by Erico Products, Inc. or equal, using the sleeves that develop the full tensile strength of the reinforcing bars, or

2. Butt-welded in accordance with the requirements of AWS D12.1.

In order to qualify operators for making cadweld production joints, each operator is required to prepare two satisfactory qualification splices for each of the splice positions to be used. Testing is by tensile testing a cadweld that simulates field conditions and uses the same materials as those to be used in the structure.

The ends of the reinforcing steel bars to be joined by the cadweld process are saw cut, flame cut, or shear cut. The ends of the bars are thoroughly cleaned of all rust, scale, grease, oil, water, or other foreign matter before the joints are made.

Cadweld Testing and Inspection

Cadweld process splices are visually inspected in accordance with Regulatory Guide 1.10. Visual inspection includes random inspection of the ends of the bars for dryness and cleanliness prior to fitting the sleeve over the ends.

Inspection is made of the completed splice for properly filled joints that have filler metal visible at both ends of the sleeve for T-series splices and the exposed end for B-series splices and at the tap hole in the center of the sleeve. Splices that do not meet all these inspection criteria are rejected.

Randomly selected cadweld splices based on separate test cycles for horizontal, vertical, and diagonal bars, size of rebar, and cadwelder are removed from the structure and tensile tested, or a combination of production and sister splices are tested in accordance with ASTM A370. Testing is in accordance with the following schedule if only production splices are tested:

1 out of first 10 splices

1 out of next 90 splices

2 out of next and each subsequent units of 100 splices.

If combinations of production and sister splices are tested, the sample frequency is as follows:

1 production splice out of the first 10 production splices

1 production and 3 sister splices out of the next 90 production splices

3 splices, either production or sister splices, for the next and subsequent units of 100 splices. At least one-fourth of the total number of splices tested are production splices.

The sample frequency for splices in curved bars with a radius of curvature less than 60 ft is as follows:

1 sister splice for the first 10 production splices

4 sister splices for the next 90 production splices

3 sister splices for the next and subsequent units of 100 production splices.

Sister splices are made using straight bars.

The tensile strength of each sample tested should equal or exceed 125 percent of the specified minimum yield strength for the grade reinforcing bar used. Failure of any splice to achieve 125 percent of the specified minimum yield strength is evaluated in accordance with Section 5 of the Procedure for Substandard Tensile Test Results as given in Regulatory Guide 1.10, Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments.

<u>Welding of Reinforcing Steel</u>

All welding of reinforcement conforms to Recommended Practices for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction, AWS D12.1.

Certified material test reports for welding electrodes are obtained from the electrode manufacturer.

The end of the bars to be joined by butt welding are prepared by sawing or flame cutting and dressing by grinding, where necessary.

In order to qualify welders for work on the reinforcing steel bars, each welder makes test welds in each position he is required to use during production. Each test weld is tension tested and each is required to meet or exceed the minimum tensile strength of the reinforcing bar.

Structural ductility is maintained by staggering critical splices wherever possible to assure that small adverse effects of multiple splices in the same plane do not occur.

Inspection and Testing of Reinforcing Steel Welds

All welds are visually inspected. Any cracks, porosity, or other defects are removed by chipping or grinding until sound metal is reached, and then repaired by welding.

Completed welded joints in reinforcing steel are selected on a random basis from seismic Category I structures and radiographically inspected in accordance with the following schedule:

1 out of first 10 splices

3 out of next 100 splices

1 out of next and subsequent units of 100 splices.

Cracks and any excessive amount of contained voids, as specified in AWS D12.1, are cause for repair or removal and replacement. Replaced welds are examined in a similar manner.

Reinforcing steel bars welded to steel embedments are tested by sister splice, in accordance with the following schedules:

1 sister splice out of the first 10 production splices

4 sister splices for the next 90 production splices, and

3 sister splices for the next and each subsequent units of 100.

Structural Steel

Structural steel material, erection, and fabrication tolerances are in accordance with the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, including the supplements, as listed in Section 3.8.4.2. In general, steel used for structural framing conforms to ASTM-A36. In areas where the design indicates that a higher strength steel is required, ASTM A440, A441, A572, A588, or A242 steel is used.

Certified copies of mill test reports showing actual chemical and physical properties are furnished for each heat of steel used in making seismic Category I structural steel.

Welding of structural steel is in accordance with AWS D1.1 with the following exceptions:

1. Low hydrogen electrodes may be stored at temperatures between 120°F and 350°F after being removed from sealed containers, or after drying.

2. Electrode ambient exposure time may be 5 hr for E70xx and 4 hr for E80xx.

3. In lieu of preproduction bend testing as described in AWS D1.1-75 (paragraph 4.29.2), threaded studs may be preproduction tested by torque testing using the provisions of AWS D1.1, paragraph 4.30.2.

4. As an option to preproduction testing of threaded studs as required in paragraph 4.29.2 of AWS D1.1-75, all production-threaded studs may be torque tested using the provisions of AWS D1.1-75, paragraph 4.30.2.

5. AWS D1.1-75, paragraph 4.28.11 calls for a 5/16-in minimum fillet weld as an option to gun welding of studs. In lieu of this minimum weld size, the following criteria (taken from AWS D1.1-82, Table 7.5.5) will apply when shielded metal arc welding of studs is performed:

Stud Diameter (in)	Minimum Fillet Size (in)
1/4 through 7/16	3/16
1/2	1/4
5/8, 3/4, 7/8	5/16
1	3/8

The material installation and inspection of high strength bolts conform to the requirements of the Specification for Structural Joints using ASTM A325 or A490 Bolts.

•→8

High strength bolts (e.g. A325 or A490) are used at River Bend Station as follows:

1. For structural steel connections on all seismic Category I Structures

2. In component support designs as follows:

   a. Large bore pipe supports
   b. Duct supports
   c. Small bore pipe supports
   d. Conduit supports
   e. Cable tray supports

8←•

3.8.4.6.1 Bearing Type Anchors (Drillco Maxi Bolt)

•→9

River Bend Station does not consider I&E Bulletin 79-02 applicable to Drillco Maxi Bolt bearing type anchors. Further, concrete performance was not addressed as a concern in I&E Bulletin 79-02.

9←•

The design assumption that the concrete is a homogeneous material is accomplished by stringent quality control measures and techniques used in mixing, placing, and curing of concrete. Actual concrete strength is always higher than the one used in design. Additionally, capacity reduction factors(ϕ) are introduced in the design calculations.

Prior to the use of Drillco Maxi Bolt, each size of bolt had been qualified by testing to ensure the ultimate load carrying capacity of each bolt. The concrete slab used for this test program is cast using the same construction procedure used on the project. During testing, the test frame is located so that the tensioning mechanism supported on concrete is outside the theoretical concrete shear cone area of the bolts being tested.

•→9

Use of this application is authorized only after the bolts are qualified. Each bolt is subjected to a stress of $0.81F_y$ during installation, thereby assuring the presence of good quality concrete around the sleeve. The above listed measures in design, installation, and testing coupled with adequate factors of safety in the allowable loads provide assurance that the anchors perform as intended without loss of function.

9←•

3.8.4.6.2 Other Materials

The materials used for waterstops, seals, compressible filler in the shake-spaces, and door gaskets (seals) in and between all Seismic Category I structures are selected to provide adequate resistance against environmental factors including radiation. The environmental resistance

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properties of each of these materials are listed in Table 3.8-9. In most cases the actual temperature ranges and the radiation levels in the Seismic Category I areas where these materials are used are within the limits specified in this table. In a few isolated instances, certain door seals are subjected to higher cumulative radiation levels during accident conditions than those listed. However, this condition affects primarily the elongation property of the door gaskets, and since the plant is operated with these doors closed, this condition is not considered to affect the functional capability of the gasket material.

3.8.4.7 Testing and Inservice Surveillance Requirements

No full-scale structural testing or in-service surveillance is anticipated for the structures described in Section 3.8.4.1. For testing of the materials used in their construction, refer to Section 3.8.4.6.

3.8.5 Foundations and Concrete Supports

3.8.5.1 Description of the Foundation and Supports

Table 3.8-7 lists the foundation systems that are used for individual Seismic Category I structures. All foundations are soil supported. The characteristics of the soil are described in Section 2.5. The relative locations of the Seismic Category I structures are shown on Fig. 1.2-1.

3.8.5.1.1 Reactor Building

Fig. 3.8-24 shows the reactor building foundation mat. The mat is a reinforced concrete structure approximately 10 ft thick and 150 ft in diameter. The mat is reinforced with both top and bottom layers of reinforcing steel. Shear reinforcing steel, for radial shear forces, is placed in the vertical direction. Bottom mat reinforcement is placed in an orthogonal grid pattern with layers at 90 deg to each other. Reinforcement for the top of the mat consists of concentric circular bars and radial bars. The reinforcing pattern for the top of the mat is arranged to maintain a uniform spacing of the bars which extend into the mat from the vertical walls above. Reinforcing steel bars are detailed for maximum length. Mat reinforcing bars are not spliced at the junction of the mat and vertical walls.

The normal high groundwater table is expected to be approximately el 57 ft msl. The reactor building mat is founded at el 60 ft msl. Adequate concrete cover is provided for protection of reinforcement and embedded steel against corrosion.

The concrete reactor vessel pedestal, weir wall, drywell wall, and containment vessel are anchored to the mat by mat embedment assemblies as shown in Fig. 3.8-2. The shield building wall is also adequately connected to the mat by vertical reinforcing dowels to resist the discontinuity moments and shears. To ensure radial shear transfer from the structural concrete fill, a shear key has been cut into the mat (Figure 3.8-1a).

3.8.5.1.2 Foundations for Other Structures

Foundations for all other Seismic Category I structures and the radwaste building are reinforced concrete mats. The turbine building foundation is a combination of a reinforced concrete mat and spread footings. These building foundations are listed in Table 3.8-7. Adjacent foundations of these structures are separated by a shake-space and are provided with waterstops as described in Section 3.8.4.1.

The foundations of the Seismic Category I structures and radwaste and turbine buildings are evaluated against the possibility of liquefaction during a seismic event, as described in Section 2.5.4.8.

Fig. 3.8-25 shows a typical reinforcing pattern at the junction of reinforced concrete vertical structural elements and a foundation mat.

3.8.5.2 Applicable Codes, Standards, and Specifications

The design codes, standards, specifications, and regulations that are used for the design and construction of Seismic Category I foundations are listed in Section 3.8.4.2 except that the design loads and loading combinations for the reactor building mat are in accordance with Article CC-3000 of the ASME Code, Section III, Division 2[3].

3.8.5.3 Loads and Loading Combinations

The loading combinations for design of the Seismic Category I foundations, other than the reactor building mat, are the same as those used in designing the Seismic Category I structures and are listed in Section 3.8.4.3.

The reactor building mat is designed for the same loading combinations as the drywell wall. They are consistent with those in Table CC-3230-1 of the ASME Code, Section III, Division 2 and are discussed in Section 3.8.3.3.1. In addition, the following load combinations are used to check against sliding and overturning due to earthquakes, winds, and tornadoes, and against floatation due to floods:

(1) D + H + OBE
(2) D + H + W
(3) D + H + SSE
(4) D + H + W_t
(5) D + F'

where D, OBE, W, SSE, W_t are as defined in Section 3.8.3.3.1, H is the lateral earth pressure, and F' is the bouyant force of the design basis flood.

3.8.5.4 Design and Analysis Procedures

The reactor building mat is analyzed and designed for the loading combinations defined in Section 3.8.5.3. The MAT-6 program, a digital computer program based upon the general methods described in Appendix 3A, is used to determine the stresses in the mat due to statically applied axisymmetric loads. This program analyzes an axisymmetrically loaded circular plate on an elastic foundation and maintains compatibility between the plate and concentric walls supported by the plate. The mat analysis includes the effects of the drywell and steel containment pressure loads generated by the design basis accident, the loads from temperature due to operating conditions and the design basis accident, the stiffness characteristics of the cylindrical shells which are considered as elastic constraints on the mat, the dead loads, and the characteristics of the supporting soil. The subgrade stiffness is based upon the Boussinesq theory, which assumes the subgrade to be a homogeneous isotropic elastic medium. The discontinuity moments and shears at the junctions of the shield building wall, steel containment vessel, drywell wall, weir wall, and pedestal wall with the mat are computed by the program by applying compatibility conditions at the interface of the mat with each of the above. Appendix 3A presents the design control measures that have been employed to demonstrate the applicability and validity of the MAT-6 program. Dynamic analysis of the reactor building provides acceleration profiles for the reactor building which are applied as static loads on the structure. Since these loads are asymmetric, the mat is analyzed using SHELL 1, a finite-difference computer program described in Appendix 3A.

The discontinuity forces, at the base of the drywell, weir wall, pedestal, shield building, and containment, obtained from these analyses are used as boundary conditions for the design of these structures.

The mat is analyzed for hydrodynamic loads as described in Appendix 6A.

With the exception of pipe tunnels and electrical tunnels, the foundations of all Seismic Category I structures and turbine and radwaste buildings are analyzed and designed using Finite Element Capability and ICES STRUDL II computer program for the loading combinations, as described in Section 3.8.4.3. The details of the computer program are discussed in Appendix 3A. The piping and electrical tunnels are analyzed and designed as described in Section 3.8.4.4.7.

3.8.5.5 Structural Acceptance Criteria

Structural design of all foundations, other than the reactor building mat, is in accordance with ACI 318-71, using ultimate strength design. Capacity reduction factors are taken as given in Section 9.2 of ACI 318-71.

The reactor building mat is designed so that the stresses in the concrete and reinforcing steel are within the limits specified by ASME Code, Section III, Division 2, Article CC-3000. Sliding and overturning factors of stability are as follows:

Loading Condition	Minimum Acceptable Stability Factor
Design Wind	1.50
OBE	1.50
SSE	1.10
Tornado	1.10
Floatation	1.10

•→12
A settlement monitoring program is included in the Technical Requirements Manual. Structural settlement is physically measured and compared to calculated predicted values. If the measured value exceeds the limits of the calculated predicted settlement then a special report is required.
12←•

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Materials, quality control, and special construction techniques used for the construction of foundations are the same as for other Seismic Category I structures, and are described in Section 3.8.4.6.

The normal high groundwater table at the station is at el 57 ft msl. The top of the reactor building foundation mat is at el 70 ft msl. During periods of extreme high flood water level, it is postulated that a small amount of water may seep through the reactor building mat and shield building wall and through the other Seismic Category I substructures.

Sumps are located at the lower level of all Seismic Category I buildings. They are capable of collecting seepage water due to flooding for removal by pumping.

3.8.5.7 Testing and Inservice Surveillance Requirements

Testing and inservice inspection is not planned for any foundation structure, other than the reactor building mat foundation which is load tested as described in Section 3.8.2.

References - 3.8

1. ANSI A58.1 - 1972, Building Code Requirements for Minimum Design Loads in Buildings and Other Structures.

2. Wind Forces on Structures, ASCE Paper No. 3269, Vol. 126, Part 2, 1961.

3. ASME Boiler and Pressure Vessel Code, Section III, Division 2, 1977.

4. Metal Bellows Corporation Report CR 682 Revision D, Documentation of Compliance for Hose Assembly, Flexible Metal Per ASME Boiler & Pressure Vessel Code Section III, Subsection NC, Class 2, February 15, 1983.

5. Metal Bellows Corporation Report CR 681 Revision E, Design Report for Hose Assembly, Flexible Metal Per ASME Boiler & Pressure Vessel Code Section III, Subsection NC, Class 2, March 4, 1983.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

Two inputs to Section 3.9 are provided: Section 3.9A applies to systems and components within SWEC scope of supply; Section 3.9B applies to systems and components within GE scope of supply.

3.9A MECHANICAL SYSTEMS AND COMPONENTS (SWEC SCOPE OF SUPPLY)

3.9.1A Special Topics For Mechanical Components

3.9.1.1A Design Transients for Non-NSSS Systems

Table 3.9A-1 lists the plant events that were used for the design and analysis of ASME Section III Class 1 components and supports. The table also shows the number of cycles per event and event classification. The application of these transients is discussed under load combinations in Section 3.9.3.1A. However, it should be noted that Transient 9a (Table 3.9A-1) is analyzed under both upset and faulted conditions.

3.9.1.2A Computer Programs Used in Analyses

The computer programs used in analyses are described, and their applicability and validity is demonstrated in Appendix 3A.

3.9.1.3A Experimental Stress Analysis

Experimental stress analysis for the design of non-NSSS equipment was not used.

3.9.1.4A Consideration for the Evaluation of the Faulted Condition

3.9.1.4.1A Equipment and Components

The elastic analysis techniques described in Section 3.7.3A are utilized in the qualification of Seismic Category I ASME code and non-code equipment within balance-of-plant scope. Stress limits utilized for the faulted plant condition are as outlined in Section 3.9.3.1A. The design conditions and stress limits defined are applicable for an elastic system (and equipment) analysis. Inelastic analyses have not been employed.

3.9.1.4.2A Piping Systems

Seismic Category I ASME Code Class 1, 2, 3 piping, and pipe supports are analyzed and designed in accordance with the requirements of ASME Section III, Subsections NB, NC, ND and NF, respectively. The analyses also comply with Appendix F of ASME Section III. The 1974 edition is used, with the following exceptions:

1. Building settlements - not applicable to the faulted condition - are analyzed according to the 1977 edition.

2. The number of OBE load cycles is based on Appendix N of the 1977 edition, Winter 1978 Addenda.

3. For pipe supports the 1974 Edition including the Summer 1974 Addenda and Summer 1976 Addenda (NF 2610 only) for standby cooling tower pipe supports.

4. Essential systems which are necessary to shut down the reactor or to mitigate the consequences of an accident comply with the functional capability requirements delineated in NEDO-21985 }2{in addition to the ASME Code.

5. For mechanical snubbers, the 1974 Edition including the addenda through Summer 1976 is utilized.

6. Weld repairs are made in accordance with Code Case N-275 with the restrictions specified in R.G. 1.84 as follows:

   "If an indication is removed and weld-metal layers still remain, it is not permitted to gouge or grind through the wall in order to avoid the MT or PT examinations. When complete weld metal removal is done to avoid the MT or PT examination, the cavity shall be MT or PT examined."

7. Code Case N-242-1 is used for material provided with mechanical snubbers.

8. For control rod drive (RDS) piping and supports installed by Reactor Controls, Inc., under SWEC Design Specification No. 228.180, the following edition of ASME III applies:

a) For pipe procurement, fabrication, and erection, 1977 Edition, including the Summer 1979 Addenda.

b) For pipe design, 1980 Edition, including the Summer 1982 Addenda.

c) For pipe supports design, procurement, fabrication, and erection, 1977 Edition, including the Summer 1979 Addenda.

Loadings considered in the faulted condition include the following:

1. Loading associated with normal plant conditions, including hydrodynamic loads associated with suppression pool phenomena

2. SSE

3. Dynamic system loading associated with the faulted plant conditions, that is, with the DBA - the break of a main steam line or a recirculation line

4. Dynamic system loading associated with the intermediate break accident (IBA) and small break accident (SBA).

Procedures for developing the loading functions in 1 and 2 above are described in Sections 3.9.1.5A and 3.7.3.8A. Loading functions in 3 and 4 are described in Section 3.6.2A. Loads associated with the suppression pool phenomena are described in Appendix 6A.

3.9.1.5A Analysis of Piping Systems

Seismic Category I piping systems (ASME Code Class 1, 2, 3), are analyzed in accordance with ASME Section III, 1974 edition, Subarticles NB-, NC-, and ND-3600. ANSI B31.1 seismically supported and nonseismic piping systems are analyzed in accordance with ANSI B31.1 Code, 1973 edition, including Summer 1974 Addenda. In addition, high energy piping systems are analyzed for pipe rupture criteria.

Analytical modeling and seismic analysis are described in Section 3.7.3.8A. Static analysis and other dynamic analyses which contribute the remaining stresses in the code stress criteria are described in the following sections.

3.9.1.5.1A Static Analysis

The static equation of equilibrium for the idealized system may be written in matrix form, as follows:

$$KU = P - Q \tag{1}$$

where:

K = Stiffness matrix for assembled system

U = Nodal displacement vector

P = External forces, weights, etc

Q = Equivalent thermal forces $= \int_0^L AE\alpha\overline{T}d$

A = Cross section area

E = Young's Modulus

α = Thermal expansion coefficient

T =Average wall temperature less 70°F installation temperature

O = Coordinate along pipe axis

L = Length of pipe

The unknown nodal displacements are obtained from NUPIPE by solving this equation using the Gauss method. The nodal displacements are then applied to the individual members, and member stiffnesses are used to find internal forces. The nodal displacements at support locations can be used along with the support stiffness to determine support reactions.

3.9.1.5.1.1A Dead Loads (Weight, Pressure) and Live Loads

The deadweight case is calculated by NUPIPE simultaneously with the pressure case. The analysis assumes all flexible restraints, such as spring hangers, to be rigid. If a pipe has different contents (medium) and therefore different weight in various flow modes, this is taken into consideration. Other details are discussed in Section 3.7.3.8.3.1A.

Live loads are considered insignificant and are therefore neglected.

3.9.1.5.1.2A Initial Displacements (Anchor Movements)

NUPIPE permits calculation of the thermal initial support displacements combined with the thermal response due to the average pipe wall temperature change.

Earthquake anchor movements are discussed in Section 3.7.3.8.3.5A. In ASME Code Class 1 analysis the loads due to OBE anchor movements are combined with the OBE inertia loads via absolute summation. In ASME Code Class 2 and 3 analysis the code permits their exclusion from occasional loads if they are included with the thermal expansion loads. NUPIPE is set up to include the earthquake anchor movement loads along with the thermal expansion loads.

3.9.1.5.1.3A Thermal Loads

A piping system may experience various operating modes. All operating modes are modeled as follows: Portions of piping with flowing medium have the temperature of the medium, while inactive branches have ambient temperature. Nonuniform temperature distributions along the pipe near branch connections of active and inactive legs are considered.

To analyze the condition following the isolation of an active branch for the steady-state temperature profile and overall average temperature of a long pipeline with one closed end, the program DET is employed.

In Class 1 analysis only, loads due to temperature distribution across the thickness of the pipe wall and due to geometric and material discontinuities during thermal transients, represented in ASME Section III, Subarticle NB-3600, by the terms with ΔT_1, ΔT_2, T_a, T_b, must be considered. These loads are obtained from the program TRHEAT or HTLOAD, based on geometry, fluid type, insulation, and environmental data. These programs are described in Appendix 3A.

3.9.1.5.2A Occasional Loads Excluding Seismic Loads

Occasional loads are also analyzed with NUPIPE. In the matrix equation of motion

$$M\ddot{U} + C\dot{U} + KU = F(t)$$

where:

M = Mass matrix

C = Damping matrix

K = Stiffness matrix

U = Displacement vector

the forcing function F(t) is applied as a set of force time histories, one for each mass degree-of-freedom which experiences a dynamic load.

3.9.1.5.2.1A Fluid Transients

Fluid transients are considered in the following systems:

1. Main steam and main steam bypass systems

2. Main steam safety/relief valve (SRV) discharge system

3. Moisture separator/reheater safety relief system

4. Feedwater system

5. Emergency core cooling systems (ECCS)

6. Standby service water (SSW) system.

The computer programs (Appendix 3A) used to calculate these force time histories due to water hammer, steam hammer, and pipe with air trapped in water lines, are WATHAM, STEHAM, and WATAIR, respectively.

3.9.1.5.2.2A Jet Impingement

The effects of direct jet impingement on piping are evaluated after all other piping analyses are completed and targets from all postulated breaks have been identified.

3.9.1.5.2.3A Relief Valve Reactions (Other Than MS SRVs)

Valves that are subjected to jet reaction forces are supported by static restraints adjacent to the valve body, in such a manner that the effects on the piping outside these restraints can be neglected.

3.9.1.5.2.4A Suppression Pool Induced Dynamic Loads in the Reactor Building

These loads are described and assessed in Appendix 6A.

3.9.1.5.3A Field-Run Piping

There is no field-run ASME Code Class piping in River Bend Station.

3.9.1.5.4A Load Combinations and Stress Criteria

In detailed analyses of ASME Code Class piping systems the individual load cases are combined as shown in the load combination tables, Tables 3.9A-2 and 3.9A-3. This is performed with NUPIPE.

In the simplified analysis for small bore piping (as defined in Section 3.7.3.8A) the same principle is followed; however, the resulting seismic spans, thermal offsets, and support loads are bounding values determined from several fundamental configurations.

The classification for ASME Class 1, 2, and 3 piping systems according to type of analysis is given in Table 3.9A-4.

3.9.1.5.5A Buried Piping

The safety-related buried piping identified in Section 3.7.3.12.1A is analyzed to ASME III Code Class 3 criteria. The load conditions analyzed and respective assumptions are as follows:

1. Internal pressure is used to determine minimum wall thickness for the piping in the event that the counterbalancing external pressure of the soil is not present at all times.

2. External pressure due to contact pressure between soil and pipe is evaluated with respect to the ASME code allowables. For a first order approximation the soil pressure is assumed uniform.

3. The buried piping is evaluated for the deadweight load of the soil acting on top of the pipe. The solution is based on a plain strain solution for a circular conduit buried in an infinite, linearly elastic medium subjected to a uniformly distributed overpressure. The medium is considered to be homogeneous, isotropic, and time independent.

Deformations are assumed to occur only in the plant perpendicular to the pipe axis.

4. Thermal expansion is evaluated for the maximum temperature range due to the restraint of axial movement from soil fraction or anchors. The axial stress developed due to the restraint of the pipe axial movement is added to the moment stresses in the evaluation of the code equations.

5. The settlements of the buildings where the piping is attached and yard areas where piping is buried are analyzed using the theory of beam on elastic foundation. The settlements are transformed into displacement boundary conditions to be imposed on the piping.

6. Concentrated surface loadings, such as traffic load, are analyzed by the same procedure as soil deadweight. The resulting pressure distribution is obtained from the Boussinesq solution of a concentrated point load on a semi-infinite elastic solid.

7. Seismic analysis of buried piping is discussed in Section 3.7.3.12.1A.

3.9.1.6A Suppression Pool Induced Dynamic Loads for Mechanical Equipment

These loads are described and assessed in Appendix 6A.

3.9.1.7A Piping Engineering and Design

A quality control representative witnesses, on a surveillance basis, the Seller's dimensional checks to ensure that the counterbore of pipe is in accordance with the specification and that the material specification minimum pipe wall thickness requirement has not been violated. In order to assure records on actual field end counterbores performed by the shop fabricator, the fabricator includes on his shop traveler the required and actual counterbore dimension as well as the required and actual pipe wall thickness after counterboring.

The preliminary stress analysis of Class 1 piping system is based on the assumption that the out-of-roundness is within 0.08t limit that is specified in Table NB-3681(a)-1 of ASME Section III. Technical justifications for these assumptions are described as follows:

1. While the out-of-roundness tolerances, established by ASME Code and piping specifications, are not sufficient to assure the 0.08t limit, the pipes are judged to be generally round (i.e., within the 0.08t limit to require no increase of k-index).

2. Out-of-roundness affects only the pressure term in the calculation of peak stresses. Its contribution to the overall peak stresses is not significant even if out-of-roundness conditions permitted by the design/fabrication specification are considered.

Final Class 1 pipe stress analyses are reviewed to account for out-of-roundness based on the fabrication limits established by the ASME Code and pipe specifications, unless more realistic field measurements are available.

3.9.2A Dynamic Testing and Analysis

3.9.2.1A Piping Vibration, Thermal Expansion, and Dynamic Effects

A detailed preoperational test program will be submitted 60 days before the start of the tests, as required by Regulatory Guide 1.68.

3.9.2.1.1A Flow Modes

Tabulated flow modes for various systems are provided as part of the above test program.

3.9.2.1.2A Preoperational and Startup Vibration Testing

Safety-related piping systems designated as Class 1, 2, or 3 are designed in accordance with ASME Section III. Each system is designed to withstand dynamic loadings from operational transient conditions that are encountered during expected service as required by NB-3622, NC-3622, and ND-3622 of the code.

During the preoperational and/or startup test program, vibration testing is performed on the following systems. Only those under the scope of SWEC are discussed in this section. Vibration testing for piping in the GE scope of supply is discussed in Section 14.2.12.3.29):

•→4

1. Residual heat removal system.

4←•

2. High-pressure core spray system
3. Low-pressure core spray system
4. Reactor core isolation cooling system
5. A portion of the feedwater system
6. Reactor water cleanup system
7. Component cooling water system
8. Main steam isolation valve seal system
9. Penetration valve leakage control system
10. Standby liquid control system
11. Fuel pool cooling and cleanup system
12. Fuel transfer system
13. Standby service water system
14. CRD system
15. Instrument air system
16. Service air system
17. Reactor plant ventilation system
18. Standby diesel generator air start system
19. Control building chilled water system.

BOP steady-state vibration testing consists of two separate testing phases. Phase I testing consists of visual screening of ASME Class 1, 2, and 3, and selected high and moderate energy piping systems at preselected locations. A specific list of monitored systems and locations at which visual observations will be made and will be contained in startup test and preoperational test procedures. These vibration visual observations are performed by engineers trained for excessive vibration screening. Any piping system viewed from the recommended observation distance which does not exhibit excessive vibration, is considered acceptable.

Calculations for observation distances (e.g., displacement) are based upon deflection equations given in the ANSI/ASME 0M-3 Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems with an allowable stress of (0.8/1.3)S for carbon steel piping and an allowable stress of S at 10^{11} cycles for stainless steel piping.

Phase II testing is performed on piping systems which exhibit excessive vibration during the visual screening. Phase II testing consists of taking a velocity and/or displacement reading using handheld vibration monitors. Criteria for displacement measurements are based on the ANSI/ASME OM-3 with assumptions as previously stated.

BOP transient vibration testing is performed at preselected data points. Two levels of acceptance criteria, Level I and II, are imposed. Level I and II criteria are defined in accordance with Sections 3.9.2.1.4.1B and 3.9.2.1.4.2B. Acceptance limits for Level I are based upon the ASME Boiler and Pressure Vessel Code Section III, Equation 9 for Class 1, 2, and 3 systems or the ANSI B31.1, Equation 12 for Class 4 (NNS) systems. Acceptance limits restrict the bending stress due to deflection plus stresses due to deadweight and pressure to a value less than the normal/upset allowable stress for occasional loads. Level II criteria are based on pipe stress and support loads not to exceed design basis predictions. Flow transients monitored are pump starts, pump stops, changes to system flows due to rapid valve position changes, and FSAR-designated system trips. A specific list of flow transients is contained in startup test and preoperational test procedures.

Small bore pipe testing of small bore piping branch connections on systems monitored for steady-state vibration is included as part of the visual observations. Control rod drive lines are instrumented for transient vibration with level 1 and level 2 acceptance limits as previously stated.

Essential instrument lines chosen for additional monitoring points are instrumented or visually examined for vibration. Acceptance criteria limits for these test points are based upon deflection equations given in ANSI/ASME OM-3 with stresses as defined above. The reactor pressure vessel level indicator instrumentation lines for monitoring both steam and water levels, main steam instrumentation lines for monitoring main steam flow, reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment, and instrumentation lines and systems identified in the INPO SER 64-83, "Fatigue Cracks and Leaks in Small Diameter Piping," dated September 12, 1983, are possible additional monitoring points for excessive vibration.

3.9.2.1.3A Preoperational Thermal Expansion Testing

Preoperational tests for BWRs are conducted near ambient conditions; therefore, thermal expansion testing during the preoperational test phase is very limited.

For the systems delineated in Section 3.9.2.1.2A that are operated at other than ambient conditions during the preoperational test phase, pipe deflections are observed or measured at selected locations. The startup expansion testing program is discussed in further detail in Section 3.9.2.1.2B.

3.9.2.1.4A Measurement Locations

The exact locations of measuring devices and identification of visual inspection points are supplied in the test program. Measurements taken at points with dynamic instrumentation show whether the stress and fatigue limits are within acceptable levels, and measurements taken at points with expansion instrumentation in an expansion test, excluding dynamic effects, are checked against displacement criteria.

3.9.2.1.5A Acceptance Criteria

Acceptance criteria for vibration on systems listed in Section 3.9.2.1.2A are specified in the vibration test program. The measured stress, representing the combined stress of pressure, deadweight, and fluid transient loads, for instance, can be combined with the analytical stress of the load cases not simulated, such as the OBE, and then compared with the combined analytical result. The allowable stresses are listed in the load combination tables, Tables 3.9A-2 and 3.9A-3.

The limits for thermal displacements depend on the equipment design parameters. Under all plant conditions the piping is not permitted to touch another object which may interfere with the operation of the piping system or equipment.

3.9.2.1.6A Corrective Actions

If during the vibration test it should be noted that the vibrations are beyond the acceptable design level, additional supports and restraints would be provided. The possibility of piping rerouting would also be considered, and a retest would be performed to assure that the design meets the acceptance criteria.

Analogously, if the design tolerances for thermal displacements are not satisfied at a point along the piping, the equipment affected can usually be realigned. Otherwise, additional supports and restraints would be provided, and pipe rerouting would also be considered.

3.9.2.2A Seismic Qualification of Safety-Related Mechanical Equipment

This section provides the qualification criteria and methods for equipment affected by seismic loads. The methods for the qualification of equipment affected by the suppression pool induced dynamic loads are provided in Appendix 6A, subsection 6A.17.

3.9.2.2.1A Seismic Qualification Criteria

The purpose of qualifying Seismic Category I mechanical equipment is to demonstrate its ability to perform a safety-related function during and after a postulated seismic occurrence, of a magnitude up to and including the SSE.

Equipment which does not perform any safety-related function, but whose failure could jeopardize the function of Seismic Category I equipment, is required only to maintain its structural integrity.

Seismic qualification of equipment is accomplished by one of the four methods discussed in Section 3.7.3.1A. Analysis is used to demonstrate structural integrity of the equipment. When mechanical equipment is qualified by analysis or test, the acceptance criteria and margins of safety are in accordance with Section 3.9.2.2.2A. Where the equipment is classified as active, additional deflection analysis and/or testing is performed. Details of qualification methods for specific equipment are contained in Table 3.9A-5.

These methods are applied to mechanical equipment as follows:

1. Analysis

    a. The listing below is for equipment where the maintenance of structural integrity only is required to assure performance of the design-intended function. This equipment is qualified by analysis:

(1) Piping

(2) Ductwork

(3) Tanks and vessels

(4) Heat exchangers

(5) HVAC components

(6) Non-active pumps and valves.

b. Analysis is also used to qualify rotating machinery items where verification must be obtained to demonstrate that deformations resulting from seismic loadings do not cause binding of the rotating element, to the extent that the component cannot perform its design-intended function. Components in this category include:

(1) Active pumps

(2) Fans.

2. Dynamic Testing - Equipment whose functional capability cannot be adequately demonstrated by analysis, is qualified by dynamic testing. Equipment qualified by dynamic testing is listed below:

a. Standby diesel generator components

b. Hydrogen recombiner control panels

c. Electric motor valve actuators, including limit switches

d. Pneumatic and hydraulic valve actuators, including limit switches and solenoid valves

e. Electrical control panels, relay boards, switchgear and motor control centers, radiation monitoring equipment

f. Control instrumentation such as flow switches, thermocouples,and transmitters

g. Batteries, battery chargers, inverters

h. Electrical penetrations.

3. Combination of Analysis with Testing - A combination of analysis with static or dynamic testing is used for seismic qualification of active valves, as follows:

    a. The natural frequencies of the valve assembly are determined by analysis or test.

    b. A static deflection test is performed to verify that deformation due to seismic loadings does not cause binding of internal valve parts, which prevents valve operations within specified time limits.

    c. The electric motor-driven valve actuator and other electrical appurtenances are qualified by dynamic testing.

    The equipment that is qualified by testing is mounted and operated in a manner similar to the actual system. For testing procedures refer to Section 3.7.3A.

3.9.2.2.2A Acceptance Criteria

The acceptance criteria used are as follows:

1. Tests, when used, demonstrate that the component performs its required safety function during and after the test. The TRS generally envelops the applicable frequency range of the RRS with the suggested margin in accordance with IEEE 323-1974. Where the TRS does not envelop the RRS with the suggested margins of IEEE 323-1974, a justification is provided.

2. Analysis, when used, verifies that stresses do not exceed the specified allowable stress limits, for the loading conditions shown in Tables 3.9A-6 and 3.9A-7 and that deformations do not exceed those that will permit the component to perform its design-intended function.

For ASME components, the specified allowable stress limits are those shown in Tables 3.9A-8 and 3.9A-9.

For non-ASME components, the Design Condition I loading has allowable stresses limited to 75 percent of the minimum yield strength at the design temperature of the material, in accordance with applicable ASTM specification. For the Design Condition II loading the stresses do not exceed the smaller of:

1. 100 percent of the minimum yield strength, or

2. 70 percent of the minimum ultimate tensile strength of the material (at temperature), in accordance with the ASTM or equivalent specification for the material.

For definitions of Design Conditions I and II, see Section 3.9.3.1.2.1A.

3.9.2.2.3A Seismic Qualification of Specific Non-NSSS Mechanical Equipment

3.9.2.2.3.1A Piping

Piping is seismically qualified by analysis only. Seismic analysis is described in Section 3.7.3.8A. In the reactor building the piping is subjected to floor- and wall-induced vibrations of a nature similar to a seismic event. The analysis of these phenomena and verification of the analysis in certain critical areas of strong vibration input by testing are described in Appendix 6A.

3.9.2.2.3.2A Tanks

The safety-related tanks have been seismically qualified as follows:

1. The seismic analysis on the buried standby diesel generator fuel oil storage tank consisted of the following:

    a. Selection of the applicable seismic acceleration factors at the elevation in the diesel generator building at which the tank is installed.

    b. Calculation of the lowest natural frequency of the filled tank in its buried environment taking into account both the mass and spring rate of this environment. This frequency occurs in the rigid range.

c. Choice of the correct frequency range for seismic factor selection by combining analysis parameters a and b.

d. Determination of loads both on the tank and support rings by static stress analysis with seismic g-factors applied to all tank and sand masses.

e. ASME Code methods, for the design of the tank shell, heads, stiffening, and support rings were used. Local stress analysis, by Bijlaard or other methods, as appropriate, were utilized in determining stresses at nozzles and support rings.

f. Analyses were performed for both normal and upset conditions (including live and dead loads, thermal and pressure stresses, and OBE seismic factors) and faulted conditions composed of live and dead loads plus full SSE inertial loads.

g. Adequacy of the tank at design pressure was determined. The tank was hydrotested at 1.5 times design-pressure in compliance with ASME Code.

2. The seismic analysis for the air damper/accumulators, the main control room chilled water compression tank and the standby diesel generator fuel oil day tank, consisted of the following:

a. An analysis of the vessel has been performed to prove that it has rigid characteristics, i.e., the natural frequency of vibration of the predominant mode of the supported vessel is in the flat portion of the applicable response spectrum curves.

b. The seismic acceleration coefficients were applied statically, and a static analysis performed on the equipment and supports. The vertical and horizontal seismic effects were applied simultaneously to the subject vessel at its gravitational center for the seismic load calculation and design.

3. Since the MSIV, ADS, and SRV accumulators are located inside the reactor building, seismic as well as hydrodynamic effects were considered in their analysis. Steps 2a and 2b were therefore done with the applicable acceleration coefficients.

3.9.2.2.3.3A Pumps

Qualification of pumps is discussed in Section 3.9.3.2A. The results of typical tests and analyses are described in Table 3.9A-5.

3.9.2.2.3.4A Valves

The qualification of active valves is discussed in Section 3.9.3.2A. The qualification type and summary of test results are described in Table 3.9A-5.

3.9.2.2.3.5A Other Mechanical Equipment

The qualification method for mechanical equipment other than the above is discussed in Section 3.7.3A.

The qualification results are described in Table 3.9A-5.

3.9.2.2.3.6A Electrical Equipment and Instrumentation

The seismic qualification criteria and methods of qualification of Seismic Category I electrical equipment and instrumentation other than those items discussed in this section, are described in Section 3.10.

3.9.2.2.3.7A Cranes

Cranes are seismically qualified in accordance with criteria that:

1. Preclude the possibility of the crane being dislodged by a seismic disturbance

2. No part of the crane becomes detached during an earthquake.

3.9.2.3A Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

See Section 3.9.2.3B.

3.9.2.4A Preoperational Flow-Induced Vibration Testing of Reactor Internals

See Section 3.9.2.4B.

3.9.2.5A Dynamic System Analysis of the Reactor Internals Under Faulted Condition

See Section 3.9.2.5B.

3.9.2.6A Correlations of Reactor Internals Vibration Tests with the Analytical Results

See Section 3.9.2.6B.

3.9.3A ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1A Loading Combinations, Design Transients, and Stress Limits

3.9.3.1.1A ASME Section III Class 1 Components

3.9.3.1.1.1A Equipment

Seismic Category I, Class 1 mechanical equipment (i.e., valves) is designed in accordance with ASME Section III, Subsection NB. This equipment is listed in Table 3.9A-11. Loading combinations for each service condition are outlined in Table 3.9A-6. These loading combinations incorporate the applicable design transients identified in Table 3.9A-1. Stress limits corresponding to each service condition are in accordance with NB-3000, as demonstrated in Table 3.9A-8.

These load descriptions include Dynamic Load 1, Dynamic Load 2, and Dynamic Load 3 notations, which are load combinations for equipment in the reactor building only, resulting from consideration of hydrodynamic loading conditions; for equipment outside the reactor building, these reduce to OBE, SSE, and OBE, respectively. Further discussion of these combinations is given in Appendix 6A.

For the conditions specified, the allowable stress limits defined in Table 3.9A-8 are applicable to stress results obtained by elastic analysis techniques. The analysis methods described in Section 3.7.3A are used in implementing this criterion. Computer programs used in these analyses are discussed in Appendix 3A.

3.9.3.1.1.2A Piping

The pipe stress analysis load combinations and stress limits for ASME Class 1 piping are given in Table 3.9A-2. The design transients and number of associated stress cycles for the various plant conditions are given in Table 3.9A-1. This table includes the dynamic load events OBE, SSE, LOCA-related load cases, and SRV discharge cases. The suppression pool events are discussed in Appendix 6A. There are several SRV cases. Table 3.9A-2 uses only the envelope of the response of all cases (SRV). However, for the calculation of the usage factor, in cases where the stress of the 16-valve case significantly exceeds the combined 1-valve/2-valve stress, these two load cases are separated, since the 16-valve case has a much smaller number of cycles.

Fatigue analysis includes all events with 25 or more significant primary or secondary stress cycles. ECCS systems are required to meet the allowable primary stresses of the normal and upset conditions, in order to assure their continued operation during the emergency or faulted event.

ASME Class 1 piping meets the criteria of ASME Section III, 1974 edition, and of 10CFR50.55a, Section (d).

Analysis of the individual load cases is described or referred to in Section 3.9.1.5A.

3.9.3.1.2A ASME Class 2 and 3 Components

3.9.3.1.2.1A Equipment

Tables 3.9A-7 and 3.9A-9 list loading conditions and stress limits for ASME Section III Class 2 and 3 components of the Seismic Category I fluid systems constructed in accordance with ASME Section III, Subsections NC and ND. These conditions are:

1. Design Condition I - Includes the specified design loads (temperature, pressure, etc), plus Dynamic Load 1 loads

2. Design Condition II - Includes the specified design loads (as above), plus Dynamic Load 2 loads, plus pipe rupture loads (if applicable).

The design loading combinations are analagous to either the Code Class 1 Normal or Upset conditions for Design Condition I and to the faulted condition for Design Condition II. See the notes following Table 3.9A-7 for the definitions of Dynamic Load 1 and Dynamic Load 2.

These requirements, which supplement the present scope of ASME Section III, Subsections NC and ND, are consistent with the present code format and philosophy. Further extension of terminology (normal, upset, etc) is not required, since Class 2 and 3 systems are not generally evaluated for such varieties of operating conditions and transients, but rather to design conditions which conservatively envelop all operating conditions.

Generally, only design conditions of pressure and temperature are necessary to satisfy ASME code requirements. These conditions envelop all service level conditions for the component such as normal, upset, emergency and faulted plant conditions. Use of design conditions plus seismic loading is considered a conservative criterion.

The stress limits and design conditions presented in Table 3.9A-9 are intended to ensure that no gross deformation of the component occurs. These limits are applicable for an elastic system (and component) analysis. No inelastic analysis has been performed for any ASME Class 2 or 3 component.

All ASME Code Class 2 and 3 components that are required to shut down the reactor or mitigate the consequences of a postulated piping failure without offsite power meet allowable stress and deformation limits. Items that are within 10 percent of allowable limits are:

1. Fuel pool cooler lower bracket anchor bolts

2. Fuel pool cooling pumps impellors

3. Standby service water pumps impellors

4. Standby diesel generator fuel oil transfer pumps column bolts, nozzle flanges, anchor bolts, discharge head flanges, and adapter plates

5. Standby diesel generator day tank nozzles, anchor bolts, and saddle-to-vessel stress

6. Standby diesel generator storage tank support ring (including webs and welds) and nozzle wall thicknesses

3.9.3.1.2.2A Piping Systems

The load combinations and stress limits for ASME Class 2 and 3 piping are given in Table 3.9A-3. They conform with the criteria of ASME Section III. These criteria imply elastic analysis. Under faulted condition, with primary stress limit 2.4S , gross inelastic deformations may occur but are permitted by the code.

Analysis of the individual load cases is described or referred to in Section 3.9.1.5A. The application of detailed or simplified analysis depends on criteria stated in Table 3.9A-4.

3.9.3.1.3A Compliance with Regulatory Guide 1.48

Table 3.9A-12 compares, for each component type and class, the load combinations and the stress limits of the regulatory guide with those used for the plant design. Exceptions to the regulatory guide are listed and justified in the right column of the table.

3.9.3.1.4A Bolting Stress Limits

The allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections are given by the following:

1. Anchor bolts used in equipment anchorage - Appendix B of ACI 349-1976, Code Requirements for Nuclear Safety Related Concrete Structures.

2. Bolts used in component supports - ASME Code, Section III, Division I (1977 Edition), Subsection NF, paragraph NF-3280, and Appendix XVII, paragraph 2460; and ASME Code Case N-71-8 (1644-8). For service levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum allowable stresses at temperature.

3. Bolts used in flanged connections - ASME Code, Section III (1974 Edition).

3.9.3.2A Pump and Valve Operability Assurance

This section provides the operability assurance programs for pumps and valves affected by seismic loads. The operability assurance programs for pumps and valves affected by the suppression pool induced dynamic loads are provided in Appendix 6A, subsection 6A.17.

Pumps and valves installed in Seismic Category I piping systems are designed in accordance with the requirements of ASME Section III, Subsections NB, NC, and ND. Tables 3.9A-10 and 3.9A-11 list the active pumps and valves, respectively. Active pumps and valves that are appurtenances of an assembly and are not individually identifiable by a mark number are not listed. Instead, the assembly is listed in Tables 3.9A-5 and 3.10A-1.

Table 3.9A-21 provides a sample comparison of RBS valve specification requirements, for motor-operated carbon steel valves 2 1/2 in and larger, with ANSI 278.1-1975, Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard, as supplemented by Regulatory Guide 1.148.

Active components are those whose operability is relied upon to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a postulated pipe break in the RCPB.

Nonactive components are those whose operability (e.g., valve opening or closure, pump operation or trip) is not relied upon to perform the system function during the transients or events considered in the respective operating condition category.

Safety-related valves are qualified by testing and analysis, and safety-related active pumps by analysis with appropriate stress limits and nozzle loads. The content of these programs is detailed in the following sections.

3.9.3.2.1A Pump Operability Program

All active pumps are qualified for operability by being subjected to tests both prior to installation in the plant and after installation in the plant. The in-shop tests include:

1. Hydrostatic tests to ASME Section III requirements

2. Seal leakage tests at the same pressure used in the hydrostatic tests

3. Performance tests while the pump is operated with flow to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements and other pump/motor parameters.

Also monitored during these operational tests are bearing temperatures and vibration levels which are shown to be below appropriate limits specified to the manufacturer for design of each active pump.

After the pump is installed in the plant, it undergoes cold hydro tests, hot functional tests, and the required periodic in-service inspection and operation as applicable. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the safety-related active pumps are qualified for operability during an SSE condition by assuring that 1) the pump is not damaged during the seismic event, and 2) the pump continues operating when subjected to the SSE loads.

The pump manufacturer is required to show that the pump operates normally when subjected to the maximum applicable amplified seismic (floor) accelerations, attached piping nozzle loads, and dynamic system loads associated with the faulted plant operating condition. Analysis procedures are utilized in accordance with those outlined in Section 3.7.3A. Natural frequencies are determined in order to obtain maximum seismic accelerations based on applicable amplified (floor) response spectra.

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the values indicated in Table 3.9A-9. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur. A static shaft deflection analysis of the rotor is performed with horizontal and vertical accelerations based on floor response levels. The deflections determined from the static shaft analysis are compared to allowable rotor clearances.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9A-9 as allowables, assures that critical parts of the pump are not damaged during the short duration of the faulted condition;

therefore, the reliability of the pump for post-faulted condition operation is not impaired by the seismic event.

In addition to the post-faulted condition operation, it is necessary to assure that the pump functions throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., no rotation. Typically, the rotor can be seized 5 full sec before a circuit breaker shuts down the pump to prevent damage to the motor. However, the high rotary inertia in the operating pump rotor, and the random nature and short duration loading characteristics of the seismic event prevent the rotor from becoming seized. In actuality, the seismic loadings cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump does not shut down during the SSE and operates at the design speed despite the SSE loads.

Seismic analysis of the assembly (i.e., pump, motor, and supporting structure) was performed to ensure pump operability and acceptable qualification of the entire assembly. Additionally, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment that is identified to be vital to the operation of the pump or pump motor, and that is not qualified for operation during the pump analysis or motor qualifications is separately qualified for operation at the accelerations that it would see at its mounting. The pump motor and vital auxiliary equipment are qualified by meeting the requirements of IEEE 344-1975.

The functional ability of active pumps after a faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads are identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

Results of analyses and tests are included in Table 3.9A-5.

3.9.3.2.2A Valve Operability Program

Safety-related active valves are required to perform their mechanical motion during and after the course of a postulated accident. Assurance must be supplied that these valves can operate during and after a seismic event. Qualification tests accompanied by analyses are conducted for all active valves.

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Valves without significant extended structures are considered seismically adequate as a result of piping seismic adequacy (see ASME Section III, Paragraph NB-3524). For valves with operators having significantly extended structures, which are essential for maintaining pressure integrity, analysis is based upon static forces resulting from equivalent seismic accelerations acting at the center of gravity of the operator. For "active" valves, as committed in Project position in Reg guide 1.48 of Table 1.8-1, operability is checked by performing a static deflection test. A static load (equivalent to that produced by SSE conditions) is applied at the operator centroid, with simultaneous operation of the pressurized valve during and after the test.

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The safety-related valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Section III requirements, main seat leakage tests, disc hydrostatic test, functional tests to verify that the valve opens and closes within the specified time limits when subjected to the design differential pressure, and operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation, etc) according to IEEE 323-1974 and IEEE 382-1972. Cold hydrodynamic qualification tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operation are performed to verify and ensure the functional ability of the valve. An analysis of the extended structure is also performed for static equivalent seismic SSE loads. The maximum stress limits allowed in these analyses assure the maintenance of structural integrity. The limits used for Class 2 and 3 active valves are shown in Table 3.9A-9.

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In addition to these tests and analyses, representative valves of each design type, pressure, and size group are tested for verification of operability during a simulated seismic event, by demonstrating operational capabilities within the specified limits. The testing procedures are described below.

The valve is mounted in a manner which conservatively represents the actual valve installation. The valve assembly includes the operator and all appurtenances normally attached to the valve in service. The operability of the valve during the faulted condition is demonstrated as follows:

1. The actuator and yoke of the valve system are statically loaded by an amount equal to that determined from an analysis as representing faulted accelerations applied at the center of gravity of the operator about the weaker axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

2. The valve is then operated while in the deflected position, i.e., from the normal operating mode to the faulted operating mode. The valve is required to perform its safety-related function within the specified operating time limits.

3. Electric motor operators and other electrical appurtenances necessary for operation are qualified in accordance with IEEE 323-1974, IEEE 344-1975, and IEEE 382-1972.

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The accelerations used for the valve qualification are 3.0 g horizontal and 3.0 g vertical. The piping design maintains the motor operator accelerations to these levels with an adequate margin of safety.

Adequacy of the valve body is demonstrated through piping analysis. Low stresses in the valve body preclude the possibility of significant distortion, and therefore the possibility of binding of internal components. Based on this consideration, pipe end loads need not be simulated during the operability tests.

For valves where the stresses in the valve body could be significant, the piping end loads are imposed during the operability tests. Examples include solenoid valves and air-operated control valves.

For selected active valve categories specific qualification programs are conducted to demonstrate operability. The method of qualification for these valves is detailed below:

Butterfly Valve

The containment and drywell vent/purge isolation valves are evaluated for operability during a postulated accident by both analysis and testing methods.

1. The valve assembly is analytically evaluated and shown to perform its safety-related function (i.e., to close within the required time). The analysis of the valve considers seismic, hydrodynamic, operating, and LOCA loads.

2. The valve assembly is statically loaded by an amount equal in magnitude to the dynamic force and applied at the actuator C.G. The design pressure of the valve is simultaneously applied and the valve is operated while in the deflected position.

3. Electrical appurtenances (limit switches and solenoid-operated valves) are qualified in accordance with the requirements of IEEE 323-1974 and 344-1975.

4. In addition, assurance of operability is demonstrated by the following tests:

   a. In-shop shell hydrostatic tests

   b. Cold cyclic tests

   c. Seat leakage tests

   d. Pre- and post-installation functional tests

Check Valves

Check valves are characteristically simple in design, and their operation is not affected by seismic accelerations or the applied nozzle loads. Check valve design is compact, and there are no extended structures or masses whose motion could cause distortions or restrict operation of the valve. The nozzle loads due to maximum dynamic excitation do not affect the functional ability of the valve since the valve disc is designed to be isolated from the casing wall. The clearance supplied by the design around the disc prevents the disc from becoming bound or restricted due to any casing distortions caused by nozzle loads. Therefore, the design of these valves is such that when the structural integrity of the valve is assured, using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valves are also subjected to the following tests and analysis:

1. Stress analysis, including the faulted loads and disc impact loads

2. In-shop hydrostatic test

3. In-shop seat leakage test

4. Periodic in situ valve exercising and inspection to assure the functional ability of the valve.

For the feedwater check valves, the operability following a postulated feedwater line break is also demonstrated. The maximum disc impact velocity and the pressure differential across the disc are determined. A stress analysis of the valve, which considers the impact and the seismic inertia loads, demonstrates the valve's adequacy.

The basic criteria used in selecting the representative valve for qualification testing are based on an evaluation of the following parameters:

1. Valve assembly weight

2. Valve size, type, and pressure ratings

3. Valve actuator type and performance characteristics

4. Mounting arrangement of the valve and its appurtenances.

The methodology utilized in assessing the degree of similarity and evaluating the differences generally follows the guidelines of ANSI/ASME Standard B16.41, Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants.

Using the methods described, all the safety-related valves in the system are qualified for operability during the faulted event. These methods conservatively simulate the faulted event and ensure that the active valves can perform their safety-related function when required.

3.9.3.3A Design and Installation Details for Mounting of Pressure-Relief Devices

The installation criteria for mounting of all pressure relief devices (safety and relief valves) and for specifying materials, fabrication, examination, testing, inspection, stamping, and reporting are in accordance with the rules in Subsections NB, NC, and ND of ASME Code, Section III, applicable to the classification of the piping systems involved.

The design criteria for all SRVs are in accordance with the rules in Paragraphs NB-3677 and NC-3677, applicable to classification of the piping system involved. In particular, the design criteria and analyses used to calculate maximum stresses and stress intensities are in accordance with Subarticles NB-3600 and NC-3600. Maximum dynamic stresses on each valve nozzle are calculated based upon full discharge dynamic loads (thrust and bending) and internal design pressure. Maximum stress intensity in the run pipe or header, under full discharge loads (thrust, bending, and torsion) and internal design pressure are also computed[(1)].

Calculation of the dynamic load factor associated with the functioning of open-type SRVs is based on modeling the valve and nozzle as a single-degree-of-freedom dynamic system. The lumped mass of this system corresponds to the weight of the valve and nozzle and is assumed to be at the valve center of gravity. The rotational degree-of-freedom of this system is considered to be in the direction that causes maximum bending stress in the nozzle at the junction of the nozzle and run-pipe. Rotational flexibility of the system is computed by a series combination of nozzle flexibility and local run-pipe flexibility (at the junction of the nozzle and run-pipe).

The rise time of the discharge force at the outlet of the safety valve elbow is assumed to be the minimum valve opening time, and the discharge force is assumed to rise linearly with time. The ratio of maximum dynamic rotations predicted by this single-degree-of-freedom system to the static rotation caused by the steady-state discharge force represents the dynamic load factor.

For safety or relief valve(s) mounted on a common header and full discharge occurring concurrently, the additional stresses induced in the header are combined with the previously computed local and primary membrane stresses to obtain the maximum stress intensity.

Steam transients in the main steam SRV discharge lines (SRVDL) are calculated using STEHAM (see Appendix 3A). This computer program solves the one-dimensional compressible flow equations of mass, momentum, and energy by the method of characteristics with finite difference approximation for time and space. The boundary conditions of relief valve mass flow and water slug dynamics are used. The water slug was modeled for the changing mass and conservation of momentum.

The solution of the three conservation equations with the boundary conditions provides the distribution of flow velocities, pressures, and densities through the pipe for each time step. From these properties the unbalanced axial forces on a piping segment are calculated by the momentum equation applied to this control volume. These unbalanced forcing functions were then applied to the piping system for dynamic analysis.

A typical isometric drawing of the discharge line and supports is shown in Fig. 3.9A-1. The corresponding set of typical forcing function plots is provided in Figures 3.9A-2 through 3.9A-12. Note that the positive direction for these forces is opposite to the direction of steam flow. The plots for segments 10 (Fig. 3.9A-11) and 11 (Fig. 3.9A-12) include the reaction forces due to clearing of the water slug.

Relief valve loads and piping reactions for other safety relief discharge piping for safety related systems are calculated by the time history method as described above for the main steam SRVs. For nonsafety related systems, the evaluation of SRV discharge piping is performed using Moody's method. Maximum thrust forces are calculated by:

$$F_{max} = \left(1.26\ P_O - P_\infty\right)A_E \text{ for ideal gas (steam hammer)}$$

$$F_{max} = 2\left(P_O - P_\infty\right)A_E \text{ for incompressible fluid (water hammer)}$$

This force (F_{max}) is fully developed in a pipe segment between two adjacent pipe bends only if the distance between these bends is equal to or larger than the wave traveling distance, $d_W = V_W\ t_V$

where:

P_O = Valve set pressure

P_∞ = Pressure at downstream of valve

A_E = Nozzle orifice area

V_w = Sonic velocity in the fluid

t_v = Valve opening time

The thrust forces in pipe segments which have lengths between two adjacent bends less than d_w are calculated by the ratio

$$F = F_{max}\left(\frac{\quad}{d_w}\right)(DLF)$$

where:

DLF = Dynamic load factor, determined in accordance with Reference 5.

Then, the forces for the pipe segment are statically applied to the piping system.

3.9.3.4A Component Supports

3.9.3.4.1A Pipe Supports

The pipe support designs, using base plates and concrete expansion anchor bolts, are performed using the flexibility criteria of NRC IE Bulletin 79-2 before they are released for fabrication. Verification of as-built conditions to address NRC IE Bulletin 79-14 is provided in Section 3.7.3.8.1A.

3.9.4.4.1A Nonnuclear Piping

Nonnuclear piping satisfies the requirement of ANSI B31.1 Code 1973, including Summer 1974 Addenda, paragraphs 120 and 121.

3.9.3.4.1.2A Nuclear Piping

Pipe supports are designed in accordance with either subsection NF of ASME III or both subsection NF of ASME III and AISC specification, depending on how the jurisdictional boundaries are considered.

When the complete pipe support assembly is considered to be under ASME III jurisdiction, the design of the pipe support follows the requirements of subsection NF of ASME III. The interpretation of jurisdictional boundaries, in these cases, is as follows:

1. Paragraph NF 1131.1, Portion A is defined as the pressure retaining component. This includes pipe and mechanical equipment.

2. Paragraph NF 1131.2, Portion B is defined as the element integrally attached to Portion A and is the intervening element for Portion C. For design purposes this is considered to be pads with their attachment welds and welds of all other integral attachments to the pipe or mechanical equipment or pad, along with the portion of the integral attachment adjacent to the weld. For procurement and installation purposes this is considered to be pads, all other elements integrally attached to the pipe or mechanical equipment or pad and their associated connection welds.

3. Paragraphs NF 1131.3, Portion C and NF 1131.4, Portion D are the elements that come under NF requirements. This includes all items not governed by Portions A, B, and E, such as the remaining portions of the integral attachment, components, frames, base plates, and all welds that connect these elements to the next load-carrying element in the load path to the building structure.

4. Paragraph NF 1131.5, Portion E is defined as the load-carrying building structure. This is considered to be the building structure and all elements in it such as expansion bolts, concrete inserts, and supplementary steel.

When the pipe support assembly is considered to be under both ASME III and non-ASME III jurisdiction, the interpretation of jurisdictional boundaries (so called alternate jurisdictional boundaries) is as follows:

1. The alternate jurisdictional boundaries for NF applications extend from the ASME III pipe transmitting loads to the first structural members or component standard supports and include welds attaching these NF-members or component standard supports to other non-NF members such as structural steel, supporting frames, base plates or embedment plates, etc.

2. The definitions of integral attachments and pressure-retaining components are the same as those described in the interpretation of those supports which fall within ASME III jurisdiction only.

When supports are designed utilizing alternate jurisdictional boundary criteria, the non-NF members and non-NF welds are designed to meet allowable stress limitations of both ASME III code and AISC specification. When the design is based on the AISC specification, it includes all the load conditions required for ASME III design (see Table 3.9A-13 for load conditions). However, for load conditions 3, 4, and 5, the allowable stress values used for load conditions 1 and 2 are increased by one-third.

When integral attachments are used, local pipe stresses are checked and connection welds are designed in accordance with Subsections NB, NC, and ND of ASME Section III. An alternate procedure for the evaluation of the design for hollow circular cross section welded attachments on Class 2 and 3 piping will be in accordance to Code Case N-392. For the design of rectangular cross section attachments on Class 2 or 3 piping, Code Case N-318 may be used and a listing of applications is given in Table 3.9A-22.

Fillet welds that fall within NF jurisdiction are designed in accordance with Subsection NF of the ASME Code, Section III, 1974 edition. However, when the weld size is controlled by minimum fillet weld size requirements based on the thickness of the two parts joined, the largest minimum fillet weld size is limited to 5/16 inch.

An alternate method to determine minimum size of fillet welds for NF linear-type pipe supports will be in accordance with Code Case N-413. (Stress allowables of Table NF-3292.1-1 of the 1974 Code will apply.)

Table 3.9A-13 lists the load conditions, load combinations, and allowable stresses. Loads are applied in whatever manner is necessary to attain the worst possible stress levels for all support elements.

The calculated compressive stresses for pipe supports and building steel members will not exceed two-thirds of the Column Research Council (CRC) column strength curve[6].

All pipe supports, except snubbers, are qualified by analysis only.

The design criteria and dynamic testing requirements for component and pipe supports listed in the following paragraphs are applicable under all plant operating conditions.

1. Component Supports - All component supports are designed, fabricated, and assembled so they cannot become disengaged by the movement of the supported pipe or equipment during operation. All component supports are designed in accordance with the rules of Subsection NF of ASME Section III.

2. Spring Hangers - The design load on spring hangers is the load caused by deadweight alone. They are calibrated to ensure that they support the deadweight at both their hot and cold load settings. Spring hangers also allow for a down-travel and up-travel in excess of the specified thermal movement to account for dynamic movement.

3. Rod Hangers - Rod hangers are only used as a rigid restraint when there is no possibility of compression.

4. Struts - The design loads on struts include those loads caused by deadweight, thermal expansion, primary seismic forces (OBE and SSE), system anchor displacements, and reaction forces caused by relief valve discharge and turbine stop valve closure, etc.

   Struts are designed in accordance with NF-3000.

5. Snubbers (Mechanical) - The design load on snubbers includes all dynamic loads such as seismic forces (OBE and SSE), system dynamic anchor movements, and reaction forces caused by short duration relief valve discharge and turbine stop valve closure, and dynamic loads produced by suppression pool phenomena.

   The snubbers are designed and load-rated in accordance with NF-3000 to be capable of carrying the design load for all dynamic operating conditions. Faulted condition design uses the criteria outlined in Appendix F of the ASME Code.

   The prototype snubbers have been tested dynamically to ensure that they can perform as required in the following manner:

   a. The snubber was subjected to a force that varied approximately as the sine wave.

b. The frequency (Hz) of the input force was verified at small increments within the specified range.

c. The resulting relative displacements and corresponding loads across the working components, including end attachments, were recorded.

d. The test was conducted with the snubber at various temperatures.

e. The peak load in both static tension and compression tests was higher than the rated load followed by an operational test.

f. The duration of the tests at each frequency was specified.

g. Snubbers were tested for various abnormal environment conditions, including salt-fog, sand and dust, and humidity followed by operational tests.

   The environmental test results are filed at the snubber manufacturer's location. The other test results are forwarded with the shipment of each snubber and are incorporated into the permanent plant file.

6. Anchors - Anchors are designed to restrain all rotations and translations of piping. Terminal anchors are anchors which are common to two independently analyzed piping subsystems, one on each side of the anchor. For each load type the anchor loads on each side of the anchor are combined to form a total anchor load for that load type. For vibratory loads the total anchor load component is ` the square root sum of the squares (SRSS) of two components. The NRC evaluation of Reference 3 and Revision 1 of NUREG-0484 }4{ provides justification for the use of the SRSS method for these dynamic loading combinations. For static loads the total anchor load is the algebraic sum of loads from both sides of the anchor.

   Design transient cyclic data are not applicable to piping supports, as no fatigue evaluation is necessary to meet the code requirements.

The use of U-bolts on safety-related piping is limited to tight access conditions and sliding arrestors on trapese spring hangers. Vendor specified load ratings developed in accordance with ASME III, Subsection NF are used in these designs.

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Code Case N-314 has been used for the partial thread engagement of lock nuts, where the lock nuts are defined as either jam nuts, heavy hex nuts cut to half (so-called half nuts), or heavy hex nuts with a minimum of 50 percent thread engagement. The 50 percent thread engagement for the heavy hex nut used as a lock nut has been established by considering it to be the same as the full engagement length for a half nut or jam nut. However, when these lock nuts are partially engaged, where partial engagement for a heavy hex nut is considered to be less than 50 percent, the design drawing used for construction is revised to show such engagement and the partial engagement length is established by analysis.

3.9.3.4.2A Pump Supports

The pump pedestal and pedestal bolt analysis includes consideration of loads from operating and seismic events, connecting pipes, temperature, and deadweight. The stress limits of ASME Section III Subsection NF are met. The analysis includes deflection of the pedestal.

3.9.3.4.3A Other Component Supports

Component supports are designed in accordance with Subsection NF of ASME Section III (1977 Edition up to and including the Winter 1978 Addenda) Classes 2 and 3. The combinations of design loadings categorized with respect to plant operating conditions identified as Service Limits A, B, C, and D, which have been specified for the design of ASME Code constructed items, are presented in Table 3.9A-14. The corresponding stress limits are as defined in ASME Section III, Subsection NF.

The stress results for each of the components listed below appear in Tables 3.9A-16 through 20. The specified component operating condition is consistent with the plant operating condition for each transient event.

1. RHR heat exchanger supports and restraints

2. Fuel pool cooling and cleanup heat exchanger supports and restraints

3. Standby diesel generator fuel oil day tank supports

4. Quencher supports and restraints. Quencher design information is discussed in Appendix 6A, Section A.6A.7.2.

5. SSW pump supports.

The RWCU regenerative and nonregenerative heat exchanger supports are designed to the AISC code.

U-bolts are not used in the installation of component supports. High strength bolts (A325 steel) are used for the mounting of the following equipment:

1. Main steam isolation valve accumulators located at el 142 ft in the reactor building and el 134 ft in the auxiliary building.

2. Chilled water compression tanks located at el 98 ft in the control building.

3.9.4A Control Rod Drive Systems

See Section 3.9.4B.

3.9.5A Reactor Pressure Vessel Internals

See Section 3.9.5B.

3.9.6A Inservice Testing of Pumps and Valves

Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves is performed in accordance with ASME OM Code for Operation and Maintenance of Nuclear Power Plants and applicable addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i).

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The examinations are in accordance with an approved ASME OM code. In addition, RBS has submitted a separate inservice inspection program document, including pumps and valves, which complies with "NRC Staff Guidance for Complying with Certain Provision of 10CFR50, 55a(g) - Inservice Inspection Requirement."
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3.9.6.1A Inservice Testing of Pumps

The pumps provided with an emergency power source, are tested according to the requirements of ASME OM Code, Subsection ISTB. The hydraulic and mechanical parameters to be measured or observed are discussed and identified in the aforementioned document.

3.9.6.2A Inservice Testing of Valves

Valves are tested according to the requirements of ASME OM Code, Subsection ISTC. Parameters to be measured or observed are discussed and defined in the aforementioned inservice inspection document. Pressure isolation valves which are to be included in the testing program are listed in Table 3.9A-15.

The following system valves that provide interface between the reactor coolant pressure boundary (RCPB) Class 1 and Class 2 piping are not included in the list:

1. Standby liquid control system
2. Main steam safety and relief valve system
3. Sampling system
4. Drain, leakage monitoring connection (LMC), and vent valves

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The standby liquid control system valves are not tested because these valves are always sealed in the closed position and are only actuated by explosive devices. Also, this system is located inside containment.
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The main steam safety and relief valves seat leakage is discharged to the suppression pool (inside containment); therefore, there are no high pressure occurrences that exceed the Class 3 piping design pressure. Also, the seat leakage during plant operation is not limited to a specific maximum amount in the closed position for fulfillment of their function as required by ASME OM Code, Subsection ISTC-1300, Valve Categories, Category A.

The sampling system reactor coolant system interface valves are not included because valve seat leakage is not limited to a specific maximum in the closed position for fulfillment of their function as required by ASME OM Code, Subsection ISTC-1300, Valve Categories, Category A. Also, this system terminates inside containment, and a restriction in the sample line limits reactor coolant loss at normal system pressure. The maximum loss of reactor coolant is well within the capability of the reactor coolant makeup system (Section 9.3.2.3).

Manual drains, leakage monitoring connections, and vent valves are not included because they are excluded by ASME OM Code, Subsection ISTC-1200. Also, these connections are provided with a threaded cap to limit leakage if valve leakage does occur.

References - 3.9A

1. K. R. Wichman, et al. Local Stress in Spherical and Cylindrical Shells due to External Loadings. Welding Research Council Bulletin WRC-107, 1965.

2. Functional Capability Criteria for Essential Mark II Piping, NEDO-21985, September 1978, prepared by Battelle Columbus Laboratories for General Electric Company.

3. Study to Demonstrate the Generic Applicability of SRSS Combination of Dynamic Responses for Mark III Nuclear Steam Supply System and Balance-of-Plant Piping and Equipment Components. Report SMA 12109.01-R001 prepared for General Electric Company, Nuclear Energy Division, by Structural Mechanics Associates, November 1981.

4. Methodology for Combining Dynamic Responses, USNRC Report, NUREG-0484, Rev. 1, May 1980.

5. Biggs, J. M. Introduction to Structural Dynamics, McGraw-Hill Book Company, New York, NY, 1964

6. Friedrich Bleich, Buckling Strength of Metal Structures, McGraw-Hill Book Company, New York, NY, 1952.

3.9B MECHANICAL SYSTEMS AND COMPONENTS (GE SCOPE OF SUPPLY)

3.9.1B Special Topics For Mechanical Components

3.9.1.1B Design Transients

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This section describes the transients which are used in the design of major NSSS ASME Section III, Code Class 1 and Class 2 components. The number of cycles or events for each transient is included. These transients are included in the design specifications and/or stress reports for components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Section III, Subsection NCA, as applicable. (The first four conditions correspond to Service Levels A, B, C, and D, respectively.) The transient descriptions listed in the following sections are based on 100% power. To obtain the actual uprated temperatures and pressures, refer to the GE cycle diagrams for 105% uprate and the certified design specification for the Thermal Power Optimization (TPO) (Appendix K) uprate.

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3.9.1.1.1B CRD Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40-yr life of the Control Rod Drive (CRD) are as follows:

Transient	Category	Cycles
1. Reactor startup/shutdown	normal/upset	120
2. Vessel pressure tests	normal/upset	130
3. Vessel overpressure	normal/upset	10
4. Scram tests	normal/upset	140
5. Startup scrams	normal/upset	160
6. Operational scrams	normal/upset	300
7. Jog cycles	normal/upset	30,000
8. Shim/drive cycles	normal/upset	1,000

In addition to the above transients, the following transients have been considered in the design and fatigue analysis of the CRD.

Transient	Category	Cycles
9. Scram with inoperative buffer	normal/upset	24
10. Operating Basis Earthquake (OBE)*	normal/upset	10
11. Safe Shutdown Earthquake	faulted	1
12. Scram with stuck control blade	faulted	1
13. Control rod ejection accident	faulted	1

All ASME Section III, Class 1 components of the CRD have been analyzed according to ASME Section III Boiler and Pressure Vessel Code.

3.9.1.1.2B CRD Housing and Incore Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

Transient	Category	Cycles
1. Startup and shutdown	normal/upset	120
2. Design pressure tests	normal/upset	403
3. Loss of feedwater pumps	normal/upset	10
4. Relief or safety valve blowdown	normal/upset	8
5. Scrams	normal/upset	180
6. Operation Basis Earthquake (OBE)*	normal/upset	10
7. Safe Shutdown Earthquake (SSE)**	emergency -CRD HSG faulted-Incore HSG	1 1
8. Stuck rod scram - CRD HSG only-	normal/upset	1
9. Scram no buffer - CRD HSG only-	normal/upset	1

3.9.1.1.3B Hydraulic Control Unit Transients

The transients used in the design and analysis of the Hydraulic Control Unit and its components are:

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* The frequency of this transient indicates an emergency category. However, for conservatism, this OBE condition is analyzed as a normal and upset event. A single event is assumed to consist of 10 stress cycles. See Section 3.7.3.2B for further discussion.

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** SSE is a faulted condition; however, in the CRD housing stress analysis report, it was treated as an emergency condition with lower stress limits thus making the comparison of results to the allowable more conservative.

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Transient	Category	Cycles
1. Reactor startup/shutdown	normal/upset	120
2. Scram tests	normal/upset	140
3. Startup scrams	normal/upset	160
4. Operational scrams	normal/upset	300
5. Jog cycles	normal/upset	30,000
6. Shim/drive cycles	normal/upset	1,000
7. Scram with stuck scram discharge valve	emergency	1
8. OBE*	normal/upset	10
9. SSE	faulted	1

3.9.1.1.4B Core Support and Reactor Internals Transients

The cycles listed in Table 3.9B-1 were considered in the design and fatigue analysis for the reactor internals.

3.9.1.1.5B Main Steam System Transients

The following transients are considered in the stress analysis of the main steam piping:

Main Steam Transients

Transient	Category	Cycles
1. Hydrotest	test	40
2. Leaktest	test	360
3. Startup	normal	120
4. Turbine trip	upset	10
5. Scram and trip isolation valves open	upset	40
6. Scram	upset	140
7. Shutdown	normal	111
8. Loss of feedwater pumps isolation valves closed	upset	10
9. Turbine bypass single relief or safety valve	upset	8
10. Reactor over pressure delayed scram	emergency	1
11. Automatic blowdown	emergency	1

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* The frequency of this transient indicates an emergency category. However, for conservatism, this OBE condition is analyzed as a normal and upset event. A single event is assumed to consist of 10 stress cycles. See Section 3.7.3.2B for further discussion.

Transient	Category	Cycles
12. Operating basis earthquake (OBE)	upset/normal	50
13. Turbine stop valve closure (TSV)	upset	600
14. Relief valve lift (RVL)	upset	5433

3.9.1.1.6B Recirculation System Transients

The following transients are considered in the stress analysis of the recirculation piping:

Recirculation Transients

Transient	Category	Cycles
1. Hydrotest	test	40
2. Startup	normal	120
3. Turbine trip	upset	10
4. Partial feedwater heater bypass	upset	70
5. Turbine generator trip F.W. on isolation valves open	upset	40
6. Scram	upset	140
7. Shutdown	normal	111
8. Loss of feedwater pumps isolation valves closed	upset	10
9. Turbine bypass single S/RV blowdown	upset	8
10. Reactor overpressure with delayed scram	emergency	1
11. Automatic blowdown	emergency	1
12. Operating Basis Earthquake (OBE)	upset/normal	50
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13. Single Loop Operation	normal	25
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3.9.1.1.7B Reactor Assembly Transients

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The reactor pressure vessel assembly includes the reactor pressure vessel, support skirt, and shroud support including leg, cylinder, and plate. The cycles listed in Table 3.9B-1 were specified in the reactor assembly design and fatigue analysis. Technical Specification component cyclic or transient limits are listed in Table 3.9B-22.

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3.9.1.1.8B Main Steam Isolation Valve Transients

The main steam isolation valves are designed for the following service conditions and thermal cycles:

Transient	Category	Cycles
1. Startup and shutdown		
a. Heating cycle @ 100°F/hr	normal/upset	300
b. Cooling cycle @ 100°F/hr	normal/upset	300
c. `29°F between 70°F and 552°F	normal/upset	600
d. `50°F step change between 70°F and 552°F	normal/upset	200
2. Loss of feedwater pump/MSLIV closure		
a. 552°F to 573°F in 3 sec (ΔT = 21°F heating)	normal/upset	10
b. 573°F to 525°F in 9 min (ΔT = 4°F cooling)	normal/upset	10
c. 525°F to 573°F in 6 min (ΔT = 48°F heating)	normal/upset	10
d. 573°F to 485°F in 7 min (ΔT = 88°F cooling)	normal/upset	10
e. 485°F to 573°F in 8 min (ΔT = 88°F heating)	normal/upset	10
f. 573°F to 485°F in 7 min (ΔT = 88°F cooling)	normal/upset	10
3. Single relief valve blowdown		
a. 552°F to 375°F in 10 min (ΔT = 177°F cooling)	normal/upset	8
4. Reactor overpressure with delayed scram		
a. 552°F to 586°F in 2 sec (ΔT = 34°F heating)	emergency	1
b. 586°F to 561°F in 30 sec (ΔT = 25°F heating)	emergency	1
5. Automatic and blowdown (ADS)		
a. 552°F to 375°F in 3.3 min (ΔT = 177°F cooling)	emergency	1
b. 375°F to 259°F in 19 min (ΔT = 116°F cooling)	emergency	1
6. Pipe rupture and blowdown		
a. 552°F to 259°F in 15 sec (ΔT = 293°F cooling)	faulted	1

3.9.1.1.9B Safety/Relief Valve Transients

The transients used in the analysis of the safety/relief valves are as follows:

Transient	Category	Cycles
1. Heating and cooldown - within the temperature limits of 70°F and 552°F at a rate of 100°F/hr	normal/upset	300
2. Small temperature changes- of 29°F (either increase or decrease) at any temperature between the limits of 70°F and 552°F	normal/upset	600
3. 50°F temperature changes - (either increase or decrease) at any temperature between the limits of 70°F and 552°F	normal/upset	200
4. Loss of feedwater pumps, isolation valve closure	normal/upset	10
5. Turbine bypass, single relief or safety valve blowdown (temperature drops from 552°F to 375°F in 10 minutes)	normal/upset	8
6. Reactor overpressure with delay scram - (temperature rise from 552°F to 586°F in 2 sec, and the pressure rises from 1050 psig to 1375 psig, immediately followed by a cooling transient in which the temperature drops from 586°F to 561°F in 30 sec and the pressure drops to 1125 psig)	emergency	1
7. Automatic blowdown -(temperature changes from 552°F to 375°F in 3.3 min, immediately followed by a change from 375°F to 259°F in 19 min)	emergency	1
8. Pipe rupture and blowdown - (temperature drops from 552°F to 259°F in 15 sec)	faulted	1

9. Installed hydrotests - valve inlet nozzle and disc shall be designed to withstand the following:		
a. hydrotests to 1045 psig at 100°F	testing	120
b. steam line flooding during plant shutdown	other	120

Paragraph NB3552 of ASME III code excludes various transients and provides means for combining those which are not excluded. Review and approval of the equipment suppliers certified calculations provides assurance of proper accounting of the specified transients.

3.9.1.1.10B Recirculation Flow Control Valve Transients

The following pressure and temperature transients were considered in the design of the recirculation system flow control valve:

Transient	Category	Cycles
1. Startup (100°F/hr heatup rate 70°F to design temperature)	normal/upset	300
2. Small temperature changes (29°F step)	normal/upset	600
3. 50°F step changes	normal/upset	200
4. Safety/relief valve blowdowns (single valve) (552°F to 375°F in 10 min)	normal/upset	8
5. Safety valve transient (110% of design pressure)	normal/upset	1
6. Installed hydrotests		
a. 1300 psig	testing	130
b. 1670 psig	testing	3
7. Automatic blowdown 552°F to 375°F in 3.3 min and 375°F to 281°F in 19 min	emergency	1
8. Improper start of pump in cold loop (100°F to 552°F over a period of 15 sec)	emergency	1

3.9.1.1.11B Recirculation Pump Transients

The following pressure and temperature transients were considered in the design of the recirculation pumps:

Transient	Category	No. of Cycles
1. Bolt up	Normal-Upset	123
2. Design hydrotest	Testing	40
3. Startup, turbine roll and increase to rated power	Normal-Upset	120
4. Daily power reduction - 75%	Normal+Upset	10,000
5. Weekly power reduction - 50%	Normal+Upset	2,000
6. Rod pattern change	Normal+Upset	400
7. Loss of feedwater heaters	Normal+Upset	80
8. Scrams	Normal+Upset	180
9. Special normal operation transients	Normal+Upset	20
10. Shutdown	Normal+Upset	111
11. Unbolt	Normal+Upset	123
12. Scram - Loss of feedwater pumps isolation valves close	Normal+Upset	10
13. Scram - Turbine bypass single relief or safety relief valve blowdown	Normal+Upset	8
14. Reactor overpressure with delayed scram feedwater stays on isolation valves stay open	Emergency	1
15. Scram - Automatic blowdown	Emergency	1
16. Improper start of pump in cold loop	Emergency	2
17. Improper startup with reactor drain shutoff followed by turbine roll and increase to rated power	Emergency	1
18. Pipe rupture and blowdown	Faulted	1

3.9.1.1.12B Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves:

Transient	Cycles
1. 70°F-575°F-70°F of 100°F/hr	300
2. ±29°F between limits of 70°F and 575°F, instantaneous	600
3. ±50°F between limits of 70°F and 575°F, instantaneous	200

4.	552°F to 375°F, in 10 min	8
5.	552°F to 281°F, in 22.3 min	1
6.	100°F to 552°F, in 15 sec	1
7.	110% of design pressure at 575°F	1
8.	1300 psi at 100°F installed hydrostatic test	130
9.	1670 psi at 100°F installed hydrostatic test	3

3.9.1.2B Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific components. (Computer programs were not used in the analysis of all components, thus, not all components are listed.)

Subsections 3.9.1.2.1B through 3.9.1.2.5B, 3.9.1.3.3B, and 3.9.1.4.3B reference computer programs utilized by GE and vendors for analyzing NSSS components. These NSSS programs can be divided into two categories:

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verificaion of input, output, and methodology is documented in GE Design Record Files.

a. SPECA
b. SNAP
c. CREEP-PLAST
d. ANSYS
e. SAP4
f. ANSI7
g. NOZAR
h. RVFOR
i. TSFOR
j. PDA
k. EZPYP
l LION
m. WTNOZ
n. WBHFN
o. CRDSSO1

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and vendors. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-Stamped equipment is in full compliance with 10CFR50, Appendix B.

Pump & Motor Vendor Programs

a. ANSYS
b. RTRMEC
c. FMAP
d. FLTFLG
e. MULTISPAN
f. 2DFMAP
g. CRISP
h. BIJLAARD

CB&I Programs

a. GENOZZ
b. NAPALM
c. 1027/BIJLAARD
d. 846
e. KALNINS
f. ASFAST
g. TEMAPR
h. PRINCESS
i TGRV
j. E0962A
k. 984
l. GASP
m. DUNHAM'S
n. 1335
o. HAP
p. 1635
q. 953
r. 1666
s. 1684
t. E1702A
u. MESHPLOT
v. 1028
w. 1038

3.9.1.2.1B Reactor Vessel and Internals

3.9.1.2.1.1B Reactor Vessel

The computer programs used in the preparation of the reactor vessel stress report are identified and their use summarized in the following paragraphs.

3.9.1.2.1.1.1B CB&I Program 7-11 - "GENOZZ"

The GENOZZ computer program is used to proportion barrel and double taper type nozzles to comply with the specifications of the ASME Code, Section III and contract documents. The program will either design such a configuration or analyze the configuration input into it. If the input configuration does not comply with the specifications, the program modifies the design and redesign it to yield an acceptable result.

3.9.1.2.1.1.2B CB&I Program 9-48 - "NAPALM"

The basis for the program NAPALM, Nozzle Analysis Program--All Loads Mechanical, is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. The program analyzes at specified locations from the point of application of the mechanical loads. At each location the maximum stress intensity is calculated for both the inside and outside surfaces of the nozzle. The program gives the maximum stress intensity for both the inside and outside surfaces of the nozzle as well as its angular location around the circumference of the nozzle from the reference location. The principal stresses are also printed. The stresses resulting from each component of loading (bending, axial, shear, and torsion) are printed, as well as the loadings which caused these stresses.

3.9.1.2.1.1.3B CB&I Program 1027/BIJLAARD

This program is a computerized version of the analysis method contained in the "Welding Research Council Bulletin 107, Dec 65."

Part of this program provides for the determination of the shell stress intensities (S) at each of four cardinal points at both the upper and lower shell plate surfaces (ordinarily considered outside and inside surfaces) around the perimeter of a loaded attachment on a cylindrical or spherical vessel. With the determination of each S, there is also determined the components of that S (2 normal stresses,
σ_x and σ_y, and one sheer stress τ). This program provides the same information as the manual calculation and the input data is essentially the geometry of the vessel and attachment.

3.9.1.2.1.1.4B CB&I Program 846

This program computes the required thickness of a hemispherical head with a large number of circular parallel penetrations by means of the area replacement method in accordance with the ASME Code, Section III.

In cases where the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

3.9.1.2.1.1.5B CB&I Program 781 - "KALNINS"

This program is a thin elastic shell program for shells of revolution. This program was developed by Dr. A. Kalnins of Lehigh University. Extensive revisions and improvements have been made by Dr. J. Endicott to yield the CB&I version of this program.

The basic method of analysis was published by Professor Kalnins in the Journal of Applied Mechanics, Volume 31, September 1964, pages 467 through 476.

The KALNINS thin shell program (Program 781) is used to establish the shell influence coefficient and to perform detail stress analysis of the vessel. The stresses and the deformations of the vessel can be computed for any combination of the following axi-symmetric loading:

1. Preload condition
2. Internal pressure
3. Thermal load

3.9.1.2.1.1.6B CB&I Program 979 - "ASFAST"

ASFAST program (Program 979) performs the stress analysis of axi-symmetric, bolted closure flanges between head and cylindrical shell.

3.9.1.2.1.1.7B CB&I Program 766 - "TEMAPR"

This program reduces any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient has the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plate thickness and actual temperature distribution. The output contains the average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV language.

3.9.1.2.1.1.8B CB&I Program 767 - "PRINCESS"

The PRINCESS computer program calculates the maximum alternating stress amplitudes from a series of stress values by the method in Section III of the ASME Code.

3.9.1.2.1.1.9B CB&I Program 928 - "TGRV"

The TGRV program is used to calculate temperature distributions in structures or vessels. Although it is primarily a program for solving the heat conduction equations, some provisions have been made for including radiation and convection effects at the surfaces of the vessel.

The TGRV program is a greatly modified version of the TIGER heat transfer program written about 1958 at Knolls Atomic Power Laboratory, by A. P. Bray. There have been many versions of TIGER in existence including TIGER II, TIGER II B, TIGER IV, and TIGER V, in addition to TGRV.

The program utilizes an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments and continually summed. Linear mathematics are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer, conduction, radiation and convection, as well as internal heat generation.

Given any odd-shaped structure, which is represented by a three-dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates, TGRV calculates and gives as output the steady state or transient temperature distributions in the structure as a function of time.

3.9.1.2.1.1.10B CB&I Program 962 - "E0962A"

Program E0962A is one of a group of programs (E0953A, E1606A, E0962A, E0992N, E1037N, and E0984N) which are used together to determine the temperature distribution and stresses in pressure vessel components by the finite element method.

Program E0962A is primarily a plotting program. Using the nodal temperatures calculated by program E1606A or program E0928A, and the node and element cards for the finite element model, it calculates and plots lines of constant temperature (isotherms). These isotherm plots are used as part of the stress report to present the results of the thermal analysis. They are also very useful in determining at which points in time the thermal stresses should be determined.

In addition to its plotting capability, the program can also determine the temperatures of some of the nodal points by interpolation. This feature of the program is intended primarily for use with the compatible TGRV and finite element models that are generated by program E0953A.

3.9.1.2.1.1.11B CB&I Program 984

Program 984 is used to calculate the stress intensity of the stress differences, on a component level, between two different stress conditions. The calculation of the stress intensity of stress component differences (the range of stress intensity) is required by Section III of the ASME Code.

3.9.1.2.1.1.12B CB&I Program 992 - GASP

The GASP computer program, originated by Prof. E. L. Wilson of the University of California at Berkeley, uses the finite element method to determine the stresses and displacements of plane or axi-symmetric structures of arbitrary geometry and is written in FORTRAN IV. For a detailed account, see the following reference document:

> Wilson, E.L.; "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties" Aerojet General Corporation, Sacramento, California. Technical Memorandum No. 23, July 1965.

As mentioned previously, the program determines the stresses and displacements of plane or axi-symmetric structures using the finite element method. The structures may have arbitrary geometry and have linear or non-linear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

The structure to be analyzed is broken up into a finite number of discrete elements or "finite-elements" which are interconnected at finite number of "nodal-points" or "nodes." The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes. The basic input to the program consists of the geometry of the stress-model and the boundary conditions. The program then gives the stress components at the center of each element and the displacements at the nodes, consistent with the prescribed boundary conditions.

3.9.1.2.1.1.13B CB&I Program 1037 - "DUNHAM'S"

DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacements of axi-symmetric structures of arbitrary geometry subjected to either axi-symmetric loads or non-axi-symmetric loads represented by Fourier series.

This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle non-axi-symmetric loads (which requires that each node have three degrees of freedom) and the material properties for DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.1.14B CB&I Program 1335

To obtain stresses in the shroud support, the baffle plate must be considered as a continuous circular plate. The program makes this modification and allows the baffle plate to be included in CB&I program 781 as two isotropic parts and an orthotropic portion at the middle (where the diffuser holes are located).

3.9.1.2.1.1.15B CB&I Programs 1606 and 1657 - "HAP"

The HAP program is an axi-symmetric nonlinear heat analysis program. It is a finite element program and is used to determine nodal temperatures in a two-dimensional or axi-symmetric body subjected to transient disturbances. Programs 1606 and 1657 are identical except that 1606 has a larger storage area allocated and can thus be used to solve larger problems. The model for program 1606 is compatible with CB&I stress programs 992 and 1037.

3.9.1.2.1.1.16B CB&I Program 1635

Program 1635 offers three features to aid the stress analyst in preparing a stress report.

1. Generates punched card input for program 767 (PRINCESS) from the stress output of program 781 (KALNINS).

2. Writes a stress table in a format such that it can be incorporated into a final stress report.

3. Has the option to remove through-wall thermal bending stress and report these results in a stress table similar to the one mentioned above.

3.9.1.2.1.1.17B CB&I Program 953

The program is a general purpose program, which does the following:

1. Prepares input cards for the thermal model.

2. Prepares the node and element cards for the finite element model.

3. Sets up the model in such a way that the nodal points in the TGRV model correspond to points in the finite element model. They have the same number so that there is no possibility of confusion in transferring temperature data from one program to the other.

3.9.1.2.1.1.18B CB&I Program 1666

This program is primarily written to calculate the temperature differences at selected critical sections of the nuclear reactor vessel components at different time points of thermal transients during its life of operation and list them all in a tabular form. Since there is no involved calculation applicable particularly to nuclear components, this program can be used with any other kind of model that sees thermal transients over a period of time. This program helps ascertain the time points in thermal transients when the thermal stresses may be critical.

3.9.1.2.1.1.19B CB&I Program 1684

This program is written to better expedite the fatigue analysis of nuclear reactor components as required by the ASME Code, Section III. Specifically, this program is an expansion of an earlier program, 984. The features of this program allow the user to easily perform the complete secondary stress and fatigue evaluation including partial fatigue usage calculation of a component in one run. An additional option allows the user to completely document the input stress values in a format suitable for a stress analysis report. The program is written to allow for a minimum amount of data handling by the user once the initial deck is established.

3.9.1.2.1.1.20B CB&I Program "E1702A"

This program evaluates the stress-intensity factor K due to pressure, temperature, and mechanical load stresses for a number of different stress conditions (times) and at a number of different locations (elements). It then calculates the maximum RT_{NDT} the actual material can have based on a 1/4T flaw size and compares it with the ordered RT_{NDT}. If the ordered RT_{NDT} is larger than the maximum RT_{NDT}, the maximum allowable flaw size is calculated. The rules of Appendix G are used except that WRC 175 can be used to calculate K_I due to pressure in a nozzle to shell junction.

For a more thorough description of the fracture problem, see Welding Research Council Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials."

3.9.1.2.1.1.21B CB&I Program 955 "MESHPLOT"

This program plots input data used for finite element analysis. The program plots the finite element mesh in one of three ways: without labels, with node labels, or with element labels. The output consists of a listing and a plot. The listing gives all node points with their coordinates and all elements with their node points. The plot is a finite element model with the requested labels.

3.9.1.2.1.1.22B CB&I Program 1028

This program calculates the necessary form factors for the nodes of the model which simulates heat transfer by radiation. Inputs are shape and dimensions of the head-to-skirt knuckle junction. The program is limited to junctions with a toroidal knuckle part.

3.9.1.2.1.1.23B CB&I Program 1038

This program calculates the loads required to satisfy the compatibility between the shroud buffle plate and the jet pump adaptors in RPV.

3.9.1.2.1.2B Reactor Internals

3.9.1.2.1.2.1B Response Spectra Program/SPECA

SPECA is a GE proprietary computer program developed to generate amplified response spectra for arbitrary piece-wise linear acceleration time histories with uniform time step. The program calculates maximum acceleration responses for a series of single degree-of-freedom damped systems subjected to the excitation of input time history motions. The requirements of Regulatory Guide 1.122 are incorporated in the program such that spectrum ordinates are computed at frequency intervals small enough to produce accurate response spectra. This program is capable of generating individual as well as enveloped/peak-broadened spectra consistent with Regulatory Guide 1.122 for multiple load cases. This program provides response spectra for various ratios of critical damping.

3.9.1.2.1.2.2B Other Programs

The following programs are used in the analysis of the core support structure and other safety related reactor internals: SNAP, CREEP-PLAST, and ANSYS. Details of these programs are provided in Section 4.1.

3.9.1.2.2B Piping

The computer programs used in the analysis of NSSS piping systems within GE's scope of supply are identified and their use summarized in the following paragraphs.

3.9.1.2.2.1B Piping Analysis/PISYS

PISYS is a computer code specialized for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include: distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option of response spectrum analysis and the results are documented in Reference 7.

3.9.1.2.2.2B Component Analysis/ANSI 7

The ANSI 7 Computer Program determines stress and accumulative usage factors in accordance with NB-3600 of ASME Code, Section III. The program was written to perform stress analysis in accordance with the ASME sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.2.3B Dynamic Forcing Functions

3.9.1.2.2.3.1B Relief Valve Discharge Pipe Forces Computer Program/RVFOR

The relief valve discharge pipe connects the relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time dependent forces to develop in the pipe wall. This computer program computes the transient fluid mechanics and the resultant pipe forces using the method of characteristics.

3.9.1.2.2.3.2B Turbine Stop Valve Closure/TSFOR

The TSFOR program computes the time history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

3.9.1.2.2.4B Piping Dynamic Analysis Program/PDA

The pipe whip analysis was performed using the PDA computer program. PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-dependent stress-strain relations are used to model the pipe and the restraint. Similar to the popular elastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using moment-rotation relations, nonlinear equations of motion are formulated using energy considerations and the equations are numerically integrated in small time steps to yield the time-history of the pipe motion.

3.9.1.2.2.5B Piping Analysis Program/EZPYP

EZPYP links the ANSI-7 and SAP program together. The EZPYP program can be used to run several SAP cases by making user specified changes to a basic SAP pipe model. By controlling files and SAP runs the EZPYP program gives the analyst the capability to perform a complete piping analysis in one computer run.

3.9.1.2.2.6B Thermal Transient Program/LION

The LION program is used to compute radial and axial thermal gradients in piping. The program calculates a time history of ΔT_1, ΔT_2, T_a, and T_b (defined in ASME Section III, Class 1 piping analysis) for uniform and tapered pipe wall thickness.

3.9.1.2.2.7B WTNOZ Computer Program

WTNOZ is a timeshare program for piping weight calculations.

3.9.1.2.3B Recirculation Pump Program/ANSYS

The ANSYS Code using finite element methods is used in the analysis of the recirculation pump casing for various thermal and mechanical loads during plant operating and postulated conditions.

In general, the finite element techniques are used to solve temperature distribution in heat transfer transient problems, and to perform stress analysis for various thermal and mechanical loadings by using the same finite element model representing the pump body. The output of these programs is in the form of temperature profiles, deflections, and stresses at the nodal points of the finite element idealization of the pump structure.

3.9.1.2.4B ECCS Pumps and Motors

3.9.1.2.4.1B Motor Rotor Program/RTRMEC

RTRMEC is a computer program which calculates and displays results of mechanical analysis of motor rotor assembly when acted upon by external forces at any point along the shaft (rotating parts only). The shaft deflection analysis including magnetic and centrifugal forces was analyzed. The calculation for the seismic condition assumes that the motor is operating and that the seismic, magnetic, and centrifugal forces all act simultaneously and in phase on the rotor-shaft assembly. Note that the distributed rotor assembly weight is lumped at the various stations, with the shaft weight at a station being the sum of one-half the weight of the incremental shaft length just before the station, plus one-half the weight of the adjacent incremental shaft length just after the station. Bending and shear effects are accounted for in the calculations.

3.9.1.2.4.2B Two-Dimensional Frame Program/FMAP

FMAP is a computer program which solves for the natural frequencies and associated mode shapes of a two-dimensional frame. The frame is defined as a system of uniform, weightless members whose ends, or joints, are rigidly attached. All weights are lumped at the joints. Each joint has three degrees of freedom: two translations in the plane of the frame and a rotation about the axis normal to the plane. The frame is in the X-Y plane, and all motion of the frame is in this plane.

3.9.1.2.4.3B Flange Stress Program/FLTFLG

FLTFLG computer program determines stresses in bolted flanged connections where the flanges are flat faced and bolted together directly or separated by a metal spacer such that there is metal to metal contact beyond the bolt circle. Calculation procedure follows rules set forth in Appendix II, Part B, ASME Section VIII.

3.9.1.2.4.4B Bending Analysis Program/MULTISPAN

MULTISPAN is a computer program which performs the bending analysis of variable cross-section continuous beams up to ten spans. The analysis yields reactions, internal forces, displacements, and maximum shear and bending stresses.

3.9.1.2.4.5B Two-Dimensional Lumped-Mass Frame Program/ 2DFMAP

2DFMAP is a computer program which solves for the natural frequencies and the associated mode shapes of a rigidly jointed, two-dimensional lumped-mass frame. The solution is based on small-deflection theory assuming linear stiffnesses for the frame. Stiffness matrix alterations can be used to add complex structural elements which cannot be represented by members. Gaussian elimination is available to reduce the size of the stiffness matrix if relatively small weights are associated with any degree of freedom. The frequencies and mode shapes are computed using the Householder-Sturm method and inverse iteration.

3.9.1.2.4.6B Rotating Elements Vibration Mode Program/ CRISP

CRISP computer program determines the fundamental and harmonic modes of lateral vibration of rotating elements of arbitrary flexural rigidity. The computational method is based on a transfer matrix representation of the rotor shaft which includes the effect of multiple supports with dissimilar elasticity and damping in the bearings and with dissimilar elasticity and mass of the bearing supports. In addition to calculating the natural frequencies, the program provides lateral deflections for the determination of rotor stresses, running clearances, and severity of vibration at the different resonant frequencies. Vibration amplitudes of the bearing supports are also provided for determining support resonant frequencies and for obtaining an optimum design through modifications of the bearing and their supports.

3.9.1.2.4.7B Nozzle Shell Program/BIJLAARD

BIJLAARD analyzes stresses at the nozzle shell intersection by methods described in Welding Research Council Bulletin No. 107. Additional description is provided in Section 3.9.1.2.1.1.3B.

3.9.1.2.5B RHR Heat Exchangers Natural Frequency Program/ WBHFN

WBHFN calculates the natural frequency of the RHR heat exchanger considering the stiffness of the supporting steel structure. In this program the heat exchanger is modeled as a flexible beam supported on two springs, the two springs being the stiffness characteristics of the upper key-way support and the lower support tiedown bracket.

Three different methods of calculating the natural frequency of the RHR heat exchanger could be used. All three methods were used to calculate the natural frequency of a typical system with all results being within 7 percent of each other.

3.9.1.3B Experimental Stress Analysis

The following subsections in this section list the only NSSS components upon which experimental stress analysis was used and provide a discussion of the analysis.

3.9.1.3.1B Experimental Stress Analysis of Piping Components

The following components have been tested to verify their design adequacy:

1. Snubbers

2. Pipe whip restraints

Descriptions of the snubber and whip restraint tests are contained in Sections 3.9.3.4B and 3.6B, respectively.

3.9.1.3.2B Orificed Fuel Support, Vertical and Horizontal Load Tests

The BWR 6 Orificed Fuel Support (OFS) under the provisions of the ASME Code, Section III, Subsection NG is classified as a core support structure and, therefore, must comply with NA-3352.1. In order to meet this requirement, an analysis was performed using the finite element method. However, the complexity of the OFS design as well as the nonlinear behavior of the OFS during analysis preempted the use of finite element analysis. Accordingly, a series of horizontal and vertical load tests were performed in order to conform to the requirements of the code and the design specification. The results of these tests indicate that the component and seismic loading of the OFS are below the stress limit allowables, including a 0.65 quality factor. The allowable stress limits were arrived at by applying a 0.65 quality factor to the ASME Code allowables of 1.5 S_m for the upset and 1.5 x 0.7 S_u for the faulted service conditions.

3.9.1.3.3B Control Rod Drive

Experimental data was used in refining the CRDSSO1 code. The output of CRDSSO1 was used in the dynamic analysis of both code and non-code parts. Pressures used in the analysis were also determined during actual testing of prototype control rod drives.

