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3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 SUMMARY DESCRIPTION

This section contains an evaluation of the design bases of the Energy Northwest Nuclear Project No. 2 (Columbia Generating Station (CGS)) nuclear generating station as compared to the NRC General Design Criteria (GDC) for Nuclear Power Plants, Appendix A of 10 CFR Part 50, effective May 21, 1971, and subsequently amended on July 7, 1971. The GDC, which are divided into six groups and total 55 in number, are intended to establish minimum requirements for the design of nuclear power plants.

The GDC 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 interpretation. For this reason, there are some cases where conformance to a particular criterion is not directly comparable. 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 section references are included to identify where detailed design information pertinent to each criterion is discussed.

Based on the content herein, Energy Northwest concludes that CGS is in compliance with the GDC.

3.1.2 CRITERION CONFORMANCE

3.1.2.1 Group I - Overall Requirements

3.1.2.1.1 Criterion 1 - Quality Standards and Records

> Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to 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 (GDC 1).

Evaluation Against Criterion 1

Structures, systems, and components important to safety are identified in Section 3.2. The quality assurance program used during the operations phase is described in EN-QA-004, Energy Northwest Operational Quality Assurance Program Description. The quality assurance program used during the design and construction of the plant was provided at the Preliminary Safety Analysis Report (PSAR) stage and has been applied to the items contained in Table 3.2-1. The intent of the quality assurance program is to ensure 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 ensures 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 keeping appropriate records. The total quality assurance program of Energy Northwest and its principal contractors is responsive to and satisfies the quality-related requirements of 10 CFR Part 50, including Appendix B.

Structures, systems, and components are first classified in Section 3.2 with respect to their location and service and their relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment to ensure a quality product in keeping with the required safety function. In cases where codes are not available or the existing code must be modified, an explanation is provided.

Documents are maintained which demonstrate that the requirements of the quality assurance 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 item of information is retrievable for reference. These records will be maintained in accordance with the Operational Quality Assurance Program Description.

The detailed quality assurance program developed by the applicant and its contractors satisfies the requirements of Criterion 1.

For further discussion see the following sections:

a. Principal design criteria 1.2

b. Plant description 1.2

c. Classification of structures, components, and systems 3.2

d. Quality assurance 17

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of 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 (GDC 2).

Evaluation Against Criterion 2

The design criteria adopted for structures, systems, and components considers the magnitude and the probability of occurrence of natural phenomena at the specific site. The designs are based on the most severe natural phenomena recorded for the site with an appropriate margin to account for uncertainties in the historical data. Detailed discussion of the various phenomena considered and the design criteria developed are presented in the sections listed below.

The design criteria developed meet the requirements of Criterion 2.

For further discussion, see the following sections:

a. Meteorology 2.3

b. Hydrologic engineering 2.4

c. Geology and seismology 2.5

d. Classification of structures, components, and systems 3.2

e. Wind and tornado design loadings 3.3

f. Water level (flood) design 3.4

g. Missile protection 3.5

h. Seismic design 3.7

i. Design of Seismic Category I structures 3.8

j. Mechanical systems and components 3.9

k. Seismic qualification of Category I instrumentation and electrical equipment 3.10

l. Environmental design of mechanical and electrical equipment 3.11

3.1.2.1.3 Criterion 3 - Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effect of fires on structures, systems, and components important to safety. Firefighting 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 (GDC 3).

Evaluation Against