3.9.1.4B Considerations for the Evaluation of Faulted Conditions

Seismic Category I equipment is evaluated for the faulted loading conditions. The following paragraphs in Subsection 3.9.1.4B show examples of the treatment of faulted conditions for the major components on a component-by-component basis. Additional discussion of faulted analysis is found in Section 3.9.3B, Section 3.9.5B, and Table 3.9B-2.

Sections 3.9.2.2B and 3.7B discuss the treatment of dynamic loads resulting from the postulated SSE. Section 3.9.2.5B discusses the dynamic analysis of loads on NSSS equipment resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas and no cases are identified where design limits, such as clearance limits, are exceeded.

3.9.1.4.1B Control Rod Drives System Components

3.9.1.4.1.1B Control Rod Drives

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The major control rod drive components that have been analyzed for the faulted conditions are: ring flange, main flange, and indicator tube. The maximum stresses for these components and for various plant operating conditions including the faulted condition are given in Table 3.9B-2t.

3.9.1.4.1.2B Hydraulic Control Unit

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The Hydraulic Control Unit (HCU) is tested for the seismic and hydraulic load conditions. Subsection 3.9.2.2.2.2B describes the methods of this analysis.

1←•

3.9.1.4.2B Standard Reactor Internal Components

3.9.1.4.2.1B CR Guide Tube

The maximum calculated stress on the CR guide tube occurs in the base during the faulted condition. The "faulted" limit is 2.4 S_m where S_m is 16,000 psi at 575°F. The analysis and the results for various plant operating conditions are summarized in Table 3.9B-2x.

14←•

3.9.1.4.2.2B Incore Housing

The maximum calculated stress on the incore housing occurs at the outer surface of the vessel penetration during the faulted condition. The "faulted" limit is the lesser of 2.4 S_m or 0.7 S_u. Stresses are summarized in Table 3.9B-2y.

3.9.1.4.2.3B Jet Pump

The dynamic analysis of the jet pump under faulted load conditions shows that the maximum stress occurs at the jet pump riser brace. The maximum allowable for this condition per ASME Code, Section III, Subsection NG, is 3.6 S_m. Stresses are summarized in Table 3.9B-2v.

3.9.1.4.2.4B LPCI Coupling

The location of the highest primary stress (P_m + P_b) is at the strut to shroud attachment weld. The smallest margin at the weld is from the faulted condition, resulting from a large line break plus SSE. The allowable stress is 2.4 x 1.5 x 0.7 S_m. A weld quality factor of 0.7 is included as required by ASME Code, Section III,

Table NG-3352-1. The analysis and results are summarized in Table 3.9B-2w.

3.9.1.4.2.5B Orificed Fuel Support

See Subsection 3.9.1.3.2B, "Orificed Fuel Support, Vertical and Horizontal Load Tests."

3.9.1.4.2.6B CRD Housing

The CRD housing is analyzed for the faulted condition. The SSE and hydrodynamic loads are considered. Table 3.9B-2u shows that the maximum calculated stresses are bounded by the allowables.

3.9.1.4.3B Reactor Pressure Vessel Assembly

The RPV assembly includes the RPV, support skirt, and shroud support. For the faulted condition, the RPV was evaluated using an elastic analysis. The support skirt and shroud support were also evaluated with an elastic analysis; accounting for the compressive buckling load. The analysis and results are summarized in Table 3.9B-2a.

3.9.1.4.4B Core Support Structures

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The core support structures are evaluated for the faulted load condition on the basis of seismic and other dynamic events described in Sections 3.7B and 3.9.5B, respectively. The calculated stresses and allowables for various plant operating conditions are summarized in Table 3.9B-2b.

14←•

3.9.1.4.5B Main Steam Isolation, Recirculation Gate and Safety/Relief Valves

Tables 3.9B-2g, 3.9B-2h, and 3.9B-2j provides a summary of the method of analyses of the safety/relief, main steam isolation, and recirculation gate valve, respectively.

Standard design rules, as defined in ASME III, are utilized in the analysis of pressure boundary components of Seismic Category I valves. Conventional, elastic stress analysis is used to evaluate components not defined in the ASME Code. The code allowable stresses are applied to determine acceptability of structure under applicable loading conditions including faulted condition.

3.9.1.4.6B Recirculation System Flow Control Valve

The recirculation system flow control valve is analyzed for faulted conditions using the elastic analysis criteria from the ASME Code, Section III. The results are summarized in Table 3.9B-2f.

3.9.1.4.7B Main Steam and Recirculation Piping

For main steam and recirculation system piping, elastic analysis methods are used for evaluating faulted loading conditions. The equivalent allowable stresses using elastic techniques are obtained from ASME Code, Section III, Appendix F, "Rules for Evaluation of Faulted Conditions," and these are above elastic limits. Additional information on the main steam and recirculation piping is in Tables 3.9B-2d and 3.9B-2e.

3.9.1.4.8B Nuclear Steam Supply System Pumps, Heat Exchanger, and Turbine

The recirculation (Table 3.9B-2i), ECCS (Table 3.9B-2n), RCIC (Table 3.9B-2r), and SLC pumps (Table 3.9B-2l), RHR heat exchangers (Table 3.9B-2o), and RCIC turbine (Table 3.9B-2q) have been analyzed for the faulted loading conditions identified in Section 3.9.1.1B. In all cases, stresses were within the elastic limits. The analytical methods, stress limits, and allowable stresses are discussed in Sections 3.9.2.2B and 3.9.3.1B.

3.9.1.4.9B Control Rod Drive Housing Supports

Examples of the calculated stresses and the allowable stress limits for faulted conditions for the control rod drive housing supports are shown in Table 3.9B-2z.

3.9.1.4.10B Fuel Storage Racks

Examples of the calculated stresses, and stress limits for the faulted conditions for the new fuel storage racks, fuel preparation machine, refueling platform, and fuel transfer tube are shown in Table 3.9B-2s.

3.9.1.4.11B Fuel Assembly (Including Channels)

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BWR fuel assembly design bases, analytical methods, and evaluation results, including those applicable to the faulted conditions, are contained in References 4 and 5 for GE fuel. The acceleration profiles are summarized in Table 3.9B-2aa for GE fuel.

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3.9.1.4.12B Reactor Refueling and Servicing Equipment

Refueling and servicing equipment which are important to safety are classified as essential components per the requirements of 10CFR50, Appendix A. This equipment and other equipment whose failure would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra and combines loads at frequencies up to 33 Hz for seismic and 60 Hz for hydrodynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC, allowables.

3.9.1.5B Reactor Asymmetric Loads Analysis Methodology

The following is a description of the methodology used in the reactor asymmetric loads analysis.

1. Pressure-Time Histories

   The pressure time histories in the annulus region between the RPV and shield wall are generated from a feedwater line break and a recirculation line break. These analyses are described in Section 6.2.1.2.

2. Concentrated Force-Time Histories

   The forcing function of jet impingement on the shield wall is obtained from the break flow transient caused by a feedwater line break and a recirculation line break. Forcing functions of jet reaction on RPV, jet impingement on RPV, and pipe whip restraint load on restraint anchors are obtained from the feedwater line break, the recirculation line break, and main steam line break.

3. Integrated Dynamic Analysis

   Beam and shell models are used to integrate pressure-time histories and concentrated force-time histories in determining the effects on the shield wall pedestal, vessel support, core support and internals, and control rod drives. These dynamic analyses yield displacements, forces, stresses and moments.

4. Attached Piping Analysis

Acceleration time history from the integrated dynamic analysis is used to generate response spectra for the stress analysis of the attached piping. This analysis covers ECCS lines, primary coolant piping, and associated pipe supports.

5. Load Combination for Vessel and Piping

Asymmetric LOCA loads in combination with SSE by the SRSS methodology are treated as a faulted condition for evaluation against the ASME Code and functional capability requirements. This is described in Table 3.9B-2 (for NSSS components).

For attached ECCS piping, the ARS that is generated for the annulus pressurization (AP) case is used in the pipe stress analysis in the same manner as it is used for the seismic SSE case, i.e., considered as faulted condition with the critical damping values of 2 and 3 percent according to NRC Regulatory Guide 1.61, and with the combination of modes and spatial components according to NRC Regulatory Guide 1.92.

The results of the pipe stress analysis due to AP are combined by the SRSS method with those due to SSE and flow transients and are used in the evaluation of Equation 9 with respect to both the ASME code allowables and functional capability requirements.

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Pipe loads exerted on the ECCS pipe supports as a result of annulus pressurization are combined with SSE by the SRSS method, under the faulted condition.

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Piping analysis methods and load combinations are discussed in Section 3.9A. Stress results for the annulus pressurization case are shown in Tables 3.9B-12 through 3.9B-16 for ECCS piping and in Tables 3.9B-17 through 3.9B-21 for ECCS piping supports.

3.9.2B Dynamic Testing and Analysis

3.9.2.1B Piping Vibration, Thermal Expansion, and Dynamic Effects

The test program is divided into three phases: piping vibration, thermal expansion, and dynamic effects.

3.9.2.1.1B Piping Vibration

3.9.2.1.1.1B Preoperational Vibration Testing of Recirculation Piping

The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation piping are within acceptable limits. This phase of the test uses remote measurements. Remote measurements are made during the following steady-state conditions:

1. Recirculation pumps minimum flow

2. Recirculation pumps at intermediate flow

3. Recirculation pumps at maximum stable preoperational flow

Section 3.9.2.1A further discusses preoperational vibration testing, including measurement locations and visual inspection points.

3.9.2.1.1.2B Startup Vibration Testing of Main Steam and Recirculation Piping

The purpose of this phase of the program is to verify that the main steam and recirculation piping vibration are within acceptable limits. Because of limited access due to high radiation levels, no visual observation is required during this phase of the test. Remote measurements are made during the following steady state conditions:

1. Main steam flow at 25% of rated

2. Main steam flow at 50% of rated

3. Main steam flow at 75% of rated

4. Main steam flow at 100% of rated.

3.9.2.1.1.3B Operating Transient Loads on Main Steam and Recirculation Piping

The purpose of the operating transient test phase is to verify that pipe stresses are within Code limits. The amplitude of displacements and number of cycles per transient of the main steam and recirculation piping are measured and the displacements compared with acceptance criteria. The deflections are correlated with stresses to verify that the pipe stresses remain within Code limits.

Remote vibration and deflection measurements are taken during the following transients:

1. Recirculation pump starts
2. Recirculation pump trip at 100% of rated flow
3. Turbine stop valve closure at 100% power
4. Manual discharge of each S/R valve at 1,000 psig and at planned transient tests that result in S/R valve discharge.

3.9.2.1.2B Thermal Expansion Testing of Main Steam and Recirculation Piping

Thermal expansion preoperational and startup testing program performed through the use of potentiometer sensors has been established to verify that normal thermal movement occurs in the piping systems. The main purpose of this program is to ensure the following:

1. The piping system during system heatup and cooldown is free to expand, contract, and move without unplanned obstruction or restraint in the x, y, and z directions.
2. The piping system does "shakedown" after a few thermal expansion cycles.
3. The piping system is working in a manner consistent with the assumption of the NSSS stress analysis.
4. There is adequate agreement between calculated values of displacements and measured value of displacement.
5. There is consistency and repeatability in thermal displacements during heatup and cooldown of the NSSS systems.

Limits of thermal expansion displacements are established prior to the start of piping testing to which the actual measured displacements are compared to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limits values by more than the specified tolerance, the piping system is responding in a manner consistent with predictions and is therefore acceptable. Two levels of limits of displacements are established to check the systems as explained in Section 3.9.2.1.4B.

3.9.2.1.3B Dynamic Effects Testing of Main Steam and Recirculation Piping

To verify that snubbers are adequately performing their intended function during plant operation, a program for dynamic testing as a part of the normal startup operation testing is planned. The main purpose of this program is to ensure the following:

1. The vibration levels from the various dynamic loadings during transient and steady-state conditions are below the predetermined acceptable limits.

2. Long-term fatigue failure does not occur due to underestimating the dynamic effects caused by cyclic loading during plant transient operations.

This dynamic testing is to account for the acoustic wave due to the safety/relief valve lifts,(RV1), safety/relief valve load resulting from air clearing (RV2), and turbine stop valve closure load (TSVC). The maximum stresses developed in the piping by the RV1, RV2, and TSVC transients analysis are used as a basis for establishing criteria which assure proper functioning of the snubbers. If field measurements exceed criteria limits, the snubbers are not operating properly. If field measurements are within criteria limits, the snubbers are functioning properly. Sample production snubbers of each size (i.e., 50 kips, 100 kips) are qualified and tested for design and faulted condition loadings prior to shipment to the field. Snubbers are tested to allow free piping movements at low velocity. During plant startup, the snubbers are checked for improper settings.

The criteria for vibration displacements are based on assumed linear relationship between displacements, snubber loads, and the magnitude of applied loads for any function and response of system. Thus, the magnitude of limits of displacements, snubber loads, and nozzle loads are all proportional. Maximum displacements (Level 1 limits) are established to prevent the maximum stress in the piping systems from exceeding the normal and <u>upset primary stresslimits</u> and/or the maximum snubber load from exceeding the maximum load to which the snubber has been tested.

Based on the above criteria, Level 1 displacement limits are established for all instrumented points in the piping system. These limits are compared with the field measured piping displacements. Method of acceptance is explained in the following subsection.

3.9.2.1.4B Test Evaluation and Acceptance Criteria for Main Steam and Recirculation Piping

The piping response to test conditions is considered acceptable if the test results verify that the piping responded in a manner consistent with the predictions of the stress report and/or that the tests verify that piping stresses are within Code limits (ASME Section III, NB-3600). Acceptable deflection and acceleration limits are determined after the completion of piping systems stress analysis and are provided in the startup test specifications. To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Levels 1 and 2, are described in the following paragraphs.

3.9.2.1.4.1B Level 1 Criteria

Level 1 establishes maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory.

If the Level 1 limit is exceeded, the plant is placed in a satisfactory hold condition, and the responsible piping design engineer is advised. Following resolution, applicable tests must be repeated to verify that requirements of the Level 1 limits are satisfied.

3.9.2.1.4.2B Level 2 Criteria

If the Level 2 criteria are satisfied for both steady state and operating transient vibrations, there is no fatigue damage to the piping system due to steady state vibration, and all operating transient vibrations are bounded by the values in the stress report.

Exceeding the Level 2 specified pipe motion requires that the responsible piping design engineer be advised. Plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Detailed evaluation is needed to develop corrective action or to show that the measurements are acceptable. Depending upon the nature of such resolution, the applicable tests may or not have to be repeated.

3.9.2.1.4.3B Acceptance Limits

For steady state vibration, the piping peak stress due to vibration only (neglecting pressure) does not exceed 10,000 psi for Level 1 criteria and 5,000 psi for Level 2 criteria. These limits are below piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of ASME Code for 10^6 cycles.

For operating transient vibration, the piping bending stress (zero to peak) due to operating transient only does not exceed $1.2S_m$ or pipe support loads do not exceed the Service Level D ratings for Level 1 criteria. The $1.2S_m$ limit ensures that the total primary stress including pressure and dead weight does not exceed $1.8S_m$, the new Code Service Level B limit. Level 2 criteria are based on pipe stress and support loads not to exceed design basis predictions. Design basis criteria require that operating transient stresses and loads not exceed any of the Service Level B limits including primary stress limits fatigue usage factors limits and allowable loads on snubbers.

3.9.2.1.5B Corrective Actions for Main Steam and Recirculation Piping

During the course of the tests, the remote measurements are regularly checked to determine compliance with Level 1 criteria. If trends indicate that Level 1 criteria may be violated, the measurements are monitored at more frequent intervals. The test is interrupted as soon as Level 1 criteria are violated. As soon as possible after the test hold or termination, the following corrective actions are taken:

1. Installation Inspection. A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. If vibration exceeds criteria, the source of the excitation is identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

2. Instrumentation Inspection. The instrumentation installation and calibration are checked, and any

discrepancies are corrected. Additional instrumentation is added, if necessary.

3. Repeat Test. If actions 1 and 2 identify discrepancies that could account for failure to meet Level 1 criteria, the test is repeated.

4. Resolution of Findings. If the Level 1 criteria are violated on the repeat test or no relevant discrepancies are identified in 1 and 2, the test results and criteria are reviewed to ensure that the test can be safely continued.

If the test measurements indicate failure to meet Level 2 criteria, the following corrective actions are taken after completion of the test:

1. Installation Inspection. A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. If vibration exceeds limits, the source of the vibration is identified. Action is taken to correct any discrepancies.

2. Instrumentation Inspection. The instrumentation installation and calibration are checked, and any discrepancies are corrected.

3. Repeat Test. If 1 and 2 above identify a malfunction or discrepancy that could account for failure to comply with Level 2 criteria and appropriate corrective action has been taken, the test is repeated.

4. Documentation of Discrepancies. If the test is not repeated, the discrepancies found under actions 1 and 2 are documented in the test evaluation report and correlated with the test condition. The test is not complete until the test results are reconciled with the acceptance criteria.

3.9.2.1.6B Measurement Locations for Main Steam and Recirculation Piping

Remote shock and vibration measurements with transducers are made in the three orthogonal directions near the first downstream S/R valve on each steam line and in the three orthogonal directions on the piping between the recirculation pump discharge and the first downstream valve. During preoperational testing of recirculation piping, visual observation and manual measurements by a hand-held vibrograph are made to supplement the remote measurements.

3.9.2.2B Seismic and Hydrodynamic Qualification of Safety-Related Mechanical Equipment

This subsection describes the dynamic, i.e., seismic, and where applicable, hydrodynamic qualification of safety-related floor-mounted, pipe-mounted, and fuel handling mechanical equipment. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit, for example, ECCS pumps. These modules are generally discussed in this section rather than in Sections 3.10B and 3.11. Operability qualification of active pumps and valves is discussed in Section 3.9.3.2B.

3.9.2.2.1B Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety function during and after the exposure to dynamic loads is demonstrated by tests and/or analysis. Selection of testing, analysis, or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. Where practical, the equipment operability is established by testing. Otherwise, the operation and/or loads are simulated by mathematical analysis and applied in addition to physical tests.

Equipment which is large and/or can be represented by a frame-type structure is usually qualified by analysis to show that the loads, stresses, and deflections are less than the allowable maximum. Analysis and/or testing are used to show that there are no equipment resonances within the frequency range of interest (generally 1 to 33 Hz for equipment subjected to seismic loads only, and 1 to 100 Hz for equipment subjected to seismic and hydrodynamic loads). If equipment resonances are discovered within the applicable frequency range, dynamic tests may be conducted and, in conjunction with mathematical analysis, used to verify operability and structural integrity under the required dynamic loads.

When the equipment is qualified by dynamic test, the response spectrum method is generally used in determining input motion. Testing is performed on prototypes of equipment and is supported by analysis to demonstrate similarity between the prototype and equipment installed at RBS.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Dynamic loads are simulated by testing using random vibration input or single frequency input at frequencies throughout the applicable range.

Whichever method is used, the equipment response during testing envelopes the actual equipment response expected during dynamic loading conditions. The TRS, where applicable, generally envelops the applicable frequency range of the RRS with margins greater than 10 percent. If poke-throughs occur, they are justified on a case-by-case basis.

The equipment being dynamically tested is mounted in a way that simulates the actual mounting and causes no dynamic coupling to the equipment. Equipment mounted on intermediate structures is qualified to the acceleration levels at the mounting location which takes into account the transmissibility of the supporting structure.

3.9.2.2.1.1B Vibration Input

Dynamic tests are generally performed using random multifrequency vibration input. However, single frequency input such as sine waves can be used provided one of the following conditions is met:

1. The device input motion is dominated by one frequency.

2. The device is rigid, or its response is adequately represented by one mode.

3. The device can be characterized as passive, in which case its safety function is satisfied by maintaining structural integrity.

4. The device is tested to a sufficiently high acceleration level to excite all modes over the frequency range of interest.

3.9.2.2.1.2B Application of Input Motion

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion may be applied to one direction at a time.

In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely retilinear resultant input is avoided.

3.9.2.2.1.3B Hydrodynamic Fatigue

A number of NSSS components are mounted inside the River Bend Station reactor building, and are subjected to hydrodynamic loads. One of the possible hydrodynamic loads (i.e., loads due to SRV actuation) is predicted to occur sufficiently often that fatigue effects might possibly develop over the projected 40-yr life of the plant.

In order to assess whether such fatigue effects could significantly degrade the NSSS components to a point where performance of their safety-related functions might be impaired, a representative sample of the NSSS components was evaluated in depth. This sample included a valve, a pump, an electric motor, and a level switch. The evaluation methods used included both test and analysis of sufficient duration to simulate 40 yrs of in-service application.

The tested components were subjected to test response spectra which enveloped the required response spectra due to hydrodynamic loads resulting from SRV actuation. The test durations were of sufficient length to simulate the number of SRV actuations expected during 40 yrs of plant operation. The fatigue testing preceded testing for the five upset events and one faulted event. In all cases, the components were demonstrated to be able to perform their safety function within predetermined acceptance criteria during and after all fatigue, upset, and faulted testing.

Fatigue usage calculations were performed on the analyzed components. Stress reversals of sufficient magnitude and number were selected to simulate the number of SRV actuations expected during 40 yrs of plant operation. In all cases, conservative calculations predicted usage factors considerably less than 1.0 for all critical elements.

Based on the work summarized above, it has been concluded that fatigue due to hydrodynamic loads resulting from SRV actuation does not constitute a safety concern for NSSS equipment at the River Bend Station.

3.9.2.2.2B Dynamic Qualification of Specific NSSS Mechanical Components

The following sections discuss the dynamic qualification of major NSSS components. A listing of Seismic Category I equipment, with the exception of active pumps and valves, is provided in Table 3.9B-11. The operability qualification of pumps and valves is described in Section 3.9.3.2B.

3.9.2.2.2.1B HPCS Diesel Generator

The HPCS diesel generator is qualified by a combination of test and analysis. The testing program consists of two phases. The first phase involves self starting of the diesel engine by using startup procedures deliberately designed to cause maximum engine vibration.

Devices on the engine which experience vibration levels greater than expected under seismic plus normal startup procedures are qualified by this technique.

Active devices not qualified by the first phase are then placed on a shaker table and seismically qualified in the normal way. All essential active devices mounted on the engine are therefore qualified by test.

The analysis program covers all passive components not qualified by the testing approach. Both static and dynamic analyses are performed, depending on whether the equipment is rigid below the seismic ZPA. In addition, the generator is analyzed dynamically since it cannot be qualified by either of the testing approaches. Deflection analyses are also performed on the generator to ensure operability under all postulated conditions.

3.9.2.2.2.2B Hydraulic Control Unit (HCU)

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The HCUs are located in the reactor building and are subjected to both seismic and hydrodynamic loads. A complete HCU assembly was qualified by multiaxis/ multifrequency testing in the frequency range from 1 to 100 Hz. The required safety function, i.e., to deenergize and initiate reactor scram was successfully demonstrated during testing. As the result of fatigue failure resulting from the over-conservatively applied hydrodynamic loads during testing, the HCUs were qualified for a limited life. Analysis of the testing conducted prior to the fatigue failure demonstrates qualification for greater than 40 years for the RBS seismic/dynamic requirements. The license amendment limiting the HCU seismic/dynamic qualification has been removed via Amendment No. 34 to NPF-47.

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3.9.2.2.2.3B Recirculation Pump and Motor Assembly

The recirculation pump and motor assembly is located inside the reactor building and is classified as a passive, safety-related component.

The recirculation pump, including its appurtenances and supports, individually and as an assembly, is designed to withstand accelerations of 4.5 g horizontal and 3.0 g vertical. This compares to the calculated RBS required accelerations of 0.75 g horizontal and 0.5 g vertical. Details of the qualification are as follows:

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1. The flooded pump, motor, and recirculation system piping assembly is analyzed as a system. The system is supported by constant support hangers from the brackets on the motor-mounted stand, with mechanical snubbers attached to brackets on the pump case and the top of the motor frame. Calculations using a finite-element model of the RBS system determined the natural frequencies, mode shapes, and maximum dynamic acceleration responses using the response spectrum method. The maximum acceleration values are less than the RBS design values.

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2. Primary stresses due to horizontal and vertical dynamic forces are considered to act simultaneously and therefore added algebraically. Horizontal and vertical dynamic forces are applied to mass centers, and equilibrium reactions are determined for motor and pump brackets.

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3. Load, shear, and moment diagrams are constructed using design values in excess of calculated live loads, dead loads, and calculated snubber reactions. Combined bending, tension, and shear stresses were determined for each major motor flange bolting and pump case.

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4. The maximum combined tensile stress in the cover bolting is calculated including tensile stress from design pressure.

5. The brackets on the pump case, were designed to withstand loads resulting from the building dynamic response.

3.9.2.2.2.4B ECCS Pump and Motor Assembly

There are five ECCS pumps, one LPCS, one HPCS, and three RHR pumps. All five pumps are located in the auxiliary building and are not subjected to hydrodynamic loads.

A prototype ECCS pump motor has been dynamically qualified via a combination of static analysis and dynamic testing. The motor assembly has been dynamically qualified by multiaxis/multifrequency testing, in accordance with IEEE Standard 344-1975. The qualification test program included demonstration of startup and shutdown capabilities, as well as no load operability during dynamic loading conditions.

For static analysis, the seismic forces of each component or assembly are obtained by concentrating its mass at the center of mass of the component or assembly and multiplying by the seismic acceleration (earthquake coefficient). The magnitude of the RBS specific earthquake coefficients is 0.430 g vertical and 1.106 g horizontal compared to an equipment capability of 2.35 g vertical and 2.1 g horizontal.

The qualification of the pump motor assembly as a unit while operating under faulted conditions was provided by analysis. A three-dimensional finite element model of the pump/motor and its support was developed and dynamically analyzed using the response spectrum analysis method. The results of the analyses demonstrated that the stresses at all critical locations are less than their corresponding allowable values when the pump/motor assembly is subjected to the applicable static and dynamic loads. Pump operability is further established by demonstrating that the calculated critical location displacements are less than the corresponding allowables.

3.9.2.2.2.5B RCIC Pump Assembly

The RCIC pump construction is a barrel-type on a large cross-section pedestal. Qualification was performed by analysis. The RCIC pump is not subjected to hydrodynamic loads. Results are obtained by using acceleration forces acting simultaneously in three directions, one vertical and two horizontal, and calculated using the square root of the sum of the squares method. The pump mass, support system, and accessory piping have been shown, by analysis, to have a natural frequency greater than 33 Hz.

The RCIC pump assembly is analytically qualified by static analysis for seismic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90 percent of the allowable.

3.9.2.2.2.6B RCIC Turbine Assembly

The RCIC turbine is not subjected to hydrodynamic loads and has been seismically qualified via a combination of static analysis and dynamic testing. The turbine assembly consists of rigid masses, wherein static analysis has been utilized, interconnected with control levers and electronic control systems, necessitating final qualification via dynamic testing. Static loading analysis has been employed to verify the structural integrity of the turbine assembly and the adequacy of bolting under operating and seismic loading conditions. The complete turbine assembly has been seismically qualified via dynamic testing, in accordance with IEEE Standard 344-1975 as interpreted by Regulatory Guide 1.100. The qualification test program included demonstration of startup and shutdown capabilities, as well as no load operability during seismic loading conditions.

3.9.2.2.2.7B Standby Liquid Control Pump and Motor Assembly

The SLC positive displacement pump and motor are mounted on a common baseplate in the reactor building. The SLC pump structural integrity and operability is demonstrated by three-dimensional finite element analysis. The analysis demonstrates that the critical location stresses are less than the allowable stress limits.

The structural integrity and operability of the motor is demonstrated by type test in accordance with IEEE Standards 323-1974 and 344-1975. The dynamic test includes vibration aging postulated as the result of hydrodynamic loads.

3.9.2.2.2.8B RHR Heat Exchangers

A three-dimensional finite element model is developed to dynamically analyze the heat exchanger and its supports using the response spectrum analysis method, and to verify that the RHR heat exchangers can withstand seismic loadings. The RHR heat exchangers are located in the auxiliary building and therefore do not experience hydrodynamic loads. The RBS specific response spectra are used in the analysis for seismic loads. The same model is used to statically analyze and evaluate the nozzles due to the effects of the external piping loads and dead weight in order to ensure that nozzle load criteria and limits are met. Critical location stresses are evaluated and compared with the allowable stress criteria. The results of the analysis demonstrate that the stresses at all investigated locations are less than their corresponding allowable values.

The seismic qualification of the RHR heat exchangers meets the requirements of Reg. Guide 1.92 and ASME Section III, Class 2 and 3.

3.9.2.2.2.9B Standby Liquid Control Storage Tank

The standby liquid control storage tank is located inside the reactor building and is subjected to seismic and hydrodynamic loads. The tank is considered a rigid body and is qualified by three-dimensional static analysis. Sloshing of the fluid within the tank is considered in the analysis.

3.9.2.2.2.10B Main Steam Isolation Valves

The main steam isolation valves are qualified for dynamic loads by a combination of test and analysis. The MSIVs are modeled in the RBS main steam piping stress analysis.

Maximum stresses and moments are calculated and compared to allowables to ensure structural integrity of the valve and yoke as a whole. The MSIV actuator, including the barret and valve stem, is dynamically tested to both seismic and hydrodynamic loads using multiaxis/multifrequency inputs. Stroke times are measured before, during, and after the dynamic testing to ensure operability under all dynamic conditions. The MSIV body and externals are not included in this testing since they are not susceptible to externally applied dynamic loads.

3.9.2.2.2.11B Main Steam Safety/Relief Valves

The main steam safety relief valves are qualified for dynamic loads by a combination of test and analysis. The SRVs are modeled in the RBS main steam piping analysis which generates RRS at the inlet flange interface to valve, as well as forces and moments the outlet flange interface. The complete valve/actuator assembly is then dynamically tested for both seismic and hydrodynamic loads using multiaxis/multifrequency inputs. Moments on the outlet flange were simulated during this testing. The SRV is required to operate within its specified limits before, during, and after the dynamic testing.

3.9.2.3B Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady State Conditions

The major reactor internal components within the vessel are subjected to extensive testing coupled with dynamic system analyses to properly describe the resulting flow-induced vibration phenomena incurred from normal reactor operation and anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analysis of the response signals measured for reactor internals of many similar designs are performed to obtain the parameters which determine the amplitudes and modal contributions in the vibration responses. These studies provide useful predictive information for extrapolating the results from tests of components with similar designs to components of different designs. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to the complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

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1. Dynamic analysis of major components and subassemblies is performed to identify vibration modes and frequencies. The analysis models used for Seismic Category I structures are similar to those outlined in Section 3.7.2B, Seismic System Analysis.

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2. Data from previous plant vibration measurements is assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar, but response amplitudes vary among BWRs of differing size and design.

3. Parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates and structural parameters such as natural frequency and significant dimensions.

4. Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function

is obtained for each major component and response mode.

5. Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions, based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of paragraph 1 above.

The dynamic modal analysis also forms the basis for interpretation of the preoperational and initial startup test results (Section 3.9.2.4B). Modal stresses are calculated, and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of `10,000 psi.

The magnitude of the jet reaction loads applied to the reactor internal structures caused by acceleration and deceleration of the flow under normal and upset conditions are negligible compared to the differential pressure loads, and generally need not be considered. Jet reaction loads that require consideration are those associated with the jet pump assembly and riser and within the steam separator itself. The upward jet reaction loads on the separator assembly are cancelled by the downward jet impingement loads at the upper surface of the shroud head dome.

Vibratory loads are continuously applied during normal operation, and the stresses are limited to `10,000 psi to prevent fatigue failure. Prediction of vibration amplitudes, mode shapes, and frequencies for normal reactor operations are based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation.

In order to evaluate the dynamic response of the jet pumps, two locations were chosen for monitoring on jet pumps in the prototype plant. These locations are the riser brace and the diffuser of the jet pump. The reasons for selecting these positions were sensitivity and accessibility. Knowing the strain response at these gage locations, the stresses at other locations can be predicted, as well as the mode of vibration, response frequency, and displacement. These values are compared to analytical criteria, and thus, their acceptability is evaluated.

The load due to cross flow from the jet pumps to the peripheral control rod guide tubes is 620 lb on the bottom 1/8 of the guide tube length, 345 lb on the next higher 1/8 of the guide tube length, and 130 lb on the next 1/4 length of the guide tube.

The stresses produced due to vibratory loads are 375.5 psi and are considered negligible.

The dynamic loads due to flow-induced vibration from the feedwater jet impingement would have no significant effect on the steam separator assembly.

The analysis has shown that the impingement feedwater jet velocity is 12 ft/sec, way below the critical velocity of 118 ft/sec. Also, the analysis has shown that the excitation frequency of the steam separator skirt is 5.1 CPS, and the natural frequency of the skirt is 50 CPS.

The load due to flow-induced vibration has no effect on the LPCI coupling since the calculated natural frequency of the coupling is over 50 Hz.

The calculated stresses due to the hydrodynamic forces during normal operating conditions are negligible compared to the design allowable stresses. Locations for which calculations were made include the weld joints, elbows, and rings.

3.9.2.4B Preoperational Flow-Induced Vibration Testing of Reactor Internals

Vibration measurement and inspection programs are conducted during preoperational and initial startup testing of first-of-a-kind reactor internals configuration*, in accordance with guidelines of Regulatory Guide 1.20 for prototype reactor internals. These programs are conducted

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*NEDE-24057-P (Class II) and NEDO-24057 (Class I), Assess-ment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, November 1977. Also, NEDO-24057-1-P (Amendment No. 1 dated December 1978), NEDE-24057-2-P (Amendment No. 2 dated June 1979), and NEDE-22146 (Clas III), Kuosheng-1 Reactor Internals Vibration Measurements, July 1982.

in the prototype plants in three phases, described as follows:

1. Preoperational Tests Prior to Fuel Loading. Steady-state test conditions include balanced (two-pump) recirculation system operation and unbalanced (single-pump) operation, over the full range of flow rates up to rated flow. Transient flow conditions include single and two-pump trips from rated flow. The specified test duration is to be 35 hr of balanced operation, plus 14 hr of single-pump operation of each recirculation loop, for a total of 63 hr. The major components are subjected to a minimum of 106 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements are obtained during this test, and a close visual inspection of internals is conducted before and after the test.

2. Precritical Testing with Fuel. This vibration measurement series is conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions include balanced, unbalanced, and transient conditions as for the first test series. This test series verifies the anticipated effect of the fuel on the vibration response of internals. Previous vibration measurements in BWRs* have shown that the fuel adds damping and reduces vibration amplitudes of major internal structures. Thus, the first test series (without fuel) is a conservative evaluation of the vibration levels of these structures.

3. Initial Startup Testing. Vibration measurements are made during reactor startup at conditions up to 100 percent rated flow and power. Balanced, unbalanced, and transient conditions of recirculation system operation are evaluated. The primary purpose of this test series is to verify the anticipated effect of two-phase flow on the

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*NEDE-24057-P (Class II) and NEDO-24057 (Class I), Assess-ment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, November 1977. Also, NEDO-24057-1-P (Amendment No. 1 dated December 1978), NEDE-24057-2-P (Amendment No. 2 dated June 1979), and NEDE-22146 (Class III), Kuosheng-1 Reactor Internals Vibration Measurements, July 1982.

vibration response of internals. Previous vibration measurements in BWRs* have shown that the effect of the two-phase flow is to broaden the frequency-response spectrum and diminish the maximum response amplitude of the shroud and core support structures.

Vibration sensor types include strain gauges, displacement sensors (linear variable transformers), and accelerometers. Accelerometers are provided with double integration signal conditioning to give a displacement output. Sensor locations include the following:

Top of shroud head, lateral acceleration (displacement)
Top of shroud, lateral displacement
Jet pump riser braces, bending and extension strains
Jet pump diffuser, bending strain
Control rod drive housings, bending strain
Incore housings, bending strain
Core spray internal piping, bending strain

In addition to the above components, vibration of the core spray sparger is measured during preoperational testing of that system at the designated prototype 218 size BWR/6 plant. In all prototype plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded on magnetic tape, and provision is made for selective online analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer, using frequency, phase, and amplitude information from the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes is then made on the basis of the analytically obtained normal mode which best approximates the observed mode.

The visual inspections conducted prior to and following preoperational testing are for the purpose of detecting evidence of vibration, wear, or loose parts. At the completion of preoperational testing, the reactor vessel

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*NEDE-24057-P (Class II) and NEDO-24057 (Class I), Assess-ment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, November 1977. Also, NEDO-24057-1-P (Amendment No. 1 dated December 1978), NEDE-24057-2-P (Amendment No. 2 dated June 1979), and NEDE-22146 (Clas III), Kuosheng-1 Reactor Internals Vibration Measurements, July 1982.

head and the shroud head are removed, the vessel drained, and major components inspected on a selected basis. The inspections cover the shroud, shroud head, and core support structures, the jet pumps, and the peripheral control rod drive and incore guide tubes. Access is provided to the reactor lower plenum for these inspections. (Reactor internals for the River Bend Station are similar to those of the designated prototype plant, Kuosheng 1. An inspection program is implemented at River Bend Station in accordance with the requirements of Regulatory Guide 1.20, Revision 2, paragraph 3.1.3, for nonprototype, Category I reactor internals. Preoperational tests are conducted at the same steady-state conditions and for the same duration as specified for the prototype plant. The inspection procedure is the same as for the prototype plant.)

Identified GE-supplied NSSS analysis, design, and/or equipment utilized in River Bend Station are in compliance with the intent of Regulatory Guide 1.20 through the incorporation of the following alternate approach.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program meets the requirements of Criterion 1, Quality Standards and Records, of Appendix A to 10CFR50 and Section 50.34, Contents of Applications; Technical Information, of 10CFR50. This Regulatory Guide is applicable to the core support structures and other reactor internals.

Vibration testing of reactor internals is performed on all GE BWR plants. At the time of the original issue of AEC Regulatory Guide 1.20, test programs for compliance were instituted. The first BWR/6 plant of each size is considered a prototype and is instrumented and subjected to preoperational and startup flow testing to demonstrate that flow-induced vibrations similar to those expected during operation cause no damage. Subsequent plants which have internals similar to those of the prototypes are also tested in compliance with the requirements of Regulatory Guide 1.20.

General Electric confirms satisfactory vibration performance of internals in these plants through preoperational flow testing followed by inspection for evidence of excessive vibration. Extensive vibration measurements in prototype plants, together with satisfactory operating experience in 11 BWR/4 plants, establish the adequacy of BWR/6 reactor internal designs. General Electric is continuing these test programs for the GESSAR plants to verify structural integrity and to establish the margin of safety.

3.9.2.5B Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to ensure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces (Fig. 3.9B-5a and 3.9B-5b), a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a comprehensive dynamic model of the RPV and internals with 12 degrees-of-freedom. Only motion in the vertical direction is considered here; hence, each structural member (between two mass points) only has an axial load. Besides the real masses of the RPV and core support structures, account is made for the water inside the RPV.

Typical curves of the variation of pressures during a steam line break are shown in Fig. 3.9B-5a and 3.9B-5b. The accident analysis method is described in Section 3.9.5.2B.

The time varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis and is detailed in Section 3.7.2.1B. The dynamic components of forces from these loads are combined with dynamic force components from other dynamic loads (including seismic), all acting in the same direction, by the Square Root of the Sum of the Squares (SRSS) method. This resultant force is then combined with other steady-state and static loads on an absolute sum basis to determine the design load in a given direction.

The loads and load combinations acting upon the jet pumps and LPCI coupling are listed in Paragraph 3.9.3.1B.

3.9.2.6B Correlations of Reactor Internals Vibration Tests With the Analytical Results

Prior to initiation of the instrumented vibration measurement program for the prototype plant*, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the

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*Kuosheng 1 is expected to be the 218 BWR/6 prototype plant.

allowable vibration levels during the vibration test. The vibration data obtained during the test are analyzed in detail. The results of the data analysis, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3B ASME Section III, Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1B Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic events for the design of safety-related ASME Code components (except containment components) which are discussed in Section 3.8.

This section also lists the major ASME Section III, Class 1, 2, and 3 pressure parts and associated equipment on a component-by-component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Design transients for ASME Section III, Class 2 equipment are not addressed in this section. They are covered in Section 3.9.1.1B, Design Transients. Seismic-related loads are discussed in Sections 3.9.2.2B, Seismic Qualification Testing of Safety-Related Mechanical Equipment, and 3.7B, Seismic Design.

Table 3.9B-2 is the major part of this section; it presents the loading combination, analytical methods (by reference or example), and also the calculated stress or other design values for the most critical areas in the design of each component. These values are also compared to applicable Code allowables.

3.9.3.1.1B Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in the ASME Code, Section III.

3.9.3.1.1.1B Normal Condition

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with condenser available), and system shutdown other than upset, emergency, faulted, or testing.

3.9.3.1.1.2B Upset Condition

These are any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. An operating basis earthquake is included in the upset condition as shown in Table 3.9B-4. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.1.3B Emergency Condition

These are deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the RCPB. The conditions have a low probability of occurrence, but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, transients caused by one of the following: a multiple valve blowdown of the reactor vessel; loss of reactor coolant from a small break or crack which does not depressurize the reactor system nor result in leakage beyond normal makeup system capacity, but which requires the safety functions of isolation of containment and reactor shutdown; improper assembly of the core during refueling.

3.9.3.1.1.4B Faulted Condition

These are combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated events are the most drastic that must be designed against and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: a control rod drop accident, a fuel handling accident, a main steam line break, a recirculation loop break, the combination of any pipe break plus the seismic motion associated with a safe shutdown earthquake plus a loss of offsite power, or the safe shutdown earthquake.

3.9.3.1.1.5B Correlation of Plant Conditions with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation can be used to identify the appropriate plant condition for any hypothesized event or sequence of events.

Plant Conditions	Event Encounter Probability Per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	$1.0 > P > 10^{-2}$
Emergency (low probability)	$10^{-2} > P > 10^{-4}$
Faulted (extremely low probability)	$10^{-4} > P > 10^{-6}$

3.9.3.1.1.6B Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment is capable of accomplishing its safety functions as required by the event and incurs no permanent changes that adversely affect its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment is capable of accomplishing its safety functions as required by the event, but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

3.9.3.1.1.7B Compliance with Regulatory Guide 1.48

GE-supplied NSSS analysis, design, and/or equipment utilized in this facility is in compliance with the intent of

Regulatory Guide 1.48 through the incorporation of the alternate approach cited in Table 3.9B-4.

Regulatory Guide 1.48 delineates acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of the Seismic Category I fluid system components. Compliance with this guide is shown in Table 3.9B-4.

3.9.3.1.2B Reactor Pressure Vessel Assembly

The reactor vessel assembly consists of the reactor pressure vessel and shroud support.

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The reactor pressure vessel and shroud support are constructed in accordance with Section III of the ASME Code. The shroud support was constructed to the requirements of the April 1973 draft of Subsection NG. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The reactor pressure vessel is an ASME Class I component. Complete stress reports on these components have been prepared in accordance with ASME requirements. Table 3.9B-2a summarizes the loading combinations for each category of plant conditions. The stress analysis performed on the reactor vessel, including the faulted conditions, was completed using elastic methods or simplified elastic-plastic analysis of ASME code Section III, Paragraph NB-3228. The shroud support was also evaluated using elastic conditions, except as noted in Subsection 3.9.1.4.3B. Load combinations and stress analyses for other reactor internals are discussed in Subsection 3.9.5B.

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3.9.3.1.3B Main Steam Piping

The main steam piping discussed in this paragraph includes that piping extending from the reactor pressure vessel to the outboard main steam isolation valve. This piping is designed in accordance with the ASME Code, Section III, Subsection NB-3600. The load combinations and stress criteria are shown in Table 3.9B-2d.

The rules contained in Appendix F of ASME Section III are used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with F-1360.

3.9.3.1.4B Recirculation Loop Piping

The recirculation system piping which is bounded by the reactor pressure vessel nozzles is designed in accordance with the ASME Code, Section III, Subsection NB-3600. The load combinations and allowables are shown in Table 3.9B-2e. The rules contained in Appendix F of ASME Section III are used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with F-1360.

3.9.3.1.5B Recirculation System Valves

The recirculation system flow control and suction and discharge gate valves are designed in accordance with the ASME Code, Section III, Class I, Subsection NB, paragraph 3500. These valves are not required to operate under the safe shutdown earthquake. Loading combinations and other stress analysis information are presented in Table 3.9B-2, parts f and j.

3.9.3.1.6B Recirculation Pump

The recirculation pumps are designed in accordance with the ASME Code, Section III. These pumps are not required to operate during the safe shutdown earthquake. The loading combinations and other stress analysis information are presented in Table 3.9B-2i.

3.9.3.1.7B Standby Liquid Control (SLC) Tank

The loads considered in the design of the SLC tank and the categorization of these loads is listed as follows:

1. Pressure (atmospheric) - Normal/Upset
2. Temperature (200°F) - Normal/Upset
3. OBE (2/3 SSE) - Upset
4. Piping Nozzle Loads - Upset
5. SSE - Faulted

The ASME Code allowable stress limits for the normal and upset category are (1.0S) for general membrane and (1.5S) for bending plus local membrane.

The ASME Code allowable stress limits for the faulted category are (1.2S) for general membrane and (1.8S) for bending plus local membrane.

A summary of the design calculations and methods used is shown in Table 3.9B-2m.

3.9.3.1.8B Residual Heat Removal Heat Exchangers

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The heat exchangers are designed in accordance with the ASME Code, Section III. The stress analysis methods, calculated and actual limits for the RHR heat exchangers are shown in Table 3.9B-2o. Heat exchanger design is also discussed in Section 3.9.2.2.2.8B.

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3.9.3.1.9B RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine is designed and fabricated following the basic guidelines for an ASME Code, Section III, Class 2 component.

Operating conditions for the RCIC turbine include:

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1. Surveillance testing - Quarterly Operation in accordance with technical specifications.

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2. Auto-Startup - 30 cycles per yr with reactor pressure at 1,150 psia, nominal, and saturated temperature, turbine exhaust pressure at 25 psia, peak, and saturated temperature.

Design conditions for the RCIC turbine include:

1. Turbine Inlet - 1,250 psig at saturated temperature

2. Turbine Exhaust - 165 psig at saturated temperature

3. Upset conditions, which control the turbine design, include:

   Design pressure
   Design temperature
   Operating basis earthquake
   Inlet and exhaust piping nozzle loads

   Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane and (1.5S) for bending plus local membrane.

4. Faulted, or emergency conditions include:

   Design pressure
   Design temperature
   Safe shutdown earthquake
   Inlet and exhaust piping nozzle loads

   Stress limits for pressure boundary are 120 percent of ASME Code allowable stress (1.2S) for general membrane and (1.8S) for bending plus local membrane.

5. Nozzle loading definition includes:

   Upset - Inlet F = (3500-M)/3
   Exhaust F = (7000-M)/3

   Faulted (or Emergency) - Inlet F = (4200-M)/3
   Exhaust F = (8400-M)/3

   Where F (lb) and M (ft-lb) are the resultant force and moment on the respective nozzle.

Table 3.9B-2q contains a summary of the RCIC turbine components calculated and allowable loads.

3.9.3.1.10B RCIC Pump

The RCIC pump is designed and fabricated to the requirements for an ASME Code Class 2 component.

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Operating conditions for the RCIC pump are tested under surveillance together with the RCIC turbine. An operation test is performed where the RCIC pump takes condensate from the aboveground storage tank and at design flow discharges condensate back to the aboveground storage tank via a closed test loop.

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Design conditions for the RCIC pump include:

1. Available NPSH - 21 feet

2. Total head - High speed - 2,980 ft
   Low speed - 610 ft

3. Constant flow rate - 625 gpm

4. Normal ambient operating temperature - 60°F to 122°F

5. Normal plus upset conditions which control the pump design include:

Design Pressure	- 1,525 psig
Design Temperature	- 40°F - 140°F
Operating Basis Earthquake	- 2/3 of SSE
Suction Nozzle Loads	- Fo = 1,940 lb Mo = 2,460 ft-lb
Discharge Nozzle Loads	- Fo = 3,715 lb Mo = 4,330 ft-lb

Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane and (1.5S) for bending plus local membrane.

6. Faulted or Emergency conditions include:

Design Pressure	- 1,525 psig
Design Temperature	- 40°F - 140°F
Safe Shutdown Earthquake:	- Horizontal - 1.5g Vertical - 1.5g
Suction Nozzle Loads	- Fo = 2,325 lb Mo = 2,950 ft-lb
Discharge Nozzle Loads	- Fo - 4,450 lb Mo = 5,200 ft-lb

Stress limits for pressure boundary are 120 percent of ASME Code allowable stress (1.2S) for general membrane and (1.8S) for bending plus local membrane.

7. Nozzle Loading:
Pump nozzles are subject to loading from the connecting pipe. The nozzle pipe reactions to the allowable forces and moments on the equipment is expressed as:

$$\left|\frac{Fi}{Fo}\right| + \left|\frac{Mi}{Mo}\right| \qquad \begin{array}{l} Fi = Fx = Fy = Fz \\ Mi = Mx = My = Mz \end{array}$$

Fo = The allowable value of Fi when all moments are zero.

Mo = The allowable value of Mi when all forces are zero. Therefore, the equipment shall be designed to be capable of:

a. Withstanding the three external orthogonal forces, all equal to Fo with no moments.

b. Withstanding the three external orthogonal moments, all equal to Mo with no forces.

Table 3.9B-2r contains a summary of the design calculations for the RCIC pump components.

3.9.3.1.11B ECCS Pumps

Design conditions for RHR, LPCS, and HPCS pumps are as follows:

	RHR	LPCS	HPCS
Design Pressure			
Suction	215 psig	115 psig	115 psig
Discharge	500 psig	600 psig	1,575 psig
Design Temperature	40-360°F	40-212°F	40-212°F

1. Normal plus upset condition:

   Design pressures are tabulated above. The operating basis earthquake seismic accelerations are 0.5 g (shaft, column, bowl, and suction barrel) or 1.5 g (discharge head) horizontal and 0.35 g vertical. Stress limits for pressure boundary are Code allowable stress (1.0S) for general membrane and (1.5S) for bending plus local membrane.

2. Faulted or emergency condition:

   Design pressures are tabulated above. The safe shutdown earthquake seismic accelerations are 0.5 g (shaft, column, bowl, and suction barrel) or 1.5 g (discharge head) horizontal and 0.7 g vertical. Stress limits for the pressure boundary are 120 percent of ASME Code allowable stress (1.2S) for general membrane and (1.8S) for bending plus local membrane.

The RHR, LPCS, and HPCS pumps are designed and fabricated in accordance with the requirements of ASME Section III.

Table 3.9B-2n summarizes the load criteria and design calculations for the ECCS pumps.

3.9.3.1.12B Standby Liquid Control Pump

The standby liquid control pump is designed and fabricated following the requirements for an ASME Code, Class 2 component.

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Operating conditions for the SLC pump and motor are functionally tested by pumping demineralized water through a closed test loop. The SLC pump is capable of injecting the net contents of the storage tank, with a sodium pentaborate enrichment of 80 atom percent boron-10, in not less then 35 minutes and not more than 125 minutes. The pump is capable of injecting flow into the reactor against zero psig up to the initial set point of the reactor relief valves.
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Design conditions for the SLC pump include:

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1. Flow rate — 43 gpm

2. Available NPSH, maximum — 12.9 psi

3. Maximum operating discharge pressure — 1,250 psig

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4. Ambient conditions:

Temperature	70°F - 122°F
Relative humidity	20% - 95%

5. Normal plus upset conditions which control the pump design include:

Design pressure	1,400 psig
Design temperature	150°F
Operating basis earthquake	2/3 of SSE
Suction nozzle loads	Fo = 770 lb Mo = 490 ft-lb
Discharge nozzle loads	Fo = 370 lb Mo = 110 ft-lb

Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane.

6. Faulted or emergency conditions include:

Design pressure	1,400 psig
Design temperature	150°F
Safe shutdown earthquake	horizontal 1.75g vertical 1.75g
Suction nozzle loads	Fo = 920 lb Mo = 590 ft-lb
Discharge nozzle	Fo = 440 lb Mo = 130 ft-lb

Stress limits for pressure boundary are 120 percent of ASME Code allowable stress (1.2S) for general membrane and (1.8S) for bending plus local membrane.

7. Nozzle loading:

Pump nozzles are subject to loading from the connecting pipe. The nozzle pipe reactions to the allowable forces and moments on the equipment is expressed as:

$$\left| \frac{Fi}{Fo} \right| + \left| \frac{Mi}{Mo} \right| \leq 1$$

where:

Fi = The largest absolute value of the three actual external orthogonal forces (Fx, Fy, Fz) that may be imposed by the pipe.

Mi = The largest absolute value of the three actual external orthogonal moments (Mx, My, Mz) permitted from the pipe when they are combined simultaneously for a spcific condition.

Fo = The allowable value of Fi when all moments are zero.

Mo = The allowable value of Mi when all forces are zero.

A summary of the design calculations for the standby liquid control pump components is contained in Table 3.9B-2L.

3.9.3.1.13B Main Steam Isolation and Safety/Relief Valves

Load combination, analytical methods, calculated stresses, and allowable limits are shown for the safety/relief and main steam isolation valves in Tables 3.9B-2g and 3.9B-2h, respectively.

3.9.3.1.14B Reactor Water Cleanup (RWCU) System Pump

The RWCU pump is not part of a safety system and is not designed to Seismic Category I requirements.

The static analysis considers static equilibrium forces on the equipment, including the effect of OBE loads. This analysis considers piping loads as well as torsional moment produced by the rotating assembly. No dynamic analysis is performed.

No experimental or inelastic stress analysis was used in the pump design.

The design loading combinations and limits for the pump include the following:

1. Normal plus upset loads: This includes the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads, dead weight loads including seismic due to operational basis earthquake (OBE) loads, plus torsional loads due to rotation of the component assembly.

2. Seismic loading: This equipment and supports are designed to withstand the OBE loads applied at the mass center, assuming that the pump is flooded.

3. Stresses in the supports and the anchor bolts due to OBE loads are combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the applicable Codes.

4. The ASME Code, Section III, is used as a guide in calculating the thickness of the pressure-retaining parts and for sizing the cover bolting.

5. Identified thermal transients: Equipment operates between 70°F - 545°F. Transient analysis is not required for Class III components in this temperature range.

Table 3.9B-2p shows the calculated stress values and allowable stress limits for the pump.

3.9.3.1.15B Reactor Water Cleanup System (RWCU) Heat Exchangers

The RWCU regenerative and nonregenerative heat exchangers are not part of a safety system and are not designed to Seismic I requirements. However, a static seismic analysis was done on these heat exchangers. Static seismic forces of 0.2g horizontal and 0.0g vertical were used in this analysis.

No experimental or inelastic stress analysis was used in the design of these heat exchangers.

The loading considered in the design of the heat exchangers includes:

1. Normal plus upset loads: This includes the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads, and dead weight loads.

2. Seismic loading: This equipment and supports are designed to withstand the static seismic forces applied.

3. Stresses in the supports and the anchor bolts due to seismic loads are combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the applicable Codes.

4. The allowable shear on anchor bolts set in concrete is in accordance with Table No. 26-1 of the Uniform Building Code.

Table 3.9B-2c shows the calculated stress values and allowable stress limits for the heat exchangers.

3.9.3.1.16B Bolting Stress Limits

3.9.3.1.16.1B Floor Mounted Equipment

1. Equipment Anchorage Bolting

   The floor anchored mechanical equipment (pumps, heat exchangers, and RCIC turbine) in the NSSS scope of supply are mounted on a concrete floor or a steel structure. The design of concrete anchor bolts and the responsibility to prescribe and meet the necessary codes and stress limits are in the AE's scope of supply. The design of attachment bolts for the equipment mounted on steel structure, and the responsibility to prescribe and meet the necessary codes and stress limits, are also in the AE's scope of supply. GE works with the interface limit of 10,000 psi in tension or shear for sizing bolt holes in the equipment base, based on the required nominal size and number of bolts for maximum loads.

2. Component Support Bolting

a. RWCU Pump

The support bolting of this pump which is not essential to safety is designed for the effects of pipe load and SSE load to the requirements of the ASME Code, Section III, Appendix XVII. The stress limits of $0.41S_y$ for tension and $0.15S_y$ for shear are used.

b. RCIC/SLC Pumps and RCIC Turbine

The equipment-to-base plate bolting satisfies the following design criteria:

For normal and upset condition, 1.0S is used for primary membrane and 1.5S for primary membrane plus bending, where S is the allowable stress limit from the ASME Code, Section III, Appendix I, Table I-7.3. For emergency and faulted conditions, the stresses shall be less than 1.2 times the allowable limits for normal and upset given above.

3.9.3.1.16.2B Piping Supports and Pipe Mounted Equipment (Valves and Pump) Supports

The supports are hanger and snubber type (including clamps) linear standard components as defined by the ASME Code Section III, the Subsection NF. The bolts used in these supports meet criteria of NF-3280 for Service Levels A and B and NF-3230 for Service Levels C and D.

For Service Levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum allowable stresses at temperature.

3.9.3.2B Pump and Valve Operability Assurance

Safety-related pumps and valves must perform a mechanical motion during the course of accomplishing a safety function.

Operability is ensured by a comprehensive program of testing and analysis. Testing includes 1) shop tests such as hydrostatic tests and performance tests, 2) preoperational tests to ensure proper installation and interfaces, 3) startup tests to verify that the active pumps and valves perform within their specified limits under a variety of normal and abnormal conditions, and 4) inservice tests to ensure continued operation within specified limits during the life of the plant. In addition, dynamic and environmental testing is performed as discussed in Section 3.9.2.2B and for equipment requiring qualification in accordance with 10CFR50.49 in Section 3.11 and the RBS Environmental Qualification Document.

The active pumps and valves are listed in Tables 3.9B-3a and 3.9B-3b, respectively. Active pumps and valves that are part of the Division 3 (HPCS) diesel generator are not identified separately since they are qualified as part of the diesel generator assembly.

3.9.3.2.1B ECCS Pumps

All active pumps are qualified for operability by first being subjected to rigid tests before and after installation in the plant and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 125 percent of the design pressure, (2) seal leakage tests, and (3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, and Net Positive Suction Head (NPSH) requirements. Also monitored during these operating tests are bearing temperatures (except water-cooled bearings) and vibration levels. Both are shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic in-service inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

The design features of the ECCS pumps, particularly the sizing of the pump internal passages are such that particulates that might pass through suction side strainers will not affect pump operability following a LOCA when debris may be present in the suppression pool.

3.9.3.2.1.1B Analysis of Loading, Stress, and Acceleration Conditions

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the material elastic limit, as indicated in Section 3.9.3.1B and Table 3.9B-2. A three-dimensional finite element model of the pump/motor and its supports is developed using the response spectrum method of dynamic analysis. The same model is analyzed for static nozzle loads, pump thrust loads, and dead weight. Critical displacements and stresses are evaluated and compared with the allowable criteria. The average membrane stress (S_m) for the faulted condition loads is maintained at 1.2S, or approximately 0.75 S (S_y = yield stress) and is the maximum stress in local fibers (S_m + bending stress S_b) is limited to 1.8S, or approximately 1.1 S. The maximum dynamic nozzle loads are also considered in an analysis of the pump supports to ensure that a system misalignment does not occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9B-2 as allowables ensures that critical parts of the pump are not damaged or excessively displaced during the faulted event and, therefore, the reliability of the pump for post-faulted condition operation will not be impaired.

A dynamic analysis is performed to determine the seismic load from the applicable floor response spectra. This analysis demonstrates that faulted condition nozzle loads and seismic accelerations do not impair the operability of the pumps during or following the faulted event.

Components of the pump having a natural frequency above 33 Hz, are essentially rigid. This frequency is sufficiently high to avoid problems with amplification between the component and structure for all seismic loads. For components with a natural frequency below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis.

3.9.3.2.1.2B Pump Operation During and Following the Faulted Loading Condition

Active pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor and the nature of the random, short duration loading characteristics of the dynamic event will prevent the rotor from becoming seized. In actuality, the dynamic loadings cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump continues to operate at the design speed while subjected to the faulted loads.

The functional ability of the active pumps after a faulted condition is assured since only normal operating loads and steady state nozzle loads exist. For the active pumps, the faulted condition is greater than the normal condition only due to seismic SSE loads on the equipment itself. The SSE event is infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps are not damaged during the faulted event, the post-faulted condition operating loads are no worse than the normal plant operating limits. This is ensured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.2B SLC Pump and Motor Assembly and RCIC Pump Assembly

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The SLC pump and motor are dynamically qualified as described in Section 3.9.2.2.2.7B. In addition, an analysis is performed to evaluate the interaction between the SLC pump and motor. The extent of this interaction is determined by assuming the SLC pump shaft and motor shaft are disconnected, and then calculating the relative displacement of the shaft centerlines when subjected to dynamic and static loads, thermal displacement, and initial misalignment. The results show that the displacements due to both continuous and intermittent loads between the two centerlines of the pump and motor shafts do not exceed the allowable limit for continuous loading of the coupling hardware used to join the shafts.

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The function of the RCIC pump/turbine is to provide makeup water to the reactor vessel in the event the vessel becomes isolated which does not result in environmental or other challenges significantly different than those encountered during normal plant operation. The RCIC pump is rigid below the seismic ZPA and is seismically qualified by analysis using a three dimensional finite element model as discussed in Section 3.9.2.2.2.5B. Small piping providing cooling water to the mechanical seals is analyzed using static coefficient analysis. A deflection analysis of the shaft is performed to ensure that minimum clearances are maintained under combined seismic and radial hydraulic thrust loads at the impellers. The RCIC turbine was dynamically qualified by type test as discussed in Section 3.9.2.2.2.6B.

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3.9.3.2.3B ECCS Motors

Qualification of the Class 1E motors used for the ECCS motors is in compliance with IEEE 323-74. The qualification of all motor sizes is based on completion of a type test, followed up with review and comparison of design and material details and seismic analysis of production units, ranging from 600 to 3,500 Bhp, with the motor used in the type test. All manufacturing, inspection, and routine tests by the motor manufacturer on production units are performed on the test motor.

The type test has been performed on a 1,250 hp vertical motor in accordance with IEEE 323-74, first simulating normal operation during the design life, then the motor being subjected to a number of seismic events, and then to the abnormal environmental condition possible during and after a loss-of-coolant-accident (LOCA). The test plan for the type test was as follows:

1. Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on extrapolation in accordance with the temperature life characteristic curve from IEEE 275-66 for the insulation type used on the ECCS motors. The amount of aging equaled the total estimated operation days at maximum insulation surface temperature.

2. Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

3. The normal induced current vibration effect on the insulation system has been simulated by 1.5 g's horizontal vibration acceleration at current frequency for a 1-hour duration.

4. Motor bearings are selected and their operating life is established based on bearing manufacturer's test and operating data using the calculated bearing loads.

5. The dynamic load deflection analysis on the rotor shaft, performed to ensure adequate rotation clearance, has been verified by static loading and deflection of the rotor for the type test motor.

6. Dynamic aging and testing has been performed on a biaxial test table in accordance with IEEE 344-75.

During this type test, the shake table was activated simulating the maximum design limit of the safe shutdown earthquake with motor starts and operation combination as may possibly occur during a plant life.

7. An environmental test simulating a 100-day LOCA condition has been performed with the test motor fully loaded, simulating pump operation. The test consisted of startup and 6 hours operation at 212°F ambient temperature and 100 percent steam environment. Another startup and operation of the test motor after 1-hour standstill in the same environment was followed by sufficient operation at high humidity and temperature, based on extrapolation in accordance with the temperature life characteristic curve from IEEE 275-1966 for the insulation type used on the ECCS motors.

3.9.3.2.4B NSSS Valves

3.9.3.2.4.1B Class 1 Active Valves

The Class 1 active valves are the main steam isolation valves, safety/relief valves, standby liquid control valves, and the high-pressure core spray injection valve. Each of these valves is designed to perform its mechanical motion in conjunction with a design basis accident. Seismic qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.4.1.1B Main Steam Isolation Valve

The MSIV is mathematically modeled in the main steam line system analysis to ensure that design limits are not exceeded for both piping input loads and actuator dynamic loads. The valve's actual input loads, amplified accelerations, and resonance frequencies are determined based on site excitation input to the system as a part of the overall steamline analysis. Pipe anchors and restraints are applied as required to limit pipe system resonance frequencies and amplified accelerations to within acceptable limits for the MSIVs.

The MSIV actuator is qualified for dynamic requirements by multiaxis/multifrequency testing over a frequency range from 1 to 100 Hz. During the test the actuator is supported in a manner which simulates the actual valve body mounting and orientation. Stem seals and stem-to-cover clearances are duplicated on the test fixture. The shake table input equals or exceeds the specified RBS dynamic loads.

The actuator was cycled from "open" to "close" during each loading condition and operated within the specified time limits.

The capability of the main steam isolation valve to close following a downstream line break was demonstrated by the type test. The test specimen was a 20-in valve of a design representative of the RBS MSIVs.

3.9.3.2.4.1.2B Main Steam Safety/Relief Valves

A mathematical model of this valve is included in the main steam line system analysis as with the MSIVs. This analysis ensures that the equipment design limits are not exceeded.

Dynamic testing consists of multiaxis/multifrequency tests of a complete valve/actuator assembly over a frequency range from 1 to 100 Hz in both the relief and safety modes during which the SRV is required to operate within the specified limits while subjected to moments and accelerations exceeding the RBS requirements.

The safety/relief valve actuator is qualified to IEEE Standards 323-74 and IEEE 344-75 by type test.

3.9.3.2.4.1.3B Standby Liquid Control Valve (Explosive Valve)

The two SLC explosive valves are qualified to IEEE Standards 323-1974, 344-1975, and 382-1980 by type test of an RBS prototypical valve/actuator assembly. The qualification test demonstrated the capability of the valve to perform its safety function during and after exposure to the postulated dynamic and environmental challenges.

3.9.3.2.4.1.4B High Pressure Core Spray Injection Valve

Qualification of the Class 1 active HPCS injection valve (E22-F004) was performed in two parts, with the actuator qualified by type testing and the valve qualified by analysis.

Type testing of an actuator was performed in accordance with IEEE 344-1975 and 382-1980. The actuator was mounted on a valve during qualification to simulate the interface. To account for the effect of dynamic aging, the actuator was subjected to both vibration aging and SRV aging prior to the upset and faulted event RRS testing (five upset and one faulted). Following the RRS testing, the actuator was subjected to an additional series of uniaxial sine beat tests simulating the faulted event for a second time. The actuator was then tested to the LOCA (chugging) post-aging condition. Actuator operability was demonstrated during the most severe faulted event testing. Dynamic similarity between the River Bend actuator and the tested actuator was established by a similarity analysis.

A stress evaluation was performed on the yoke legs and the valve body to demonstrate structural integrity of these components when subjected to the RBS dynamic load requirements. The evaluation was performed in accordance with the rules of ASME B&PV Code Section III where applicable. At locations where the ASME code does not specifically apply (e.g., yoke legs), methods employed by vendor or methods based on principles of stress analysis were used. All stresses were shown to be within the allowable limits.

The stress evaluation also included a fatigue calculation on critical valve components. Stress cycles due to SRV blowdowns, seismic events (both upset and faulted), and chugging were considered. All components were found to satisfy the fatigue requirement. As for valve operability, qualification was demonstrated using static bend test data. The subject valve belongs to a family of block valves supplied by Anchor/Darling that had been generically tested for operability using the static bend test method. A stem deflection analysis was also performed to further demonstrate that the valve would remain operable under the most severe loading conditions.

3.9.3.2.4.2B ASME Code Class 2 and 3 Active Valves

There are six valves within the NSSS scope of supply which are Class 2 active and no Class 3 active valves. These six Class 2 active motor-operated valves are used in the HPCS system.

Qualification of these Class 2 active HPCS valves was performed similar to that for the injection valve described in Paragraph 3.9.3.2.4.1.4B, with the actuators qualified by type testing and the valves qualified by analysis. The valves were evaluated for fatigue capability where applicable and valve operability was demonstrated using static bend test data. These standard valves are seismically qualified to IEEE Standards 344-1975 and

382-1980. Qualification of the actuator furthermore meets the requirements for IEEE Standard 323-1974.

3.9.3.3B Design and Installation of Pressure Relief Devices

3.9.3.3.1B Main Steam Safety/Relief Valves

Safety/relief valve lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period from opening of the safety/relief valve until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the safety/relief valve cause the safety/relief valve discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The safety/relief valves (SRVs) are designed to the requirements of Articles NCA-3350 and NB-3560 of ASME Section III. In order to ensure that the structural integrity is capable of withstanding the dynamic effects of SRV actuation, the SRV piping is dynamically analyzed by using direct integration time history analysis of initial blowdown forces (using proper valve opening times) and response spectrum analysis for subsequent effect of the containment pedestal acceleration. The fluid transient properties are calculated at numerous locations along the pipe based on the maximum set pressure specified in the steam system specification and the value of ASME flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves and single valve openings are considered in the analysis. Simultaneous discharge is assumed in order to induce maximum stress in the piping. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid friction terms. Fig. 3.9B-6 shows a set of fluid property and pipe section load transients typical of those produced by relief valve discharge. The method of analysis does not utilize dynamic load factors. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the safety/relief valve, the main steam line, and the discharge piping are combined with loads due to other effects as specified in Section 3.9.3.1B and Table 3.9B-2.

From piping dynamic analysis, the maximum g value is defined. The SRVs are then tested to show that they can withstand higher accelerations than the maximum g value. The Code stress limits corresponding to load combinations classification as normal, upset, emergency, and faulted are applied to the steam and discharge pipe.

3.9.3.4B Component Supports

All Class 1 linear plate and shell supports within GE scope of supply are designed in complete accordance with the rules and regulations of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. These design requirements include the analysis and/or tests to demonstrate that all such component supports will not deform under faulted plant conditions to the extent that would impair the required operability of the supported components to perform a safety function for a safe shutdown of the plant.

Supports are either designed by load rating in accordance with paragraph NF-3260 or to the stress limits for linear supports in accordance with paragraph NF-3231. To avoid buckling in the component supports, Appendices F and XVII of the ASME Code, Section III require that the allowable loads be limited to two-thirds of the critical buckling loads. The critical buckling loads for ASME Code Class 1 component supports in the NSSS scope subjected to faulted loads which are more severe than normal, upset, and emergency loads, are determined by the vendor, using the methods discussed in Appendix F of the ASME Code. In general, the load combinations for the conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet the Code requirements.

All component supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed. All component supports are designed in accordance with the rules of Subsection NF of the Code. For the NSSS scope of supply, valve operators which are mounted on Class 1 piping are not used as component supports.

3.9.3.4.1B Piping

1. Hangers

   The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement. The design of hanger support is in accordance with the rules of ANSI Code B31.7.

2. Snubbers

a. Required Load Capacity and Snubber Location

The entire piping system, including valves and suspension system between anchor points, is mathematically modeled for complete structural analysis. In the mathematical model, the snubbers are modeled as a spring with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on snubbers includes those loads caused by seismic forces (operating basis earthquake and safe shutdown earthquake), system anchor movements and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

The snubber location and loading direction are first decided by estimation to confine the stresses in the piping system to acceptable values. The snubber locations and direction are refined by performing the computer analysis on the piping system as described above.

The spring constant required by the suspension design specification for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and load direction have been confirmed. If the spring constants are not in agreement, they are brought in agreement, and the system analysis is redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are compatible.

b. Design Requirements

To ensure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed.

The snubbers are required by the suspension design specification to be designed in accordance with all of the rules and regulations of the ASME Code, Section III, Subsection NF. This design requirement includes analysis where in the stresses in the snubber component parts are calculated under normal, upset, emergency, and faulted loads. These calculated stresses are then compared against the allowable stresses of the material as given in ASME Code, Section III to make sure that they are below the allowable limit.

c. Test Descriptions

Snubbers are tested dynamically to ensure that they could perform as required under upset loading conditions in the following manner:

(1) The snubbers were subjected to a force that varied approximately as a sine wave.

(2) The frequency (Hz) of the input force was in increments of 5 Hz within the range of 3 to 33 Hz.

(3) The test was conducted with the snubber at room temperature and at 200°F.

(4) The peak load in both tension and compression was equal to or higher than the rated load of the snubbers.

(5) The duration of the test at each frequency was 10 seconds or more.

Snubbers are tested dynamically to ensure that they could perform as required under emergency and faulted loading conditions in the following manner:

(1) The snubbers were subjected to forces that varied approximately as a sine wave.

(2) The test was conducted with the snubbers at room temperature.

(3) The peak load in both tension and compression was equal to 1.5 times the rated load of the snubbers.

(4) The duration of the test was 10 seconds.

The snubbers are modeled as linear elastic springs in the dynamic analysis of the piping system. The vast majority of all dynamic loadings occur with frequencies ranging from 3 to 33 Hz. By using the results of the dynamic testing, spring constants are calculated. These constants increase with higher frequencies. The average spring constant, including all lost motions (dead band, etc) of the snubber, is then used in the analytical model of the snubber.

In addition to the testing of the snubbers by themselves, General Electric has subjected the SRV piping to valve discharge while monitoring the piping system for stresses. The SRV discharge creates accoustic waves that propagate through the discharge piping and impose momentary forces on the pipe at each change in direction. The results of this testing of the piping system, with the measured frequencies at 5 Hz to 50 Hz, show a satisfactory correlation between actual stresses and predicted stresses in the pipe. Since the analytical model of the piping system uses the spring constants obtained from the aforementioned snubber test, this correlation serves as a calibration of the snubber spring constant as well as demonstrates the snubber capability above 33 Hz.

d. Snubber Installation Requirements

An installation instruction manual is required by the suspension design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing, which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

The suspension design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

e. Inspection, Testing, Repair, and/or Replacement of Snubbers

The suspension design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance, and repair of the snubber. It also contains inspection points and the period for inspection.

3. Struts

The design load on struts includes those loads caused by dead weight, thermal expansion, primary seismic forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE), system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with the ASME Code, Section III, Subsection NF-3000 to be capable of carrying the design load for all operating conditions.

4. E-BARs

The E-BAR is a simple passive device that is fabricated to be a “drop in” replacement for a snubber with the same or greater load capacity. During normal system operation, the E-BAR functions like a gapped support, which allows free thermal expansion within a predetermined range. Under dynamic loading, the E-BAR limits the pipe motion and functions as a seismic restraint.

The E-BAR is an engineered gapped strut with stiffness comparable to that of a similar sized strut. Some E-BAR components are actually typical strut components such as the rear end brackets, pinned joints, and clamps. A shaft moves through the center of two annular tapered wedges, each force fit into the E-BAR's housing (a cylinder). The E-BAR allows free (no drag or friction resistance) thermal movement through a pre-set gap during system plant operation.

The E-BARs are designed to ASME B&PV Code 1974 Edition including addenda through Winter 1979, Section III, Subsection NF, Class 1, 2 or 3 as applicable.

3.9.3.4.2B ECCS Pumps

The HPCS, LPCS, and RHR pumps are tested as defined in Section 3.9.3.2B. These tests prove the adequacy of the support structure for the pump assembly under operating conditions. Furthermore, the stress calculation summary provided in Section 3.9.3.1B defines the stress levels in the critical support areas, namely, the pressure boundary parts and nonpressure boundary parts. The stress level margins prove the adequacy of the equipment.

3.9.3.4.3B RCIC Turbine

The RCIC turbine assembly has been tested as defined in Section 3.9.2.2B. These tests proved the adequacy of the support structure for the turbine assembly under actual operating conditions. Futhermore, the calculation summary provided in Section 3.9.3.1B defined the stress levels in the critical support areas, namely, the stop valve yoke and the pedestal dowel pins and bolts. The substantial stress level margins prove the adequacy of the equipment.

3.9.3.4.4B Reactor Water Cleanup System Pump

The pump pedestal and pedestal bolts have been analyzed as discussed in Section 3.9.3.1B. Loads from seismic, dead weight, connecting pipes, and temperature were considered.

The stress limits of ASME Section III, Subsection NF were met.

3.9.4B Control Rod Drive System

This plant is equipped with an hydraulic control rod drive system. The discussion in paragraph 3.9.4B includes the control rod drive mechanism (CRDM), the hydraulic control unit (HCU), the condensate supply system, and the scram discharge volume and extends to the coupling interface with the control rods.

3.9.4.1B Descriptive Information on CRDS

Descriptive information on the control rod drives as well as the entire control and drive system is contained in Section 4.6.

3.9.4.2B Applicable CRDS Design Specifications

The control rod drive system (CRDS) is designed to meet the functional design criteria as outlined in Section 4.6 and consists of the following:

1. Locking piston control rod drive

2. Hydraulic control unit

3. Hydraulic power supply (pumps)

4. Interconnecting piping

5. Flow and pressure and isolation valves

6. Instrumentation and electrical controls.

Those components of the CRD forming part of the primary pressure boundary are designed according to ASME Code Section III.

The safety classification of the CRD and the CRD hydraulic system is outlined in Table 3.2-1, and the components are designed according to the codes and standards governing the individual safety classes.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in Section 3.9.1.1B, faulted conditions in Section 3.9.1.4B, seismic testing in Section 3.9.2.2B, loading combinations and stress limits in Table 3.9B-2t.

3.9.4.3B Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components and the CRDs have been evaluated analytically, and the design load combinations and stress limits are listed in Table 3.9B-2t. For the noncode components, experimental testing was used to determine the CRD performance under all possible conditions as described in paragraph 3.9.4.4B. The CRD piping has been designed to withstand the HCU floor dynamic response to the impact loadings associated with a LOCA pool swell event.

3.9.4.4B CRD Performance Assurance Program

The CRD test program consists of the following tests:

1. Development tests

2. Factory Quality Control tests

3. 5-yr maintenance life tests

4. 1.5X design life tests

5. Operational tests

6. Acceptance tests

7. Surveillance tests.

All of the above tests except 3 and 4 are discussed in paragraphs 4.6.3 through 4.6.3.1.1.5. Tests 3 and 4 are discussed below:

Test 3 - 5-Year Maintenance Life Tests

Four control rod drives are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and 1/6 of the cycles specified in paragraph 3.9.1.1B.

Upon completion of the test program, control rod drives must meet or surpass the minimum specified performance requirements.

Test 4 - 1.5X Design Life Tests

When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the life test cycles specified in paragraph 3.9.1.1B.

Two CRDs have undergone such testing in 1976. Upon completion of the test program, these CRDs met or surpassed the minimum specified performance requirements.

3.9.5B Reactor Pressure Vessel Internals

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel internals.

3.9.5.1B Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are identified below:

Core Support Structures
- Shroud
- Shroud support cylinder, plate, and legs (part of the RPV core plate and core plate hardware)
- Grid (only that portion below the bottom weld in the cylindrical portion is core support structure. The grid is a part of the top guide assembly.)
- Top guide (hardware studs, nuts, and eccentric sleeves between top guide and shroud)

Orificed fuel supports (except for the orifices which do not support or restrain the core)
CRD Housing (only that portion above the first weld that is above the housing to pressure vessel weld)
Control rod guide tubes

Reactor Internals
Jet pump assemblies, braces, and instrumentation
**Feedwater spargers
Vessel head spray nozzle
Differential pressure and liquid control lines
**In-core flux monitor guide tube
**Initial startup neutron sources
**Surveillance sample holders
Core spray lines and spargers
**In-core instrument housings
LPCI coupling

A general assembly drawing of the important reactor components is shown in Fig. 3.9B-7.

The floodable inner volume of the reactor pressure vessel can be seen in Fig. 3.9B-8. It is the volume inside the core shroud up to the level of the jet pump suction inlet.

The design arrangement of the reactor internals, such as the jet pumps steam separators and guide tube, is such that one end is unrestricted and thus free to expand.

The LPCI couplings incorporate vertically oriented slip fit joints to allow free thermal expansion.

3.9.5.1.1B Core Support Structures

The core support structures consist of those items listed in Section 3.9.5.1B. These structures form partitions within the reactor vessel to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Fig. 3.9B-8 shows the reactor vessel internal flow paths.

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**Non-safety class component.

3.9.5.1.1.1B Shroud

The shroud support, shroud, and top guide make up 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 this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide's grid plate below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the top by the grid plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the reactor pressure vessel shroud support.

3.9.5.1.1.2B Shroud Support

The shroud support is designed to support the shroud and to support and locate the jet pumps. The shroud support provides an annular baffle between the reactor pressure vessel and the shroud. The jet pump discharge diffusers penetrate the shroud support to introduce the coolant to the inlet plenum below the core.

3.9.5.1.1.3B Shroud Head and Steam Separator Assembly

This component is not a core support structure or safety class component. It is discussed here to describe the coolant flow paths in the reactor pressure vessel. The shroud head and steam separator assembly is bolted to the top of the top guide 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 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 vanes that impart a 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 downcomer annulus.

3.9.5.1.1.4B Core Plate

The core plate consists of a circular stainless steel plate with bored holes stiffened with a rim and beam structure. The plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core support plate.

The entire assembly is bolted to a support ledge on the lower portions of the shroud.

3.9.5.1.1.5B Top Guide

The top guide consists of a circular grid plate with square openings welded to the bottom of the top guide cylinder. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Notches are provided in the bottom of the intersections to anchor the in-core flux monitors and startup neutron sources. The top guide is bolted to the shroud. The core spray spargers are installed in the upper portion of the top guide cylinder.

3.9.5.1.1.6B Fuel Support

The fuel supports shown in Fig. 3.9B-9 are of two basic types: namely, peripheral supports and four-lobed orificed fuel supports. The peripheral fuel support is located at the outer edge of the active core and is not adjacent to control rods. Each peripheral fuel support supports one fuel assembly and contains a single orifice assembly designed to ensure proper coolant flow to the peripheral fuel assembly. Each four-lobed orificed fuel support supports four fuel assemblies and is provided with four orifice plates to ensure proper coolant flow distribution to each rod-controlled fuel assembly. The four-lobed orificed fuel supports rest in the top of the control rod guide tubes, which are supported laterally by the core plate. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell. (See Section 4.4.2, Fuel System Design Description and Drawings.)

3.9.5.1.1.7B Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings up through holes in the core plate.

Each tube is designed as the guide for a control rod and as the vertical support for a four-lobed orificed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing, which in turn transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the control rod drive 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 control rod drive housing to hold the thermal sleeve in position.

3.9.5.1.1.8B Jet Pump Assemblies

The jet pump assemblies are not core support structures but are discussed here to describe coolant flow paths in the vessel. The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the jet pump is covered in detail in References 1 and 2. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (see Fig. 3.9B-10). The driving nozzles, suction inlet, and throat comprise the inlet mixer assembly which is removable as a unit. The diffuser is welded to the shroud support ledge. High pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace which provides lateral support for the riser pipe assembly is welded to the riser pipe and to pads on the reactor vessel walls.

The inlet mixer assembly is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The inlet mixer 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.

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The preload on RBS Jet Pump Beams is 25 kips in accordance with General Electric Recommendations. Inservice Inspection of these beams will be performed in accordance with the recommendations of the BWRVIP.

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3.9.5.1.1.9B Steam Dryers

The steam dryer assembly is not a core support structure nor safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryers remove moisture from the wet steam leaving the steam separators.

The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing 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 during installation 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, which may occur under accident conditions, is restricted by steam dryer holddown brackets attached to the reactor vessel top head.

3.9.5.1.1.10B Feedwater Spargers

The feedwater nozzle and sparger design follows the resolution presented in Reference 6. These components are not core support structures nor safety class components. They are discussed here to describe flow paths in the vessel. The feedwater spargers are stainless steel headers in a forged tee design 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 pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.1.11B Core Spray Lines

This component is not a core support structure. It is discussed here to describe a safety class feature inside the reactor pressure vessel. The core spray lines are the means for directing flow to the core spray nozzles, which distribute coolant during accident conditions.

Two core spray lines enter the reactor vessel through the two core spray nozzles. (See Section 5.4, Component and Subsystem Design.) 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 top guide cylinder immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the top guide cylinder. The two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the top guide and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the top guide cylinder. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers. (See Section 6.3, Emergency Core Cooling System.)

3.9.5.1.1.12B Vent Nozzle

This component is not a core support structure. It is included here to describe a safety class feature in the reactor pressure vessel. The vent assembly performs the function of providing a vent for the noncondensible gases in the vessel head. The vent assembly is bolted to a mating flange on the reactor vessel head nozzle. External piping is connected to the assembly by means of standard flanges. (See Section 5.4.7, Residual Heat Removal System.)

3.9.5.1.1.13B Differential Pressure and Liquid Control Line

This component is not a core support structure or safety class component. It is discussed here to describe the coolant paths in the reactor vessel. The differential pressure and liquid control lines enter the vessel through two bottom head penetrations and serves a dual function within the reactor vessel - to sense the differential pressure across the core support plate (described in Section 5.4, Component and Subsystem Design) and to provide a path for the injection of the liquid control solution into the coolant stream. One line terminates near the lower shroud with a perforated length below the core support plate. It is used to sense the pressure below the core support plate during normal operation and to inject liquid control solution if required. This location facilitates good mixing and dispersion. The other line terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.1.14B In-Core Flux Monitor Guide Tubes

This component is not a core support structure or safety class component. It is discussed here to describe the coolant flow paths in the reactor vessel. Provide a means of positioning fixed detectors in the core as well as provide a path for calibration monitors (TIP System).

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing (see Section 5.4, Component and Subsystem Design) in the lower plenum to the top of the core support plate. 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. A latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.9.5.1.1.15B Surveillance Sample Holders

This component is not a core support structure or a safety class component. It is discussed here to describe the coolant flow paths in the reactor vessel. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (see Section 5.4, Component and Subsystem Design). The baskets hang from the 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.

3.9.5.1.1.16B Low Pressure Coolant Injection Lines
This component is not a core support structure but is discussed here to describe the coolant flow paths in the reactor vessel. Three LPCI lines penetrate the core shroud through separate LPCI nozzles. Coolant is discharged inside the core shroud.

3.9.5.2B Design Loading Conditions

3.9.5.2.1B Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied by core support structures and engineered safety features components reveals three significant faulted events:

1. Recirculation Line Break: A break in a recirculation line between the reactor vessel and the recirculation pump suction.

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 structures within the reactor.

3. Earthquake: subjects the core support structures and reactor internals to significant forces as a result of ground motion.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other engineered safety feature reactor internals are less severe than these three postulated events. The faulted conditions for the reactor pressure vessel internals are discussed in Section 3.9.1.4B. Loading combination and analysis for the reactor pressure vessel internals are discussed in Section 3.9.3.1B, Table 3.9B-1 and Table 3.9B-2. The core support structures are designed in accordance with Subsection NG of Section III of the ASME Code.

3.9.5.2.2B Pressure Differential During Rapid Depressurization

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A digital computer code is used to analyze the transient conditions within the reactor vessel following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes, which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressure in the various regions of the reactor. The computer code initially used is the General Electric Short-Term Thermal-Hydraulic Model described in Reference 3. This model has been approved for use in ECCS conformance evaluation under 10CFR50, Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are, therefore, not described in Reference 3. For 105% Power Uprate, additional analysis were performed using LAMB thermal-hydraulic computer core. The analysis results remain bounding of TPO (Appendix K) uprate conditions. The main steam line (MSL) break inside containment is postulated for calculating Reactor Internal Pressure Differentials (IPDS), except for the steam dryer RIPD for which the steam line break outside containment is postulated. The locations at which RIPDS are applicable and calculated are shown is Figure 3.9B-11. The bounding RIPD values which result from the range of possible reactor power and flow conditions are then determined.

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These additional features are discussed below:

1. The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

2. The flow path between the bypass region and the shroud head is more accurately modeled since the fuel assembly pressure differential is influenced by flashing in the guide tubes and bypass region for a steamline break. In the ECCS analysis, the momentum equation is solved in this flow path, but its irreversible loss coefficient is conservatively set at an arbitrary low value.

3. The enthalpies in the guide tubes and the bypass are calculated separately, since the fuel assembly ΔP is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3B Recirculation Line and Steam Line Break

3.9.5.2.3.1B Accident Definition

Both a recirculation line break (the largest liquid break) and an inside steam line break (the largest steam break) are considered in determining the design basis accident for the engineered safety feature reactor internals. The recirculation line break is the same as the design basis loss-of-coolant accident described in Section 6.3, Emergency Core Cooling Systems. A sudden, complete circumferential break is assumed to occur in one recirculation loop. The pressure differentials on the reactor internals and core support structures are in all cases lower than for the main steam line break.

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 and thus a faster depressurization rate than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor internal structures.

The steam line break accident produces significantly higher pressure differentials across the reactor internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steam line break. Therefore, the steam line break is the design basis accident for internal pressure differentials.

3.9.5.2.3.2B Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant, the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. The core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and, consequently, the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the reactor pressure vessel and thus the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

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To ensure that the calculated pressure differences bound those which could be expected if a steam line break should occur, an analysis is conducted at a low power, high recirculation flow condition (cavitation interlock) in addition to the standard safety analysis condition at high power, rated recirculation flow. The bounding RIPD values for MSL break from the cavitation interlock or full power operation is determined and used for reactor internal structural evaluations. The cavitation interlock is designated Case 2 and full power operation is designated Case 1.
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This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

Table 3.9B-5 summarizes the maximum pressure differentials. Case 1 is the safety analysis condition; Case 2 is the low power - high flow condition. Comparison of these values illustrates the statements made in the foregoing paragraphs.

3.9.5.2.4B Earthquake

The seismic loads acting on the structures within the reactor vessel are based on a dynamic analysis as described in Section 3.7B. Seismic analysis is performed by coupling the lumped mass model of the reactor vessel and internals, as described in Section 3.7B, with the building model to determine the system natural frequencies and node shapes. The relative displacement, acceleration, and load response is then determined by either the time-history method or the response-spectrum method. In the time-history method, the dynamic response is determined for each node of interest and added algebraically for each instant of time. Resulting response time histories are then examined, and the maximum value of displacement, acceleration, shears, and moments are used for design calculations.

In the response-spectrum method, the relative displacements, accelerations, shears, and moments are determined for each node of interest. The square root of the sum of the squares of these individual responses is then used for design calculations.

The detailed descriptions of the earthquake analysis are given in Section 3.7B. The detailed description of the dynamic response analysis to these forcing functions is given in Section 3.9.2.5B.

3.9.5.3B Design Bases

3.9.5.3.1B Safety Design Bases

The reactor core support structures and internals meet the following safety design bases:

1. They 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. Deformation is limited to assure that the control rods and core standby cooling systems can perform their safety functions.

3. Mechanical design of applicable structures assures that safety design bases (1) and (2), above, are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.3.2B Power Generation Design Bases

The reactor core support structures and internals are designed to the following power generation design bases:

1. They provide the proper coolant distribution during all anticipated normal operating conditions up to full power operation of the core without fuel damage.

2. They are arranged to facilitate refueling operations.

3. They are designed to facilitate inspection.

3.9.5.3.3B Design Loading Categories

Loading combinations for the core support structures are shown in Table 3.9B-6. The basis for determining faulted loads on the reactor internals is shown for seismic loads in Section 3.7B and for pipe rupture loads in Sections 3.9.5.2.3B and 3.9.5.3.4B. Table 3.9B-2x provides the same type of information for the CRD guide tube. Tables 3.9B-2v, 3.9B-2w, and 3.9B-2y show the loading combinations for the jet pump, LPCI coupling, and in-core housing, respectively.

Core support structure and safety class internals stress limits are consistent with ASME Code Section III, Categorization of Loading Conditions (NA-2140), and associated stress limits contained in Addenda dated through Summer 1976. Levels A, B, C, and D service limits defined in Winter 1976 Addenda which replace normal, upset, emergency, and faulted condition limits are not reflected in design documents for core support structures and other safety class internals for this reactor. However, for these components Levels A, B, C, and D service limits are judged to be equivalent to the normal, upset, emergency, and faulted loading conditions limits, and therefore, for clarity, both sets of nomenclatures are retained herein.

Stress intensity and other design limits are discussed in Section 3.9.5.3.5B. The core support structures which are fabricated as part of the reactor pressure vessel assembly are discussed in Section 3.9.1.3B.

The design requirements for equipment classified as "other internals," e.g., steam dryers and shroud heads, were specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it is to operate. Where possible, design requirements are based on applicable industry codes and standards. If these are not available, the designer relies on accepted industry or engineering practices.

3.9.5.3.4B Response of Internals Due to Inside Steam Break Accident

The maximum pressure loads acting on the reactor internal components result from an inside steam line break, and on some components the loads are maximum when operating at the minimum power associated with the maximum core flow (Table 3.9B-5, Case 2). This has been substantiated by the analytical comparison of liquid versus steam breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that it is possible but not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be as listed under Case 1 in Table 3.9B-5.

3.9.5.3.5B Stress, Deformation, and Fatigue Limits for Engineered Safety Feature Reactor Internals (Except Core Support Structure)

The stress deformation and fatigue criteria listed in Tables 3.9B-7, 3.9B-8, 3.9B-9, and 3.9B-10 are used or the criteria are based on the criteria established in applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical methods based on field experience and testing. For the quantity SFmin (minimum safety factor) appearing in those tables, the following values were used.

Service Level	Design Condition	SF
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Faulted	1.125

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined for adequate clearances during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Section 3.9.2.5B.

The ASME Code Article NG-3000 is used as a guide for the safety-related reactor internals other than the core support structures. The reactor internals are designed such that they do not adversely affect the integrity of the core support structure.

3.9.5.3.6B Stress and Fatigue Limits for Core Support Structures

The stress, fatigue, and other limits for the core support structures are in accordance with ASME Section III, Subsection NG.

3.9.6B Inservice Testing of Pumps and Valves

See Section 3.9.6A.

References - 3.9B

1. Design and Performance of G.E. BWR Jet Pumps. General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

2. Moen, H.H. Testing of Improved Jet Pumps for the BWR/6 Nuclear System. General Electric Company, Atomic Power Equipment Department, NEDO-10602, June 1972.

3. Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K. Proprietary Document, General Electric Company, NEDE-20566.

4. BWR Fuel Channel Mechanical Design and Deflection. NEDE-21354-P, September 1976.

5. BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings. NEDE-21175-P, November 1976, and NEDE-21175-3-P, July 1982.

6. Boiling Water Reactor Feedwater Nozzle/Sparger Final Report, NEDO-21821, March, 1978.

7. PISYS Analysis of NRC Problem, NEDO-24210, August 1979.

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3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Two inputs are provided for Section 3.10: Section 3.10A applies to the SWEC scope of supply, and Section 3.10B applies to the GE scope of supply.

3.10A SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT (SWEC SCOPE OF SUPPLY)

This section provides the qualification criteria and methods for equipment affected by seismic loads. The methods for the qualification of equipment affected by the suppression pool induced dynamic loads are provided in Appendix 6A, subsection 6A.17.

3.10.1A Seismic Qualification Criteria

Table 3.10A-1 provides a listing of Seismic Category I instrumentation and electrical equipment requiring seismic qualification.

Parameters used to develop seismic loadings and criteria for Seismic Category I structures, systems, and components are described in Section 3.7. From the basic ground input data, a series of response spectra curves at various building elevations was developed. The magnitude and frequency of the SSE loadings that each component experiences are determined by the locations within the plant. These seismic data were included in the purchase specifications for Seismic Category I equipment and systems. For equipment located at various areas throughout the plant, the purchase specification includes response spectrum curves which envelop the response spectra at all locations where the instrumentation is used.

Qualification and documentation procedures used for Class 1E equipment and/or systems meet the provisions of IEEE 344-1975[1], as supplemented by Regulatory Guide 1.100, Revision 1.

Seismic Category I instrumentation is divided into two classifications: 1) equipment that is designed to maintain its functional capability during and after an SSE; and 2) equipment that, although not required to maintain its functional capability, is designed to maintain the pressure boundary integrity of the system of which it is a part during and after an SSE. The requirements for instrumentation, equipment, and systems which are required to maintain pressure boundary integrity are in accordance with the ASME Code, Section III, 1974, or later, depending on time of purchase of equipment. The performance requirements of Seismic Category I electrical and instrumentation items and their respective supports may be structural as well as functional. The structural design is in accordance with applicable codes.

If no codes are applicable, the stress level for the OBE combined with operating loads is limited to 75 percent of the minimum yield for the material in accordance with the ASTM specification. For the SSE combined with operating loads, the stress level does not exceed the smaller of:

1. 100 percent of the minimum yield strength, or

2. 70 percent of the minimum ultimate tensile strength of the material (at temperature), in accordance with the ASTM specification.

Seismic analysis, without testing, is performed on equipment whose functional operability is assured by its structural integrity alone. When complete seismic testing is impractical, a combination of tests and analyses is performed. See Table 3.10A-1 for the seismic qualification methods applicable to specific equipment.

3.10.2A Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The methods by which the supplier can qualify equipment are as follows:

1. Testing

2. Type-testing (prototype)

3. Analysis

4. Combination of 1 or 2 and 3.

These methods, including the factors for selection of an analytical or test option, test objectives, and acceptability criteria, are described in Section 3.7.3.1.1A. Qualification and documentation procedures used for Seismic Category I equipment and/or systems meet the provisions of IEEE 344-1975, as supplemented by the requirements of Regulatory Guide 1.100, Revision 1.

3.10.2.1A Testing

Seismic tests are performed by subjecting equipment to vibratory motion that conservatively simulates the SSE at the equipment mounting. Vibratory tests are conducted over the range of 1 to 33 Hz.

Whenever feasible, seismic qualification tests on equipment are performed while the equipment is subjected to normal operating loads. However, occasionally an operational configuration is difficult to simulate correctly, and where it can be demonstrated that operating loads such as pressure, torque, flow, voltage, current, or temperature do not cause significant stress loads within the equipment, or where such operating loads are not significant to a determination of equipment operability, operation under load is not specified. The equipment is monitored and evaluated during and after the test for malfunction or failure and, upon completion of the test, is tested for proper operation.

In seismic qualification testing, equipment auxiliary components, such as relays, switches, and instruments necessary for proper operation, are mounted in a manner similar to which they are to be installed, and then tested and qualified along with the equipment.

For multicabinet assemblies, the test prototype units sometimes consist of smaller number of cabinets than the assembly being qualified. In such cases, an evaluation of the responses due to the front-to-back, side-to-side, vertical, and torsional modes of the multicabinet assemblies with respect to those of the tested units is made. This evaluation ensures the adequate qualification of the multicabinet assemblies and of the electrical components located in them.

The input motion is applied to the vertical axis, combined with each one of the principal horizontal axes. The maximum input motion acceleration is equal to, or is in excess of, the maximum seismic acceleration expected at the equipment mounting location. Following the requirements of Regulatory Guide 1.100, Revision 1, it is specified that the Test Response Spectrum closely envelops applicable portions of the Required Response Spectrum in verifying the adequacy of test input motion.

3.10.2.2A Prototype Testing

In some cases, where groups of equipment have similar characteristics, the test program is based upon testing of a prototype item of equipment. The test reports, furnished by the equipment supplier, are reviewed for assurance that the group of components qualified by the prototype is dynamically similar. If any extrapolation as to dimension or mass is used, the vendor is required to justify similarity of the dynamic characteristics.

3.10.2.3A Analysis

Analysis without testing is acceptable only if structural integrity alone could assure the design-intended function.

Responses are calculated for the three-directional seismic loadings individually and combined by the square root of the sum of the squares (SRSS) method. The seismic response is added to the operating load response on an absolute basis to establish the combined effects, and compared with allowable stress, strain, or deflections, as the basis for acceptable qualification.

3.10.2.4A Combined Analysis and Testing

When the equipment cannot be practically qualified by analysis or testing alone because of its complexity or size, combined analysis and testing is used. When this procedure is employed, the major component is qualified by analyses, and the motors, operators, and appurtenances necessary for operation are qualified by testing. The auxiliary equipment is tested and qualified to the acceleration level at its mounted location, and its equivalent seismic loading is applied to the major component being analyzed.

3.10.3A Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

A design objective, when feasible, is to provide supports, for electrical equipment, instrumentation, and control systems, with fundamental natural frequencies above the cutoff frequency of the relevant amplified response spectra curves. This assures that amplification of floor accelerations through supporting members to mounted equipment is minimized.

The response of racks, panels, cabinets, and consoles is considered in assessing the capability of instrumentation and electrical equipment. Electrical equipment and instrumentation is tested, wherever feasible, with their supporting structures in their installed configuration. Mounted components are qualified to acceleration levels at their mounting location which takes into account the transmissibility of the supporting structure.

Determination of amplification and seismic adequacy of instrumentation and electrical equipment is implemented by the analysis and testing methods outlined in Section 3.7.3A.

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Cable trays are designed for static acceleration loads equal to 1.3 times the applicable peak amplified resonant response at the support elevations using values of 4-percent damping (OBE) and 8-percent damping (SSE). The use of 4- and 8-percent damping is based on the results of vibration tests performed on field-installed systems[2]. The dynamic analysis method is used to determine support parameters (geometry, size, and spacing) for cable tray systems, with OBE and SSE loadings based on 2-percent and 4-percent critical damping, respectively. The design is established so that no adverse deformation or failure occurs during the SSE. Normal working stresses are maintained during OBE conditions.

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Additionally, restraints are used as necessary to limit the horizontal lateral loads to allowable design values, established on the basis of cable tray loading and unsupported span lengths. Design provisions for significant differential motions between buildings are made by breaks in the trays, if these relative displacements result in unacceptable equipment or support loadings.

3.10.4A Operating License Review

The results of all seismic tests and analyses performed by outside vendors requires review and approval. These results become a permanent onsite record. A summary of seismic test and/or analysis results is given in Table 3.10A-1.

References - 3.10A

1. IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations. IEEE 344-1975, The Institute of Electrical and Electronic Engineers, 1975.

2. Final Report on Determination and Utilization of Modal Properties of Cable Tray Systems. Report No. 1787, prepared by Structural Dynamics Research Corp. for Stone & Webster Engineering Corporation, September 26, 1974.

3.10B SEISMIC AND HYDRODYNAMIC QUALIFICATIONS OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT (GE SCOPE OF SUPPLY)

3.10.1B Dynamic Qualification Criteria

3.10.1.1B Seismic Category I Equipment Identification

Seismic Category I instrumentation and electrical equipment is listed in Table 3.2-1. "Active" NSSS pumps, motors, valves and valve-mounted equipment are listed in Tables 3.9B-3a and 3.9B-3b.

Seismic Category I instrumentation and electrical equipment is designed to withstand the faulted event without functional impairment.

The Class 1E instrumentation, electrical equipment, and support structures supplied by GE requiring seismic qualification are identified in Table 3.10B-1. The seismic qualification of these instrumentation, equipment, and supports is described in the following subsections.

Section 3.9.2.2B addresses the dynamic qualification testing and analysis of the Category I mechanical components, equipment, and their supports, including the integral or associated electrical components such as valve-mounted components and pump motors.

3.10.1.2B Dynamic Design Criteria

3.10.1.2.1B NSSS Equipment

The seismic criterion used in the design and subsequent qualification of all Class lE instrumentation and electrical equipment supplied by GE is as in the following paragraph.

The Class lE equipment is capable of performing its safety-related functions during 1) normal plant operation, 2) anticipated transients, 3) design basis accidents, and 4) post-accident operation while being subjected to, and after the cessation of, the accelerations resulting from the seismic and hydrodynamic loads at the point of attachment of the equipment to the building or supporting structure.

The criteria for each of the devices used in the Class lE systems depend on the use in a given system; for example, a relay in one system may have as its safety function to deenergize and open its contacts within a certain time, while in another system it must energize and close its contacts. Since GE supplies many devices for many applications, the approach taken was to test the device in the worst case configuration. In this way, the capability of protective action initiation and the proper operation of fail-safe circuits is assured.

•→14

From the basic input ground motion data, a series of response curves at various building elevations are developed after the building layout is completed. Standard requirement levels that meet or exceed the maximum expected unique plant information are included in the design specifications for Seismic Category I equipment. Equipment is dynamically qualified either by GE or by the supplier. In either case test data, operating experience, and/or calculations are used to substantiate that the components, systems, etc, do not suffer loss of their safety function during or after exposure to seismic and hydrodynamic loads. The magnitude and frequency of the SSE loadings which each component may experience are determined by its specific location within the plant.

14←•

3.10.2B Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

3.10.2.1B Methods of Showing NSSS Equipment Compliance with IEEE 344-1975 and Regulatory Guide 1.100

Procedures

GE supplied Class 1E equipment meets the requirement that the dynamic qualification should demonstrate the capability to perform the required safety function during and after the seismic and hydrodynamic loads. Both analysis and testing were used but most equipment was tested. Analysis was primarily used to determine the adequacy of mechanical strength such as mounting bolts and pressure boundaries.

Analysis - GE supplied Class 1E equipment performing primarily a mechanical safety function (pressure boundary devices, etc) was analyzed since the passive nature of their critical safety role usually made testing unnecessary. Analytical methods sanctioned by IEEE 344–1975 were utilized in such cases (See Table 3.10B-1 for indication of which items were qualified by analysis).

Testing - GE supplied Class 1E equipment having an active electrical safety function was tested in compliance with IEEE 344-1975.

Documentation

Available documentation verifies that the seismic qualification of GE supplied Class 1E equipment is in accordance with the requirements of IEEE 344-1975 and Regulatory Guide 1.100.

3.10.2.2B Testing Procedures for Qualifying Electrical Equipment and Instrumentation

The test procedure required that the device be mounted on the table of the vibration machine in a manner similar to its normal, installed configuration. The device was tested in the operating states as if it were performing its Class 1E functions, and these states were monitored before, during, and after the test to assure proper function and absence of spurious function. In the case of the relay example, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay is used in those configurations in its Class 1E functions.

The dynamic excitation was a random multiple frequency test in which the applied vibration was a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. The vibratory excitation was applied in two orthogonal axes (horizontal and vertical) simultaneously with the axes chosen as those coincident with the most probable mounting configuration. The device was then rotated 90 deg in the horizontal plane and the test repeated. Each device therefore has been tested in the three major orthogonal axes.

The first step was usually a search for resonances in each axis. This was done since resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search was usually run at low acceleration levels in order to avoid damaging the test sample in case a severe resonance was encountered. The resonance search was performed for the applicable frequency range in accordance with IEEE 344; if the device was large enough, the vibrations were monitored by accelerometers placed at critical locations from which resonances were determined by comparing the acceleration level with that at the table of the vibration machine. Sometimes, the devices either were too small for an accelerometer, with their critical parts in an inaccessible location, or had critical parts that would be adversely affected by the mounting of an accelerometer. The vibrations were monitored at the closest location.

Following the frequency scan and resonance determination, the devices were tested to determine their dynamic capability. For multifrequency testing, five OBE and one SSE test were run at the appropriate TRS. In some cases, the TRS was gradually increased until device malfunction occurred or the shake table limit was reached. For single-frequency testing, a malfunction limit test was run at each resonant frequency as determined by the frequency scan. In this test, the acceleration level was gradually increased until either the device malfunctioned or the limit of the vibration machine was reached. If no resonances were detected (as was usually the case), the device was considered to be rigid (all parts move in unison) and the malfunction limit was therefore independent of frequency. To achieve maximum acceleration from the vibration machine, rigid devices were malfunction tested at the upper test frequency since that allowed the maximum acceleration to be obtained from deflection-limited machines.

The summary of the tests on the devices used in Class 1E applications is given in Table 3.10B-1.

The above procedures were required of purchased devices as well as those made by GE. Vendor test results were reviewed and if unacceptable, the tests were repeated either by GE or the vendor. If the vendor tests were adequate, the device was considered qualified to the limits of the test.

3.10.2.3B Qualification of Valve Operators

The qualification of valve operators is discussed in Section 3.9.2.2B.

3.10.2.4B Qualification of NSSS Motors

Seismic qualification of NSSS motors is discussed in Section 3.9.2.2B.

3.10.3B Methods and Procedure of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

3.10.3.1B Dynamic Analysis and Testing Procedures

3.10.3.1.1B Panel-Mounted Equipment

The Class 1E equipment supplied by GE is used in many systems on many different plants and is subjected to widely varying dynamic loads. The qualification tests were performed to envelop the applicable frequency range. For supports subjected to seismic loads only, the tested frequencies range from 1 to 33 Hz. Where testing below 5 Hz was limited by the capability of the test facility, a combination of test and analysis was used to ensure that there were no untested resonances.

For multicabinet assemblies that are too large for the test table, one or two bays of the assembly are tested which gives representative results in the front-to-back and vertical directions. The side-to-side results are evaluated and are generally found to be conservative due to the increased flexibility of the narrower section. If conservatism cannot be established, the panel is accurately modeled and a computer analysis of its structural response is performed.

Some GE supplied Class 1E devices were qualified by analysis only. Analysis was used for passive mechanical devices and was sometimes used in combination with testing for larger assemblies containing Class 1E devices. For instance, a test might have been run to determine if there were natural frequencies in the equipment within the critical frequency range. If the equipment was determined to be free of natural frequencies within the critical seismic frequency range, then it was assumed to be rigid and a static analysis was performed. If it had natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations were determined to see if Class 1E devices mounted in the assembly would operate without malfunctioning. In general, the testing of Class 1E equipment was accomplished using the following procedure.

Assemblies (i.e., control panels and local racks) containing devices with established seismic and hydrodynamic malfunction limits were mounted on the table of a vibration machine in the manner it was to be mounted when in use. All control panel and local rack tests have been performed according to the requirements of IEEE 344-1975. The initial vibration test in each case was a low level resonance search. As with the devices tested to IEEE 344-1975, the assemblies were tested in the three major orthogonal axes. The resonance search was run in the same manner as described for devices. If resonances were present, the transmissibility between the input and the location of each Class 1E device was determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to conservatively determine the input motion at any Class 1E device location for any given input to the base of the assembly.

The full acceleration level tests described disclosed that the panel types had more than adequate mechanical strength and that acceptability was just a function of its amplification factor and the malfunction levels of the devices mounted in it. Many devices were mounted in the test panel or rack and qualified as an assembly. Other devices were tested individually as described above. Sometimes panels were tested at lower acceleration levels and the transmissibilities measured to the various devices. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input the panel seismic qualification level could be determined. Several high level tests have been run on selected generic panel designs to assure the conservativeness in using the transmissibility analysis described.

3.10.4B Operating License Review

The dynamic test results for safety-related panels and control equipment within the NSSS scope are maintained in a permanent file by GE and can be readily audited in all cases. The equipment used in Class 1E applications at RBS passed the prescribed tests.

A summary of the test results for the devices used in Class 1E applications is given in Table 3.10B-1.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

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Safety-related mechanical and electrical (including instrumentation and control) equipment is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCA. Safety-related equipment located in a harsh environment area (i.e., where the environment resulting from an accident would be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences) is furthermore environmentally qualified to demonstrate the absence of common mode failures. Seismic qualification is addressed in Sections 3.9 and 3.10.

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3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Equipment Identification

Safety-related electrical equipment includes all three categories of 10CFR50.49(b)[1]. Safety-related mechanical equipment includes active equipment, i.e., equipment which must move or change position to perform its design safety function (e.g., pumps, motor-operated valves, safety-relief valves, or check valves).

•→7

The systems and subsystems required to mitigate an accident are listed in Table 3.11-1 and Chapter 15. Table 3.11-2 provides a matrix of these systems with the accidents for which they are required to be operable. The functions of some of the systems are limited under the postulated accident conditions. These limitations are described as follows. The only portion of the HVY system which is required to operate for an accident is the portion servicing the standby service water pumphouse to ensure that an adequate environment is provided for the SWP pumps and associated components.

•→15

The RCIC system is listed for all accidents where steam is available to supply the RCIC turbine. RCIC is not required to directly mitigate the consequences of any of the listed accidents.

15←• •→14

The SFC system is indicated as being required for all accidents since it must operate to remove decay heat whenever spent fuel is stored in the spent fuel pool. It is not required to directly mitigate the consequences of any of the listed accidents. In addition, the HVF system must operate to ensure an adequate environment for the SFC system, and to maintain Fuel Building integrity during movement of recently irradiated fuel.

14←•

The SLS system is required to operate only in the event that the normal scram system (RPS) or control rods do not function to shut down the reactor. Since each accident listed results in a scram, the SLS is not required for all accidents, but is considered to be available.

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The containment isolation system includes the containment isolation valves associated with the systems listed in Table 3.11-4. The SWEC/GE system cross reference is listed in Table 3.11-3.

The standby power system is capable of supplying ac power for electrical loads which are required for safe shutdown of the reactor. The systems included in the SPS are listed in Table 3.11-5.

The remote safe shutdown system includes the Division I and II remote shutdown panels, and is designed to achieve safe shutdown from either panel. The RRS utilizes some of the existing systems used for normal reactor shutdown operation to shut down the reactor from outside the main control room.

Equipment must be qualified for the length of time it is required to perform its safety function and must remain in a safe mode after the function is performed. The length of time the equipment is required to function following the onset of an accident is its post-accident operability time (PAOT).

The required PAOT is determined by an analysis of the functional performance requirements for each applicable event. For each device the required active function time, based upon the intended safety function, is determined. The determined operability time ranges from very short periods, where safety functions are performed early in the event and subsequent actions are not called for, to longer periods of time where process variables are stabilized. For long periods of required operability, the parameters selected to environmentally qualify needed products assure the capability to bring the reactor to, and maintain a safe shutdown condition.

The PAOT is stated in the Equipment Qualification Master List.

3.11.1.2 Environmental Conditions

The indoor environmental design basis conditions which have been used to establish the design basis for RBS are specified for normal, abnormal, and accident conditions. The environmental data for temperature, pressure, humidity, and radiation are defined for each building zone.

Environmental zones are classified as either harsh or mild. The classification of mild zones is in compliance with 10CFR50.49(c)[1]. There is no significant change in environmental conditions, except radiation, in these zones. Additionally for a zone to be classified as mild, the total integrated dose for 40 years plus 180 days post-accident service is no more than 10^4 rads. Tables 3.11-6, 3.11-7, and 3.11-8 list and describe the harsh and mild environmental zones. Illustrations of the physical zones are provided on the environmental zone maps, Figures 3.11-1 through 3.11-5.
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3.11.1.2.1 Temperature, Pressure, Humidity Environment

The plant ventilation/air-conditioning system is designed to maintain maximum normal operating temperature and pressure conditions for all normal reactor operating modes, including startup, power range, hot standby, shutdown and refueling. The humidity is generally uncontrolled. Normal conditions are assumed to prevail until an abnormal or accident condition occurs, in which case the abnormal or accident condition prevails for the specified duration. Following the abnormal or accident condition, normal conditions are again assumed to exist.

Abnormal operating conditions are any reasonably expected or anticipated deviations from abnormal conditions (excluding accident conditions). Abnormal operating conditions include:

1. Transients that result from main steam line isolation (loss of condenser vacuum, turbine trip) and MSIV closure (including a stuck open relief valve).

2. Transients caused by a single failure of one division of redundant essential HVAC equipment.

3. Transients caused by the loss of nonessential drywell HVAC equipment as the result of a loss of offsite power.

•→12
The abnormal temperatures calculated for the auxiliary building under condition 2 (above) envelop conservative design maxima that have a low probability of occurrence since they are associated with normal plant operating modes that prevail for less than 1 percent of the plant operating time (i.e., residual heat removal and shutdown cooling). Equipment in the auxiliary building will not be qualified for these temperatures since the plant can be shut down using redundant equipment that is not affected by the postulated abnormal event. The drywell temperatures associated with condition 3 (above) are enveloped by the applicable accident conditions; however, their effect on the qualified life of equipment generally is not considered. Following occurrence of abnormal temperatures in the auxiliary building or drywell, a review will be performed to ensure that the equipment remains qualified. Requalification may be based on actual temperatures recorded during the event.
12←•
Accident conditions, defined as the failure of high-energy and moderate-energy piping, are postulated in accordance with NRC Regulatory Guide 1.46, Branch Technical Positions MEB 3-1 and APCSB 3-1.
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3.11.1.2.2 Radiation Environment

Integrated radiation environments are specified in terms of rads for gamma and beta radiation. The gamma values are based on energy deposition in tissue (rads) or exposure in air (Roentgen). However, the corresponding absorbed dose which would occur in equipment materials (e.g., carbon) when exposed to the environment would differ only slightly in magnitude. For equipment qualification testing, the equivalence of 1 rad to 1 roentgen is an appropriate assumption. The beta environment is stated in terms of a surface air dose and does not account for any shielding between the airborne or plateout activity and the material of interest. The total integrated dose equals the normal plus the accident conditions. Neutron environments are specified in terms of neutron fluence (neutrons/cm^2) for that portion of the spectrum ≥ 1 Mev.

For normal operating conditions, the radiological environments are specified as doses integrated over a 40-yr plant life for gamma and beta radiation. A plant capacity factor of 0.8 is used to develop the integrated dose for all equipment which operates in conjunction with normal reactor operation. Expected operation time over the 40-yr life of the plant is used to determine integrated doses in the vicinity of other auxiliary systems and equipment, such as radwaste and fuel handling systems. Radiation dose contributions due to abnormal conditions that are expected during the life of the plant are included in the 40-yr normal operating conditions.

For abnormal condition, radiation dose contributions due to abnormal conditions are for the MSIV isolation event resulting from a transient caused by a loss of condenser vacuum, an MSIV closure, or a turbine trip. The integrated dose contribution from this event is included in the normal 40-yr values.

For accident conditions, accident radiological doses are in addition to normal operational conditions. The accident dose contribution is determined for the single most limiting accident. Dose profiles as a function of time (t) following the accident are specified. The actual accident dose is determined based on the required operation time for the equipment following an accident. In most cases, the post-LOCA (DBA) environmental conditions will be the basis for the radiological requirements. High energy pipe breaks and fuel handling accidents are also considered. Accident integrated doses include combined dose contributions from airborne and contained sources and represent the maximum dose for the area specified.

3.11.1.2.3 Chemical Environment

Engineered Safety Feature (ESF) systems are designed to perform their safety functions in the temperature, pressure, and humidity conditions described in the Environmental Design Criteria.

Following an accident, the containment and drywell atmospheres are maintained below 4 percent (by volume) hydrogen, as discussed in USAR Section 6.2.5.
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River Bend Station does not utilize any chemical additives to the water recirculated by the ECCS during normal or accident conditions. The water in these systems is not chemically inhibited. The maximum limits for the suppression pool water are compatible with those of the primary coolant and are listed as follows:

Parameter	Reactor Water Limits Shutdown Condition	Pressure Suppression Pool Water Quality Expected	Suppression Pool Water Maximum Limits
Conductivity	≤ 10 μmho/cm @ 25°C	≤ 3 μmho/cm @ 25°C	≤ 10 μmho/cm @ 25°C
Chlorides (as Cl^-)	≤ 0.5 ppm	≤ 0.2 ppm	≤ 0.2 ppm
pH	5.3 to 8.6 @ 25°C	5.3 to 8.6 @ 25°C	5.3 to 8.6 @ 25°C
Total suspended solids		≤ 1 ppm	≤ 5 ppm

During layup, the RHR system is filled with water of the following limits:

Parameter	RHR System Maximum Limit
Conductivity	≤ 3 μmho/cm @ 25°C
Chlorides (as Cl^-)	≤ 0.05 ppm
pH	5.3 to 7.5 @ 25°C

3.11.1.2.4 Submergence

The approach to the design of RBS is to locate devices above expected submergence levels. Flood levels have been determined for the buildings, and for compartments within the buildings, for the natural phenomena and accident conditions that could cause flooding.
•→8
Equipment located inside the drywell and containment that will be submerged during normal plant operation and as a result of a design basis accident is identified in the Equipment Qualification Master List. Equipment located inside the containment and subjected to submergence conditions is qualified to perform its intended function while submerged. Equipment located inside the drywell that is subjected to submergence is not required to perform an active safety function. Evaluation has demonstrated that subsequent failure of this equipment is without significant consequences.
7←• 8←•

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Equipment located in the containment below elevation 109 ft. will be submerged for a duration of up to 7 seconds during suppression pool swell following a LOCA. This equipment is qualified to withstand the submergence conditions to which it is exposed.

For areas outside the containment, flooding analyses were performed as described in USAR Appendix 3C. These analyses demonstrate that electrical equipment required for safe shutdown of the plant is either located above the flood elevation or protective measures (e.g., curbs) are provided to prevent submergence.

Non-Class 1E electrical equipment supplied from Class 1E power sources are identified in Table 8.3-7. This equipment, as described in USAR Section 8.3.1.4.3, is electrically separated. Redundant Class 1E systems are not degraded as a result of submergence of non-Class 1E electrical equipment.

3.11.2 Qualification Tests and Analyses

Safety related electrical equipment that is located in a harsh environment has been qualified by test or other methods as permitted by 10CFR50.49 (f)[1]. Equipment type test is the preferred method of qualification.

3.11.2.1 Conformance With Regulatory Requirements

The requirements of General Design Criteria (GDC) 1, 4, 23, and 50 of Appendix A to 10CFR50 and Criterion III of Appendix B to 10CFR50 are met as outlined below:

GDC 1 of 10CFR50, Appendix A, requirements are achieved by incorporating performance, design, construction, and testing requirements into equipment specifications and by the establishment of a system of reviews to ensure conformance with these specified requirements.

Appropriate auditable records are maintained in a permanent file. Refer to Chapter 17 for a further definition of how Criterion III of Appendix B to 10CFR50 is met.

GDC 4 requirements are met for harsh environment equipment by designing and qualifying the equipment for satisfactory operation and proper safety function performance during normal, abnormal, test, and DBA environments.

Because components are procured to withstand the environments resulting from both abnormal events and accidents, the RBS safety related electric equipment meets the requirements of GDC 23.

GDC 50 requirements are achieved by analysis and testing of pressure boundary components to ensure containment integrity.

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A discussion of compliance with Regulatory Guides is provided in Section 1.8.

3.11.2.2 Qualification Methodology for Safety Related Equipment

3.11.2.2.1 Harsh Environment BOP Electrical Equipment

The methodology for the RBS equipment qualification program is in accordance with the guidelines provided in NUREG-0588 for Category I plants, and in compliance with the regulation of 10CFR50.49. The methodology consists of developing the temperature, pressure, humidity, and radiation dose levels for normal, abnormal, and accident conditions. Post-accident operability time is developed to assure that the equipment will be qualified to maintain a safety function during a post-accident event.

These requirements are included in the procurement specification for the electrical equipment important to safety. The specification mandates that the qualification will be accomplished in accordance with IEEE 323-1974 and in accordance with the quality assurance program referenced in 10CFR50 Appendix B.

Based on these requirements the equipment manufacturer develops an equipment qualification program.

The environmental qualification documents for the equipment are obtained for engineering evaluation from equipment vendors, equipment manufacturers and/or testing facilities. These documents, in the form of qualification procedures, reports, and supplementary information, are evaluated in accordance with NUREG-0588 and IEEE 323-1974. Review of these documents includes assurance that they are technically adequate and conform to the environmental qualification requirements of the applicable emergency conditions, operability times, and service conditions.
•→13
Each reviewed vendor qualification document is referenced in the Equipment Qualification Assessment Report (EQAR).
7←•
3.11.2.2.1.1 Aging

The effects of age are documented as part of the qualification program.
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Arrhenius aging methodology is the preferred method for evaluating equipment aging and in general is used as a basis for determining qualified life. When other methods are used, appropriate justification is provided. The thermal aging methodology used for qualifying BOP safety related devices is described in the qualification test reports referenced in the EQAR.
8←• 13←•

•→7
Prior to simulating seismic and DBA events, equipment or its age-sensitive components were preconditioned to their end-of-qualified life condition.

If it was known that aging improved performance capability, new or partially aged devices were used in testing.

Advanced life conditioning was accomplished by applying an appropriate combination of operational and environmental cycling to simulate the expected service life listed in the equipment specification and by subjecting the device to physical and chemical stresses that are known to degrade the device. Normal cycling of in-plant conditions was performed in any combination aging procedure.

In the case where accelerated aging was used, the procedure employed considered the expected application and design life of the device being tested.

Synergistic effects were considered when these effects were known to have a significant effect on equipment performance.
•→13 •→8
Where required, a maintenance or replacement schedule consistent with qualified life is provided as part of the support documentation and is referenced in the EQAR.
8←• 13←•
When type testing was selected as the qualification method, the type test was run on the device(s) in a specified sequence that was set down as part of the written test procedure. All sequential testing was performed on the same unit(s) including aging. The sequence recommended in IEEE 323-1974, paragraph 6.3.2, is used. Where an alternate sequence can be justified technically; this justification can be documented in the qualification reports. The test specimen has been subjected to all normal manufacturing and QA procedures and is representative of the device supplied.

3.11.2.2.1.2 Margin

Margins are not included in the parameters given in the equipment specifications. However, the specification does include an insert that requires that margin be added to comply with IEEE 323-1974 requirements.

Qualification type test results were reviewed to verify that adequate margin exists between the most severe specified service conditions for the equipment and the conditions used in the type testing. Margins are in addition to any conservatism applied during the derivation of local environmental conditions of the equipment. Margin accounts for the production variations of equipment and inaccuracies in test instrumentation. Increased levels of testing, number of test cycles, and test duration are among the methods used for ensuring adequate margin.
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Some equipment is required by the design to perform its safety function only within the first 10 hours of an accident. For this equipment in general, a time margin of at least 1 hour in excess of the time assumed in the accident analysis was used.
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3.11.2.2.1.3 Dose Rate and Synergistic Effects

Qualification for radiation was based on the calculated total integrated dose. Safety related electrical equipment qualified for use in a nuclear radiation environment was exposed to radiation which simulated the conservatively calculated integrated dose (normal and accident) that the equipment is expected to withstand prior to completion of its intended safety function. In general, a gamma radiation source, typically CO-60, is used to simulate expected radiation exposure. Where beta and gamma radiation exposure is expected, beta radiation is taken into account either during simulated exposure (directly or as a gamma equivalent) or during evaluation of the results. Reduction in the total beta dose was allowed after considering appropriate shielding factors. If the beta radiation dose contribution to the equipment or component was calculated to be less than 10 percent of the total gamma radiation dose to which the equipment or component has been qualified, then the equipment was considered qualified for the beta and gamma radiation environment.

Dose rate effects were considered when these effects were known to have a significant effect on equipment performance.

The dose rate, energy spectrum, or particle type was addressed to arrive at a gamma equivalent total dose to which the equipment must be exposed. Actual testing using dose rate, energy spectrum, or particle type was not considered.

Therefore, synergistic effects involving dose rate are not addressed. However; where synergistic effects of radiation and temperature were identified prior to the initiation of the qualification, they are included in the program.
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3.11.2.2.2 Deleted
•←8

3.11.2.2.3 Harsh Environment NSSS Electrical Equipment

The approach taken by General Electric to environmentally qualify safety related equipment within the NSSS Scope of Supply for RBS to a level consistent with NUREG-0588 is described in GE Licensing Topical Report NEDE-24326-1-P[2]. This report has been approved by the NRC. The methodology is consistent with applicable regulations (10CFR50, Appendix A); applicable Regulatory Guides; and with applicable consensus National Standards (ANSI and IEEE). The work performed under this guidance is controlled in a manner consistent with the commitments contained in the NRC-approved GE Licensing Topical Report on Quality Assurance.
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The approach to qualification described in NEDE-24326-1-P[2] is predicated on type testing being the preferred approach. Depending upon either the unique characteristics of the specific devices or on the availability of other sources of qualification data, other approaches such as partial type test with justification by analysis, operating experience, analysis or combination of the above mentioned approaches may be used. For any of these approaches the eventual approach used is justified in the accompanying qualification report. This justification is based on the demonstrated ability of the product to meet its intended safety function.

•→12
Where type testing is performed, the approach taken is to assure the device is functional under normal conditions as well as under extremes of such conditions; next the devices are aged to an end-of-qualified-life condition, next the device is subjected to dynamic simulation; next the device is subject to design basis event conditions and post design basis event conditions; and lastly the device is inspected for failures which may not have been apparent during the operational testing which occurs during each exposure to an environmental extreme. When a product is tested, where practical, the interface associated with the product is included in the test. The specific sequences of environments applied during the testing are determined, using engineering judgment, to best select the sequence to which the product would be subjected during actual installation. Furthermore, where synergisms between environments are known, these effects are taken into consideration during the planning and conducting of the test. All tests that are conducted include adequate margins as required in NUREG-0588. NRC personnel from the Region IV Office of Inspection Enforcement have routinely audited General Electric's environmental qualification efforts and have found no indication that what was being performed for RBS did not demonstrate adequate qualification.

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Following the completion of the tests all of the associated documentation that led to the test and was generated during the test is formally assembled into a qualification report. The report is available for NRC audit.

For devices not qualified by test (e.g., devices classified as safety related solely because they perform a pressure boundary function; devices that perform their safety function prior to the onset of harsh environments in which they do not contribute to the mitigation of the event after performance of the intended safety function, etc.) qualification reports are also prepared demonstrating the adequacy of their qualification. As with devices qualified by test, these qualification reports are in an auditable form. The last step of qualification is to ensure that the device tested is similar to the device installed in the field. Therefore, before full qualification can be assured, there is a verification of the similarity between the tested device and the installed device.

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3.11.2.2.4 Deleted

3.11.3 QUALIFICATION TEST RESULTS

3.11.3.1 Master Equipment List

The master list of all equipment within the scope of the environmental qualification program is controlled at RBS using approved procedures.
7←• 8←•
The electrical equipment master list contains all equipment requiring qualification in accordance with 10CFR50.49.

1. Equipment requiring qualification under 10CFR50.49b(1) includes all safety-related equipment required to perform its safety function in a harsh environment. Also included is equipment in any directly mechanically connected auxiliary systems with electrical components (e.g., cooling water or lubricating systems) which support the safety function of other safety-related equipment.

2. Equipment requiring qualification under 10CFR50.49b(2) includes all equipment electrically connected directly into the control or power circuitry of the safety-related equipment whose failure under postulated environmental conditions could adversely affect the safety function of other equipment. The identification of this equipment utilized, among other measures, review of applicable elementary wiring diagrams.

3. Equipment requiring qualification under 10CFR50.49b(3) includes all post-accident monitoring equipment in accordance with the RBS position to Regulatory Guide 1.97.

•→8
For each device, the Master List provides a summary of the key elements of the Environmental Qualification Program. Table 3.11-10 and Figure 3.11-6 contain the heading for the Master List, with a description of each entry. The first three characters of the device indicate the major system in which the device is used. The subsequent characters are used to further segregate the devices by specific type and number.
•→13
3.11.3.2 Deleted

3.11.3.3 Equipment Qualification Assessment Report (EQAR)

The EQAR provides a text based discussion of the basis for considering each Class 1E component qualified.
7←• 8←• 13←•

•→13 •→8 •→7
Section 2 of the EQAR provides the summary level information that had been provided in the SCEW sheet.
13←•
The EQAR provides a logical discussion of each facet of the qualification process. It elucidates a comparison between the required attributes necessary for qualification of individual components and the test, analysis, or combination thereof, that demonstrates the capability of the equipment to perform its required safety function.

The EQAR identifies the reference documents used in development of the qualification basis. Documents that are referenced may include test reports, equipment specifications, engineering calculations, failure modes and effects analyses, and supplemental analyses. The EQAR contains references to all of the supportive environmental qualification documents that demonstrate that the equipment is qualified to perform its safety function in the postulated environmental conditions. EQARs are controlled and maintained at RBS using approved procedures.
•←8
3.11.4 Loss of Ventilation

The following design features preclude the possibility of a total system failure for ventilation systems serving areas where equipment required to function during and following a DBA is located:

1. All HVAC systems serving these equipment areas are designed to Seismic Category I requirements (see Section 9.4).

2. Sufficient redundancy in equipment and power supplies is provided so that no single active component failure can result in loss of HVAC system function.

3. Redundant HVAC systems are connected to separate and independent onsite standby power supplies to assure system operation upon loss of offsite power (see Section 8.3).

4. Failure modes for isolation valves and dampers are described in Section 9.4. Valves or dampers required for operation after postulated accidents fail in the safe position.

5. Equipment outside the containment building required to operate following a LOCA or a high-energy pipe break is so located that it is not exposed to resultant post-accident ambient conditions or is designed to withstand these severe conditions.

6. Instrumentation and controls which incorporated audible and visual alarms enable the operator to continuously monitor the HVAC systems' performances. In the event of system malfunction, the operator has the capability to switch manually to the HVAC standby equipment.

7←•

•→7
Based on the above features and the detailed HVAC systems' evaluation in Section 9.4, only partial loss of the ventilation or air-conditioning system could occur in areas where equipment required to function during and following a DBA is located. This loss would not adversely affect the availability of the safety-related equipment to function during and following a DBA.

Equipment is qualified for the limiting environmental service conditions for which it must function assuming loss of non-Class 1E HVAC systems.

3.11.5 Estimated Chemical and Radiation Environment

The chemical composition and resulting pH to which safety-related equipment is exposed during normal operation and design basis accident conditions is reported in Section 3.11.1.2.3.

The methodology for developing design source terms for the radiation environment for normal operation and design basis accident conditions is described in Section 3.11.1.2.2. The design basis accident source term is based on the assumptions of Section 1.4 of NUREG-0588[3] and Regulatory Guide 1.89[4]. River Bend has implemented Alternate Source Term (AST), per Regulatory Guide 1.183, which revised the design basis source term from TID-14844 to NUREG-1465. Equipment Qualification doses were evaluated using Regulatory Guide 1.183 assumptions, and updated as necessary.

The design basis source term for the release of hydrogen following an accident complies with the assumptions of Regulatory Guide 1.7[5]. The post-LOCA hydrogen concentration of the containment and drywell atmosphere is maintained below 4 percent (by volume), as discussed in Section 6.2.5.

3.11.6 MAINTENANCE AND SURVEILLANCE PROGRAM

The RBS Maintenance and Surveillance Program was developed to maintain Category I structures and safety-related systems and components at the quality required to perform their intended functions and withstand design basis events. The program ensures filing of documentation in an auditable and retrievable manner. The objectives of the maintenance and surveillance program, as related to qualification, are accomplished by reviewing qualification data, vendor manuals, NRC correspondence, S.E. reports. etc. This review identifies:

1. Components with a qualified life of less than 40 years.
2. Components requiring part replacement to continue qualification.
3. Routine preventive maintenance or surveillance requirements for all equipment, regardless of qualified life.
4. Special interfaces and configuration.

7←•

•→8 •→7

The above requirements are incorporated into the existing preventive maintenance program which provides a computerized data base that includes performance frequency, assigned responsibility, applicable procedures, instructions, and requirements. The program allows access to equipment and maintenance history. Additionally, it provides a means to identify material requirements for better inventory control. Based on a work order issued as a result of the preventive maintenance schedule, the appropriate department completes the required maintenance. The documented results are sent to the appropriate discipline for technical evaluation and possible alteration of maintenance and surveillance requirements. The initially developed maintenance and surveillance program is modified during plant life if additional information such as corrective maintenance frequency, surveillance testing, and industry/operating experience identified any unanticipated degradation trends. Controls are established for such activities as procurement, storage and station modification to prevent compromising qualification. Implementation of RBS procedures accomplishes the following:

8←•

1. Procurement of qualified components for the intended safety function.

2. Procurement of qualified spare parts and replacement components.

3. Assurance that the required maintenance during storage is properly performed and documented.

4. Control of station modifications to prevent compromising qualification.

5. Review of station modification packages and procurement documents to address equipment qualification concerns.

The RBS Quality Assurance program conforms to 10CFR50, Appendix B, and requires inspections, verifications, and audits to ensure that maintenance requirements are properly implemented in a timely manner. The RBS maintenance and surveillance program is consistent with NRC requirements and ensures that safety-related equipment is maintained and monitored under controls to assure continued qualification throughout the life of the plant.

7←•

•→7
References - 3.11

1. Title 10, Code of Federal Regulations, Paragraph 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants. Federal Register Vol. 48, No. 15, January 21, 1983.

2. Shirley, N.C., et al, General Electric Qualification Program Licensing Topical Report NEDE-24326-1-P, January 1983.

3. A. J. Szukiewicz et al. Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, NUREG-0588, Rev. 1, July 1981.

4. NRC Regulatory Guide 1.89, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants. Proposed Revision 1, November 1983.

5. NRC Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following A Loss-of-Coolant Accident, Rev. 2, November 1978.

7←•

6. NRC Regulatory Guide 1.183, Alternate Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors.

APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

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APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

INTRODUCTION

The following computer programs are used for the stress analysis of the steel containment system:

1. SHELL 1
2. ASAAS
3. TAC2D

The following computer programs are used in dynamic and static analysis of Seismic Category I structures:

1. MAT 6
2. STRUDL
3. Time History (TIMHIS6) Program
4. SHELL 1
5. WILSON-GHOSH

•→10

6. MICAS PLUS
7. ME-323 (PC PREPS)

10←•

The following computer programs are used in the analysis of Seismic Category I equipment and components:

1. ME 121
2. DINASAW
3. LIMITA II
4. LIMITA III
5. STARDYNE
6. BIJLAARD
7. MISSILE
8. PSPECTRA

The following computer programs are used for the analysis of Seismic Category I piping systems, including pipe supports:

1. NUPIPE
2. TRHEAT
3. HTLOAD
4. WATAIR
5. PSPECTRA
6. STEHAM
7. WATHAM
8. PITRUST
9. PILUG
10. ANSYS

For each computer program, there is a brief description of the program's theoretical basis, the assumptions and references used in the program, the extent of its application, and a summary of manual or comparison qualification.

The computer programs used in the analyses of Seismic Category I code and non-code items are documented in accordance with the requirements of existing SWEC engineering department procedures. In meeting these requirements, compliance to the following has been demonstrated.

1. Documentation of computer programs includes the author, source, dated version, and facility.

2. Documentation of computer programs includes the extent and limitations of its application.

3. Test problems demonstrate the qualification of each program.

3A.1 SHELL 1

This is a finite-difference stress analysis computer code. It can be used to determine the forces, moments, shears, displacements, rotations, and stresses in a thin shell of revolution subject to arbitrary loads expanded in Fourier series of up to 150 terms. Single-layer shells with up to 30 simply connected branches may be analyzed. Poisson's ratio may change at discontinuity points, and Young's modulus and the thermal coefficient of expansion may be different at each point. The allowed types of loading include elastic restraints, pressures in three orthogonal directions, temperature changes which may have a gradient through the shell thickness, and simplified input for weight of the shell or earthquake forces.

The equilibrium equations for a thin shell are based on the linear theory of Sanders[2]. Sander's equations are expanded and modified slightly to handle a broader range of problems. All pertinent load, stress, and deformation variables are expanded into Fourier series. The individual Fourier components of stress and deflection are found separately by solution of the finite-difference forms of the appropriate differential equations. The algorithm used to solve these equations is a minor modification of the Gaussian elimination method.

<u>Sample Problem: Thin Wall Cylinder</u>

A long thin-walled circular cylinder is subjected to a constant internal pressure distribution. A solution of this problem may be obtained by the use of Reference 1.

The pertinent parameters of the cylinder are:

<u>Dimension and Properties</u>

$R = 25$ in

$\ell = 20$ in
$t = 0.5$ in

$E = 28 \times 10^6$ psi
$\upsilon = 0.3$

<u>Loading and Boundary Conditions</u>

$$F_R\Big|_{z=0} = M\Big|_{z=0} = \delta z\Big|_{z=0} = 0$$

$$F_R\Big|_{z=1} = M\Big|_{z=1} = F_Z\Big|_{z=1} = 0$$

$P_i = 75$ psi

The following solution can be verified by consulting Reference 1:

$$\delta_R = \frac{pR^2}{Et} \qquad (3A.1\text{-}1)$$

$$\sigma_\theta = \frac{pR}{t} \qquad (3A.1\text{-}2)$$

The cylinder is idealized by 10 elements (Fig. 3A.1-1).

In Table 3A.1-1, the computer results are compared with the results obtained from Equations 3A.1-1 and 3A.1-2. The results compare very favorably. Therefore, this problem serves to demonstrate the accuracy of SHELL 1.

References - Section 3A.1

1. Roark, R. J. Formulas for Stress and Strain. McGraw-Hill Book Company, Fourth edition, New York, NY, 1965.

2. Sanders, Jr., J. L. An Improved First Approximation Theory for Thin Shells. NASA Technical Report R-24.

TABLE 3A.1-1

EXACT AND COMPUTER STRESSES FOR THIN-WALL CYLINDER

SHELL 1 COMPUTER PROGRAM

Variable	Exact	SHELL 1
δ_R	3.348×10^{-3} in	3.342×10^{-3} in
σ_θ	3,750 psi	3,750 psi

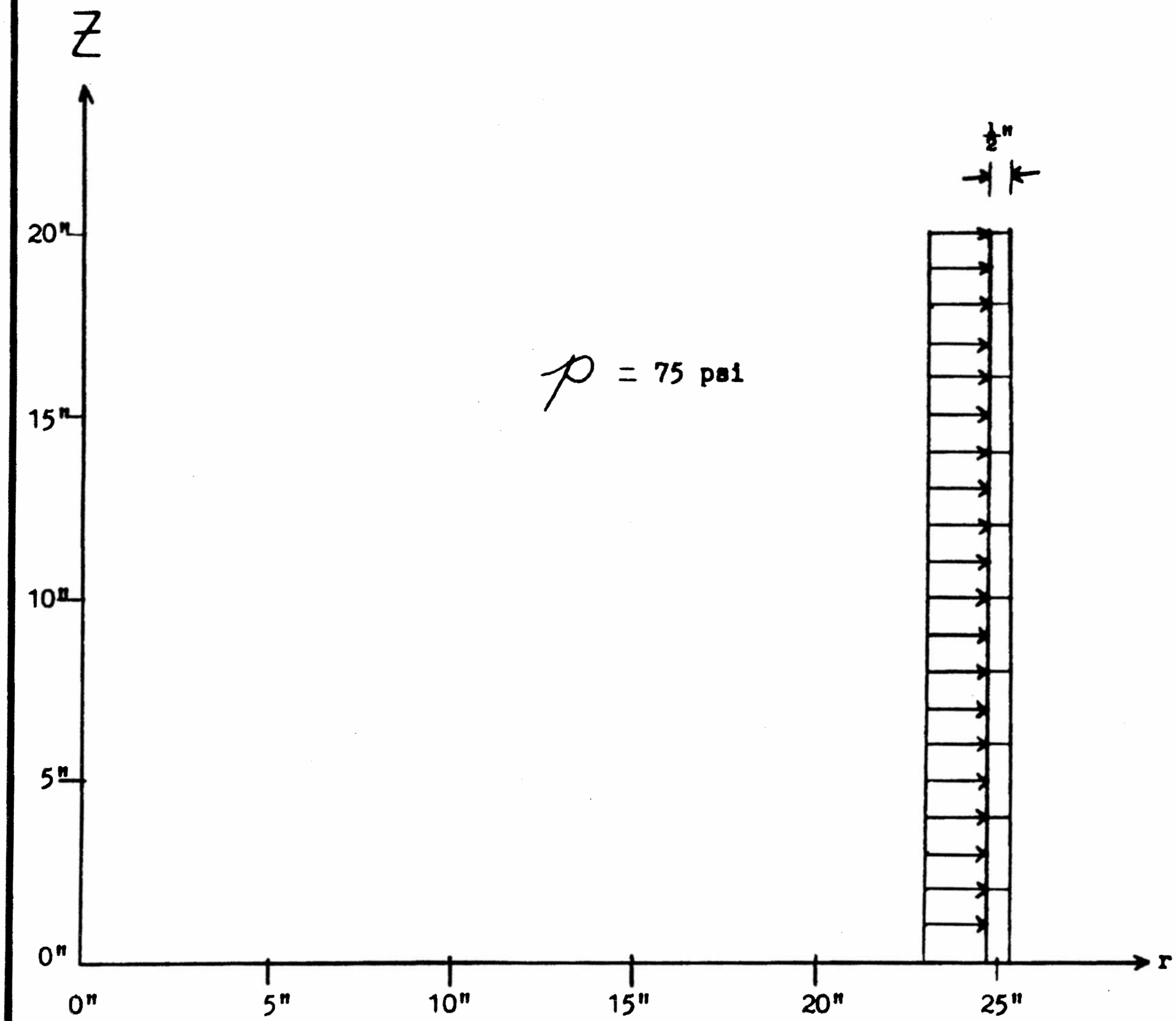

FIGURE 3A.1-1

TEN ELEMENT IDEALIZATION
OF THIN WALL CYLINDER

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3A.2 ASAAS

This is a finite element computer code. It can be used to determine stresses and displacements in arbitrary axisymmetric solids, including problems involving asymmetric mechanical and thermal loads and asymmetric temperature-dependent mechanical properties. All dependent variables, including the mechanical properties, are input by Fourier series expansions of the circumferential coordinate. The mechanical loads can be surface pressures, surface shears, and nodal point forces.

The explicit stiffness relations for the axisymmetric solid ring elements of triangular cross section are based on the classical theorem of potential energy and the assumption that within any element the displacement variation in the R-Z plane is linear. All dependent variables, including the material properties, are expanded into Fourier series. The harmonics are coupled, and all the equilibrium equations are solved simultaneously. The algorithm used to solve the equations is a block-modified, square root, Cholesky method with iterative refinement.

ASAAS is a recognized program in the public domain[1].

<u>Sample Problem: Harmonic Axisymmetric Plane Strain</u>

An infinitely long, solid, circular cylinder is subjected to cos θ and cos ;θ pressure distributions. A closed-form solution of this problem may be obtained by the use of Reference 2.

The pertinent parameters of the cylinder are:

Dimension and Properties	Loading and Boundary Conditions
$r_o = a$	$\sigma_r = P_o(\cos\theta + \cos 2\theta)$
$l = a$	$\sigma_{r\theta} = P_o \sin\theta$
$E = 10 \times 10^6$ psi	$u_z = 0$
$\upsilon = 0.25$	$u_r \big\|_{r=0} = 0$
$a = 1$ in	$P_o = 10,000$ psi

The following solution can be verified by consulting Reference 2.

$$\sigma_r = p_o \left(\frac{\gamma}{a} \cos \theta + \cos 2\theta\right) \qquad (3A.2\text{-}1)$$

$$\sigma_\theta = p_o \left[3\frac{\gamma}{a} \cos \theta + \frac{2r^2 - a^2}{a^2} \cos 2\theta\right] \qquad (3A.2\text{-}2)$$

$$\sigma_{r\theta} = p_o \left[\frac{\gamma}{a} \sin\theta + \frac{r^2 - a^2}{a^2} \sin 2\theta\right] \qquad (3A.2\text{-}3)$$

$$u_r = p_o \left[\frac{(1-4\nu)\ (1+\nu) r^2}{2Ea} \cos \theta + \frac{1+\nu}{E}\left(r - \frac{2\nu r^3}{3a^2}\right) \cos 2\theta\right] \qquad (3A.2\text{-}4)$$

$$u_\theta = p_o \left\{\frac{(5-4\nu)\ (1+\nu) r^2}{2Ea} \sin \theta + \frac{1+\nu}{E}\left[\left(1 - \frac{2\nu}{3}\right)\frac{r^3}{a^2} - r\right] \sin 2\theta\right\} \qquad (3A.2\text{-}5)$$

The cylinder is idealized by 16 elements (Fig. 3A.2-1). Computer results are depicted in Fig. 3A.2-2, along with the exact results obtained from Equations 3A.2-4 and 3A.2-5. The computer results are very close to the exact results. Therefore, this problem serves to verify the accuracy of ASAAS for mechanical loading problems where material properties are not variable.

References - Section 3A.2

1. Crose, J. G., ASAAS Asymmetric Stress Analysis of Axisymmetric Solids with Orthotropic Temperature-Dependent Material Properties That Can Vary Circumferentially. Air Force Report No. SAMSO-TR-71-297, Aerospace Report No. TR-0172 (S2816-15)-1, December 29, 1971.

2. Love, A. E. H. A Treatise on the Mathematical Theory of Elasticity. Dover Publications, New York, NY, 1944.

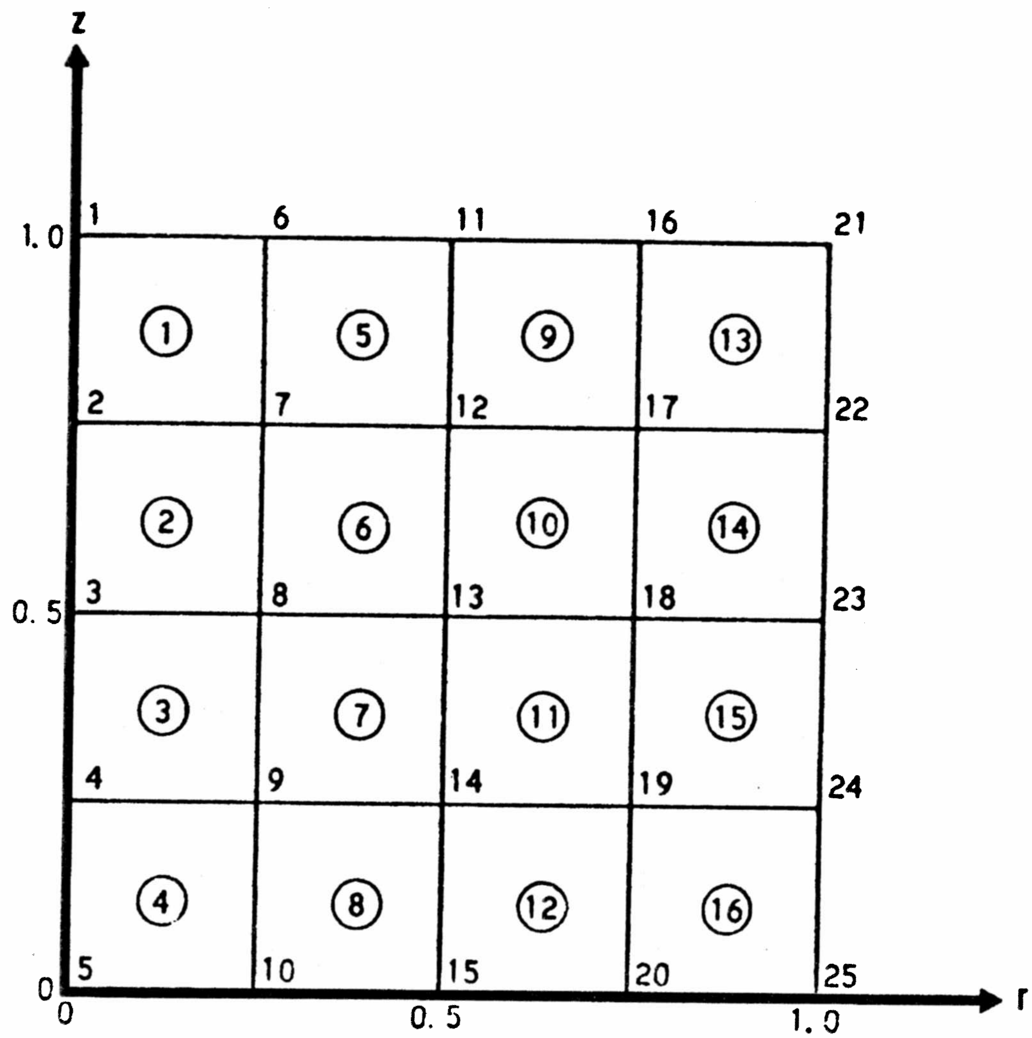

FIGURE 3A.2-1

ELEMENT PLOT

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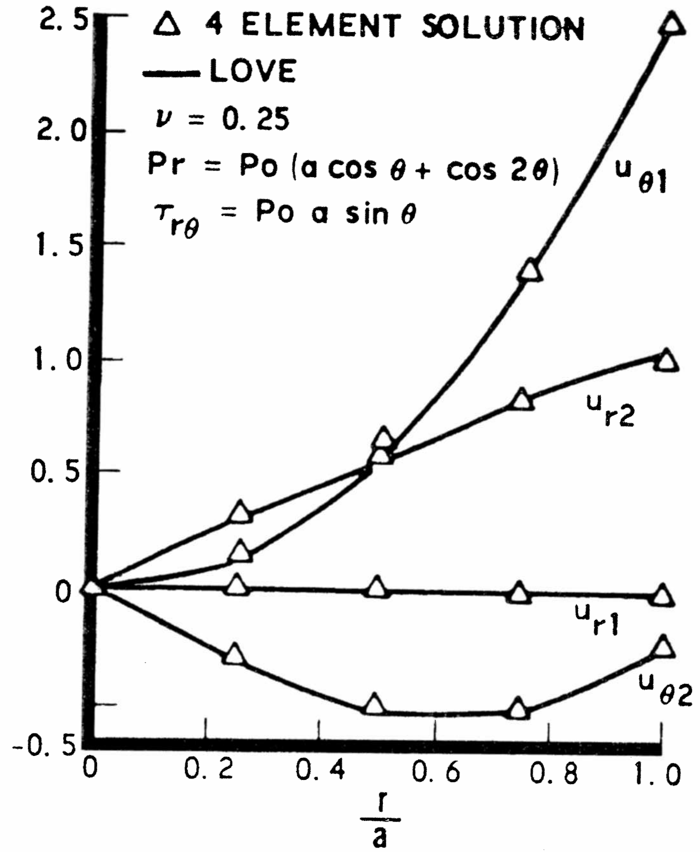

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FIGURE 3A.2-2

HARMONIC AXISYMMETRIC
PLANE STRAIN

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3A.3 TAC2D

TAC2D is a general-purpose, two-dimensional heat transfer computer code. It is a finite difference computer code. It can be used to determine steady-state and transient temperatures in two-dimensional problems. The configuration of the body to be analyzed is described in the rectanglar, cylindrical, or circular (polar) coordinate system by orthogonal lines of constant coordinate called grid lines. These grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite-difference equation is formulated for each nodal point in terms of its capacitance, heat generation, and heat flow paths to neighboring nodal points. The equations for all the nodal points are assembled and solved using an implicit alternating gradient algorithm. TAC2D is a recognized program in the public domain[1].

<u>Sample Problem</u>

A sample problem is presented to compare the results from TAC2D with an analytical solution. The objective is to show that the TAC2D program yields the correct solution.

<u>Problem Description</u>

The problem is to determine the transient temperature distribution in a right circular cylinder which is initially at temperature T_1. At time $t = 0$, the temperature at the surface is instantaneously changed to T_2 and maintained at that value. Mathematically, the problem is defined by the following equations:

$$\frac{1}{r} \frac{\delta}{\delta r} \; r \quad \frac{\delta^2 T}{\delta Z^2} = \frac{1}{k} \quad \frac{\delta}{\delta t} \qquad \delta \; ; \quad 0 \le r \le R \qquad (3A.3\text{-}1)$$

$$T(r,z,0) = T_1 \qquad (3A.3\text{-}2)$$

$$T(R,z,t) = T_2 \qquad (3A.3\text{-}3)$$

$$T(r,\pm L/2,t) = T_2 \qquad (3A.3\text{-}4)$$

Where t is the time, r is the radius, z the axial coordinate, R the outside radius of the cylinder, L the length of the cylinder, and k the diffusivity.

Further,

$$\kappa = \frac{k}{\rho c} \qquad (3A.3-5)$$

where:

k is the thermal conductivity, ρ the density, and c the specific heat capacity. For the specific problem analyzed, the following numerical values were used:

R = 12.0 in

L = 48.0 in

k = 20.0 Btu/hr-ft-°F

ρc = 40.0 Btu/cu ft-°F

T_1 = 0.0 °F

T_2 = 1,000.0 °F

Analytical Solution

It may be shown[2], that the solution is:

$$\frac{T - T_1}{T_2 - T_1} = 1 - f\,(z,t)g(r,t) \qquad (3A.3-6)$$

$$f(z,t) = \frac{4}{\pi}\sum_{n=1}^{\infty} \frac{(-1)^{n-1}}{(2n-1)} \cos\left[(2n-1)\frac{\pi z}{L}\right] e^{-\kappa\left(\frac{(2n-1)\pi}{L}\right)^2 t} \qquad (3A.3-7)$$

$$g(r,t) = 2\sum_{n=1}^{\infty} \frac{J_0\left(\frac{r}{R}\right)\gamma_m}{\gamma_m J_1(\gamma_m)} e^{-\kappa\gamma_m^2 t} \qquad (3A-.3-8)$$

where the γm are the roots of (3A.3-9)

$J_o(\gamma m) = 0$

The roots Υ_m of Equation 3A.3-9 and the functions J_1 are tabulated in Reference 3 and need not be computed.

From the definition of the problem, there is symmetry about the geometric center of the cylinder and the origin of the coordinate system taken at that point, as is reflected in the boundary conditions, Equations 3A.3-3 and 3A.3-4.

Numerical Solution with TAC2D

A cross section of the problem model for TAC2D is shown on Fig. 3A.3-1. The model extends only to the axial midplane of the cylinder, where an adiabatic boundary may be specified by virtue of the symmetry condition described previously. The solid material is represented by one material block. The boundary conditions on the four external boundaries are described by Coolants 1 through 4 (specifically, Coolant Blocks 1 through 4). The material and coolant thermal parameters, as specified by the input functions, are given in Table 3A.3-1. All coolants have the standard specific heat of 1.0 Btu/lb-°F. Coolants 1 and 2, which represent the adiabatic external boundaries, have the standard heat transfer coefficient of 10^{-6}Btu/hr-sq ft-°F and the standard flow rate of 10^{6} lb/hr.

Comparison of the TAC2D Solution with the Analytical Solution

A comparison of the output from the code with the series solution is shown on Fig. 3A.3-2. The temperature versus time function is plotted at three representative points within the cylinder. It can be seen that the results from TAC2D are almost identical to the series solution results. The maximum difference between the two sets of results is about 2°F out of a mean magnitude of 100°F.

TABLE 3a.3-1

INPUT THERMAL PARAMETER FUNCTIONS FOR TAC2D
SAMPLE PROBLEM

References - Section 3A.3

1. Peterson, J. F.
TAC2D-A General Purpose Two-Dimensional Heat Transfer Computer Code. AEC Research and Development Report, Gulf General Atomic Inc., GA-9262, September 6, 1969.

2. Carlslaw, H. S. and Jaeger, J. C.
Conduction of Heat in Solids. Oxford at the Clarendon Press, 1959, p 227.

3. Jahnke, E. and Ende, F.
Tables of Functions. Dover Publications, Fourth Edition, 1945.

TABLE 3a.3-1

INPUT THERMAL PARAMETER FUNCTIONS FOR TAC2D
SAMPLE PROBLEM

```
C     MATERIAL THERMAL PARAMETERS

            SPEC1(X)=40.0
            RCON1(X)=20.0
            ACON1(X)=20.0


      C     COOLANT THERMAL PARAMETERS

            H3A(X)=1.0E+08
            FLO3A(X)=1.0E+08
            TIN3A(X)=1460.0
            H4A(X)=1.0E+08
            FLO4A(X)=1.0 E+08
            TIN4A(X)=1460.0
```

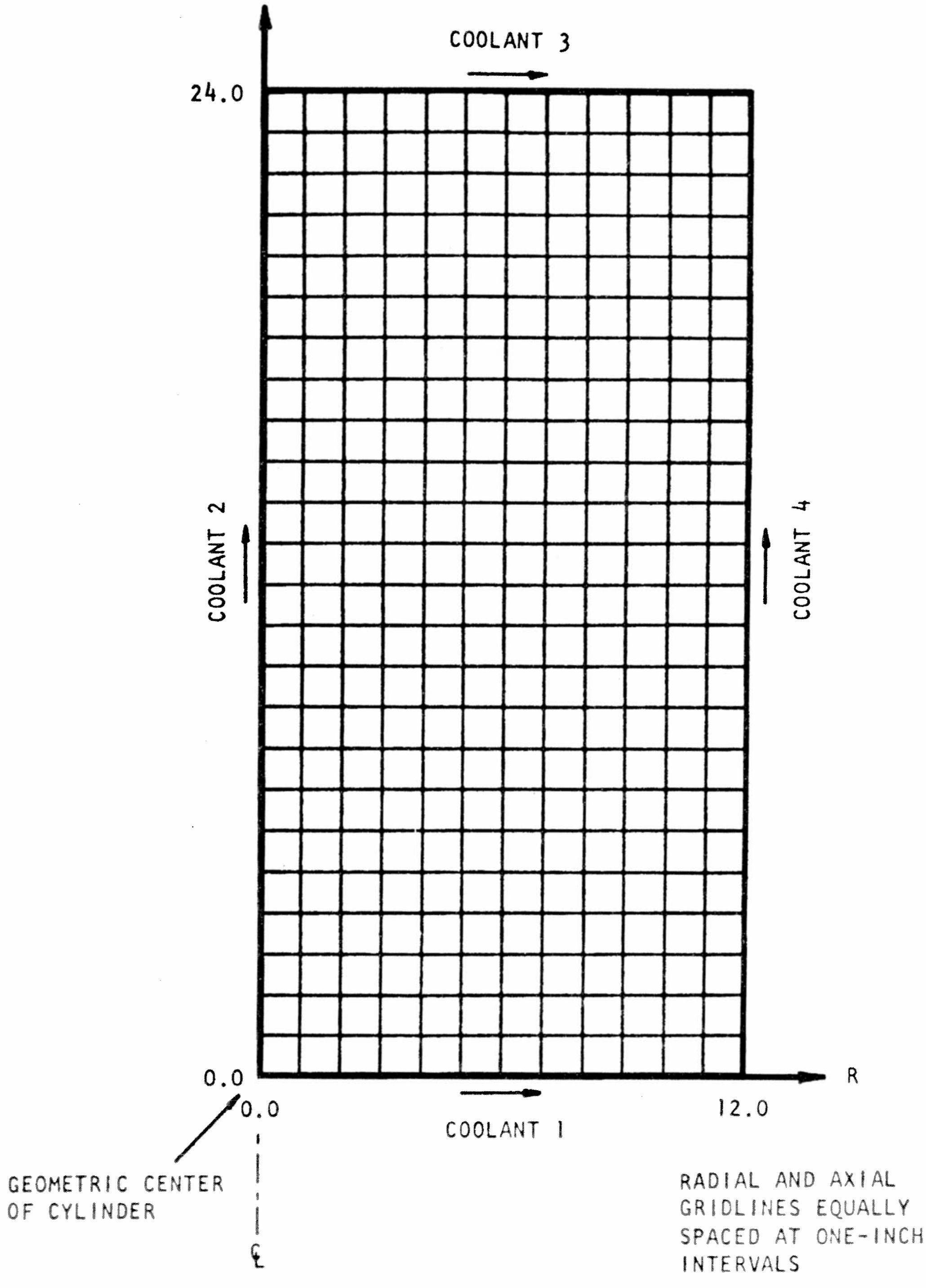

FIGURE 3A.3-1

TAC2D SAMPLE PROBLEM,
THERMAL MODEL

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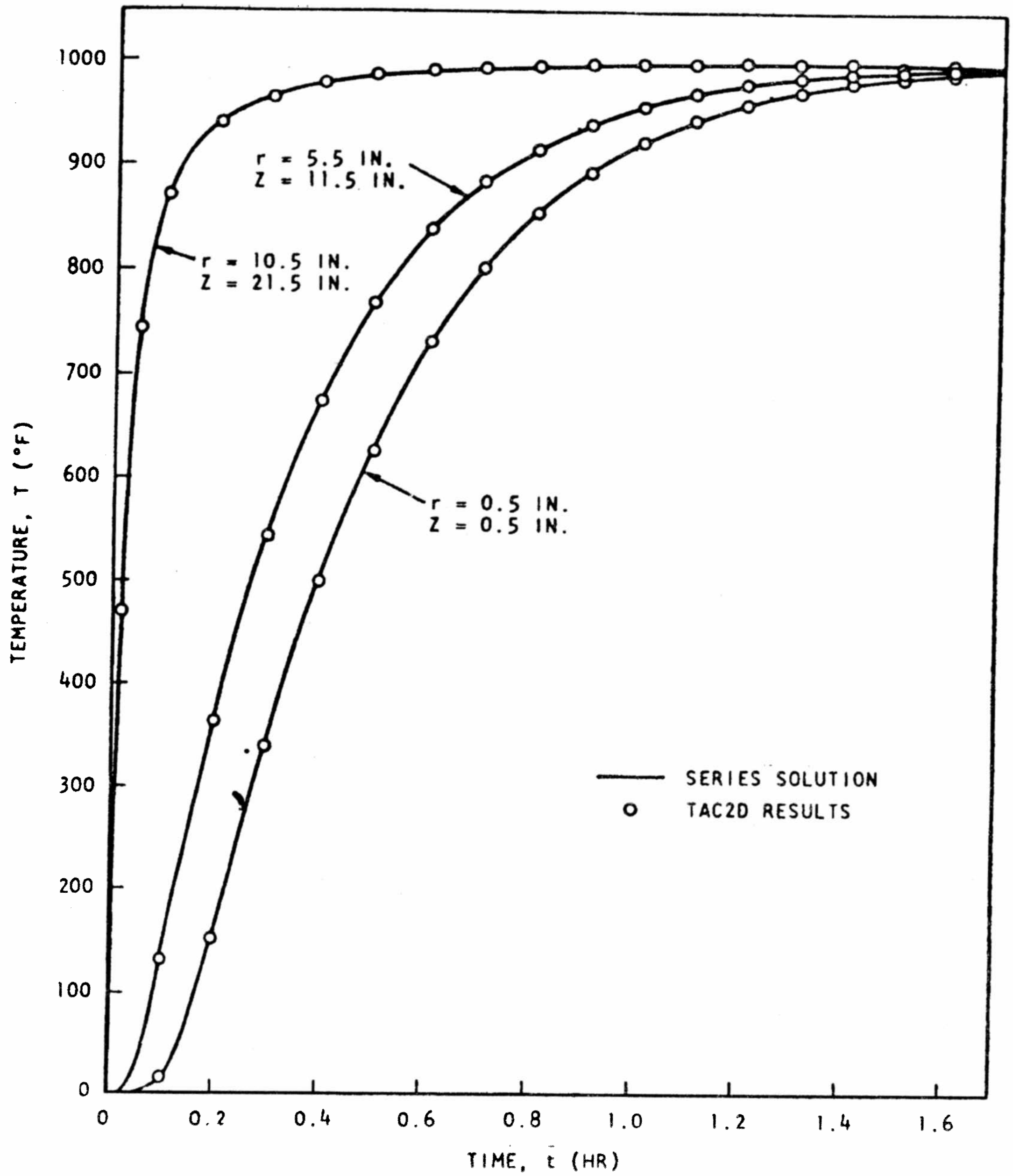

FIGURE 3A.3-2

TRANSIENT TEMPERATURE IN A RIGHT CIRCULAR CYLINDER COMPARISON OF TAC2D RESULTS WITH SERIES SOLUTION

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3A.4 MAT 6

This program analyzes a symmetrically loaded circular plate on an elastic foundation and maintains compatibility between: 1) the plate (foundation mat) and the subgrade, and 2) the plate and the circular walls supported thereon. The program computes the discontinuity effects at the interface of the mat and circular walls and includes those effects in the analysis.

The general method is described by Zhemochkin[1].

The displacements and stresses of the subgrade are derived from a Boussinesq solution for a circular plate supported on a semi-infinite, elastic half space[2]. The elastic behavior of the mat and circular wall (or walls) is basically described by Timoshenko and Woinowsky-Krieger[3] .

This program is used to analyze the containment foundation mat and to provide the contact pressure and the discontinuity forces at the junction of the mat and superstructure (i.e., the containment wall, crane wall, and reactor support wall).

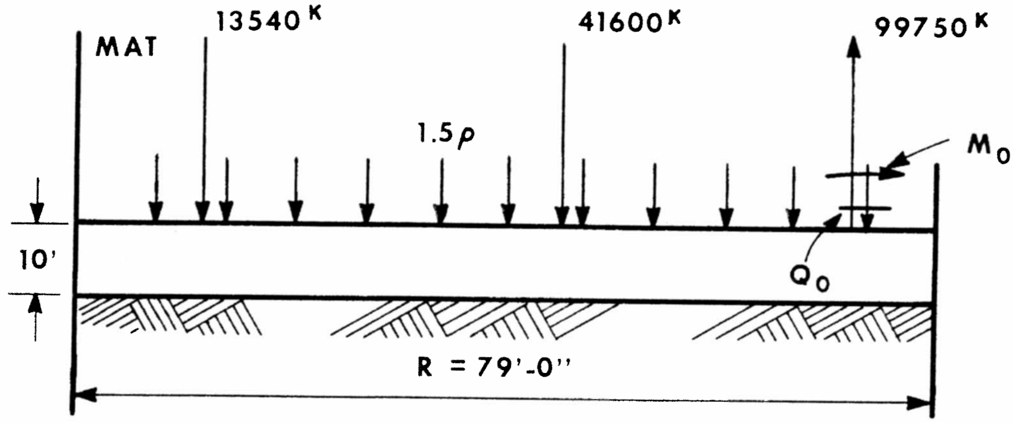
Included are plots (Fig 3A.4-1 through 3A.4-4) of radial and tangential bending moments and the radial shear in the mat for a MAT 6 solution versus a SHELL 1 solution (Section 3A.1). Also shown are the discontinuity forces at the interface of the mat and the containment wall.

References - Section 3A.4

1. Zhemochkin, B. N. Practical Methods for Analysis of Beams and Plates on Elastic Foundations. 2nd Edition, Moscow, l962.

2. Timoshenko, S. P. and Goodier, J. N. Theory of Elasticity. 3rd Edition, McGraw-Hill Book Company, Inc., New York, NY, 1970.

3. Timoshenko, S. and Woinowsky-Krieger, S. Theory of Plates and Shells. 2nd Edition, McGraw-Hill Book Company, Inc., New York, NY 1959.



CIRCULAR MAT.

M_0 = 1283 MAT 6
1104 SHELL 1
Q_0 = 136 MAT 6
140 SHELL

FIGURE 3A.4-1

STRUCTURE CONFIGURATION AND LOADING FOR MAT 6 VS SHELL 1

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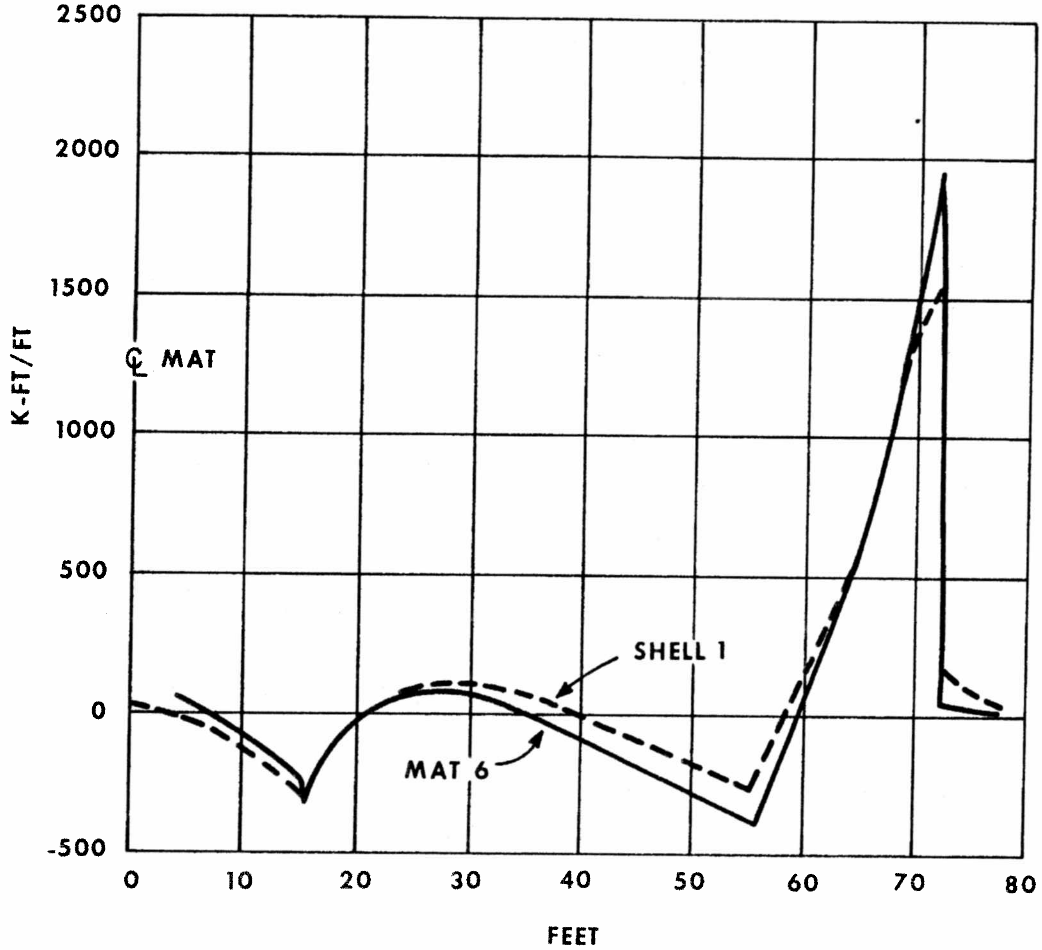

FIGURE 3A.4-2

COMPARISON OF RESULTS OF MAT 6
VS SHELL 1-MAT RADIAL MOMENT

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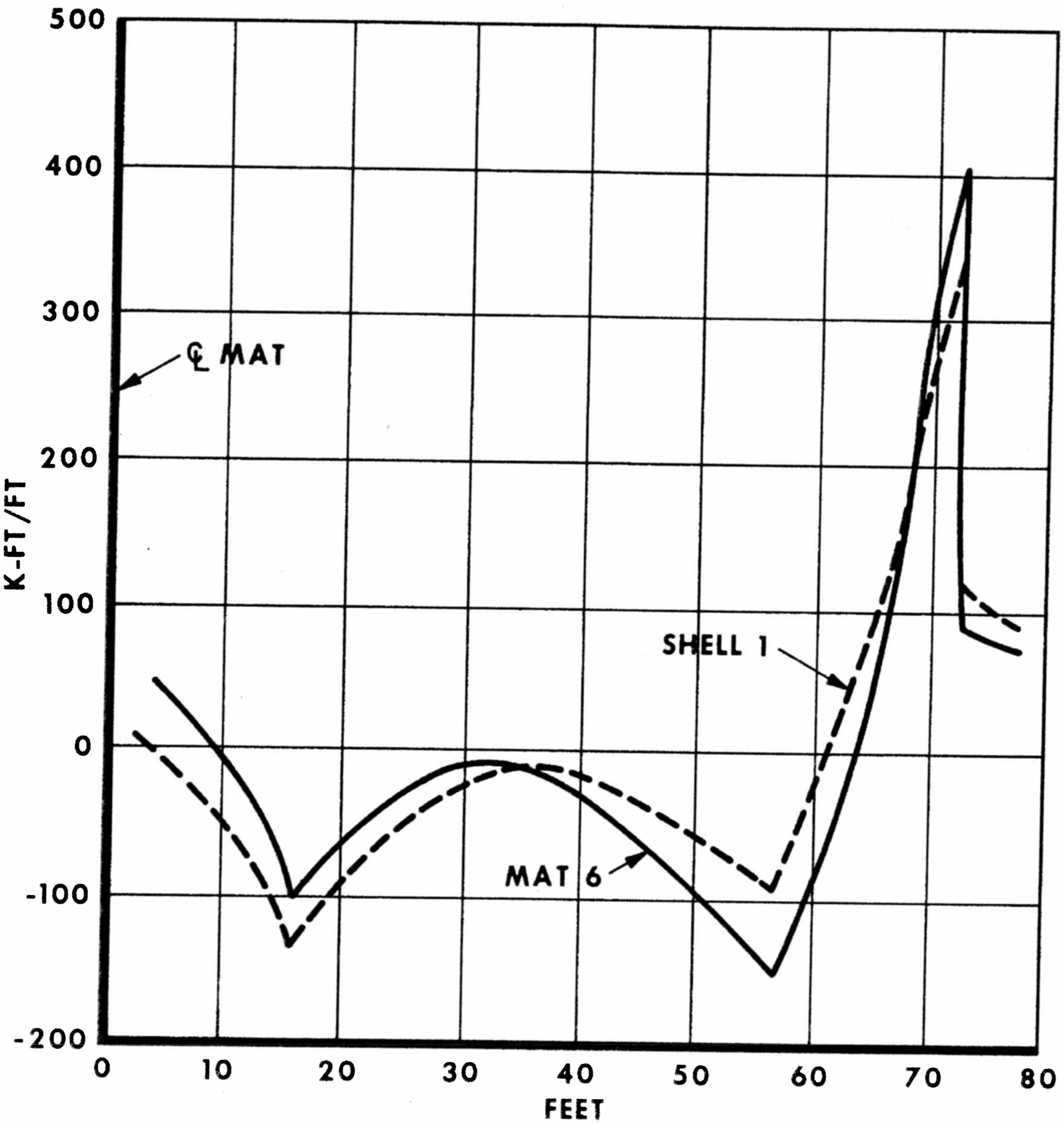

FIGURE 3A.4-3

COMPARISON OF RESULTS OF
MAT 6 VS SHELL 1-MAT
TANGENTIAL MOMENT

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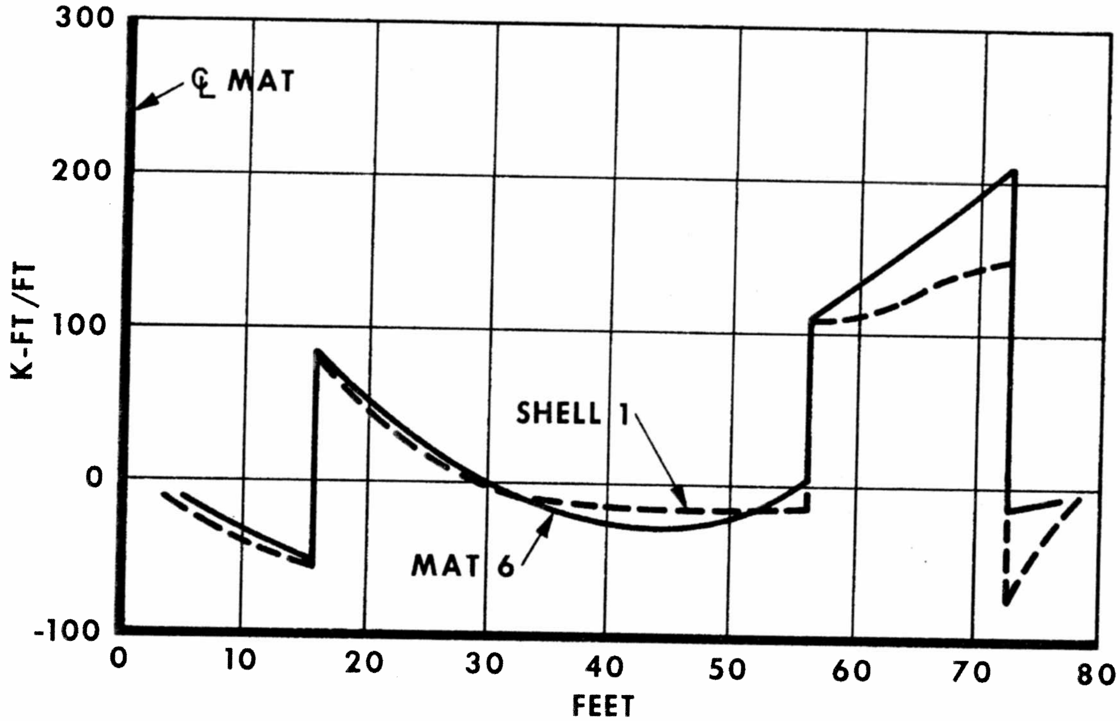

FIGURE 3A.4-4

COMPARISON OF RESULTS OF MAT 6 VS SHELL 1-MAT RADIAL SHEAR

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3A.5 STRUDL (Structural Design Language)

3A.5.1 General Description

The STRUDL computer code used within SWEC was developed from Version 2, Modification 2 (June 1972) of the Integrated Civil Engineering System (ICES) STRUDL II program which was designed and formulated by the Department of Civil Engineering at the Massachusetts Institute of Technology. STRUDL II is a recognized program in the public domain. The software system is IBM-MVS Release 3.8. The hardware configuration is IBM-3033.

The finite element method provides for the solution of a wide range of solid mechanics problems[2] . Its implementation within the context of the STRUDL analysis facilities expands these for the treatment of plane stress, plane strain, plate bending, shallow shell, and three-dimensional stress analysis problems.

STRUDL also provides a dynamic analysis capability for linear elastic structures undergoing small displacements. Either free or forced vibrational response may be obtained and, in the latter case, the forcing functions may be in the form of time histories or response spectra.

The three-dimensional finite element capability of STRUDL is used to analyze the drywell at the region of the equipment hatch and personnel door assembly and other regions of interest.

Seismic Category I structures are analyzed for seismic effect using the dynamic analysis capability of STRUDL. The analysis yields frequencies of vibration, mode shapes, displacements, velocities, accelerations, and forces.

3A.5.2 Program Verification

Comparisons of results for five test problems performed by both STRUDL and GT-STRUDL are provided herein. GT-STRUDL is a recognized program in the public domain, developed by the GT-ICES Systems Laboratory, School of Civil Engineering, Georgia Institute of Technology, Atlanta, Georgia. In all cases, there is excellent agreement of results between STRUDL and GT-STRUDL.

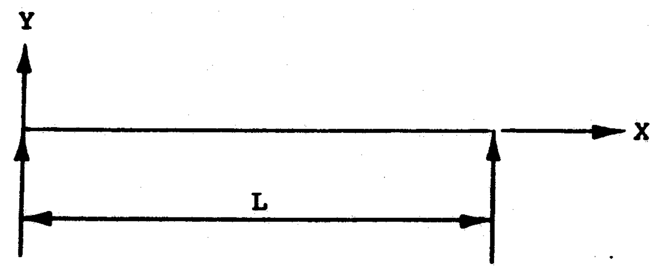
3A.5.2.1 Comparison of STRUDL Versus GT-STRUDL Results for Dynamic Analysis Capability of STRUDL

Problem No. 1

Find the natural frequencies F (I) of vibration for an I-beam with simply supported end, vibrating in the plane of its web,

The pertinent parameters of the beam:

Length	=	30 ft
Modulus of elasticity	=	30×10^6 psi
Moment of inertia	=	3021 $in.^4$
Weight per foot	=	100 lb



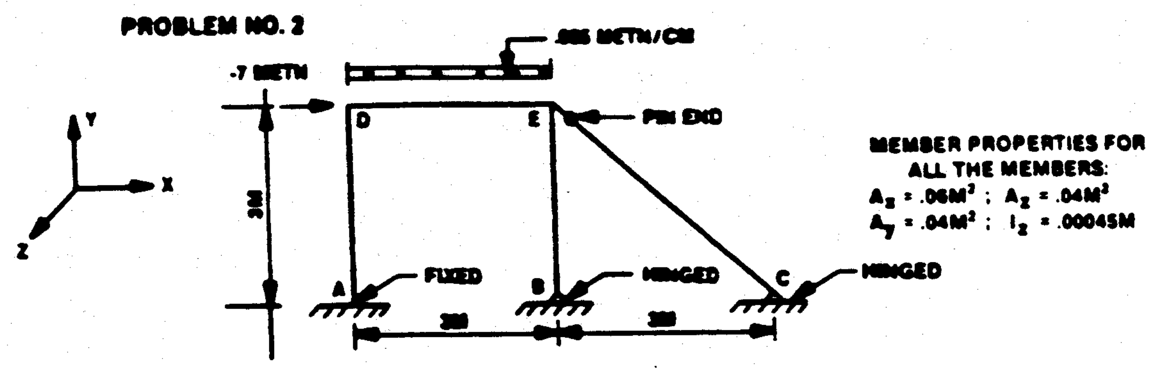
The theoretical results can be verified from Vibration Problems in Engineering (Fourth Edition, S. Timoshenko, D.W. Young, W. Weaver), p. 423, Problem 1.

Results: Natural Frequency F(I), where:

I = mode number
= 24.8 $(I)^2$ cycles/sec
= 155.82 $(I)^2$ rad/sec

The comparison of results (i.e., eigenvalues and eigenvectors) of the theoretical values, STRUDL and GT-STRUDL, are tabulated in Tables 3A.5-1 and 3A.5-2. The eigenvalues for STRUDL and GT-STRUDL agree with each other (Table 3A.5-1). The eigenvectors for STRUDL and GT-STRUDL agree with each other (Table 3A.5-2).

3A.5.2.2 Comparison of STRUDL Versus GT-STRUDL Results for Static Analysis Capability of STRUDL



The frame as shown in the sketch was tested for the loads as shown in the sketch. Also, the frame was tested for joint displacement of joints A and B in the Y direction and also the joint displacement of joint A in the X direction. The member forces and the joint forces of the STRUDL run agreed with the GT-STRUDL run. The comparison of the results are tabulated in Tables 3A.5-3 and 3A.5-4.

LOADING CONDITION 1

Member DE force Y uniform W-.005 Metn/cm

Joint D load force X-0.7 Metn

LOADING CONDITION 2

Joint A displaced Y -0.8 cm

Joint B displaced Y -0.3 cm

LOADING CONDITION 3

Joint A displaced x -0.2 cm

3A.5.2.3 Comparison of STRUDL Versus GT-STRUDL Results for Finite Element Capability of STRUDL

Problem No. 3

A foundation mat was analyzed using the finite element capability of STRUDL for a variety of loading combinations. A comparison check is performed for a loading condition which combines the self weight of the substructure and superstructure, dead load of 2.5 ft of soil above the mat, and east-west tornado loading, by using the finite element capability of GT-STRUDL, a computer program in the public domain. A finite element model is provided on Figure 3A.5-1. Sign convention details are provided on Figure 3A.5-2. Refer to Tables 3A.5-5 and 3A.5-6 for comparison between the results obtained from STRUDL and GT-STRUDL.

3A.5.2.4 Comparison of STRUDL Versus GT-STRUDL Results for Static Analysis Capability of STRUDL

Problem No. 4

A comparison check is performed for suspended ceiling design using static analysis capability of STRUDL versus GT-STRUDL. A model is provided on Figure 3A.5-3. The loading condition accounts for the dead loads of the ceiling. Refer to Tables 3A.5-7 and 3A.5-8 for comparison between the results obtained from STRUDL and GT-STRUDL.

3A.5.2.5 Comparison of STRUDL Versus GT-STRUDL Results for Dynamic Analysis (Response Spectra) Capability of STRUDL

Problem No. 5

A comparison check is performed for suspended ceiling design using dynamic analysis capability of STRUDL versus GT-STRUDL. A model is provided on Figure 3A.5-3. The loading condition accounts for the dynamic seismic loads resulting from ceiling dead load. Refer to Tables 3A.5-9, 3A.5-10, and 3A.5-11 for comparison between the results obtained from STRUDL and GT-STRUDL.

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3A.5.3 GT-STRUDL (Georgia Tech. Structural Design Language)

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After April 1988, RBS will utilize the public domain program GT-STRUDL to perform structural and finite element design analysis. GT-STRUDL performs structural and finite element analysis similar to the SWEC version of STRUDL. GT-STRUDL is substantially similar to the SWEC version of STRUDL. GT-STRUDL is a more current program and contains additional features and output processing capabilities. Section 3A.5.2 shows that SWEC used GT-STRUDL as a verifying program for the SWEC version of STRUDL. The version of GT-STRUDL used by RBS is a newer and more refined version than the version indicated in Section 3A.5.2. Versions of GT-STRUDL that are used by RBS have been qualified to assure that the program performs as stated, that the program correctly computes the phenomena of interest, and that RBS is proficient in utilizing the program. GT-STRUDL qualification is in accordance with the RBS general site and Engineering Department procedures.

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References - Section 3A.5

1. ICES STRUDL - II Structural Design Language Engineering Users' Manual, Vol. I Frame Analysis, Nov. 1968, Vol. II Addition Design and Analysis Facilities (Chapters III and IV), June 1971.

2. Zienkiewicz, O.C. and Cheung, Y.K. The Finite Element Method. McGraw-Hill Book Company, Inc., New York, NY, 1967.

3. GT-STRUDL - GT-ICES Systems Laboratory, School of Civil Engineering, Georgia Institute of Technology, Atlanta, GA.

TABLE 3A.5-1

COMPARISON OF EIGENVALUES FROM THEORETICAL RESULTS,
STRUDL RESULTS, AND GT-STRUDL RESULTS
(PROBLEM NO. 1)

"I" Mode No.	Theoretical Results: Frequency Cycles/sec	STRUDL Results: Frequency Cycles/sec	GT-STRUDL Results: Frequency Cycles/sec
1	24.8	24.84	24.84
2	99.2	99.33	99.33
3	223.2	223.37	223.38
4	396.8	396.39	396.40

NOTE: For comparison purposes, the results of four modes have been tabulated.

TABLE 3A.5-2

COMPARISON OF EIGENVECTORS FROM STRUDL AND GT-STRUDL
(PROBLEM NO. 1)

	Joint	Y-Displacement STRUDL	Y-Displacement GT-STRUDL
MODE 1	1	0.0	0.0
	2	0.309	0.309
	3	0.588	0.588
	4	0.809	0.809
	5	0.951	0.951
	6	1.000	1.000
	7	0.951	0.951
	8	0.809	0.809
	9	0.588	0.588
	10	0.309	0.309
	11	0.0	0.0
MODE 2	1	0.000	0.000
	2	0.618	0.618
	3	1.000	1.000
	4	1.000	1.000
	5	0.618	0.618
	6	0.000	0.000
	7	-0.618	-0.618
	8	-1.000	-1.000
	9	-1.000	-1.000
	10	-0.618	-0.618
	11	0.000	0.000
MODE 3	1	0.0	0.0
	2	-0.809	-0.809
	3	-0.951	-0.951
	4	-0.309	-0.309
	5	0.588	0.588
	6	1.000	1.000
	7	0.588	0.588
	8	-0.309	-0.309
	9	-0.951	-0.951
	10	-0.809	-0.809
	11	0.0	0.0

TABLE 3A.5-2 (Cont)

	Joint	Y-Displacement	
		STRUDL	GT-STRUDL
MODE 4	1	0.0	0.0
	2	1.0	-1.0
	3	0.618	-0.618
	4	-0.618	0.618
	5	-1.0	-1.0
	6	0.000	0.00
	7	1.0	-1.0
	8	0.618	-0.618
	9	-0.618	0.618
	10	-1.0	1.0
	11	0.0	0.0

Note: For comparison purposes, the results of four modes have been tabulated.

TABLE 3A.5-3

THE MEMBER FORCES FROM STRUDL AND GT-STRUDL
COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS
(PROBLEM NO. 2)

			STRUDL			GT-STRUDL		
Member	Loading Condition	Joint	Axial	Shear Y	Mom Z	Axial	Shear Y	Mom Z
AD	1	A	1652.25	-237.33	-7958.75	1652.25	-237.33	-7958.75
		D	-1652.25	237.33	-20071.84	-1652.25	237.33	-20071.86
	2	A	-583.75	-675.01	-104030.94	-583.75	-675.01	-104031.4
		D	583.75	675.01	24305.86	583.75	675.01	24305.9
	3	A	-0694.20	1480.59	107811.12	-694.20	1480.59	107811.6
		D	694.20	-1480.59	67060.81	694.20	-1480.59	67061.1
DE	1	D	1780.56	1652.25	20071.84	1780.56	1652.25	20071.8
		E	-1780.56	1654.68	-20215.77	-1780.56	1654.69	-20215.9
	2	D	675.01	-583.75	-24305.86	675.01	-583.75	-24305.9
		E	-675.01	583.75	-44640.70	-675.01	583.75	-44640.8
	3	D	-1480.59	-694.20	-67060.81	-1480.59	-694.20	-67061.1
		E	1480.59	694.20	-14930.68	1480.59	694.20	-14930.7

TABLE 3A.5-3 (Cont.)

Member	Loading Condition	Joint	STRUDL Axial	STRUDL Shear Y	STRUDL Mom Z	GT-STRUDL Axial	GT-STRUDL Shear Y	GT-STRUDL Mom Z
BE	1	B	45.28	171.16	0.00	45.29	171.16	0.00
		E	-45.28	-171.16	20215.80	-45.29	-171.16	20215.93
	2	B	286.70	377.96	-0.01	286.70	377.96	0.00
		E	-286.70	-377.96	44640.70	-286.70	-377.96	44640.84
	3	B	2301.20	126.41	0.00	2301.20	126.41	0.00
		E	-2301.20	-126.41	-14930.65	-2301.20	-126.41	14930.73
EC	1	E	2276.03	0.0	0.0	2276.04	0.0	0.0
		C	-2276.03	0.0	0.0	-2276.04	0.0	0.0
	2	E	420.09	0.0	0.0	420.09	0.0	0.0
		C	-420.09	0.0	0.0	-420.09	0.0	0.0
	3	E	-2272.64	0.0	0.0	-2272.65	0.0	0.0
		C	2272.64	0.0	0.0	2272.65	0.0	0.0

TABLE 3A.5-4

THE JOINT LOADS (AT SUPPORTS) FROM STRUDL AND GT-STRUDL
COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS

Problem No. 2

		STRUDL			GT-STRUDL		
Joint	Loading	X Force	Y Force	Z Mom	X Force	Y Force	Z Mom
A	1	237.33	1652.25	-7958.75	237.33	1652.25	-7958.75
	2	675.01	-583.75	-104030.94	675.01	-583.75	-104031.48
	3	-1480.59	-694.20	107811.12	-1480.59	-694.20	107811.62
B	1	-171.16	45.28	0.00	-171.16	45.29	0.00
	2	-377.96	286.70	-0.01	-377.96	286.70	0.00
	3	-126.41	2301.20	0.00	-126.41	2301.20	0.00
C	1	-1609.40	1609.40	0.00	-1609.40	1609.40	0.00
	2	-297.05	297.05	0.00	-297.05	297.05	0.00
	3	1607.00	-1607.00	0.00	1607.00	-1607.00	0.00

TABLE 3A.5-5
COMPARISON OF ELEMENT (RANDOMLY SELECTED) STRESSES (PROBLEM NO. 3)

		Mxx		Myy		Mxy		Vxx		Vyy	
Element	Node	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1	-12.546	-12.543	-78.334	-78.354	19.545	19.547	13.417	13.412	3.617	3.618
	2	-4.098	-4.109	-18.611	-18.642	19.107	19.109	13.417	13.412	-8.769	-8.756
	10	-13.852	-13.831	-69.887	-69.860	46.293	46.298	-3.549	-3.538	-8.769	-8.756
	9	-8.936	-8.952	-56.769	-56.764	46.730	46.735	-3.549	-3.538	-3.617	3.618
6	6	-2.145	-2.146	-16.671	-16.678	13.396	13.400	-0.595	-0.597	-0.158	-0.158
	7	-2.876	-2.879	-19.390	-19.409	10.630	10.635	-0.595	-0.597	0.916	0.924
	15	-3.956	-3.940	-11.933	-11.916	9.555	9.558	0.694	0.701	0.916	0.924
	14	-6.974	-6.986	-12.939	-12.935	12.320	12.323	0.694	0.701	-0.158	-0.158
10	11	-10.414	-10.417	-4.335	-4.336	65.119	65.122	1.027	1.026	-13.922	-13.924
	12	-15.984	-15.983	7.190	7.180	13.283	13.285	1.027	1.026	-9.953	-9.948
	20	-74.387	-74.369	-3.683	-3.672	22.101	22.102	5.789	5.798	-9.953	-9.948
	19	-99.338	-99.357	-12.307	-12.310	73.937	73.939	5.789	5.798	-13.922	-13.924
16	18	3.765	3.747	26.738	26.721	94.743	94.475	-15.849	-15.844	0.332	0.335
	19	-56.181	-56.179	-5.239	-5.250	71.624	71.624	-15.849	-15.844	-9.083	-9.078
	27	-117.077	-117.062	-7.560	-7.547	49.705	49.704	-27.147	-27.139	-9.083	-9.078
	26	1.424	1.396	31.393	31.403	72.554	72.555	-27.147	-27.139	0.332	0.335
22	25	-2.301	-2.323	56.058	56.053	74.477	74.481	-3.966	-3.962	-1.428	-1.429
	26	0.446	0.451	33.162	33.148	71.634	71.637	-3.966	-3.962	-2.312	-2.305
	34	-11.675	-11.652	29.188	29.207	49.249	49.251	-5.178	-5.163	-2.312	-2.305
	33	-1.561	-1.594	45.381	45.381	52.093	52.094	-5.178	-5.163	-1.428	-1.429
45	51	-149.275	-149.281	-2.222	-2.223	29.027	29.028	6.244	6.243	0.660	0.658
	52	-113.469	-113.472	-1.815	-1.825	26.628	26.629	6.244	6.243	-1.319	-1.314
	60	-124.368	-124.354	-0.097	-0.085	23.243	23.244	3.869	3.877	-1.319	-1.314
	59	-144.766	-144.787	-2.139	-2.142	25.642	25.643	3.869	3.877	0.660	0.658

TABLE 3A.5-5 (Cont)

Element	Node	Mxx STRUDL	Mxx GT-STRUDL	Myy STRUDL	Myy GT-STRUDL	Mxy STRUDL	Mxy GT-STRUDL	Vxx STRUDL	Vxx GT-STRUDL	Vyy STRUDL	Vyy GT-STRUDL
52	59	-144.756	-144.773	-2.047	-2.055	24.031	24.030	3.874	3.879	2.303	2.305
	60	-124.346	-124.339	0.0146	0.0123	22.659	22.658	3.874	3.879	-2.621	-2.620
	68	-143.165	-143.158	0.591	0.595	24.111	24.111	-2.034	-2.030	-2.621	-2.620
	67	-129.535	-129.548	-1.240	-1.237	25.483	25.482	-2.034	-2.030	2.303	2.305
60	68	-130.197	-130.198	2.811	2.810	22.296	22.298	22.805	22.805	-4.389	-4.388
	69	-20.373	-20.368	25.257	25.246	11.012	11.013	22.805	22.805	5.429	5.433
	77	-19.106	-19.097	61.775	61.791	34.229	34.230	34.586	34.591	5.429	5.433
	76	-157.545	-157.547	-0.386	-0.386	45.512	45.514	34.586	34.591	-4.389	-4.388
68	77	-24.513	-24.517	54.034	54.028	35.968	35.968	14.908	14.911	0.413	0.414
	78	11.657	11.666	104.330	104.326	39.457	39.457	14.908	14.911	1.380	1.384
	86	21.811	21.826	103.782	103.797	82.856	82.858	16.069	16.074	1.380	1.384
	85	-24.572	-24.578	56.964	56.973	79.367	79.369	16.069	16.074	0.413	0.414
80	91	-36.410	-36.423	2.194	2.186	30.708	30.708	1.185	1.190	7.587	7.593
	92	35.756	-35.747	8.412	8.413	65.545	65.546	1.185	1.190	6.573	6.572
	100	-7.093	-7.089	8.341	8.342	69.355	69.356	0.424	0.424	6.573	6.572
	99	-4.321	-4.324	3.110	3.117	34.518	34.518	0.424	0.424	7.587	7.593

TABLE 3A.5-6

COMPARISON OF RESULTANT (RANDOMLY SELECTED) JOINT DISP. SUPPORTS
(GLOBAL)

(Problem No. 3)

	Z Displacement		X Rotation		Y Rotation	
Joint	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	-0.0447149	-0.0447119	0.0004818	0.0004817	0.0001927	0.0001925
5	-0.0484141	-0.0484130	0.0003599	0.0003598	0.0001907	0.0001908
15	-0.0483233	-0.0483236	0.0003430	0.0003429	0.0002060	0.0002060
20	-0.0427901	-0.0427890	0.0003791	0.0003790	0.0002582	0.0002580
30	-0.0435345	-0.0435547	0.0002832	0.0002832	0.0002928	0.0002920
40	-0.0442976	-0.0442990	0.0002317	0.0002317	0.0002164	0.0002163
70	-0.0354678	-0.0354688	0.0002588	0.0002586	0.0004367	0.0004369
100	-0.0259728	-0.0259738	0.0001780	0.0001779	0.0008314	0.0008313
112	-0.0426962	-0.0426994	0.0001656	0.0001656	0.0008937	0.0008930

TABLE 3A.5-7

COMPARISON OF MEMBERS (RANDOMLY SELECTED) FORCES AND MOMENTS (PROBLEM NO. 4)

Member	Joint	Axial		Shear Y		Shear Z		Torsional Moment		Bending Y Moment		Bending Z Moment	
		STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1	0.0	0.0	0.0	0.0	0.527	0.527	0.0	0.0	0.0	0.0	0.0	0.0
	2	0.0	0.0	0.0	0.0	-0.371	-0.371	0.0	0.0	-21.567	-21.568	0.0	0.0
4	4	0.0	0.0	0.0	0.0	-0.713	-0.713	0.0	0.0	5.158	5.158	0.0	0.0
	5	0.0	0.0	0.0	0.0	0.869	0.869	0.0	0.0	32.818	32.818	0.0	0.0
10	10	0.0	0.0	0.0	0.0	0.226	0.226	0.0	0.0	5.627	5.627	0.0	0.0
	11	0.0	0.0	0.0	0.0	-0.070	-0.070	0.0	0.0	-12.738	-12.738	0.0	0.0
20	3	0.0	0.0	0.0	0.0	0.158	0.158	0.0	0.0	0.0	0.0	0.0	0.0
	23	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-5.670	-5.670	0.0	0.0
25	14	0.0	0.0	0.0	0.0	0.117	0.117	0.0	0.0	1.818	1.818	0.0	0.0
	17	0.0	0.0	0.0	0.0	-0.093	-0.093	0.0	0.0	-4.338	-4.338	0.0	0.0
30	36	0.0	0.0	0.0	0.0	-0.093	-0.093	0.0	0.0	4.338	4.338	0.0	0.0
	39	0.0	0.0	0.0	0.0	0.117	0.117	0.0	0.0	-1.818	-1.818	0.0	0.0
50	12	0.0	0.0	0.0	0.0	0.426	0.426	0.0	0.0	0.0	0.0	0.0	0.0
	54	0.0	0.0	0.0	0.0	0.000	0.000	0.0	0.0	-15.336	-15.336	0.0	0.0
60	47	0.0	0.0	0.0	0.0	-0.069	-0.069	0.0	0.0	12.621	12.621	0.0	0.0
	48	0.0	0.0	0.0	0.0	0.225	0.225	0.0	0.0	-5.569	-5.569	0.0	0.0

TABLE 3A.5-8

COMPARISON OF RESULTANT JOINT LOADS - SUPPORTS (PROBLEM NO. 4)
(GLOBAL)

Joint	X Force		Y Force		Z Force		X Mom		Y Mom		Z Mom	
	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	0.0	0.0	0.0	0.0	0.953	0.953	0.0	0.0	0.0	0.0	0.0	0.0
5	0.0	0.0	0.0	0.0	1.885	1.885	0.0	0.0	0.0	0.0	0.0	0.0
9	0.0	0.0	0.0	0.0	1.536	1.536	0.0	0.0	0.0	0.0	0.0	0.0
12	0.0	0.0	0.0	0.0	0.769	0.769	0.0	0.0	0.0	0.0	0.0	0.0
41	0.0	0.0	0.0	0.0	0.953	0.953	0.0	0.0	0.0	0.0	0.0	0.0
45	0.0	0.0	0.0	0.0	1.885	1.885	0.0	0.0	0.0	0.0	0.0	0.0
49	0.0	0.0	0.0	0.0	1.536	1.536	0.0	0.0	0.0	0.0	0.0	0.0
52	0.0	0.0	0.0	0.0	0.769	0.769	0.0	0.0	0.0	0.0	0.0	0.0

TABLE 3A.5-9

COMPARISON OF EIGENVALUES, FREQUENCIES, AND PERIODS (PROBLEM NO. 5)

Mode	Eigenvalue		Frequency (Cycles/Time Unit)		Period (Time Unit/Cycle)	
	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1.791123D 03	1.791171D+03	6.735703D 00	6.735791D+00	1.484626D-01	1.484607D-01
2	1.892565D 03	1.892596D+03	6.923816D 00	6.923873D+00	1.444290D-01	1.444278D-01
3	1.925683D 03	1.925715D+03	6.934134D 00	6.984191D+00	1.431817D-01	1.431805D-01
4	1.935117D 03	1.938148D+03	7.006645D 00	7.006702D+00	1.427217D-01	1.427205D-01
5	1.947453D 03	1.947485D+03	7.023501D 00	7.023558D+00	1.423791D-01	1.423780D-01
6	1.949031D 03	1.949063D+03	7.026346D 00	7.026403D+00	1.423215D-01	1.423203D-01
7	1.949454D 03	1.949486D+03	7.027108D 00	7.027166D+00	1.423060D-01	1.423049D-01
8	1.949909D 03	1.949940D+03	7.027927D 00	7.027984D+00	1.422895D-01	1.422883D-01
9	2.894955D 03	2.895024D+03	8.563297D 00	8.563400D+00	1.167775D-01	1.167760D-01
10	2.935117D 03	2.935188D+03	8.622493D 00	8.622597D+00	1.159757D-01	1.159743D-01
11	2.935117D 03	2.935188D+03	8.622493D 00	8.622597D+00	1.159757D-01	1.159743D-01
12	3.566743D 03	3.566836D+03	9.505086D 00	9.505210D+00	1.052068D-01	1.052055D-01
13	3.941438D 03	3.941542D+03	9.991886D 00	9.992018D+00	1.000812D-01	1.000799D-01
14	8.732367D 03	8.732600D+03	1.487257D 01	1.487277D+01	6.723786D-02	6.723696D-02
15	1.345767D 04	1.345793D+04	1.846312D 01	1.846330D+01	5.416204D-02	5.416151D-02
16	1.346419D 04	1.346445D+04	1.846759D 01	1.846777D+01	5.414892D-02	5.414839D-02
17	1.346597D 04	1.346623D+04	1.846881D 01	1.846899D+01	5.414534D-02	5.414481D-02
18	2.573237D 04	2.573314D+04	2.553054D 01	2.553092D+01	3.916877D-02	3.916819D-02
19	2.650393D 04	2.650463D+04	2.591047D 01	2.591081D+01	3.859444D-02	3.859393D-02
20	2.745771D 04	2.745854D+04	2.637256D 01	2.637296D+01	3.791820D-02	3.791763D-02
21	3.177105D 04	3.177167D+04	2.836847D 01	2.836875D+01	3.525040D-02	3.525006D-02
22	3.248699D 04	3.248762D+04	2.868632D 01	2.868660D+01	3.485952D-02	3.485948D-02
23	3.525845D 04	3.525916D+04	2.988489D 01	2.988520D+01	3.346172D-02	3.346138D-02
24	3.953962D 04	3.954066D+04	3.164728D 01	3.164769D+01	3.159829D-02	3.159788D-02
25	5.170876D 04	5.171001D+04	3.619113D 01	3.619157D+01	2.763108D-02	2.763074D-02
26	5.592949D 04	5.593058D+04	3.763922D 01	3.763958D+01	2.656804D-02	2.656778D-02
27	5.592949D 04	5.593058D+04	3.763922D 01	3.763958D+01	2.656804D-02	2.656778D-02
28	5.959983D 04	5.960165D+04	3.885462D 01	3.885521D+01	2.573697D-02	2.573657D-02
29	6.538255D 04	6.538428D+04	4.069594D 01	4.069648D+01	2.457248D-02	2.457215D-02
30	8.206529D 04	8.206758D+04	4.559318D 01	4.559382D+01	2.193311D-02	2.193280D-02

TABLE 3A.5-10

COMPARISON OF EIGENVECTORS
FOR FEW RANDOMLY SELECTED MODES AND JOINTS (PROBLEM NO. 5)

Eigenvectors (Global)

		X-Displacement		Z-Displacement	
Mode	Joint	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1	0.0	0.0	0.0	0.0
	2	0.721	-0.721	0.000	0.000
	13	0.721	-0.721	0.000	0.000
	23	1.00	-1.00	0.000	0.000
	43	0.999	-0.999	0.000	0.000
6	1	0.0	0.0	0.0	0.0
	2	0.000	0.000	-0.0009	-0.0009
	13	0.000	0.000	-0.0014	-0.0014
	23	0.000	0.000	-0.567	-0.567
	43	0.000	0.000	-0.0005	-0.0005
3	1	0.0	0.0	0.0	0.0
	2	0.000	0.000	0.0095	0.0095
	13	0.000	0.000	0.0153	0.0153
	23	0.000	0.000	1.000	1.000
	43	0.000	0.000	0.013	0.013

Y-displacements are approximately 0.000 for all three modes.

TABLE 3A.5-11

COMPARISON OF JOINT DISPLACEMENTS AT THE FREE JOINTS FOR
RANDOMLY SELECTED JOINTS (GLOBAL)

Problem No. 5

	Response	X-Displacement		Y-Displacement		Z-Displacement	
Joint	Type	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
	RMS	0.000	0.000	0.000	0.000	0.00099	0.00099
7	ABS SUM	0.000	0.000	0.000	0.000	0.00213	0.00212
	CSM	0.000	0.000	0.000	0.000	0.00117	0.00116
	RMS	0.000	0.000	0.000	0.000	0.0105	0.0104
15	ABS SUM	0.000	0.000	0.000	0.000	0.0105	0.0104
	CSM	0.000	0.000	0.000	0.000	0.0105	0.0104
	RMS	0.000	0.000	0.000	0.000	0.0424	0.0424
22	ABS SUM	0.000	0.000	0.000	0.000	0.0449	0.0449
	CSM	0.000	0.000	0.000	0.000	0.0424	0.0424
	RMS	0.000	0.000	0.000	0.000	0.0287	0.0288
36	ABS SUM	0.000	0.000	0.000	0.000	0.0287	0.2920
	CSM	0.000	0.000	0.000	0.000	0.0287	0.2920

RMS = Root Mean Square
ABS SUM = Absolute Sum
CSM = Closely Spaced Mode

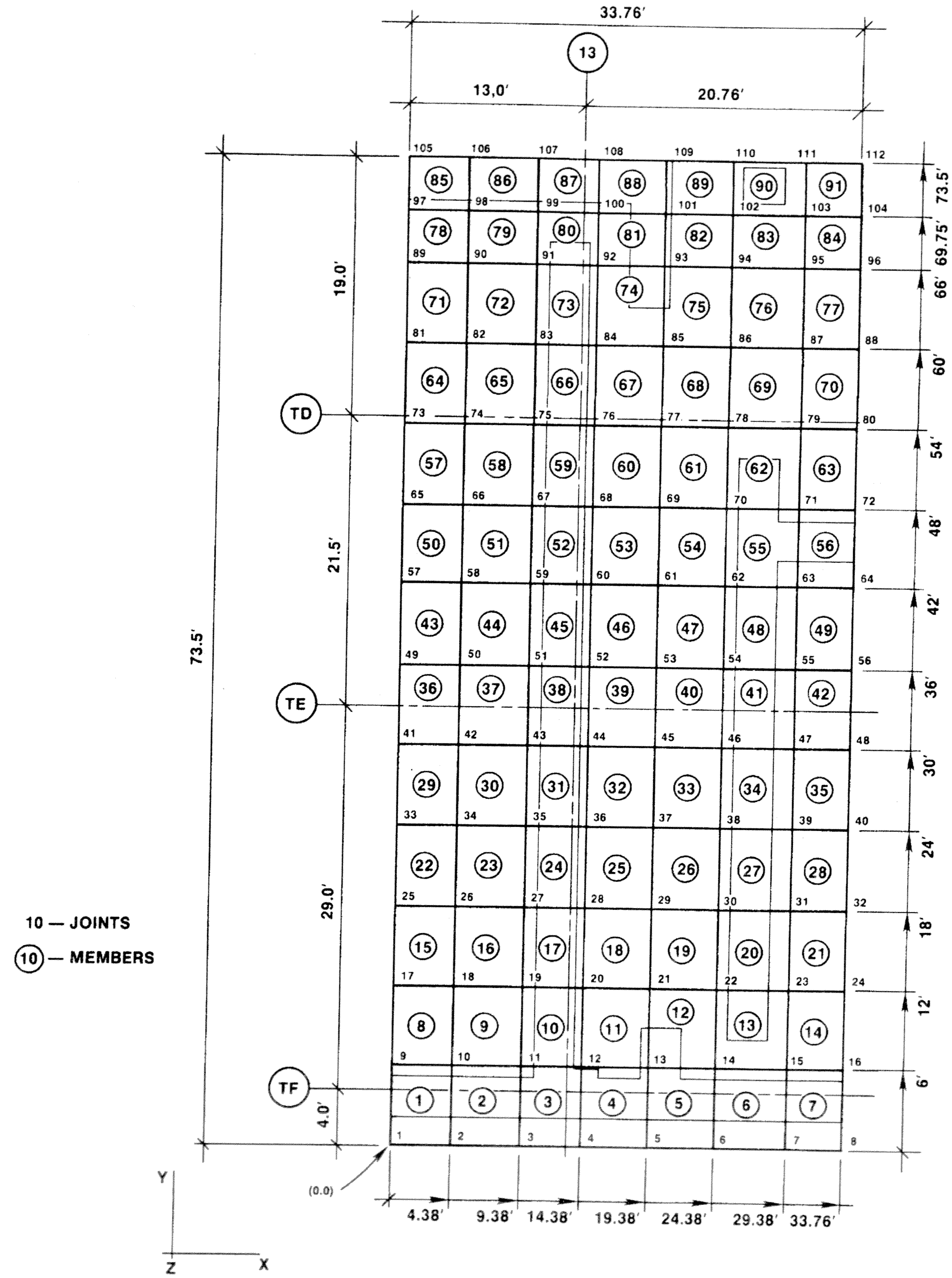

FIGURE 3A.5-1

FINITE ELEMENT MODEL OF THE FOUNDATION MAT FOR A PORTION OF THE OFF-GAS BUILDING

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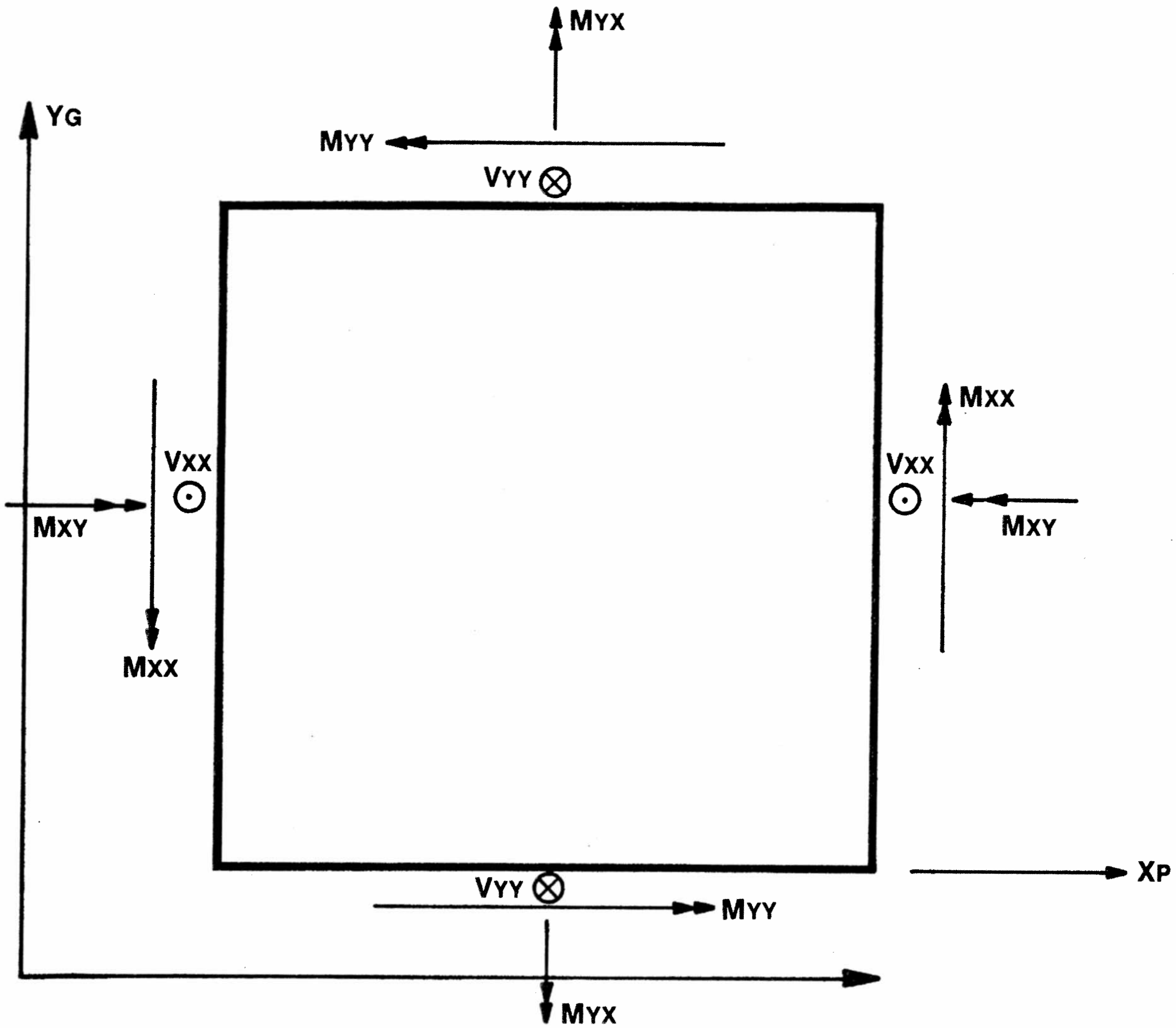

NOTES:

MXX, MYY, MXY, THE MOMENT RESULTANTS,ARE OUTPUT AT THE AVAILABLE LOCATIONS ON THE ELEMENTS.
VXX, VYY ARE ALSO OUTPUT FOR THE TRANSVERSE SHEAR RESULTANTS
⊙ POSITIVE DIRECTION COMING OUT OF PAPER
⊗ POSITIVE DIRECTION GOING INTO THE PAPER
XP, YP — PLANER COORDINATE SYSTEM
XG,YG — GLOBAL COORDINATE SYSTEM
XP IS PARALLEL TO XG
YP IS PARALLEL TO YG

FIGURE 3A.5-2

POSITIVE SIGN CONVENTION FOR RESULTS OF PLATE BENDING ELEMENT

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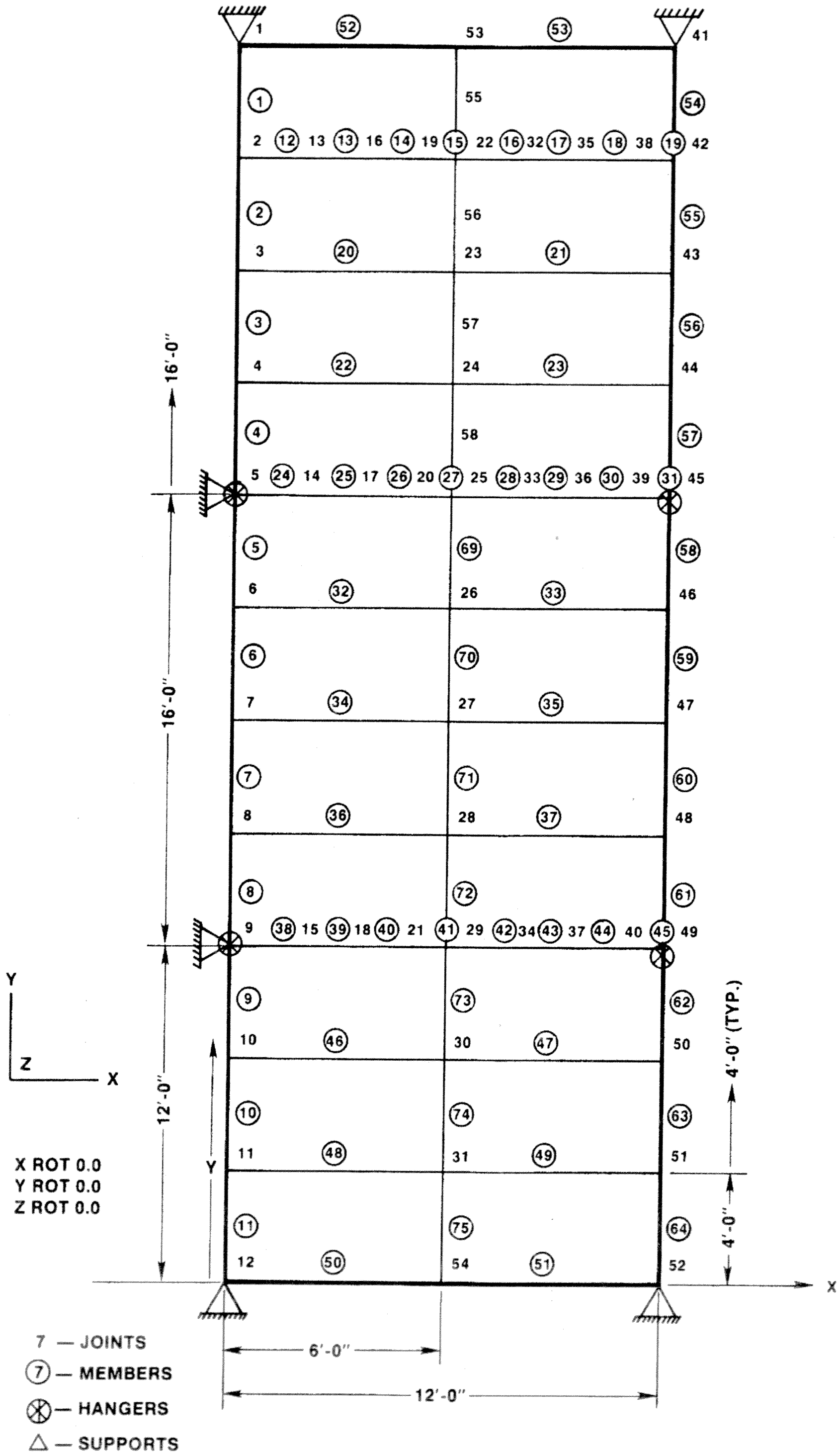

FIGURE 3A.5-3

MODEL — SUSPENDED CEILING

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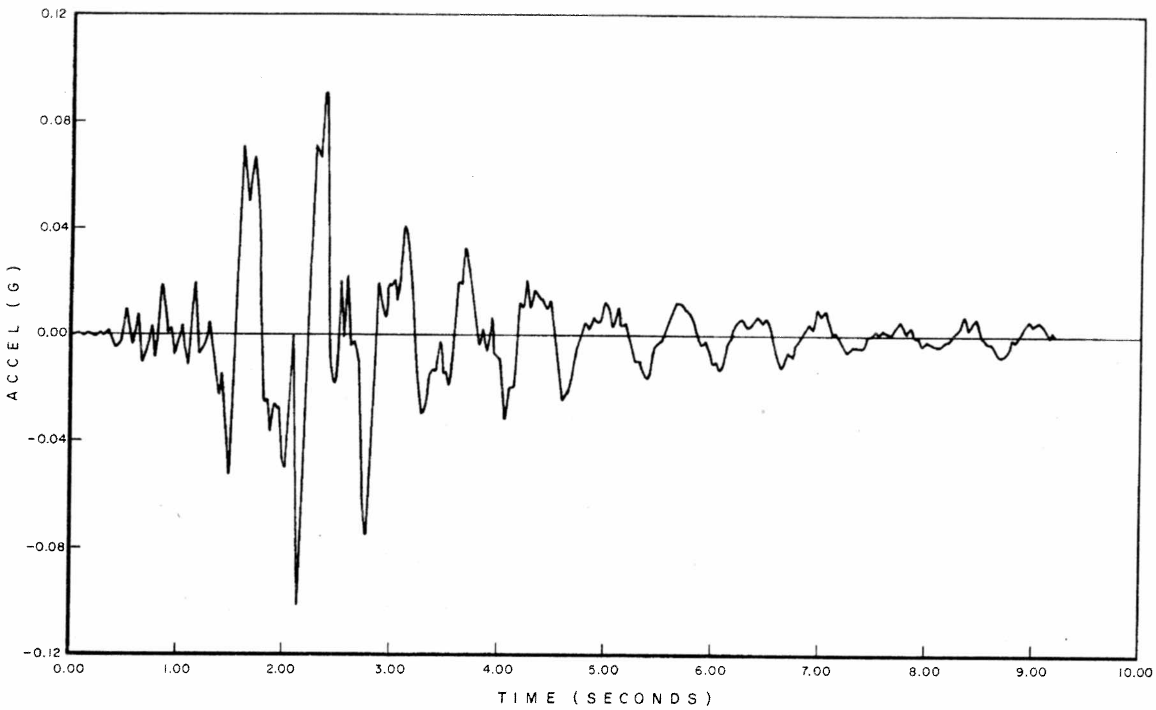
3A.6 TIME HISTORY (TIMHIS6) PROGRAM

The TIME HISTORY PROGRAM computes time history response and amplified response spectra at any mass point location of a lumped-mass, spring-connected system due to a synthetic earthquake, time-motion record input. The program calculates the time history response at the selected mass locations by standard modal superposition. The responses are computed by integration of the modal equations of the system by the "Exact Method[5]." The analytical procedure is described in Section 3.7.2.1A. The program's main application is the generation of amplified response spectra used for design of Seismic Category I equipment and piping.

The TIME HISTORY PROGRAM'S solution to a test problem is substantially identical to the solution obtained using STRUDL II. The test problem uses an actual containment structure subjected to an earthquake, time-motion record input of Helena East-West normalized to 0.06 g. The time history response of the structure is computed at the operating floor level by the TIME HISTORY PROGRAM and STRUDL II. The results of these two analyses (Fig. 3A.6-1 and 3A.6-2, respectively) agree extremely well with each other.

References - Section 3A.6

1. Wigan, N. C. and Jennings, P. C. Digital Calculation of Response-Spectrum from Strong-Motion Earthquake Records. National Science Foundation, June 1968.



MAX. ACCEL. -0.1017
AT TIME 2.150

FIGURE 3A.6-1

RESULTS OF TIME HISTORY PROGRAM
CONTAINMENT STRUCTURE-OPERATING
FLOOR LEVEL TIME HISTORY
OF STRUCTURAL RESPONSE

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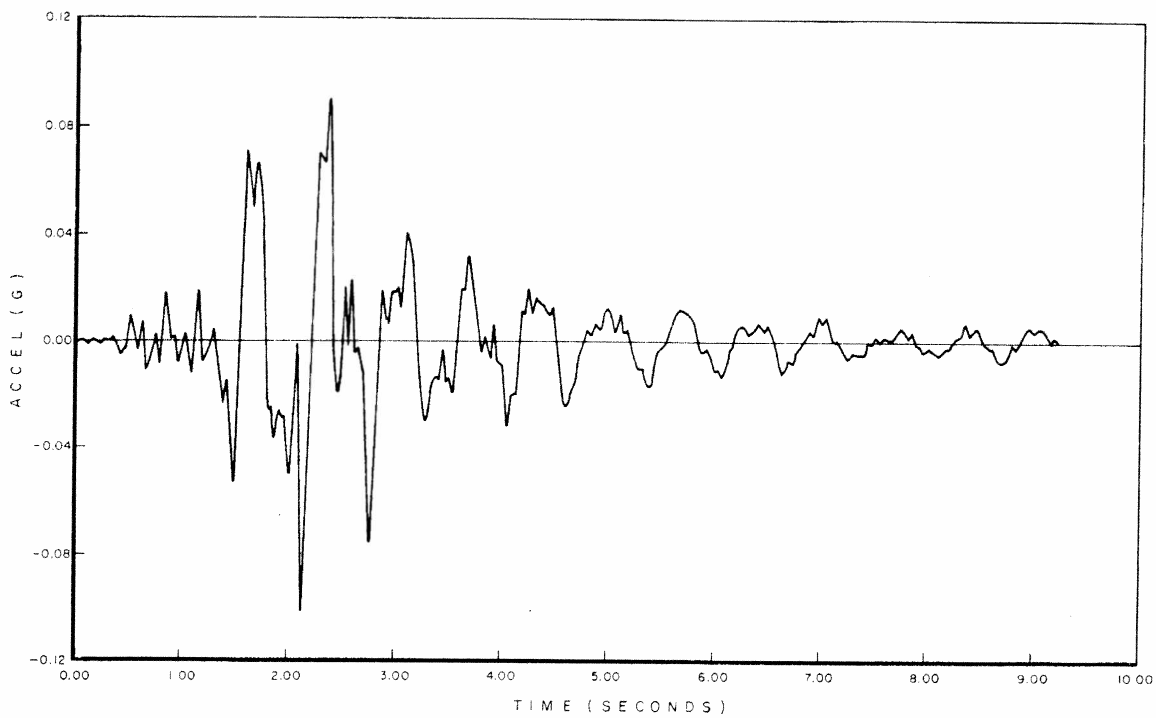

MAX. ACCEL. -0.1015
AT TIME 2.150

FIGURE 3A.6-2

RESULTS OF STUDL II ANALYSIS
CONTAINMENT STRUCTURE -
OPERATING FLOOR LEVEL TIME
HISTORY OF STRUCTURAL RESPONSE

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3A.7 ME-121

3A.7.1 General Description

ME-121 is a seismic data generation program, written and fully documented by Stone & Webster Engineering Corporation for in-house use. This program generates seismic data tables and plots which are necessary for seismic analysis and/or seismic testing of floor- and wall-mounted equipment.

ME-121 utilizes existing seismic data (Amplified Response Spectra) in terms of acceleration (g's) versus period (seconds). It is used to spread the acceleration peaks and generate a new set of seismic data (Required Response Spectra) in terms of acceleration (g's) versus frequency (Hz).

3A.7.2 Program Verification

A comparison of input seismic data, Fig. 3A.7-1 (Amplified Response Spectra), versus output seismic data, Fig. 3A.7-2 (Required Response Spectrum), demonstrates the function and adequacy of the program. A seismic data table is also generated for the new spectra (Table 3A.7-1). It can be seen that table values g_2 = (1.3 static equivalent factor times the peak acceleration value in the resonant range of 1 Hz to 33 Hz), g_6 = (rigid range acceleration value at cutoff frequency 32 Hz), and g_{10} = (zero period acceleration) are in agreement with the Required Response Spectra (Fig. 3A.7-2).

Peak spreading, by the appropriate plus and minus percentage, is performed by the program. Sloped sides from the spread peak, parallel with the lines forming the original spectrum peak, are added by hand, in conformance with the requirements of Regulatory Guide 1.122.

TABLE 3A.7-1

SEISMIC DATA TABLE

		G Data for Static Analysis									G Data for Testing			
		Resonance Range LT FC					Rigid Range GT FC				Zero Period Acceleration			
Building	El (ft)	g(1)	g(2)	g(3)	g(4)	FC CPS	g(5)	g(6)	g(7)	g(8)	g(9)	g(10)	g(11)	g(12)
Secondary	411.33	6.83	4.62	7.28	4.85	33	0.74	0.36	1.01	0.52	0.65	0.33	0.90	0.49
Containment	387.33	5.88	4.35	6.27	4.59	33	0.46	0.33	0.70	0.49	0.43	0.31	0.66	0.46
(typical,	352.50	5.03	4.19	5.39	4.43	33	0.36	0.32	0.55	0.47	0.35	0.30	0.54	0.45
i.e., not	330.08	4.80	4.08	5.05	4.32	33	0.31	0.31	0.47	0.46	0.31	0.29	0.46	0.44
RBS-	287.75	3.82	2.59	4.05	2.84	33	0.30	0.23	0.47	0.35	0.26	0.20	0.42	0.31
specific	259.75	2.85	1.38	3.08	1.70	33	0.30	0.18	0.44	0.28	0.25	0.14	0.39	0.23
data)	238.75	2.01	1.13	2.04	1.45	33	0.21	0.16	0.34	0.26	0.19	0.13	0.31	0.22
	213.75	1.58	0.88	1.73	1.19	33	0.17	0.21	0.29	0.36	0.16	0.11	0.27	0.20
	197.00	1.06	0.69	1.23	0.97	33	0.14	0.11	0.24	0.19	0.12	0.10	0.22	0.18

Key

FC = cut off frequency
LT = lower than
GT = greater than
CPS = cycles/sec (Hz)

NOTE: Damping factor SSE - 3.0 percent
OBE - 2.0 percent

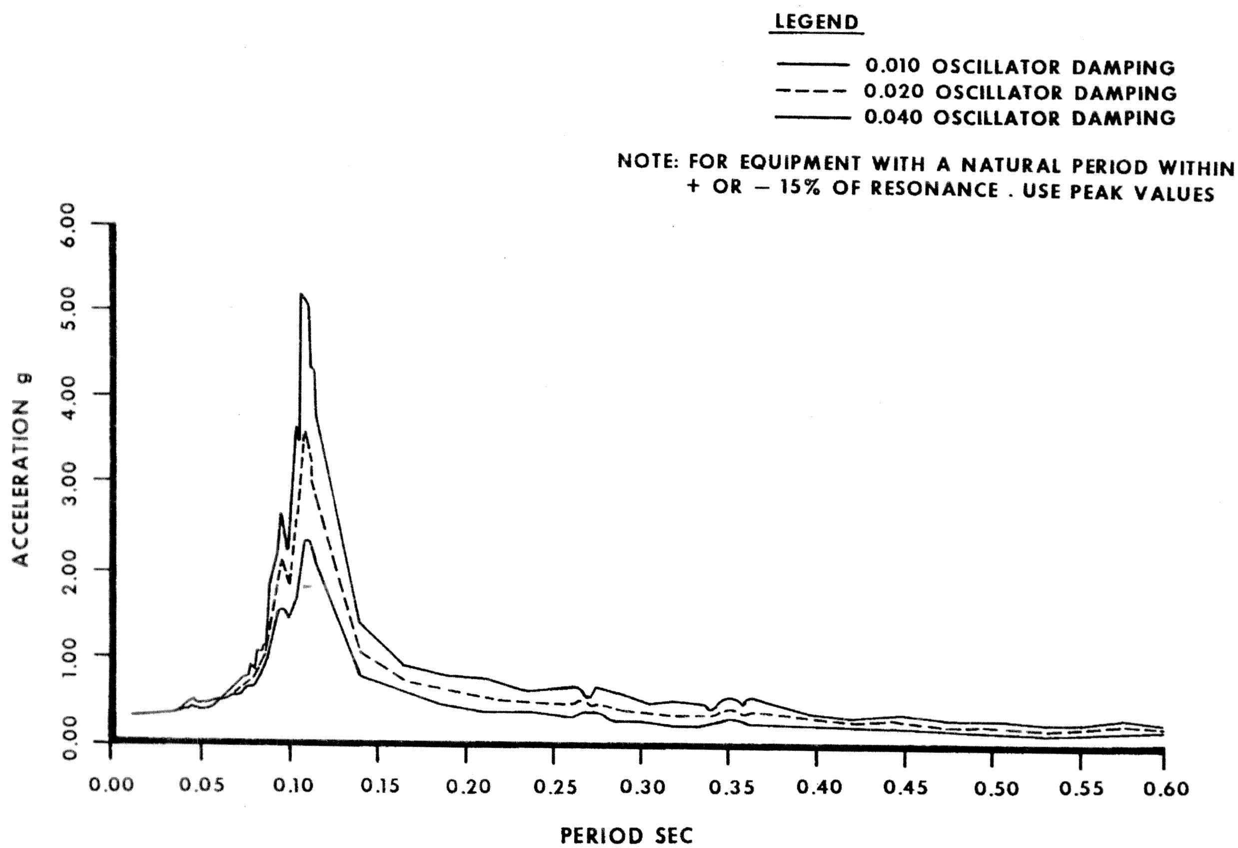

FIGURE 3A.7-1

INPUT ARS TO ME-121

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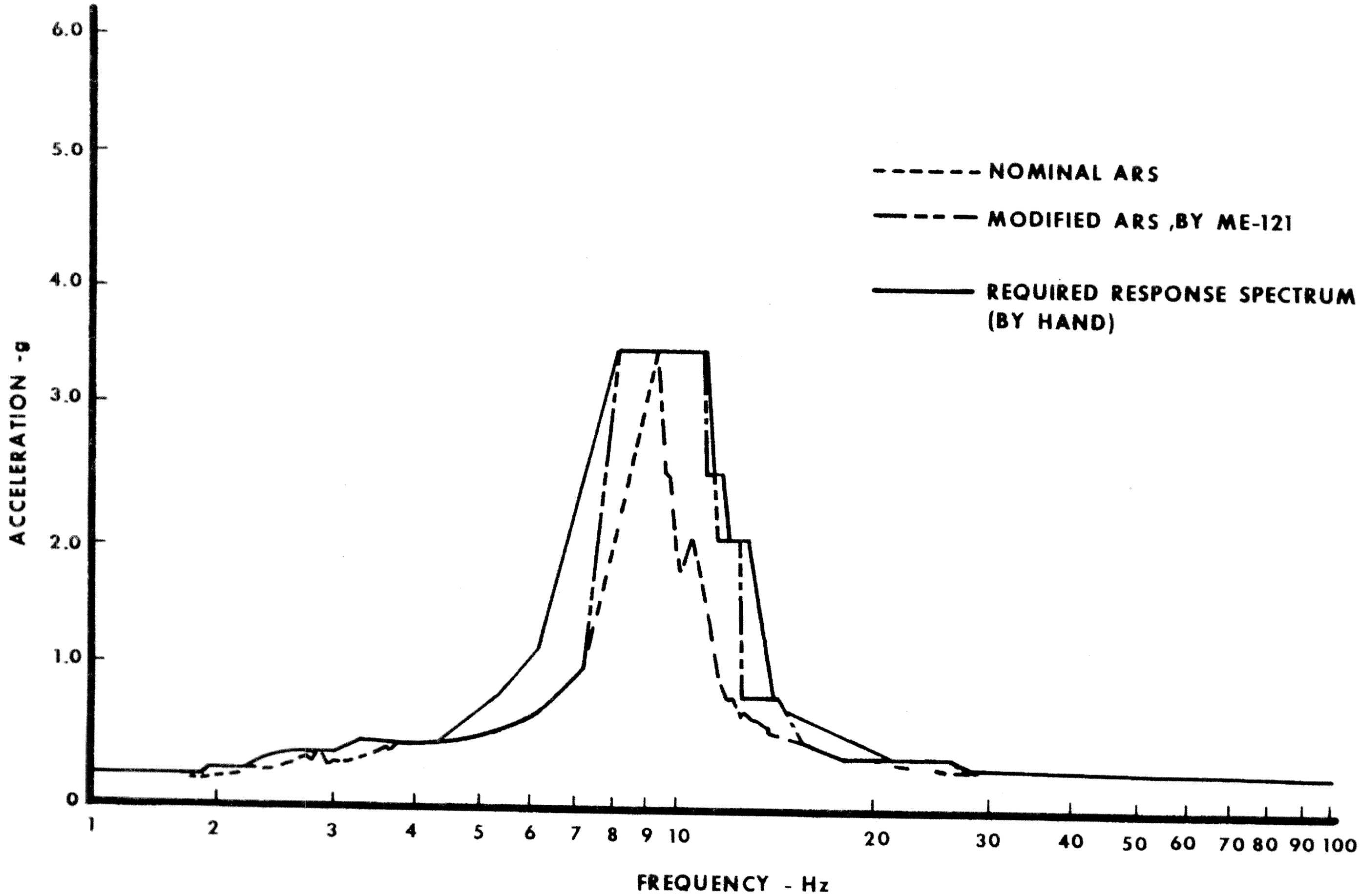

FIGURE 3A.7-2

COMPARISON OF AMPLIFIED RESPONSE SPECTRA VS REQUIRED RESPONSE SPECTRA

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3A.8 DINASAW (Dynamic Inelastic Nonlinear Analysis by Stone & Webster)

3A.8.1 General Description

DINASAW is a modification and extension of a lumped-mass elastic-plastic dynamic analysis code used to predict the large-deflection behavior of beams and rings[1]. DINASAW extends this analysis to cover pipes (tubular cross sections) which may impact walls or restraints.

The analysis, as derived, employs the spatial finite-element method in which the tangential and normal displacement fields are represented by cubic interpolations[1,2]. By applying the principle of virtual work in conjunction with D'Alembert's principle, the equations of motion may be derived in the following form:

$$[M]\ [\ddot{q}] = (F) - (P) - (H)[q]$$

where:

$[q]$ and $[\ddot{q}]$	=	Generalized displacements and generalized accelerations, respectively, for the complete assembled discretized structure, defined with respect to a global coordinate system
$[M]$	=	Lumped-mass matrix for the complete assembled discretized structure
(F)	=	Assembled vector of externally applied loading
(P)	=	Assembled internal force matrix (replaces conventional stiffness matrix)
$(H)\ [q]$	=	Generalized loads arising from both large deflection and plastic behavior

3A.8.2 Program Verification

Two examples are discussed here. The first involves a ring subjected to a radial blast wave over a portion of its circumference[1]. The resulting deformation severely distorts the ring, flattening it considerably. The computer code results closely follow both the displacement field and the strain time history.

The second case involves the impact of a rotor segment onto a ring or shroud[2]. Again the program, in conjunction with the Collision Imparted Velocity Method (CIVM), follows experimental results very closely.

References - Section 3A.8

1. Wu, R. and Witmer, E. Finite-Element Analysis of Large Transient Elastic-Plastic Deformations of Simple Structures, with Application to the Engine Rotor Fragment Containment-Deflection Problem. Aeroelastic and Structures Research Laboratory, Department of Aeronautics and Astronautics, Massachusetts Institute of Technology, January 1972.

2. Collins, T. and Witmer, E. Application of the Collision Imparted Velocity Method for Analyzing the Responses of Containment and Reflector Structures to Engine Rotor Fragment Impact. Aeroelastic and Structures Research Laboratory, Department of Aeronautics and Astronautics, Massachusetts Institute of Technology, August 1973.

3A.9 LIMITA II

3A.9.1 General Description

LIMITA II is a two-dimensional, nonlinear, transient dynamic analysis computer code. A plane frame is simulated as a lumped parameter system, consisting of an assembly of discrete lumped masses connected by beam members. Under any loading, the equilibrium at the rth mass point is ensured by the equation of motion:

$$M_r \ddot{q}_r + \sum^{i} C_{ri}\dot{q}_i + \sum K_{ri} q_i = f_r \qquad (3A.9\text{-}1)$$

Here the summation indicates series with one term for each of the i displacements where:

C_{ri} = Damping coefficient, which applies to the ith velocity in the rth equation of motion

K_{ri} = Member stiffness, which is defined as the force necessary to hold the structural member from moving in the rth degree of freedom when the ith degree of freedom is given a unit displacement and all other degrees of freedom and are restrained from moving[1,2]

f_r = External load factor

To take account of nonlinear effects, such as plasticity and large deflections, Equation 3A.9-1 is solved by an incremental method[3]. At any particular time, t, the displacement increment is obtained from:

$$\boxed{M_r\ddot{q}_r^t + \sum^{i} C_{ri}^t \dot{q}_i^t + \sum^{i} K_{ri}^t \Delta q_i^t = f_r^t - \sum_{s=0}^{t-\Delta t}\left(\sum^{i} K_{ri}^s \Delta q_i^s\right)} \qquad (3A.9\text{-}2)$$

where:

C_{ri}^{t} = Current updated damping coefficient

K_{ri}^{t} = Current updated member stiffness

$$\boxed{f_{ri}^{t} - \sum_{s=0}^{t-\Delta t}\left(\sum^{i} K_{ri}^{s}\Delta q_{ri}^{s}\right)}$$ = Forcing function

which are calculated based on the current deformed structure[4] and assumed constant through the time step, Δt.

The displacement and member forces are thus given by

$$q_r = \sum_{s=0}^{t} \Delta q_r^s$$

$$Q_r = \sum_{s=0}^{t} \left(\sum^{i} K_{ri}^{s} \Delta q_{ri}^{s}\right) \qquad (3A.9\text{-}3)$$

The second order differential system equations (Equation 3A.9-3) are solved by a linear acceleration implicit method[5].

Since no external loading is applied to a member between nodes, the maximum value of the internal force acting on a member occurs at its end sections. The transition from the elastic to the fully plastic state is disregarded, and the end sections are assumed to remain linearly elastic up to the full plastic yield surface. The yield surface is defined by a scalar function of the internal member forces, Q, of the form[6,7,8]

$$\phi\ (Q^{t}) = 1$$

Here the function Φ is obtained by integrating the stress across the section with the stress fully developed over the section and satisfying the Von-Mises (or Tresca) yield criterion.

$$\sigma^2 + \gamma^2 \tau^2 = \sigma_y^2$$

where:

σ = normal stress

τ = shear stress

σ_y = yield stress in simple tension

γ^2 = 3 (Von Mises) or 4 (Tresca)

Thus, the function Φ depends on the shape of the cross section and the force components being considered.

For a frame structure, the yielding normally occurs due to either a predominant bending moment or to a predominant tension or compression. Thus, two plastic models are provided:

1. Bending predominant members

   Since a section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic.

   b. End A is yielding and B is elastic.

   c. End A is elastic and B is yielding.

   d. Both ends A and B are yielding.

   A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear curve(9,10). In situations where force reversal occurs, the stiffness of the hinged member is restored, providing unloading along the elastic line (isotropic strain hardening model).

2. Tension or compression predominant members

   There are only two possible states:

   a. Entire member is elastic.

   b. Entire member is plastic.

When the member yields, the member is elastic but Young's modulus is replaced by a plastic tangent modulus and the force-displacement curve follows a bilinear curve. If the member unloads, the elastic modulus is restored.

The damping forces are determined approximately by two sets of dampers, one associated with the member stiffness and the other with the masses[11,12]. The damping forces are assumed to be proportional to relative velocity in the first case and absolute velocity in the latter case. Namely, the damping coefficient, C, in Equation 3A.9-1 is given by

$$C_{ri} = C^k K_{ri} + C^m M_r \delta_{ri}$$

where:

δ_{ri} = Kronecker delta; the values of C and C are assumed constant and may be determined either by an approximate analytical approach or from experimental data.

3A.9.2 Program Verification

Stone & Webster sponsored an experimental investigation performed by the Massachusetts Institute of Technology (MIT)[13]). The problem consisted of the cantilevered pipe (Fig. 3A.9-1) subjected to an impulsive load at its free end. The impulse is imparted by the detonation of a sheet of high explosive, separated from the pipe by a buffer material. A nearly uniform initial velocity is produced in the loaded region and is determined by high-speed photography.

This problem was analyzed by LIMITA II. The results were compared with experimental data and output from another computer program, DINASAW.

The stress-strain curves used in the LIMITA II and DINASAW calculations are shown on Fig. 3A.9-2 with the experimentally derived curve. Fig. 3A.9-3 shows the lumped-mass models used for both computer solutions. The impulsive load, idealized as initial nodal velocities, is also shown on Fig. 3A.9-3. Time history plots of the x and y displacements of the free end of the pipe for the LIMITA II and DINASAW runs are shown on Fig. 3A.9-4 and 3A.9-5, respectively. The moment reaction at the clamped end of the pipe is shown on Fig. 3A.9-6. A comparison of the permanent pipe deformations predicted by the experiment, DINASAW, and LIMITA II is illustrated on Fig. 3A.9-7. As shown on Fig. 3A.9-4 through 3A.9-7, agreement is good in all cases.

References - Section 3A.9

1. Martin, H. C. Introduction to Matrix Methods of Structural Analysis. McGraw-Hill Book Company, Inc., New York, NY, 1966.

2. Przemieniecki, J. S. Theory of Matrix Structural Analysis. McGraw-Hill Book Company, Inc., New York, NY, 1968.

3. Clough, R. W. and Wilson, E. L. Dynamic Response by Step-by-Step Matrix Analysis. Symposium on Use of Computers in Civil Engineering, Lisbon, Portugal, 1962, p 45.1-45.14.

4. Martin, H. C. On the Derivation of Stiff Matrices for the Analysis of Large Deflection and Stability Problems. Proc. Conf. Matrix Methods Structure Mech, Wright-Patterson Air Force Base, Ohio, October 26-28, 1965, AFFBL TR 66-80, 1966.

5. Hildebrand, F. B. Introduction to Numerical Analysis. McGraw-Hill Book Company, Inc., New York, NY, 1956.

6. Hodge, P. G. Plastic Analysis of Structures. McGraw-Hill Book Company, Inc., New York, NY, 1959.

7. Neal, B. G. The Effect of Shear and Normal Forces on the Fully Plastic Moment of a Beam of Rectangular Cross Section. Journal of Applied Mechanics, June 1961, p 269-274.

8. Stokey, W. F.; Peterson, D. B.; and Wruder, R. A. Limit for Tubes Under Internal Pressure, Bending Moment, Axial Force and Torsion. Nuclear Engineering and Design, 4, North-Holland Publishing Company, Amsterdam, 1966, p 193-261.

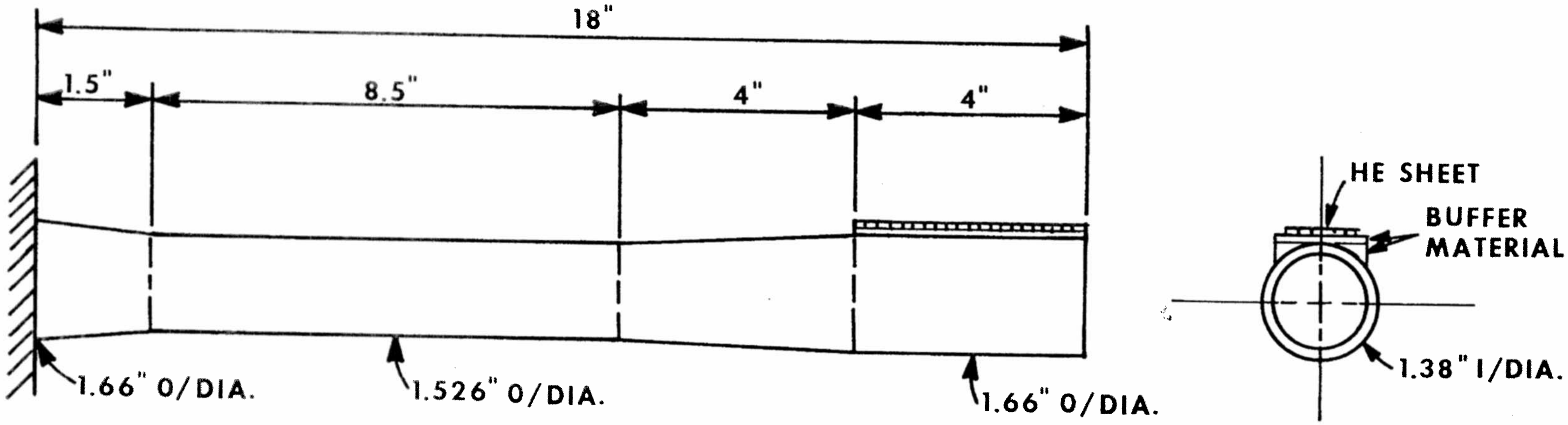
9. Clough, R. W.; Benuska, K. L.; and Wilson, E. L. Inelastic Earthquake Response of Tall Buildings. Proceedings of the Third World Conference on Earthquake Engineering, Vol II, Auckland and Wellington, New Zealand, January 1965, p 68-69.

10. Giberson, M. F. The Response of Non-Linear Multi-Story Structures Subjected to Earthquake Excitation. Earthquake Engineering Research Lab, California Institute of Technology, Pasadena, CA, June 1967.

11. Wilson, E. L. A Computer Program for the Dynamic Stress Analysis of Underground Structures. Report to Waterways Experimental Station, U.S. Army Corps of Engineers, Report No. 68-1, Structural Engineering Laboratory, University of California, Berkeley, CA, January 1968.

12. Biggs, J. M. Introduction to Structural Dynamics. McGraw-Hill Book Company, Inc., New York, NY, 1964.

13. Pirotin, S. O. and East, G. H. Large Deflector, Elastic-Plastic Response of Piping: Experiment, Analysis, and Application. Transactions of the Fourth SMIRT Conference, Paper F3/1, San Francisco, CA, August 1977.



PIPE MATERIAL: A106 STEEL.

NOTE:
NOT DRAWN TO SCALE.

FIGURE 3A.9-1

CANTILEVER PIPE
USED IN MIT EXPERIMENT

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

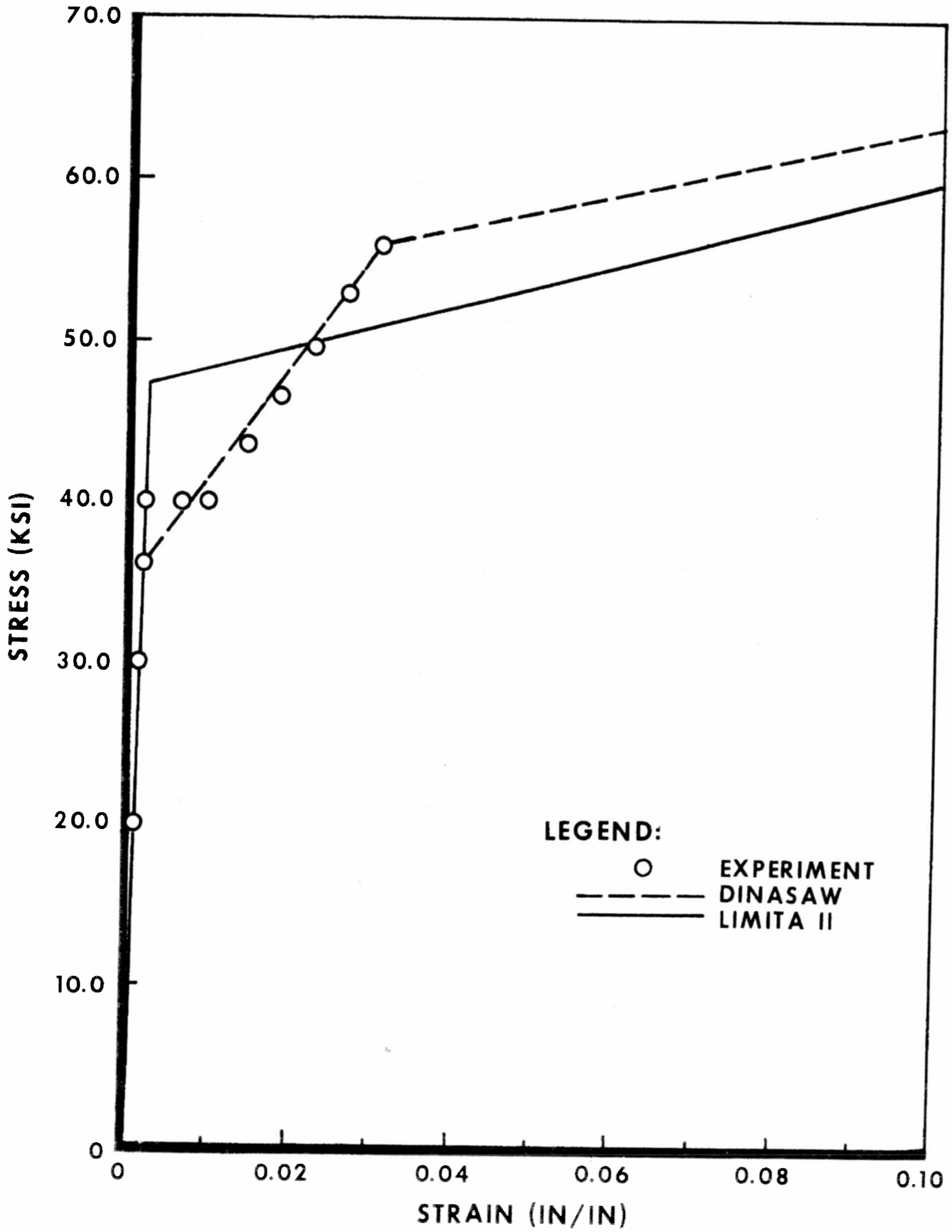
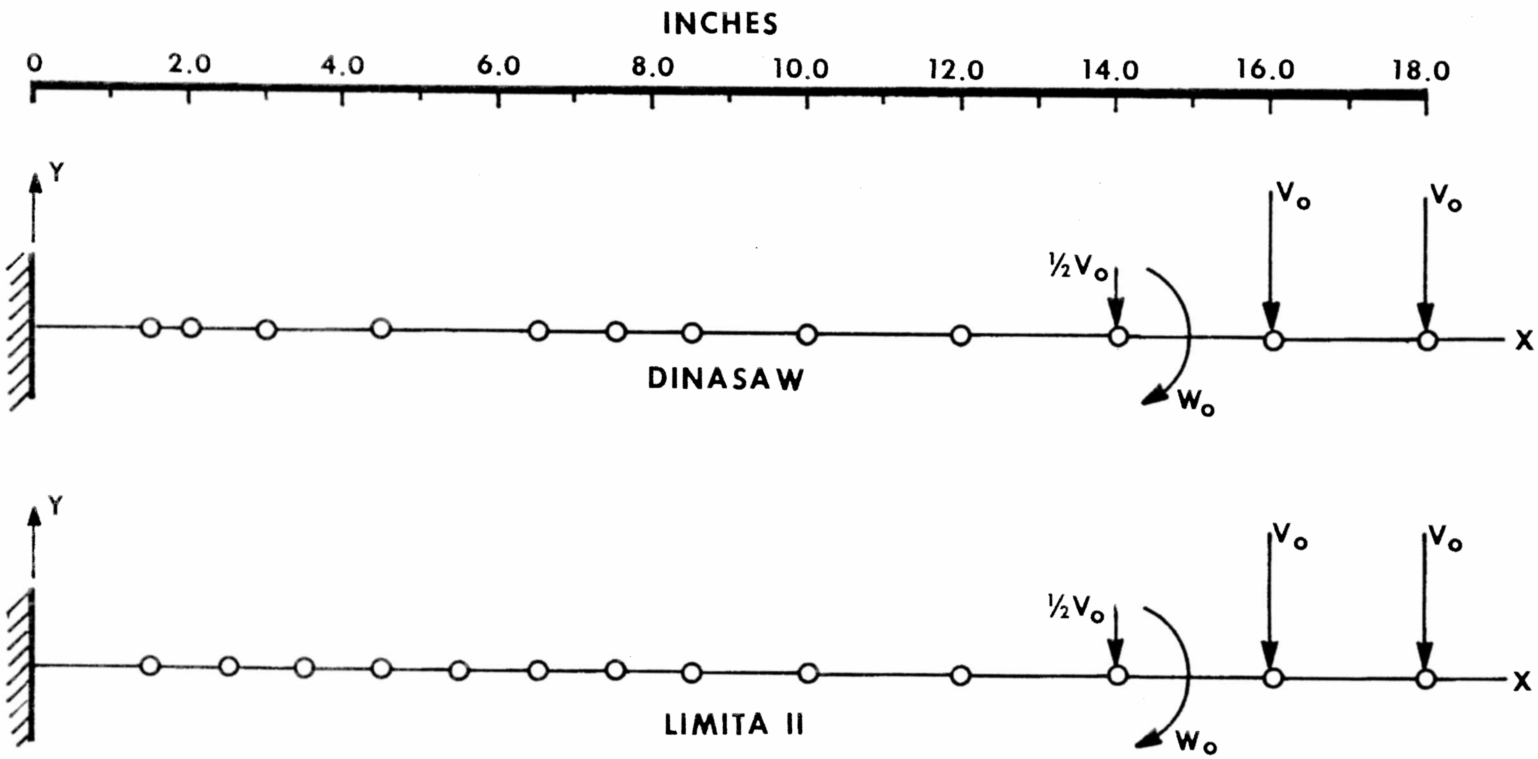


FIGURE 3A.9-2

MIT EXPERIMENT
STRESS-STRAIN CURVES

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT



NOTES:

V_o =1,400 IN/SEC

W_o =600 RAD/SEC

FIGURE 3A.9-3

DINASAW AND LIMITA II LUMPED MASS MODELS

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT

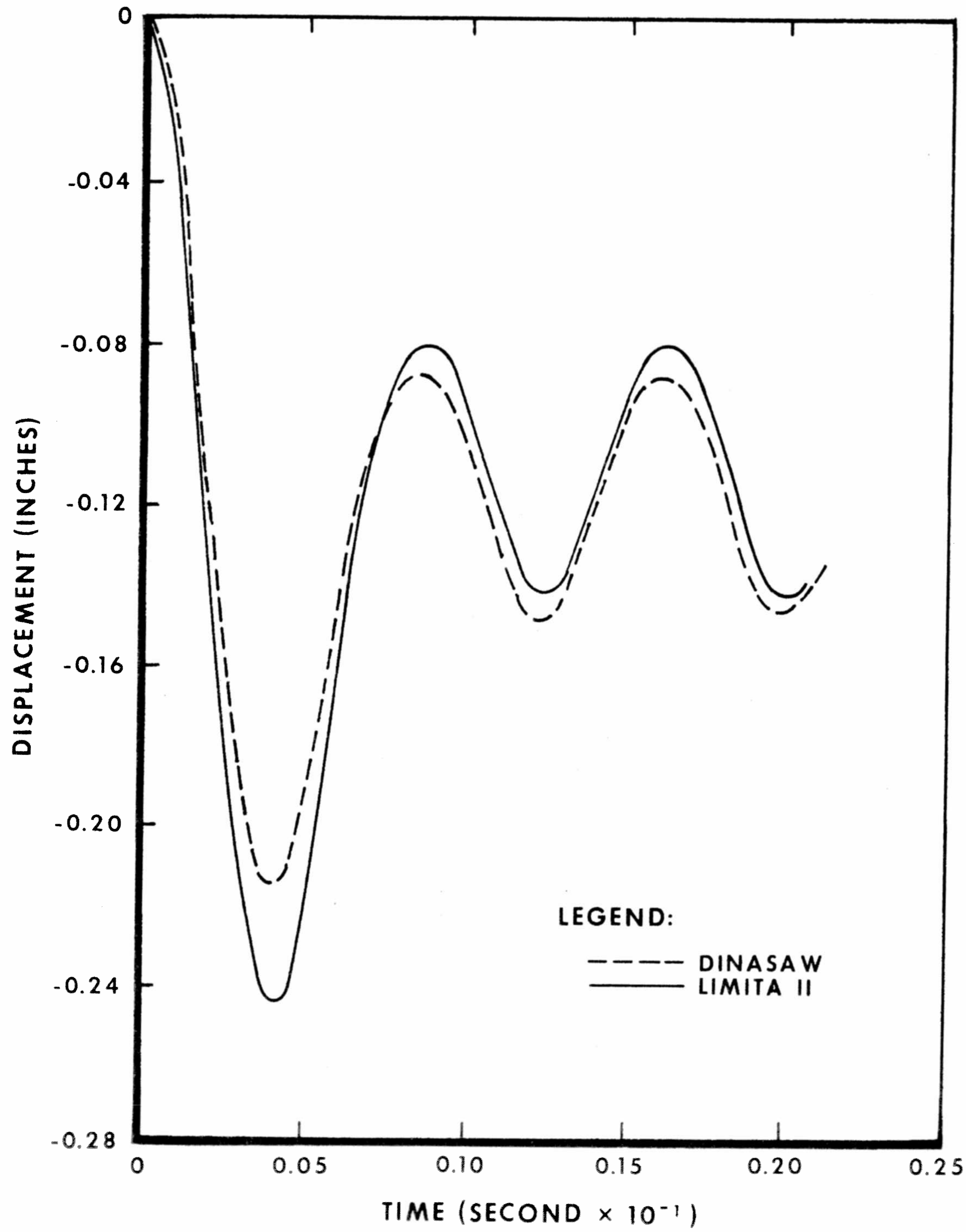


FIGURE 3A.9-4

TIME HISTORY PLOTS OF THE X-DISPLACEMENT AT THE FREE END

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT

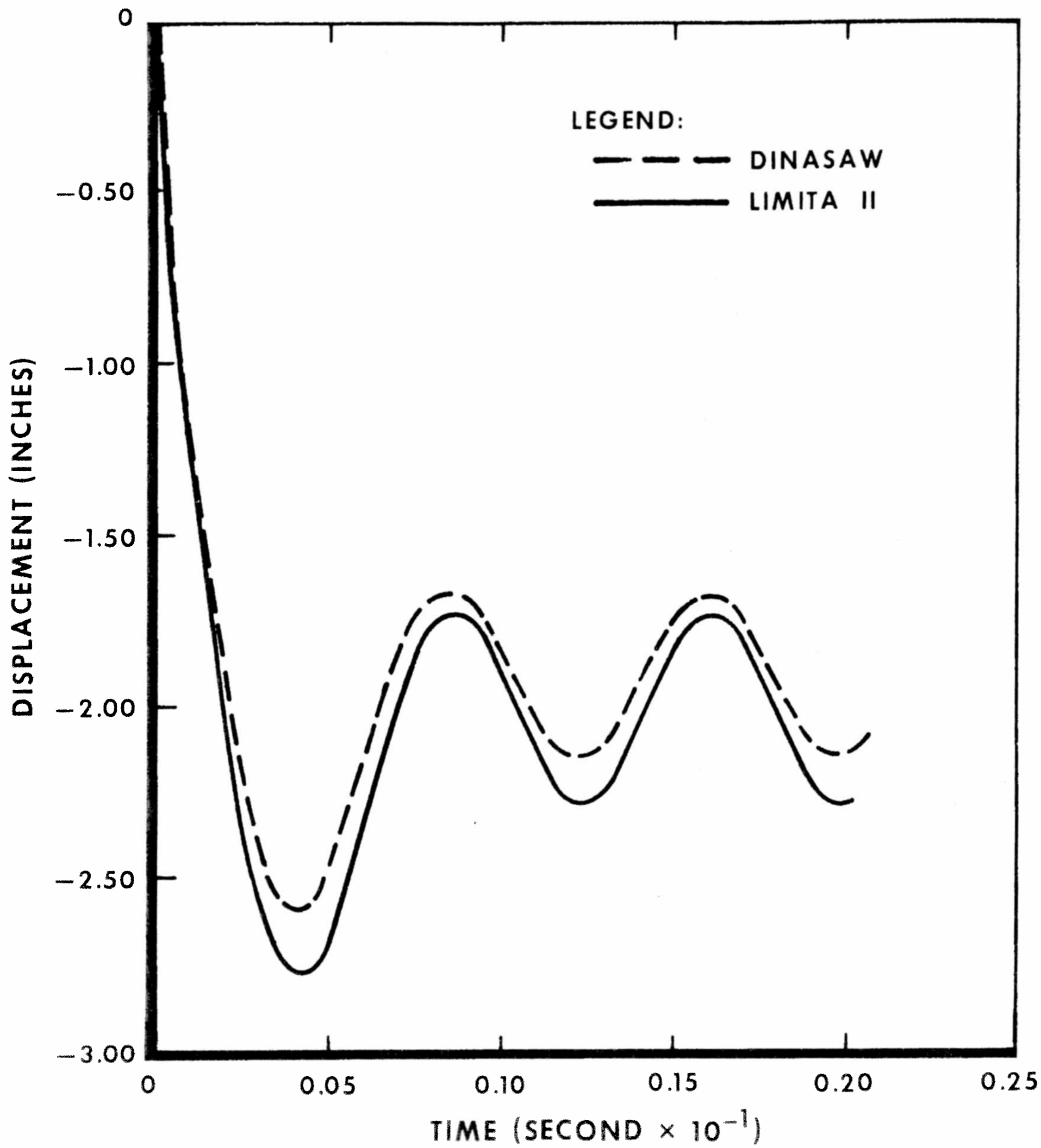


FIGURE 3A.9-5

TIME HISTORY PLOTS OF THE Y-DISPLACEMENT AT THE FREE END

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

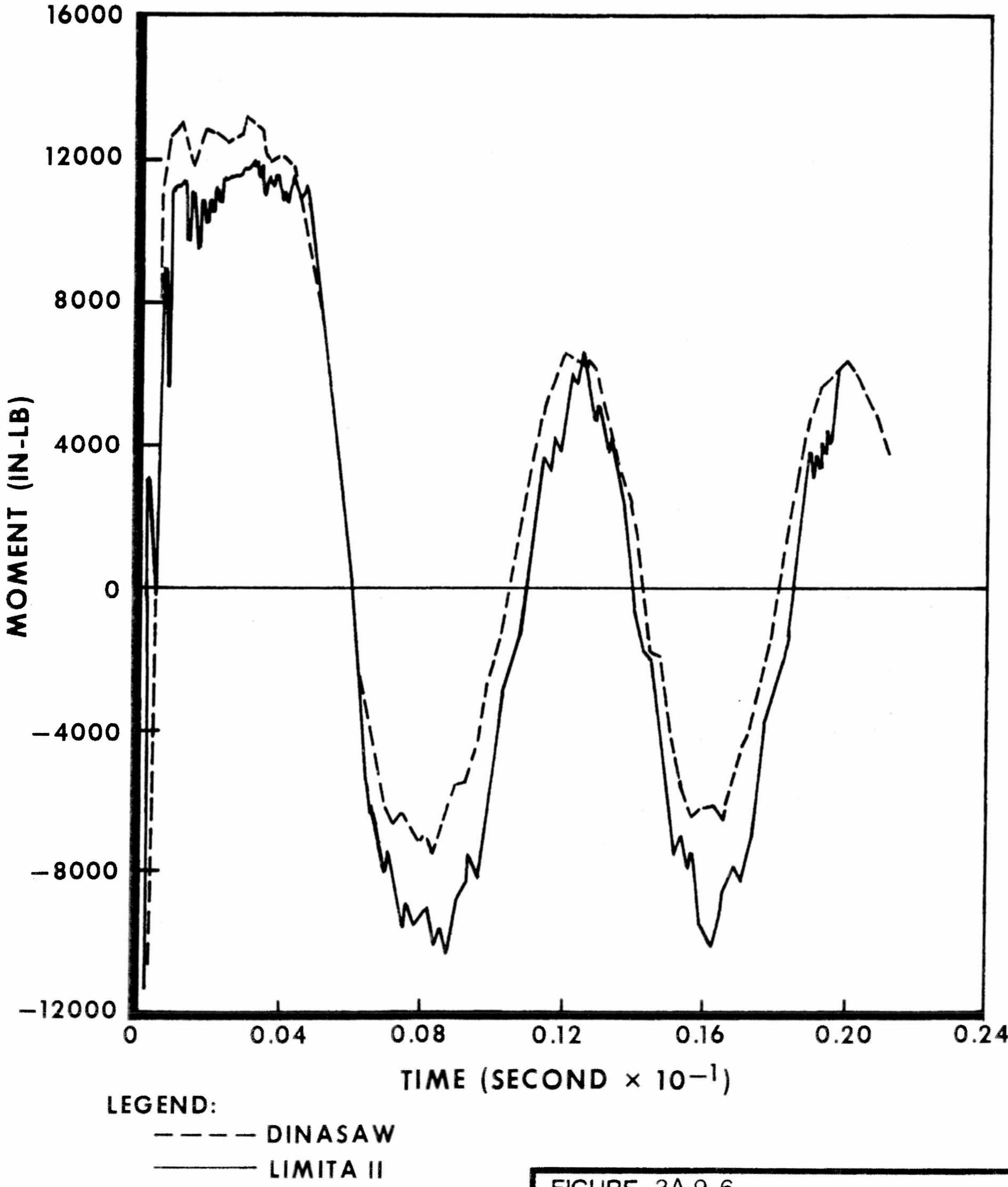


FIGURE 3A.9-6

TIME HISTORY PLOTS OF THE MOMENT REACTION AT THE CLAMPED END

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

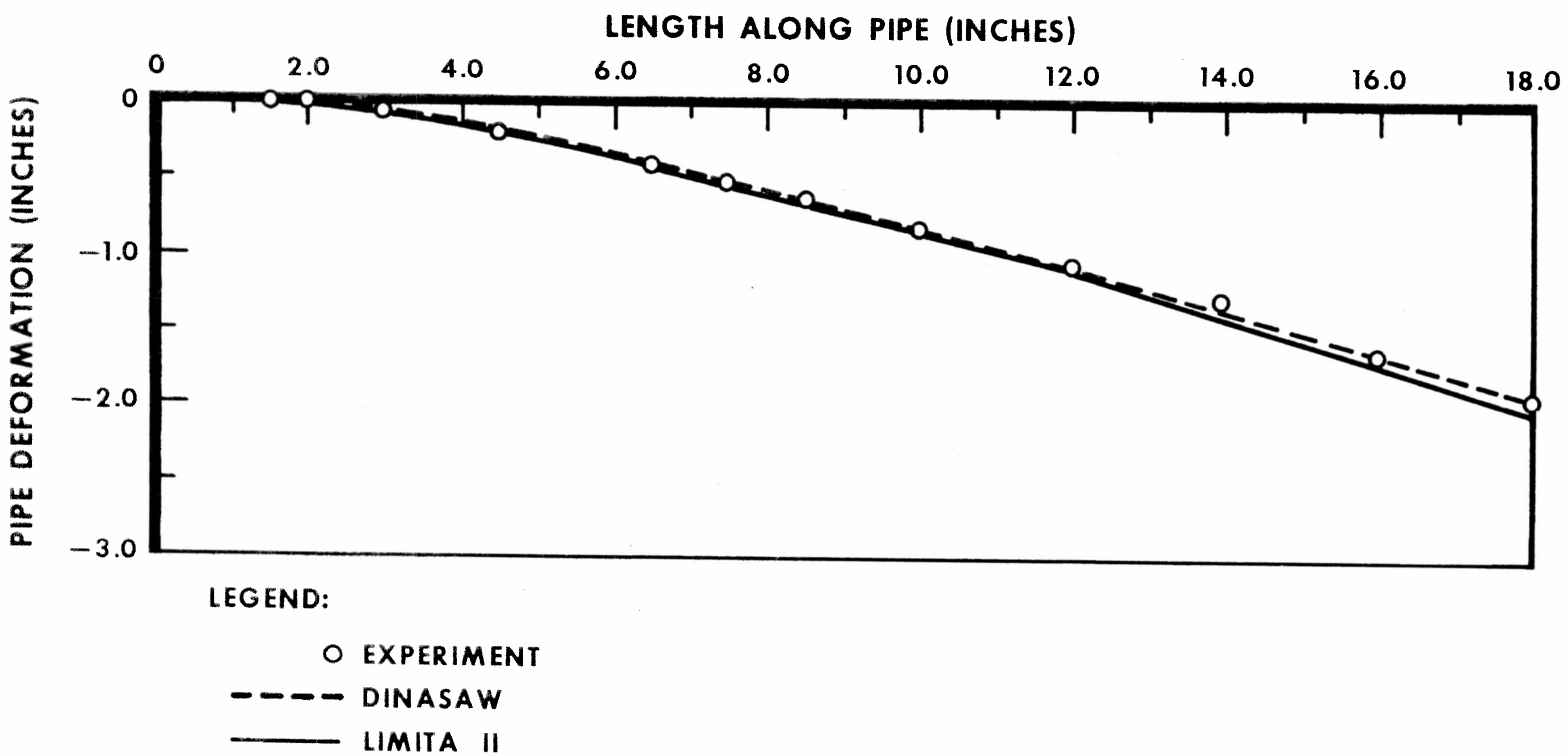


FIGURE 3A.9-7

PERMANENT PIPE DEFORMATION FOR THE MIT EXPERIMENT

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

3A.10 LIMITA III

3A.10.1 General Description

LIMITA III is a computer code which predicts the nonlinear, dynamic response of three-dimensional structures. Its formulation is identical to that of LIMITA II, with the exception that the equations are applicable to a general three-dimensional problem. For a space frame, yielding normally occurs due to either a predominant bending moment or a predominant torsion. Therefore, two plastic models are provided:

1. Bending Yield Model

   Since a beam section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic.

   b. End A is plastic, end B is elastic.

   c. End A is elastic, end B is plastic.

   d. Both ends A and B are plastic.

   A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear curve(1,2). In situations where force reversal occurs, the elastic stiffness of the hinged member is restored, providing elastic unloading (isotropic strain hardening model).

2. Torsional Yield Model

   There are only two possible states:

   a. Entire member is elastic.

   b. Entire member is plastic.

   When the member yields, the member elastic modulus is replaced by a plastic tangent modulus and the force-displacement relation follows a bilinear curve. If the member unloads, the elastic modulus is restored.

The damping forces are approximated by two sets of dampers, one associated with the member stiffness and the other with the masses(3,4). In the former, the forces are proportional to relative velocity; in the latter, they are proportional to absolute velocity. That is, the damping matrix is given by

$$[C] = Ck[K] + Cm[M]$$

The values of Ck and Cm are assumed constant and may be determined either by an approximate analytical approach or from experimental data.

3A.10.2 Program Verification

3A.10.2.1 Elastic Example

As a checkout of LIMITA III, a space frame (Fig. 3A.10-1) is considered. All members are W14x500. A step load of 30 kips is applied vertically at joint 6. This problem was analyzed by LIMITA III and ICES STRUDL II elastically. The results of displacements and moment Z at joint 6 were plotted against each other on Fig. 3A.10-2 and 3A.10-3, respectively. As shown, there is excellent agreement.

3A.10.2.2 Plastic Example

This example is provided in the description of LIMITA II in Section 3A.9.

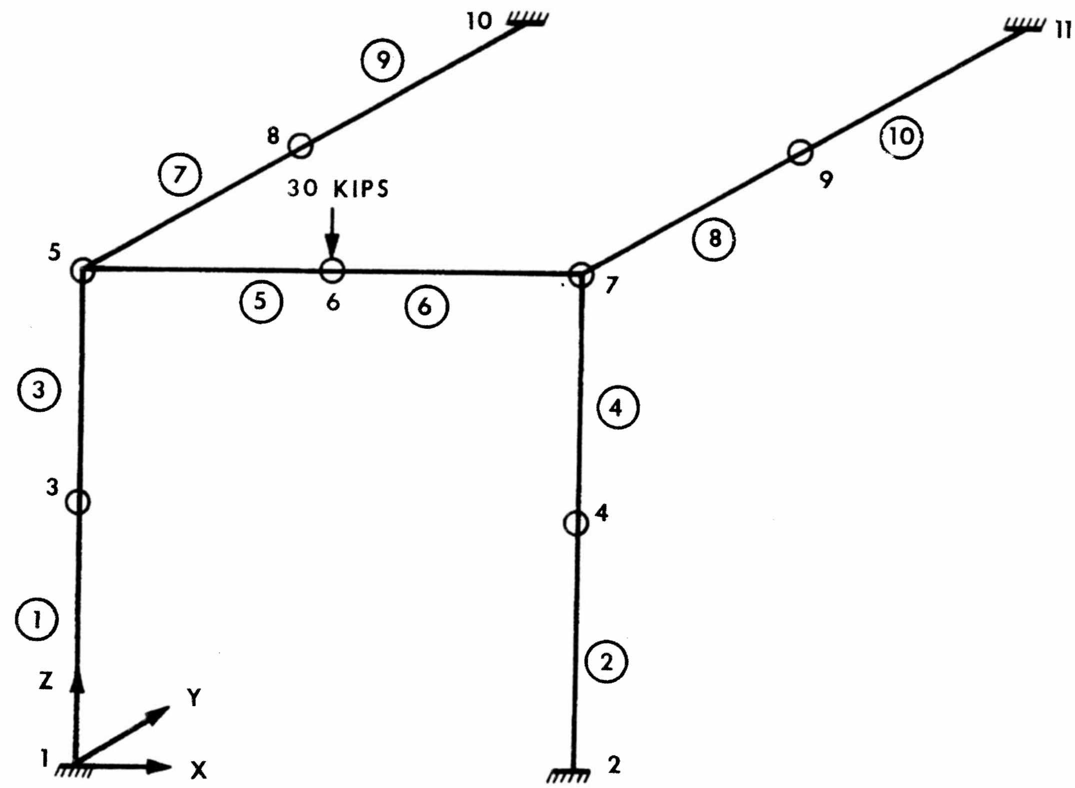
References - Section 3A.10

1. Clough, R. W.; Benuska, K. L.; and Wilson, E. L. Inelastic Earthquake Response of Tall Buildings. Proceedings of the Third World Conference on Earthquake Engineering, Vol II, Auckland and Wellington, New Zeland, January 1965, p 68-69.

2. Giberson, M. F. The Response of Non-Linear, Multi-Story Structures Subjected to Earthquake Excitation. Earthquake Engineering Research Lab, California Institute of Technology, Pasadena, CA, June 1967.

3. Collins, T. and Witmer, E. Application of the Collision Imparted Velocity Method for Analyzing the Responses of Containment and Reflector Structures to Engine Rotor Fragment Impact. Aeroelastic and Structures Research Laboratory, Department of Aeronautics and Astronautics, Massachusetts Institute of Technology, August 1973.

4. Wilson, E. L. A Computer Program for the Dynamic Stress Analysis of Underground Structures. Report to Waterways Experimental Station, U.S. Army Corps of Engineers, Report No. 68-1, Structural Engineering Laboratory, University of California, Berkeley, CA, January 1968.



NOTE:
ALL MEMBERS W14 × 500

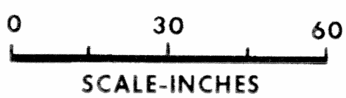


FIGURE 3A.10-1

LIMITA III ELASTIC SPACE FRAME

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

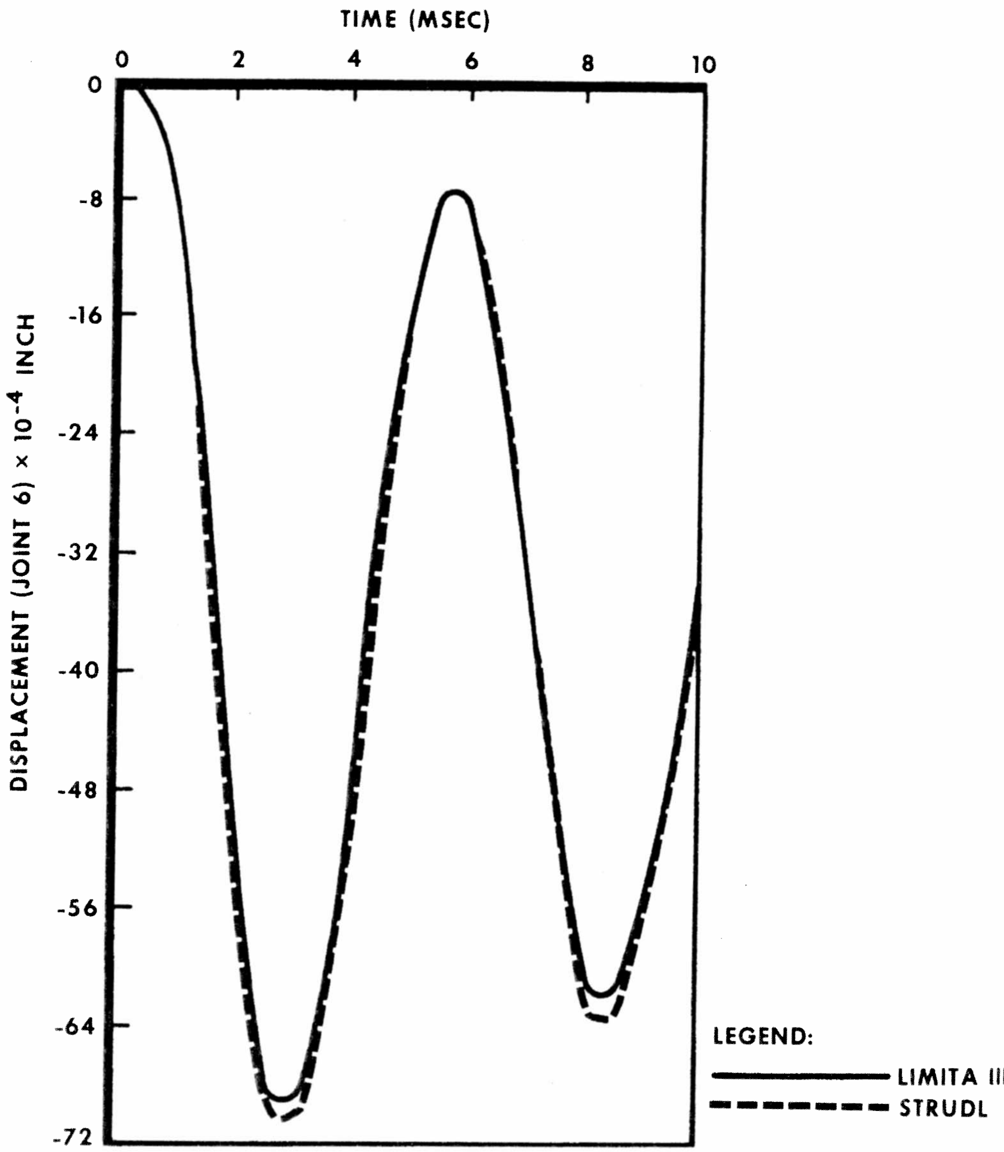


FIGURE 3A.10-2

LIMITA III DISPLACEMENT
TIME HISTORY AT JOINT 6

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

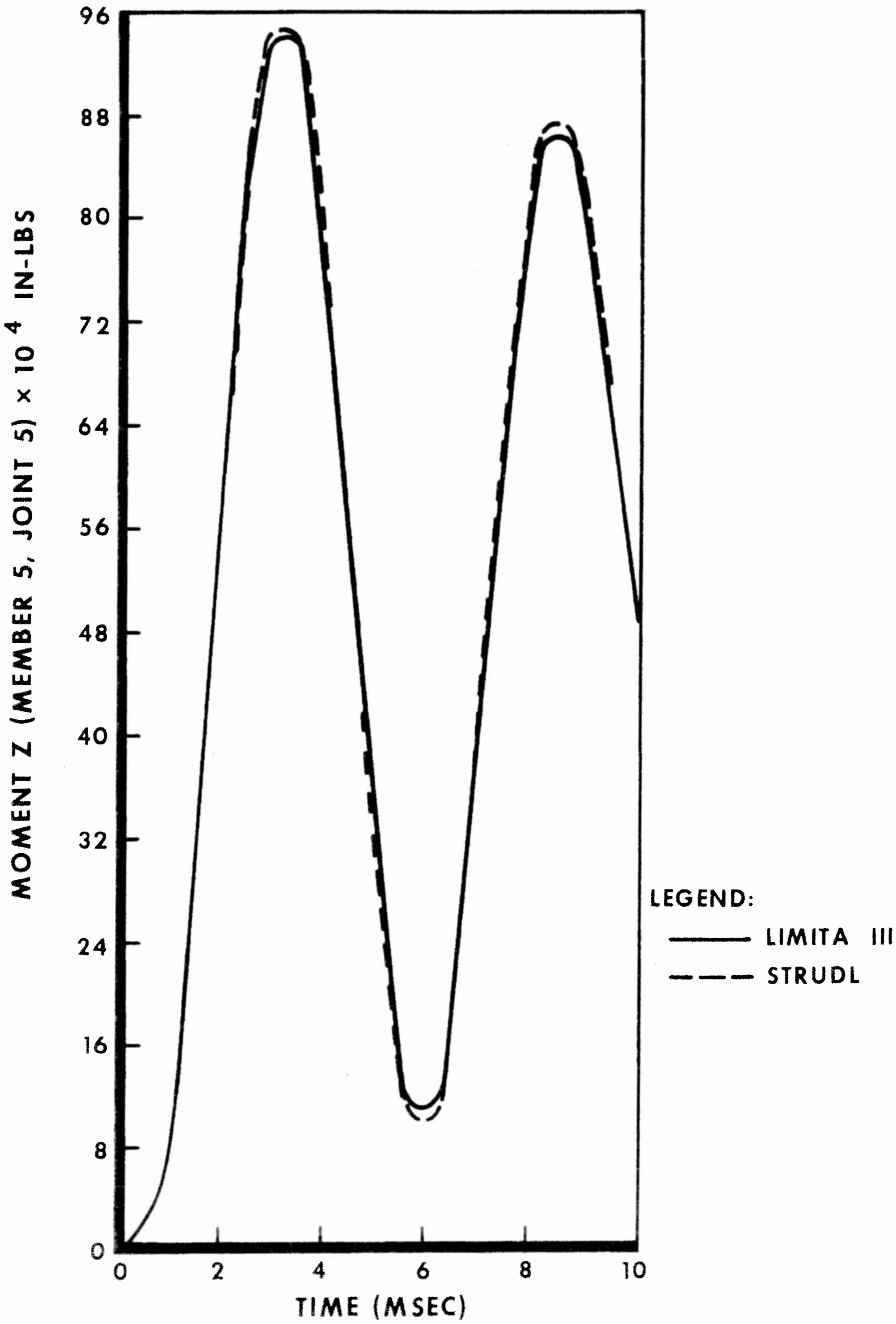


FIGURE 3A.10-3

LIMITA III TIME HISTORY OF MOMENT Z IN MEMBER 5, JOINT 6

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT

3A.11 STARDYNE

The STARDYNE Structural Analysis System, written by Mechanics Research, Inc., of Los Angeles, California, is a fully warranted and documented computer program available at Control Data Corporation.

The MRI STARDYNE System consists of a series of compatible, digital computer programs designed to analyze linear and nonlinear elastic structural methods. The system encompasses the full range of static and dynamic analyses.

The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal-applied loads and prescribed displacements.

Utilizing the normal mode technique, linear dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady-state harmonic, random, and shock spectra excitation types. Dynamic response results can be presented as structural deformations and internal member loads.

The nonlinear dynamic analysis program is integrated in the rest of the STARDYNE system. The equations of motion for the linear portion of the structural model are generated and modified to account for the nonlinear springs. The resulting nonlinear equations of motion are directly integrated, using either the Newmark or Wilson implicit integration operators. The user may enter sets of structural loadings, which vary with time, and specify time points at which the program is to output the structural response.

This computer program is considered verified by constant use and by the vendor's original documentation and qualification.

3A.12 BIJLAARD

3A.12.1 General Description

This computer code performs various analyses on tanks and pressure vessels. All of the analyses are concerned with local stresses at penetrations. Typical problems which can be handled include the following:

1. Applied load stresses at vessel-nozzle junction for:
    a. Rigid attachment to cylinder
    b. Rigid attachment to sphere
    c. Hollow attachment to sphere
2. Pressure discontinuity analysis for thin shell interaction
3. Allowable load functions on nozzles for each case
4. Area compensation analysis in accordance with the ASME Code
5. Maximum forces on supports of vessel based on allowable loads on nozzles.

Local stresses due to nozzle loads are found by the method prescribed by P. P. Bijlaard[(1)]. The method prescribed by Johns and Orange is used for pressure discontinuity stresses[(2)].

3A.12.2 Program Verification

A sample problem of a thin-walled cylindrical vessel is subjected to applied loads from a rigid cylindrical attachment. This problem may be solved using Johns and Orange's method[(2)].

A summary of the parameters and results of the manual calculations is shown on Fig. 3A.12-1. The computer calculations for the same problem are summarized on Fig. 3A.12-2. As shown on these figures, the computer results are very close to the exact results. Therefore, this problem serves to verify the accuracy of the Vessel Penetration Analysis.

References - Section 3A.12

1. Wichman, K. R.; Hopper, A. G.; and Mershon, J. L. Local Stresses in Spherical and Cylindrical Shells Due to External Loading. Welding Research Council Bulletin, WRC-107, 1965.

2. Johns, R. H. and Orange, T. W. Theoretical Elastic Stress Distribution Arising from Discontinuities and Edge Loads in Several Shell Type Structures. NASA Technical Report R-103, 1961.

1. APPLIED LOADS

Radial Load	P	= 1000 lb.
Circ. Moment	M_C	= 1000 in. lb.
Long. Moments	M_L	= 1000 in. lb.
Torsion Moment	M_T	= 1000 in. lb.
Shear Load	V_C	= 1000 lb.
Shear Load	V_L	= 1000 lb.

2. GEOMETRY

Vessel Thickness	T	= 0.375 in
Attachment Radius	r_o	= 8.0 in.
Vessel Radius	R_m	= 19.1875 in.

3. GEOMETRIC PARAMETERS

$$\gamma = \frac{R_m}{T} \quad 51.167$$

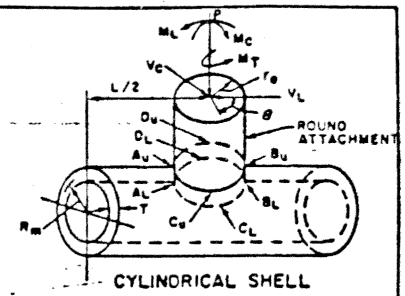
$$\beta = (0.875)\frac{r_o}{R_m} \quad 0.3648$$

STRESS CONCENTRATION DUE TO:

a) Membrane Load $K_n = 1.0$

b) Bending Load $K_b = 1.0$

NOTE: Enter all force values in accordance with sign convention.



FROM FIG. (*)	READ CURVES FOR	COMPUTE ABSOLUTE VALUES OF STRESS AND ENTER RESULTS	STRESSES - if load is opposite that shown, reverse signs shown							
			A_U	A_L	B_U	B_L	C_U	C_L	D_U	D_L
3C	$\frac{N\phi}{P/R_m} = 1.78$	$K_n\left(\frac{N\phi}{P/R_m}\right)\cdot\frac{P}{R_m T} = 247$	-247	-247	-247	-247	-247	-247	-247	-247
1C	$\frac{M\phi}{P} = 0.023$	$K_b\left(\frac{M\phi}{P}\right)\cdot\frac{6P}{T^2} = 981$	-981	+981	-981	+981	-981	+981	-981	+981
3A	$\frac{N\phi}{M_C/R_m^2\beta} = 1.41$	$K_n\left(\frac{N\phi}{M_C/R_m^2\beta}\right)\cdot\frac{M_C}{R_m^2\beta T} = 28$					-28	-28	+28	+28
1A	$\frac{M\phi}{M_C/R_m\beta} = 0.059$	$K_b\left(\frac{M\phi}{M_C/R_m\beta}\right)\cdot\frac{6M_C}{R_m\beta T^2} = 360$					-360	+360	+360	-360
3B	$\frac{N\phi}{M_L/R_m^2\beta} = 2.63$	$K_n\left(\frac{N\phi}{M_L/R_m^2\beta}\right)\cdot\frac{M_L}{R_m^2\beta T} = 52$	-52	-52	+52	+52				
1B or 1B-1	$\frac{M\phi}{M_L/R_m\beta} = 0.0067$	$K_b\left(\frac{M\phi}{M_L/R_m\beta}\right)\cdot\frac{6M_L}{R_m\beta T^2} = 41$	-41	+41	+41	-41				
Add algebraically for summation of ϕ stresses $\sigma\phi$			-1321	+723	-1135	+745	-1616	+1066	-840	+402
4C	$\frac{N_x}{P/R_m} = 4.4$	$K_n\left(\frac{N_x}{P/R_m}\right)\cdot\frac{P}{R_m T} = 612$	-612	-612	-612	-612	-612	-612	-612	-612
2C	$\frac{M_x}{P} = 0.0088$	$K_b\left(\frac{M_x}{P}\right)\cdot\frac{6P}{T^2} = 375$	-375	+375	-375	+375	-375	+375	-375	+375
4A	$\frac{N_x}{M_C/R_m^2\beta} = 5.0$	$K_n\left(\frac{N_x}{M_C/R_m^2\beta}\right)\cdot\frac{M_C}{R_m^2\beta T} = 99$					-99	-99	+99	+99
2A	$\frac{M_x}{M_C/R_m\beta} = 0.0235$	$K_b\left(\frac{M_x}{M_C/R_m\beta}\right)\cdot\frac{6M_C}{R_m\beta T^2} = 143$					-143	+143	+143	-143
4B	$\frac{N_x}{M_L/R_m^2\beta} = 1.52$	$K_n\left(\frac{N_x}{M_L/R_m^2\beta}\right)\cdot\frac{M_L}{R_m^2\beta T} = 30$	-30	-30	+30	+30				
2B or 2B-1	$\frac{M_x}{M_L/R_m\beta} = 0.01$	$K_b\left(\frac{M_x}{M_L/R_m\beta}\right)\cdot\frac{6M_L}{R_m\beta T^2} = 61$	-61	+61	+61	-61				
Add algebraically for summation of X stresses σ_x			-1078	-206	-896	-268	-1229	-193	-745	-281
Shear stress due to torsion M_T		$\tau\phi_x = \tau_x\phi = \frac{M_T}{2\pi r_o^2 T} = 7$	+7	+7	+7	+7	+7	+7	+7	+7
Shear stress due to load V_C		$\tau_x\phi = \frac{V_C}{\pi r_o T} = 106$	+106	+106	-106	-106				
Shear stress due to load V_L		$\tau_x\phi = \frac{V_L}{\pi r_o T} = 106$					-106	-106	+106	+106
Add algebraically for summation of shear stresses τ			+113	+113	-99	-99	-99	-99	+113	+113

SOURCE:

* WICHMAN, K.R.; HOPPER, A.G.; AND MERSHON, J.L. LOCAL STRESSES IN SPHERICAL AND CYLINDRICAL SHELLS DUE TO EXTERNAL LOADING. WELDING RESEARCH COUNCIL BULLETIN, WRC-107, 1965.

FIGURE 3A.12-1

SUMMARY OF MANUAL CALCULATIONS FOR LOCAL STRESSES IN CYLINDRICAL SHELLS

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT

1. APPLIED LOADS

Radial Load	P	= 1000 lb.
Circ. Moment	M_C	= 1000 in. lb.
Long. Moments	M_L	= 1000 in. lb.
Torsion Moment	M_T	= 1000 in. lb.
Shear Load	V_C	= 1000 lb.
Shear Load	V_L	= 1000 lb.

GEOMETRY

Vessel Thickness	T	= 0.375 in
Attachment Radius	r_o	= 8.0 in.
Vessel Radius	R_m	= 19.1875 in.

3. GEOMETRIC PARAMETERS

$\gamma = \frac{R_m}{T}$ 51.167

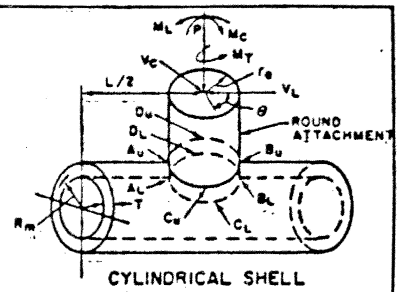
$\beta = (0.875)\cdot\frac{r_o}{R_m}$ 0.3648

STRESS CONCENTRATION DUE TO:

a) Membrane Load Kn = 1.0

b) Bending Load Kb = 1.0

NOTE: Enter all force values in accordance with sign convention.



FROM FIG. (*)	READ CURVES FOR	COMPUTE ABSOLUTE VALUES OF STRESS AND ENTER RESULTS	STRESSES - if load is opposite that shown, reverse signs shown							
			A_u	A_L	B_u	B_L	C_u	C_L	D_u	D_L
3C	$\frac{N\phi}{P/R_m} = 1.77$	$Kn\left(\frac{N\phi}{P/R_m}\right)\cdot\frac{P}{R_m T} = 246$	-246	-246	-246	-246	-246	-246	-246	-246
1C	$\frac{M\phi}{P} = 0.022$	$Kb\left(\frac{M\phi}{P}\right)\cdot\frac{6P}{T^2} = 937$	-937	+937	-937	+937	-937	+937	-937	+937
3A	$\frac{N\phi}{M_C/R_m^2\beta} = 1.41$	$Kn\left(\frac{N\phi}{M_C/R_m^2\beta}\right)\cdot\frac{M_C}{R_m^2\beta T} = 28$					-28	-28	+28	+28
1A	$\frac{M\phi}{M_C/R_m\beta} = 0.059$	$Kb\left(\frac{M\phi}{M_C/R_m\beta}\right)\cdot\frac{6M_C}{R_m\beta T^2} = 357$					-357	+357	+357	-357
3B	$\frac{N\phi}{M_L/R_m^2\beta} = 2.62$	$Kn\left(\frac{N\phi}{M_L/R_m^2\beta}\right)\cdot\frac{M_L}{R_m^2\beta T} = 52$	-52	-52	+52	+52				
1B or 1B-1	$\frac{M\phi}{M_L/R_m\beta} = 0.0066$	$Kb\left(\frac{M\phi}{M_L/R_m\beta}\right)\cdot\frac{6M_L}{R_m\beta T^2} = 40$	-40	+40	+40	-40				
Add algebraically for summation of ϕ stresses $\sigma\phi$			-1275	+679	-1091	+703	-1568	+1020	-798	+362
4C	$\frac{N_x}{P/R_m} = 4.4$	$Kn\left(\frac{N_x}{P/R_m}\right)\cdot\frac{P}{R_m T} = 614$	-614	-614	-614	-614	-614	-614	-614	-614
2C	$\frac{M_x}{P} = 0.0087$	$Kb\left(\frac{M_x}{P}\right)\cdot\frac{6P}{T^2} = 373$	-373	+373	-373	+373	-373	+373	-373	+373
4A	$\frac{N_x}{M_C/R_m^2\beta} = 5.0$	$Kn\left(\frac{N_x}{M_C/R_m^2\beta}\right)\cdot\frac{M_C}{R_m^2\beta T} = 99$					-99	-99	+99	+99
2A	$\frac{M_x}{M_C/R_m\beta} = 0.0233$	$Kb\left(\frac{M_x}{M_C/R_m\beta}\right)\cdot\frac{6M_C}{R_m\beta T^2} = 142$					-142	+142	+142	-142
4B	$\frac{N_x}{M_L/R_m^2\beta} = 1.53$	$Kn\left(\frac{N_x}{M_L/R_m^2\beta}\right)\cdot\frac{M_L}{R_m^2\beta T} = 30$	-30	-30	+30	+30				
2B or 2B-1	$\frac{M_x}{M_L/R_m\beta} = 0.01$	$Kb\left(\frac{M_x}{M_L/R_m\beta}\right)\cdot\frac{6M_L}{R_m\beta T^2} = 62$	-62	+62	+62	-62				
Add algebraically for summation of X stresses σ_x			-1079	-209	-895	-280	-1228	-198	-746	-284
Shear stress due to torsion M_T		$\tau\phi_x = \tau_x\phi = \frac{M_T}{2\pi r_o^2 T} = 7$	+7	+7	+7	+7	+7	+7	+7	+7
Shear stress due to load V_C		$\tau_x\phi = \frac{V_C}{\pi r_o T} = 106$	+106	+106	-106	-106				
Shear stress due to load V_L		$\tau_x\phi = \frac{V_L}{\pi r_o T} = 106$					-106	-106	+106	+106
Add algebraically for summation of shear stresses τ			+113	+113	-99	-99	-99	-99	+113	+113

SOURCE:

* WICHMAN, K.R.; HOPPER, A.G.; AND MERSHON, J.L. LOCAL STRESSES IN SPHERICAL AND CYLINDRICAL SHELLS DUE TO EXTERNAL LOADING. WELDING RESEARCH COUNCIL BULLETIN, WRC-107, 1965.

FIGURE 3A.12-2

SUMMARY OF COMPUTERS CALCULATIONS FOR LOCAL STRESSES IN CYLINDRICAL SHELLS

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT

3A.13 LION

LION is a digital computer program which is used to solve three-dimensional transient and steady-state temperature distribution problems. The program may also consider subcooled nucleate boiling and coolant heat transfer effects. The surface conditions may be forced convection, free convection, or radiation and heat may be externally or internally generated. Input to the program consists of structural geometry, physical properties, boundary conditions, internal heat generation rates, coolant flow properties, and flow rates.

The program solves the transient heat conduction equations for a three-dimensional field using a first forward difference method. To ensure the temperature calculation stability, LION can determine the suitable time increment, if the specified input time increment is too large.

Since the original program was developed, subsequent versions have evolved to solve larger and more complex problems[1,2,3,4,5,6].

LION is a recognized program in the public domain and has been used extensively.

This computer program is considered verified by constant use and by the vendor's original documentation and qualification.

References - Section 3A.13

1. Bray, A.P. TIGER-Temperatures from Internal Generation Rates. KAPL, Schenectady, New York, NY, May 1954.

2. Bray, A.P. and MacCracken, S.J. TIGER-II-Temperatures from Internal Generation Rates. KAPL-2044, May 29, 1959.

3. Personal Communication with Mrs. M. Helme concerning TIGER-IV, Thermal Design Memorandum No. 88 (Date).

4. Briggs, D.L. TIGER-Temperatures from Internal Generation Rates. KAPL-M-EC-29, February 1963.

5. Lechliter, G.L.; Liedel, A.L.; and Schmid, J.R. Mathematics Programs Available on Philco 2000 Computer, Part II, Curve Plotting. KAPL-M-6416 (EC-40), October 1964.

6. Schmid, J.R.; Lechliter, G.L.; and Fisher, W.W. LION-Temperature Distribution for Arbitrary Shapes and Complicated Boundary Condition. KAPL-M-6532 (EC-57), Revision IV, December 1969.

3A.14 MISSILE

3A.14.1 General Description

The MISSILE Program calculates the impact probability (P_2) of postulated turbine missiles on specified targets. The solid angle method is used to calculate P_2:

$$P_2 = \frac{1}{\Omega_m} \int d\,\Omega$$

where:

Ω = Solid angle subtended by the target

Ω_m = Total solid angle subtended by all possible missile trajectories

The integral is evaluated by numerical integration, with consideration of the missile ejection velocity and the relative positions of the turbine and target (Fig. 3A.14-1).

3A.14.2 High-Trajectory Verification

Westinghouse has derived a formula to predict the probability of impact for high-trajectory missiles[(1)]. Some adjustments to the formula are necessary to enable direct comparison with the program results. The formula has been derived on the basis that the initial velocity is random and uniformly distributed between V_1 and V_2. The program uses a deterministic initial velocity. The formula may be specialized to this condition by setting V_1 equal to V_2 after applying L'Hopital's Rule. Also, the formula has been derived assuming that a missile fragment occurs in the quadrant of the target; whereas, the program assumes that a missile fragment can occur in any of the four quadrants. These differing assumptions can be reconciled by using four fragments for program input.

After making the above adjustments, the high-trajectory formula becomes:

$$P = G^2/(2\ \pi\ \Delta\ V^4)$$

where:

P = Impact probability per square foot of target

G = Acceleration of gravity (ft/sec^2)

Δ = Deflection angle range (radians)

V = Initial velocity (fps).

Comparison of the probability calculated by the formula and the results of the computer program are given in Table 3A.14-1.

3A.14.3 Low-Trajectory Verification

The probability of impact for low-trajectory missiles (LTM) calculated by this program was verified by comparison with Bush[(2)]. The LTM was identified as four fragments of the outer disk resulting from a turbine failure. The two different initial ejection velocities were 300 fps and 600 fps. The geometry is shown on Fig. 3A.14-2.

Since only half of Bush's 4,800-sq ft target lies in the reported interval ($0 \leq \delta < 25°$), only a 2,400-sq ft portion was modeled in MISSILE.

Comparison of the probability listed in Bush[(2)] and the results of the computer program is provided in Table 3A.14-1.

References - Section 3A.14

1. Westinghouse Electric Corporation. Analysis of the Probability of the Generation and Strike of Missiles from a Nuclear Turbine. Steam Turbine Division, March 1974,p 48.

2. Bush, S.H. Probability of Damage to Nuclear Components Due to Turbine Failure. Nuclear Safety, Vol 14, No. 3, May-June 1973, p 197.

TABLE 3A.14-1

MISSILE PROGRAM VERIFICATION

Comparison of High-Trajectory Probabilities

Velocity (fps)	Deflection Angle	Formula[1]	Program
300	5°	0.254E-3	0.259E-3
300	25°	0.508E-4	0.534E-4
600	5°	0.162E-4	0.16OE-4
600	25°	0.318E-5	0.330E-5

Comparison of Low-Trajectory Probabilities

Velocity (fps)	Deflection Angle	Bush[2]	Program
300	0° ≤ <25°	0.113	0.111
600	0° ≤ <25°	0.11	0.112

Sources:

1. Westinghouse Electric Corporation. Analysis of the Probability of the Generation and Strike of Missiles from a Nuclear Turbine. Steam Turbine Division, March 1974, p 48.

2. Bush, S.H. Probability of Damage to Nuclear Components Due to Turbine Failure. Nuclear Safety, Vol 14, No. 3, May-June 1973, p 197.

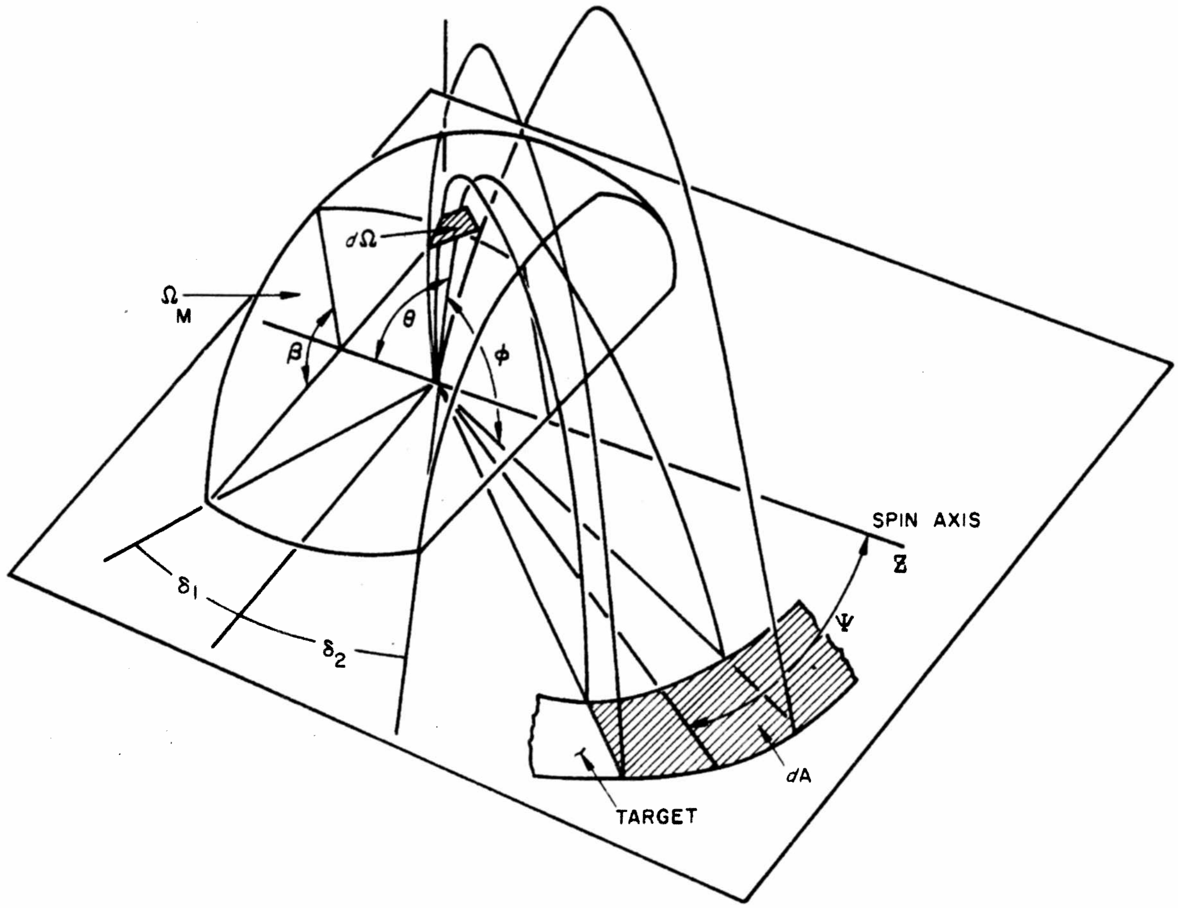


FIGURE 3A.14-1

NOMENCLATURE FOR MISSILE PROGRAM

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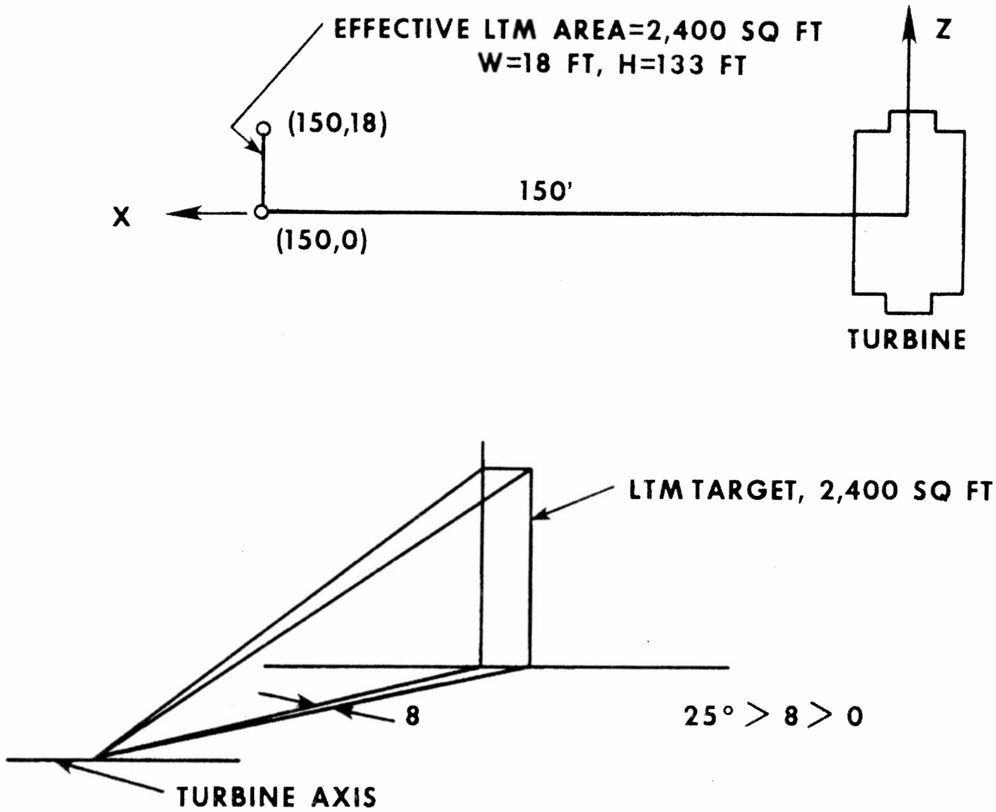

SOURCE:

BUSH S.H. PROBABILITY OF DAMAGE TO NUCLEAR COMPONENTS DUE TO TURBINE FAILURE.
NUCLEAR SAFETY, VOL 14, NO. 3
MAY-JUNE 1973.

FIGURE 3A.14-2

TARGET FOR LOW TRAJECTORY MISSILE

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3A.15 NUPIPE

NUPIPE was developed by the Nuclear Services Corporation and is fully documented. The SWEC version of NUPIPE used differs slightly from the public domain program NUPIPE in the postprocessing of the analytical results.

NUPIPE performs a linear elastic analysis of three-dimensional piping systems subjected to thermal, static, and dynamic loads. It utilizes the finite-element method of analysis with special features incorporated to accommodate specific requirements in piping analysis. In addition, it checks analytical conformance to ASME Section III and ANSI B31.1.0. This program accepts the complete geometric and physical description of the piping system, provides a complete error and coordinate check for the inputs, and computes internal forces and moments, support and equipment reactions, and displacements and stress values for a variety of loading cases.

NUPIPE has been verified with ADLPIPE[3] for thermal, weight, and response spectrum seismic analyses. The results from both programs are presented in Tables 3A.15-1 through 3A.l5-7. The model used for this comparison is presented in Fig. 3A.15-1.

The comparison is also made with ASME Benchmark solution for force time-history dynamic response[2]. The model used for this comparison is shown on Fig. 3A.15-2. The results for comparisons are presented in the form of plots on Fig. 3A.15-2. The natural frequencies are given in Table 3A.15-8.

The Class 1 piping stress conforms with the hand calculations. The model used is shown on Fig. 3A.l5-3. The results are tabulated in Tables 3A.15-9 and 3A.15-10.

In addition, NUPIPE has been verified in accordance with NRC IE Bulletin 79-07, which was accepted by the NRC in Reference 4. Finally, NUPIPE has been verified using problem numbers 1, 2, 4, and 7 of NUREG/CR-1677.

•→12 •→1

3A.15.1 NUPIPE II

After April 1988, GSU will utilize the public domain program NUPIPE II to perform piping stress analysis. NUPIPE II performs linear elastic analysis of piping systems using the finite element method similar to the SWEC version of NUPIPE. NUPIPE II is substantially similar to the SWEC version of NUPIPE and differs mainly in the post processing

1←• 12←•

●→12 ●→1
of results. Versions of NUPIPE II that are used by RBS have been qualified to assure that the program performs as stated, that the program correctly computes the phenomena of interest, and that RBS is proficient in utilizing the program. NUPIPE II qualification is in accordance with RBS general site and Engineering Department procedures. NUPIPE II was utilized by SWEC in addition to the SWEC version of NUPIPE during RBS construction.
1←● 12←●

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References – Section 3A.15

1. ASME Boiler and Pressure Vessel Code – Section III – Nuclear Power Plant Components. American Society of Mechanical Engineers, 1971; Including Summer 1972 Addenda.

2. American Society of Mechanical Engineers. Pressure Vessel and Piping; 1972 Computer Programs Verification Problem No. 5.

3. Arthur D. Little, 40Inc. ADLPIPE: Static, Dynamic, Thermal Pipe Stress Analysis.

4. Letter dated December 18, 1979, L. C. Shao (NRC) to Stone & Webster Engineering Corp. (Attn. W. G. White, Jr.).

TABLE 3A.15-1

COMPARISON OF SUPPORT REACTION DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	Program	Forces (lb)			Moments (in-lb)		
		FX	FY	FZ	MX	MY	MZ
170	NUPIPE	-9,154	7,541	4,492	-5,952	-823,420	1,241,512
	ADLPIPE	-9,178	7,540	4,492	-5,529	-823,420	1,241,512
218	NUPIPE			16,650			
	ADLPIPE			16,622			
330	NUPIPE	34,532	-33,620	-31,750	-486,338	-1,516,811	573,673
	ADLPIPE	34,511	-33,608	-31,736	-486,386	-1,519,359	573,438
390	NUPIPE		8,631				
	ADLPIPE		8,678				
430	NUPIPE	1,702	798	12,553	-28,147	164,346	248,852
	ADLPIPE	1,746	768	12,541	-26,917	166,180	250,956

TABLE 3A.15-2

COMPARISON OF DEFLECTIONS AND ROTATIONS DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	Program	Deflection (in)			Rotation (rad)		
		DX	DY	DZ	RX	RY	RZ
197	NUPIPE	0.0348	-0.141	0.230	-0.0026	0.0025	-0.0084
	ADLPIPE	0.348	-0.141	0.229	-0.0026	0.0025	-0.0084
212	NUPIPE	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
	ADLPIPE	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
230	NUPIPE	1.276	-0.028	-0.548	-0.0066	-0.0044	0.0024
	ADLPIPE	1.276	-0.027	-0.548	-0.0066	-0.0044	0.0024
260	NUPIPE	0.512	-0.001	-0.520	-0.0034	-0.0005	0.0035
	ADLPIPE	0.512	-0.000	-0.520	-0.0005	-0.0005	0.0035
390	NUPIPE	0.066	-0.000	0.249	-0.0026	0.0026	-0.0020
	ADLPIPE	0.067	-0.000	0.248	-0.0010	0.0026	-0.0020
420	NUPIPE	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007
	ADLPIPE	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007

TABLE 3A.15-3

COMPARISON OF STRESS DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	Stress (psi)	
	NUPIPE	**ADLPIPE**
180	18,989	19,013
199	17,703	17,731
214	23,958	23,955
236	14,427	14,416
265	6,254	6,25l
305	12,539	12,532
344	11,845	11,838
370	6,295	6,296
395	3,476	3,473
430	3,282	3,308

TABLE 3A.15-4

COMPARISON OF INTERNAL FORCES DUE TO DEADWEIGHT ANALYSIS

		Forces (lb)			Moments (in-lb)		
Node	Program	FX	FY	FZ	MX	MY	MZ
197	NUPIPE	295	2,337	14	-35,864	5,218	51,979
	ADLPIPE	290	2,341	15	-35,108	5,231	52,081
212	NUPIPE	295	3,306	14	59,390	5,394	14,010
	ADLPIPE	299	3,310	15	59,735	-5,500	14,542
360	NUPIPE	330	2,781	-29	30,930	-22,748	-84,971
	ADLPIPE	326	2,783	-32	31,920	-23,105	-82,784
390	NUPIPE	330	4,933	-29	-255,351	710	126,476
	ADLPIPE	336	4,707	-32	-256,444	916	126,716
420	NUPIPE	330	-492	-29	-8,972	27,075	82,202
	ADLPIPE	336	-497	-32	-9,181	27,724	80,676

TABLE 3A.15-5

COMPARISON OF DEFLECTIONS AND ROTATION DUE TO DEADWEIGHT

Node	Program	Deflections (in)			Rotations (rad)		
		DX	**DY**	**DZ**	**RX**	**RY**	**RZ**
197	NUPIPE	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
	ADLUPIPE	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
212	NUPIPE	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
	ADLPIPE	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
360	NUPIPE	-0.008	-0.068	0.024	0.0004	-0.0000	-0.0004
	ADLPIPE	-0.009	-0.069	0.024	0.0004	0.0000	-0.0004
390	NUPIPE	-0.014	-0.000	-0.003	0.0002	-0.0003	-0.0005
	ADLPIPE	-0.015	-0.000	-0.003	0.0002	-0.0002	-0.0005
420	NUPIPE	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002
	ADLPIPE	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002

TABLE 3A.15-6

COMPARISON OF STRESSES DUE TO DEADWEIGHT

Node	**NUPIPE (psi)**	**ADLPIPE (psi)**
180	685	694
199	448	458
214	667	679
236	2,472	2,449
265	530	524
305	515	522
344	635	631
370	679	677
395	575	580
430	1,101	1,091

TABLE 3A.15-7

COMPA7RISON OF NATURAL FREQUENCIES (HZ)

	Mode				
	1st	**2nd**	**3rd**	**4th**	**5th**
NUPIPE	7.109	9.328	12.297	14.681	18.043
ADLPIPE	7.118	9.329	12.492	14.427	17.714

TABLE 3A.15-8

COMPARISON OF NATURAL FREQUENCIES

	Mode	
	1st	**2nd**
NUPIPE	2.407	13.537
Benchmark Pr.	2.3288	13.0808

TABLE 3A.15-9

NUPIPE VS HAND CALCULATION

Point No. 20	Hand Calculation	NUPIPE
Min Wall Thickness	0.032 in	0.032 in
Primary Stress (Eq. 9)	3,713 psi	3,712 psi
Primary & Secondary Stress (Eq. 10)	16,041 psi	16,038 psi
Alternating Stress (Eq. 11 & 14)	13,468 psi	13,465 psi
Usage Factor	0.0654	0.0631

Point No. 30		
Min Wall Thickness	0.047 in	0.047 in
Primary Stress (Eq. 9)	8,748 psi	8,741 psi
Primary & Secondary Stress (Eq. 10)	117,655 psi	117,546 psi
Expansion Stress (Eq. 12)	99,884 psi	99,781 psi
Eq. 13	18,252 psi	18,246 psi
Alternate Stress (Eq. 14)	218,258 psi	217,811 psi
Usage Factor	Out of Range	

NOTE: Equation numbers refer to Subarticle NB-3650 of ASME Section III.

TABLE 3A.15-10

INDIVIDUAL PAIR USAGE FACTOR FOR POINT NO. 30

Pair	**Hand Calculation**	**NUPIPE**
1,5	0.183	0.1803
1,8	1.660	1.7361
1,9	0.0001	0.0001
1,10	Not in Range	
5,8	Not in Range	
5,9	0.221	0.2646
5,10	0.747	0.8051
8,9	0.857	0.8832
8,10	5.5518	5.8608
9,10	0.0001	0.0001

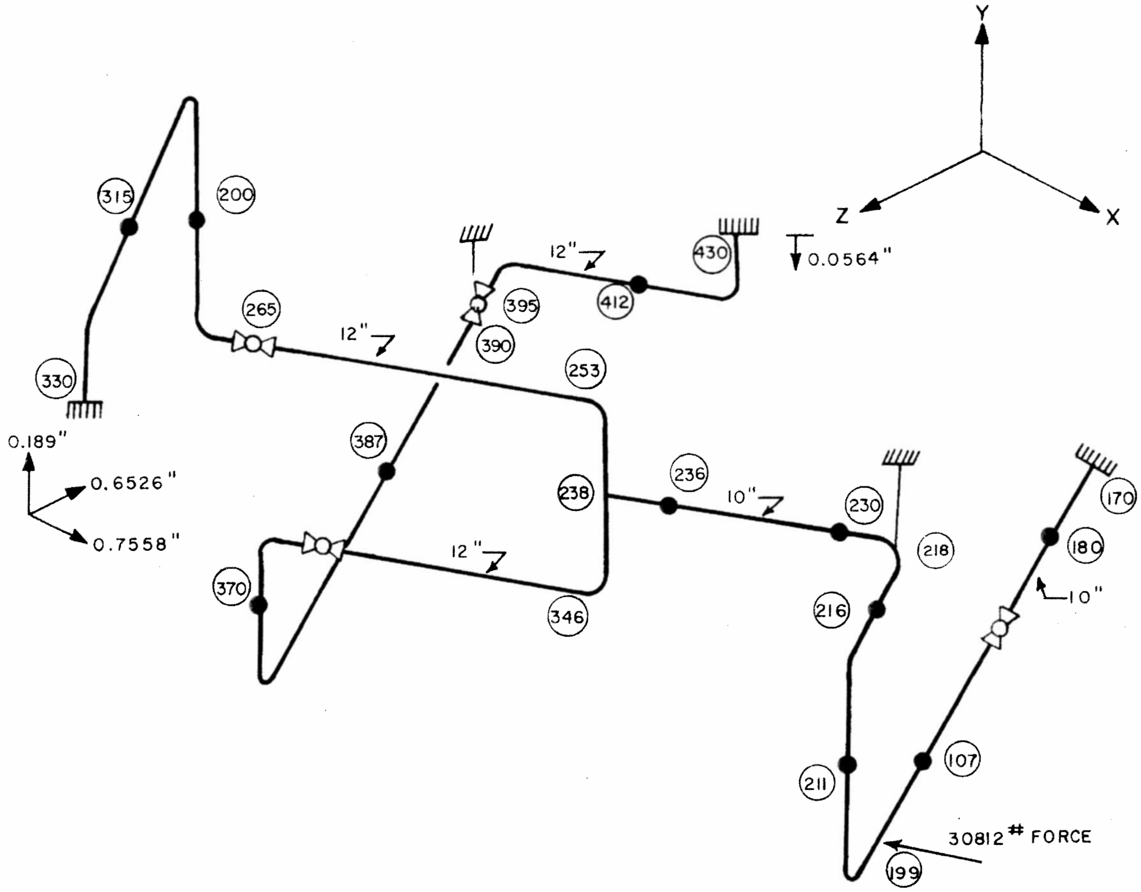

FIGURE 3A.15-1

MATHEMATICAL MODEL FOR FLEXIBILITY ANALYSIS VERIFICATION

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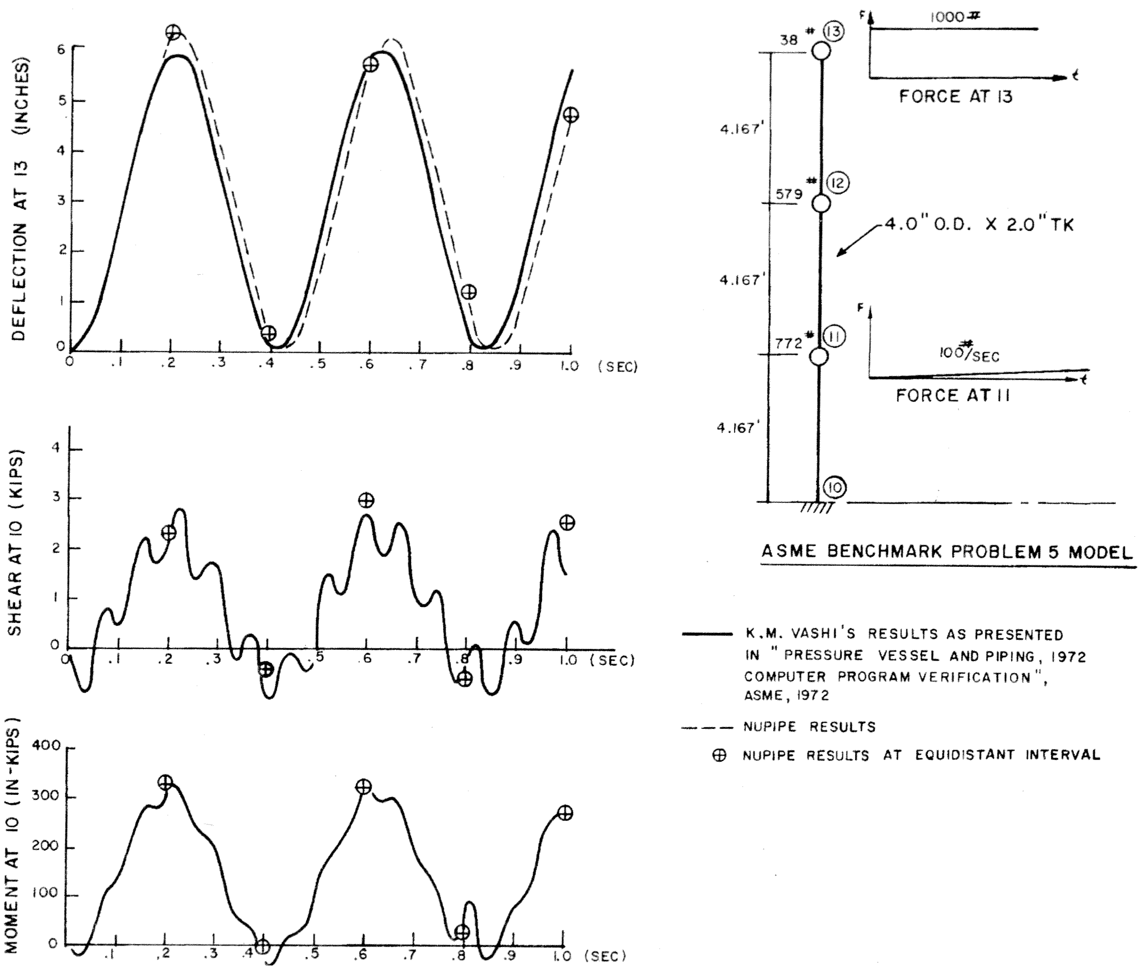
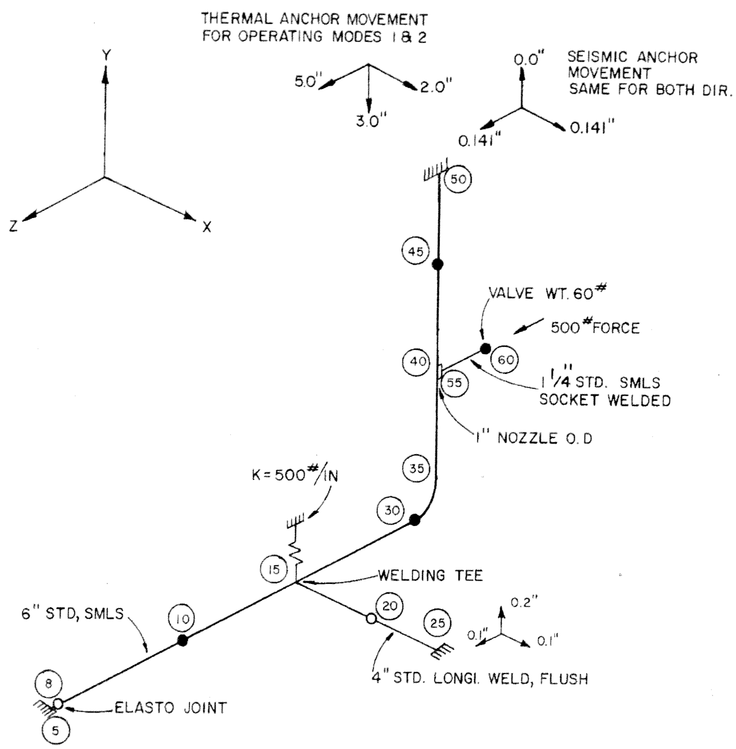


FIGURE 3A.15-2

NUPIPE PROGRAM FORCE
TIME-HISTORY VERIFICATION



OPERATING CONDITIONS			
OPER MODE	PIPE	PRESS (PSI)	TEMP (°F)
1	6"	200	400
	4"	200	400
	1 1/4"	200	400
$E/\alpha_1 T_1 - \alpha_0 T_0$ AT 15 = 440 PSI			
2	6"	200	700
	4"	0	70
	1 1/4"	200	700
$\alpha\Delta T_1 = 0.0002$, $\alpha\Delta T_2 = 0.0004$ IN/FT			
3	6"	700	70
	4"	700	70
	1 1/4"	700	70

FIGURE 3A.15-3

MATHEMATICAL MODEL FOR CLASS 1 STRESS VERIFICATION

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3A.16 TRHEAT

TRHEAT is a program which determines the temperature response of a pipe due to a temperature transient in the contained fluid. TRHEAT results include the equivalent linear and nonlinear pipe wall temperature gradients and the discontinuity temperature differences required for calculating piping stresses in accordance with the requirements for Class 1 piping specified in ASME Section III. The method of analysis used is a closed-form solution to the basic heat transfer partial differential equation.

The sample problem selected for solution by TRHEAT consists of a 2-in schedule 160, stainless steel pipe, with one end connected to a socket-welded fitting. Saturated water flowing within the piping system changes temperature from 400°F to 500°F in a period of 10 sec. Velocity of fluid is 7,560 ft/hr. Results from TRHEAT program for pipe and fluid properties are compared with standard reference values in Tables 3A.16-1 and 3A.16-2.

Reynolds number and heat transfer coefficients are compared with hand calculations and are given in Table 3A.16-3.

Comparison between TRHEAT and Brock and McNeill's charts for ΔT_1 and ΔT_2 are given in Table 3A.16-4. Table 3A.16-5 represents the comparison between MARCHEAT(1) and TRHEAT for ΔT_1, ΔT_2 and T_a-T_b, as defined in Equation 10 of ASME Section III, Paragraph NB3653.

Reference - Section 3A.16

1. MARC Analysis Corporation. MARCHEAT: A Finite Element Transient Heat Conduction Program, 1971.

TABLE 3A.16-1

PIPE MATERIAL PROPERTIES

Property	Temperature (°F)	Reference Value	TRHEAT Calculation
Thermal Conductivity	450	10.00 Btu/°F-hr-ft (See source, Table I-4.0)	10.01 Btu/°F-hr-ft
Thermal Diffusivity	450	0.164 ft/hr (See source, Table I-4.0)	0.164 ft^2/hr
Young's Modulus	70	28.3 x 10^6 (See source, Table I-6.0)	28.3 x 10^6 psi
Coefficient of Thermal Expansion	70	9.11 x 10^{-6} in/in/°F (See source, Table I-5.0)	9.11 x 10^{-6} in/in/°F

Source: Meyer, McClintock, et al. 1967 ASME Steam Tables.

TABLE 3A.16-2

FLUID MATERIAL/THERMAL PROPERTIES

Property	Temperature (°F)	Reference Value	TRHEAT Calculation
Density	450	51.467 lb/ft^3 (Source 1, p.84)	51.300 lb/ft^3
Viscosity	450	0.288 lb/hr-ft (Source 2, Table A-3)	0.2920 lb/hr/ft
Specific Heat	450	1.12 Btu/lb-°F (Source 2, Table A-3)	1.135 Btu/lb-°F
Conductivity	450	0.367 Btu/°F-hr-ft (Source 2, Table A-3)	0.3650 Btu/°F-hr-ft
Volume Expansion Coefficient	450	0.0009/°F (Source 2, Table A-3)	0.0009/°F

Sources:

1. Meyer, McClintock, et al. 1967 ASME Steam Tables.
2. Kreith, F., Principles of Heat Transfer. International Textbook Company, 1964.

TABLE 3A.16-3

TRHEAT VS HAND CALCULATION

	TRHEAT	Hand Calculation
Reynolds Number	186,900	186,941
Heat Transfer Coefficient	946.6 Btu/F-hr-ft^2	946.6 Btu/F-hr-ft^2

TABLE 3A.16-4

COMPARISON OF TRHEAT WITH CHARTS OF BROCK AND MCNEILL

Parameter	Charts	TRHEAT
Maximum ΔT_1 (°F)	43.31	44.43
Maximum ΔT_2 (°F)	8.50	8.64

TABLE 3A.16-5

COMPARISON OF TRHEAT WITH MARCHEAT

Parameter	MARCHEAT	TRHEAT
Maximum ΔT_1 (°F)	43.32	44.43
Maximum ΔT_2 (°F)	9.32	8.64
Maximum T_a - T_b (°F)	51.68	49.39

3A.17 HTLOAD

3A.17.1 General Description

HTLOAD is a computer program which performs a finite difference method analysis of piping system response to thermal transients of its contained fluid. The output gives overall thermal growth, linear and nonlinear temperature distribution through the pipe wall, gross discontinuity information (T_a- T_b), and Equation 10 and Equation 11 results of subarticle NB-3600 of ASME Section III.

HTLOAD can analyze piping with or without thermal sleeve that is subject to changes in fluid temperature, velocity, and/or state. The properties of subcooled or saturated water and superheated or saturated steam are taken from the ASME steam tables[1]. The pressure range is from 0.45 psia to 6,210 psia.

This computer program also performs thermal analysis for pipes with different insulating conditions ranging from noninsulated to perfectly insulated. It has stored properties for insulation such as unibestos, asbestos, reflective aluminum, reflective stainless, and calcium silicate. Provision is further made for hand input properties of other insulation types.

Also stored in the program are the piping material properties of carbon steel, austenitic stainless, low-chrome steel, high-chrome steel, and nickel-chrome iron for the temperature range of 32°F to 1,600°F.

Program input includes piping material insulation information, time lapse for initial to final fluid temperature, calculation time limit, fluid velocities, initial and final temperature and pressure, and pipe and thermal sleeve dimensions.

HTLOAD requires that each thermal transient be input as a step change, a ramp change, or as a 12-point arbitrary function.

Output results are used in the calculation of piping stress in accordance with Article NB-3600 of ASME, Section III. HTLOAD also performs the primary plus secondary stress intensity range check (Equation 10) and the peak stress intensity range calculation (Equation 11) from Article NB-3600.

3A.17.2 Program Verification

The sample problem selected for solution by HTLOAD consists of a 2-in schedule 160, stainless steel pipe, with one end connected to a 1/2-in thick, socket-welded fitting. Saturated water flowing within the piping system changes temperature from 400°F to 500°F in a period of 10 sec. Velocity of fluid is 7,560 ft/hr. Input properties are listed in Tables 3A.17-1 and 3A.17-2.

Reynolds number and heat transfer coefficients are compared with hand calculations[2] and are given in Table 3A.17-3.

Comparison between HTLOAD and Brock and McNeill's charts[3] for ΔT_1 and ΔT_2 are given in Table 3A.17-4. Table 3A.17-5 represents the comparison between TRHEAT[4] and HTLOAD for ΔT_1, ΔT_2, and T_a-T_b.

References - Section 3A.17

1. Meyer, McClintock, et al. 1967 ASME Steam Tables.

2. Kreith, F. Principles of Heat Transfer. International Textbook Company, 1964.

3. McNeill, D. R. and Brock, J. E. Charts for Transient Temperature in Pipes, Heating/Piping/Air Conditioning. November 1971.

4. TRHEAT: Computer Code for Transient Heat Analysis of Nuclear Piping, Nuclear Services Corporation, 1972.

TABLE 3A.17-1

PIPE MATERIAL PROPERTIES

Property	Temperature (°F)	Value
Thermal Conductivity	450	10.01 Btu/°F-hr-ft
Thermal Diffusivity	450	0.164 ft^2/hr
Young's Modulus	70	28.3×10^6 psi
Coefficient of Thermal Expansion	70	9.11×10^6 in/in/°F

TABLE 3A.17-2

FLUID MATERIAL/THERMAL PROPERTIES

Property	Value at 450°F
Density	51.300 lb/ft^3
Viscosity	0.2920 lb/hr/ft
Specific Heat	1.135 Btu/lb-°F
Conductivity	0.3650 Btu/°F-hr-ft
Volume Expansion Coefficient	0.0009/°F

TABLE 3A.17-3

COMPARISON OF HTLOAD WITH HAND CALCULATION

	HTLOAD	Hand Calculation
Reynolds Number	186,700	186,700
Heat Transfer Coefficient	946.8 Btu/°F-hr-ft^2	946.8 Btu/°F-hr-ft^2

TABLE 3A.17-4

COMPARISON OF HTLOAD WITH CHARTS OF BROCK AND MCNEILL

Parameter	Charts	HTLOAD
Maximum ΔT_1 (°F)	43.31	45.14
Maximum ΔT_2 (°F)	8.50	8.36

TABLE 3A.17-5

COMPARISON OF HTLOAD WITH TRHEAT

Parameter		TRHEAT	HTLOAD
Maximum	ΔT_1 (°F)	44.70	45.14
Maximum	ΔT_2 (°F)	8.69	8.36
Maximum	$T_a - T_b$ (°F)	19.03	19.08

3A.18 GHOSH-WILSON

3A.18.1 General Description

Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loadings, to be known as the GHOSH-WILSON computer code, is a finite-element based computer program developed by S. Ghosh and E. Wilson[1] and modified by Stone & Webster as Code ST-200.

GHOSH-WILSON is capable of performing static and dynamic analysis of complex axisymmetric structures subjected to any arbitrary static (mechanical and temperature) and dynamic loading.

The method used to represent the three-dimensional continuum is either as an axisymmetric thin shell, a solid of revolution, or a combination of both. The arbitrary loading in the circumferencial direction is represented by a Fourier series, and the analysis is carried out for each term and summed up for the total response.

Hamilton's variational principle is used to derive the equation of motion. This leads to a diagonal mass matrix and a stiffness matrix and load vector which is consistent with the assumed displacement field. The equations of motion are solved numerically in the time-domain by direct integration using the Wilson method[2].

The input required by GHOSH-WILSON is a description of geometry, materials, and boundary conditions. Loadings, damping factors, and time intervals for integration should be provided for each Fourier term. Additional inertias can be added at joints during a dynamic analysis.

GHOSH-WILSON provides time-history responses of the resultant forces, moments, shears, displacements, rotations, accelerations, and stresses at each node for the dynamic analysis. Maximum responses can also be obtained for each Fourier term.

3A.18.2 Program Verification

3A.18.2.1 Sample Problem No. 1: <u>Cylinder Under Internal Pressure</u>

A cylinder is subjected to a constant internal pressure. The cylinder is modeled using the shell element, rectangular element, and triangular element. The solution to this problem is found in References 3 and 4.

Pertinent parameters

1. Dimensions and properties

Pressure	$P = 1$ ksf
Mean radius	$R = 40$ ft
Height	$l = 20$ ft
Thickness	$t = 2$ ft
Young's Modulus	$E = 3 \times 10^6$ psi
Poisson's ratio	$\nu = 0.15$

2. Loading and boundary conditions

$l = 0, \quad M = \delta_z = 0$

$l = 20, \quad M = F = 0$

From Reference 3 using thin shell solution.

$$\sigma_\theta = \frac{PR}{t}$$

$$\delta_R = \frac{PR^2}{Et}$$

From Reference 4 based on the theory of elasticity:

$$\sigma_\theta = \frac{a^2 p}{b^2 - a^2}(1 + b^2/r^2)$$

$$\delta_r = \frac{a^2 p}{b^2 - a^2}(1 - b^2/r^2)$$

$$\Delta R = \frac{a^2 p}{E} \frac{1}{b^2 - a^2}\left[b^2 \frac{(1+\upsilon)}{r} + (1-\upsilon)\ r\right]$$

where:

a = Inside radius
b = Outside radius
r = Radius where results are computed
P = Internal pressure

Table 3A.18-1 shows the results of the GHOSH-WILSON solution compared to the theoretical solution. The results compare favorably.

3A.18.2.2 Sample Problem No. 2: <u>Cylinder Subjected to Suddenly Applied Load</u>

A cylinder simply supported at both ends is subjected to a suddenly applied load at midspan. The solution of the equations of motion is obtained by the direct integration method. The dimensions of the cylinder and the loading time history are shown on Fig. 3A.18-1. The cylinder is modeled using rectangular elements. The GHOSH-WILSON solution (displacement under the applied load) is compared to the solution using the ANSYS computer code. The results are shown on Fig. 3A.18-2, and they compare favorably.

References - Section 3A.18

1. Ghosh, S. and Wilson, E. Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading. Report EERC-69-10, University of California at Berkley, September 1969. Modified as Stone & Webster Engineering Corporation Computer Code ST-200, September 1973.

2. Bathe, Klaus-Jurgen and Wilson, E. Numerical Methods in Finite Element Analysis, Prentice-Hall, Inc., Englewood Cliffs, NJ, 1976.

3. Roark, R. J. Formulas for Stress and Strain, Fourth Ed., McGraw-Hill Book Company, NY, 1965.

4. Timoshenko, S. and Goodier, J. N. Theory of Elasticity, McGraw-Hill Book Company, NY, p 58-60, 1951.

TABLE 3A.18-1

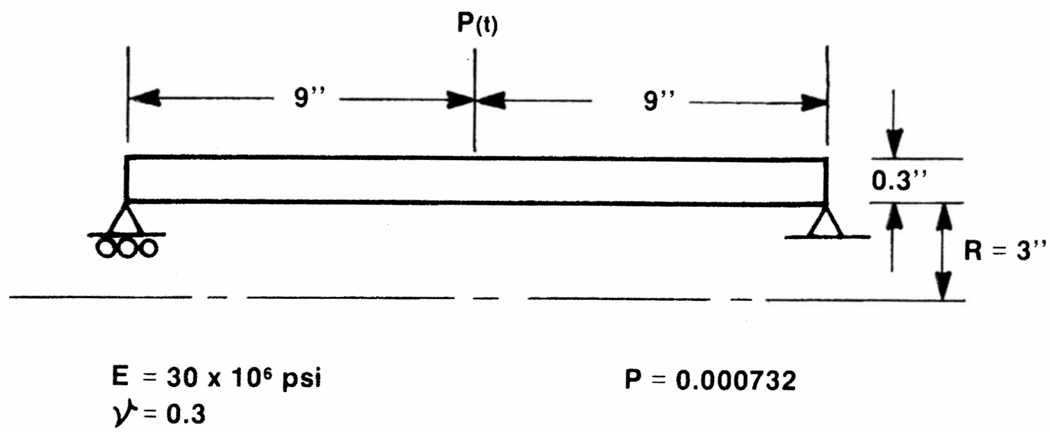
COMPARISON OF GHOSH-WILSON RESULTS VERSUS THEORETICAL SOLUTIONS
FOR A CYLINDER UNDER STATIC INTERNAL PRESSURE

	Radius (R)-ft	Theoretical Solution	Thin-Shell Theory Shell Element	GHOSH-WILSON Results Rectangular Element	GHOSH-WILSON Results Triangular Element
	R=39.5	19.748	-	19.74	19.755
σ_θ	R=40	20.00	20.27	-	-
Ksf	R=40.5	19.249	-	19.24	19.255
σ_r	R=39.5	-0.7357	-	-0.7365	-0.683
Ksf	R=40.5	-0.236	-	-0.2369	-0.188
	R=39	1.82×10^{-3}	-	1.82×10^{-3}	1.819×10^{-3}
ΔR	R=40	1.812×10^{-3}	1.87×10^{-3}	1.811×10^{-3}	1.811×10^{-3}
ft	R=41	1.804×10^{-3}	-	1.804×10^{-3}	1.804×10^{-3}

σ_θ = hoop stress

σ_r = radial stress

ΔR = displacement in radial direction



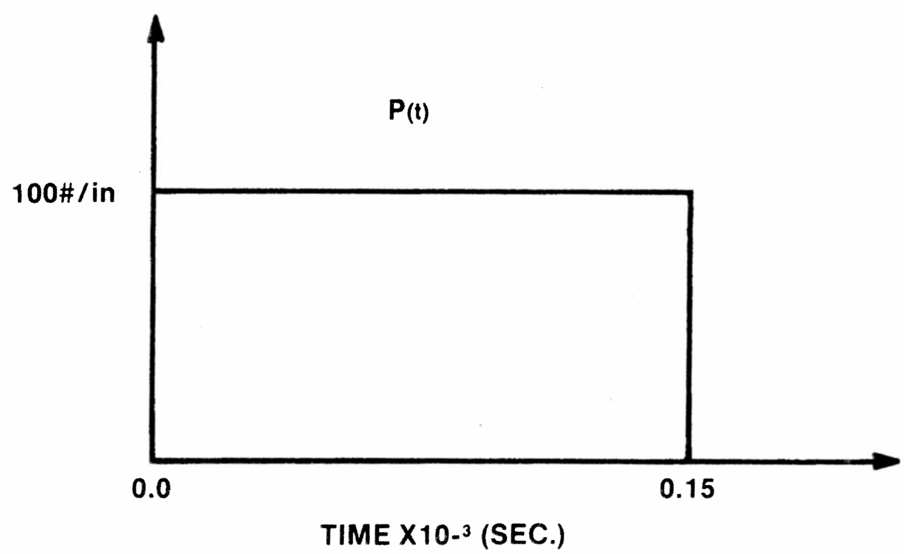


FIGURE 3A.18-1

SAMPLE PROBLEM NO. 2
FOR GHOSH-WILSON

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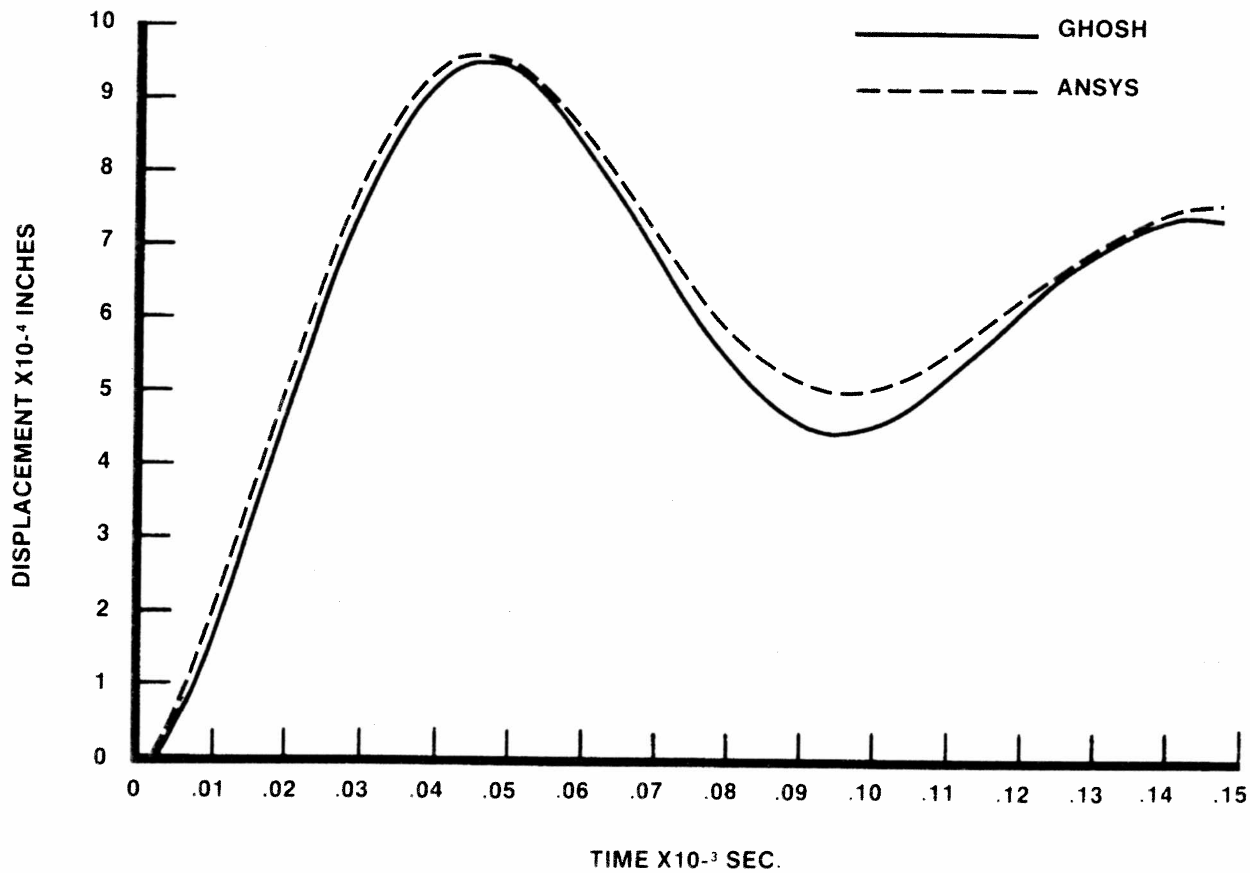

FIGURE 3A.18-2

COMPARISON OF RESULTS FROM GHOSH AND ANSYS FOR A CYLINDER SUBJECTED TO A SUDDENLY APPLIED LOAD

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3A.19 PSPECTRA

3A.19.1 General Description

PSPECTRA (ME-164) is a data-generating program written and fully documented by Stone & Webster Engineering Corporation for inhouse use. It is used to combine amplified response spectra of seismic and other dynamic events. The methods of spectrum combination include absolute summation, square root of the sum of the squares (SRSS), and maximum value enveloping. PSPECTRA is also used to generate required response spectra which are in accordance with Regulatory Guide 1.122, Rev. 1. This involves spreading the peak accelerations and sloping the sides parallel to the original peaks of the input amplified response spectra. The output curves can be generated in terms of acceleration (g's) and either period (sec) or frequency (Hz).

3A.19.2 Program Verification

A comparison of a generated response spectrum versus the two input response spectra that were combined by absolute summation is provided on Fig. 3A.19-1. Fig. 3A.19-2 provides a generated required response spectrum with spread peaks and parallel sloped sides superimposed on the input amplified response spectrum (ARS). The ARS is generated by the time-history method. These figures demonstrate the function and adequacy of the program.

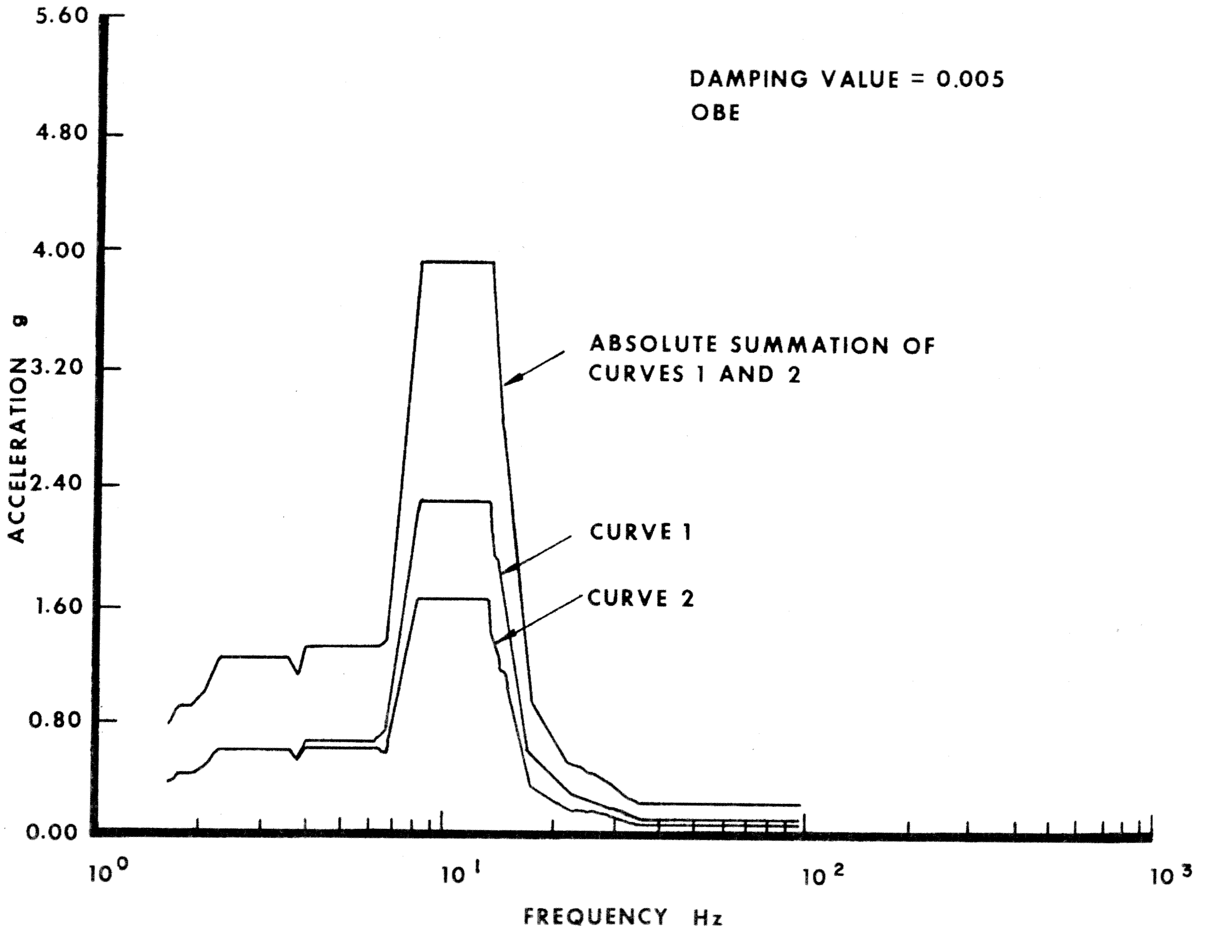

FIGURE 3A.19-1

PSPECTRA-ABSOLUTE SUMMATION OF ARS CURVES

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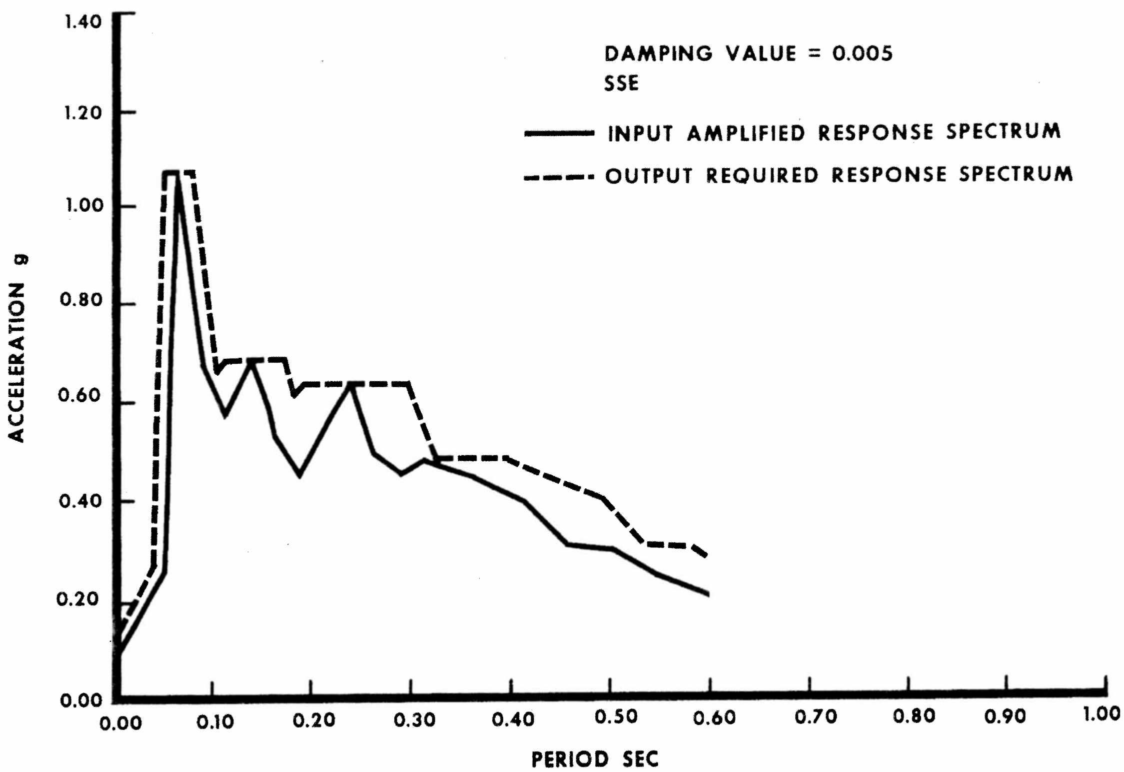

FIGURE 3A.19-2

PSPECTRA-REQUIRED RESPONSE SPECTRUM GENERATION

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3A.20 STEHAM

3A.20.1 General Description

STEHAM is a computer program which is used to determine the steamhammer transients of piping systems. This program uses the method of characteristics with finite difference approximations both in space and in time[1, 2, 3]. It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as the stop valve and the safety/relief valve. Flow characteristics of piping components are mathematically formulated as boundary conditions in the program. These components include the flow control valve, the stop valve, the safety/relief valve, the steam manifold, and the steam reservoir. Frictional effects are taken into consideration.

This program accepts the following as input: 1) the flow network representation of the piping system, 2) the initial flow conditions along the piping system, and 3) time-dependent flow characteristics of piping components. Output consists of time-histories of flow pressures, flow densities, flow velocities, inertia, and momentum functions.

3A.20.2 Program Verification

STEHAM is verified by comparing its solutions of a test problem (Fig. 3A.20-1 and 3A.20-2) to the results of the same problem obtained by an independent analytical approach, as well as an experimental measurement, as published in References 4 and 5. A comparison of results for time-history pressure responses is plotted on Fig. 3A.20-3, 3A.20-4, and 3A.20-5. The forcing functions developed for nodal points of the piping system calculated from the relation $F = (p + \rho V^2/g)A - p_a A$ has also been checked by hand calculations as tabulated in Table 3A.20-1.

References - Section 3A.20

1. Jonsson, V.K.; Matthews, L.; and Spalding, D.B. Numerical Solution Procedure for Calculating the Unsteady One-Dimensional Flow of Compressible Fluid. ASME Paper No. 73-FE-30.

2. Luk, C.H. Effects of the Steam Chest on Steamhammer Analysis for Nuclear Piping Systems. ASME Paper No. 75-PVP-61.

3. Moody, F.J. Time-Dependent Pipe Forces Caused by Blowdown and Flow Stoppage. ASME Paper No. 73-FE-23.

4. Progelhof, R.C. and Owczarek, J.A. The Rapid Discharge of a Gas from a Cylindrical Vessel Through a Nozzle. AIAA Journal, Vol 1, No. 9, September 1963, p 2182-2184.

5. Progelhof, R.C. and Owczarek, J.A. The Rapid Discharge of a Gas from a Cylindrical Vessel Through an Orifice. ASME Paper No. 63-WA-10.

TABLE 3A.20-1

NODAL FORCE COMPARISON

Diameter D = 0.25 ft

Area A = π D/4 = 0.0490874 ft^2

Nodal Force = $(p + \rho V^2/g)A - p_a A$

p = pressure lb/ft^2

ρ = density lb/ft^3

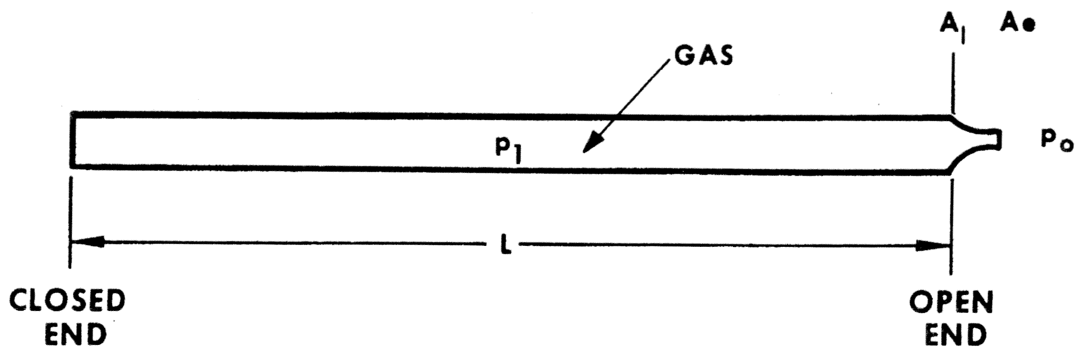
V = velocity ft/sec

g = gravitational constant 32.2 ft/sec^2

p_a = ambient pressure (14.7x144 lb/ft^2)

at time t = 0.00650 sec

Node No.	Pressure (psia)	Velocity (fps)	Density (lb/ft)	Force: STEHAM (lb)	Force: Hand Calculation (lb)
1	42.523	0.0	0.23954	186.57	196.67
5	42.785	5.7843	0.24076	198.43	198.53
10	44.231	31.219	0.24647	209.00	209.11
15	47.003	78.172	0.25737	230.62	230.73
20	50.214	129.89	0.26979	257.84	257.97
25	52.095	159.43	0.27697	274.93	275.06
30	52.209	161.97	0.27742	276.09	276.23
35	52.168	162.21	0.27731	275.83	275.97



CASE (A) FOR COMPARISON WITH ANALYTICAL RESULTS

INITIAL CONDITIONS =

$p_1/p_o = 4.72$, $a_o/a_1 = 0.80$

DIMENSIONS =

$A_e/A_1 = 0.6$, L = PIPE LENGTH

SPECIFIC HEAT RATIO =

$\gamma = C_p/C_v = 1.4$

p = PRESSURE, a = SOUND VELOCITY, A = FLOW AREA

FIGURE 3A.20-1

SUDDEN DISCHARGE OF A GAS FROM A PIPELINE THROUGH A NOZZLE

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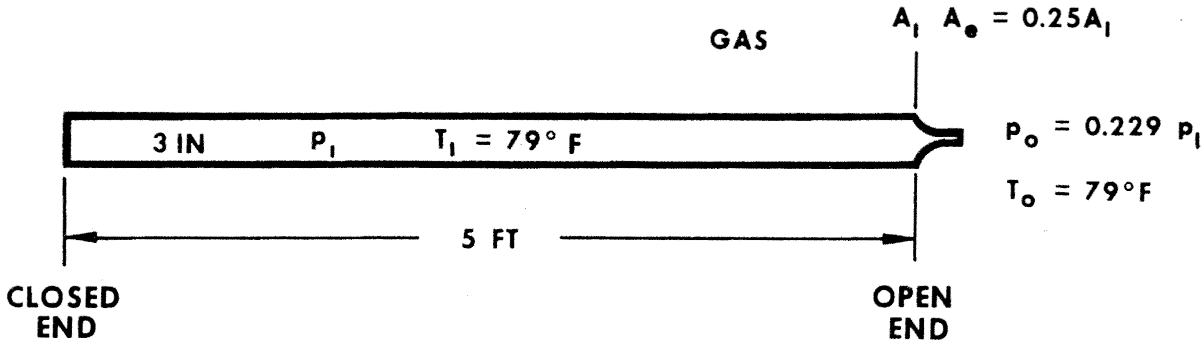

CASE (B) FOR COMPARISON WITH EXPERIMENTAL DATA AND HAND CALCULATION

PRESSURE = p_o = 14.7 psia, p_1 = 14.7/0.229 = 64.2 psia

AREA = $A_1 = \frac{\pi}{4}(0.25)^2 = 0.0491$ ft^2, $A_e = 0.25A_1 = 0.0123$ ft^2

GAS CONSTANT = R = 53.35 ft - lb_f/lb°R

TEMPERATURE = T_o = 79°F = 539°R, $T_1 = T_o$

DENSITY $= \rho_o = \frac{p_o}{RT_o} = 0.0736$ lb/ft^3, $\rho_1 = \frac{p_1}{RT_1} = 0.3215$ lb/ft^3

FIGURE 3A.20-2

SUDDEN DISCHARGE OF A GAS FROM A PIPELINE THROUGH A NOZZLE

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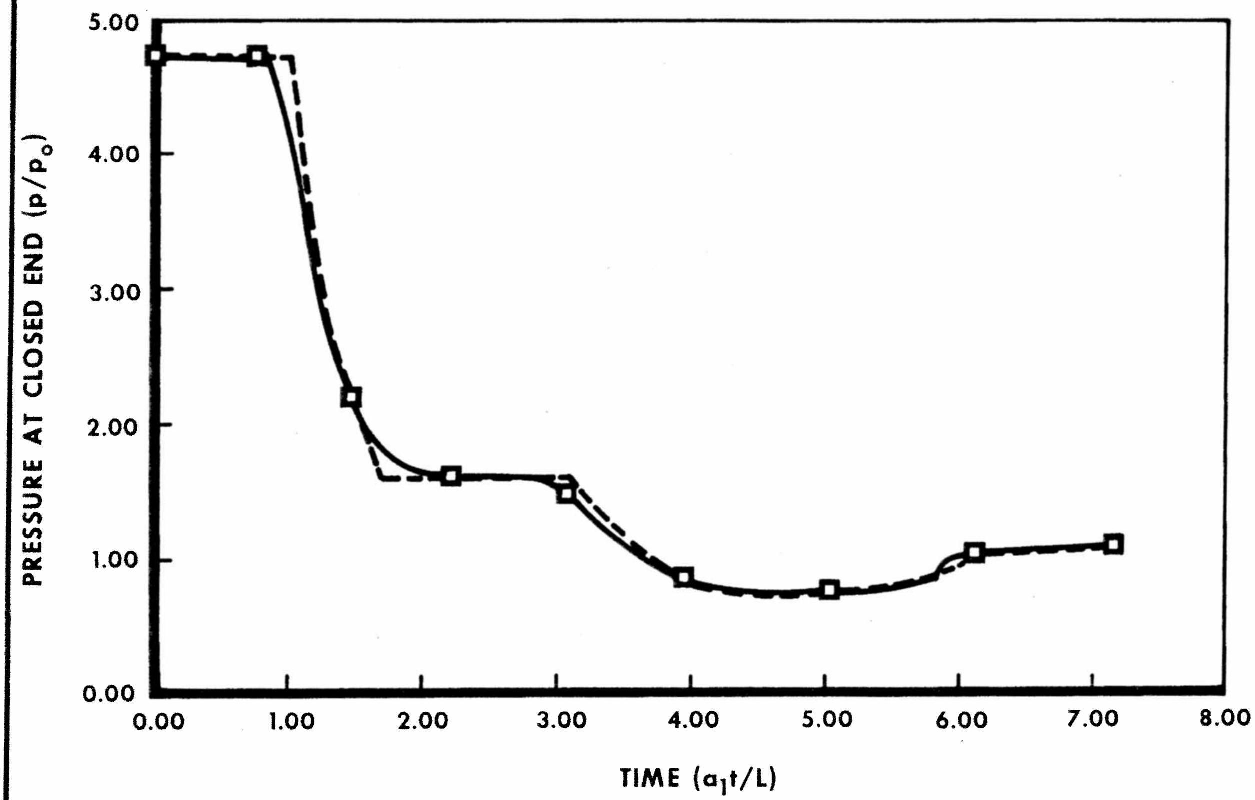

LEGEND

(CASE A)

——— STEHAM

– – – – ANALYTICAL SOLUTION (REF 4 & 5)

FIGURE 3A.20-3

COMPARISON OF PRESSURE RESPONSE AT THE CLOSED END

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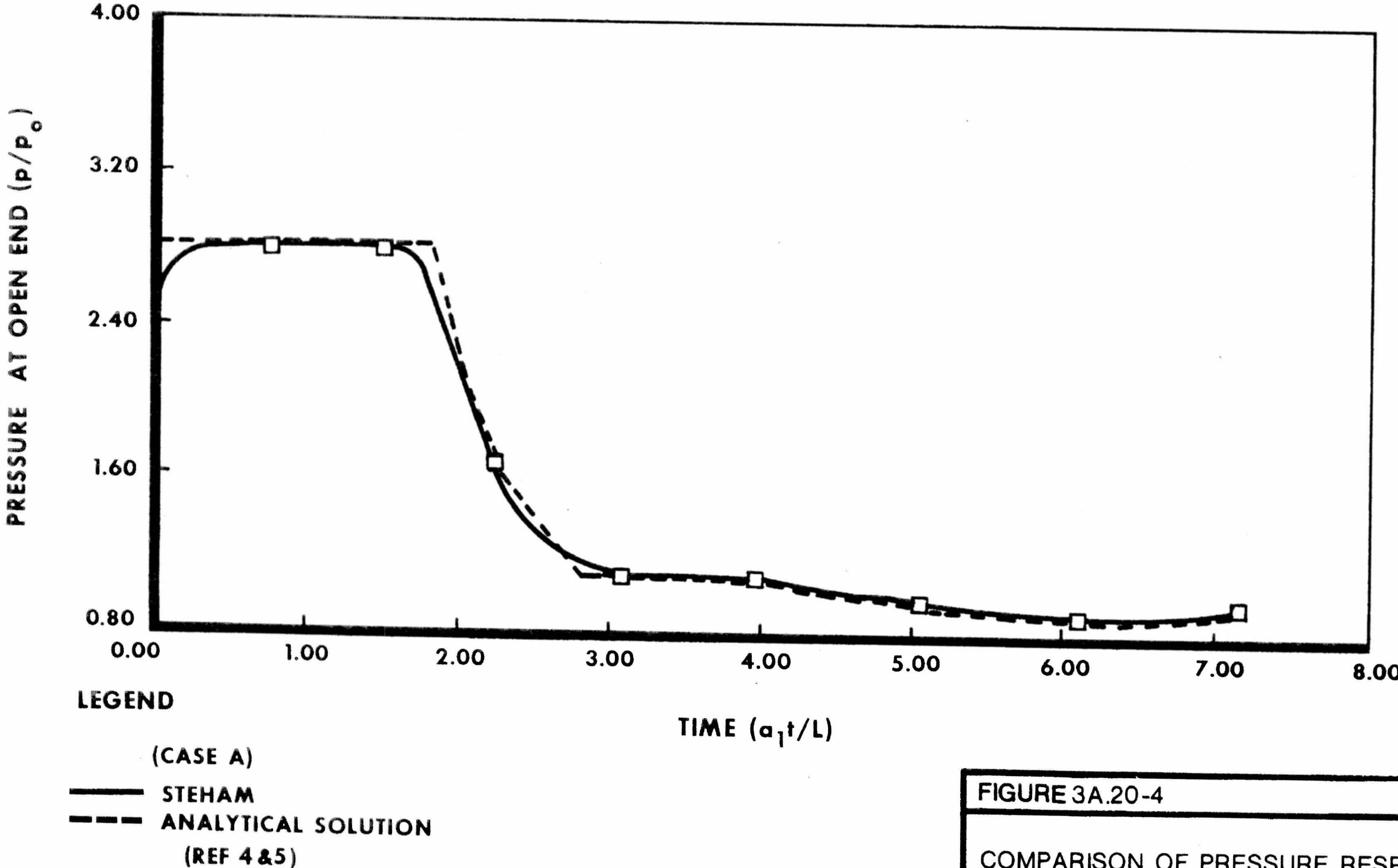

FIGURE 3A.20-4

COMPARISON OF PRESSURE RESPONSE AT THE OPEN END

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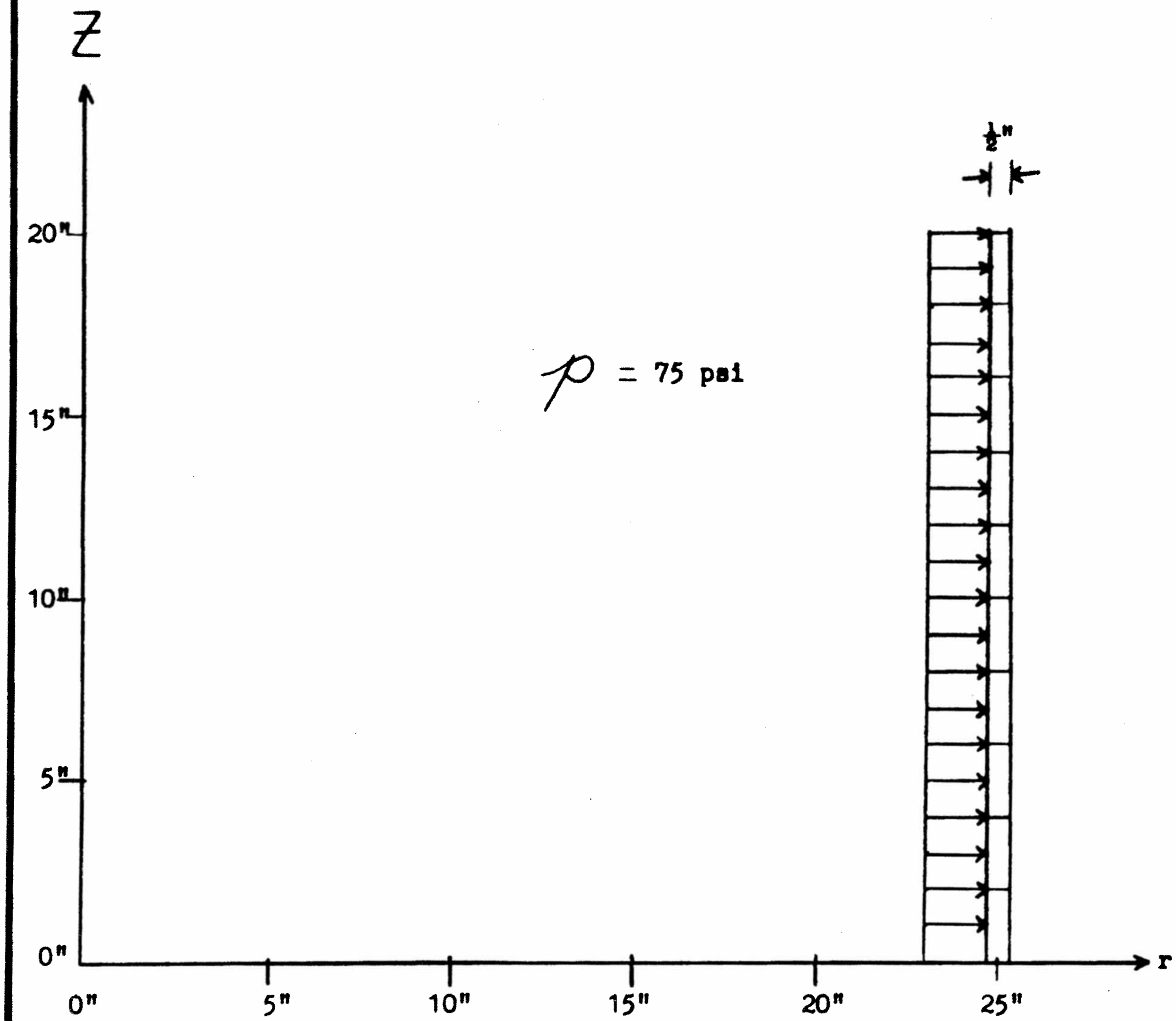

FIGURE 3A.1-1

TEN ELEMENT IDEALIZATION OF THIN WALL CYLINDER

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3A.21 WATHAM

3A.21.1 General Description

WATHAM is a computer program which is used to determine the flow-induced forcing functions acting on piping systems due to waterhammer. These forcing functions may then be used as input to a structural dynamic analysis, such as a NUPIPE program run.

WATHAM is applicable to a waterhammer problem or, more generally, any unsteady, incompressible fluid flow. These events may be caused by normal or abnormal operational changes of piping components, such as the startup and trip of pumps or the rapid opening and closing of valves.

The analysis is based upon the method of characteristics with finite-difference approximations, both in time and space for the solution of one-dimensional liquid flows. Influences of piping components, including flow valves, pipe connections, reservoirs, and pumps, have been considered in the analysis.

WATHAM input requires the geometry of the piping system, pipe properties, water properties, operational characteristics of pump and valve, flow frictional coefficients, and the initial water flow conditions. The output provides the time history functions of piezometric heads, velocities, and nodal forces for all nodes and the inertial unbalanced force for each segment. It also gives the maximum value of all the preceding functions and their occurring time in the process of flow-transient.

3A.21.2 Program Verification

Fig. 3A.21-1 depicts a flow network with nine pipes, its geometrical properties, and steady-state flow conditions. The flow-transient mode analyzed is the sudden closure of a valve at the downstream end. Fig. 3A.21-2 shows the hydraulic network for WATHAM. Table 3A.21-1 illustrates the input data needed for WATHAM run. Fig. 3A.21-3 and 3A.21-4 show a comparison of head-time curves [1, 2] with WATHAM. Table 3A.21-2 presents the comparison of nodal forces between hand calculation and WATHAM computation.

In general, WATHAM 3 results are in agreement with Streeter's results [1]. The small discrepancy is attributed to the modeling of reservoir boundary condition. In WATHAM, the energy equation between the reservoir is utilized, rather than assuming the head of pipe entrance is the same as that of the reservoir.

References - Section 3A.21

1. Streeter, V.L. and Wylie, E.G. Hydraulic Transients, McGraw-Hill Book Company, New York, NY, 1967.

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

TABLE 3A.21-1

INPUT DATA FOR WATHAM

Pipe No.	Total Length (ft)	Inside Diameter (ft)	Friction Factor	No. of Nodes	Nodal Span (ft)	Thickness (in)	Velocity (fps)
1	2,000	3.0	0.03	7	333.33	0.30824	4.24413
2	3,000	2.5	0.028	9	375	0.44	2.92132
3	2,000	2.0	0.024	6	400	0.50026	4.98473
4	1,800	1.5	0.02	7	300	0.11108	3.59336
5	1,500	1.5	0.022*	5	375	0.264	4.52142
6	1,600	1.5	0.025	6	320	0.13796	2.29183
7	2,200	2.5	0.04	8	314.29	0.21534	3.65878
8	1,500	2.0	0.03	6	300	0.14811	3.83245
9	2,000	3.0	0.024	7	333.33	0.30824	4.24413

NOTE: The initial heads of all nodes are calculated by using the Darcy-Weisbach equation.

* Friction factor in Pipe 5, 0.022, differs slightly from that of hand calculation, 0.020.

TABLE 3A.21-2

COMPARISON OF NODAL FORCE CALCULATION
AT TIME = 2.34 SEC

Pipe No.	Node No.	Force (kip) (WATHAM)	Force (kip) Hand (Calculation)
1	1	276.34	276.48
1	2	300.46	300.62
1	3	317.78	317.94
1	4	329.59	329.76
1	5	341.39	341.56
1	6	355.31	355.49
1	7	369.52	369.71

Nodal force calculation is based on the following equation:

$$F = A\left(\rho H + \frac{\rho}{g} V^2\right)$$

where:

F = nodal force, lb
ρ = density, lb/cu ft
H = nodal head, ft
g = 32.2 ft/sec^2
V = nodal velocity, fps
A = pipe area, sq ft

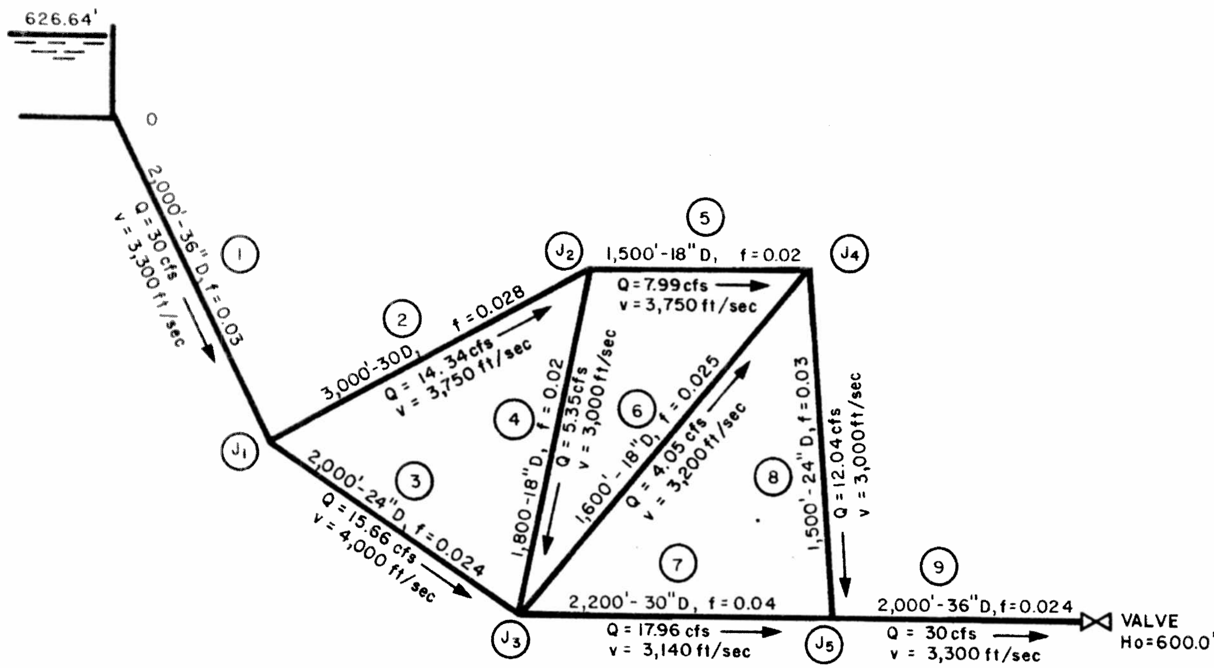

FIGURE 3A.21-1

HYDRAULIC NETWORK FOR VERIFICATION PROBLEM

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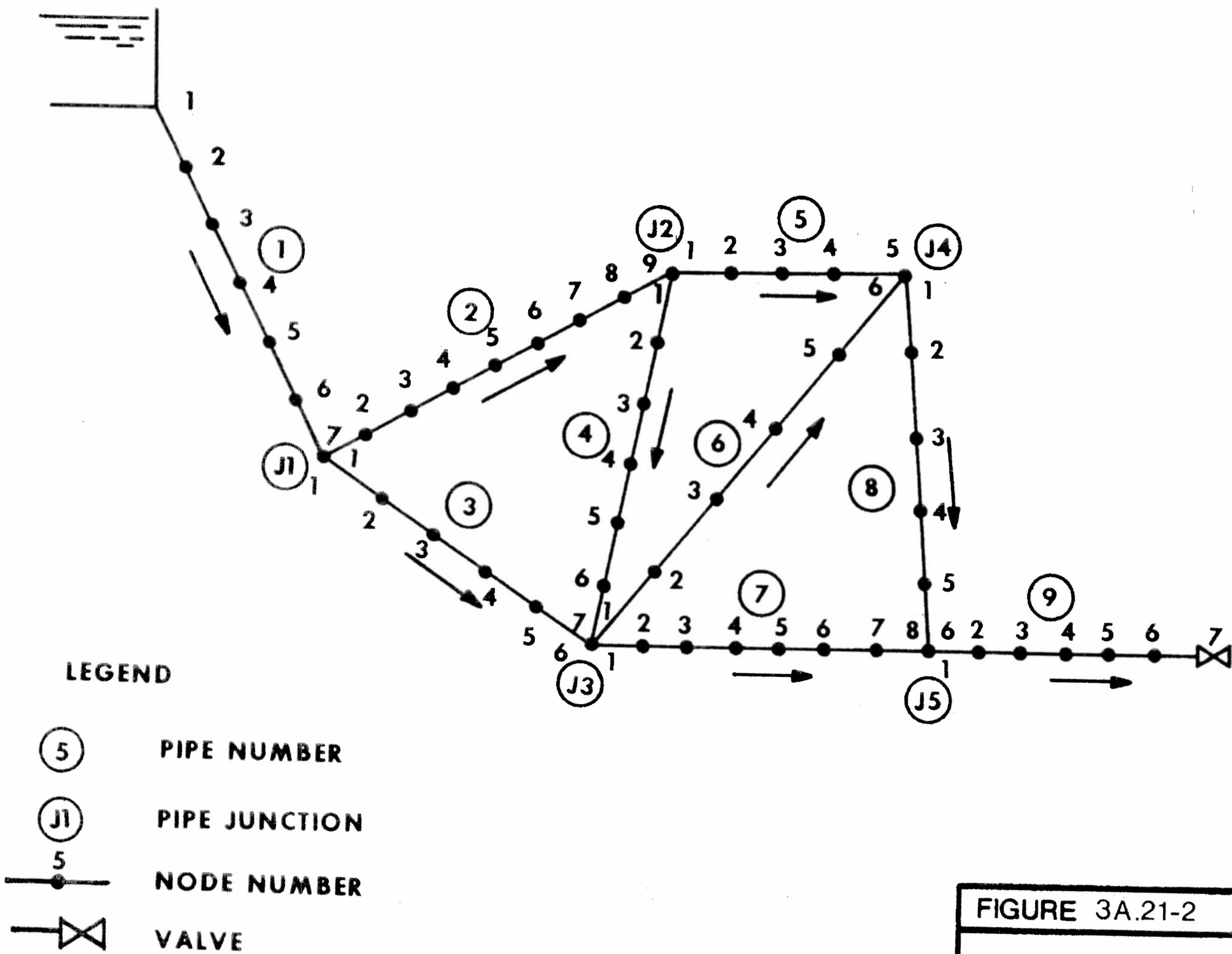

FIGURE 3A.21-2

HYDRAULIC NETWORK FOR
WATHAM VERIFICATION

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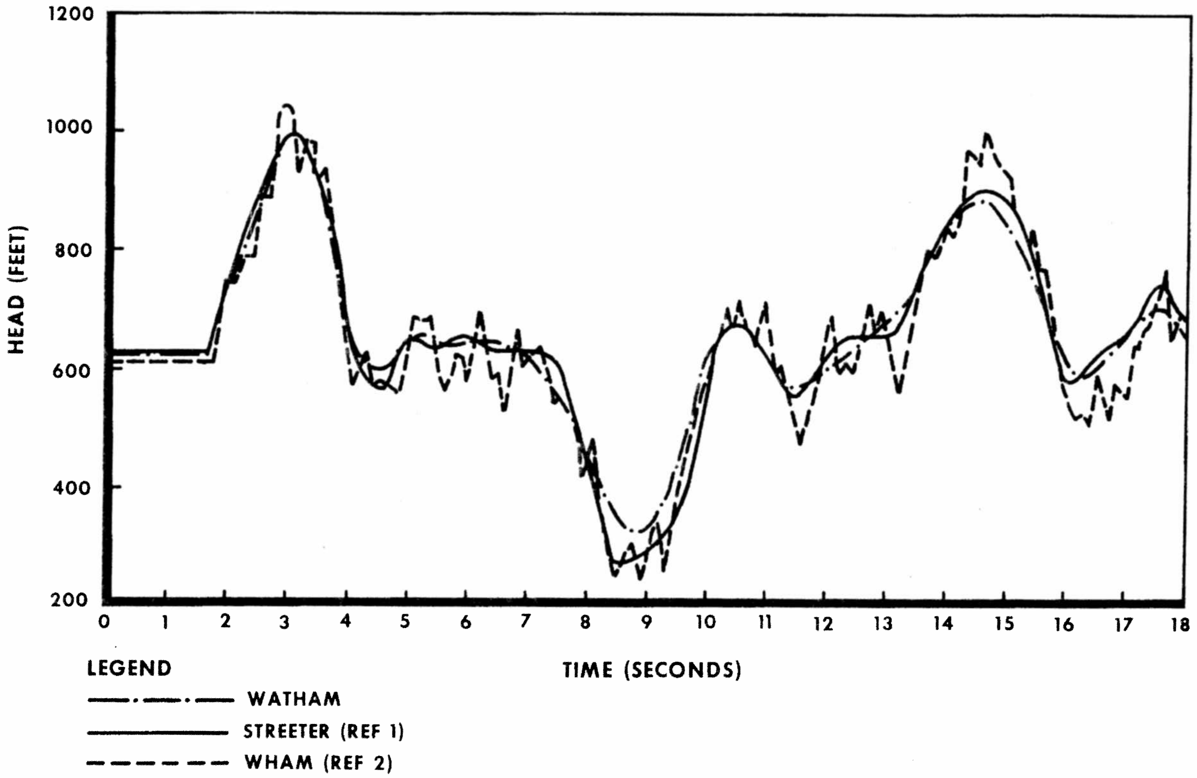

FIGURE 3A.21-3

HEAD-VERSUS-TIME PLOT
FOR JUNCTION J

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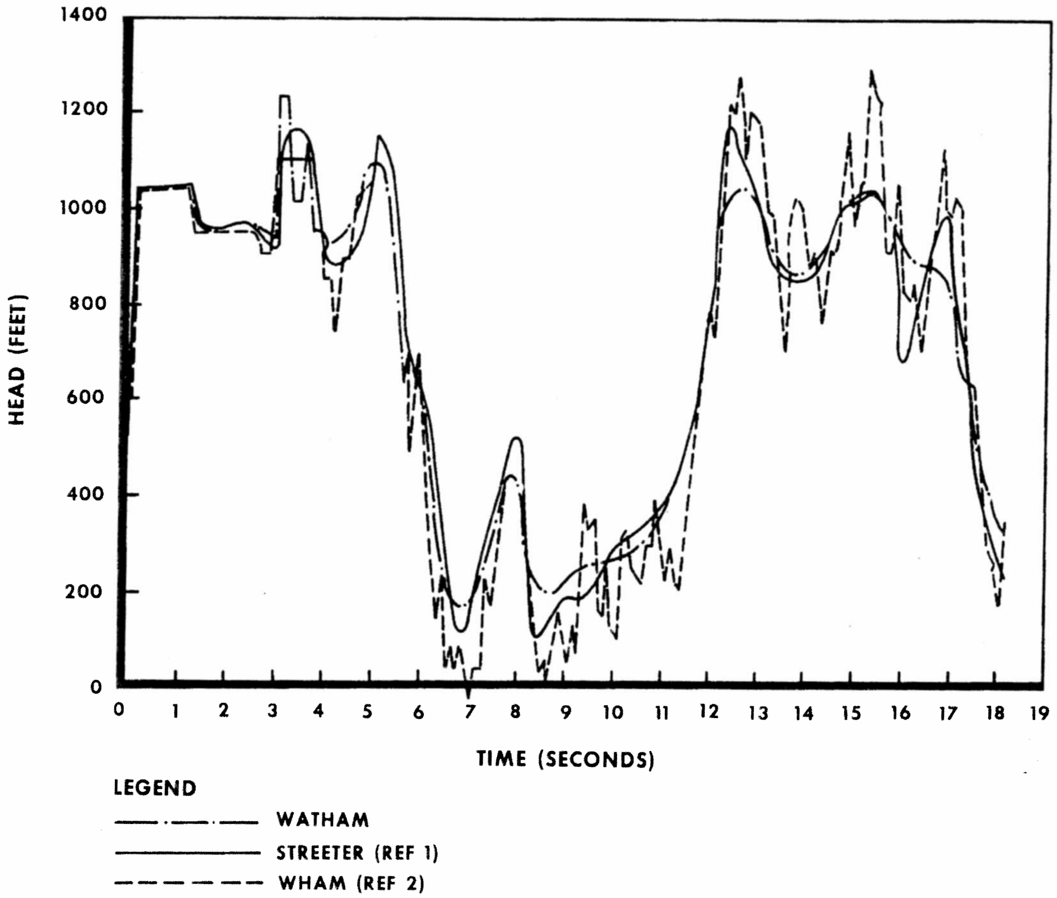

FIGURE 3A.21-4

HEAD-VERSUS-TIME PLOT AT VALVE

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3A.22 PITRUST

PITRUST is a program to calculate local stresses in the pipe caused by cylindrical welded attachments under external loadings. This program uses the Bijlaard method to calculate local stresses in the pipe wall caused by cylindrical welded attachments under external loadings, including pressure, dead load, thermal load, and combinations of maximum dynamic loads[(1)].

PITRUST has been verified by comparing its solution of a test problem to the solution of the same problem by an independently written piping local stress program, CYLNOZ, in the public domain. The CYLNOZ piping local stress program was written by Franklin Institute (Philadelphia, PA) and is presently used by engineering companies. The test problem is of a 72.375-in outside diameter by 0.375-in thick run pipe, reacting under an external loading condition of 1,000-lb force (normal and shear) and 1,000 in-lb bending and torsional moments transmitted by a 16-in outside diameter nozzle. A comparison of results is tabulated in Table 3A.22-1. The forces and moments are defined on Fig. 3A.221. PITRUST has also been verified by comparing its solution of the test problem to the experimental results obtained in Reference 2. A comparison of these results is tabulated in Table 3A.22-1.

References - Section 3A.22

1. Local Stress in Spherical and Cylindrical Shells due to External Loading. Welding Research Council Bulletin, WRC-107, 1965.

2. Corum, J. M. and Greenstreet, W. L. Experimental Elastic Stress Analysis of Cylinder to Cylinder Shell Models and Comparison with Theoretical Predictions. First International Conference on Structural Mechanics in Reactor Technology (Berlin, Preprints Vol. 3, Part G, 1971).

TABLE 3A.22-1

COMPARISON OF PITRUST WITH FRANKLIN INSTITUTE PROGRAM, CYLNOZ, AND HAND CALCULATION

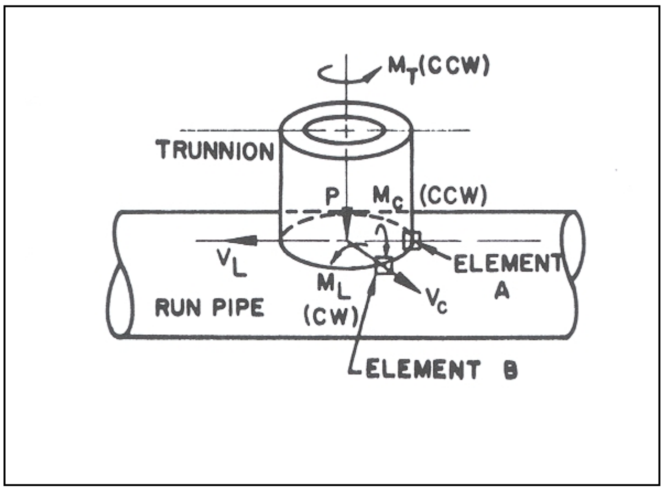
Source of Stress	Stress (psi) Franklin Institute Corrected Values	Output from PITRUST	Hand Calculation
Circumferential			
P (normal)	395	399	399.99
P (bending)	1,875	1,833	1,877.3
M_c (normal)	35.85	35.57	36.06
M_c (bending)	364.7	366.6	354.3
M_L (normal)	79.05	79.66	79.54
M_L (bending)	90.52	80.57	79.42
Axial			
P (normal)	813	812	814.8
P (bending)	812.3	827	810.6
M_c (normal)	91.79	105	95.45
M_c (bending)	158.8	160	158.8
M_L (normal)	37.06	37	37.12
M_L (bending)	117.9	105	103.85
Shear Stress by M_T	6.63	6.63	6.63
Shear Stress by V_c	106.1	106.1	106.1
Shear Stress by V_L	106.1	106.1	106.1

NOTE: For illustration of forces and moments see Fig. on Table 3A.22-2.

TABLE 3A.22-2

COMPARISON OF PITRUST WITH REFERENCE 2 RESULTS

Location and Cause	**PITRUST Results (psi)**	**Experimental Results (Ref.2) (psi)**
Element A		
Longitudinal Moment, M_L		
Circumferential Stress	20,438.9	20,000
Axial Stress	26,292.6	25,000
Element B		
Circumferential Moment, M_C		
Circumferential Stress	22,016.2	24,000
Axial Stress	13,105.8	13,000



3A.23 PILUG

PILUG is a program to calculate local stresses in the pipe wall caused by rectangular welded attachments under external loadings. This program uses the Bijlaard method to calculate local stresses in pipe wall caused by rectangular welded attachments under external loadings, including pressure, dead load, thermal load, and combinations of maximum dynamic loads[1]).

PILUG has been verified by comparing its solution of a test problem to results obtained by hand calculations using the formulations of Reference 1. A comparison of results is tabulated in Table 3A.23-1.

Reference - Section 3A.23

1. Local Stress in Spherical and Cylindrical Shells due to External Loading. Welding Research Council Bulletin, WRC-107, 1965.

TABLE 3A.23-1

COMPARISON OF PILUG COMPUTER PROGRAM OUTPUT
WITH HAND CALCULATIONS

Test Problem: Run Pipe Outside Diameter=17 in
Run Pipe Thickness=0.812 in
Axial Length of LUG=12 in
Width of LUG along Circumference=3 in
Loads: P=3300 lb; V_c =-1788 lb; V_L =2478 lb;
M_C =81834 in-lb; M_L =103320 in-lb
M_T =76284 in-lb

Stress in Circumferential Direction (psi):

Fig.[1]	β	Stress from Hand Calculation	Stress from Computer Output	Remarks
3C	0.5485	387	330	Membrane stress due to P
1C	0.326	2,165	2,160	Bending stress due to P
3A	0.294	671	629	Membrane stress due to M_c
1A	0.388	18,976	19,904	Bending stress due to M_c
3B	0.467	3,014	2,961	Membrane stress due to M_L
1B	0.416	6,143	5,969	Bending stress due to M_L

Stress in Axial Direction (psi):

Fig.[1]	β	Stress from Hand Calculation	Stress from Computer Output	Remarks
4C	0.4447	683	690	Membrane stress due to P
2C	0.4632	773	792	Bending stress due to P
4A	0.294	1,897	1,864	Membrane stress due to M_c
2A	0.550	6,357	5,942	Bending stress due to M_c
4B	0.467	2,365	2,328	Membrane stress due to M_L
2B	0.582	4,989.7	4,842	Bending stress due to M_L

TABLE 3A.23-1 (Cont)

Fig.[1]	β	Stress from Hand Calculation	Stress from Computer Output	Remarks
Shear Stress (psi):				
--	---	1,304.8	1,304.8	Shear stress due to M_T
--	---	-366.99	-366.99	Shear stress due to V_L
--	---	127.15	127.16	Shear stress due to V_c

NOTE: All the terms used in the test problem are defined in Reference 1.

(1) Local Stress in Spherical and Cylindrical Shells due to External Loading. Welding Research Council Bulletin, WRC-107, 1965.

3A.24 WATAIR

General Description

WATAIR is a computer program which is used to determine the waterhammer load on piping systems with trapped air. It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as pump startup and valve opening.

The analysis is based on a one-dimensional separated two phase flow model with ideal gas trapped between two incompressible liquids. Numerical integration is used to obtain the solution of the governing equations.

WATAIR input requires the geometry of the piping system, flow frictional coefficients, operational characteristics of pump and valve, and the initial flow conditions. The output provides the time history functions of the flow velocities, the pressure head of the air pocket, the pump discharge head and the inertial unbalanced force for each segment. It also lists the maximum value, and the time of its occurrence for each of the above parameters.

Program Verification

WATAIR is verified by comparing its solution of a test problem (Fig. 3A.24-1) to the results of the same problem obtained by an independent and verified computer program WATHAM[4]. Fig. 3A.24-2 gives the plot of the forcing function produced from WATAIR, while Fig. 3A.24-3 is from WATHAM. Table 3A.24-1 lists the imput data of the sample problem. Table 3A.24-2 compares the peak values of the unbalanced force and their time of occurrence. The WATAIR results are in good agreement with those from WATHAM both in shape and in values. Minor differences are due to the modeling differences. WATAIR uses incompressible flow solution for water; therefore, the effects of the acoustic waves are lost.

References-Section 3A.24

1. Martin, C.S. Entrapped Air in Pipelines. Second International Conference on Pressure Surges, September 22 through 24, 1976. London, BHRA Fluid Engineering Paper F.

2. Streeter, V.L. and Wylie, E.B. Hydraulic Transients. McGraw-Hill Company, Inc., New York, N.Y., 1967.

3. Karassik, I.J. et al. Pump Handbook. McGraw-Hill Company, Inc., New York, N.Y., 1976.

4. WATHAM, Stone & Webster Engineering Corporation Computer Program ME 168, Version 02.

TABLE 3A.24-1

INPUT DATA FOR WATAIR AND WATHAM

Pump	Suction head	42.9868 ft
	Rated head	2,980 ft
	Rated discharge velocity	7.5824 ft/sec
	Rated speed	4,550 rpm
	Accelerating time	5 sec
Pipe	Inside diameter	0.4801 ft
	Total length	287.3396 ft

TABLE 3A.24-2

COMPARISON OF THE FIVE LARGEST WATER HAMMER LOADS ON PIPING SEGMENTS

Segment Number	From WATAIR (lb_f)	Time (sec)	From WATHAM (lb_f)	Time (sec)
4	114.33	4.465	110.0	4.476
5	78.296	4.465	75.04	4.476
11	100.549	4.479	99.25	4.479
14	582.652	4.479	590.1	4.449
19	154.176	4.479	166.9	4.431

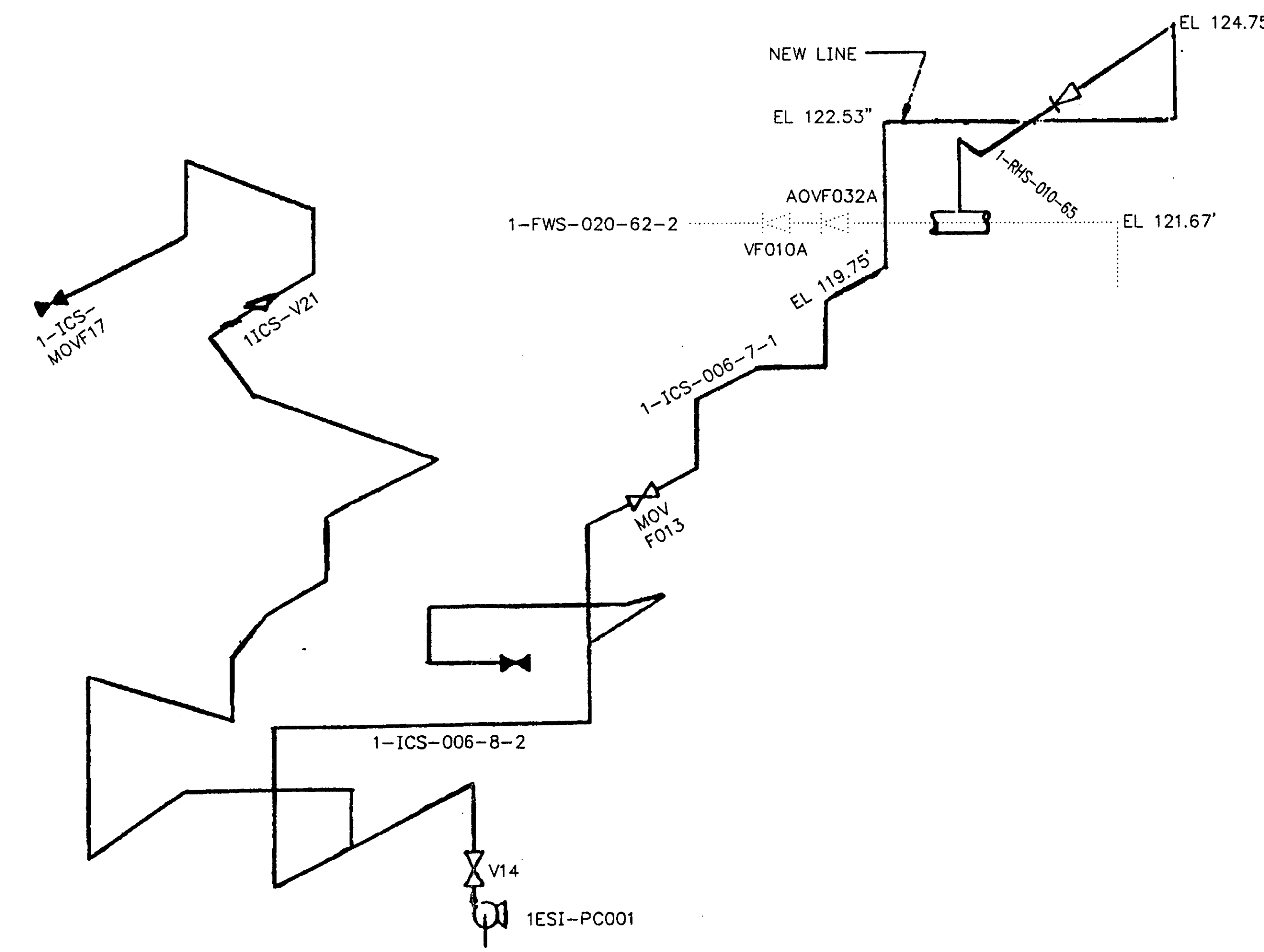


FIGURE 3A.24-1

RCIC PIPING SYSTEM USED IN THE ANALYSIS OF WATER HAMMER LOADS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

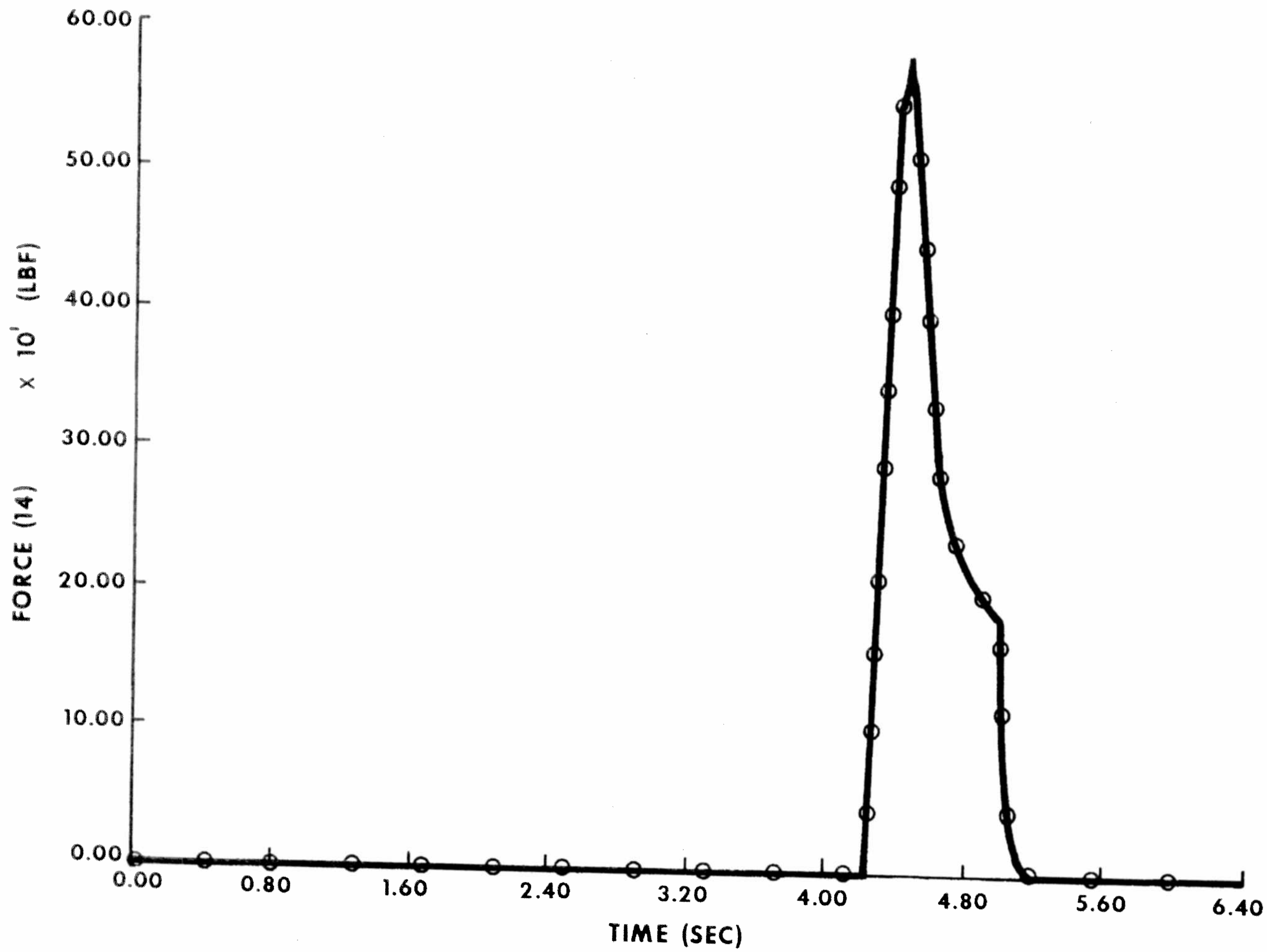


FIGURE 3A.24-2

TRANSIENT FORCE BY WATAIR IN PIPE SEGMENT 14

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

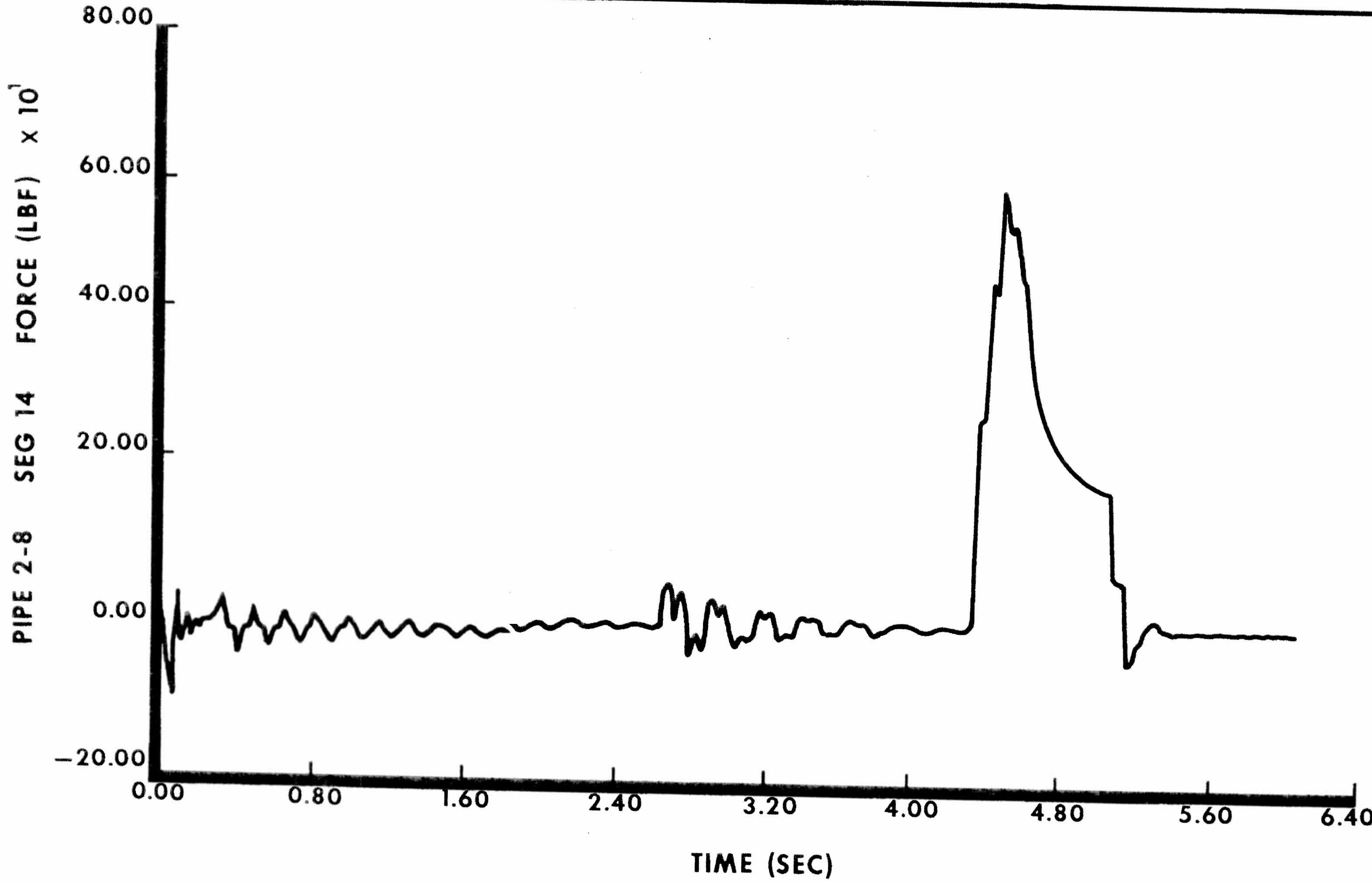


FIGURE 3A.24-3

TRANSIENT FORCE BY WATHAM IN PIPE SEGMENT 14

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

3A.25 ANSYS

The ANSYS engineering analysis system, developed by the Swanson Analysis System, Inc., is a fully warranted and documented computer program available at Control Data Corporation's 6600 data centers.

The ANSYS computer program, which has been used for production analysis since early 1970, is a large-scale, general-purpose computer program for the solution of several classes of engineering analysis problems. Analysis capabilities include: static and dynamic; plastic, creep, and swelling; small and large deflections; steady-state and transient heat transfer; and steady-state fluid flow.

The matrix displacement method of analysis, based upon finite element idealization, is employed throughout the program. The library of finite elements available contains more than 30 elements for static and dynamic analyses and more than 10 for heat transfer and fluid flow analyses. This variety of elements gives the ANSYS program the capability of analyzing frame structures (two-dimensional frames, grids, and three-dimensional frames), piping systems, two-dimensional plane and axisymmetric solids, flat plates, three-dimensional solids, axisymmetric and three-dimensional shells and nonlinear problems, including interfaces and cables.

Loading on the structure may be forces, displacements, pressures, temperatures, or response spectra. Loadings may be arbitrary time functions for linear and nonlinear dynamic analyses. Loadings for heat transfer analyses include: internal heat generation, convection, and radiation boundaries and specified temperatures or heat flows.

The ANSYS computer program was used in the analysis of the following items:

1. Drywell, containment vessel, and shield building access openings (Section 3.8.2.4.3)

2. Reactor building polar crane

3. Spent fuel cask crane

4. Standby diesel generator sets

5. Monorail systems

6. Dry transformers

7. Axial flow fans.

Qualification of the above equipment was performed by the applicable vendors and reviewed by Stone & Webster

Engineering Corporation. This computer program is considered verified by constant use and by the vendor's original documentation and qualification.

3A.26 PIPEDRAW/QUICKPIPE

The PIPEDRAW/QUICKPIPE computer programs, written by IMPELL Corporation of Norcross, GA, are fully warranted and documented computer programs.

The PIPEDRAW (ME-250) computer program is an application module for creating the three-dimensional modeling of piping system geometry and for generating the input data to interface with the QUICKPIPE stress analysis computer program.

QUICKPIPE (ME-251) performs the stress analysis for gravity, thermal, and dynamic load cases, as well as incorporating an algorithm to optimize pipe support locations and types. It was designed to interface primarily with the piping geometry created graphically by PIPEDRAW. Pipe stiffnesses are computed on an elemental basis which incorporates straight and curved elements and shear stiffness corrections. The elemental stiffnesses are combined via the direct stiffness method. The solution to the static equilibrium equations is a two-step process: triangularization followed by back substitution. Intermediate stresses for individual elements are then obtained by interpolation. For dynamic analyses, the mass distribution is checked by the program and if necessary, additional masses are automatically inserted. The mode shape is computed via the subspace iteration method.

These computer programs are verified by IMPELL Corporation in accordance with its documentation and Quality Assurance Program and audited by SWEC in accordance with the SWEC Engineering Assurance Program.

3A.27

The following additional computer programs have been used by Stone and Webster Engineering Corporation (SWEC) in the static and dynamic analysis of RBS Cat. I structures, systems, components, and supports. The verification and validation of these programs has been performed in accordance with the applicable Engineering Assurance Procedures, which are a part of the SWEC Quality Assurance Program. All qualification and history data for these programs is on file in the SWEC Computer Library.

PROGRAM	NUMBER	TITLE
ANCCOMB	(ME-145)	ANCHOR LOAD COMBINATION
ANISND	(NU-146)	A ONE DIMENSIONAL DISCRETE ORDINATES TRANSPORT CODE WITH ANSIOTROPIC SCATTERING
ANNULUS	(NU-099)	PRESSURE TEMPERATURE TRANSIENTS IN BUILDINGS BEING EXHAUSTED BY FANS
ANSYS-BCS	(ST-360)	ENGINEERING ANALYSIS SYSTEM (BCS)
APE	(ST-378)	ANCHORED PLATE EVALUATION
ARSINV	(ST-342)	ARS COMPUTATION FROM TIME HISTORY OUTPUT OF INVTRAN
BAP	(ST-383)	BASEPLATE ANALYSIS PROCESSOR
BASEPLATE II	(ME-225)	A PRE & POST PROCESSOR TO STARDYNE FOR ANALYSIS OF BASE PLATES
BEARST	(ME-155)	A PIPE BEARING STRESS
BIP	(ST-361)	BASEPLATE INVESTIGATION PROPROCESSORS
BLOAD	(ST-290)	NODAL FORCES DUE TO BODY FORCES ON SOLID FOR “STRUDL-HATCH”
BSPLT	(ME-237)	A PROGRAM FOR SIMPLE BASE PLATE DESIGNS
CCN-318	(ME-272)	ASME III CODE CASE N-318
CCN-392	(ME-262)	ASME III CODE CASE N-392
CHPLOT	(ME-179)	TIME HISTORY PLOTTING
COHORT II	(NU-145)	GENERAL PURPOSE MONTE CARLO RADIATION TRANSPORT CODE
CONSBA	(NU-169)	CONTAINMENT SMALL BREAK ACCIDENT CODE
CONSEC	(ST-266)	STRESS ANALYSIS OF REINFORCED CONCRETE SECTION
CONTORT	(NU-163)	CONTAINMENT AND REACTOR VESSEL TRANSIENT CODE
CORHYD	(NU-111)	POST DBA HYDROGEN CONCENTRATION IN CONTAINMENT
CWL	(ME-214)	CONTAINMENT WALL LOADING
DAMPING	(ST-240)	MODAL DAMPING
DET	(ME-128)	DEAD END TEMPERATURE
DPR0C2	(MS-081)	DATA PROCESSOR FOR LIMITA2

PROGRAM	NUMBER	TITLE
DPROC3	(MS-082)	DATA PROCESSOR FOR LIMITA3
DRAGON	(NU-115)	DOSE AND RADIOACTIVITY FROM NUCLEAR FACILITY GASEOUS OUTFLOWS
ELBOW	(ME-160)	DETAILED STRESSES IN ELBOWS
ENVELOPE	(ST-241)	ENVELOPING CURVES
ETA	(ME-284)	ELBOW TRUNNION ANALYSIS
FCASA	(ST-231)	FOUIER COEFFICIENTS OF DEFORMATIONS FOR ASAAS INPUT
FORCEO	(ST-242)	THIN WALL SECTION PROPERTY
FORCETH3	(MS-071)	PUNCHED FORCES FROM LIMITA3 FOR STARDYNE AND/OR TIMHIS6
FOURCO	(ST-245)	FOURIER DECOMPOSITION
FRIDAY	(ST-243)	FREQUENCY RESPONSE INTERACTION DYNAMIC ANALYSIS
FSERIES	(ST-251)	FOURIER COEFFICIENT
GAMTRAN1	(NU-003)	GAMA TRANSPORT BY POINT-KERNEL TECHNIQUE
GETARS	(ST-339)	ACCELERATION AMPLIFIED RESPONSE SPECTRA COMPUTATION
HEATING 6	(ME-266)	A MULTIDIMENSIONAL HEAT CONDUCTION ANALYSIS
INTBSL	(ST-307)	BASELINE CORRECTION AND INTEGRATION
INVTRAN	(ST-341)	COMPUTATION OF RESPONSE TIME HISTORY OF FORCING FUNCTION
ION EXCHANGER	(NU-009)	ION EXCHANGER
KFACTR	(ME-126)	FLEXIBILITY FACTORS FOR SPECIAL ELBOWS
LDCMB	(ME-257)	LOAD COMBINATION OF CLASS 1 AND 2 PIPING PENETRATION
LDCMBW	(ME-256)	CLASS 2 LOAD COMBINATION FOR DRYWELL WALL PENETRATIONS
LOADCM	(ME-169)	PIPE SUPPORT LOAD COMBINATION
LOADMC	(ME-258)	CLASS MC LOAD COMBINATION FOR CONTAINMENT PENETRATIONS
LUGSTR	(ME-170)	CLASS 1 PIPE LUG STRESS ANALYSIS
MASS	(ST-237)	MASS AND MASS MOMENTS OF INERTIA
MAX2	(MS-085)	DATA POST PROCESSOR FOR LIMITA2
MAX 3	(MS-084)	DATA POST PROCESSOR FOR LIMITA3
MESH 3D	(ST-234)	3D MESH GENERATOR FOR “STRUDL-HATCH” INPUT
NEWSECT	(ST-246)	REINFORCED CONCRETE SECTION STRESS ANALYSIS
NUDL	(ME-268)	NUPIPE TO STRUDL SUPPORT LOAD COMBINATIONS
NUPIPE II	(ME-207)	NUPIPE II (CONTROL DATA VERSION)
OEDCALC	(GT-019)	CONSOLIDATION TEST DATA REDUCTION

PROGRAM	NUMBER	TITLE
OEDPLOT	(GT-024)	CONSOLIDATION TEST PLOT
•→14		
PERC2	(NU-226)	PASSIVE/EVOLUTIONARY REGULATORY CONSEQUENCE CODE
PITRIFE	(ME-211)	PIPE TRUNNION INTERPOLATED STRESSES
PLOAD	(ST-235)	NODAL FORCES DUE TO PRESSURE LOAD FOR "STRUDL-HATCH"
PLOTIT2	(MS-087)	PLOTTER FOR LIMITA2
PLOTIT3	(MS-086)	PLOTTER FOR LIMITA3
PROFILE	(ST-207)	PROFILE PLOTTING PROGRAM
QADMOD	(NU-137)	POINT KERNAL TRANSPORT
QUAD-CGGP	(NU-222)	A COMBINATION GEOMETRY VERSION OF QAD-P5A
14←•		
QUAKE	(ST-306)	EARTHQUAKE SIMULATION
QUAKE2	(ST-312)	TIME HISTORY RESPONSE SPECTRA ANALYSIS RADIOISOTOPE
RADIOISOTOPE	(NU-007)	RADIOISOTOPE
REFUND	(ST-232)	STIFFNESS FUNCTIONS OF RECTANGULAR FOUNDATIONS
RIG3	(ST-248)	TRANSFORMATION FOR STIFFNESS MATRIX
RIG4	(ST-249)	STIFFNESS TRANSFORMATION
SANDUL	(ME-222)	STRUCTURAL ANALYSIS AND DESIGN BY THE UNIT LOAD METHOD
SBMMI	(ST-331)	SINGLE BARRIER MASS MISSILE IMPACT
SECPROP3	(ST-244)	GEOMETRIC PROPERTY
SETTLE-II	(GT-012)	SETTLEMENT OF A MULTI-LAYERED SOIL PROFILE
SNUFFE	(ME-267)	SUPPLEMENT TO NUPIPE-SW FOR FATIGUE EVALUATION
SSLAM	(ME-242)	SUBMERGED STRUCTURE LOAD ACCOUSTIC MODEL
SSLOAD	(ME-229)	SUBMERGED STRUCTURE LOAD
STARDYNE-BCS	(ST-362)	STARDYNE (BCS)
STRUDAT	(ME-223)	STRUDL POST PROCESSOR FOR SANDUL INPUT
STRUDL-GT-BCS	(ST-359)	STRUDL - GT (BCS)
STRUDL-GE-CDC	(ST-349)	STRUDL - GT (CONTROL DATA CORP)
STRUDL-SW	(ST-346)	STRUCTURAL DESIGN LANGUAGE
SUMSTRESS	(ST-303)	SUMMATION OF ASAAS STRESSES
TABLE	(ST-077)	ICES FILE STORAGE SUBSYSTEM
TCOEF	(ME-182)	THERMAL COEFFICIENTS FOR TAC2D
TOT	(ME-116)	THERMAL OUTPUT TRANSFORM
TRANFUN	(ST-340)	COMPUTATION OF TRANSFER FUNCTION AT A GIVEN NODE AND ANGLE
TRIPAT1	(ST-313)	TIME HISTORY RESPONSE SPECTRA PLOTTING
TRIPAT2	(ST-314)	RESPONSE SPECTRA PLOTTING
TRIPAT3	(ST-315)	SMOOTH RESPONSE SPECTRA PLOTTING
VES PEN ANAL	(ST-147)	VESSEL PENETRATION ANALYSIS
WALLMC	(ME-259)	CLASS (MC) LOAD COMBINATION FOR DRYWELL WALL PENETRATION
XLOAD	(ST-291)	NODAL FORCES DUE TO BODY FORCES ON PLATE FOR "STRUDL-HATCH"

3A.28

The following programs have been used by SWEC vendors in their static and dynamic analysis of Cat I structures, systems, components, and supports. The verification and validation of these programs was performed by the vendor, and is documented in the stress reports listed. The stress reports, which are on file at River Bend Station, also give the titles and functions of these programs.

PROGRAM	VENDOR	STRESS REPORT
AEVIBES	ACTION ENVIRONMENTAL TESTING INC	4242.444-275-005E
ANSYS	ACTION ENVIRONMENTAL TESTING INC	4242.444-275-010A
STARDYNE	ACTION ENVIRONMENTAL TESTING INC	4242.444-275-005E
NASTRAN	AEROFIN	4215.252-057-005C
STRESST	AMERICAN AIR FILTER	4225.220-115-003F
STRUDL/ DYNAL	AMERICAN AIR FILTER	4225.220-115-003F
ANSYS	BUFFALO FORGE CO.	4215.252-057-005C
NO. 706	CERAMIC COOLING TOWER	6232.530-087-001D
NP490100	CERAMIC COOLING TOWER	4232.530-087-012E
NP490110	CERAMIC COOLING TOWER	4232.530-087-012E
PIPESD	CERAMIC COOLING TOWER	4232.530-087-005C
PIPESD/ HEAT	CERAMIC COOLING TOWER	4232.530-087-005C
SEISMIC4	FISHER VALVE CO.	4247.491-163-002A
SAP4	GOULDS PUMPS INC.	4237.160-108-012A
STRUDL-II	GOULDS PUMPS INC.	4237.160-108-001B
STRUDL-II	HAYWARD TYLER PUMP CO.	4232.920-257-001C
NASTRAN	JAMESBURY CORP.	4228.243-105-006B
P-STAR	JAMESBURY CORP.	4228.243-105-001A
STARDYNE	JAMESBURY CORP.	4228.243-105-001A
ANSYS	M.P.R. INC.	4219.717-056-001D
ASHAD	M.P.R. INC.	4219.717-056-001D
STARDYNE	NUCLEAR QUALIFICATION SERVICES	4247-411-296-003C
ANCHOR	PATEL ENTERPRISES INC.	4242.533-265-001C
CAPOFF	REACTOR CONTROLS INC.	4228.180-285-001I
E17POST	REACTOR CONTROLS INC.	4228.180-285-045C
E2A17	REACTOR CONTROLS INC.	4228.180-285-001I
E2PLOT	REACTOR CONTROLS INC.	4228.180-285-045C
EASE2	REACTOR CONTROLS INC.	4228.180-285-001I
EPLATE	REACTOR CONTROLS INC.	4228.180-285-001I
EWELD	REACTOR CONTROLS INC.	4228.180-285-001I
FREEA17	REACTOR CONTROLS INC.	4228.180-285-045C
SPECONV	REACTOR CONTROLS INC.	4228.180-285-001I
SPECTRA	REACTOR CONTROLS INC.	4228.180-285-001I
SPLOT	REACTOR CONTROLS INC.	4228.180-285-045C

PROGRAM	VENDOR	STRESS REPORT
TPIPE	REACTOR CONTROLS INC.	4228.180-285-001I
TPRLS	REACTOR CONTROLS INC.	4228.180-285-045C
ANSYS	TRANS AMERICA DELAVAL	4244.700-041-005A
PIPESD	TRANS AMERICA DELAVAL	4244.700-041-003C
SPICE 2	TRANS AMERICA DELAVAL	6244-700-041-030E
SUPERB	TRANS AMERICA DELAVAL	4244.700-041-008C
TREMOR	TRANS AMERICA DELAVAL	4244.700-041-001E
HYDN	VELAN VALVE CORP.	4228.211-049-007C
SA1	VELAN VALVE CORP.	4228.211-049-007C
SA1FL	VELAN VALVE CORP.	4228.211-049-007C
SA3	VELAN VALVE CORP.	4228.211-049-007C
SA476.2	VELAN VALVE CORP.	4228.211-049-007C
SA4PR	VELAN VALVE CORP.	4228.211-049-007C
SA6	VELAN VALVE CORP.	4228.211-049-007C
SA7	VELAN VALVE CORP.	4228.211-049-007C
SA8	VELAN VALVE CORP.	4228.211-049-007C
S01	VELAN VALVE CORP.	4228.212-049-015C
ME-7701	WESTINGHOUSE	6215.400-071-028B
ANSYS	WOOLLEY	4219.711-056-001E
WSTP	WOOLLEY	6219.710-056-016B

•→7

The following program has been used by a SWEC vendor for the thermohydraulic analysis required to support small bore pipe stress analysis and finite element analysis of the reactor water level condensing chambers. This was done in support of the design of the reactor water level reference leg continuous backfill modification MR 93-0034. SWEC has verified that this program is appropriately qualified for performance of safety related analyses. This program is refeenced in calculation G13.18.4.0*23, 3-D Thermal Hydraulic Analyses of Condensing Chamber for MR93-0034.

PROGRAM	VENDOR
FLUENT	FLUENT Incorporated

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3A.29

The following program has been used by EBASCO in analysis of steam piping for installation of nozzle check valves.

DST/PIPESTRESS 4203, COMPUTER PROGRAM FOR PIPING STRESS ANALYSIS

•→9
3A.30 SUPERPIPE

SUPERPIPE, a public domain program developed by Vectra Technologies Inc., is a linear finite element program for the static and dynamic analysis of elbows, tees, reducers, socket or butt welds, flexible couplings, and flanges, with the appropriate flexibility factors and stress indices accounted for. Support types may include rigid, spring, constant force, snubber, anchor, or user-specified, and may have any desired orientation.

Analyses performed include thermal, weight, applied load, frequency and mode shape, response spectrum, and time-history. Following the static and dynamic analysis phase, the program performs a complete ASME B&PV Code, Section III Class 1 stress check, combining analysis results in any manner specified by the user to create the appropriate loading cases applicable for each of the ASME code stress equations. The user also supplies the number of occurrences of each steady-state and transient load state, with which the program performs a complete fatigue damage calculation.

SUPERPIPE has been thoroughly tested and verified for a comprehensive set of sample problems, including extensive comparison with several publicly available programs and ASME benchmark problems. All verification analyses have been documented in accordance with established Vectra Quality Assurance procedures.

SUPERPIPE satisfies all quality assurance and verification checks required by the United States Nuclear Regulatory Commission (USNRC). Specifically SUPERPIPE has been verified to benchmark problems contained in:

> NUREG/CR-1677 Vol. 1 Piping Benchmark Problems, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

In addition, SUPERPIPE has been verified to benchmark problems for multiple response spectra method in response to the USNRC request for additional verification of computer codes for analysis of nuclear piping systems.
9←•

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3A.31

The following program has been used by SWEC in analysis of Suppression Pool Platform to evaluate the platform and to determine various dynamic load factors (DLFs):

MICAS PLUS

Micas Plus is an integrated structural engineering package providing powerful tools for structural modeling, analysis, design and drawing production. It is written and marketed by Intergraph Corporation.

The Micas Plus program has three modules: Modeldraft, Analysis and Design. Modeldraft is a drafting package and is intended for geometry input and subsequent creation of a mathematical model. For indeterminate or determinate structural analysis, the "Analysis" module would be used. The Design module is used to design members using AISC codes. Since all three modules share the same database, the loading input for the analysis in the "Analysis" module can be used by the "Modeldraft" module to generate load location drawings. It integrates the engineering and 2D/3D graphics capabilities of the Intergraph System by using a stand-alone workstation with a UNIX operating system. This stand-alone workstation has the capability to emulate DOS for compatibility with PC-based programs.

3A.31.1

The following program has been used by SWEC in analysis of Suppression Pool Platform for the evaluation of the containment weld pads:

ME-323 (PC PREPS), Preparation and Revision of Pipe Support Analysis.

The PC PREPS Computer program written by Stone and Webster Engineering Corporation is a fully approved and documented computer program for Category I application. PC PREPS is a PC-based, integrated pipe support analysis software package. It is interactive, menu-driven, with built-in structural analysis and graphics capabilities. This program allows a pipe support analyst to prepare data, view associated graphics, and execute frame and baseplate analysis.
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3A.32 Computer programs used by GSU (Gulf State Utility) and Entergy RBS

The following computer programs are added for Dynamic and Static Analysis of Seismic Category I Structures, Equipment, and Components. These computer programs have been verified to the requirements of applicable site and corporate procedures.

3A.32.1 HYTRAN (Hydraulic Transient Analysis Program)

HYTRAN is a PC-based program for fluid transient analysis that computes the time-dependent forces, the pressures, and flow velocities in the legs of a liquid-filled piping system. Starting from an initial steady state, the solution is advanced step by step in time until the user-selected end time is reached. Transients may be initiated by pump start-up or trip, valve opening or closure, or by variations of pressures or flow at an exterior node in the network.

The input consists of an ASCII file describing the system configuration. The input data formatting is described in the user manual. Printed output of head and flow velocity at the ends of each leg and of the unbalanced forces acting on them is available at the user's option. Screen or printer plots of leg force or nodal pressure/flow time histories can also be created.

HYTRAN is needed to expedite and insure accurate calculations of forces caused by hydraulic transient events in the piping systems. Calculations are to be performed on actual or postulated fluid transients.

3A.32.2 pc-CRACK (Fracture Mechanics Software for Personal Computers)

pc-CRACK is a stand-alone, efficient software package for performing fracture mechanics analyses on a wide variety of structural components and materials. The purpose of this software is to provide an easy-to-use efficient tool for engineers and metallurgists to perform analyses and immediately visualize results so that appropriate engineering decisions can be formulated in the shortest possible time.

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- Creep Fracture Mechanics (CFM)
- Codes and Standards Evaluation
- Graphics
- File Management

The program is designed in modular form, thus enabling it to be easily upgraded to incorporate newly developed technologies in the area of fracture mechanics and failure analysis as they become available. The currently released version is 2.1.

3A.32.3 FatiguePro (River Bend Station - FatiguePro Fatigue Monitoring System)

Fatigue Pro is a computer program developed by EPRI and Structural Integrity Associates (SI), which automatically identifies and records plant transient events (cycle counting) and provides fatigue damage monitoring and stress evaluation for critical components in reactor coolant systems using plant instrument data as input. The cycle counting function uses algorithms that logically evaluate plant instruments to identify the occurrence of certain predefined plant events as specified in Project Plan. The fatigue monitoring and stress evaluation function uses Green's Functions and/or global-to-local functions to develop transient stresses and compute fatigue at fatigue-critical locations in the plant as listed in the Project Plan.

The FatiguePro software has been designed to reduce conservatism in assessment of power plant components. Standard methods of life assessment assume an upper bound for damage from any event, based on design-basis events. FatiguePro allows a utility to account for the actual events, which are generally less severe than the design basis. It does this by performing a real-time stress analysis using actual plant data, so that actual damage can be computed. This damage is accumulated into "fatigue usage factor", a number between 0 and 1 representing the amount of life that has been consumed by operation so far.

3A.32.4 COLLECT (Data Collection Program for FatiguePro Fatigue Monitoring System)

COLLECT computer program was developed for the purpose of data collection at River Bend Station (RBS). The purpose of this software is to provide a reliable and efficient way to collect and permanently store plant instrument data for further use by FatiguePro Fatigue Monitoring Program.

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The COLLECT program collects data sent through serial link from the ERIS computer, and stores it in daily files. The data is being sent from the ERIS computer by a communication program named FMS, which is written in-house by RBS engineers.

3A.32.5 ME101 (Linear Elastic Analysis of Piping)

ME1O1 is a finite element computer program that performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program. In addition, modifications aimed at achieving an improved model are performed automatically. The output may be used directly for piping design and for conformation to code and other ANSI Standard regulatory requirements. Two piping codes, ANSI Code, 1974 and ANSI Standard B31.1, Summer 1973 Addenda, are incorporated into the program to the extent of computing flexibility factors, stress intensification factors and stresses.

ME101 may be used for static, seismic and time history analysis of piping systems.

ME101 generates isometric plots of the piping configuration with optional node numbering.

3A.32.6 ME150 (FAPPS) (Frame Analysis Program for Pipe Supports)

The Frame Analysis Program for Pipe Supports (FAPPS) is an interactive computer program for the analysis and design of standard (simplified input) and nonstandard pipe support frames. It includes the capability to optimize member sizes, welds, baseplates and embedments contingent upon various user-specified design limits.

3A.32.7 ME035 (BASEPLATE)

BASEPLATE is a computer program for analysis of standard (simplified input) and nonstandard baseplates and embedment plates for safety and non-safety related applications.
12←•

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3A.32.8 ME153 (Miscellaneous Application Programs for Pipe Supports)

The Miscellaneous Application Programs for Pipe Supports (MAPPS) is an interactive computer program that enables the user to access any or all of the following eleven suppose analysis computer programs within one run:

Uniform Weld, Non-Uniform Weld (NuWELD), Beta Angle, Clip Angle, Bolt Spacing, Anchor Plate, Local Effects, Clamp (PiCLAMP), BR2PRE, 4bolt, Loading Transformation.

The MAPPS program displays these eleven programs on the terminal screen in the form of a menu, and allows the user to choose as many of the programs as necessary, without requiring the user to log in and out.

3A.32.9 ME152 (Standard Frame Analysis Program for Pipe Supports)

The Standard Frame Analysis Program for Pipe Supports (SMAPPS) is a user friendly, interactive computer program which can develop new pipe support designs, as well as qualify existing, as-built designs.

SMAPPS has complete analysis capabilities for six commonly encountered support frame configurations. Support can be analyzed to satisfy all design requirements for member stress, deflection, stiffness, welds and baseplates, all within one run. Therefore, no supplementary hand or computer calculations are necessary.

3A.32.10 ALGOR (Finite Element Linear and Non-linear Analysis Program)

ALGOR is a high-end finite element structural analysis and design software package. Program capabilities include two dimensional (2D) and three dimensional (3D) finite element analysis using variety of element types, which include 4 to 8 node bricks and 4 to 10 node tetrahedral, just to name a few. Input/Graphic files may be input to create by using ALGOR's graphics editors. Output files may also be viewed buy using graphical editors/viewers in ALGOR.

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ALGOR's capabilities include:

- Linear Stress Analysis
- Linear Natural Frequency Analysis
- Time History Analysis
- Response Spectrum Analysis
- Buckling Analysis (for beams and shells)
- Linear natural frequency Analysis (with load stiffening)
- Mass Properties Analysis
- Heat Transfer Analysis (foe steady state and transient conditions),
- Electrostatics Analysis
- Composite Stress Analysis
- Gap/Cable Stress Analysis
- Composite Natural Frequency Analysis
- Random Vibration Analysis
- Two Dimensional Fluid Flow Analysis
- Three Dimensional Fluid Flow Analysis

3A.32.11 <u>IMAGES</u> (Finite Element Analysis Program)

IMAGES is a full scale software program which performs finite element and structural analysis and design functions.

The program contains three (3) main modules:

- IMAGES- 3D for 2D add 3D Static, Modal and Dynamic Analyses
- IMAGES- THERMAL for conducting Thermal Analysis
- IMAGES- AISC performs design of steel structures to AISC rules

IMAGES has a variety of load combination options available including algebraic and absolute value summations and SRSS. The program compute element displacements, forces, reactions, & stresses.

3A.32.12 <u>STAAD III</u> (Structural Analysis And Design)

STAAD III is a full-scale structural analysis and design software.

STAAD III capabilities include 2D and 3D static, dynamic, P-delta analysis with frame, shell, and plate finite elements.

STAAD III can be used for steel, concrete, and timber as design materials.

STAAD III supports AISC, LRFD, ACI and AASHTO design codes.
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3A.32.13 <u>NISA II</u> (Numerically Integrated elements for System Analysis)

NISA II(Numerically Integrated elements for System Analysis - static analysis) is a general purpose finite element program developed and maintained by Engineering Mechanics Research Corporation (ENMC), Troy, MI. NISA II provides Linear and Nonlinear Structural, Buckling, Eigenvalue, and Linear Direct Transient Analysis capabilities. NISA II interfaces with EMRCs 3-D color graphics finite element pre- and post-processor, DISPLAY III, which can be used to generate input files and interpret results.

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APPENDIX 3B

PRESSURE ANALYSIS FOR SUBCOMPARTMENTS OUTSIDE CONTAINMENT

APPENDIX 3B

PRESSURE ANALYSIS FOR SUBCOMPARTMENTS OUTSIDE CONTAINMENT

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PRESSURE ANALYSIS FOR SUBCOMPARTMENTS OUTSIDE CONTAINMENT

The current computer code used for High Energy Line Break (HELB) analysis is GOTHIC. This computer code is described in submittals to the NRC dated May 14, 2002, June 27 and July 9, 2003, April 7 and May 12, 2004, as approved in Amendment 139 to NPF-47 dated May 20, 2004.

This computer code includes credit for fluid friction described in the initial submittal and included in the NRC SER, Section 3.4.

Also when performing these analyses, RBS assumes homogenous equilibrium conditions and 100% water entrainment for all breaks unless it is more conservative to not employ these assumptions as in the case of breaks involving fluid which is initially highly subcooled. This analysis is accomplished by disabling the forced equilibrium (i.e., enabling thermal hydraulic non-equilibrium model) and enabling the drop-liquid conversion model in GOTHIC.

3B.1 DESIGN BASES

Pressure response analyses were performed for the structural design basis of the main steam tunnel and other subcompartments in the auxiliary building for postulated ruptures of high-energy piping. The definitions for high energy and criteria for protection against dynamic effects associated with postulated rupture of piping are given in Section 3.6A. The analyses were performed using SWEC computer code THREED (Appendix 6B). The Auxiliary Building analyses were performed using the GOTHIC computer program as described above.

The auxiliary building model consists of 50 control volumes of which 34 control volumes represent physical spaces in the auxiliary building, the remainder representing mixing volumes and boundary volumes. The 34 control volumes represent 31 sub-compartments / environmental zones. The main steam tunnel was divided into four separate subcompartments for its design evaluation. A fifth node was used to represent the turbine building, and a sixth node represents the outside atmosphere. The subcompartment boundaries were chosen to represent physical restrictions to flow and to reflect additional detail in the vicinity of the high-energy lines.

Breaks were postulated in each auxiliary building volume containing a high-energy line. Breaks were postulated in the main steam tunnel on both sides of the jet impingement shield wall which bounds the break exclusion zone. All breaks were considered to be instantaneous circumferential double-ended ruptures (DER), i.e., the break area was equal to twice the effective cross-sectional flow area of the pipe, except that single-ended ruptures (SER) were considered in the main steam tunnel break exclusion zone. Section 3.6A defines the complete set of break locations in high-energy piping outside containment from which the design basis breaks for subcompartment pressurization were selected.

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During isolation valve closure, the flow area used for mass and energy release calculations was assumed to be constant until the valve area equaled the flow limiting area. Subsequently, the limiting flow area was linearly reduced to zero.

Auxiliary building high-energy lines were identified in the reactor water cleanup (RWCU) system, the reactor core isolation cooling (RCIC) system.

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A total of three break locations were postulated and analyzed. Peak calculated pressure differentials for the 20 subcompartments were generated by two of the four postulated breaks. Table 3B-1 lists all postulated breaks and identifies the two breaks that determined the design differential pressures.

The main steam tunnel analysis considered feedwater, RCIC, and main steam line breaks. Main steam line break analyses were performed assuming a two-phase blowdown. Four combinations of break locations and blowdown conditions were postulated and analyzed. Peak differential pressure values were generated by the two-phase blowdown breaks. Table 3B-2 lists the postulated line breaks and identifies the two breaks that determined the design differential pressures for the steam tunnel.

3B.2 DESIGN FEATURES

Fig. 1.2-13 through 1.2-19 show the piping and equipment in the subcompartments. The original plant design included a louver arrangement in the main steam tunnel chimney area. There are six louvered panels, three on the east side and three on the west side of the chimney (el 170'-0"). These louvers opened at a differential pressure of 3.25 psi, with an opening time of 0.3 sec. The louvers' blade skins have been removed and the current design now uses blowout panels instead.

All high-energy piping with a potential for producing high pressure and/or temperature environmental conditions in the auxiliary building is routed from the primary containment through the main steam tunnel. The RWCU pump rooms and RCIC turbine pump room are located directly below the steam tunnel, thus minimizing the length of high-energy piping outside the tunnel.

Fast closing, motor-operated isolation valves are located inside and outside containment on each high-energy line except feedwater lines, which utilize check valves to isolate reverse flow from the reactor to postulated pipe breaks outside containment. The outboard isolation valves are located in the steam tunnel break exclusion zone. The isolation valves are automatically closed by signals from the leak detection system, e.g., high local area temperature. Isolation of pipe breaks is also initiated by system high flow and other signals as described in Section 6.2.4.

Pressuretight doors designed to withstand a differential pressure of 3.0 psi are utilized to isolate ECCS equipment cubicles from the effects of high-energy line breaks. These doors are administratively controlled closed.

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Two fire doors, A95/8 and A95/9, are maintained open for pressure relief purposes by fusible links which allow the doors to close at temperatures of 225° F or more. The pressure analysis assumed these doors to be only 50-percent open, and the maximum temperature in this area after the worst-case high-energy line break is less than 225° F.

3B.3 DESIGN EVALUATION

Subcompartment nodalization schemes were selected to maximize differential pressures across node boundaries. Structural components were selected as node boundaries. The differential pressure transients across node boundaries are used to determine the structural adequacy and component support design.

The nodalization scheme used in the auxiliary building analysis is sufficiently complex that a graphic representation is not practical. Fig. 3B-22 shows the nodalization scheme for the main steam tunnel.

Table 3B-3 provides the nodal descriptions and gives the peak calculated and design differential pressures within the auxiliary building. Table 3B-4 similarly shows the subcompartment nodal descriptions for the main steam tunnel and identifies the calculated and design peak differential pressures.

In calculating the pressure differentials across the auxiliary building subcompartment walls, it is possible to take credit for the pressurization of the volume on the opposite side of the wall in question. This procedure, however, leads to slightly different pressure differentials for all walls of the subcompartment in question. To minimize the number of differential pressures to be considered and for conservatism, a single differential pressure was calculated for each volume by subtracting 14.7 psia from each of the calculated nodal absolute pressures.

Peak pressure values for the main steam tunnel subcompartments also were calculated by subtracting 14.7 psia from the peak pressure values.

Table 3B-5 gives vent flow path data for the auxiliary building. Table 3B-6 presents the vent path description corresponding to that shown on Fig. 3B-22 for the main steam tunnel.

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Tables 3B-7 through 3B-9 provide the mass and energy release data for the breaks that determine the design differential pressures within the auxiliary building.

In general, Moody[1] or Henry-Fauske[2] flow was assumed (for saturated and subcooled flows, respectively) at the limiting downstream and upstream flow areas. During the inventory period, the mass and energy release data were calculated using the methodology of NEDO-20533[3], except that the Henry-Fauske model was used to calculate subcooled flow.

Partial credit is taken for the effect of friction on reducing the rate of blowdown.

The mass and energy release data used for the postulated main steam tunnel pipe breaks are presented in Tables 3B-11 through 3B-14. These blowdowns were based entirely on frictionless Moody flow with a constant reservoir pressure. The blowdown was considered to be all steam for the first second after the accident. After 1 sec, the two-phase froth level rising in the vessel was assumed to discharge through the main steam lines. The quality of this part of the blowdown was assumed to be 7 percent.

The exposed surfaces of concrete and steel in each auxiliary building node were modeled as heat sinks in the analysis. The 2-ft thick concrete walls, ceiling, and floors were assumed to be only 1-ft thick, absorbing heat from the transient thermal environment in the respective node and insulated on the other side. The steel heat sinks include the beams, columns, posts, stairs, and platforms in the respective node. An equivalent steel slab was derived by dividing the total steel volume by the total exposed steel surface area. The UCHIDA heat transfer coefficient was applied, and condensate revaporization was assumed to be limited to 8 percent. The heat sink slabs for the auxiliary building 20-node model are defined in Table 3B-15.

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Concrete and steel heat sinks were modeled similarly in the steam tunnel 6-node model, except that the concrete slabs were assumed to be 1-ft thick, based on actual slabs which are 4-ft thick. Table 3B-16 summarizes these heat slabs.

The initial conditions in each node were assumed to be the maximum normal temperature, 14.7-psia pressure, and 100-percent relative humidity.

Fig. 3B-2 through 3B-21 and 27 through 37 provide the absolute pressure transient plots for the 20 subcompartments in the auxiliary building.

Fig. 3B-23 and 3B-25 provide the absolute pressure transient plots for the break of a 24" main steam line for the two main steam tunnel subcompartments within the auxiliary building portion of the tunnel. The peak pressures resulting from the breaks of the 8" RCIC/RHR steam line are provided in Table 3B-17.

1←•

References - 3B.4

1. Moody, F. J. Maximum Flow Rate of a Single Component Two-Phase Mixture, Journal of Heat Transfer, Trans. ASME, 87, February 1965, p 134-142.

2. Henry, R. E. and Fauske, H. K. The Two-Phase Critical Flow of One Component Mixtures in Nozzles, Orifices, and Short Tubes, Journal of Heat Transfer, Trans. ASME, 93, May 1971, p 179-187.

3. NEDO-20533, Mark III Containment System Analytical Model, Appendix B, Pipe Inventory Blowdown, June 1974.

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4. Lahey, R. T. and Moody, F. J. The Thermal-Hydraulics of a Boiling Water Nuclear Reactor, ANS, 1977.

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TABLE 3B-1

HIGH-ENERGY LINE BREAKS
AUXILIARY BUILDING

Break No.	Line	Type	Break In Node	Design Break for Nodes [2]
1	3″ RWCU	Liquid	9	[2]
2	6″ RWCU	Liquid	11	All Nodes
3	4″ RCIC	Steam	3	[3]

[1]All breaks are assumed to be double-ended ruptures.

[2]Subcompartment nodes are defined in Table 3B-3.

[3]Break does not generate design pressure for any node.

TABLE 3B-2

HIGH-ENERGY LINE BREAKS
MAIN STEAM TUNNEL
6-NODE MODEL

Break No.	Line	Break in Node	Design Break for Nodes(1)
1	24" main steam line DER[2]	2	2,3,5
2	24" main steam line SER[3]	1	1
3	8" RCIC DER[2]	2	[4]
4	8" RCIC SER[3]	1	[4]

[1]Subcompartment nodes are defined in Table 3B-4 and on Fig. 3B-22.

[2]Double-ended rupture (DER).

[3]Single-ended rupture (SER) assumes split or lomgitudinal rupture with flow area equal to the pipe cross-sectional area.

[4]Break does not generate design pressure for any node.

TABLE 3B-3
SUBCOMPARTMENT NODAL DESCRIPTION
AUXILIARY BUILDING

Control Volume	EDC Zone	Description	Net Volume (ft^3)	Limiting Break[1]	Absolute Peak Pressure (psia)	Calculated Peak Pressure Differential[2] (psid)	Design Peak Differential Pressure (psid)
1	AB-070-1	LPCS Pump Room	13,992	6-RWCU	16.07	1.37	2.40
2	AB-070-2	RHR "A" Pump Room	22,733	6-RWCU	16.07	1.37	2.40
3	AB-070-3	RCIC Pump Room	12,524	6-RWCU	16.79	2.09	2.40
4	AB-070-4	RHR "C" Pump Room	9,685	6-RWCU	16.05	1.35	2.40
5	AB-070-5	RHR "B" Pump Room	22,733	6-RWCU	16.06	1.36	2.40
6	AB-070-6	HPCS Pump Room	13,927	6-RWCU	16.05	1.35	2.40
7	AB-095-1 AB-095-9	LPCS Hatch Area & CRD Work Room	11,548	6-RWCU	16.04	1.34	2.40
8	AB-095-2	RHR "A" Equipment Room	16,402	6-RWCU	16.06	1.36	2.40
9	AB-095-3	RWCU Pump Rooms	1,567	6-RWCU	17.22	2.52	3.30
10	AB-095-8	RPCCW Equipment Area	22,845	6-RWCU	16.04	1.34	2.40
11	AB-095-4	Hoist Area	12,614	6-RWCU	16.62	1.92	2.40
12	AB-095-5	RHR "B" Equipment Room	16,402	6-RWCU	16.04	1.34	2.40
13	AB-095-6	HPCS Hatch Area	22,734	6-RWCU	16.04	1.34	2.40
14	AB-095-7	Elevator Area	21,864	6-RWCU	16.04	1.34	2.40
15	AB-114-1 AB-114-8A	MCC Area (West)	55,573	6-RWCU	16.03	1.33	2.40
17	AB-114-6	RPCCW Equipment Area	34,584	6-RWCU	16.03	1.33	2.40
18	AB-095-4	Hoist Area	2,535	6-RWCU	16.04	1.34	2.40
19	AB-114-3	MCC Area (East)	30,381	6-RWCU	16.03	1.33	2.40
20	AB-114-5	Elevator Area	31,873	6-RWCU	16.03	1.33	2.40
21	AB-141-5	SGTS "A" Room	45,330	6-RWCU	16.02	1.32	2.40
22	AB-141-2	Equipment Area (East)	70,772	6-RWCU	16.02	1.32	2.40
23	AB-141-4	RPCCW Equipment Area	40,273	6-RWCU	16.02	1.32	2.40
24	AB-141-1	Equipment Area (West)	62,074	6-RWCU	16.02	1.32	2.40
25	AB-141-3	Elevator Area	39,813	6-RWCU	16.02	1.32	2.40
26	AB-170-1	Annulus Mixing Fan Area	12,172	6-RWCU	16.01	1.31	2.40
27	AB-141-6	SGTS "B" Room	42,256	6-RWCU	16.02	1.32	2.40
28	AB-170-3	Elevator Room	1,313	6-RWCU	16.00	1.30	2.40
29	AB-170-2	Continuous Filter Room	9,962	6-RWCU	16.01	1.31	2.40
30	AB-170-1	Radiation Monitor Area	3,336	6-RWCU	16.01	1.31	2.40
31	AB-114-4	PASS Room	1,945	6-RWCU	16.04	1.34	2.40
32	AB-170-1	Annulus Mixing Fan Area	1,930	6-RWCU	16.00	1.30	2.40
33	AB-070-7	Elevator Area	35,720	6-RWCU	16.05	1.35	2.40
34	AB-070-8	RPCCW Equipment Area	35,720	6-RWCU	16.06	1.36	2.40
50	AB-114-8B	MCC Area (East)	24,613	6-RWCU	16.03	1.33	2.40

[1] All breaks are double-ended ruptures (i.e., break flow area is twice the pipe cross-sectional area).
[2] Calculated by subtracting 14.7 psia from the maximum absolute pressure for each node.

TABLE 3B-4

●→14

SUBCOMPARTMENT NODAL DESCRIPTION
MAIN STEAM TUNNEL

14←●

Node Number	Net Volume (ft^3)	Description of Volume	Break Location	Break Type	Break Line	Absolute Peak Pressure (psia)	Calculated Peak Pressure Differential[1] (psid)	Design Peak Differential Pressure (psid)
●→14								
1	26775	MSIV Area, EDC Zone AB-114-2	Node 1	2-phase[2]	24" MS SER[3]	26.15	11.45	20.00
2	118157	MS Expansion Loop Bay, EDC Zones AB-095-10 and AB-114-7	Node 2	2-phase	24" MS DER	20.18	5.48	10.00
3	10,310	MS Header Area	Node 2	2-phase	24" MS DER	19.32	4.62	NA[5]
●→11								
4	$1x10^{12}$[2]	Outside Atmosphere	-	-	-	14.70[4]	0.0	NA
11←●								
5	38,203	MS Stop Valve Area	Node 2	2-phase	24" MS DER	15.60	0.90	NA[5]
●→11 14←●								
6	$6x10^{8}$[2]	Turbine Building	-	-	-	14.70[4]	0.0	NA[5]
11←●								

(1) Calculated by subtracting 14.7 psia from the maximum absolute pressure for each node.
(2) Assumed blowdown changes from all steam to 7-percent quality at 1.05 sec.
(3) Single-ended rupture (SER) assumes split or longitudinal rupture with flow area equal to the pipe cross-sectional area.
(4) Assume large volume to maintain pressure at 14.7 psia.
(5) Turbine building - design for pressure differential is not applicable.

TABLE 3B-5

SUBCOMPARTMENT VENT PATH DESCRIPTION
AUXILIARY BUILDING

Flow Path #	From Control Volume No.	To Control Volume No.	Flow Area (ft^2)	Inertia Length (ft)	Friction Length (ft)	Head Loss Coefficient
1	34	10	271.30	22.00	2.00	3.20
2	33	14	115.00	22.00	2.00	2.70
3	5	12	882.02	22.00	1.00	0.50
4	4	18	105.00	22.00	2.00	2.70
5	3	11	114.75	21.00	2.00	3.20
6	2	8	882.02	22.00	1.00	0.50
7	11	9	9.11	19.00	10.00	10.33
8	11	9	9.11	19.00	10.00	10.33
9	11	18	1.00	24.00	2.00	1.60
10	11	18	1.00	24.00	2.00	1.60
11	11	14	6.55	46.00	4.00	1.90
12	11	14	6.55	46.00	4.00	1.90
13	10	11	5.38	46.00	4.00	1.80
14	10	11	5.38	46.00	4.00	1.80
15	14	20	115.00	23.00	2.00	2.70
16	10	17	272.43	23.00	2.00	2.70
17	20	19	32.00	23.00	3.00	2.60
18	20	19	32.00	23.00	3.00	2.60
19	17	15	32.00	24.00	3.00	2.60
20	17	15	32.00	24.00	3.00	2.60
21	20	25	115.00	28.00	2.00	2.70
22	17	23	391.00	28.00	2.00	3.30
23	25	22	32.00	60.00	3.00	2.60
24	25	22	32.00	60.00	3.00	2.60
25	23	24	32.00	60.00	3.00	2.60
26	23	24	32.00	60.00	3.00	2.60
27	22	27	10.50	46.00	12.00	4.00
28	22	27	10.50	46.00	12.00	4.00
29	24	21	10.50	46.00	12.00	4.00
30	24	21	10.50	46.00	12.00	4.00
31	23	25	78.75	60.00	1.00	0.10
32	23	25	78.75	60.00	1.00	0.10
33	25	22	413.07	25.00	3.00	2.80
34	25	22	413.07	25.00	3.00	2.80
35	23	24	413.07	25.00	3.00	2.80
36	23	24	413.07	25.00	3.00	2.80
37	26	30	73.31	29.00	2.00	2.70
38	26	30	73.31	29.00	2.00	2.70
39	26	32	96.80	15.00	2.50	0.50
40	16	1P	134.00	36.00	1.00	1.50
41	17	23	77.25	28.00	2.00	2.70
42	20	25	92.46	28.00	2.00	2.70
43	20	25	50.45	28.00	2.00	2.70
44	7	10	0.89	39.00	3.00	2.50
45	13	14	1.50	39.00	3.00	2.60
46	3	2	1.00	18.00	2.00	2.60
47	17	15	0.84	39.00	3.00	2.50
48	31	19	0.59	12.00	1.00	2.80
49	29	30	1.13	36.00	2.00	2.60
50	28	32	0.75	15.00	2.00	2.60

SUBCOMPARTMENT VENT PATH DESCRIPTION
AUXILIARY BUILDING

Flow Path #	From Control Volume No.	To Control Volume No.	Flow Area (ft^2)	Inertia Length (ft)	Friction Length (ft)	Head Loss Coefficient
51	26	22	3.75	28.00	2.00	1.70
52	27	22	1.50	36.00	2.00	2.70
53	21	24	1.50	36.00	2.00	2.70
54	9	35	9.00	20.00	2.00	1.50
55	35	9	9.00	20.00	2.00	1.50
56	35	11	9.00	20.00	2.00	1.50
57	10	36	9.00	20.00	2.00	1.50
58	36	7	9.00	20.00	2.00	1.50
59	36	10	9.00	20.00	2.00	1.50
60	36	34	9.00	20.00	2.00	1.50
61	14	37	9.00	20.00	2.00	1.50
62	37	14	9.00	20.00	2.00	1.50
63	37	13	9.00	20.00	2.00	1.50
64	37	33	9.00	20.00	2.00	1.50
65	6	38	9.00	20.00	2.00	1.50
66	38	6	9.00	20.00	2.00	1.50
67	1	39	9.00	20.00	2.00	1.50
68	8	39	9.00	20.00	2.00	1.50
69	39	40	9.00	20.00	2.00	1.50
70	40	1	9.00	20.00	2.00	1.50
71	40	2	9.00	20.00	2.00	1.50
72	40	3	9.00	20.00	2.00	1.50
73	40	8	9.00	20.00	2.00	1.50
74	15	41	9.00	20.00	2.00	1.50
75	41	15	9.00	20.00	2.00	1.50
76	41	17	9.00	20.00	2.00	1.50
77	16	42	9.00	20.00	2.00	1.50
78	42	16	9.00	20.00	2.00	1.50
79	4	43	9.00	20.00	2.00	1.50
80	12	43	9.00	20.00	2.00	1.50
81	43	44	9.00	20.00	2.00	1.50
82	44	4	9.00	20.00	2.00	1.50
83	44	5	9.00	20.00	2.00	1.50
84	44	12	9.00	20.00	2.00	1.50
85	19	45	9.00	20.00	2.00	1.50
86	45	19	9.00	20.00	2.00	1.50
87	45	20	9.00	20.00	2.00	1.50
88	45	31	9.00	20.00	2.00	1.50
89	22	47	9.00	20.00	2.00	1.50
90	24	47	9.00	20.00	2.00	1.50
91	46	24	9.00	20.00	2.00	1.50
92	46	21	9.00	20.00	2.00	1.50
93	46	23	9.00	20.00	2.00	1.50
94	46	22	9.00	20.00	2.00	1.50
95	46	25	9.00	20.00	2.00	1.50
96	46	27	9.00	20.00	2.00	1.50
97	46	28	9.00	20.00	2.00	1.50
98	46	26	9.00	20.00	2.00	1.50
99	46	29	9.00	20.00	2.00	1.50
100	46	30	9.00	20.00	2.00	1.50

TABLE 3B-5

SUBCOMPARTMENT VENT PATH DESCRIPTION
AUXILIARY BUILDING

Flow Path #	From Control Volume No.	To Control Volume No.	Flow Area (ft^2)	Inertia Length (ft)	Friction Length (ft)	Head Loss Coefficient
101	3	48	0.50	18.00	10.00	2.78
102	48	4	1.00	12.00	28.00	2.78
103	48	5	1.50	12.00	40.00	2.78
104	48	12	1.00	18.00	53.00	2.78
105	48	22	3.00	9.00	113.00	2.78
106	9	11	11.56	7.00	2.00	2.78
107	49	1	1.50	14.00	40.00	2.78
108	48	6	1.39	15.00	40.00	2.78
109	49	2	1.50	16.00	28.00	2.78
110	49	9	1.11	5.00	14.00	2.78
111	49	8	1.36	18.00	4.00	2.78
112	3	49	0.50	18.00	10.00	2.78
113	49	48	2.33	1.00	18.00	2.78
114	11	8	1.11	25.00	6.00	2.78
115	8	10	2.00	15.00	4.00	2.78
116	12	14	2.33	26.00	3.00	2.78
117	50	22	3.39	24.00	2.00	2.78
118	19	22	0.90	24.00	2.00	2.78
119	50	12	22.81	21.00	2.00	2.78
120	19	13	20.50	21.00	2.00	2.78
121	19	13	0.90	21.00	2.00	2.78
122	47	46	9.00	20.00	2.00	1.50
123	7	10	0.89	39.00	3.00	2.50
124	13	14	1.50	39.00	3.00	2.60
125	3	2	1.00	18.00	2.00	2.60
126	17	15	0.84	39.00	3.00	2.50
127	31	19	0.59	12.00	1.00	2.80
128	29	30	1.13	36.00	2.00	2.60
129	28	32	0.75	15.00	2.00	2.60
130	27	22	1.00	36.00	2.00	2.70
131	21	24	1.00	36.00	2.00	2.70
132	3	2	7.33	18.00	2.00	2.78
133	3	4	1.56	18.00	2.00	2.78
134	3	2F	0.08	9.00	1.00	0.00
135	9	3F	0.05	5.00	1.00	0.00
136	11	4F	0.18	5.00	1.00	0.00
137	50	19	24.50	20.00	2.00	2.78
138	50	19	297.00	20.00	2.00	2.78
139	50	19	297.00	20.00	2.00	2.78
140	42	20	0.01	1.00	1.00	2.78

TABLE 3B-6

SUBCOMPARTMENT VENT PATH DESCRIPTION
MAIN STEAM TUNNEL
6-NODE MODEL

Vent Path No.	From Vol. Node No.	To Vol. Node No.	Vent Area (ft^2)	Inertia Factor, L/A (ft^{-1})	Head Loss Coefficient					
					Contraction	Expansion	Obstruction[1]	Friction	Turning Loss	Total
J1	1	2	151	0.017	-	-	-	-	_	2.215[2]
J2	1	2	41	0.172	-	-	-	-	_	4.001[3]
J3	2	3	297	0.049	0.481	-	-	-	_	0.481
J4[4]	2	4	138	0.01	0.485	-	4.366	-	0.8	5.651
J5	5	6	119	0.041	0.462	1.0	1.036	0.008	-	2.506
J6	3	5	213	0.073	-	0.545	-	-	-	0.545
J7	5	6	442	0.029	-	1.0	-	-	-	1.0

----

[1] Includes grating, orifice, mesh door, and any other form losses, as applicable.
[2] Combined loss for eight openings in the jet impingement wall. Each includes contraction, expansion, thick orifice, and friction terms which are summed and combined in parallel.
[3] Combined loss for two parallel vent openings in top of jet impingement wall comprised of contraction, expansion, thick orifice, friction, and 90° bend loss components.
[4] Six louvered blowout panels modeled to open at 3.5 psid with 0.3-sec delay. The current design consists of blowout panels which are bounded by the louvered panels in the calculation.

TABLE 3B-7

MASS AND ENERGY RELEASE
3-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
0.0000	357.78	198,956
0.1000	357.78	198,956
0.2000	357.78	198,956
0.3000	357.78	198,956
0.4000	357.78	198,956
0.5000	357.78	198,956
0.6000	357.78	198,956
0.7000	357.78	198,956
0.8000	357.78	198,956
0.9000	357.78	198,956
1.0000	357.78	198,956
1.1000	357.78	198,956
1.2000	357.78	198,956
1.3000	357.78	198,956
1.4000	357.78	198,956
1.5000	357.78	198,956
1.6000	357.78	198,956
1.7000	357.78	198,956
1.8000	357.78	198,956
1.9000	357.78	198,956
2.0000	349.56	194,387
2.1000	349.56	194,387
2.2000	349.56	194,387
2.3000	349.56	194,387
2.4000	349.56	194,387
2.5000	349.56	194,387
2.6000	349.56	194,387
2.7000	349.56	194,387
2.8000	349.56	194,387
2.9000	349.56	194,387
3.0000	349.56	194,387
3.1000	349.56	194,387
3.2000	349.56	194,387
3.3000	349.56	194,387
3.4000	349.56	194,387
3.5000	349.56	194,387
3.6000	349.56	194,387
3.7000	349.56	194,387
3.8000	349.56	194,387
3.9000	288.26	160,297
4.0000	288.26	160,297
4.1000	288.26	160,297
4.2000	288.26	160,297
4.3000	288.26	160,297

TABLE 3B-7

MASS AND ENERGY RELEASE
3-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
4.4000	288.26	160,297
4.5000	288.26	160,297
4.6000	288.26	160,297
4.7000	288.26	160,297
4.8000	288.26	160,297
4.9000	288.26	160,297
5.0000	288.26	160,297
5.1000	288.26	160,297
5.2000	288.26	160,297
5.3000	288.26	160,297
5.4000	288.26	160,297
5.5000	288.26	160,297
5.6000	288.26	160,297
5.7000	288.26	160,297
5.8000	288.26	160,297
5.9000	288.26	160,297
6.0000	288.26	160,297
6.1000	288.26	160,297
6.2000	288.26	160,297
6.3000	288.26	160,297
6.4000	288.26	160,297
6.5000	288.26	160,297
6.6000	288.26	160,297
6.7000	288.26	160,297
6.8000	288.26	160,297
6.9000	288.26	160,297
7.0000	288.26	160,297
7.1000	288.26	160,297
7.2000	288.26	160,297
7.3000	288.26	160,297
7.4000	288.26	160,297
7.5000	288.26	160,297
7.6000	288.26	160,297
7.7000	288.26	160,297
7.8000	288.26	160,297
7.9000	288.26	160,297
8.0000	288.26	160,297
8.1000	288.26	160,297
8.2000	288.26	160,297
8.3000	288.26	160,297
8.4000	288.26	160,297
8.5000	288.26	160,297
8.6000	288.26	160,296
8.7000	288.26	160,295

TABLE 3B-7

MASS AND ENERGY RELEASE
3-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
8.8000	288.26	160,294
8.9000	288.26	160,294
9.0000	288.25	160,293
9.1000	288.25	160,292
9.2000	288.25	160,292
9.3000	288.25	160,291
9.4000	288.25	160,290
9.5000	288.25	160,289
9.6000	288.25	160,289
9.7000	288.24	160,288
9.8000	288.24	160,287
9.9000	288.24	160,287
10.0000	288.24	160,286
10.1000	288.24	160,285
10.2000	288.24	160,285
10.3000	288.24	160,284
10.4000	288.24	160,283
10.5000	288.23	160,282
10.6000	288.23	160,281
10.7000	288.23	160,279
10.8000	288.23	160,278
10.9000	288.22	160,276
11.0000	288.22	160,275
11.1000	288.22	160,273
11.2000	288.22	160,272
11.3000	288.21	160,270
11.4000	288.21	160,268
11.5000	288.21	160,267
11.6000	288.20	160,265
11.7000	288.20	160,264
11.8000	288.20	160,262
11.9000	288.20	160,261
12.0000	288.19	160,259
12.1000	288.19	160,258
12.2000	288.19	160,256
12.3000	288.18	160,255
12.4000	288.18	160,253
12.5000	288.18	160,251
12.6000	288.18	160,249
12.7000	288.17	160,247
12.8000	288.17	160,245
12.9000	286.33	159,223
13.0000	284.12	157,996
13.1000	280.87	156,187

TABLE 3B-7

MASS AND ENERGY RELEASE
3-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
13.2000	277.27	154,186
13.3000	273.67	152,184
13.4000	270.07	150,182
13.5000	266.47	148,180
13.6000	262.87	146,177
13.7000	259.27	144,174
13.8000	243.53	135,421
13.9000	232.94	129,533
14.0000	228.74	127,197
14.1000	224.54	124,861
14.2000	220.34	122,525
14.3000	211.86	117,809
14.4000	184.02	102,331
14.5000	164.36	91,396
14.6000	164.35	91,392
14.7000	158.42	88,094
14.8000	144.68	80,455
14.9000	130.87	72,773
15.0000	117.37	65,268
15.1000	117.36	65,264
15.2000	117.36	65,263
15.3000	117.36	65,262
15.4000	117.36	65,261
15.5000	117.36	65,260
15.6000	117.35	65,258
15.7000	117.35	65,257
15.8000	117.35	65,256
15.9000	117.35	65,255
16.0000	117.34	65,253
16.1000	117.34	65,252
16.2000	117.34	65,251
16.3000	117.34	65,250
16.4000	117.34	65,249
16.5000	117.33	65,246
16.6000	117.33	65,244
16.7000	117.32	65,241
16.8000	117.32	65,239
16.9000	117.31	65,237
17.0000	117.31	65,234
17.1000	117.31	65,232
17.2000	117.30	65,229
17.3000	117.30	65,227
17.4000	117.29	65,224
17.5000	117.29	65,222

TABLE 3B-7

MASS AND ENERGY RELEASE
3-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
17.6000	117.28	65,219
17.7000	117.28	65,217
17.8000	117.27	65,215
17.9000	117.27	65,212
18.0000	117.27	65,210
18.1000	117.26	65,207
18.2000	117.26	65,205
18.3000	117.25	65,202
18.4000	117.25	65,200
18.5000	117.24	65,196
18.6000	117.23	65,192
18.7000	117.23	65,188
18.8000	117.22	65,184
18.9000	117.21	65,180
19.0000	117.20	65,176
19.1000	117.20	65,171
19.2000	117.19	65,167
19.3000	117.18	65,163
19.4000	117.18	65,159
19.5000	117.17	65,155
19.6000	117.16	65,151
19.7000	117.15	65,147
19.8000	117.15	65,143
19.9000	117.14	65,139
20.0000	117.13	65,135
20.1000	117.12	65,131
20.2000	117.12	65,127
20.3000	117.11	65,123
20.4000	117.10	65,117
20.5000	117.08	65,108
20.6000	117.07	65,098
20.7000	117.05	65,089
20.8000	117.03	65,079
20.9000	117.01	65,070
21.0000	117.00	65,061
21.1000	116.98	65,051
21.2000	116.96	65,042
21.3000	116.95	65,032
21.4000	116.93	65,023
21.5000	116.91	65,013
21.6000	116.90	65,004
21.7000	116.88	64,994
21.8000	116.86	64,985
21.9000	116.84	64,975

TABLE 3B-7

MASS AND ENERGY RELEASE
3-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
22.0000	116.83	64,966
22.1000	116.81	64,956
22.2000	116.79	64,947
22.3000	116.78	64,937
22.4000	116.75	64,923
22.5000	116.71	64,901
22.6000	116.67	64,879
22.7000	116.63	64,858
22.8000	116.59	64,836
22.9000	116.55	64,814
23.0000	116.52	64,792
23.1000	116.48	64,771
23.2000	116.44	64,749
23.3000	116.40	64,727
23.4000	116.36	64,706
23.5000	116.32	64,684
23.6000	116.28	64,662
23.7000	116.24	64,641
23.8000	116.20	64,619
23.9000	116.16	64,597
24.0000	116.13	64,575
24.1000	116.09	64,554
24.2000	116.05	64,532
24.3000	116.01	64,510
24.4000	115.89	64,445
24.5000	115.72	64,350
24.6000	115.55	64,255
24.7000	115.38	64,160
24.8000	115.21	64,065
24.9000	115.04	63,970
25.0000	114.87	63,875
25.1000	114.69	63,780
25.2000	114.52	63,685
25.3000	114.35	63,590
25.4000	114.18	63,495
25.5000	114.01	63,400
25.6000	113.84	63,305
25.7000	113.67	63,210
25.8000	113.50	63,115
25.9000	113.33	63,020
26.0000	113.16	62,925
26.1000	112.99	62,830
26.2000	112.82	62,735
26.3000	112.65	62,640

TABLE 3B-7

MASS AND ENERGY RELEASE
3-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
26.4000	109.88	61,102
26.5000	106.46	59,203
26.6000	103.05	57,303
26.7000	99.63	55,404
26.8000	96.22	53,505
26.9000	92.80	51,606
27.0000	89.39	49,707
27.1000	85.97	47,807
27.2000	82.56	45,908
27.3000	79.46	44,185
27.4000	46.95	26,106
27.5000	46.95	26,106
27.6000	46.95	26,106
27.7000	46.95	26,106
27.8000	46.95	26,106
27.9000	40.31	22,415
28.0000	30.07	16,721
28.1000	19.79	11,007
28.2000	9.50	5,281
28.3000	0.00	0

TABLE 3B-8

MASS AND ENERGY RELEASE
6-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
0.0000	1411.94	785,159
0.1000	1411.94	785,159
0.2000	1411.94	785,159
0.3000	1411.94	785,159
0.4000	1411.94	785,159
0.5000	1411.94	785,159
0.6000	1411.94	785,159
0.7000	1411.94	785,159
0.8000	1411.94	785,159
0.9000	1411.94	785,159
1.0000	705.97	392,579
1.1000	178.37	99,186
1.2000	178.37	99,186
1.3000	178.37	99,186
1.4000	178.37	99,186
1.5000	178.37	99,186
1.6000	178.37	99,186
1.7000	178.37	99,186
1.8000	178.37	99,186
1.9000	178.37	99,186
2.0000	178.37	99,186
2.1000	178.37	99,186
2.2000	178.37	99,186
2.3000	178.37	99,186
2.4000	178.37	99,186
2.5000	178.37	99,186
2.6000	178.37	99,186
2.7000	178.37	99,186
2.8000	178.37	99,186
2.9000	178.37	99,186
3.0000	178.37	99,186
3.1000	178.37	99,186
3.2000	178.37	99,186
3.3000	178.37	99,186
3.4000	178.37	99,186
3.5000	178.37	99,186
3.6000	178.37	99,186
3.7000	178.37	99,186
3.8000	178.37	99,186
3.9000	178.37	99,186
4.0000	178.37	99,186
4.1000	178.37	99,186
4.2000	178.37	99,186
4.3000	178.37	99,186
4.4000	178.37	99,186

TABLE 3B-8

MASS AND ENERGY RELEASE
6-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
4.5000	178.37	99,186
4.6000	178.37	99,186
4.7000	178.37	99,186
4.8000	178.37	99,186
4.9000	178.37	99,186
5.0000	178.37	99,186
5.1000	178.37	99,186
5.2000	178.37	99,186
5.3000	178.37	99,186
5.4000	178.37	99,186
5.5000	178.37	99,186
5.6000	178.37	99,186
5.7000	178.37	99,186
5.8000	178.37	99,186
5.9000	178.37	99,186
6.0000	178.37	99,186
6.1000	178.37	99,186
6.2000	178.37	99,186
6.3000	178.37	99,186
6.4000	178.37	99,186
6.5000	178.37	99,186
6.6000	178.37	99,186
6.7000	178.37	99,186
6.8000	178.37	99,186
6.9000	178.37	99,186
7.0000	178.37	99,186
7.1000	178.37	99,186
7.2000	178.37	99,186
7.3000	178.37	99,186
7.4000	178.37	99,186
7.5000	178.37	99,186
7.6000	178.37	99,186
7.7000	178.37	99,186
7.8000	178.37	99,186
7.9000	178.37	99,186
8.0000	178.37	99,186
8.1000	178.37	99,186
8.2000	178.37	99,186
8.3000	178.37	99,186
8.4000	178.37	99,186
8.5000	178.37	99,186
8.6000	178.37	99,186
8.7000	178.37	99,186
8.8000	178.37	99,186
8.9000	178.37	99,186

TABLE 3B-8

MASS AND ENERGY RELEASE
6-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
9.0000	178.37	99,186
9.1000	178.37	99,186
9.2000	178.37	99,186
9.3000	178.37	99,186
9.4000	178.37	99,186
9.5000	178.37	99,186
9.6000	178.37	99,186
9.7000	178.37	99,186
9.8000	178.37	99,186
9.9000	178.37	99,186
10.0000	178.37	99,186
10.1000	178.37	99,186
10.2000	178.37	99,186
10.3000	178.37	99,186
10.4000	178.37	99,186
10.5000	178.37	99,186
10.6000	178.37	99,186
10.7000	178.37	99,186
10.8000	178.37	99,186
10.9000	178.37	99,186
11.0000	178.37	99,186
11.1000	178.37	99,186
11.2000	178.37	99,186
11.3000	178.37	99,186
11.4000	178.37	99,186
11.5000	178.37	99,186
11.6000	178.37	99,186
11.7000	178.37	99,186
11.8000	178.37	99,186
11.9000	178.37	99,186
12.0000	178.37	99,186
12.1000	178.37	99,186
12.2000	178.37	99,186
12.3000	178.37	99,186
12.4000	178.37	99,186
12.5000	178.37	99,186
12.6000	178.37	99,186
12.7000	178.37	99,186
12.8000	178.37	99,186
12.9000	178.37	99,186
13.0000	178.37	99,186
13.1000	178.37	99,186
13.2000	178.37	99,186
13.3000	178.37	99,186
13.4000	178.37	99,186

TABLE 3B-8

MASS AND ENERGY RELEASE
6-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
13.5000	178.37	99,186
13.6000	178.37	99,186
13.7000	178.37	99,186
13.8000	178.37	99,186
13.9000	178.37	99,186
14.0000	178.37	99,186
14.1000	178.37	99,186
14.2000	178.37	99,186
14.3000	178.37	99,186
14.4000	178.37	99,186
14.5000	178.37	99,186
14.6000	178.37	99,186
14.7000	178.37	99,186
14.8000	178.37	99,186
14.9000	178.37	99,186
15.0000	178.37	99,186
15.1000	178.37	99,186
15.2000	178.37	99,186
15.3000	178.37	99,186
15.4000	178.37	99,186
15.5000	178.37	99,186
15.6000	178.37	99,186
15.7000	178.37	99,186
15.8000	178.37	99,186
15.9000	178.37	99,186
16.0000	178.37	99,186
16.1000	178.37	99,186
16.2000	178.37	99,186
16.3000	178.37	99,186
16.4000	178.37	99,186
16.5000	178.37	99,186
16.6000	178.37	99,186
16.7000	178.37	99,186
16.8000	178.37	99,186
16.9000	178.37	99,186
17.0000	178.37	99,186
17.1000	178.37	99,186
17.2000	178.37	99,186
17.3000	178.37	99,186
17.4000	178.37	99,186
17.5000	178.37	99,186
17.6000	178.37	99,186
17.7000	178.37	99,186
17.8000	178.37	99,186
17.9000	178.37	99,186

TABLE 3B-8

MASS AND ENERGY RELEASE
6-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
18.0000	178.37	99,186
18.1000	178.37	99,186
18.2000	178.37	99,186
18.3000	178.37	99,186
18.4000	178.37	99,186
18.5000	178.37	99,186
18.6000	178.37	99,186
18.7000	178.37	99,186
18.8000	178.37	99,186
18.9000	178.37	99,186
19.0000	178.37	99,186
19.1000	178.37	99,186
19.2000	178.37	99,186
19.3000	178.37	99,186
19.4000	178.37	99,186
19.5000	178.37	99,186
19.6000	178.37	99,186
19.7000	178.37	99,186
19.8000	178.37	99,186
19.9000	178.37	99,186
20.0000	178.37	99,186
20.1000	178.37	99,186
20.2000	178.37	99,186
20.3000	178.37	99,186
20.4000	178.37	99,186
20.5000	178.37	99,186
20.6000	178.37	99,186
20.7000	178.37	99,186
20.8000	178.37	99,186
20.9000	178.37	99,186
21.0000	178.37	99,186
21.1000	178.37	99,186
21.2000	178.37	99,186
21.3000	178.37	99,186
21.4000	178.37	99,186
21.5000	178.37	99,186
21.6000	178.37	99,186
21.7000	178.37	99,186
21.8000	178.37	99,186
21.9000	178.37	99,186
22.0000	178.37	99,186
22.1000	178.37	99,186
22.2000	178.37	99,186
22.3000	178.37	99,186
22.4000	178.37	99,186

TABLE 3B-8

MASS AND ENERGY RELEASE
6-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
22.5000	178.37	99,186
22.6000	178.37	99,186
22.7000	178.37	99,186
22.8000	178.37	99,186
22.9000	178.37	99,186
23.0000	178.37	99,186
23.1000	178.37	99,186
23.2000	178.37	99,186
23.3000	178.37	99,186
23.4000	178.37	99,186
23.5000	178.37	99,186
23.6000	178.37	99,186
23.7000	178.37	99,186
23.8000	178.37	99,186
23.9000	178.37	99,186
24.0000	178.37	99,186
24.1000	178.37	99,186
24.2000	178.37	99,186
24.3000	178.37	99,186
24.4000	178.37	99,186
24.5000	178.37	99,186
24.6000	178.37	99,186
24.7000	178.37	99,186
24.8000	178.37	99,186
24.9000	178.37	99,186
25.0000	178.37	99,186
25.1000	178.37	99,186
25.2000	178.37	99,186
25.3000	178.37	99,186
25.4000	178.37	99,186
25.5000	178.37	99,186
25.6000	178.37	99,186
25.7000	178.37	99,186
25.8000	178.37	99,186
25.9000	178.37	99,186
26.0000	178.37	99,186
26.1000	178.37	99,186
26.2000	178.37	99,186
26.3000	178.37	99,186
26.4000	178.37	99,186
26.5000	174.46	97,014
26.6000	165.69	92,140
26.7000	156.76	87,173
26.8000	147.69	82,127
26.9000	138.50	77,015

TABLE 3B-8

MASS AND ENERGY RELEASE
6-IN RWCU DER IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
27.0000	129.15	71,819
27.1000	119.69	66,557
27.2000	110.11	61,228
27.3000	100.41	55,834
27.4000	90.60	50,381
27.5000	80.70	44,874
27.6000	70.71	39,319
27.7000	60.64	33,722
27.8000	50.50	28,084
27.9000	40.31	22,415
28.0000	30.07	16,721
28.1000	19.79	11,007
28.2000	9.50	5,281
28.3000	0.00	0

TABLE 3B-9

MASS AND ENERGY RELEASE
4-IN RCIC DER
IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
0.0000	134.83	160,373
0.1000	134.83	160,373
0.2000	134.83	160,373
0.2446	134.83	160,373
0.2500	72.86	86,666
1.0000	72.86	86,666
2.0000	72.86	86,666
3.0000	72.86	86,666
4.0000	72.86	86,666
5.0000	72.86	86,666
5.1000	72.86	86,666
5.2000	72.86	86,666
5.3000	72.86	86,666
5.4000	72.86	86,666
5.5000	72.86	86,666
5.6000	72.86	86,666
5.7000	72.86	86,666
5.8000	72.86	86,666
5.9000	72.86	86,666
6.0000	72.86	86,666
6.5000	72.86	86,666
7.0000	72.86	86,666
7.5000	72.86	86,666
8.0000	72.86	86,666
8.5000	72.86	86,666
9.0000	72.86	86,666
9.5000	72.86	86,666
10.0000	72.86	86,666
10.5000	72.86	86,666
11.0000	72.86	86,666
11.5000	72.86	86,666
12.0000	72.86	86,666
12.5000	72.85	86,651
13.0000	72.84	86,636
13.1000	72.83	86,629
13.2000	72.82	86,623
13.3000	72.82	86,616
13.4000	72.81	86,610
13.5000	72.81	86,603
13.6000	72.80	86,597
13.7000	72.80	86,591
13.8000	72.79	86,584
13.9000	72.79	86,578

TABLE 3B-9

MASS AND ENERGY RELEASE
4-IN RCIC DER
IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
14.0000	72.78	86,571
14.1000	72.77	86,562
14.2000	72.77	86,554
14.3000	72.76	86,545
14.4000	72.75	86,537
14.5000	72.74	86,527
14.6000	72.73	86,514
14.7000	72.72	86,501
14.8000	72.71	86,488
14.9000	72.70	86,475
15.0000	72.69	86,461
15.1000	72.67	86,441
15.2000	72.66	86,422
15.3000	72.64	86,403
15.4000	72.62	86,383
15.5000	72.61	86,364
15.6000	72.59	86,345
15.7000	72.57	86,326
15.8000	72.56	86,306
15.9000	72.54	86,287
16.0000	72.52	86,260
16.1000	72.49	86,221
16.2000	72.45	86,183
16.3000	72.42	86,144
16.4000	72.39	86,106
16.5000	72.36	86,067
16.6000	72.32	86,028
16.7000	72.29	85,990
16.8000	72.26	85,951
16.9000	72.23	85,913
17.0000	72.18	85,861
17.1000	72.13	85,797
17.2000	72.08	85,732
17.3000	72.02	85,668
17.4000	71.97	85,604
17.5000	71.91	85,539
17.6000	71.86	85,475
17.7000	71.81	85,411
17.8000	71.75	85,346
17.9000	71.70	85,282
18.0000	71.60	85,166
18.1000	71.47	85,016
18.2000	71.35	84,866

TABLE 3B-9

MASS AND ENERGY RELEASE
4-IN RCIC DER
IN AUXILIARY BUILDING

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
18.3000	71.22	84,716
18.4000	71.10	84,566
18.5000	70.97	84,416
18.6000	70.84	84,266
18.7000	70.72	84,116
18.8000	70.59	83,965
18.9000	70.46	83,815
19.0000	70.22	83,530
19.1000	69.94	83,187
19.2000	69.65	82,844
19.3000	69.36	82,501
19.4000	69.07	82,158
19.5000	68.78	81,815
19.6000	68.49	81,472
19.7000	68.21	81,128
19.8000	67.92	80,785
19.9000	67.63	80,442
20.0000	66.56	79,173
20.1000	65.30	77,672
20.2000	64.04	76,171
20.3000	62.78	74,670
20.4000	61.51	73,169
20.5000	60.25	71,668
20.6000	58.99	70,167
20.7000	57.73	68,666
20.8000	56.47	67,164
20.9000	55.20	65,663
21.0000	44.12	52,481
21.1000	37.61	44,733
21.2000	33.83	40,242
21.3000	31.73	37,740
21.4000	28.81	34,273
21.5000	3.14	3,737
21.6000	2.35	2,794
21.7000	1.56	1,850
21.8000	0.76	907
21.9000	0.00	0

TABLE 3B-10

MASS AND ENERGY RELEASE
8-IN RHR DER
IN AUXILIARY BUILDING – NODE 12

DELETED

TABLE 3B-11

MASS AND ENERGY RELEASE
24-IN MAIN STEAM LINE DER
IN STEAM TUNNEL - NODE 2

●→14

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.0	0.0	0.0
0.001	7,866	9,390,000
0.122	7,886	9,390,000
0.123	5,336	6,370,000
0.219	5,336	6,370,000
0.220	5.611	6,700,000
1.050	5.611	6,700,000
1.051	18,446	10,750,000
7.830	18,446	10,750,000
10.500	0.0	0.0

14←●

TABLE 3B-12

MASS AND ENERGY RELEASE
24-IN MAIN STEAM LINE SER
IN STEAM TUNNEL - NODE 1

•→14

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.0	0.0	0.0
0.001	3,933	4,700,000
0.116	3,933	4,700,000
0.117	3,369	4,020,000
0.219	3,369	4,020,000
0.220	4,025	4,810,000
1.050	4,025	4,810,000
1.051	13,232	7,720,000
7.830	13,232	7,720,000
8.840	11,488	6,700,000
10.500	0.0	0.0

14←•

TABLE 3B-13

MASS AND ENERGY RELEASE
8-IN RCIC STEAM LINE DER
IN STEAM TUNNEL - NODE 2

• →14

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.0000E+00	1.0640E+03	6.3414E+05
1.4000E-02	1.0640E+03	6.3414E+05
1.4000E-02	7.1100E+02	6.3379E+05
1.9500E-01	7.1100E+02	6.3379E+05
1.9500E-01	8.8800E+02	8.4460E+05
3.3800E-01	8.8800E+02	8.4460E+05
3.3800E-01	7.0900E+02	8.4442E+05
1.2000E+01	7.0900E+02	8.4442E+05
2.3000E+01	0.0000E+00	0.0000E+00

14←•

TABLE 3B-14

MASS AND ENERGY RELEASE
8-IN RCIC STEAM LINE SER
IN STEAM TUNNEL - NODE 1

• →14

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.0000E+00	5.3200E+02	6.3308E+05
3.6000E-02	5.3200E+02	6.3308E+05
3.6000E-02	4.4500E+02	5.2955E+05
1.8900E-01	4.4500E+02	5.2955E+05
1.8900E-01	5.3400E+02	6.3546E+05
3.4500E-01	5.3400E+02	6.3546E+05
3.4500E-01	3.5500E+02	4.2245E+05
1.7500E+01	3.5500E+02	4.2245E+05
2.3000E+01	0.0000E+00	0.0000E+00

14←•

TABLE 3B-15

HEAT SINK SLAB DESCRIPTION
AUXILIARY BUILDING

Thermal Conductor #	Control Volume[1] Side A	Side B	Surface Area (ft^2)	Material[2]	Thickness (ft)
1	1	1	686.40	Concrete	2.00
2	1	34	352.70	Concrete	3.00
3	1	2	787.80	Concrete	2.50
4	1	1	575.90	Concrete	4.00
5	1	1	324.70	Concrete	3.00
6	1	1	620.00	Concrete	10.00
7	1	7	602.00	Concrete	2.00
8	2	34	684.70	Concrete	3.00
9	2	2	684.70	Concrete	4.00
10	2	2	956.30	Concrete	10.00
11	3	34	395.90	Concrete	3.00
12	3	4	787.80	Concrete	2.00
13	3	3	395.90	Concrete	4.00
14	3	2	787.90	Concrete	4.00
15	3	3	553.00	Concrete	10.00
16	3	11	233.30	Concrete	2.00
17	4	33	308.80	Concrete	3.00
18	4	5	787.80	Concrete	4.00
19	4	4	308.80	Concrete	4.00
20	4	4	431.00	Concrete	10.00
21	4	11	253.00	Concrete	2.00
22	5	33	684.70	Concrete	3.00
23	5	5	684.70	Concrete	4.00
24	5	5	956.30	Concrete	10.00
25	6	6	686.40	Concrete	2.00
26	6	33	352.70	Concrete	3.00
27	6	6	324.70	Concrete	3.00
28	6	6	575.90	Concrete	4.00
29	6	5	787.80	Concrete	2.50
30	6	6	620.00	Concrete	10.00
31	6	13	602.00	Concrete	2.00
32	33	33	682.80	Concrete	1.00
33	33	33	1645.90	Concrete	2.50
34	33	33	418.00	Concrete	6.00
35	33	33	732.20	Concrete	3.00
36	33	33	223.30	Concrete	4.00
37	33	33	1602.50	Concrete	10.00
38	33	14	1351.30	Concrete	2.00
39	34	34	1645.90	Concrete	2.50
40	34	34	418.00	Concrete	6.00
41	34	34	223.30	Concrete	4.00
42	34	34	940.00	Concrete	3.00
43	34	34	1602.50	Concrete	10.00
44	34	10	1036.50	Concrete	2.00
45	34	11	75.00	Concrete	2.00
46	7	7	474.30	Concrete	2.00
47	7	10	243.80	Concrete	3.00
48	7	8	539.00	Concrete	2.50

TABLE 3B-15

HEAT SINK SLAB DESCRIPTION
AUXILIARY BUILDING

Thermal Conductor #	Control Volume[1] Side A	Side B	Surface Area (ft^2)	Material[2]	Thickness (ft)
49	7	7	398.10	Concrete	1.00
50	7	7	243.80	Concrete	2.00
51	7	15	629.10	Concrete	2.00
52	8	10	468.50	Concrete	3.00
53	8	8	468.50	Concrete	4.00
54	8	15	956.30	Concrete	2.00
55	9	11	141.70	Concrete	2.00
56	9	11	136.00	Concrete	2.00
57	9	11	141.70	Concrete	2.00
58	9	8	85.00	Concrete	4.00
59	9	10	51.00	Concrete	4.00
60	9	3	166.70	Concrete	2.00
61	9	34	100.00	Concrete	2.00
62	9	11	266.70	Concrete	1.50
63	9	9	324.40	Concrete	1.00
64	10	10	286.00	Concrete	3.00
65	10	10	813.00	Concrete	2.50
66	10	11	67.90	Concrete	4.00
67	10	10	160.20	Concrete	3.00
68	10	10	643.20	Concrete	2.00
69	10	17	1148.00	Concrete	2.00
70	11	11	441.30	Concrete	2.50
71	11	14	153.20	Concrete	4.00
72	11	12	269.60	Concrete	4.00
73	11	18	383.60	Concrete	2.00
74	11	11	266.00	Concrete	4.00
75	11	8	360.00	Concrete	4.00
76	11	33	136.20	Concrete	2.00
77	11	16	1202.20	Concrete	4.00
78	12	14	468.50	Concrete	3.00
79	12	12	468.50	Concrete	4.00
80	12	18	231.60	Concrete	4.00
81	12	50	956.30	Concrete	2.00
82	13	14	243.80	Concrete	3.00
83	13	13	474.30	Concrete	2.00
84	13	13	785.40	Concrete	2.00
85	13	13	398.10	Concrete	4.00
86	13	13	552.50	Concrete	4.00
87	13	12	539.00	Concrete	2.50
88	13	13	833.00	Concrete	2.00
89	13	19	1312.50	Concrete	2.00
90	13	31	161.50	Concrete	2.00
91	14	14	813.00	Concrete	2.50
92	14	14	286.00	Concrete	3.00
93	14	14	467.20	Concrete	1.00
94	14	14	501.00	Concrete	2.00
95	14	14	160.20	Concrete	2.00
96	14	20	1312.00	Concrete	2.00

TABLE 3B-15

HEAT SINK SLAB DESCRIPTION
AUXILIARY BUILDING

Thermal Conductor #	Control Volume[1] Side A	Side B	Surface Area (ft^2)	Material[2]	Thickness (ft)
97	18	18	205.90	Concrete	4.00
98	18	16	180.50	Concrete	4.00
99	15	17	1095.80	Concrete	3.00
100	15	16	829.30	Concrete	4.00
101	15	15	1570.80	Concrete	4.00
102	15	15	587.50	Concrete	2.00
103	15	15	1196.00	Concrete	2.00
104	15	15	720.80	Concrete	2.00
105	15	15	799.00	Concrete	2.00
106	15	24	1759.30	Concrete	2.00
107	15	21	625.00	Concrete	2.00
108	16	16	853.50	Concrete	2.50
109	16	20	290.30	Concrete	4.00
110	16	50	835.80	Concrete	4.00
111	16	15	1.00	Concrete	4.00
112	16	17	290.30	Concrete	4.00
113	16	22	519.80	Concrete	4.00
114	16	23	168.50	Concrete	4.00
115	16	24	519.80	Concrete	4.00
116	16	25	168.50	Concrete	4.00
117	17	17	440.00	Concrete	3.00
118	17	17	1250.80	Concrete	2.50
119	17	17	237.50	Concrete	3.00
120	17	17	989.50	Concrete	2.00
121	17	23	1097.00	Concrete	2.00
122	19	20	321.00	Concrete	3.00
123	19	19	583.00	Concrete	2.00
124	19	31	355.30	Concrete	1.00
125	19	19	928.50	Concrete	2.00
126	19	19	587.50	Concrete	2.00
127	19	19	800.00	Concrete	4.00
128	19	22	920.00	Concrete	2.00
129	19	21	315.00	Concrete	2.00
130	31	31	137.80	Concrete	2.00
131	31	31	246.50	Concrete	2.00
132	31	22	161.50	Concrete	2.00
133	20	20	1250.80	Concrete	2.50
134	20	20	440.00	Concrete	3.00
135	20	20	718.80	Concrete	1.00
136	20	20	770.80	Concrete	2.00
137	20	20	237.50	Concrete	3.00
138	20	25	1234.00	Concrete	2.00
139	21	24	2095.60	Concrete	2.00
140	21	27	397.50	Concrete	2.00
141	21	21	477.00	Concrete	2.00
142	21	21	1777.60	Concrete	2.00
143	21	21	583.00	Concrete	2.00
144	21	21	653.00	Concrete	2.00

TABLE 3B-15

HEAT SINK SLAB DESCRIPTION
AUXILIARY BUILDING

Thermal Conductor #	Control Volume[(1)] Side A	Side B	Surface Area (ft^2)	Material[(2)]	Thickness (ft)
145	21	21	1993.60	Concrete	2.00
146	27	24	254.10	Concrete	2.00
147	27	22	2380.50	Concrete	2.00
148	27	27	1760.10	Concrete	2.00
149	27	27	874.50	Concrete	2.00
150	27	27	1732.00	Concrete	2.00
151	27	27	1700.50	Concrete	2.00
152	27	29	346.50	Concrete	2.00
153	22	25	1043.80	Concrete	3.00
154	22	22	761.30	Concrete	2.00
155	22	22	1320.50	Concrete	2.00
156	22	22	371.00	Concrete	2.00
157	22	29	1181.70	Concrete	2.00
158	22	30	311.20	Concrete	2.00
159	22	22	1672.10	Concrete	2.00
160	23	23	457.60	Concrete	3.00
161	23	23	1037.40	Concrete	2.50
162	23	24	1024.10	Concrete	3.00
163	23	23	244.40	Concrete	3.00
164	23	23	1029.10	Concrete	2.00
165	23	23	1132.20	Concrete	2.00
166	24	24	684.80	Concrete	2.00
167	24	24	761.30	Concrete	2.00
168	24	24	2594.00	Concrete	2.00
169	25	25	1097.30	Concrete	2.50
170	25	25	484.00	Concrete	3.00
171	25	25	790.60	Concrete	1.00
172	25	25	847.80	Concrete	2.00
173	25	25	258.50	Concrete	3.00
174	25	26	1312.00	Concrete	2.00
175	25	25	168.50	Concrete	2.00
176	26	26	188.10	Concrete	2.50
177	26	26	162.20	Concrete	2.00
178	26	26	215.60	Concrete	1.00
179	26	26	203.20	Concrete	2.00
180	26	26	108.10	Concrete	2.00
181	26	30	162.20	Concrete	2.00
182	26	26	44.94	Concrete	2.00
183	26	28	96.80	Concrete	2.00
184	26	26	1093.40	Concrete	2.00
185	32	32	122.50	Concrete	1.00
186	32	32	526.80	Concrete	2.00
187	30	30	241.50	Concrete	2.00
188	30	29	112.10	Concrete	2.00
189	30	30	57.50	Concrete	2.00
190	30	30	241.50	Concrete	2.00
191	30	30	305.70	Concrete	2.00
192	29	29	106.40	Concrete	2.00

TABLE 3B-15

HEAT SINK SLAB DESCRIPTION
AUXILIARY BUILDING

Thermal Conductor #	Control Volume[1] Side A	Control Volume[1] Side B	Surface Area (ft^2)	Material[2]	Thickness (ft)
193	29	29	575.00	Concrete	2.00
194	29	29	278.90	Concrete	2.00
195	29	29	299.00	Concrete	2.00
196	29	29	276.00	Concrete	2.00
197	29	29	1070.50	Concrete	2.00
198	28	28	596.80	Concrete	2.00
199	28	32	147.00	Concrete	1.00
200	28	26	162.80	Concrete	2.00
201	1	1	1114.00	Steel	0.03
202	2	2	2455.00	Steel	0.03
203	3	3	935.00	Steel	0.03
204	4	4	743.00	Steel	0.03
205	5	5	2492.00	Steel	0.03
206	6	6	1253.00	Steel	0.03
207	33	33	3672.00	Steel	0.05
208	34	34	3672.00	Steel	0.05
209	7	7	481.00	Steel	0.04
210	8	8	1045.00	Steel	0.04
211	10	10	1873.00	Steel	0.05
212	11	11	897.00	Steel	0.04
213	12	12	1047.00	Steel	0.04
214	13	13	1127.00	Steel	0.04
215	14	14	1214.00	Steel	0.06
216	18	18	75.00	Steel	0.04
217	15	15	2178.00	Steel	0.05
218	16	16	1743.00	Steel	0.09
219	17	17	1795.00	Steel	0.05
220	19	19	1086.00	Steel	0.05
221	20	20	1396.00	Steel	0.06
222	21	21	1366.00	Steel	0.06
223	27	27	1331.00	Steel	0.06
224	22	22	2757.00	Steel	0.06
225	23	23	1909.00	Steel	0.05
226	24	24	2717.00	Steel	0.06
227	25	25	1738.00	Steel	0.05
228	26	26	1022.00	Steel	0.05
229	28	28	53.00	Steel	0.03
230	50	50	1085.00	Steel	0.05
231	50	50	750.00	Concrete	4.00
232	50	22	984.00	Concrete	2.00
233	50	20	775.00	Concrete	3.00

(1) Control Volume numbers are defined in Table 3B-3.
(2) Thermal Properties:

	Concrete	Carbon Steel
Conductivity, Btu/hr-°F-ft^3	0.92	27.00
Volumetric heat capacity, Btu/°F-ft^3	22.62	58.8

TABLE 3B-16

HEAT SINK SLAB DECRIPTION
MAIN STEAM TUNNEL
6-NODE MODEL

Slab No.	Node Exposure[1] Left	Right	Exposed Surface Area (ft^2)	Material	Thickness (ft)
1	1	0[2]	5,253	Concrete[3]	1.0
2	2	0	15,292	Concrete[3]	1.0
3	2	0	2,529	Concrete[3]	1.0
4	3	0	2,964	Concrete[3]	1.0
5	5	0	6,390	Concrete[3]	1.0
6	2	2	7,249	Carbon Steel[3]	0.044
7	2	2	617	Carbon Steel[3]	0.144
8	1	1	1,183	Carbon Steel[3]	0.053
9	2	1	488	Carbon Steel[3]	0.166

---

[1] Node numbers are defined in Table 3B-4 and on Fig. 3B-22.
[2] Zero exposure indicates an insulated boundary assumption with zero heat transfer at this boundary.
[3] Thermal Properties:

	Concrete	Carbon Steel
Conductivity, Btu/hr-°F-ft^3	0.8	26.0
Volumetric heat capacity, Btu/°F-f^3	23.2	53.9

TABLE 3B-17

PEAK PRESSURE
MAIN STEAM TUNNEL
8” RCIC STEAM LINE BREAK

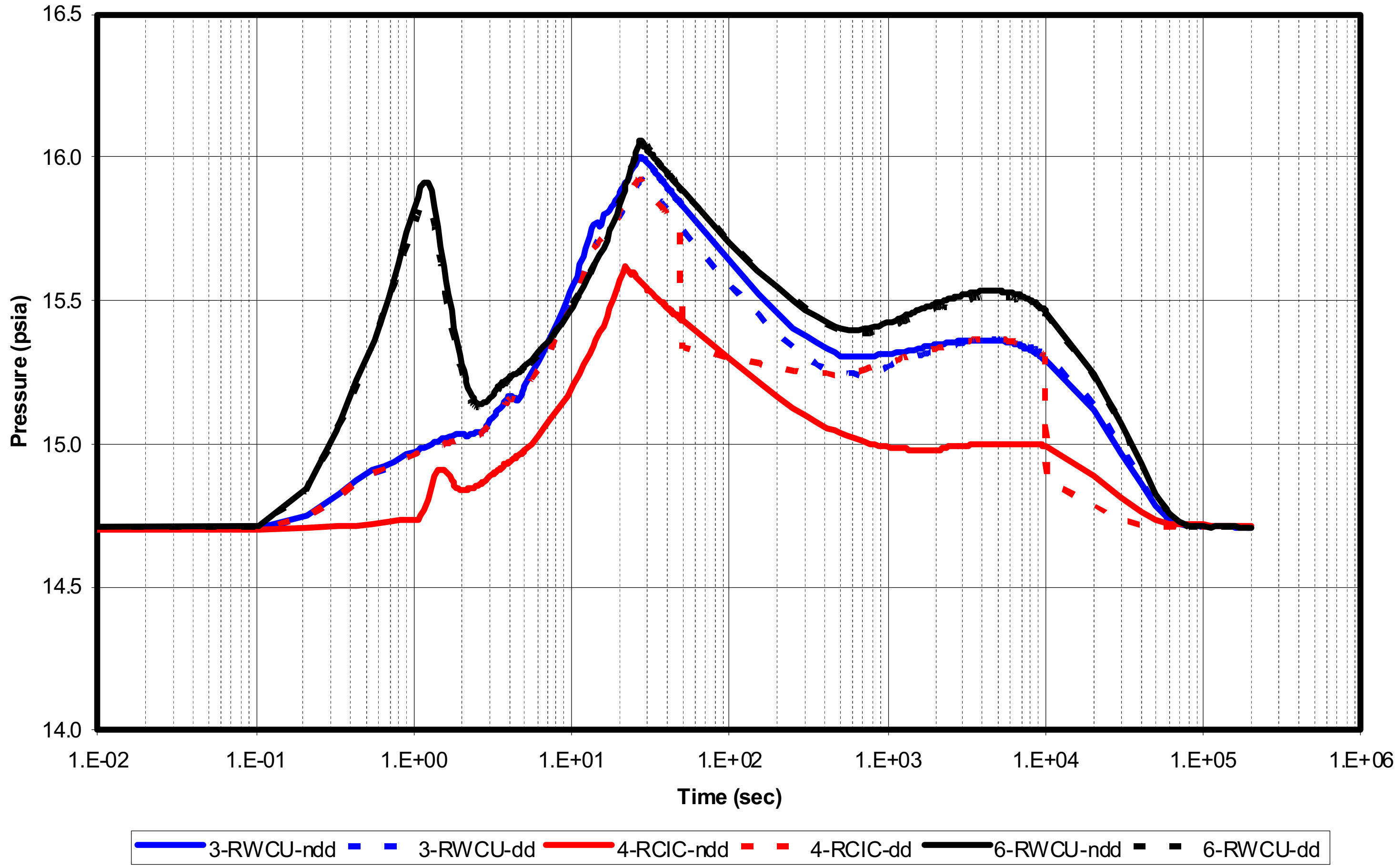
	NODE 1		NODE 2	
	PRESSURE (PSIA)	TIME AT PEAK (SEC)	PRESSURE (PSIA)	TIME AT PEAK (SEC)
8” RCIC SER IN NODE 1	15.07	0.380	14.97	0.392
8” RCIC DER IN NODE 2	15.26	1.280	15.26	1.285

DELETED

FIGURE 3B-1

NODALIZATION DIAGRAM
AUXILIARY BUILDING
20 NODE MODEL

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT



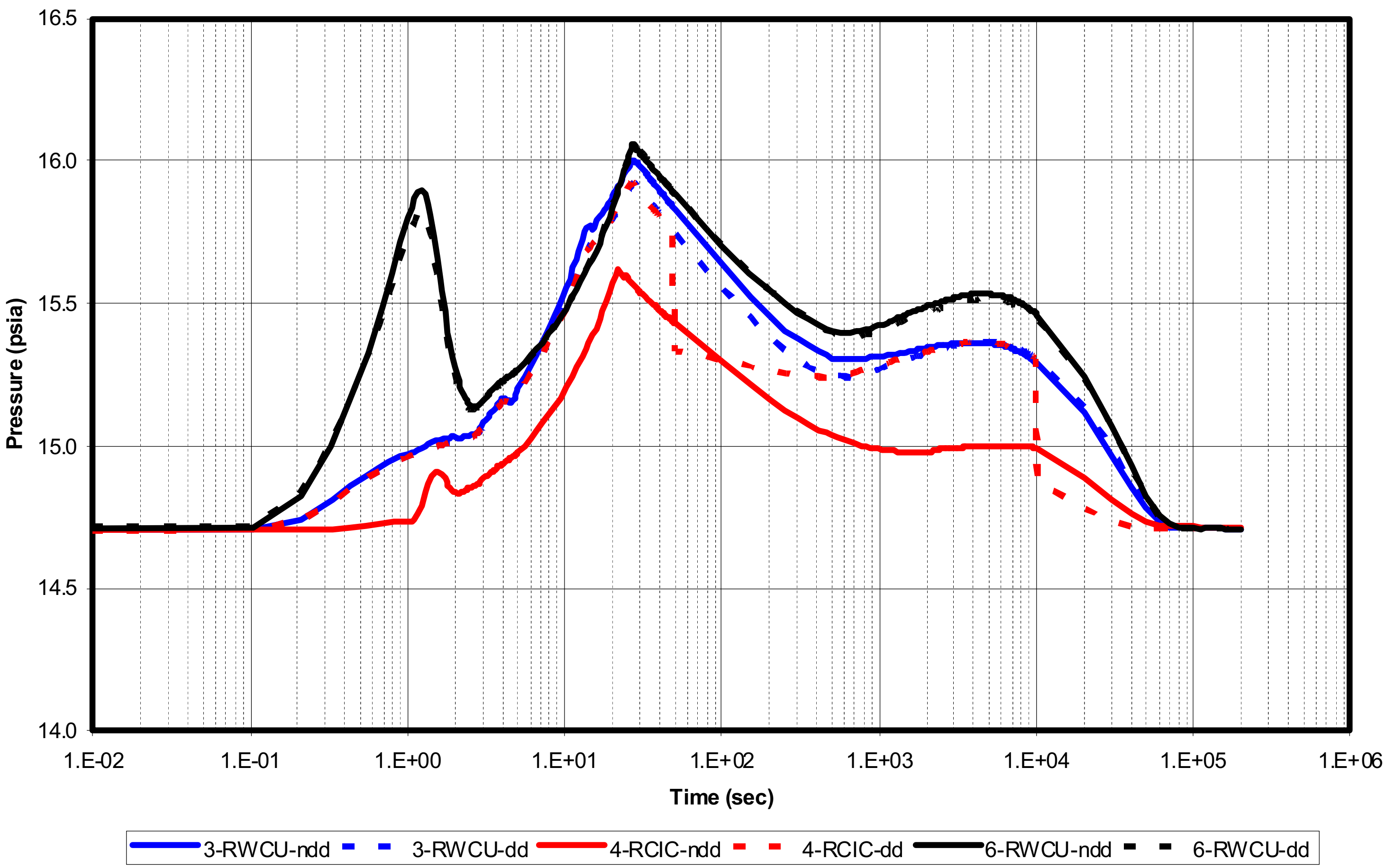
EDC ZONE AB-070-1

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-2

PRESSURE TRANSIENTS IN CV 1
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



EDC ZONE AB-070-2

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-3

PRESSURE TRANSIENTS IN CV 2
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
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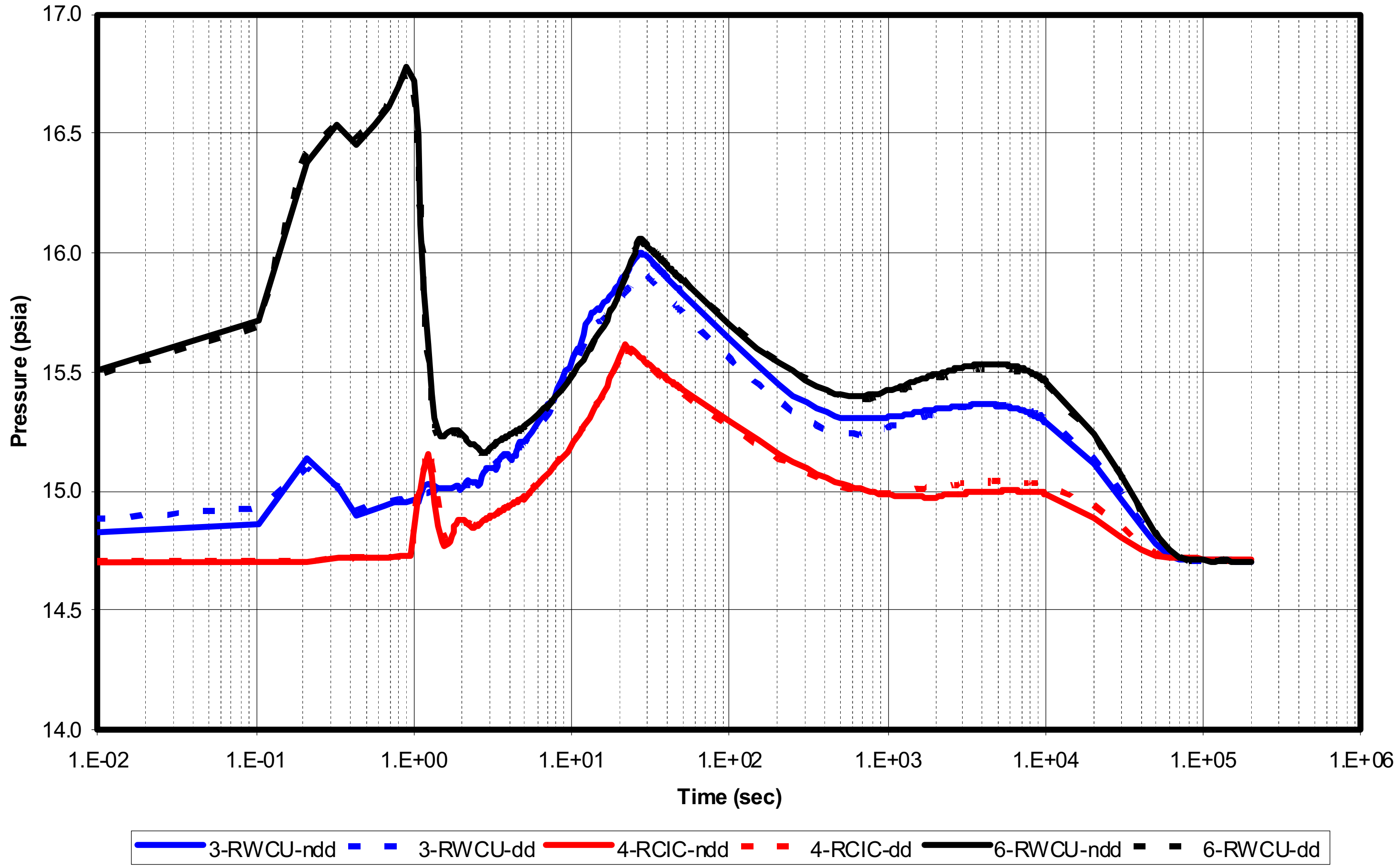


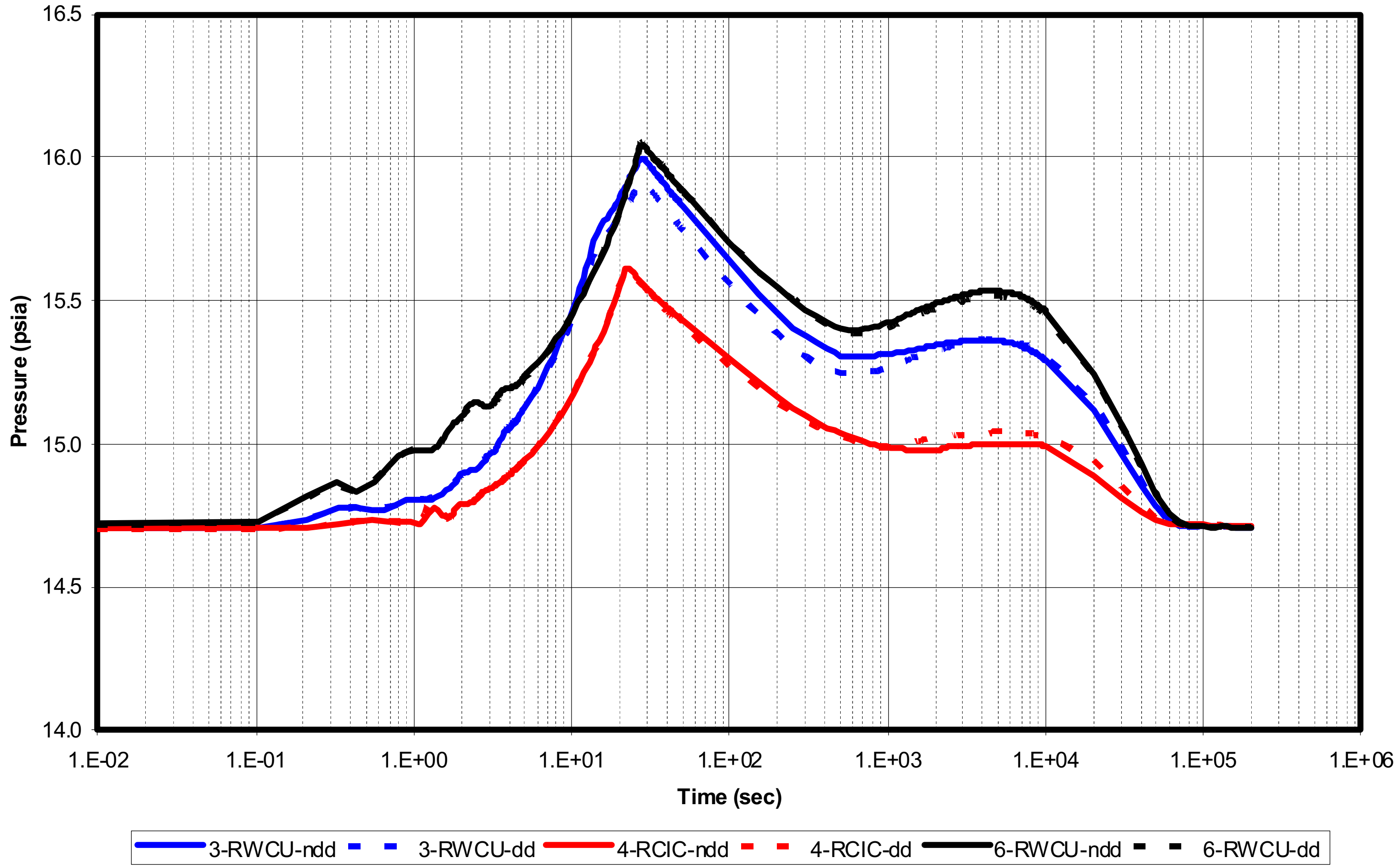
EDC ZONE AB-070-3

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-4

PRESSURE TRANSIENTS IN CV 3
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



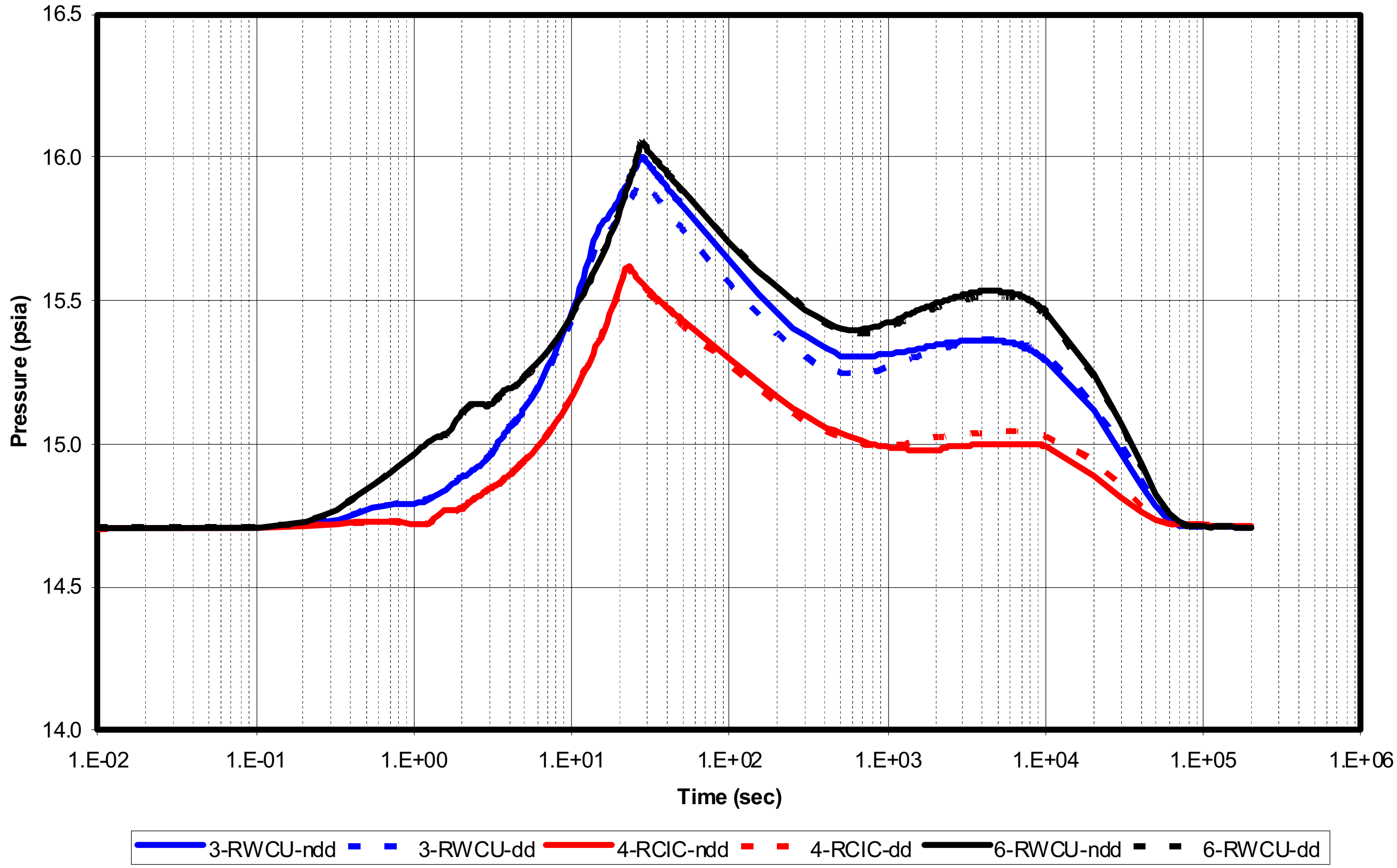
EDC ZONE AB-070-4

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-5

PRESSURE TRANSIENTS IN CV 4
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



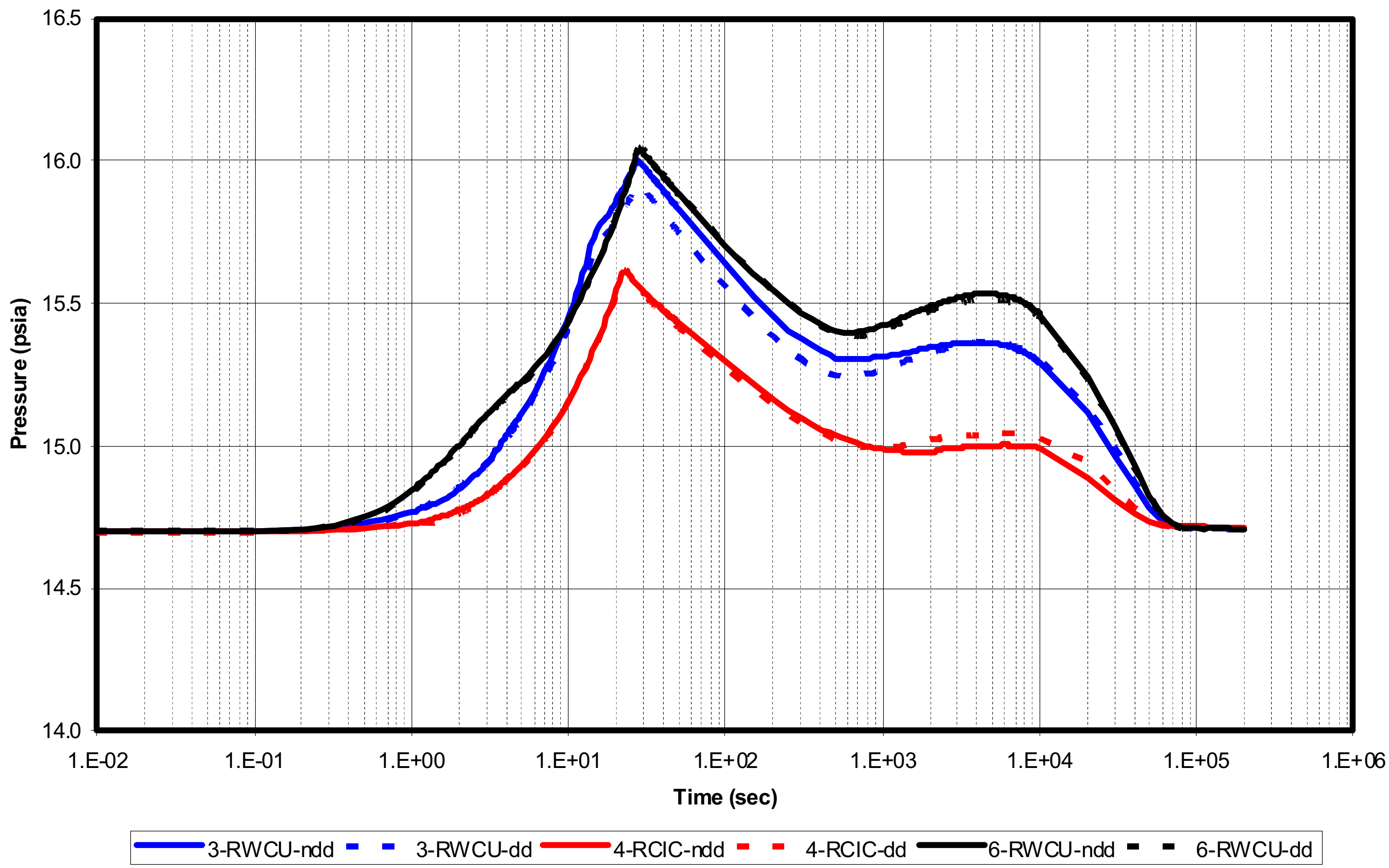
EDC ZONE AB-070-5

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-6

PRESSURE TRANSIENTS IN CV 5
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



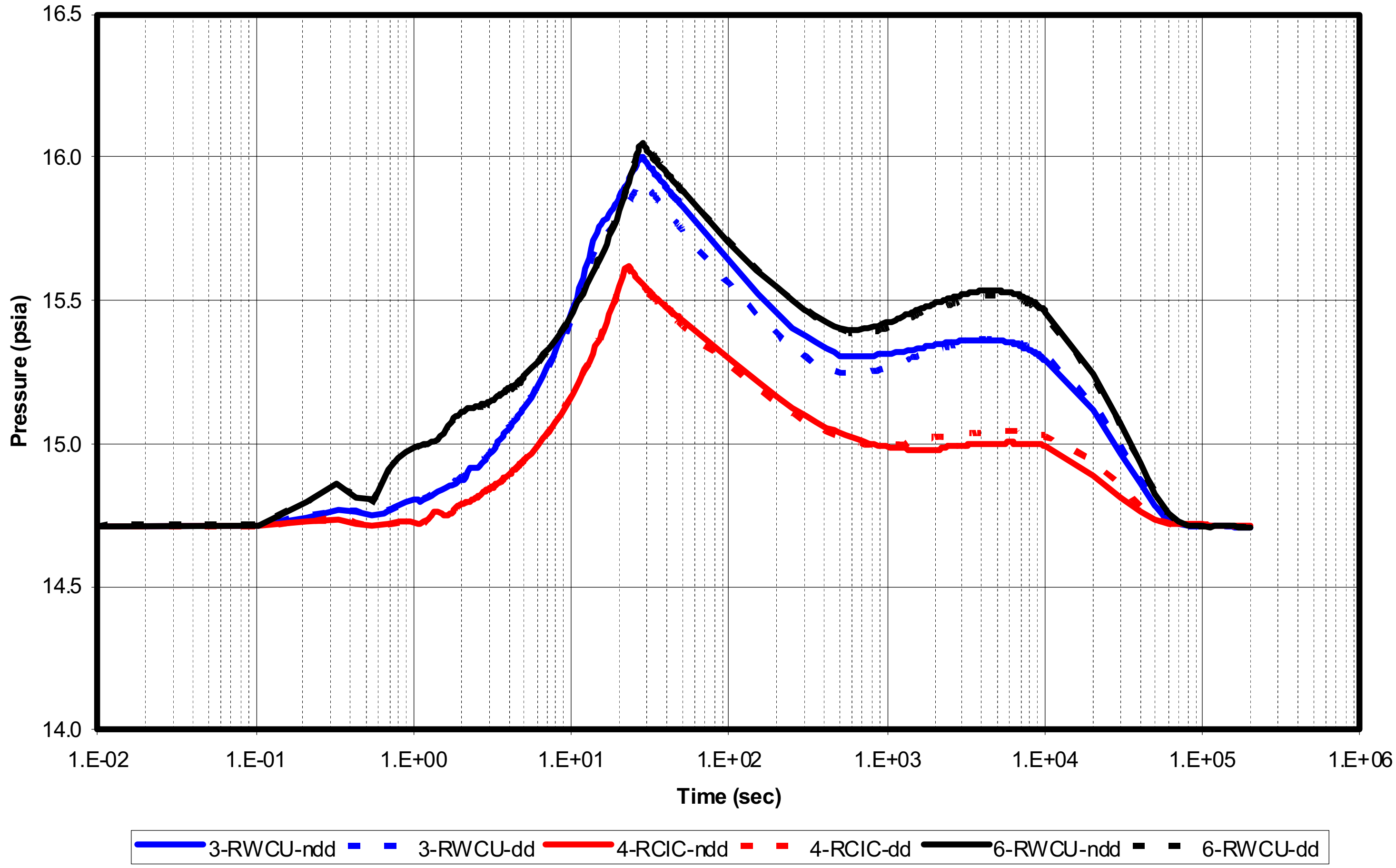
EDC ZONE AB-070-6

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-7

PRESSURE TRANSIENTS IN CV 6
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



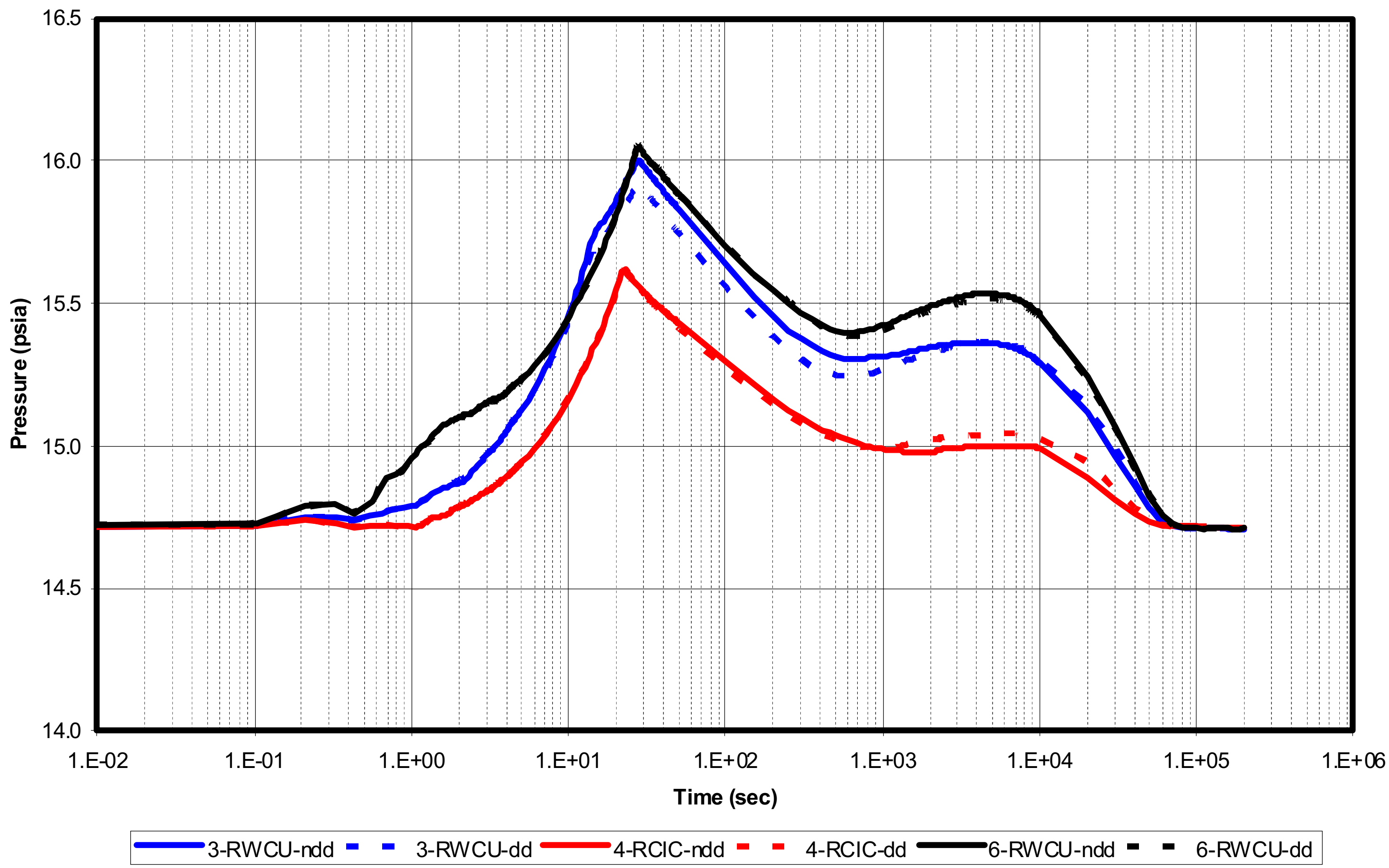
EDC ZONE AB-070-7

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-8

PRESSURE TRANSIENTS IN CV 33
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



EDC ZONE AB-070-8

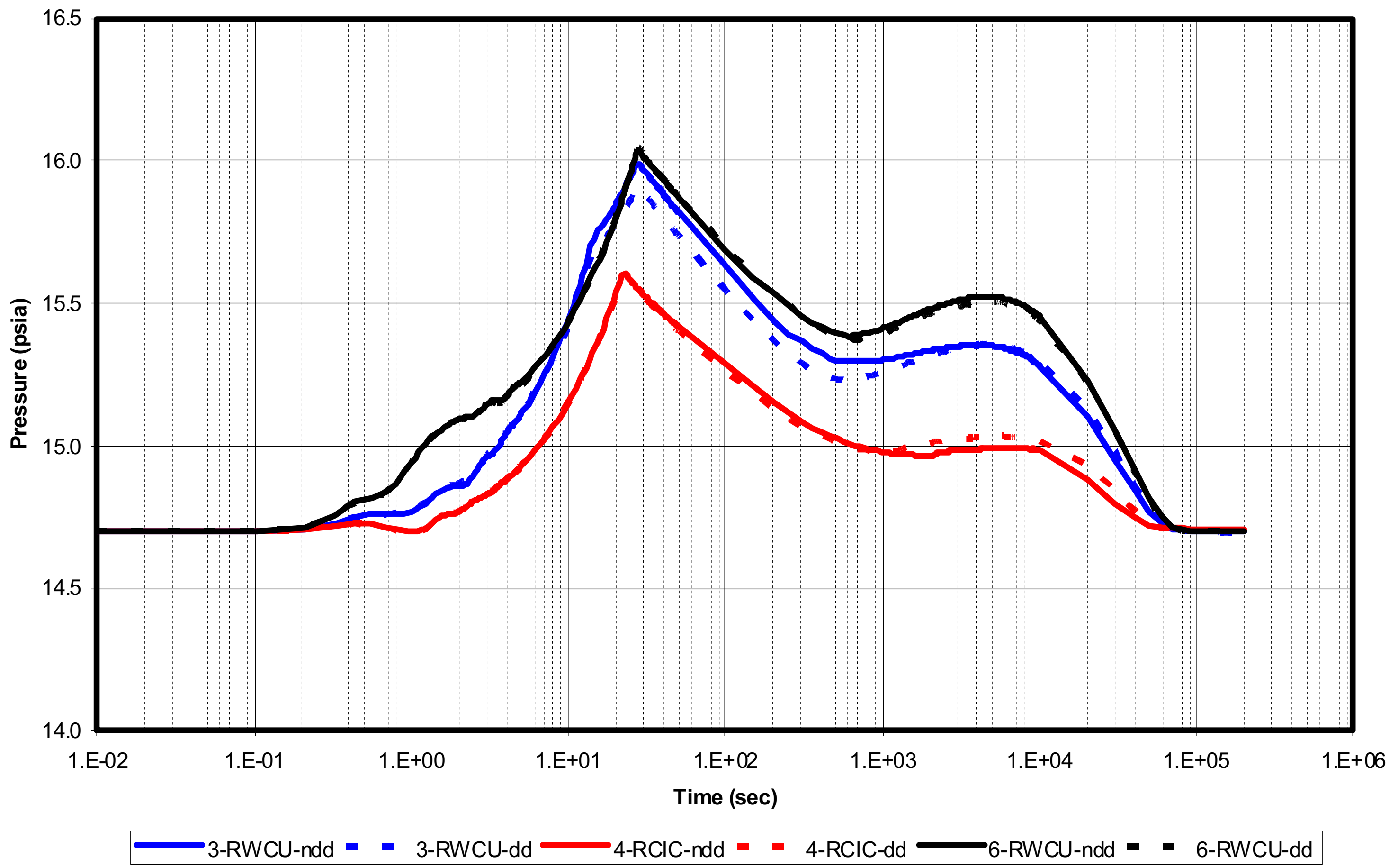
dd = Duct Destruction

ndd = No Duct Destruction

FIGURE 3B-9

PRESSURE TRANSIENTS IN NODE 34
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



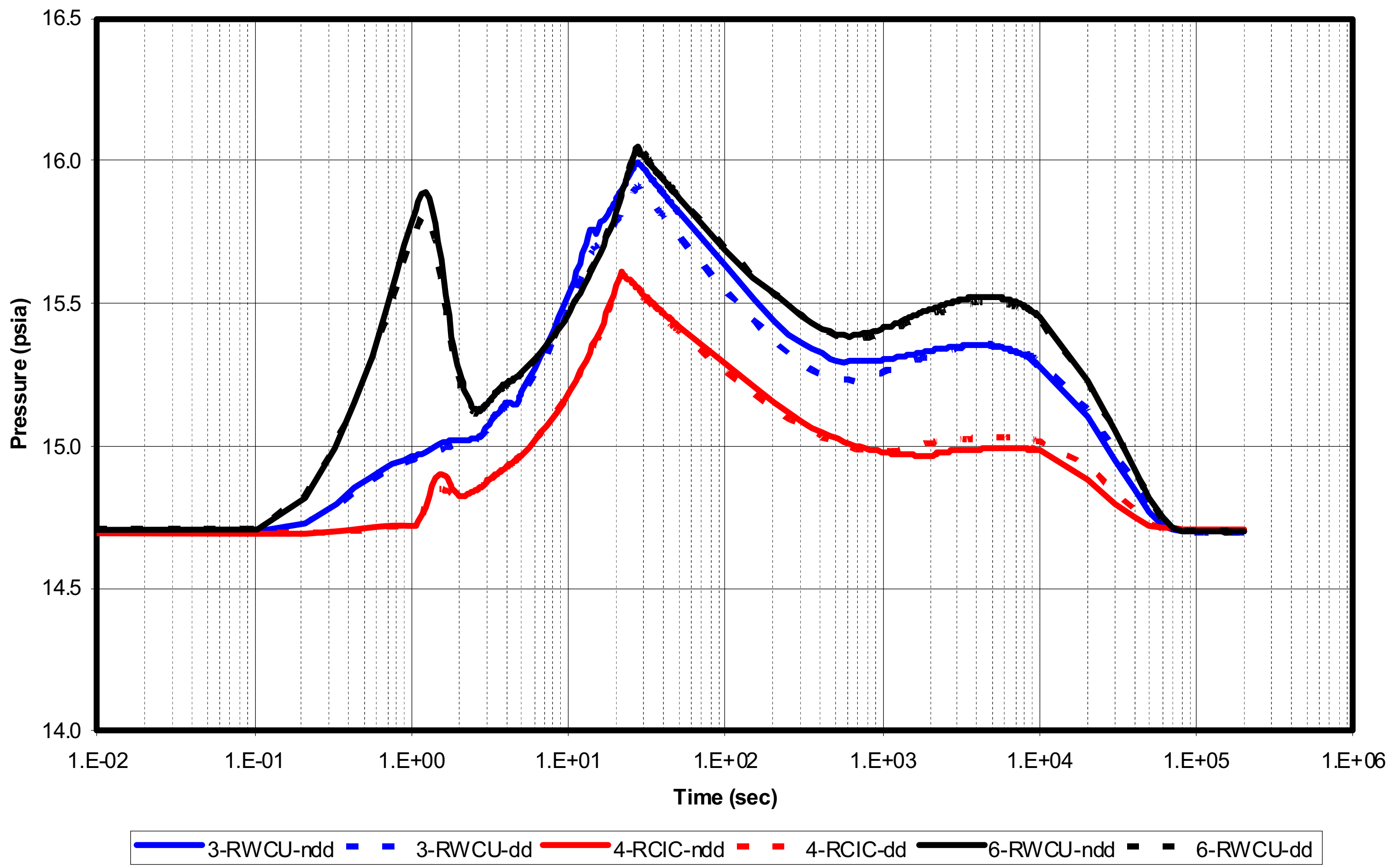
EDC ZONE AB-095-1

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-10

PRESSURE TRANSIENTS IN CV 7
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



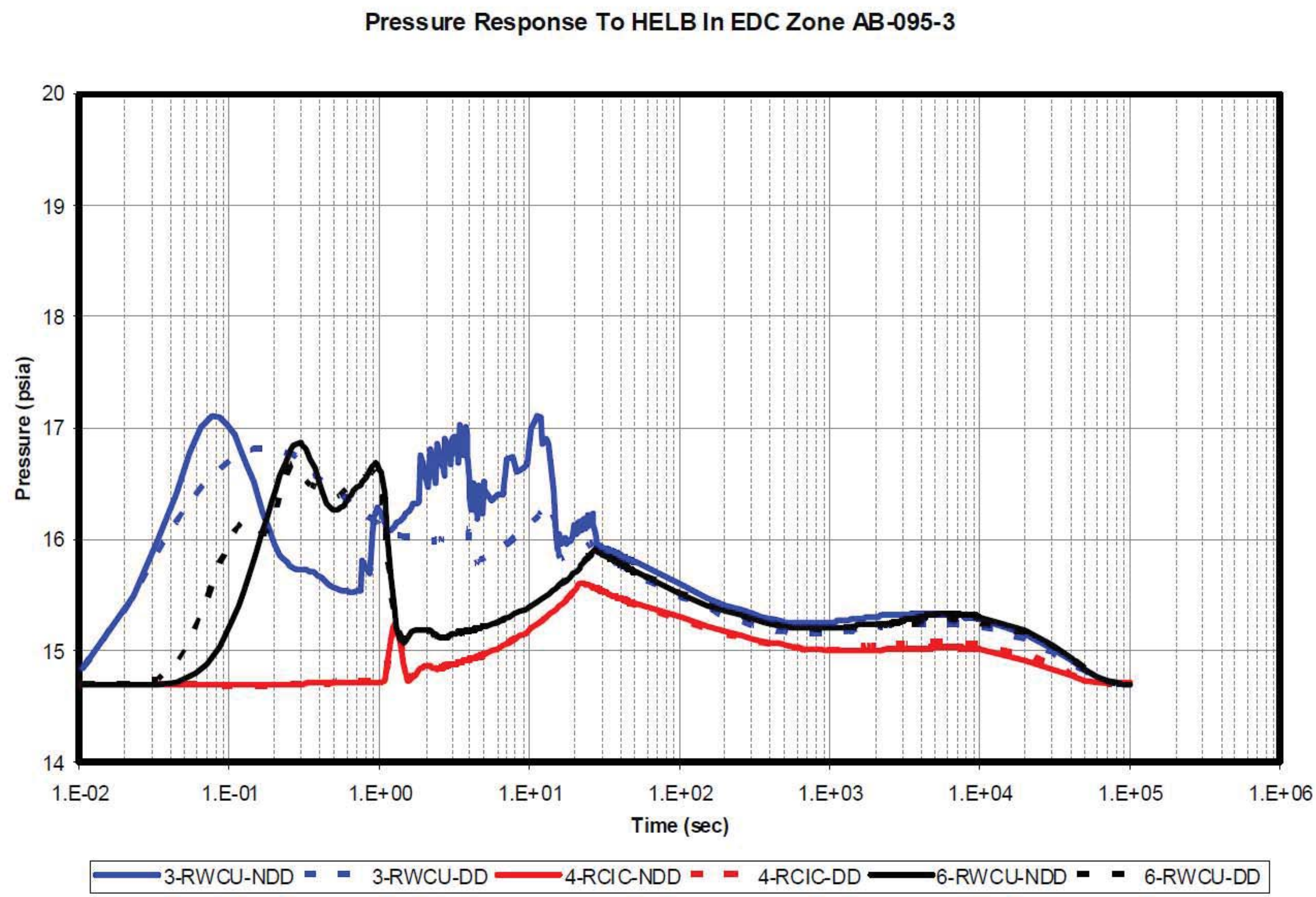
EDC ZONE AB-095-2

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-11

PRESSURE TRANSIENTS IN CV 8
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



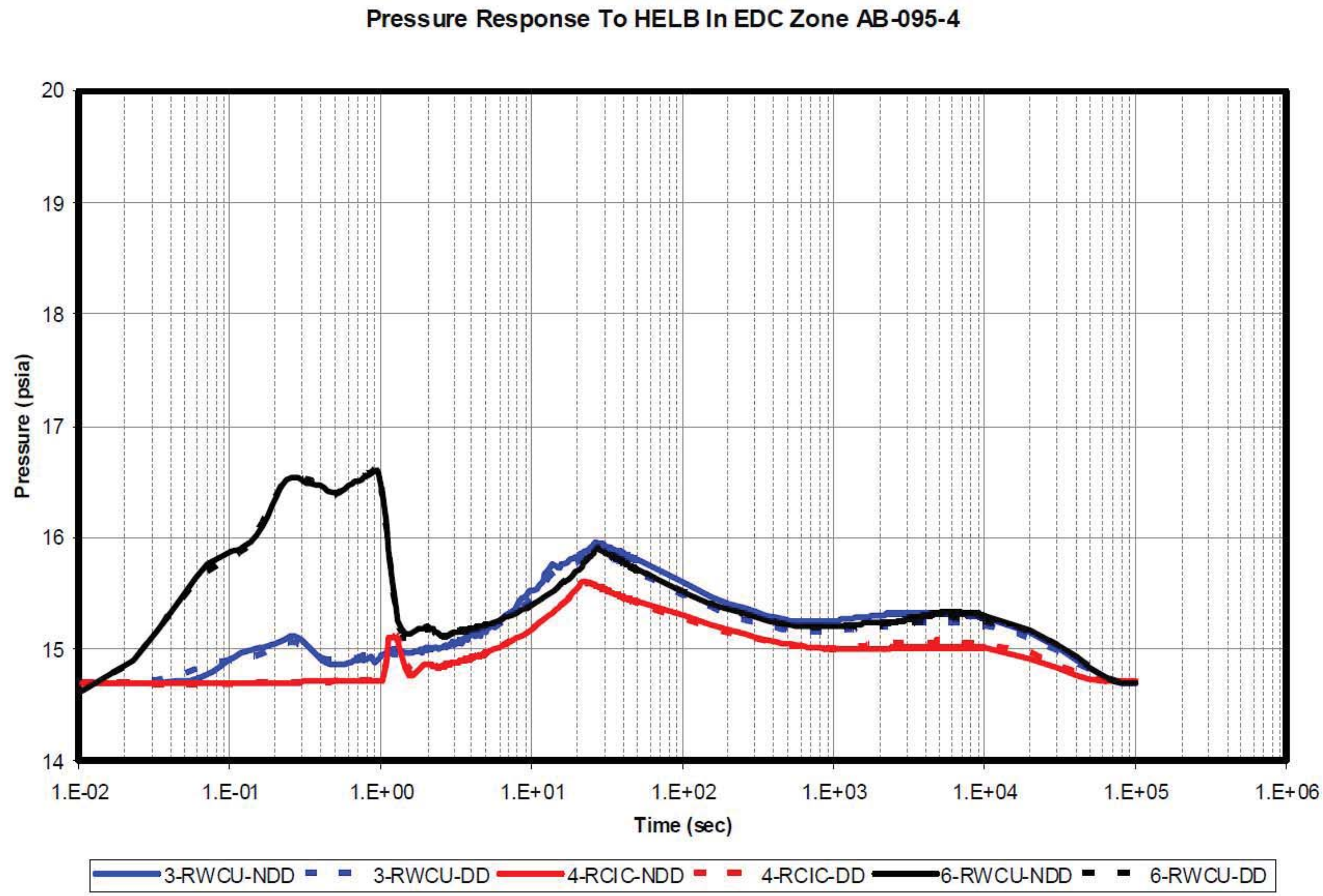
EDC ZONE AB-095-3

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-12

PRESSURE TRANSIENTS IN CV 9
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 23



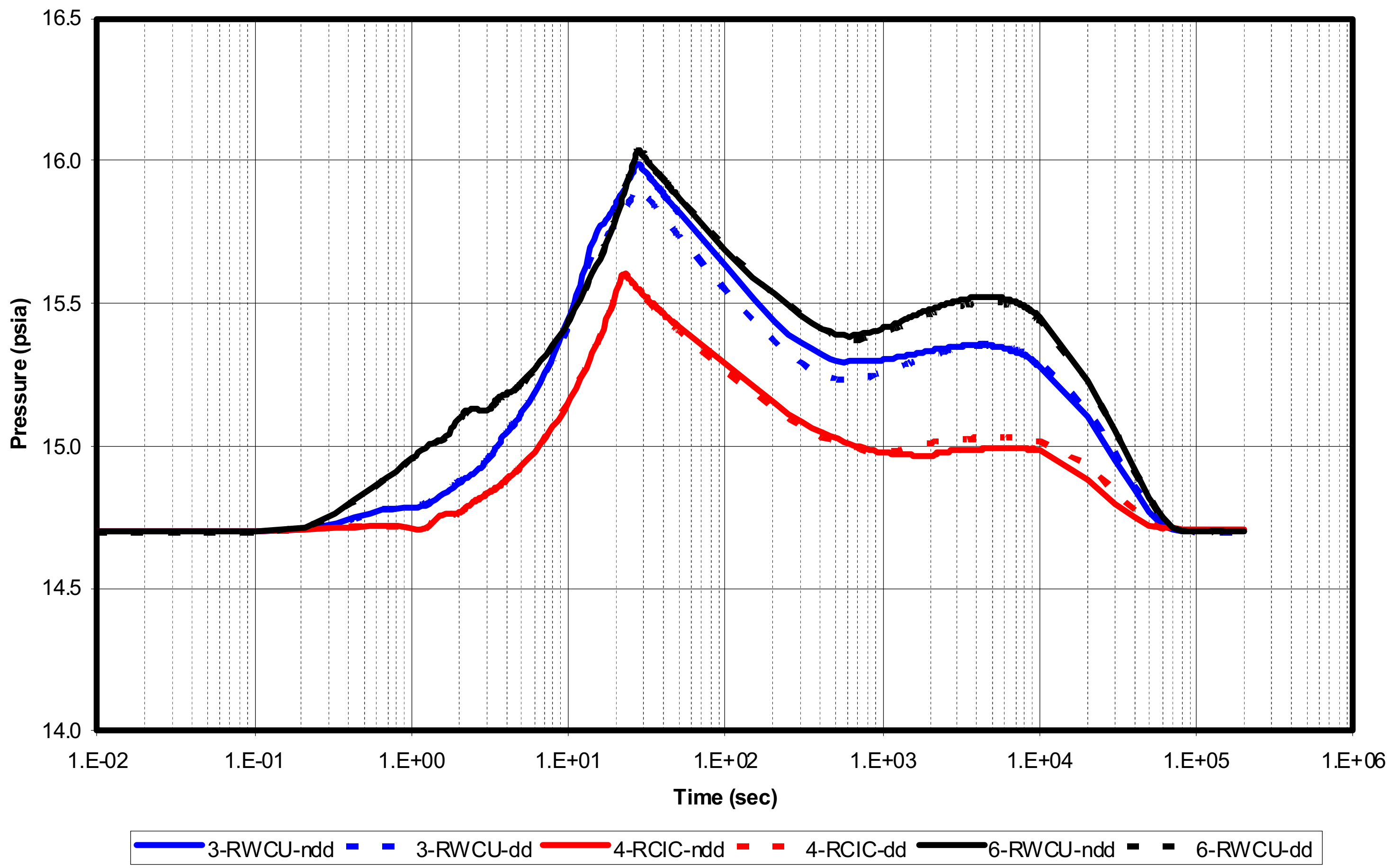
EDC ZONE AB-095-4

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-13

PRESSURE TRANSIENTS IN CV 11
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 23



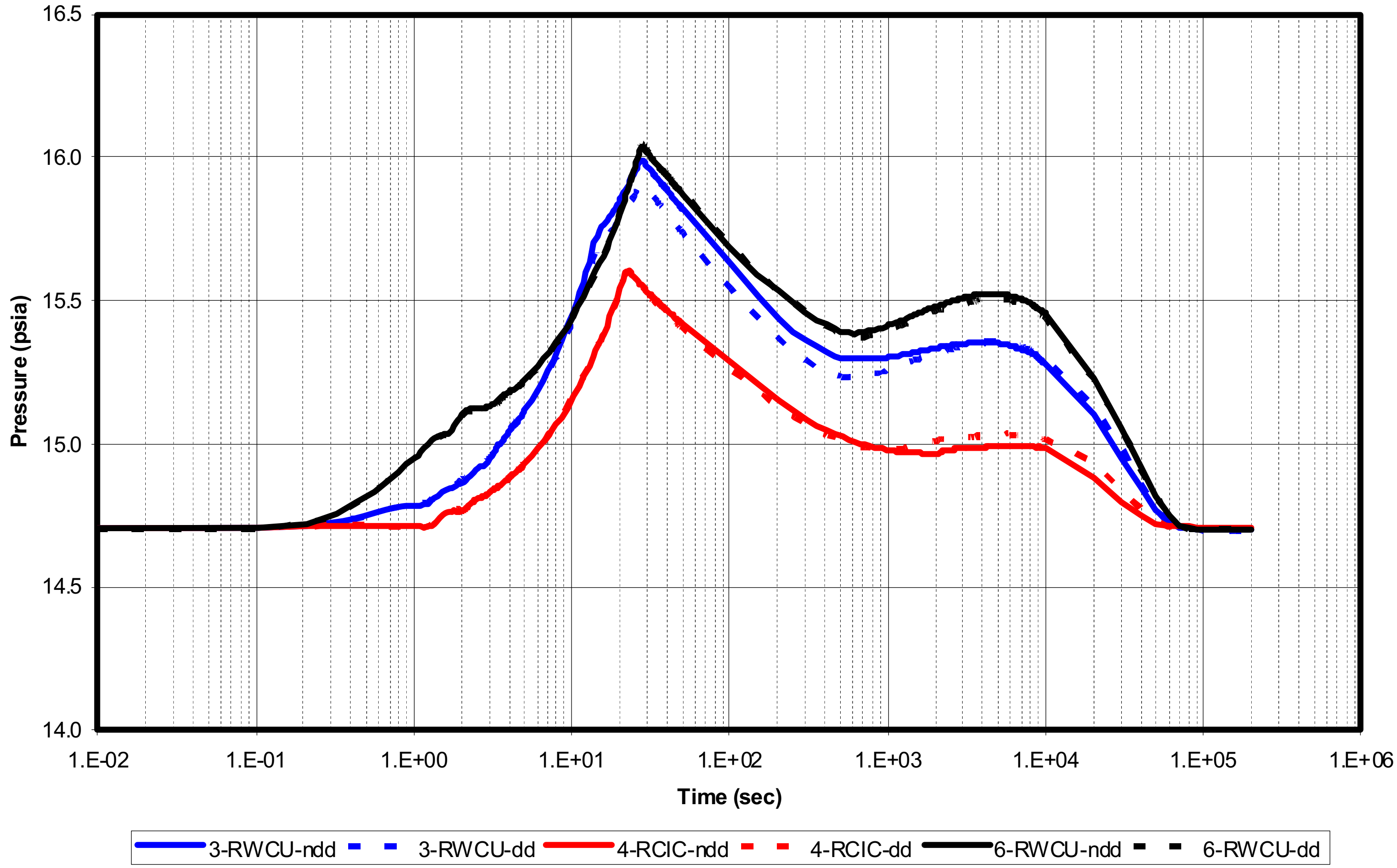
EDC ZONE AB-095-5

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-14

PRESSURE TRANSIENTS IN CV 12
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



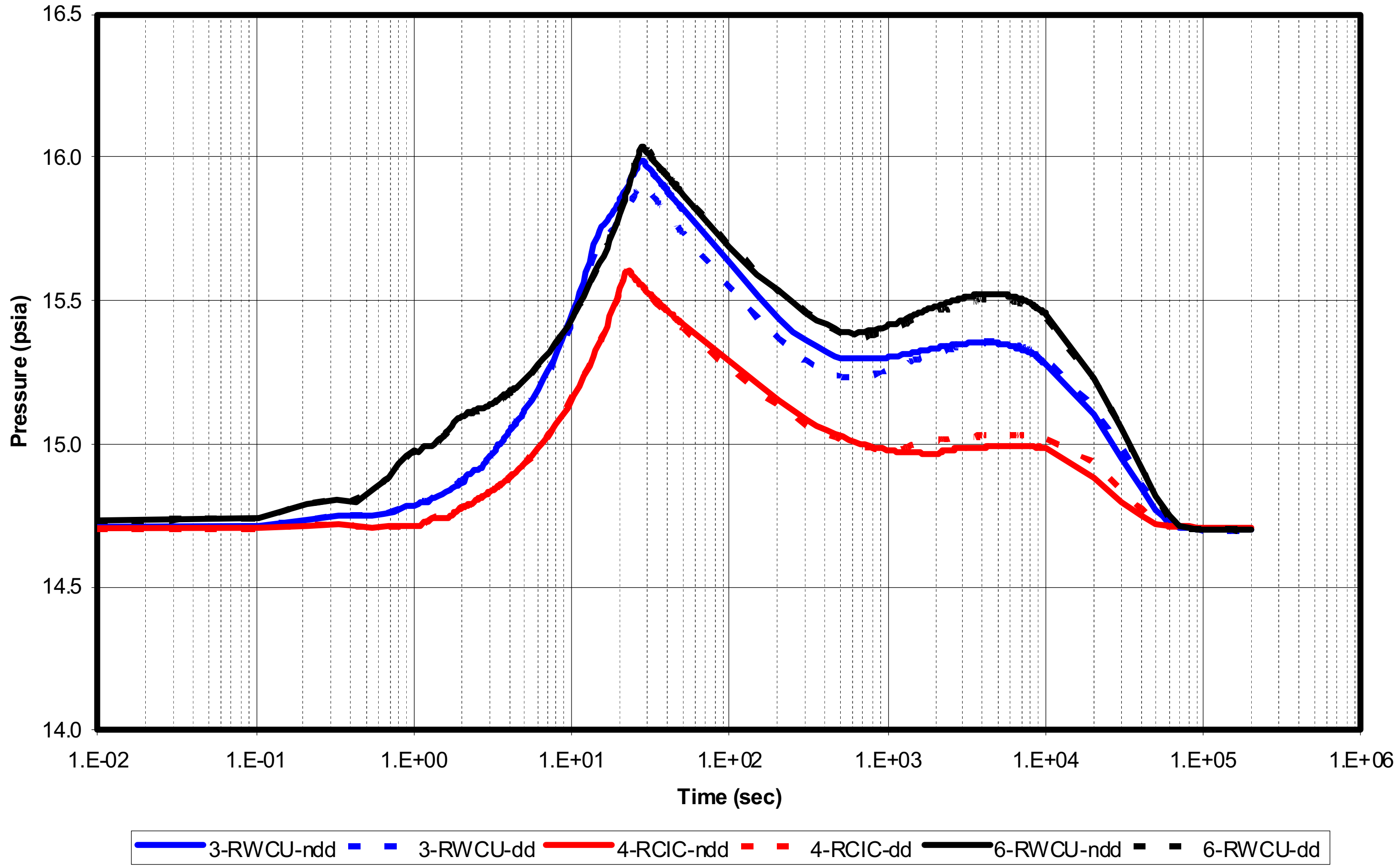
EDC ZONE AB-095-6

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-15

PRESSURE TRANSIENTS IN CV 13
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



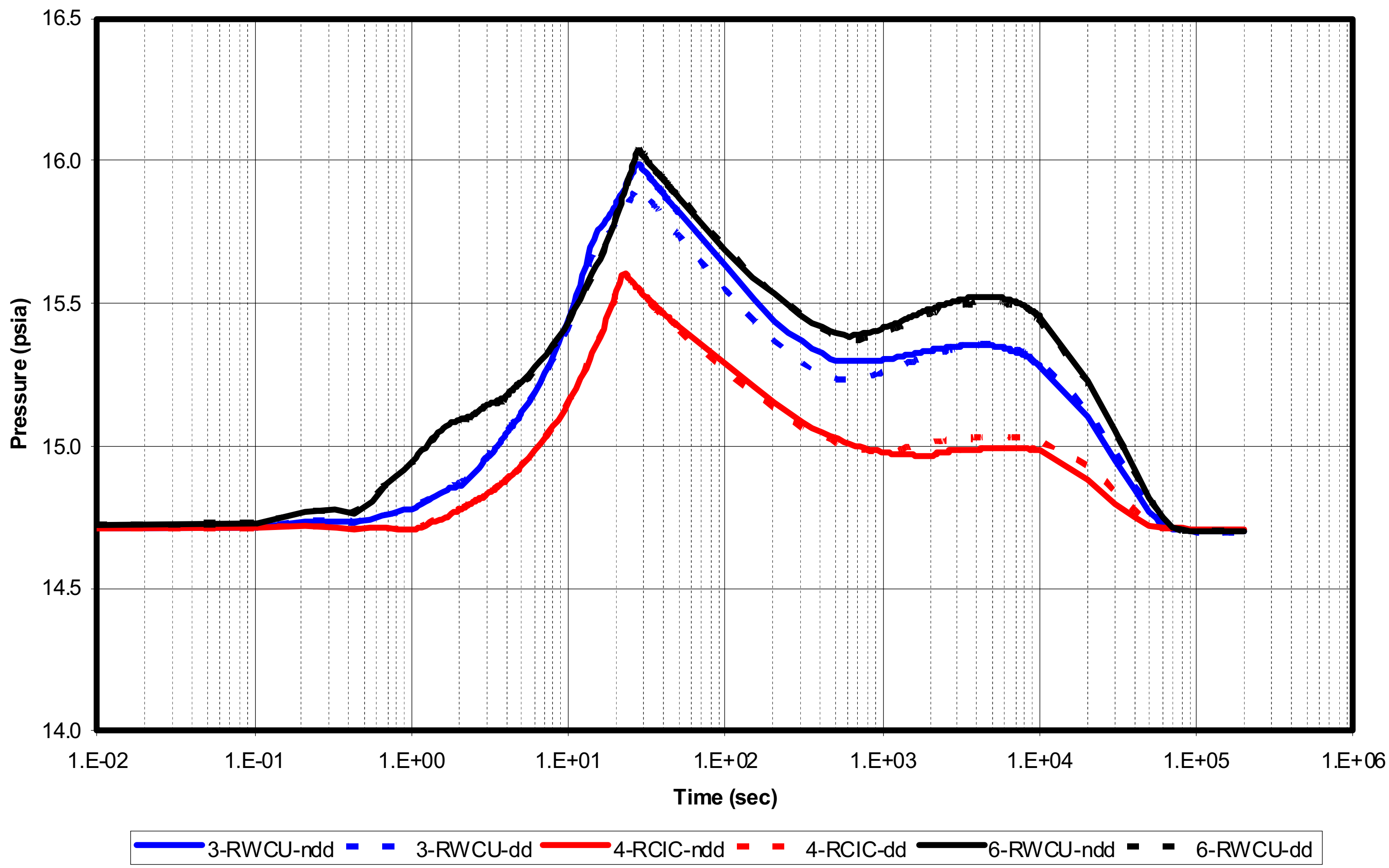
EDC ZONE AB-095-7

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-16

PRESSURE TRANSIENTS IN CV 14
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



EDC ZONE AB-095-8

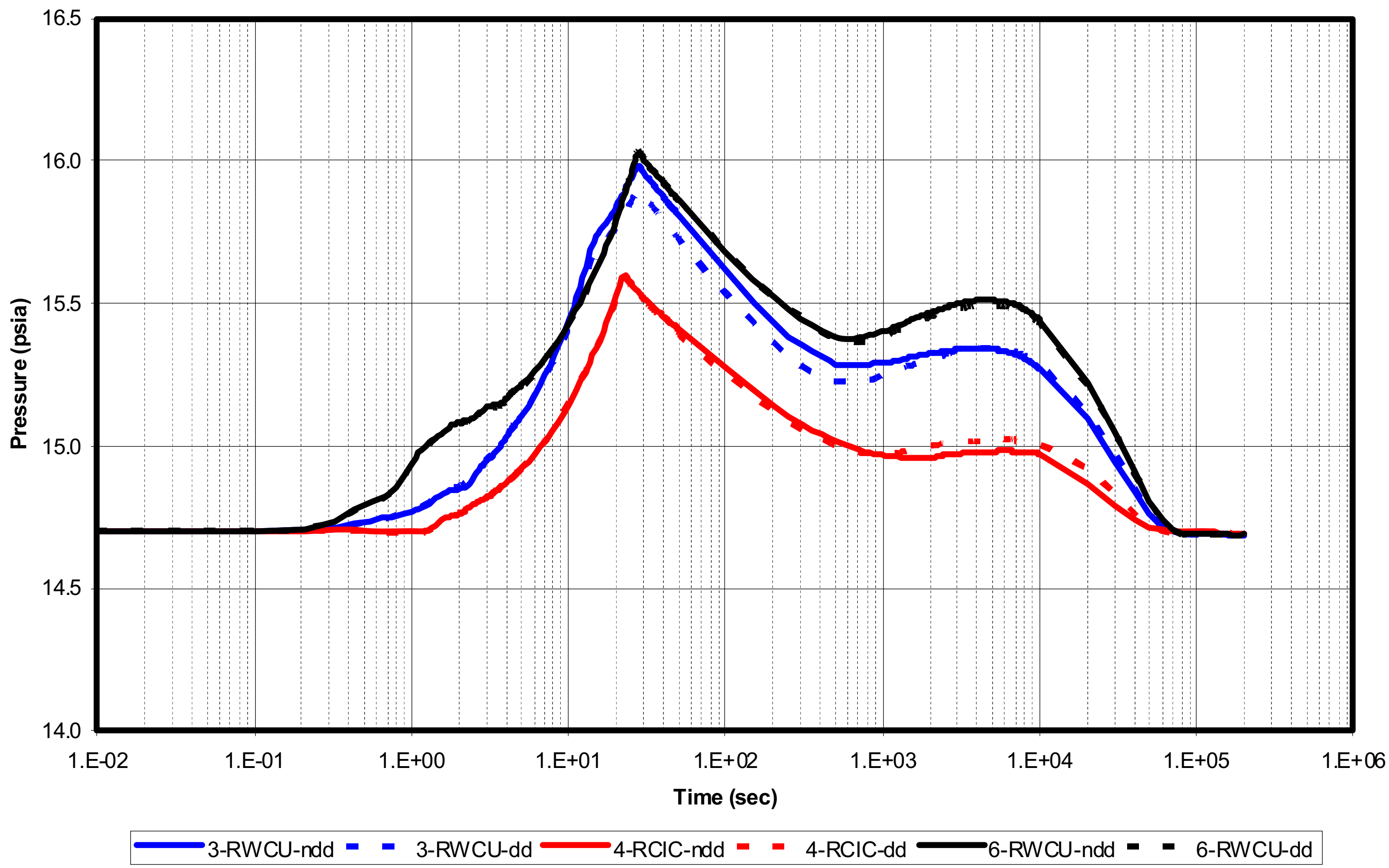
dd = Duct Destruction

ndd = No Duct Destruction

FIGURE 3B-17

PRESSURE TRANSIENTS IN CV 10
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



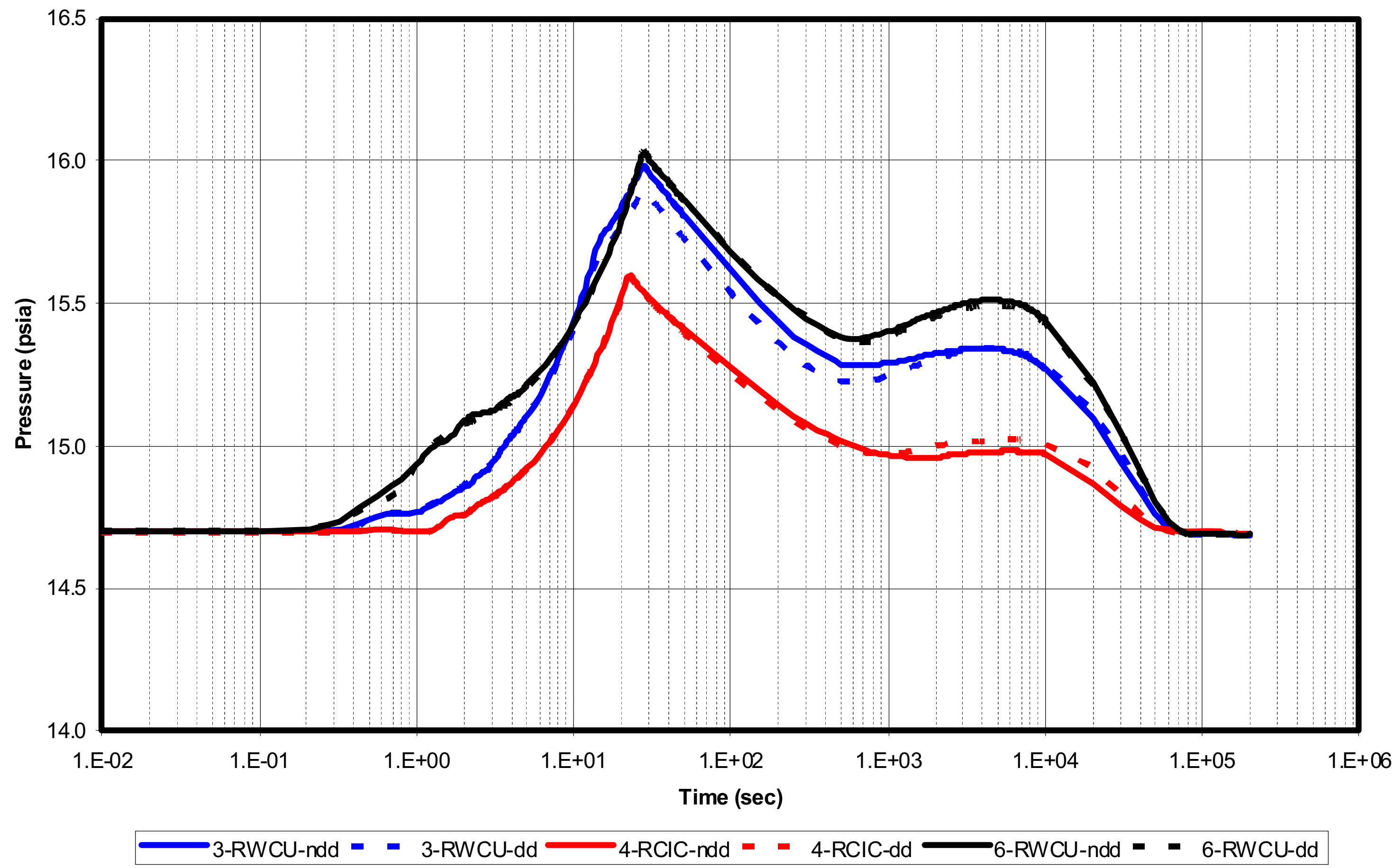
EDC ZONE AB-114-1

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-18

PRESSURE TRANSIENTS IN CV 15
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



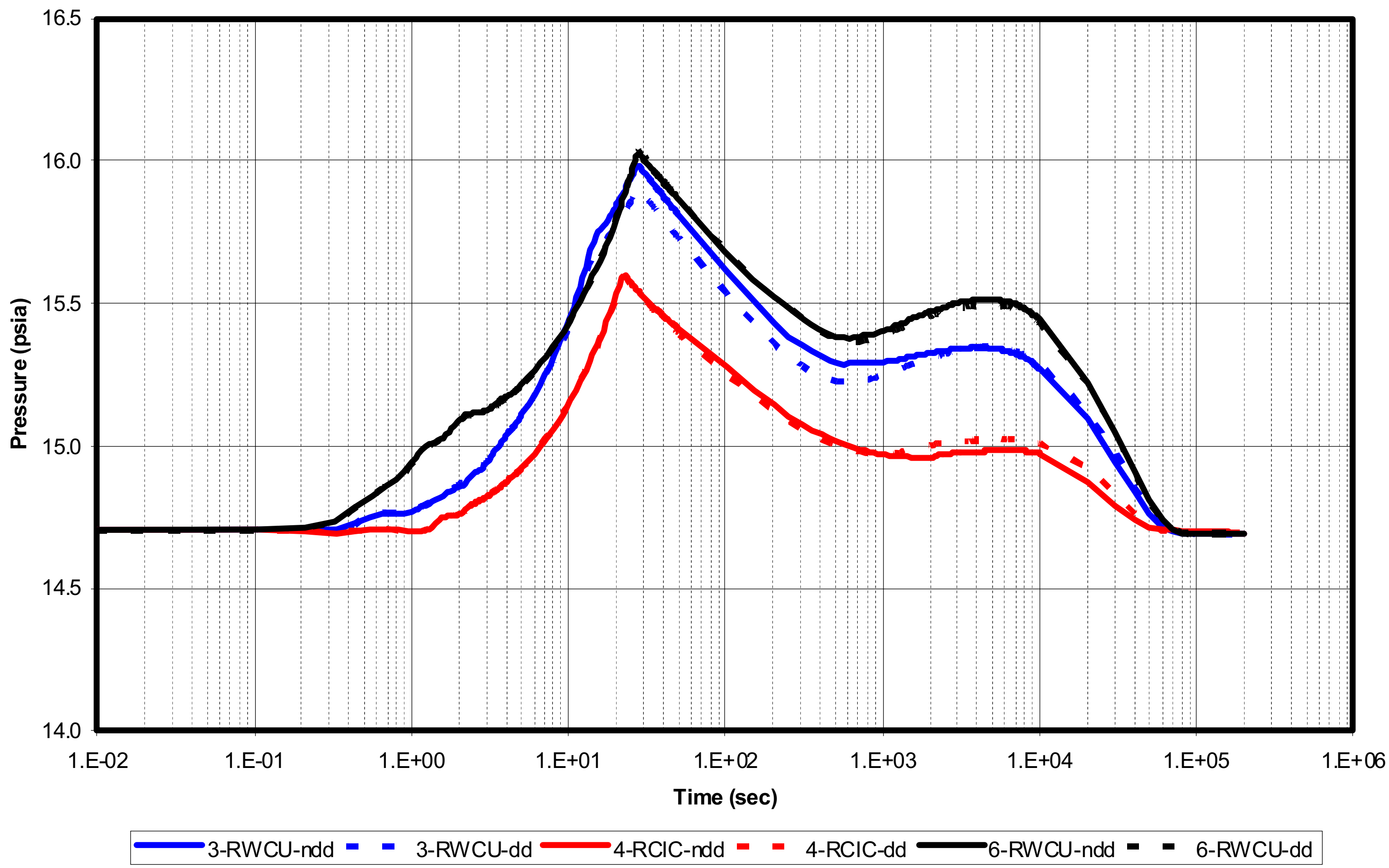
EDC ZONE AB-114-3

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-19

PRESSURE TRANSIENTS IN CV 19
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



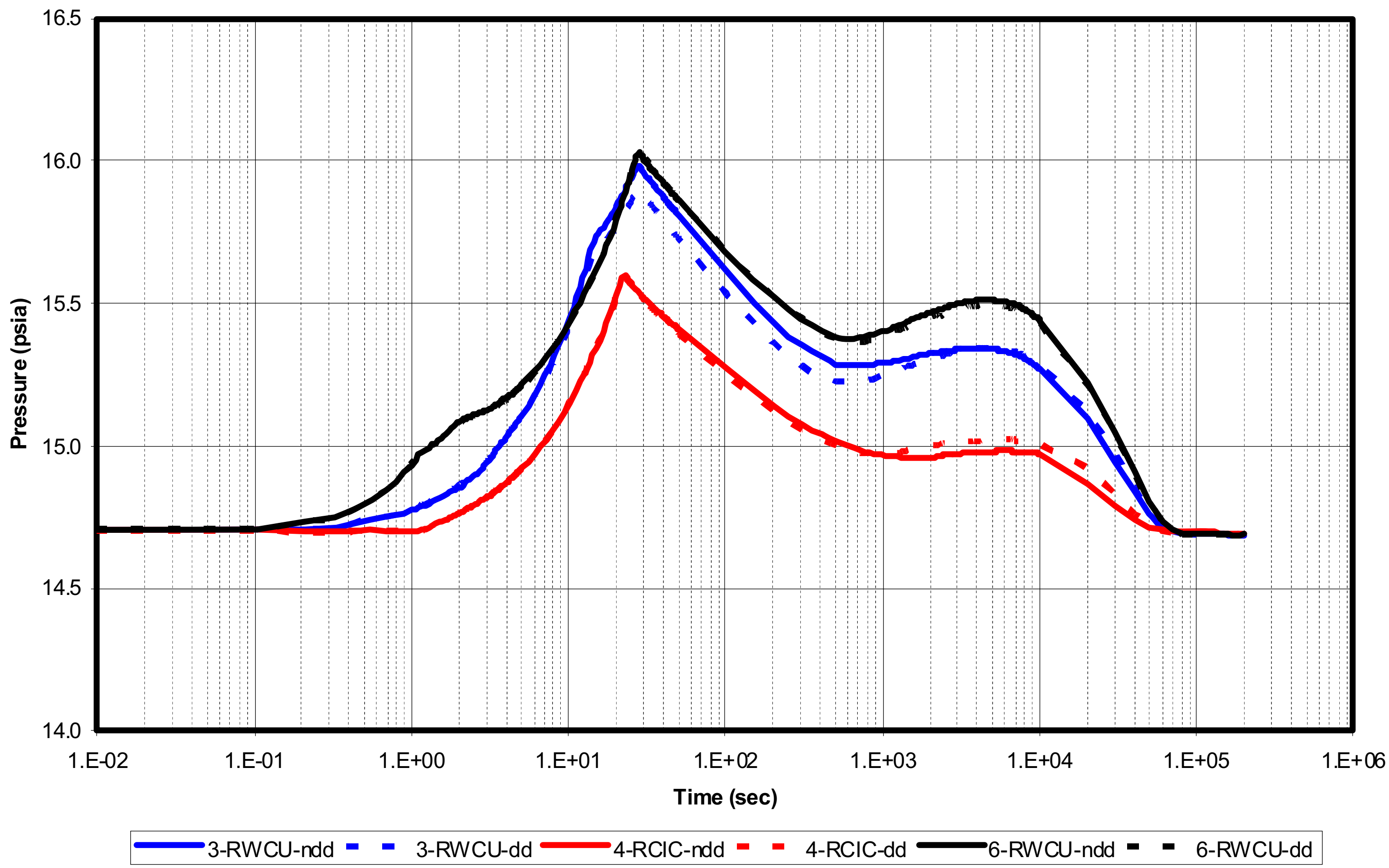
EDC ZONE AB-114-4

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-20

PRESSURE TRANSIENTS IN CV 31
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



EDC ZONE AB-114-5

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-21

PRESSURE TRANSIENTS IN CV 20
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22

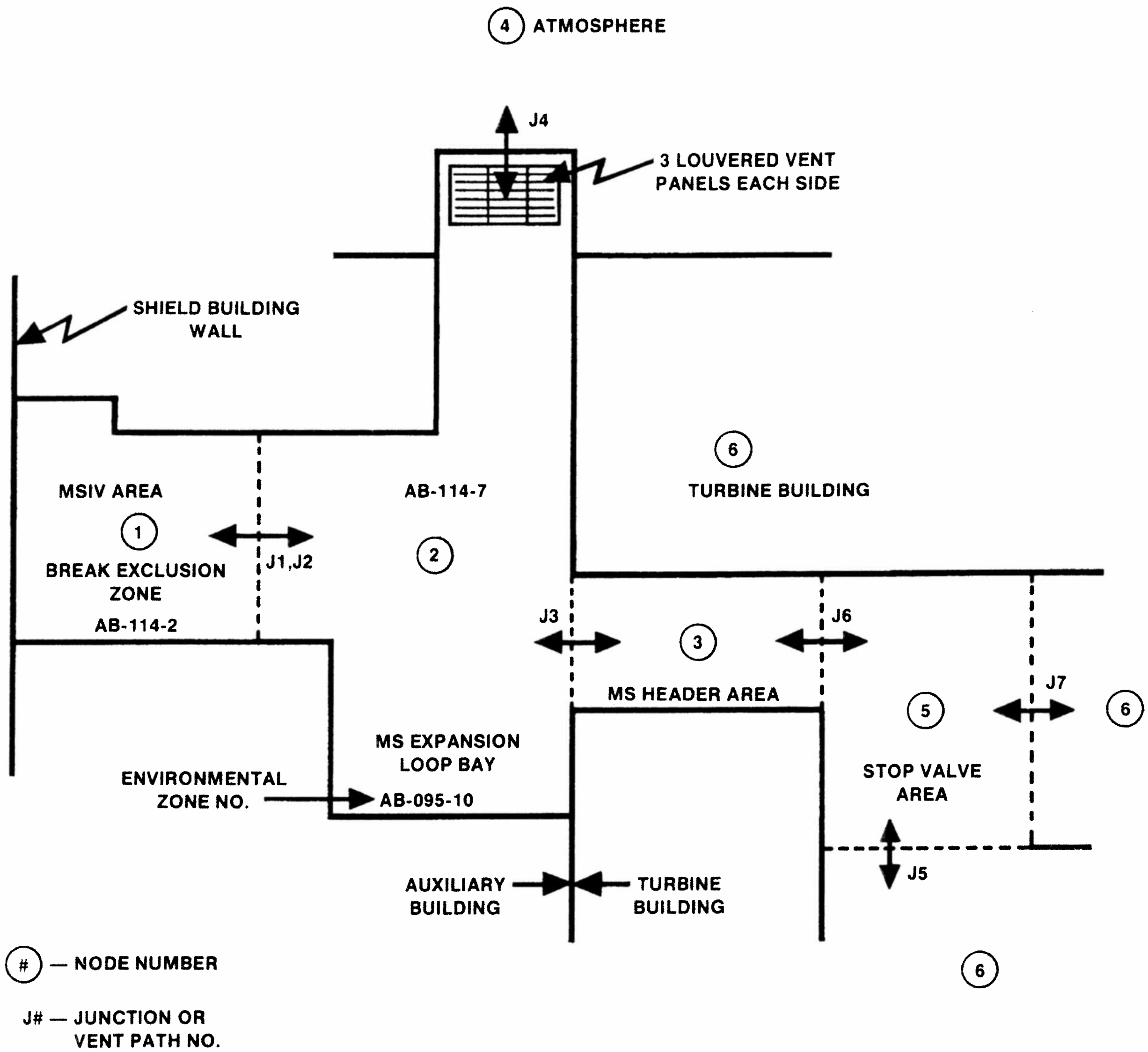
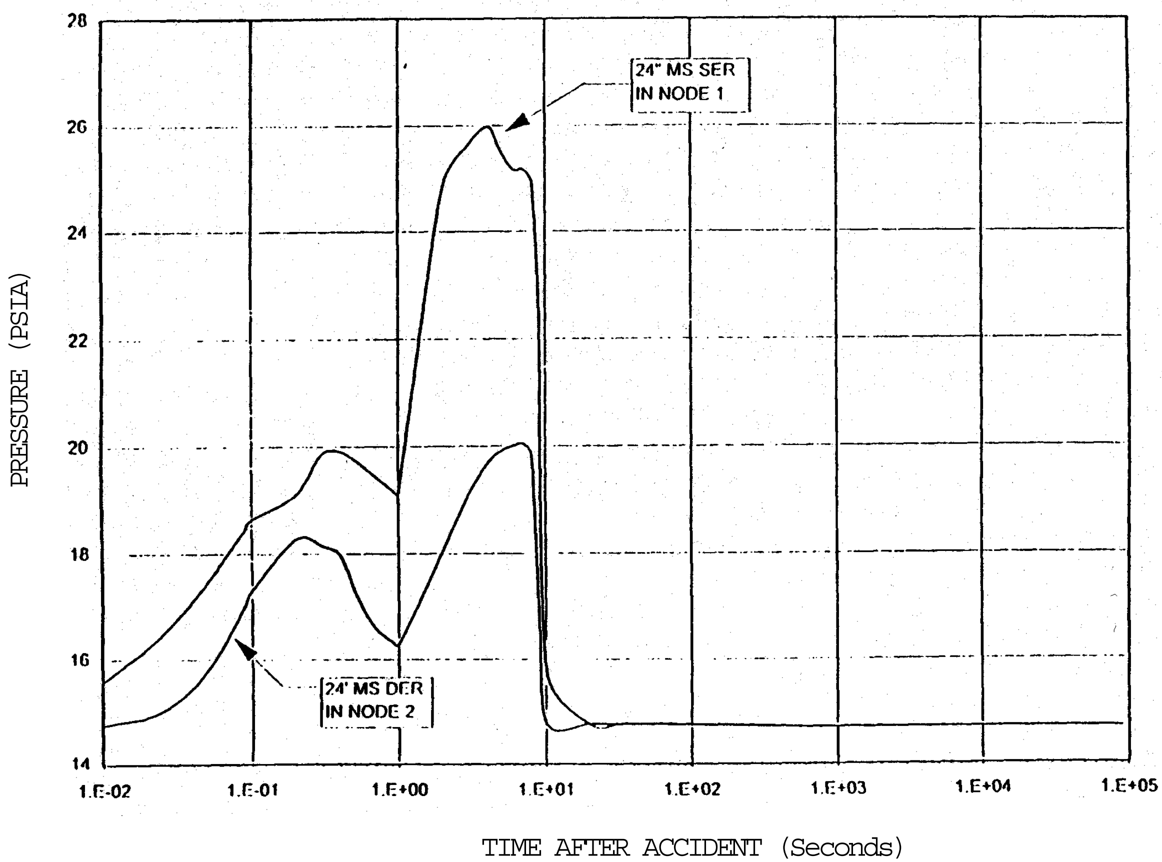


FIGURE 3B-22

NODALIZATION DIAGRAM
MAIN STEAM TUNNEL
6 NODE MODEL

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT



DER - DOUBLE ENDED RUPTURE
SER - SINGLE ENDED RUPTURE

FIGURE 3B-23

PRESSURE TRANSIENTS IN NODE 1
MAIN STEAM TUNNEL
HIGH ENERGY LINE BREAK ANALYSIS

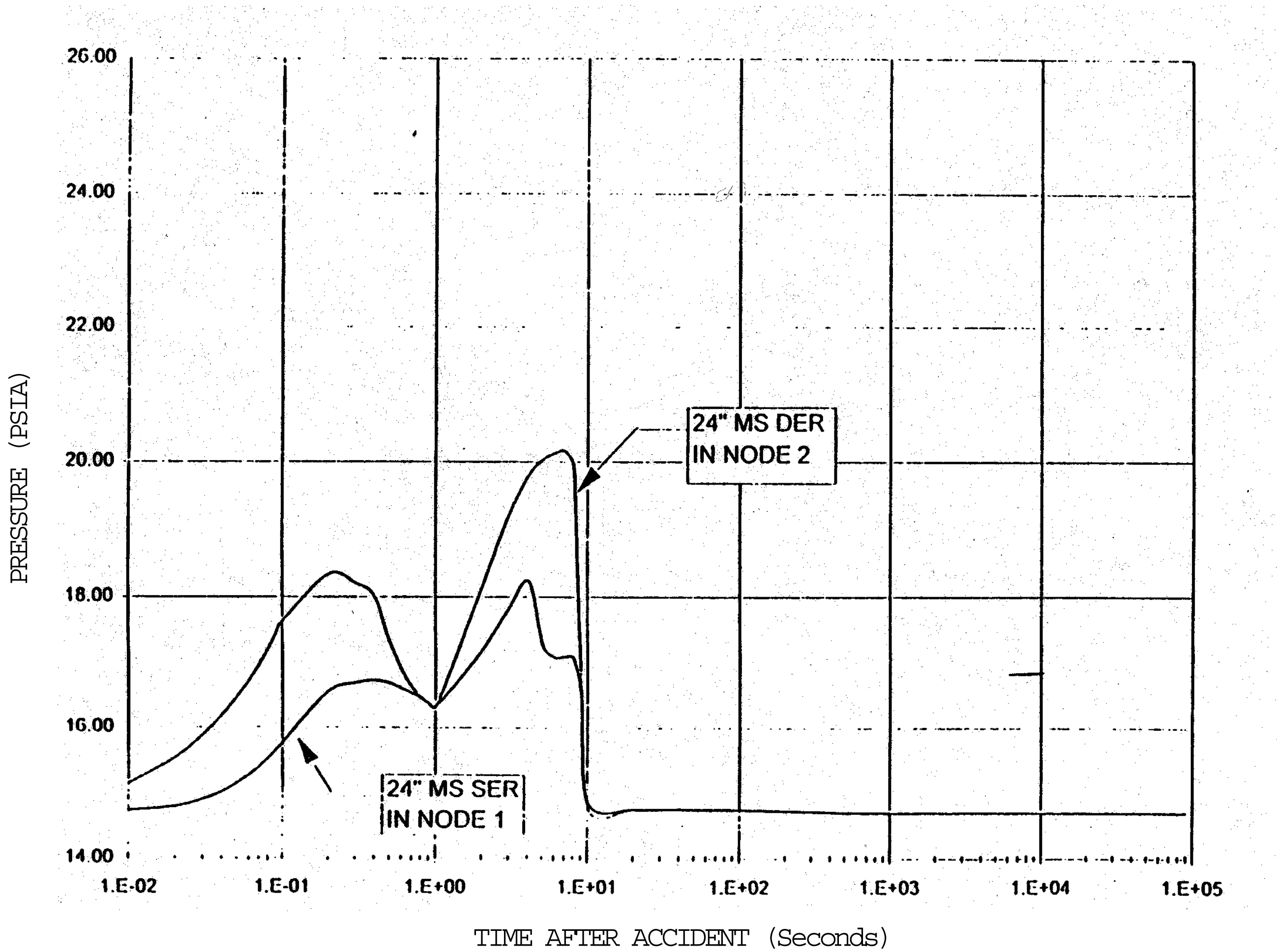
RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

THIS FIGURE HAS BEEN DELETED

FIGURE 3B-24

PRESSURE TRANSIENTS IN NODE 1
MAIN STEAM TUNNEL
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT



DER - DOUBLE ENDED RUPTURE
SER - SINGLE ENDED RUPTURE

FIGURE 3B-25

PRESSURE TRANSIENTS IN NODE 2
MAIN STEAM TUNNEL
HIGH ENERGY LINE BREAK ANALYSIS

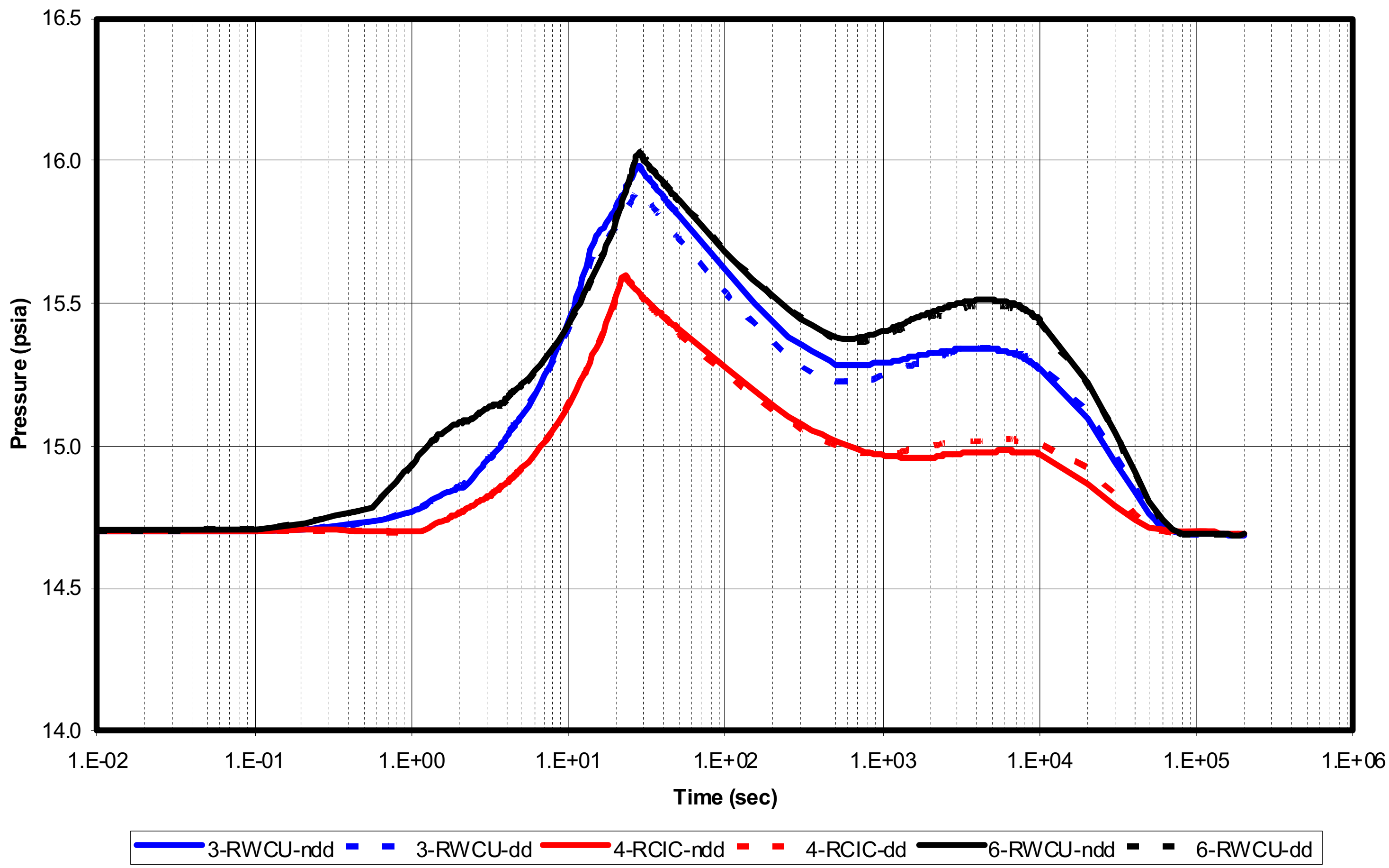
RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

THIS FIGURE HAS BEEN DELETED

FIGURE 3B-26

PRESSURE TRANSIENTS IN NODE 2
MAIN STEAM TUNNEL
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT



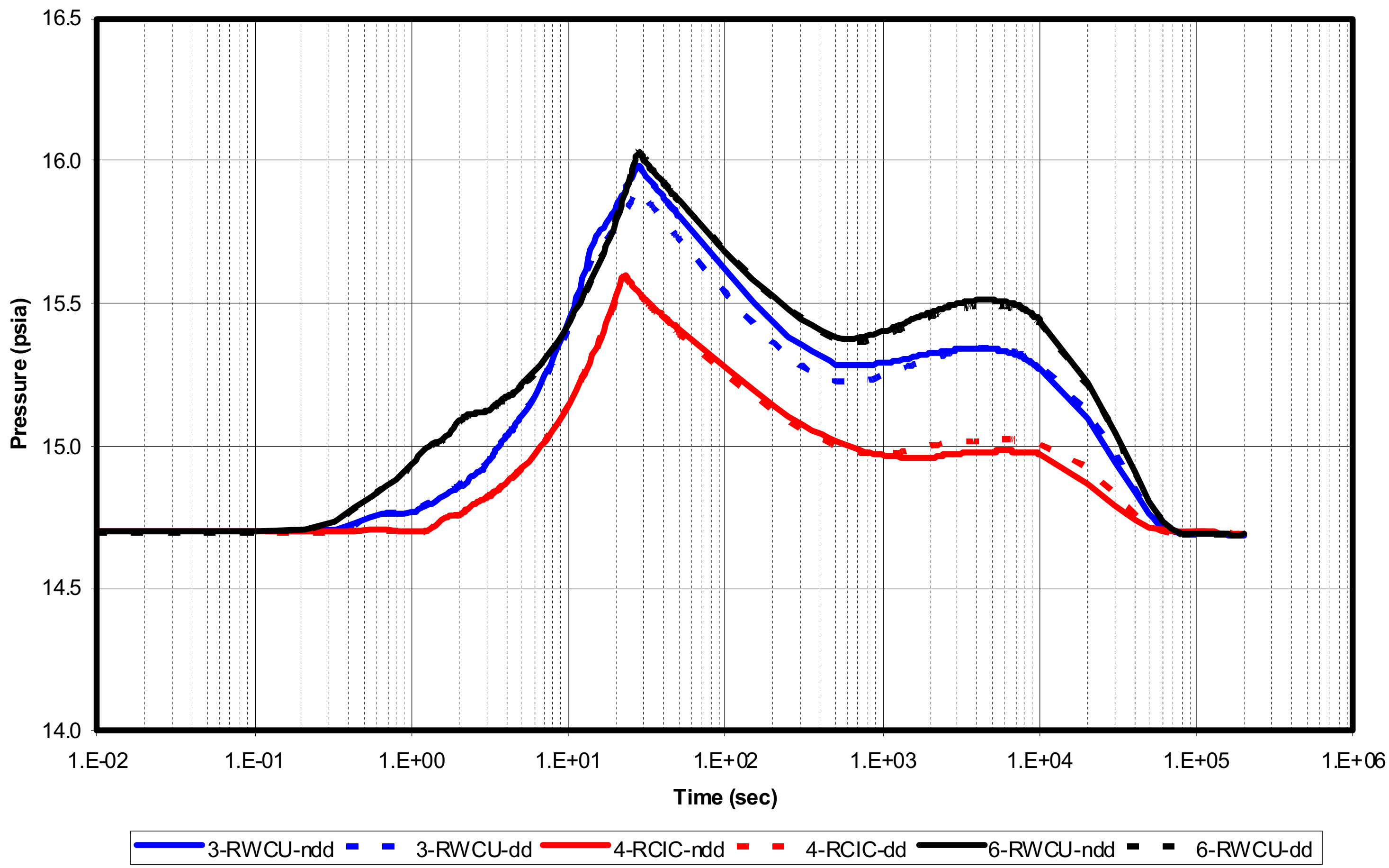
EDC ZONE AB-114-6

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-27

PRESSURE TRANSIENTS IN CV 17
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



EDC ZONE AB-114-8

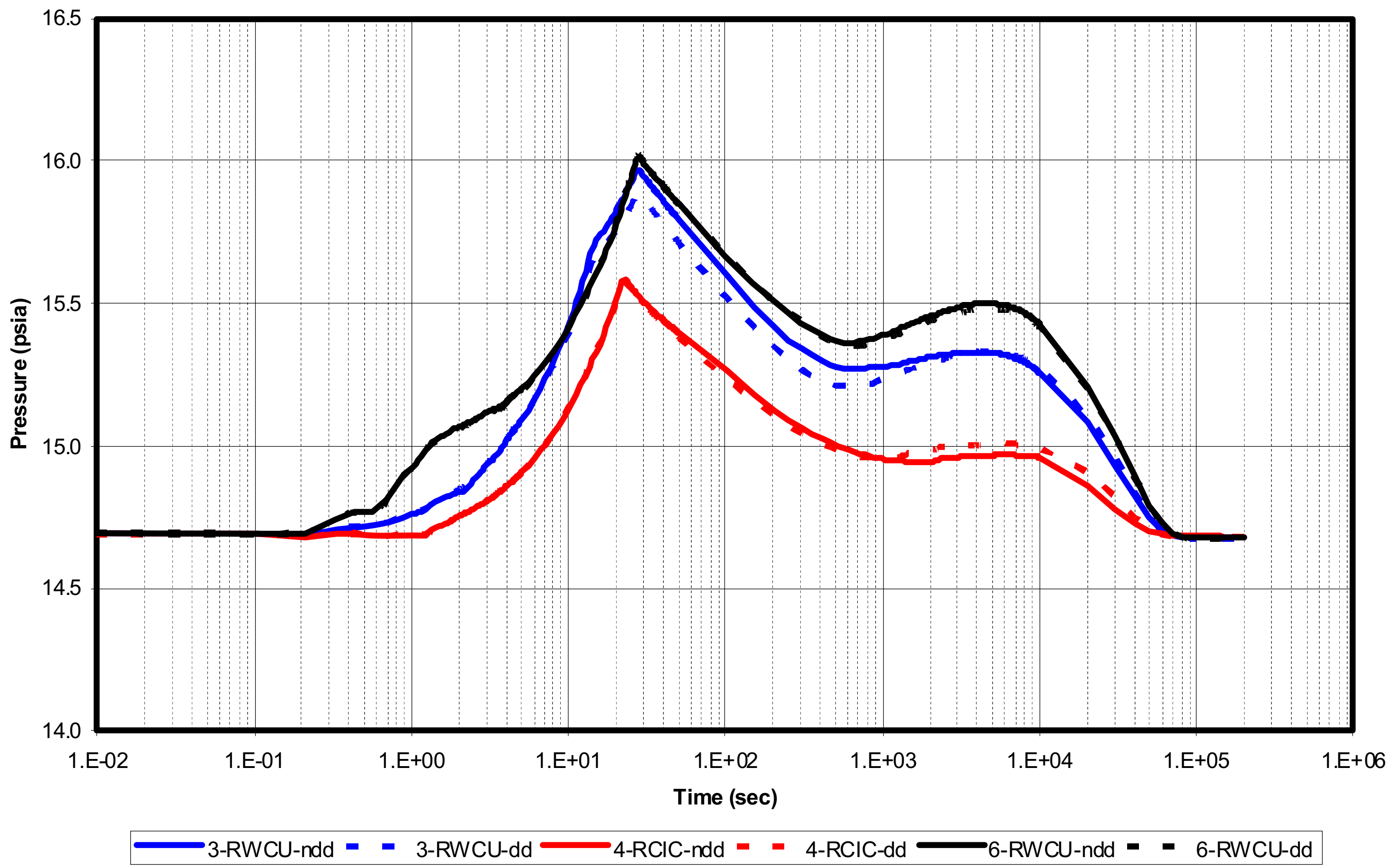
dd = Duct Destruction

ndd = No Duct Destruction

FIGURE 3B-28

PRESSURE TRANSIENTS IN CV 50
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



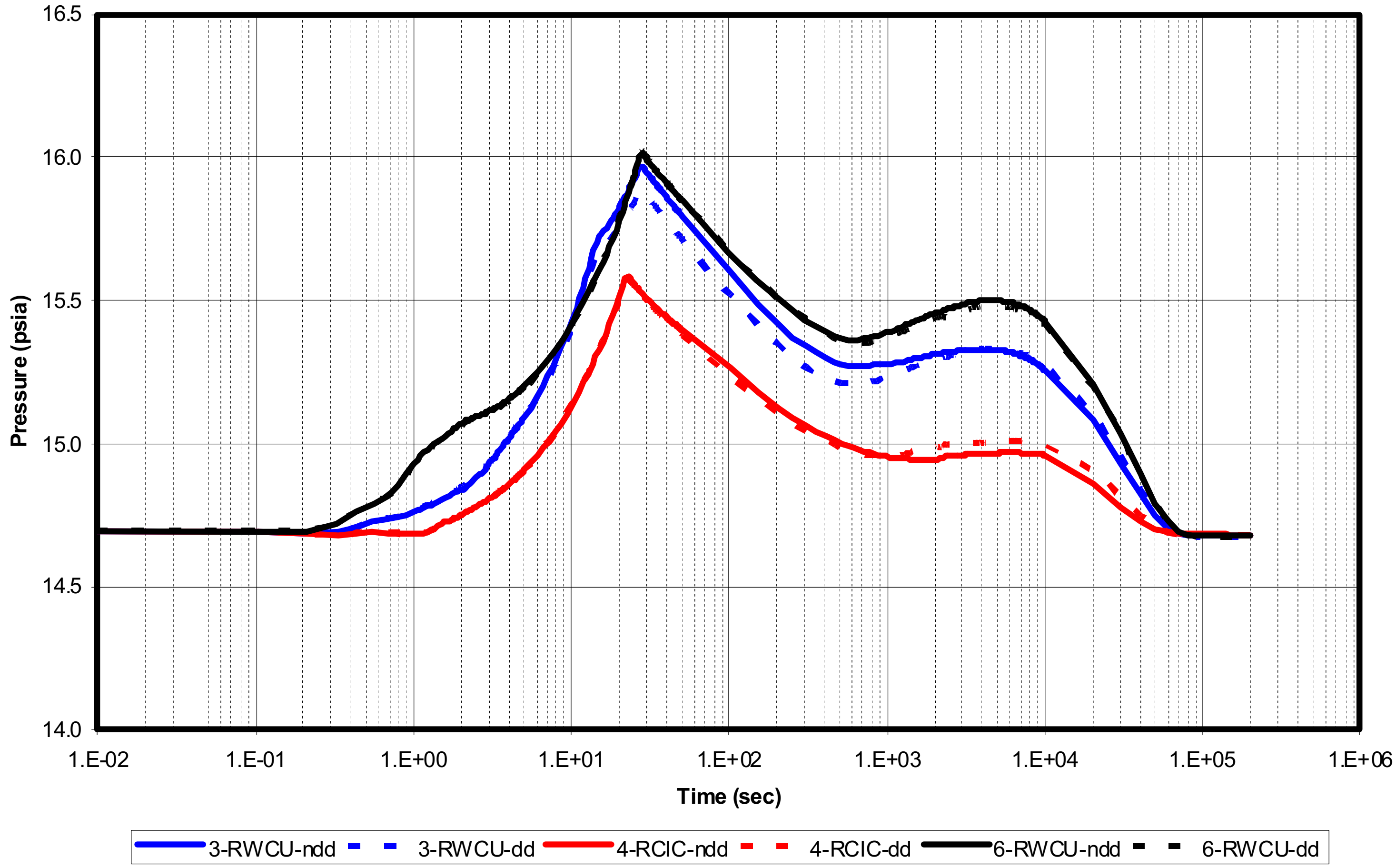
EDC ZONE AB-141-1

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-29

PRESSURE TRANSIENTS IN CV 24
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



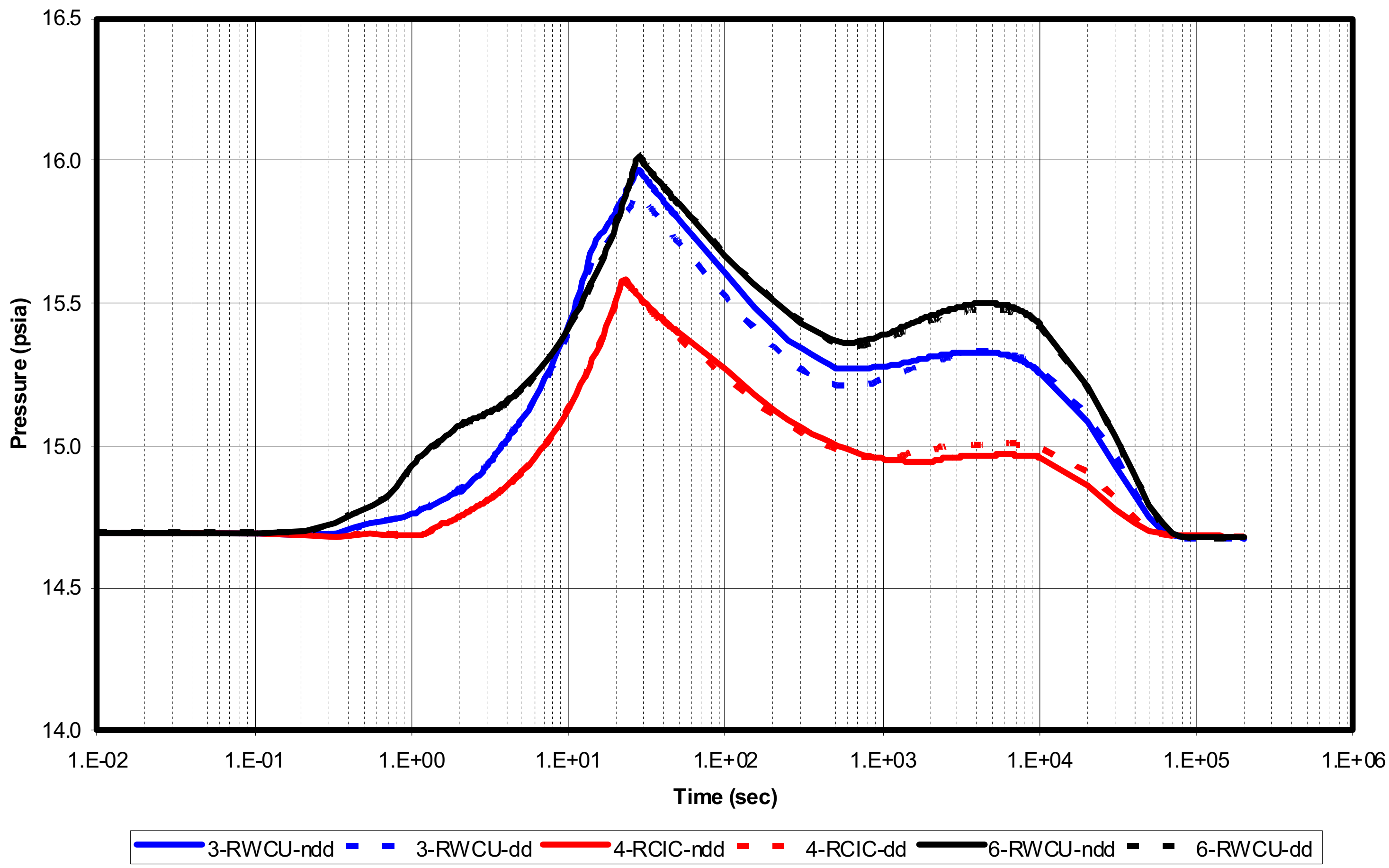
EDC ZONE AB-141-2

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-30

PRESSURE TRANSIENTS IN CV 22
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



EDC ZONE AB-141-3

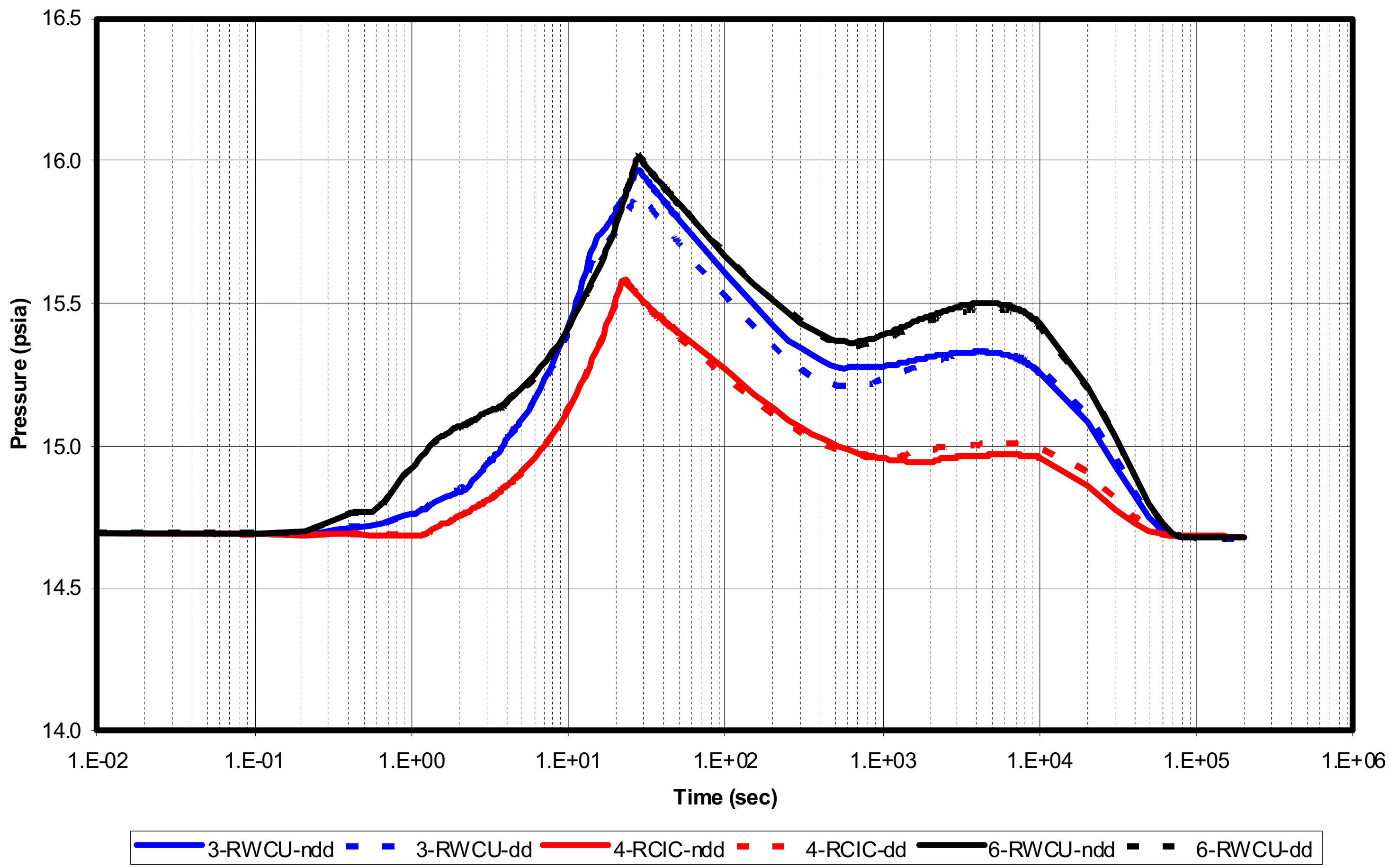
dd = Duct Destruction

ndd = No Duct Destruction

FIGURE 3B-31

PRESSURE TRANSIENTS IN CV 25
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



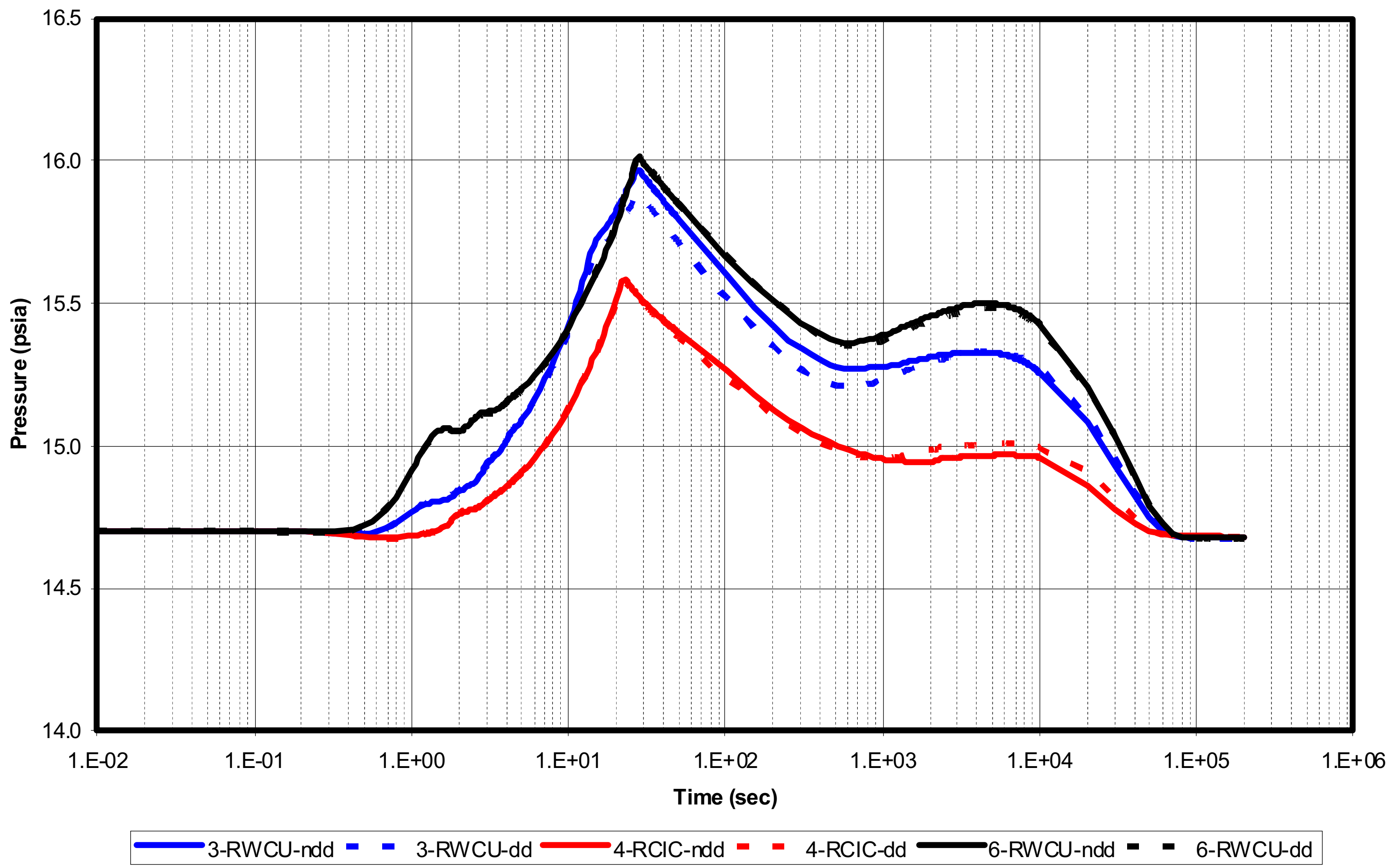
EDC ZONE AB-141-4

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-32

PRESSURE TRANSIENTS IN CV 23
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



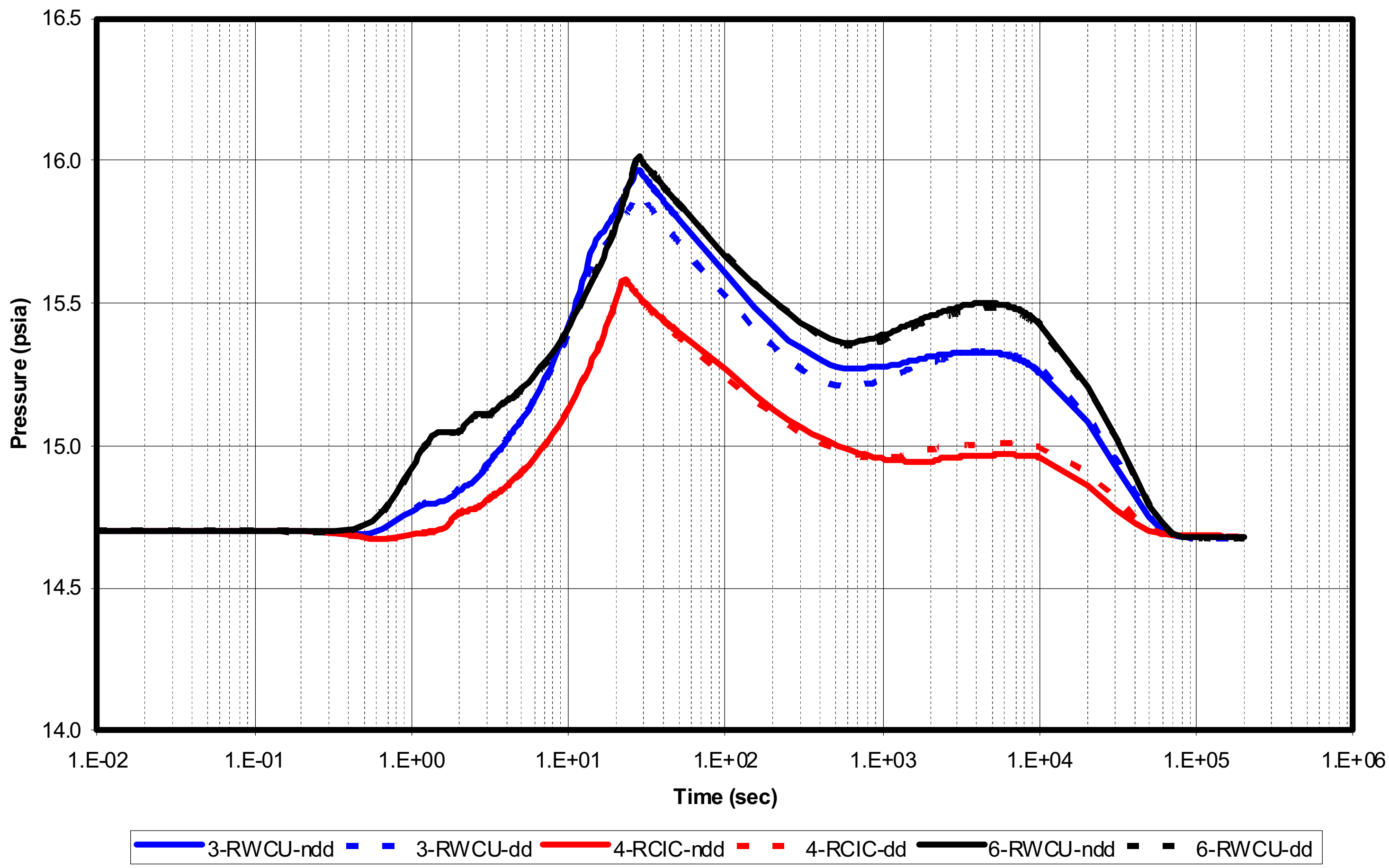
EDC ZONE AB-141-5

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-33

PRESSURE TRANSIENTS IN CV 21
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



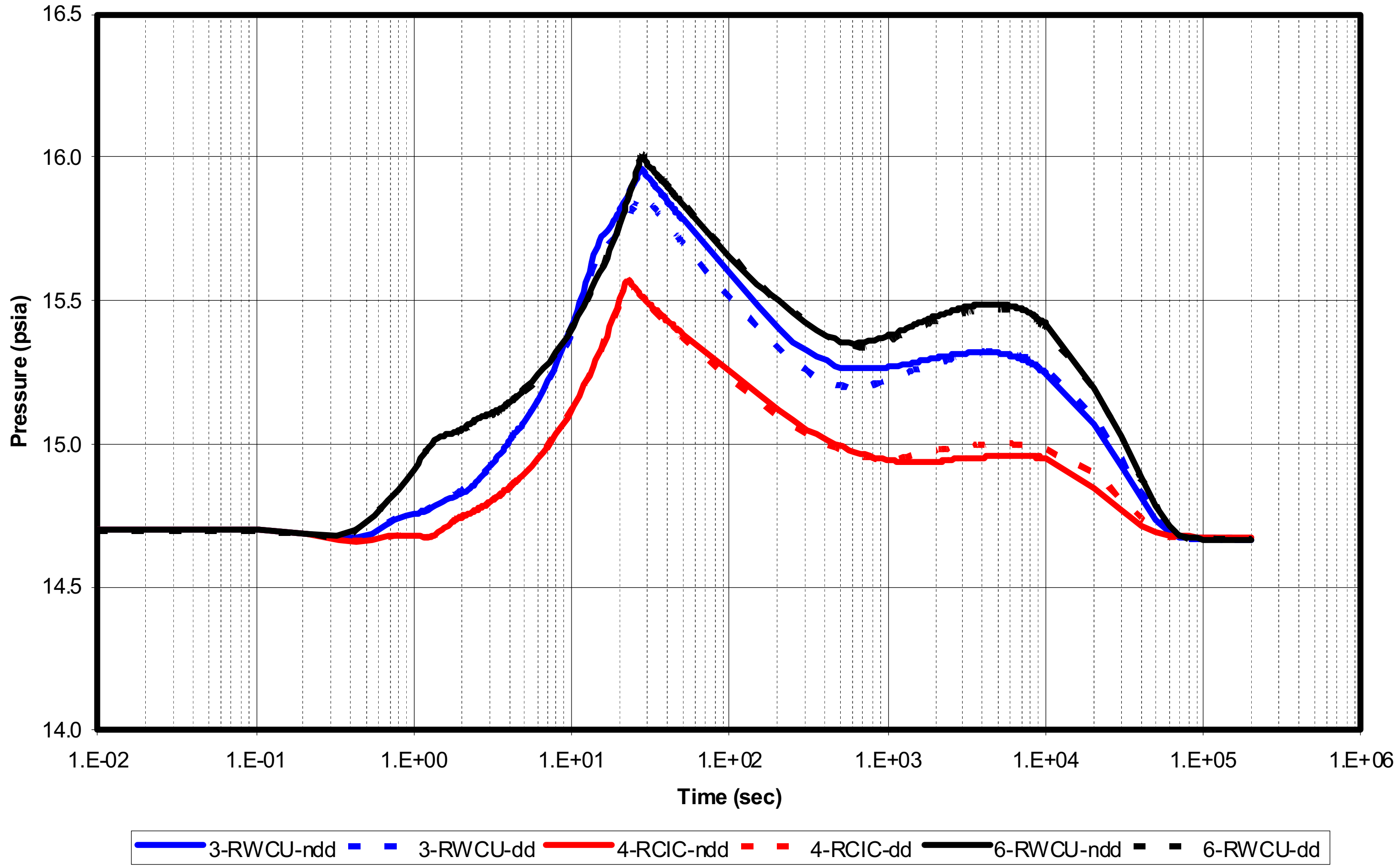
EDC ZONE AB-141-6

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-34

PRESSURE TRANSIENTS IN CV 27
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



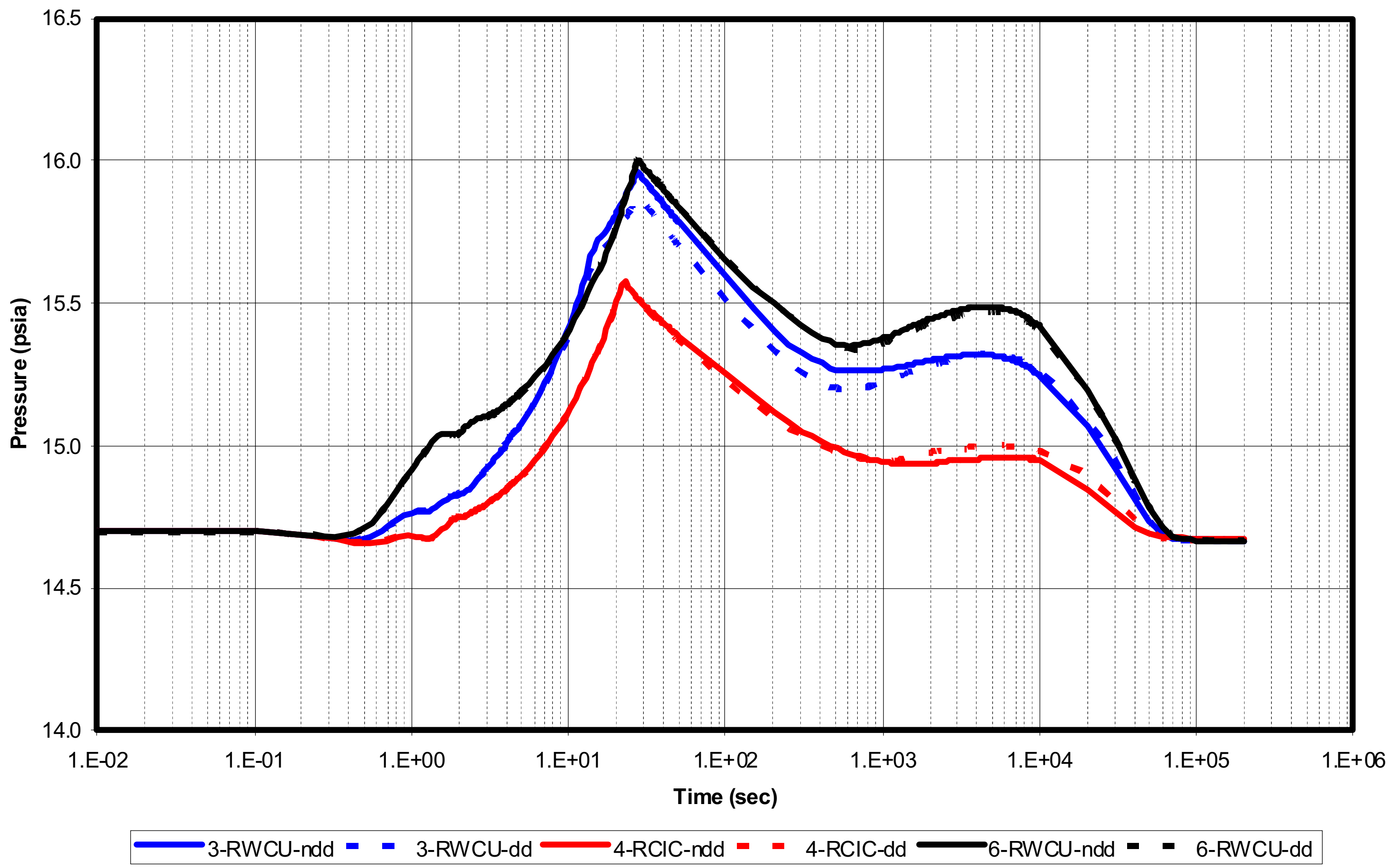
EDC ZONE AB-170-1

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-35

PRESSURE TRANSIENTS IN CV 26
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



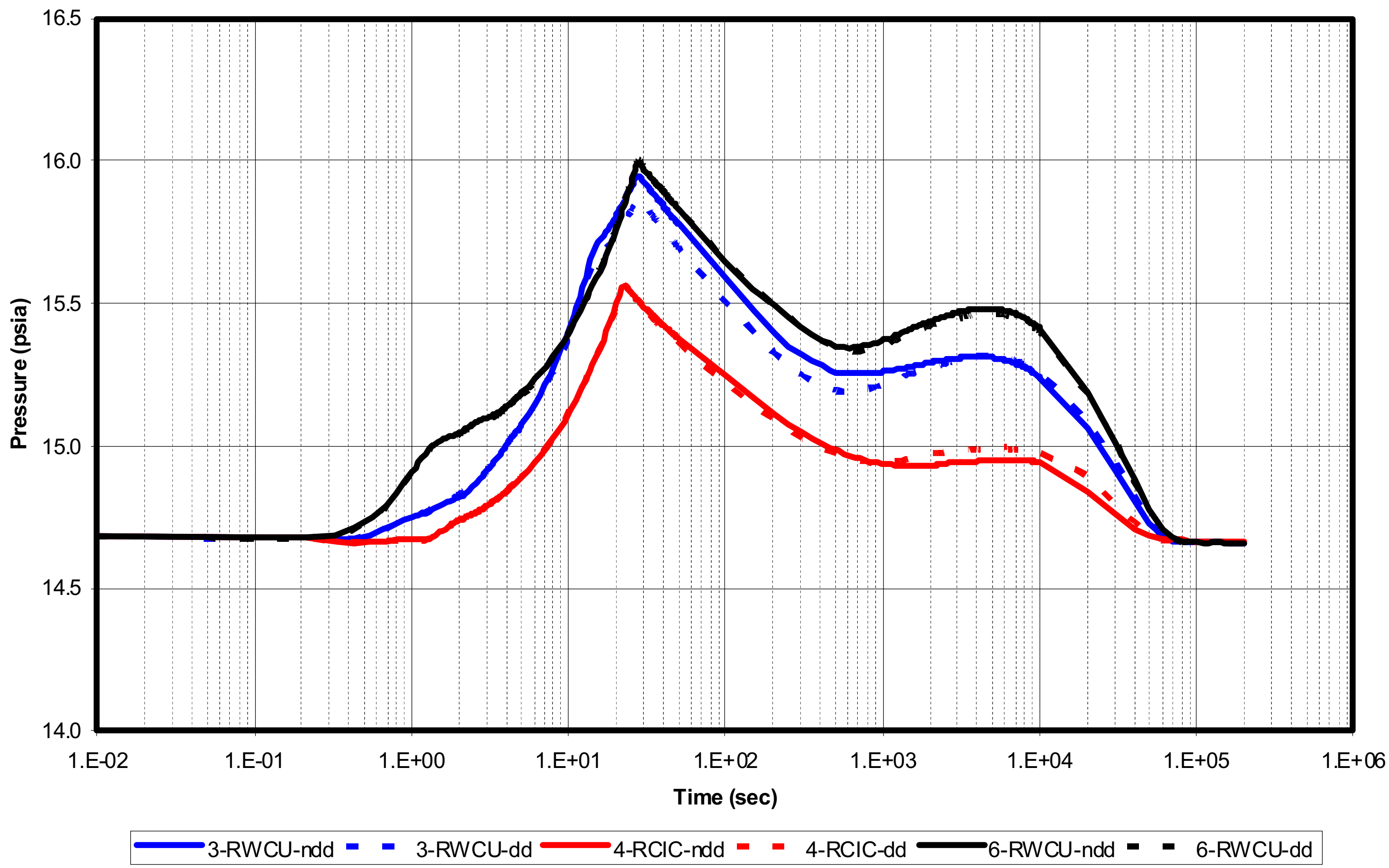
EDC ZONE AB-170-2

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-36

PRESSURE TRANSIENTS IN CV 29
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22



EDC ZONE AB-170-3

dd = Duct Destruction
ndd = No Duct Destruction

FIGURE 3B-37

PRESSURE TRANSIENTS IN CV 28
AUXILIARY BUILDING
HIGH ENERGY LINE BREAK ANALYSIS

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT
REVISION 22

APPENDIX 3C

FAILURE MODE ANALYSIS
FOR PIPE BREAKS AND CRACKS

APPENDIX 3C

FAILURE MODE ANALYSIS FOR PIPE BREAKS AND CRACKS

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3C.1 GENERAL

This appendix describes the specific pipe failure protection provided to satisfy the requirements of Section 3.6.1A and demonstrates that the essential systems, components, and equipment are not adversely affected by pipe breaks or cracks.

The information provided by this appendix is separated into three sections: 3C.2, a discussion of high-energy pipe breaks and the effects of pipe whip and jet impingement; 3C.3, a discussion of moderate-energy pipe cracks and the effects of spraying; and 3C.4, a discussion of flooding as a result of breaks or cracks.

Subcompartment pressurization is discussed in detail in Section 6.2.1.2 (for inside the containment) and in Appendix 3B (for outside the containment).

This appendix does not address the specific protection of field-routed essential instrument tubing or electrical conduit. However, these items are protected in accordance with the requirements of Section 3.6.1A.

For a detailed discussion of break/crack locations and types, break exclusion areas, guard pipes, and whip restraints which are frequently mentioned in this appendix, refer to Section 3.6.2A.

Environmental temperature and pressure parameters resulting from moderate-energy line cracks and high-energy line breaks are addressed in Section 3.11.

3C.2 HIGH-ENERGY PIPE BREAKS AND EFFECTS OF PIPE WHIP AND JET IMPINGEMENT

The following systems are described in the noted sections:

Main Steam Piping System, Including RPV Vent and MS Drain Piping	3C.2.1
Feedwater Piping System	3C.2.2
Reactor Recirculation	3C.2.3
RCIC and Connected RHR Systems	3C.2.4
LPCS/HPCS System	3C.2.5
LPCI Mode of RHR	3C.2.6
RHR System	3C.2.7
RWCU System	3C.2.8
RCIC Head Spray	3C.2.9
3-In and Smaller High-Energy Piping	3C.2.10

Each section references appropriate isometric drawings with break location and restraints. In addition, composite drawings showing pipe/equipment/room configurations have been provided in Section 3.6A, but are not specifically referenced.

The only pipe breaks of concern in the non-Seismic Category I turbine building are those with the potential to have an impact on safety-related equipment in adjacent buildings. Although the reactor protection SCRAM sensors for the turbine stop and control valves are located in the turbine building, they are not considered essential.

Only the essential jet impingement targets have been mentioned in the following sections. Nonessential targets, such as structural targets, have been evaluated to ensure structural integrity in order to isolate and mitigate jet impingement effects.

An evaluation has been performed also to verify that the plant can be safely shut down considering the effects of jet impingement from longitudinal cracks in main steam or feedwater piping in the break exclusion area of the main steam tunnel. Cracks in this portion of piping are considered highly unlikely due to the quality of material and quality assurance requirements specified for the fabrication and installation of this piping. In addition, the stress criteria for no postulated breaks as discussed in Section 3.6A have been met. The potential jet impingement targets in this area were identified and assumed to fail to function due to the jet effects. Also, a structural evaluation was performed to verify that structural integrity was maintained considering the effects of jet impingement, pressure, and flooding in this area.

3C.2.1 Main Steam Piping, Including RPV Vent and MS Drain Piping

The locations of postulated pipe breaks and pipe whip restraints for the three systems are shown on Fig. 3.6A-12 through 3.6A-16 for main steam, Fig. 3.6A-14a for RPV vent, land Fig. 3.6A-33b-1 through 3.6A-33d for MS drain. The results of the associated stress calculations are summarized in Tables 3.6A-1 through 3.6A-8, Table 3.6A-4a, and Tables 3.6A-17a through 3.6A-17c for the three systems, respectively.

General

Each of the four 24-in main steam lines is welded to the appropriate reactor nozzle (el 155 ft - 0 1/2 in) above the top of the shield wall. After the first elbow, each line runs downward to an approximate el 129 ft - 0 in, and then horizontally through the drywell and containment penetrations, the auxiliary building steam tunnel, and then into the turbine building. The portion of each line from the drywell wall to just before the second isolation valve is fully enclosed in a guard pipe.

The "break exclusion zone," as described in Section 3.6.2.1.5-2.lA, Item 2, starts inboard of the moment-limiting restraint adjacent to the inboard isolation valve and extends just beyond the jet impingement wall of the auxiliary building steam tunnel which contains the outboard moment-limiting restraint. The safety classifications of the various portions of the main steam piping are given in Table 3.2-1.

A total of 16 safety/relief valves are mounted on the horizontal runs between the reactor and the first isolation valves inside the drywell. The discharge piping from these valves is normally unpressurized; the pipe whip of this piping is due solely to the jet thrust resulting from the pipe breaks postulated at the connection of the main steam lines and the SRV lines and has been taken into account in this section.

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In addition, an 8-in line branching from main steam line A supplies steam to the RCIC turbine. This line

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passes through the steam tunnels and is discussed in the analysis of the RCIC system.

For the RPV vent system, a 4-in line leaves the RPV head at el 173 ft-0 5/16 in and reduces to a 2-in line before traveling downward through the refueling seal support penetration. At el 156 ft-8 in, a 2-in line branches off this line and travels in a northwesterly direction to the normally closed valve F002. The line from the RPV head keeps running downward and eventually joins the vertical leg of the main steam line A at el 148 ft-6 1/2 in.

Two-in main steam drain lines begin from each of the horizontal legs of the four main steam lines at el 127 ft-9 7/16 in inside the drywell and join to form a 3-in line. The 3-in line goes down through the platform at el 120 ft-9 in and drops to an elevation of 115 ft-7 1/16 in before traveling eastward to the inner isolation valve F016 and through the drywell and containment penetrations, into the auxiliary building steam tunnel. In the auxiliary building it travels through two more valves before going through the jet impingement wall. The line then joins with four other drain lines from the main steam lines in the auxiliary building and becomes a 3-in line before running into the turbine building. Four other 1 1/2-in drain lines from the main steam isolation valves in the steam tunnel travel through isolation valves before passing through the jet impingement wall and joining to form a 3-in line which runs through to the turbine building.

Inside Drywell

The main steam piping, if allowed to whip, can impact targets such as the 8-in RCIC piping, the 10-in LPCI piping, 10-in LPCS piping, 10-in HPCS piping, feedwater piping, service water piping, structural steel at various elevations, the drywell wall, and the primary shield wall.

To preclude any likelihood of loss of a system, or structural integrity of the drywell, required for a safe plant shutdown, a total of 22 restraints have been installed inside the drywell for the main steam system.

RCIC piping is protected by Restraint lMSS-PRR-813.

The LPCI piping is protected by Restraints IMSS-PRR-804 and 822.

The LPCS piping is protected by Restraints lMSS-PRR-812 and 813.

The HPCS piping is protected by Restraints lMSS-PRR-821,831, and 833.

The feedwater piping is protected by Restraints lMSS-PRR-802, 805, 814, 821, 822, 824, and 834.

The service water piping is protected by Restraints lMSS-PRR-801, 811, 821, and 831.

The drywell wall is protected by Restraints lMSS-PRR-801, 804, 805, 811, 813, 821, 824, and 831.

The primary shield wall is protected by Restraints lMSS-PRR-803, 813, 814, 823, 824, 833, and 834.

Restraints lMSS-PRR-806, 815, 826, and 835 are to protect the MSIV and to ensure that the stress allowables are within the limits as defined for the break exclusion region.

Essential targets that a jet discharging from a ruptured main steam line could impact include other main steam lines and supports required for safe shutdown and containment isolation. Since these target lines are the same size as the ruptured line, the jet would not affect the safety function of the main steam system. Other piping systems impinged by the jet and required for safe shutdown are the CSH and RHS lines and their respective supports. The impingement of these lines was found to be acceptable since alternate systems were available to meet safety requirements. For the RPV level and pressure instrument lines required for safe shutdown and being impinged by the jet, additional jet impingement restraints were incorporated in the plant design to ensure that the lines could withstand the impact loads and perform their safety function. All essential field-routed small bore piping has been evaluated for the effects of jet impingement.

Essential conduit targets impinged by a jet include conduits associated with resistance temperature detectors (RTDs) required for-containment monitoring systems (CMS) and those associated with automatic depressurization systems (ADS). Of the four RTDs associated with post-accident monitoring, the failure of one due to jet impingement was found to be acceptable, since this failure and a worst single active or passive failure will cause a maximum of three of the RTDs to be inoperative with at least one RTD remaining in service. In the case of conduits associated with the ADS, it was found that for any given main steam rupture event, a maximum of three of the seven valves involved were affected between pipe whip and jet impingement (leaving four valves available for safety functions), and this was found to be acceptable.

For the RPV vent line, four restraints have been added to preclude the piping from whipping into targets such as the RPV and drywell dome. Other targets that could be impacted are structural steel at various elevations and the refueling seal penetration, both of which have been evaluated to ensure their structural integrity, and the MSS line, which would not be damaged since it is larger than the whipping pipe. A ruptured vent line could also whip into conduits associated with RTDs whose failure is acceptable since they are not required for post-accident monitoring. Conduits associated with RTDs are also jet impingement targets and, again, their failure is acceptable since they are not required for post-accident monitoring. Other jet impingement targets include conduits for two valves required for the ADS. Failure of one of these valves would not affect the ADS since only six ADS valves are required even with a single failure of HPCS. The conduit of the other ADS valve has been evaluated to ensure that it is capable of withstanding the jet loading.

Potential targets such as the drywell wall, RCIC and FWS isolation valves, and the RCIC and DTM containment penetrations are protected from an unrestrained whipping drain line by a total of eight restraints and their supporting structural steel. Other pipe whip targets include structural steel at various elevations, which has been evaluated to ensure structural integrity, and RCIC and FWS lines and an FWS check valve, none of which would be damaged since the targets belong to a piping system with lines larger than the whipping line.

Essential jet impingement targets for the drain line include conduits for RCS and CCP valves; these are acceptable since the valves of these essential systems are not required for safe shutdown. Another essential conduit target is the conduit for an RHS valve required for containment isolation. Failure of this valve to close is acceptable since piping inside the containment that is associated with this penetration will remain full of water from the RPV and thereby provide a water seal. Other essential jet impingement targets are MSS and ICS valves, conduits for an ICS valve, and an ICS line, all of which are required for containment isolation and which are acceptable since an analysis of the systems indicates that the system requirements can still be met after the rupture event.

Inside the Steam Tunnels

The main steam piping, from the moment-limiting restraint inboard of the first isolation valve (inside the drywell) to and including the moment-limiting restraint at the jet I impingement wall, meets the stress criteria for no postulated breaks, as discussed in Section 3.6A.

The four zero-gap restraints provided for the drain lines primarily protect the isolation valves and the break exclusion zone from the potential whipping of the lines.

In the Auxiliary Building

From the steam tunnel, the four 24-in main steam lines (MSL) (A,B,C, and D) enter the auxiliary building at the center of the north wall at approximate el 128 ft-0 in. Lines B and C drop to an elevation of 115 ft-0 in (line C is a mirror image of line B, and line A is a mirror image of line D). MSLs A and C run along the perimeter of the western half of the auxiliary building, while B and D run along the perimeter of the eastern half of the auxiliary building until they meet at the center of the south wall, where lines A and D drop to the elevation of approximately 114 ft-0 in. From this point all four lines run south into the turbine building.

Pipe whip of the MSLs in the auxiliary building has been precluded by the placement of restraints. Restraints lMSS-PRR-902 (zero gap), 903, and 904 (omnidirectional) keep the northern portion of line C from whipping in the auxiliary building. Restraints lMSS-PRR-922 (zero gap), 1923, and 924 (omnidirectional) do the same for line B. Similarly, restraints lMSS-PRR-912 (zero gap) and 913 (omnidirectional) for line A and lMSS-PRR-932 (zero gap) and 1933 (omnidirectional) for line D are provided for the same purpose.

Bumper or omnidirectional restraints are provided at the elbows of the main steam piping in the four corners of the auxiliary building to prevent damage to the walls due to pipe whip. Strap restraints are provided to prevent whipping of the southern portion of the MSLs into the center of the auxiliary building.

A total of five zero-gap moment-limiting restraints have been installed adjacent to the jet impingement wall, outside the containment, on the four drain lines running in the steam tunnel area and on the 3-in DTM line in the auxiliary building. These restraints protect the break exclusion area from the impact of a ruptured pipe as well as keep stresses within acceptable limits in the break exclusion zone.

Essential targets for a jet discharging from a ruptured main steam line primarily are conduits for valves of the following systems: MSS, FWS, DTM, penetration valve leakage control, and main steam line isolation valve seal. However, since these particular valves in these systems are not required for safe shutdown or break isolation, failure of these targets is acceptable.

The potential targets that could be impacted by a whipping drain line, either in the steam tunnel or auxiliary building, consist primarily of piping lines and their valves. However, a review of the targets in question revealed that their failure was acceptable since none of these particular portions of the essential systems were required for either safe shutdown or break isolation. Other targets include walls and floors, all of which have been designed to ensure their structural integrity.

Essential targets impinged by a jet discharging from a drain line in either the steam tunnel or auxiliary building are essentially conduits serving area temperature monitors and MSS and FWS isolation valves. Since failure of the area temperature monitors will automatically trip the reactor protection system (fail safe) and since the area temperature monitors are not required to isolate the subject break, their failure is acceptable. A detailed review of the valve targets in question revealed that their failure was acceptable since these particular portions of the essential systems were not required for safe shutdown and break isolation.

Turbine Building

There are no breaks postulated in the turbine building because there are no essential targets in the turbine building.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the MSLs can cause additional damage which could impair the ability to safely shut down the reactor, or which could increase the offsite radiation effects beyond the limits of 10CFR50.67. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

3C.2.2 Feedwater Piping System

The piping, break locations, and restraints are shown on Fig. 3.6A-17 and 3.6A-18. The stress analyses used to determine the break locations are summarized in Tables 3.6A-9a, 3.6A-9b, 3.6A-10a, and 3.6A-10b.

General

From the turbine building, each of the two 20-in feedwater lines enters the auxiliary building steam tunnel, passes through the containment steam tunnel (enclosed by guard pipes in the tunnel) into the drywell at el 122 ft-0 in where each branches into two 12-in risers and passes through the RPV shield wall to the RPV nozzles at el 142 ft-3 1/2 in.

The break exclusion zone includes the moment-limiting restraint just inboard of the inner isolation valve and extends through the guard pipe and the auxiliary building steam tunnel, to and including the moment-limiting restraint located in the jet impingement wall of the auxiliary building.

Inside the Drywell

Postulated breaks could presumably result in a feedwater pipe whipping into several targets. The targets include CRD, ICS, RHS, SVV, FWS, and MSS lines, MSS isolation valves, structural steel at various elevations, the primary shield wall, and the drywell wall.

To preclude damage caused by a whipping feedwater pipe, a total of 22 restraints have been installed on the feedwater system inside the drywell.

All restraints on the feedwater system, except four, are omnidirectional restraints. 1FWS*PRR-811 and -831 are moment-limiting zero gap restraints that keep stresses within acceptable limits in the isolation valves and break exclusion zones, and 1FWS*PRR-810 and -830 are omnidirectional restraints.

Essential targets impinged by the jet from a ruptured feedwater line include a number of conduits for essential items. Failure of a conduit for a hydrogen igniter is acceptable since this failure and a worst single active or passive failure will not affect the safety function of this system. Conduits for the RPV level and pressure instrumentation are also affected; a few of them are acceptable since they are either not associated with ECCS or, even if they are associated with ECCS instrumentation, they are not required to automatically initiate ECCS. For the remaining RPV level and pressure instrumentation conduits, jet impingement restraints have been incorporated in the plant design to ensure that the lines would withstand the jet impingement loading. Failure of conduits and SVV lines serving ADS valves is acceptable since only two of the seven ADS valves, if HPCS is available, and three of the seven valves, if HPCS is unavailable, are required for safe shutdown. Conduits for CRD position indicator probes are other conduits affected by the jet for which jet impingement restraints have been designed so the line could withstand the loading. Targets required for plant safe shutdown and designed or analyzed to withstand the impingement loading include CRD bundles and a conduit for a termination cabinet serving the RCS, DTM, ADS, and CMS lines. Failure of the CMS line required for safe shutdown is acceptable since the drywell sample flow would still be maintained after the rupture event. Targets which are required for containment isolation and which have been analyzed to ensure that system requirements could still be met after the rupture event include RHS and WCS lines and supports, ICS and DTM lines, and MSS and ICS isolation valves. Essential jet impingement targets also include SVV lines and a tank associated with ADS valves, and LPCI lines, all of which are required for safe shutdown. Their failure is acceptable since alternative systems are available to shut down the plant, even if loss of site power and a single active failure were considered to be coincident with the rupture event. Main steam lines and supports also impinged by the jet would not be damaged since they are larger than the ruptured pipe.

All equipment inside the drywell, the operation of which during or after a LOCA is required for safe shutdown, is qualified for the post-LOCA drywell environment as discussed in Section 3.11.

Inside the Steam Tunnel

All feedwater piping from inboard of the first moment-limiting (zero gap) restraint in the drywell to outboard of the jet impingement wall meets the criteria for no postulated breaks as discussed in Section 3.6A.

In the Auxiliary Building

The reactor vessel water is protected from blowdown, following a postulated rupture of the feedwater piping outside the containment, by check valves 1B21*FO10A, B inside the containment and by testable check valves 1B21*AOVF32A, B outside the containment. Breaks are not postulated in the piping between the valves because that region is classified as a break exclusion area. Analyses were performed to demonstrate that the feedwater isolation check valves can perform their function following a postulated pipe break of the feedwater line outside the containment.

The reverse flow caused by the sudden pressure reduction at the break rapidly closes both valves. A dynamic analysis was performed to obtain the forcing function for use in the valve stress analysis. First, a flow transient analysis was performed for the feedwater system to simulate the pipe break condition. The reverse flow condition at the check valve location was determined using the SWEC computer program WATHAM (Section 3A.21). Hydrodynamic torque exerted on the valve disk by the reverse flow was applied to determine the valve closing time and the disk impact speed on its seat.

A stress analysis was conducted to determine the ability of these isolation valves to withstand impact of the disk on the seat, at the speeds obtained from that dynamic analysis. The acceptance criterion is that gross leak rates do not occur because of disk rupture, serious fracture of the seat/disk interface, or misalignment of the disk.

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An inelastic analysis was performed in accordance with Appendix F of the ASME III Code (1977) for Class 1 service, using the ANSYS computer program (Appendix 3A, Section 3A.2S). The non-linear stress/strain relationship was conservatively approximated by a bilinear curve with the strain at ultimate stress equal to 2/3 the elongation at temperature as provided in ASME II, adjusted for strain rate and temperature effects. This analysis has verified that structural integrity of the feedwater check valves is maintained. Note that, as discussed in Section 1S.6.6, a feedwater line break outside containment is less limiting than other postulated LOCAs.

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From the drywell, the two 20-in feedwater lines enter the auxiliary building from the north side of the north-south centerline (steam tunnel) at approximate el 121 ft-6 in, and then drop vertically to approximate el 109 ft-0 in. A 90-deg elbow directs them horizontally west along the north auxiliary building wall. At a mean distance approximately 13 ft from the west wall the lines turn south, cross the auxiliary building, and enter the turbine building at approximate el 108 ft-4 in.

Four circumferential and two longitudinal breaks at four breakpoint locations have been postulated for each line. Pipe whip targets for these breaks include RHS, MSS, WCS, FWS, SAS and CNS lines, RHS and WCS valves, the auxiliary building wall, and structural steel at various elevations. All of these targets have been protected by proper installation of pipe whip restraints.

Restraints 1FWS-PRR-901 and 911 are moment-limiting restraints to keep stresses within acceptable limits in the break exclusion zone.

Bumper type restraints 1FWS-PRR-903 and 913 keep the feedwater lines from deflecting downward.

Restraints IFWS-PRR-904 and 905 protect the north and west auxiliary building walls, as well as the CNS and SAS lines, from lateral displacement of the feedwater lines.

Restraints 1FWS-PRR-902 and 912 (omnidirectional) prohibit horizontal pipe whip of the vertical sections of feedwater piping. Restraint 1FWS-PRR-914 (omnidirectional) prohibits the east-west translation of feedwater pipe 1FWS-020-31-4.

The possible essential targets that could be impinged by a jet from a ruptured feedwater line essentially consist of conduits serving MSS, FWS, and DTM valves. Since this portion of the essential system is neither required for safe shutdown nor isolation of the break, failure of these conduits is acceptable.

Conclusions

Because of spatial separation and the installation of pipe whip restraints, no postulated failure of this piping can prevent the safe shutdown of the plant.

3C.2.3 Reactor Recirculation

For the recirculation system, the locations of the postulated breaks and restraints are shown on Fig. 3.6B-4. The results of the associated stress calculations are summarized in Table 3.6B-3.

General

Each of the two reactor recirculation loops leaves the RPV at el 115 ft-10 1/2 in as a 20-in line, drops vertically to el 84 ft-4 1/2 in, turns horizontally through the suction isolation valve F023A (or F023B for loop B), and turns up into the pump suction port. The 20-in pump discharge line runs horizontally at approximate el 89 ft through the flow control valve F060A (or F060B) and the isolation valve F067 and turns up to el 108 ft-3 1/2 in where it joins the C-shaped 16-in horizontal header. From this header, five 12-in risers go up and enter the RPV at el 116 ft-3 1/2 in. In addition, from loop B only, an 18-in RHR suction line branches off from the vertical run between the reactor outlet and valve F023B at el 91 ft-6 in, turns up through the normally open valve FOIO and normally closed valve F009, and turns out to leave the drywell and containment at el 116 ft-10 in.

GE is responsible for the location and design of restraints for the recirculation system.

Recirculation Loop A

A total of ten restraints have been installed on this loop to prevent the whipping of the piping in the event of a rupture. A restraint has been installed on the vertical leg of each of the six risers to limit the travel of the ruptured pipe radially from the RPV. Four restraints have been installed on the header to limit both radial and downward travel.

If the pipe were to whip totally unrestrained, the possible targets would include primary shield wall, RDS tube bundle, and HVAC ducting. Essential jet impingement targets for a break in the RCS line include the MSS lines required for containment isolation, but since the target lines are larger than the ruptured line, the jet would not affect the safety function of the main steam system. For the RPV level and pressure instrument lines which are impinged by the jet and are required for safe shutdown, additional jet impingement restraints were incorporated in the plant design to ensure that the line could withstand impact loads and perform its safety function. Conduits for hydrogen ignitors are also impinged by jets. However, this failure is acceptable since even with this failure and a worst single active or passive failure either the safety function of this system will not be affected or the distance from any given point in the drywell to an unaffected hydrogen ignitor does not exceed 30 ft. Other essential lines impinged by a jet from an RCS break and required for either containment isolation or safe shutdown include the RDS, ICS, and DER lines. While alternate safety systems were available for some of these systems, an analysis of the others was done to ensure that they could withstand impact loads. All essential field-routed small bore piping has been evaluated for the effects of jet impingement.

Recirculation Loop B

The restraint locations are equivalent to Loop A except that an additional two restraints have been installed just above and below the tee where the RHR suction line joins in.

The potential pipe whip targets and jet impingement targets essentially remain similar to those in Loop A.

3C.2.4 RCIC and Connected RHR Systems

The locations of postulated pipe breaks and restraints for both systems are shown on Fig. 3.6A-12 and Fig. 3.6A-19. The results of the associated stress analyses are summarized in Tables 3.6A-lla and 3.16A-12.

General

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An 8-in branch line from main steam line A delivers steam to drive the RCIC turbine.

Inside the drywell, the line starts at el 146 ft-3 3/4 in on the vertical leg of main steam line A, azimuth 72 degrees; and follows the outer radius of the RPV shield wall horizontally to azimuth 0 degree; then drops to an elevation of 125 ft-6 3/4 in and goes horizontally through the inner isolation valve F063 and through the drywell and containment penetrations into the auxiliary building steam tunnel. The steam piping in the containment building steam tunnel is enclosed in a guard pipe.

Inside the auxiliary building, after the line passes through the outer isolation valve (FO64) and the jet impingement wall, the line branches into two 8-in lines that at one time supplied the two RHR heat exchangers. One of the 8-in lines runs east through the steam tunnel wall at approximate el 116 ft-0 in to a blank flange while the other runs west through the steam tunnel wall at approximate el 116 ft- 4 in to a blank flange. The lines are classified as high energy up to these flanges, while beyond the flanges,

12←•

to the RHR heat exchangers, they are classified as moderate energy.

A 4-in line supplying the RCIC turbine branches off the 8-in line running west, drops down through the tunnel floor (el 114 ft-0 in) and again through floor el 95 ft-9 in to an elevation of 73 ft-9 1/4 in and then runs horizontally to the normally closed valve F045. Up to this valve the line is classified as high energy while from the valve to the RCIC turbine, it is classified as moderate energy.

From the moment-limiting device adjacent to the inner isolation valve F063, through the guard pipe and outer isolation valve F064, up to the moment-limiting device adjacent to the jet impingement wall, the piping meets the stress criteria for no postulated breaks, as discussed in Section 3.6A.

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Also, the 6-inch RCIC injection line is routed to the 20-inch Feedwater A loop by tapping into the 10-inch RHS shutdown cooling mode return line to Feedwater inside the Main Steam Tunnel just south of the Jet Impingement Wall.

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Inside the Drywell

For stress analyses, the RCIC steam line within the drywell was modeled as a branch of the main steam piping. Due to the postulated breaks, the line could potentially whip into several targets, including the unit cooler, MSS, FWS, and ICS restraints, FWS and ICS lines and supports, and the shield wall.

To preclude the damage that could be caused by the whipping RCIC line, a total of six restraints have been installed along the RCIC lines inside the drywell. All of these restraints are omnidirectional except for restraints PRR-805 and PRR-806 which are moment-limiting (zero gap) restraints to keep the stress within acceptable limits in the isolation valve and the break exclusion zone.

Essential jet impingement targets for the ICS piping system include conduits for the RCIC isolation valve and main steam safety relief valves. Since the conduits are capable of withstanding jet impingement loads, the safety function of these targets would not be affected. A conduit for a hydrogen ignitor is also impinged by a jet, but the failure of one ignitor is acceptable since this failure and a worst single active or passive failure will not affect the safety function of this system. For the RPV level and pressure instrument line impinged by the jet, jet impingement restraints have been incorporated to ensure that the line could withstand the load and perform its safety function.

Inside the Steam Tunnel

All RCIC piping, from inboard of the first moment-limiting (zero gap) restraint in the drywell to outboard of the second moment-limiting (zero gap) restraint in the auxiliary building, meets the stress criteria for no postulated breaks, as discussed in Section 3.6A.

Inside the Auxiliary Building

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For stress analyses, the RHS line between the two blank flanges on the 8″ branch line was modeled together with the RCIC piping.

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The only postulated breaks for the 4-in line entering the auxiliary building are at each of the terminal ends, and any whipping of the pipe is precluded by the two restraints on each leg.

Due to the postulated breaks, the 8-in line could potentially whip into the ICS and WCS lines and the jet impingement wall, and this is precluded by Restraint PRR-914.

For the 4-in line, which branches vertically downwards from the 8-in line, apart from the terminal end breaks, three other breaks are postulated at the elbows. In this instance, however, since the potential structural and piping targets have been designed to withstand the pipe whip loading, restraints are not required.

The essential conduit targets included conduits associated with an ICS containment isolation valve which was impinged by a jet discharging through a penetration hole in the jet impingement wall. The conduit was protected by providing a shield at the jet impingement wall.

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For the 6-inch RCIC injection line, a circumferential break is postulated at the high/moderate energy line interface at the RCIC check valve interface with the high energy RHS line. The following are the potential HELB targets from this pipe break and their evaluations: (1) 24″ MSS piping, 20″ FWS piping and valve— These targets are acceptable because the targeted piping sizes are larger than the ruptured pipe, (2) 4″ WCS piping, valve, and supports, 2″ WCS tank drain line—- These targets are acceptable since they are non-essential for plant shutdown and the combined HELB flow area of the failed piping and the subject RCIC HELB is bounded by the 20″ Feedwater double ended rupture in this volume, (3) steel platform/grating at el 124′-9″—- Failure of this potential target is acceptable because no essential systems can be adversely affected by the failure of this platform, (4) Main Steam Isolation Valve Seal System valve E33-VF303B and inlet line, and cables and conduits for FWS valve—- These targets are not required for safe shutdown or to isolate the break, (5) Leak Detection system thermocouples 1LDS*RTD2A & 2B—- These are non-essential items whose failure will annunciate alarms in the main control room.

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3C.2.5 LPCS/HPCS System

These systems do not operate during normal plant operation, and hence only a small portion of the piping which is normally exposed to reactor pressure is classified as high energy.

The postulated pipe break locations and restraints for the LPCS and HPCS systems are shown on Fig. 3.6A-22 and 3.6A-21, respectively. The results of the associated stress analyses are summarized in Tables 3.6A-14 and 3.6A-13, respectively.

LPCS

The 10-in LPCS piping is attached to the RPV nozzle at el 140 ft-9 in and azimuth 90 degrees. After passing through the primary shield wall, the line passes through the locked-open valve F007 and normally closed check valve F006, at which point the high-energy portion ends. The line beyond that point is classified as moderate energy.

Apart from the circumferential breaks at each terminal end, two intermediate breaks at each end of the locked-open valve F007 have also been postulated on the horizontal leg of the line.

Unrestrained whipping of dead-end piping would impact the drywell wall and structural steel at various elevations and affect the drywell wall penetration. To avoid such an occurrence, a restraint has been installed on the vertical leg of the piping, just beyond check valve F006. Pipe whip of dead-end piping is due solely to the jet thrust resulting from the pipe breaks postulated on the high-energy portion of the line.

Essential targets that a jet discharging from a ruptured LPCS line could impact include conduits for ADS valves, RPV level and pressure instrument tubing, and an RHS line required for safe shutdown. In the case of ADS valves, since the break has the equivalent flow area of approximately two and one-half times the flow area of a safety relief valve and since HPCS is available, failure of these three ADS valves is acceptable. Failure of instrument tubing would be acceptable since it is not associated with ECCS; and, since alternative safety systems are available, failure of the RHR line would be acceptable as well. A conduit for a hydrogen ignitor is also impinged by a jet, but the failure of one ignitor is acceptable since this failure and a worst single active or passive failure will not affect the safety function of this system. Other essential jet impingement targets are main steam line supports and isolation valves on the 1PCS line; they are acceptable since an analysis of the systems indicates that system requirements could still be met after the rupture event.

HPCS

The 10-in HPCS piping is very similar to the LPCS piping discussed above, except that it is located at azimuth 270 degrees. The two valves it runs through are the locked-open valve F036 and the normally closed valve FOO5. Targets similar to those on the LPCS line are protected by a similar restraint. Essential jet impingement targets include the HPCS line and supports, a valve on the HPCS line, and MSS supports, all of which are required for containment isolation. However, an analysis of these systems indicates that system requirements could still be met after the rupture event; hence, these targets are acceptable.

3C.2.6 LPCI Mode of RHR

The LPCI systems are not in use during normal plant operation, and, as a result, only that portion of the piping that is pressurized is classified as high energy.

The postulated pipe break location and restraints for the LPCI function of the RHR system are shown on Fig. 3.6A-25. The results of the associated stress analyses are summarized in Table 3.6A-16.

LPCI A

From the RPV nozzle, el 136 ft-3 1/4 in, azimuth 45 degrees, the line passes through the primary shield wall and through the locked-open valve F039A and normally closed check valve F041A. The line is classified as high energy form the RPV nozzle up to check valve F041A and as moderate energy beyond that.

In addition to the circumferential breaks postulated at each terminal end, breaks have also been postulated at either end of locked-open valve F039A. Due to postulated breaks, the lines could potentially whip into the drywell and primary shield walls, containment penetration, CSL and SVV lines, and structural steel at various elevations. To protect these targets and to preclude the possibility of pipe whip, PRR-801 has been installed on the vertical leg of the piping, just beyond check valve F041A.

Essential jet impingement targets for the LPCI system include conduits and piping associated with ADS valves. Since the breaks have a flow area of approximately two and one-half times the flow area of a safety relief valve and since HPCS is available, failure of three of the ADS valves due to jet impingement is acceptable. Conduit targets associated with ADS valves, other than the three whose failure was acceptable, were analyzed for jet impingement loads to ensure their structural integrity and found to be acceptable. Other targets include an RHS line and supports in the LPCS system. Even though the line is part of the

ECCS, failure of the 3/4-in line will not degrade the ECCS function. RCS lines associated with RPV level and pressure instrumentation are also impinged by jets; this is acceptable since they are not associated with ECCS. Essential targets required for containment isolation include the MSS and RHS lines and supports, CSL lines, conduits for various containment isolation valves, and MSS and LPCI valves. However, all these targets are acceptable since an analysis of the systems indicates that system requirements could still be met after the pipe rupture event.

LPCI B and C

The piping for Loops B and C is very similar to Loop A described above. The major differences are the RPV nozzle azimuth is 225 degrees and 135 degrees for Loops B and C, respectively; and the locked-open valves and the normally closed valves the lines run through are F039B and F041B for Loop B and F039C and F041C for Loop C. For Loops B and C, in addition to the circumferential breaks postulated at each terminal end, circumferential breaks have also been postulated at one end of the locked-open valve and at the location where the branch line joins the 10-in line. For both loops, the potential targets due to pipe whip caused by these breaks include the drywell wall and SAS lines. Restraints equivalent to those in Loop A have been provided to protect these targets and preclude the possibility of pipe whip.

Essential jet impingement targets for Loop B include RPV level and pressure instrumentation lines and LPCI-B piping, supports, and valves. Failure of the instrumentation lines is acceptable since they are not associated with ECCS; and, no failure consequence needs to be considered for impingement of the LPCI-B piping system since the rupture line impacts its own system.

For Loop C, the essential jet impingement targets include a conduit for ADS valves and LPCI-C piping, supports, and valves. The conduit has been designed to withstand the jet impingement loading; and, no failure consequence needs to be considered for the LPCI-C piping system since the rupture line impacts its own system.

3C.2.7 RHR System

The locations of postulated pipe breaks and restraints are shown on Fig. 3.6A-24 and 3.6A-25a. The results of the associated stress analyses are summarized in Tables 3.6A-15 and 3.6A-16a.

General

Inside the drywell, an 18-in line branches off the vertical leg of recirculation system Loop B at el 91 ft-6 in and rises vertically, passing through two valves. The line up to the normally closed valve F009 is classified as high energy and is classified as moderate energy past the valve.

In the auxiliary building there are two separate sections of piping classified as high energy. Two 10-in lines go through normally closed valves, join with 4-in WCS lines, and end in feedwater Loops A and B. The sections of piping between the normally closed valves F050A and F050B and the feedwater lines are classified as high energy.

Inside Drywell

If the pipe were to whip totally unrestrained, it could conceivably damage the drywell wall and floor framing steel. To preclude such an occurrence, a restraint has been installed on the vertical leg of the piping, just below the two valves. Other targets that could be impacted by a whipping line include RCS Loop B piping and the feedwater pipe rupture restraint PRR-811 structure. The RCS piping would not be damaged by the impact since it is larger than the whipping line, and the pipe rupture restraint structure has been evaluated to ensure that it could withstand the impact loading. Since the structural integrity of the restraint structure is maintained, the structure also acts as a barrier and prevents the pipe from whipping into an essential conduit for RHP, isolation valve F009, which is required for containment isolation.

Essential jet impingement targets include conduits for a hydrogen ignitor and valves required for the ADS, and the RPV pressure reference leg instrumentation line associated with ECCS. Failure of a hydrogen ignitor due to jet impingement is acceptable since this failure and a worst single active failure will not affect the safety function of the system. Failure of the ADS valves will not affect safe shutdown since only three ADS valves are required for a large break inside the containment. The instrumentation line has been protected by incorporating jet impingement restraints in the plant design such that the line would accept the jet impingement load. Other essential targets required for containment isolation include MSS, RHS, ICS, SVV, DER, and DTM lines, and RHR valves. They are all acceptable since an analysis of these systems indicates that system requirements could still be met after the rupture event.

RBS USAR

Inside the Auxiliary Building

Due to the postulated breaks, the lines could potentially whip into the main steam lines, but this would not cause any damage since the main steam lines are much larger than the whipping lines. The jet discharging from the lines could potentially impact the main steam lines, the WCS lines, and the walls of the steam tunnel, but all of these nonessential targets have been designed to ensure their structural integrity.

3C.2.8 RWCU System

The locations of postulated pipe breaks and of pipe whip restraints are shown on Fig. 3.6A-23 and 3.6A-26 through 3.6A-33a. The results of the associated stress calculations are summarized in Tables 3.6A-19 and 3.6A-20.

General

The reactor water cleanup system, which is classified as a high-energy system, could be considered as a loop from the reactor, through the heat exchangers, through the filter/demineralizers, back through the regenerative heat exchangers, and back to the reactor again. The system is not required for safe plant shutdown and, from the standpoint of piping failure, the only concern is the possible detrimental effects on other equipment.

One 4-in line from each reactor recirculation loop and a 3-in line from the reactor lower head drain join into a 6-in line inside the drywell at el 90 ft-0 in. This line progresses upwards, leaving the drywell at el 116 ft-0 in. Enclosed in a guard pipe, this line passes through the drywell wall and the primary shield wall, entering the auxiliary building. Isolation valves are located near each end of this guard pipe. The line then drops down through the floor at el 114 ft-0 in and at el 106 ft-9 in divides into two 6-in lines, which in turn become 3-in lines and supply the two pumps located in separate rooms. From the pumps' discharge, the two 3-in lines join again into a 4-in line at el 106 ft-9 in, which goes up back into the steam tunnel and then through a penetration into the containment. In the containment building the line runs upwards through the floor, into the heat exchanger room, and then into the heat exchangers. After leaving the heat exchangers, the 4-in line splits into two 3-in lines at el 165 ft-9 in, and they run into the filter/demineralizers. After the filter/demineralizers, the two 3-in lines drop down and join again into a 4-in line at el 154 ft-3 in and return to the regenerative heat exchangers.

From there, the line drops to an elevation of 117 ft-9 1/2 in, runs through the containment penetration into the auxiliary building where it joins the RHS piping to the feedwater, and returns to the RPV. In every instance, when the line goes through the penetration, from the innermost zero-gap restraint in the containment building through the penetration, and up to the outer zero-gap restraint in the auxiliary building, the piping meets the stress criteria for no postulated breaks, as discussed in Section 3.6A.

Inside the Containment

Because of the arrangement of the piping, there is very little equipment that could be damaged by the impact of a ruptured pipe. Among the possible targets, the more significant include the weir wall, SVV, and DER lines. To prevent this potential damage, a total of 11 restraints have been installed, including six zero-gap restraints by the containment penetrations. Targets that are not protected by restraints and that have been evaluated to ensure their structural integrity include the RPV and pedestal, CRD housing, heat exchangers, cubicle walls, and structural steel at various elevations.

All RWCU piping, from inboard of the first moment-limiting (zero-gap) restraint in the containment to outboard of the moment-limiting (zero-gap) restraint in the auxiliary building, meets the stress criteria for no postulated breaks, as discussed in Section 3.6A.

Essential targets impinged by the jet produced by postulated rupturing of the pipe include conduits for various temperature elements inside the containment.

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Ambient temperature elements in each volume provide the automatic break-point isolation. In the case of conduits serving some of the leak detection temperature elements, either jet impingement restraints or shields were incorporated in the plant design to ensure the structural integrity of these conduits. Failure of the conduits for the RPV and RCS thermocouple temperature elements is acceptable since they are not required for safe shutdown, even though part of essential systems. Other jet impingement targets include WCS isolation valves and conduits for these valves. Failure of the isolation valve

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targets is acceptable because the break isolation can be provided by other redundant check valves. However, if break isolation is provided by valve 1G33*MOVFOO4, drywell flooding may occur; this is acceptable since it will not degrade safe shutdown. Other jet impingement targets required for containment isolation include WCS lines and supports, SVV lines, and RHS valves; they are acceptable since an analysis of these systems indicates that the system requirements could still be met after the rupture event.

Outside the Containment

Three zero-gap restraints have been installed outside the containment, adjacent to the penetrations, to protect the break exclusion area from the consequences of a ruptured pipe. Targets that could be impacted by a whipping line due to an RWCU piping break include RHS, ICS, and FWS lines and ICS restraints. However, in all the above instances, since the whipping line is smaller than the target line, the target cannot be damaged. Other pipe whip targets include various walls and floors, all of which have been structurally designed to withstand the pipe whip loading, and a ventilation duct that is not required for safe shutdown.

Even though targets impinged by a jet from a ruptured RWCU line include essential conduits leading to an RCIC fill pump motor and various valves of essential systems, a more detailed review revealed that these particular portions of the essential systems were not required for safe shutdown. Other targets affected by the jet include conduits for flow transmitters used to detect leakage. However, once a break occurs in a particular volume, the flow transmitters will not be required since area temperature monitors will detect and isolate the break. Hence, the failure of these targets is acceptable. The jet impingement targets also include conduits from area ambient temperature elements required for breakpoint isolation, but this is acceptable since, in this instance, the elements are not in the postulated breakpoint volume.

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3C.2.9 RCIC

No high energy line breaks associated with RCIC inside containment are postulated, and considerations of RCIC line rupture, pipewhip or targets are no longer applicable. However, the failure of the removed 6” RCIC head spray line and information that is part of the original design will be maintained as the bounding conditions.

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3C.2.1 3-In and Smaller High-Energy Piping

3C.2.10.1 Control Rod Drive Hydraulic System

The piping and break locations are shown on Figures 3.6A-24b and 3.6A-24c. The stress analyses used to determine the break locations are summarized in Tables 3.6A-18a and 3.6A-18b.

General

From the condensate storage tank, the CRD lines enter the fuel building and, after passing through two filters, connect to two drive water pumps. No breaks are postulated in this portion of the piping since it is not considered high-energy piping. The pressurized lines that leave the pumps go through two more filters before entering the containment building and the flow control station.

The high-energy portion of the supply piping that leaves the control station is comprised of the following four lines:

1. The charging line, which provides a constant flow of pressurized water to charge the scram accumulators in the hydraulic control units (HCUs).

2. The cooling line, which maintains proper cooling of the drive mechanisms by providing a bypass flow of water to each of the drives, via the HCUs, during normal operation periods when rod drive movement is not required.

3. The drive line, which supplies the HCU with the water required for rod positioning during normal operation of the system.

4. The exhaust line, which displaces excess cooling and exhaust water generated by normal drive motion to the RPV.

The supply piping emerges from the control station as a bundled group of various sized lines. This bundle of piping extends toward both the 90-deg and 270-deg side HCU banks. Upon reaching the HCU banks, each of the supply lines branches out over each bay of the HCUs and extends down into the HCUs.

The supply piping provides the necessary water and air for the proper functioning of the HCU during normal rod movement. The HCU provides the interface valving between the supply/exhaust piping and the insert/withdraw piping that operates the drives. The insert/withdraw pipings, starting from the HCU scram valves, are bundled in groups and enter the drywell wall through the penetrations approximately at el 130 ft at both the 90-deg and 270-deg sides. The pipe bundles drop to a lower elevation, extend toward the RPV, and enter the RPV through the CRD housing at approximately el 96 ft.

<u>Inside the Containment</u>

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Essential jet impingement targets for the CRD piping system are primarily conduits for essential items. Failure of the conduits for the containment ambient temperature monitoring RTDs is acceptable since this failure and a single active or passive failure will cause a maximum of six of nine RTDs to be inoperative, leaving a minimum of three RTDs in service. Failure of the conduits serving the suppression pool temperature monitoring RTDs is acceptable since failure of three RTDs due to jet impingement and a worst single active or passive failure will cause a maximum of 10 of 14 RTDs to be inoperative, leaving a minimum of four RTDs in service. Failure of conduits for the RPV level and pressure instrumentation panels is acceptable since this failure and a worst single active or passive failure will cause a maximum of three of four panels to be inoperative, leaving at least one panel in service. The above conduits also serve either HPCS or LPCI RPV level transmitters, and their failure is acceptable since, if the failed instrumentation is HPCS, then ADS with LPCS or LPCI and if it is LPCI, then ADS with LPCS and HPCS, would be available following this rupture event. Other conduits permitted to fail include

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conduits for the CRD hydraulic control unit (HCU) pilot scram solenoid valves. Their failure is acceptable since the valves will fail open, their fail safe position, on a loss of power. Conduits for radiation monitors are impacted also, and this is acceptable since they monitor radiation in the drywell during post-LOCA and this particular break is not associated with LOCA. Failure of conduits for the CRD scram discharge volume water level indicator is acceptable since this failure and a worst single active or passive failure will cause a maximum of four of five instruments to be inoperative, leaving at least one instrument in service. In addition, this instrumentation performs its safety function until a CRD scram is initiated, after which time the instrument is not required. Conduits for the CRD scram discharge valves are acceptable as targets since that portion of the essential system is not required for either safe shutdown or break isolation. Other targets whose failure is acceptable for the above reason include conduits for the main steam flow instrumentation panel, hydrogen mixing system valves, and SLC lines. Essential CRD valves impinged by the jet have been analyzed to ensure that they can withstand the jet impingement loading.
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The potential targets that could be impacted by whipping CRD lines are similar to the jet impingement targets previously mentioned. Since all these targets have been evaluated and found to be acceptable for jet impingement loads, the targets are considered to be acceptable for pipe whip loads as well. Other targets that the lines could impact include a column of a multifunctional support and some withdraw lines. Both targets have been designed and analyzed to ensure that they maintain their structural integrity after the rupture event.

<u>In the Fuel Building</u>

Because of the piping arrangement in the fuel building, the only targets that a whipping pipe could impact are the floors and walls of the building; however, these floors and walls have been designed to ensure their structural integrity.

Essential jet impingement targets required for plant safe shutdown include a conduit for a service water line flow transmitter. This flow transmitter monitors flow from the Division 2 standby service water pumps and flow into the standby service water system cooling tower. The failure of this target is acceptable since the operator could verify the flow by monitoring pump discharge pressure and pump motor run current, both of which are indicated in the control room. Other essential targets include certain cable trays providing power for the fuel building ventilation system fans which cool the spent fuel pool area. Failure of these cable trays is acceptable since repairs can be made in 4 hr. During this period, the spent fuel pool temperature will not increase to an unacceptable level.

3C.2.10.2 Standby Liquid Control (SLC) System

The SLC systems are not in use during normal plant operation and, as a result, only that portion of piping that is pressurized is classified as high energy.

The postulated pipe break locations for the SLC system are shown on Figure 3.6A-24a; the results of the associated stress analyses are summarized in Table 3.6A-20a.

General

From the RPV nozzle, el 102 ft-8 1/8 in, azimuth 225 deg, the 1 1/2-in line drops down, passes through the pedestal penetration at elevation 96 ft~-10 3/4 in, and through locked-open valve F008 and normally closed check valve F007. The line is classified as high energy from the RPV nozzle up to check valve F007 and as moderate energy beyond that.

Apart from the circumferential breaks postulated at the terminal ends, breaks have also been postulated at either end of locked-open valve F008.

Inside the Drywell

Due to postulated breaks, the only essential target the pipe could whip into is the CRD housing. However, since the CRD housing pipe is much larger than the ruptured line, the target would not be damaged. Hence, restraints are not necessary on this line. The jet emanating from the ruptured pipe would also impact the CRD housing, and this is acceptable for the same reasons given above.

3C.3 MODERATE-ENERGY PIPE CRACKS AND EFFECTS OF SPRAYING

3C.3.1 Discussion

The components and/or equipment required for safe shutdown of the reactor were evaluated for the effects of spraying from through-wall leakage cracks in moderate-energy systems. The evaluation demonstrates that the plant can be safely shut down, assuming a concurrent single active failure in systems necessary to mitigate the consequences of the postulated piping failure and shut down the reactor. Where necessary, measures will be provided to protect and ensure component operability. Flooding effects from cracks in moderate-energy systems are discussed in Section 3C.4.

Moderate-energy piping, as defined in Section 3.6.2.1.2A, includes piping systems where the maximum operating temperature is 200°F or less and maximum operating pressure is 275 psig or less. It also includes some systems that qualify as high-energy systems for short operational periods and moderate-energy for major operational periods.

Only high-energy piping is capable of producing breaks (Section 3.6.2.1.3A). Moderate-energy piping produces only through-wall leakage cracks. The most limiting moderate-energy piping crack, i.e., RHR system, produces environmental conditions as severe as high-energy breaks.

The criteria used to define the location of cracks in moderate-energy systems outside containment are defined in Section 3.6.2.1.5.2.2A, and the criteria for calculating crack flow rates are given in Section 3.6.2.1.6.3A.

3C.3.2 Evaluation Procedure - Spraying

The evaluation was conducted in accordance with NRC Branch Technical Position ASB 3-1, which states that a leakage crack in moderate-energy piping is considered separately as a single, postulated initial event occurring during normal plant conditions. The essential equipment that must operate under these conditions is that required to bring the plant to a safe shutdown condition and maintain long-term cooling. Fig. 3C.3-1 defines four pathways to hot shutdown and two pathways to long-term cooling of the reactor, including continued cooling of the spent fuel pool. The essential components making up these pathways (the targets) were located by environmental zones. The evaluation of effects of spraying from moderate-energy cracks proceeded in all environmental zones containing targets. Included in the evaluation were the reactor building, auxiliary building, fuel building, diesel generator building, control building, standby service water cooling tower, and piping and electrical tunnels. Excluded were the turbine, radwaste, normal switchgear, and other nonseismic Category I buildings. These nonseismic structures contain no equipment that is required to safely shut down the reactor, maintain long-term cooling, or maintain spent fuel pool cooling.

The following summary outlines the procedure used to evaluate spraying effects from moderate-energy cracks.

1. List by environmental zone all components and/or equipment (targets) required for safe shutdown in all buildings.

2. Evaluate all components and/or equipment to determine if they are waterproof (not susceptible to failure from spraying) and can withstand the effects of water temperature. Table 3C.3-2 shows the maximum spray temperatures in each building.

3. Identify water sources in environmental zones that contain potential spray-susceptible targets (cracks are not postulated for spray evaluation in zones without targets). If there is a water source in the zone, assume that all potential targets are sprayed. If there is no water source in the zone, evaluate the susceptibility of the equipment to failure as the result of dripping water from other zones.

4. Assume the failure of all targets in the zone that are not waterproof and identify available paths for safe shutdown and maintenance of long-term cooling. Fig. 3C.3-1 depicts the safe shutdown paths.

   If it is concluded through this evaluation that the plant could not be shut down safely, a more detailed approach is taken to determine if components are actually sprayed and rendered inoperable. Using this basis, a reexamination of paths for safe shutdown is then conducted.

5. The spraying evaluation is conducted in conjunction with a flooding evaluation (Section 3C.4). If a spray source in a given zone is large enough to cause potential flooding problems in the given zone (or other zones), failures from flooding are combined with failures from spraying to evaluate available safe shutdown equipment.

6. In addition to the direct consequences of pipe crack, a single active failure is assumed in those systems required to mitigate the consequences of the piping failure and shut down the reactor, in accordance with Section 3C.3.3.

3C.3.3 Evaluation Guidelines - Spraying

The basic guidelines used to evaluate the effects of spraying are as follows:

1. If a water pipe is within an environmental zone, all targets within that zone are assumed to be sprayed. If this assumption yields unacceptable results, a more detailed review of spraying and component shielding is conducted.

2. Qualification for spraying is determined by a review of component specifications and test data.

3. All Class 1E electrical components which have NEMA 4 (or equivalent) enclosures are not assumed to fail as the result of water spray.

4. Unit cooler and fan motors are not assumed to fail since they are enclosed within the unit cooler housing or ductwork, which shields them from direct spraying.

5. Cables and splices are waterproof and unaffected by water spray.

6. All junction and terminal boxes for safe shutdown equipment containing termination boards have NEMA 12 (or equivalent) enclosures and are not assumed to fail as a result of dripping water, but are assumed to fail from spray.

7. If the actions required to stop the flow of water from the crack cause additional safe shutdown equipment to become inoperable, these systems will be assumed to fail as a consequence of the postulated pipe crack.

8. If the postulated piping failure results in a reactor or turbine trip, loss of offsite power is assumed.

9. Guidelines for single-failure evaluation are as follows:

a. Plant shutdown is assumed to be a consequence of the pipe crack, and a single active failure is assumed in the safe shutdown systems.

b. Where the postulated piping failure is assumed to occur in one train of a dual-purpose, moderate-energy, safe shutdown system (e.g., safety-related RHR and service water are subsystems comprising such a safe shutdown system; SFC and safety-related chilled water are other examples of such systems), a single failure is not postulated in the redundant safety-related train of that system or subsystem. Dual purpose systems are identified as discussed in Paragraph B.3.b (3) of NUREG-0800, Branch Technical Position ASB 3-1.

10. In determining alternate paths to safe shutdown, credit was taken for all available systems (as defined by the above criteria).

3C.3.4 Analytical Methods

As described in the spraying evaluation procedure (Section 3C.3.2), all targets in a given zone were assumed to be sprayed by any water sources in the zone. Analytical calculations of spraying distance were not utilized in reevaluating problem areas. In these instances, shielding, moving equipment, and other modifications were considered.

3C.3.5 Results of Evaluation - Spraying

The following subsections present the results of the spraying evaluation building-by-building using the procedures and guidelines discussed in Sections 3C.3.2 and 3C.3.3.

The evaluation verifies that the plant can be safely shut down in the event of pipe cracks in fluid systems. As noted below, protective measures ensure that required system functional capability is maintained. A list of moderate-energy piping systems and system parameters is provided in Tables 3C.3-1 and 3C.3-2 for those buildings housing equipment required for safe shutdown.

3C.3.5.1 Reactor Building (Including Drywell, Containment, and Annulus)

In the reactor building, all safety-related targets required for safe shutdown have been qualified for spray. All junction boxes and cable terminations supporting these targets have spliced connections which do not fail from spray.

3C.3.5.2 Auxiliary Building

In the auxiliary building, spray sources include both safety-related and nonsafety-related systems. Components susceptible to failure from spray are motors and motor control centers for RCIC, HPCS, RHR, and LPCS system pumps. A single spray source will not affect more than one of these pump motors. Failure of an RCIC, HPCS, or LPCS motor is acceptable; sufficient redundancy exists to safely shut down the plant when considering an additional single active failure as described in Section 3C.3.3, Item 9. The RHR pump motors will be protected from spray as required to ensure safe shutdown of the plant. Motor control centers for these pumps are also protected from spray. The spray sources which would fail these components do not fail the redundant trains by flooding (Section 3C.4).

3C.3.5.3 Control Building

The spray sources in the control building include chilled, service, makeup, domestic, and fire protection water systems (Table 3C.3-2). The spray-susceptible targets are the control panels, ventilation systems, and pump motors. The chiller equipment room is divided into two compartments, and Division A and B equipment is physically separated. However, service water for the Division B compartment passes through, and can spray, targets in the Division A compartment. Additionally, in the Division B compartment, Division B targets may be sprayed by a nonsafety-related makeup water line. These components are shielded from potential spraying, as required, to ensure availability of the system safe shutdown function when considering an additional single active failure as described in Section 3C.3.3, Item 9.

3C.3-5.4 Diesel Generator Building

The only potential spray source in the diesel generator building is service water (Table 3C.3-2).

Although there are many spray-susceptible targets in the diesel generator building, the diesels are each housed within a separate concrete structural cubicle which provides sufficient separation such that any given spray source could potentially fail only one division of emergency power. This is acceptable since the spray would not cause a reactor or turbine trip, and offsite power would still be available. The plant can be safety shut down considering an additional single active failure as described in Section 3C.3-3, Item 9. Potential flooding from the spray source may result in loss of the redundant trains of emergency power (Section 3C.4.5.4). However, this is not a problem since the postulated spray source does not cause a trip of the turbine. In this case, offsite power is available and the emergency power systems are not required.

3C.3.5.5 Piping Tunnels

There are no spray-susceptible targets in the piping tunnels.

3C.3.5.6 Electrical Tunnels

There are no spray-susceptible targets in the electrical tunnels.

3C.3.5.7 Standby Service Water Cooling Tower

The sources of water in the standby service water cooling tower are service water and makeup water. The spray-susceptible targets are the standby service water pumps, their associated MCCs, and the cooling tower fan motors. There is adequate physical separation such that only one division (A or B) of standby service water could potentially be failed by spray from as ingle MELC. A MELC in these zones would not cause a unit trip. Offsite power would be available, and safe shutdown could be achieved using the normal service water system. Flooding from the postulated cracks does not affect the redundant trains (Section 3C.4.5.7).

3C.3.5.8 Fuel Building

The water sources in the fuel building are listed in Table 3C.3-1. The spray-susceptible targets are the SFC pump motors and associated SFC components. These pump motors and components are housed within separate concrete structural cubicles which provide sufficient physical separation that spray from SFC Division A will not affect SFC Division B components, and vice versa. No single failure was postulated in the opposite train of SFC since the SFC system is qualified as a dual-purpose moderate-energy system as discussed in Section 3C.3.3, Item 9.

TABLE 3C.3-1
MODERATE ENERGY SYSTEMS LOCATED IN BUILDINGS CONTAINING
SAFE SHUTDOWN EQUIPMENT

	Building										
Moderate Energy Piping System	Reactor Drywell	Reactor Contain/ Annulus	Auxiliary/ Main Steam Tunnel	Fuel	Control	Diesel Generator	Standby Service Water Pump-house	Piping & Electrical Tunnels PT-1A	PT-4	PT-3	PT-1,2,3
Condensate Makeup and Drawoff (CNS)		X	X	X						X	X
Fire Protection (FPW)		X	X	X	X	X					X
Reactor Plant Component Cooling Water (CCP)	X	X	X	X							X
Service Water (SPW)	X	X	X		X	X	X	X	X	X	X
Makeup Water (MWS)			X		X		X	X	X	X	X
Turbine Plant Sampling (SST)			X								
Reactor Plant Sampling (SSR)	X	X	X								
Ventilation Chilled Water (HVN)		X	X								X
High-Pressure Core Spray (HPCS)	X	X	X	X						X	X
Low-Pressure Core Spray (LPCS)	X	X	X								
Reactor Core Isolation Cooling (ICS)	X	X	X							X	
Residual Heat Removal (RHR)	X	X	X							X	

TABLE 3C.3-1 (Cont)

Moderate Energy Piping System	Reactor Building: Drywell	Reactor Building: Contain/ Annulus	Building: Auxiliary/ Main Steam Tunnel	Building: Fuel	Building: Control	Building: Diesel Generator	Standby Service Water Pump-house	Piping & Electrical Tunnels: PT-1A	PT-4	PT-3	PT-1,2,3
Radioactive Liquid Waste (LWS)			X							X	X
Fuel Pool Cooling and Cleanup (SFC)		X	X	X						X	X
Control Rod Drive (RDS)	X	X	X	X							X
Fuel Transfer (SPT)				X							
Domestic Water (DWS)					X						
Control Building Chilled Water (HVK)					X						
Standby Liquid Control (SLS)	X	X									

TABLE 3C.3-2

MAXIMUM LEAKAGE RATES FOR EACH BUILDING
CONTAINING SAFE SHUTDOWN EQUIPMENT

Flooding Safety-Related Location	System with Maximum Leakage Rate	System Maximum Operating Conditions Pressure (psig)	/Temp (°F)[4]	Nominal Line Size (in)[5]	Maximum Leakage Rate (gpm)
Reactor building drywell	Residual heat removal (RHR)[1]	160	350[6]	18	1320
Containment/annulus	Residual heat removal[1]	180	350[6]	12	870
Auxiliary building/main steam tunnel	Residual heat removal[1,2]	160	350[6]	20	1610
Fuel building	Reactor plant component	100	125[7]	12	540
Fuel building (el 148'-0")	Fire protection (FPW)	120	70	4	100
Control building	Fire protection	120	70	6	190
Diesel generator building	Service water (SWP)	120	95	8	290
Standby service water pumphouse	Makeup water (MWS}	150	90[8]	4	120
Tunnel PT-4	Service water	120	95	16	910
Tunnel PT-3	Service water	100	95	18	1040
Tunnels PT-1, 2, 3 interconnected	Service water	100	95	24	1800

[1] The RHR system leakage rates are associated with the shutdown cooling mode.

[2] The leakage rates from the HPCS, LPCS, RHR, LPCIA, B, C, and PCIC (ICS) systems were based on the standby mode of operation. These leakage rates are exceeded by the RHR system shutdown cooling rode leakage rate.

[3] The reactor plant component cooling water (CCP) system is a closed system that is automatically served by the service water system when the CCP systems pressure is low.

[4] The maximum system operating pressures are established to the next higher (psig) in increments of 20 psig for calculation envelopes.

[5] Piping schedule 80 was used for calculation envelopes; for line sizes greater than 24 in, specified piping wall thicknesses were applied.

[6] This is the maximum temperature during the RHR shutdown cooling mode. Note that the spray temperature heating any components would be 212°F since the fluid would flash to atmospheric pressure on leaving the pipe.

[7] The maximum temperature is based upon the spent fuel pool cooling system.

[8] The maximum temperature is based upon the service water system.

SAFE SHUTDOWN PATHS			
DIVISION A — PATH 1	DIVISION B — PATH 2	DIVISION A - PATH 3	DIVISION B — PATH 4
SHORT TERM	SHORT TERM	SHORT TERM	SHORT TERM
RCIC	HPCS	ADS-A, 6 OF 7 SRV'S/LSV	ADS-B, 6 OF 7 SRV'S/LSV
LSV/3 OF 7 SRVS — ADS-A	LSV/3 OF 7 SRVS — ADS-B	LPCS	LPCI (RHR-C)
RHR-A (SUP. POOL COOL.)	RHR-B (SUP. POOL COOL.)	RHR-A (SUP. POOL COOL.)	RHR-B (SUP. POOL COOL.)
SWP-A	SWP-B	SWP-A	SWP-B
SFC-A	SFC-B	SFC-A	SFC-B
HVAC, MCC'S, CONTROLS ETC.	HVAC, MCC'S, CONTROLS ETC.	HVAC, MCC'S, CONTROLS ETC.	HVAC, MCC'S, CONTROLS ETC.
LONG TERM	LONG TERM	LONG TERM	LONG TERM
3 OF 7 SRVS/LSV, RHR-A (ALTERNATE) SHUTDOWN COOL.	3 OF 7 SRVS/LSV, RHR-B (ALTERNATE) SHUTDOWN COOL.	3 OF 7 SRVS/LSV, RHR-A (ALTERNATE) SHUTDOWN COOL.	3 OF 7 SRVS/LSV, RHR-B (ALTERNATE) SHUTDOWN COOL
SWP-A	SWP-B	SWP-A	SWP-B
SFC-A	SFC-B	SFC-A	SFC-B
HVAC, MCCS ETC.	HVAC, MCCS ETC.	HVAC, MCCS ETC.	HVAC, MCCS ETC

LEGEND:

RCIC	REACTOR CORE ISOLATION COOLING SYSTEM
SRV	MAIN STEAM SYSTEM SAFETY RELIEF VALVES
RHR	RESIDUAL HEAT REMOVAL SYSTEM
SWP	STANDBY SERVICE WATER SYSTEM
SFC	SPENT FUEL POOL COOLING SYSTEM
HVAC	VENTILATION AND COOLING SYSTEMS
MCC	MOTOR CONTROL CENTERS
LSV	PENETRATION VALVE LEAK CONTROL SYSTEM (SRV AIR SUPPLY)

NOTES:

1. WHEN A LOSS OF OFFSITE POWER IS POSTULATED, ALL SAFE SHUTDOWN PATHS REQUIRE THE EMERGENCY DIESELS AND SUPPORT SYSTEMS.

FIGURE 3C.3-1

SAFE SHUTDOWN PATHS, INCLUDING LONG TERM COOLING OF REACTOR AND CONTINUED SPENT FUEL POOL COOLING

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

3C.4 COMPARTMENT FLOODING AS A RESULT OF BREAKS OR CRACKS

3C.4.1 Discussion

The components and/or equipment required for safe shutdown of the reactor were evaluated for the effects of flooding from through-wall leakage cracks in moderate-energy systems, breaks in high-energy lines, and failure of nonseismic tanks, vessels, and pipes. The evaluation verifies that the plant can be safely shut down, assuming a concurrent single active failure in systems necessary to safely shut down the reactor and maintain long-term cooling. Where necessary, measures are provided to ensure component operability. Spraying effects from cracks in moderate-energy systems are discussed in Section 3C.3.

A detailed discussion of break/crack locations and types is provided in Sections 3.6.IA and 3.6.2A.

As discussed in the following sections, flooding effects from high-energy pipe breaks outside of containment are enveloped by moderate-energy crack flooding. This is primarily due to rapid detection and isolation of high-energy pipe breaks based on automatic isolation on area high temperature.

The total mass released by high-energy pipe breaks is shown in Table 3C.4-1, and the capacity of nonseismic tanks and vessels inside buildings containing safe shutdown equipment is shown in Table 3C.4-2. Flooding effects from external water sources are discussed in Section 3.4.

The nonseismic tanks and vessels were assumed to fail in accordance with the guidance in NUREG-0800, Standard Review Plan 3.4.1, II.3. Leakage from these sources through cracks in the connected piping was considered. These leakage rates are enveloped by the maximum leakage rates postulated for each associated building volume.

The areas that require leak detection provide, as a minimum, an alarm in the main control room. The alarm response will provide for plant area surveillance to locate and isolate the leak source. Thirty minutes from receipt of an alarm is sufficient time to locate and isolate the leakage source such that safe shutdown equipment is unaffected by the maximum potential leakage source.

Where curbs are incorporated into the plant design to prevent flooding of safety-related equipment, they are provided with sufficient height to prevent flow into the area, based on the maximum calculated flood level.

3C.4.2 Evaluation Procedure - Flooding

The approach for the flooding evaluation was similar to the procedure described in Section 3C.3.2 for the spraying evaluation. The evaluation was conducted utilizing the essential components making up the four pathways to safe shutdown defined in Table 3C.4-3 and located by environmental zones.

The following summary outlines the procedure used to evaluate flooding effects:

1. List by environmental zone all components and/or equipment required for safe shutdown in all buildings (see Fig. 3C.3-1).

2. Locate all safe shutdown targets by elevation.

3. Identify the hydraulic boundaries of each area to determine the extent of flooding. These were generally more extensive than the environmental zones.

4. Identify flood sources and calculate either maximum mass released or limiting crack flow rate (Section 3C.4.4) from postulated water sources.

5. Determine flood levels within each hydraulic boundary based on either total mass released or balance of flow in/out of the boundary. In this determination no credit is taken initially for the normal plant drainage system.

6. Identify all safe shutdown targets which could possibly be submerged and rendered inoperable. Evaluate all components and/or equipment to determine if they are waterproof (not susceptible to failure from submergence) and can withstand the effects of the water temperature. Table 3C.3-2 shows the maximum spray temperatures in each building.

7. Assume the failure of all targets in the hydraulic boundary that are determined to be below flood level and susceptible to failure. Identify the available paths to safe shutdown and maintenance of long-term cooling.

If it were concluded through this evaluation that the plant could not be shut down safely, a more detailed evaluation was conducted. No credit was taken for plant drainage systems, which were assumed to have failed.

8. In addition to the direct consequences of flooding, a single active failure is assumed in those systems required to mitigate the consequences of the piping failure and shut down the reactor.

9. Review drainage systems to ensure that leakage from one failed redundant train does not backflow through drains and flood the other train.

3C.4.3 Evaluation Guidelines - Flooding

The basic guidelines used to evaluate the effects of flooding were:

1. Within a given hydraulic boundary, the largest water source located anywhere in that boundary is used to calculate flood heights for all areas included. In many cases this leads to the largest water source being used for flood calculations on all floors within a building. A cross-check was made for sources from one building flooding into another building.

2. Credit is taken for flood protection by doorways and penetrations only if the particular doorway or penetration is specified as watertight.

3. All motors, including valve motor operators and solenoids, are assumed to fail if submerged.

4. All junction and terminal boxes containing termination boards are assumed to fail if submerged.

5. All instruments are assumed to fail if submerged.

6. All cables are nonhydroscopic and are not assumed to fail if submerged.

7. Motor control centers and switchgear are assumed to fail if submerged.

8. Guidelines for single active failure are the same as those assumed for failure due to spraying (Section 3C.3.3).

9. Credit is taken for operator action to isolate the leak 30 min after detection.

10. Flood detection is provided by an alarm in the main control room (MCR) as a minimum, except in the diesel generator building. The emergency core cooling systems pump cubicles each have a safety-related Class 1E cubicle flood level transmitter and associated level indication in the MCR. In addition, each cubicle has a nonsafety-related level switch with an alarm in the MCR. The auxiliary building crescent area has two safety-related Class 1E level switches on one division that are powered by an uninterruptible power supply (UPS) and alarm in the MCR. Other areas that require leak detection have redundant safetyrelated Class 1E level switches that alarm in the MCR.

3C.4.4 Analytical Methods

For a pipe in any given area, a through-wall leakage crack is assumed to occur at a location that would result in the most severe consequences due to flooding. The flow rate of the fluid is evaluated by assuming that the crack acts as an orifice. The following equation is used:

$$Q = 19.65 \cdot C \cdot d^2\sqrt{h_L}$$

where:

Q = Crack flow (gpm)

C = orifice coefficient

d = Equivalent diameter of crack (in)

h_L = Fluid head (ft)

The diameter of the crack is determined by assuming that the crack area is circular in shape. The area is defined as:

$A = (D/2)(t/2)$

where:

A = Crack area (in^2)

D = Nominal pipe diameter (in)

t = Nominal wall thickness (in)

The equivalent crack diameter is then defined as:

$$\mathbf{d} = \left(\frac{4A}{\pi}\right)^{1/2}$$

In calculating flow over stairways, hatches, and other floor openings or curbs, weir flow is assumed to determine the height of the water above the top of the weir as follows:

$$h_W = \left(\frac{q}{3.33L}\right)^{2/3}$$

where:

h_W = Water head above weir (ft)

q = Flow (ft3/sec)

L = Length of weir (ft)

If there is an intervening door which is not watertight, an additional head loss (modeled as a thick-edged orifice) is assumed for the door.

3C.4.5 Results of Evaluation - Flooding

The following subsections present, building-by-building, the results of the flooding evaluation using the procedures and guidelines discussed in Sections 3C.4.2 and 3C.4.3. The evaluation verifies that the plant can be safely shut down in the event of pipe cracks in fluid systems.

3C.4.5.1 Reactor Building (Including Drywell, Containment, and Annulus)

Leakage from a moderate-energy system within the drywell would result in a flood height to the top of the drywell weir wall. Once this level is reached, additional leakage would spill over the weir wall into the suppression pool. All equipment within the drywell which must operate during or after a LOCA is qualified for the appropriate environmental conditions as described in Section 3.11.

Leakage from a moderate-energy system is within the bounds of that qualification; therefore, the ability to safely shut down the plant is not impaired by this leakage.

Leakage from a moderate-energy system within the containment causes flood levels that do not affect equipment required for safe shutdown. The general floor elevations, except for el 186 ft-3 in, consist mostly of grating; therefore, no water accumulation can occur. Leakage into el 186 ft-3 in would result in a maximum flood height of approximately 4 in. Buildup above this level is prevented by spillage through grating. All leakage into general areas will spill into the suppression pool.

Cubicle volumes within the containment may flood to elevations greater than 10 in; however, these volumes do not contain equipment that is required for safe shutdown or spent fuel pool cooling.

In the annulus volume there is no equipment required for safe shutdown or spent fuel pool cooling, however flooding of this area is unacceptable for structural loading. The maximum limiting flood elevation is approximately 24 in, which is based upon redundant safety-related level switches that alarm in the MCR and 30 min for operator action to isolate the flood source. This flood level is acceptable for structural loading.

There are no external flooding sources to the reactor building.

3C.4.5.2 Auxiliary Building (Including Main Steam Tunnel)

The maximum flood height on the upper levels of the auxiliary building is approximately 6 in. in the general floor areas and 12 in. in cubicles. These flood heights are based on steady state water levels for weir flow over curbs surrounding equipment hatches and other openings, plus additional head losses for flow under doors.

The lowest elevation of the auxiliary building (el 70 ft) is comprised of separate watertight ECCS pump rooms and a crescent area containing the isolation valve. The crescent area contains two safety-related Class 1E level switches on one division that are powered by an uninterruptible power supply (UPS) and alarm in the control room. The maximum flood level in the crescent area is below all safe shutdown equipment, allowing 30 min for operator action to isolate the leakage.

Flooding in any one of the pump rooms will not affect the other ECCS pump rooms. Each drain line that penetrates the cubicles has redundant safety-related back-flow check valves. The RCIC, LPCI, LPCS, RHR, and HPCS pump rooms each have a single safety-related Class 1E level transmitter with level indication in the main control room. Also, each cubicle has a second nonsafety-related level detector which alarms in the main control room.

Operator action within 30 min after detection of flooding in any of these rooms is sufficient to keep water from flowing through ventilation openings high up in these cubicles and affecting the redundant ECCS pump rooms. These cubicles are capable of withstanding the additional structural loads due to this flooding.

Flooding on el 95 ft could potentially enter both the LPCS and HPCS cubicles at the same time from above. In this instance, the level detectors in each cubicle, as described above, provide redundant level detection, such that operator action within 30 min would prevent the failure of any safe shutdown equipment.

There is no leakage from external sources into this building. External doors that may be subject to flooding are designed as watertight.

The auxiliary building main steam tunnel may flood to an elevation of approximately 110 ft-0 in. This flood level is limited by spillage through piping penetrations into the turbine building. There is no equipment located in this volume that is required for safe shutdown.

3C.4.5.3 Control Building

Leakage from a moderate-energy system within this building could result in flood levels from approximately 2 to 14 in in the upper elevations. A buildup above these levels is prevented by spillage through doorways and stairwells. Safe shutdown equipment is above these flood levels except for electrical switchgear on el 98 ft-0 in. This area has an approximate flood level of 2 in. Curbs have been incorporated into the plant design to prevent the switchgear areas from flooding. There are no water sources within these areas, and the penetrations from above are water sealed.

The basement elevation has a limiting flood level of approximately 22 in, which is based upon once in 12-hr surveillance detection plus 30 min for operator action to isolate the flood source. Safe shutdown equipment items are above this flood level.

There is one external source of flooding to this building which is from the diesel generator building. This is a nonwatertight door that provides access between the control and diesel generator buildings at el 98 ft-0 in.

The potential maximum flooding flow rate from the diesel generator building to the control building is enveloped by the maximum flooding flow rate that is postulated for the control building.

3C.4.5.4 Diesel Generator Building

Technical Specifications require plant shutdown based upon standby diesel generator availability. Leakage from a moderate-energy system within this building would affect the emergency power sources only and not result in a trip of the turbine generator or reactor protection system. Therefore, safe shutdown is performed using offsite power.

The flood detection in the diesel generator building is by plant surveillance once per shift (every 12 hr). The maximum postulated leakage rate of the service water system has no significant effect on the service water system and is insufficient to affect more than one standby power source between surveillances.

3C.4.5.5 Piping and Electrical Tunnels

The three tunnel volumes have limiting flood levels of approximately 12 to 14 in. These flood levels are limited by redundant safety-related level switches that alarm in the MCR and 30 min for operator action to isolate the flood source. Safe shutdown equipment within the tunnels that is susceptible to flooding is above the flood levels.

There are no external flooding sources to two tunnel volumes because of watertight access doors and sealed penetrations. One tunnel volume has an external flood source from the standby service water cooling tower pumphouse. This external flood source flow rate is enveloped by the maximum postulated flooding flow rate postulated for this tunnel volume.

3C.4.5.6 Standby Service Water Cooling Tower Pumphouse

Leakage from a moderate-energy system within these areas could result in flood levels from approximately 2 in to

12 in. A buildup above these levels is prevented by spillage through doorways and stairwells into the piping tunnel (Section 3C.4.5.5). Safe shutdown equipment items are above the flood levels.

There is one external source of flooding to this area which is the piping tunnel. The maximum flood level from the tunnel source is the limiting flood elevation, approximately 12 in for the pumphouse (Section 3C.4.5.5). The safe shutdown items are above this flood level. Leak detection for both volumes is in the tunnel, which has a greater leakage rate than the pumphouse.

3C.4-S.7 Fuel Building

Leakage from moderate-energy systems within this building could result in flood levels from approximately 2 to 27 in in the upper elevations. A buildup above these levels is prevented by spillage through doorways and stairwells. Equipment required for spent fuel pool cooling is above these flood levels.

The basement has a limiting flood elevation of approximately 11 in, which is based upon redundant safety-related level switches that alarm in the MCR and 30 min for operator action to isolate the flood source. Equipment required for spent fuel pool cooling is above this flood level.

There are no external flooding sources to this building because of watertight doors and sealed penetrations.

TABLE 3C.4-1

TOTAL MASS RELEASED BY HIGH-ENERGY LINE BREAKS (HELBs)

Building	HELB	Total Mass (lb)
Auxiliary building[1]	RWCU	7,776
Control building	None	-
Diesel generator building	None	-
Piping and electrical tunnels	Main steam line	164,352
Standby service water cooling tower	None	-
Fuel building	None	-
Reactor building	[2]	-

----

[1] Mass released by the high-energy liquid line (RWCU) envelopes the RCIC steam line break releases.

[2] Included in LOCA analyses. Refer to Sections 6.2.1 and 6.2.2.

TABLE 3C.4-2

CAPACITY OF NONSEISMIC TANKS AND VESSELS
WITIN BUILDINGS CONTAINING SAFE
SHUTDOWN EQUIPMENT

Building	Mark No.	Capacity (gal) (total)
Reactor building	None	-
Auxiliary building	1CCP-TK1	3,000
Control building	None	-
Diesel generator building	1EGF-TK3A 1EGF-TK3B	35 35
Piping tunnels	None	-
Electrical tunnels	None	-
Standby service water cooling tower	None	-
Fuel building	1SFC-TK2 1SFT-TK1	560 1,525

APPENDIX 3D

DESIGN CRITERIA FOR PROCESS AND GUARD PIPES IN MARK III BWR CONTAINMENTS

APPENDIX 3D

DESIGN CRITERIA FOR PROCESS

AND GUARD PIPES IN MARK III

BWR CONTAINMENTS

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APPENDIX 3D
DESIGN CRITERIA FOR PROCESS AND GUARD PIPES IN MARK III BWR CONTAINMENTS

3D.1 GENERAL

A guard pipe is defined as a process pipe enclosure used to direct high-energy fluids which may escape from the process pipe as it passes through the containment back into the drywell.

Guard pipes are installed on those process lines which form part of the reactor coolant pressure boundary and carry high-energy fluids through the drywell and containment. Guard pipes are provided when the following criteria apply to the process line:

1. A crack or leak occurring in the process pipe within the containment along with a single active failure of the drywell isolation valve would otherwise result in overpressurization of the containment vessel.

2. Either the temperature of the process fluid exceeds 200°F or the pressure of the fluid exceeds 275 psig.

Guard pipes are installed on the following process lines:

4 - 24-in diameter main steam lines
2 - 20-in diameter feedwater lines
1 - 5-in diameter steam drain line
1 - 8-in diameter RCIC steam supply
•→12
12←•
1 - 18-in diameter RHR shutdown suction line
1 - 6-in diameter RWCU suction line

A typical guard pipe assembly is shown in Fig. 3.8-4. The guard pipe is joined to the process pipe by a flued head forged extension. The guard pipe is anchored to the reinforced concrete drywell wall. A metal bellows is used to provide a flexible seal between the guard pipe and the containment vessel. A nonmetallic seal is also installed around the guard pipe where it penetrates the shield building. A lateral restraint is provided between the shield building and guard pipe to restrict the lateral movement of the guard pipe due to dynamic loads and also to limit the stresses and deformations in the guard pipe. Wherever required, mid-guard restraints are provided to the guard pipe to limit the stresses and deformations in the process pipe. These restraints are located between the drywell wall and the steel containment.

3D.2 DESIGN CRITERIA

3D.2.1 Process Pipe

For the process pipe within the guard pipe, the design criteria are:

1 The process pipe is required to be seamless.

2. Moment limiting restraints are to be provided to resist both bending and torsional moments on all high-energy lines. These restraints are not located within the guard pipes, but are beyond the isolation valves inboard and outboard of the guard pipes. Mid-guard restraints, where required, are located within the guard pipes.

3. The design meets the requirements of the designated ASME Code Class 1.

4. The design loading combinations and stress limits are discussed in Section 3D.2.5.

3D.2.2 Flued Heads

Flued heads are provided between the process pipe and guard pipe and between the guard pipe and the penetration bellows.

3D.2.2.1 Flued Heads between the Guard Pipe and the Process Pipe

The design criteria for flued heads between the guard pipe and the process pipe are as follows:

1. The design meets the requirements of ASME Code Class 1.

2. The connecting weld between the flued head and the guard pipe also meets the requirements of ASME Code Class 1 and serves as the boundary between the ASME Code Class MC rules applicable to the guard pipe and the ASME Code Class 1 rules applicable to the flued head.

3. The loading combinations and stress limits are as delineated in Section 3D.2.5.

3D.2.2.2 Flued Heads between the Guard Pipe and the Expansion Bellows

The design criteria for flued heads between the guard pipe and the expansion bellows are as follows:

1. The design meets the requirements of ASME Code Class MC.

2. The connecting welds between the guard pipe, flued head, and expansion bellows meet the requirements of ASME Code Class MC.

3. The loading combinations and stress limits are as delineated in Section 3D.2.5.

3D.2.3 Guard Pipe

For the guard pipe which encloses the process pipe, the design criteria are:

1. The portion of the guard pipe from the drywell wall up to the weld connecting to the process pipe flued head is designated as Class MC and, therefore, is designed and constructed in accordance with Subsection NE, Section III, of the ASME Code, augmented by the applicable provisions of Regulatory Guide 1.57.

2. The design loading combinations and stress limits are as delineated in Section 3D.2.5.

3. The design pressure and temperature of the guard pipe are equal to the maximum operating pressure and temperature of the enclosed process pipe under normal plant conditions.

4. All guard pipe assemblies are subjected to a single pressure test at a pressure not less than their design pressures.

3D.2.4 Augmented Inservice Inspection of the Guard Pipe and Flued Head Weld

The flued head to the guard pipe and welds within the guard pipe are inspectable in accordance with the requirements of Section XI of the ASME Code. Additionally, the weld between the flued head and the process pipe, when such welds are used, is inspectable in accordance with the requirements of Section XI of the ASME Code.

3D.2.5 Design Loading Combinations and Stress Limits

The design loading combinations and stress limits for process pipes and flued heads which serve as process pipe boundary are tabulated in Table 3.8-5. The loading combinations are in accordance with Regulatory Guide 1.48, and the stress limits meet the criteria for break exclusion zones as delineated in Section 3.6A.

The design loading combinations and stress limits for guard pipes and flued heads which serve as the transition between the guard pipe and the expansion bellows are tabulated in Table 3.8-6.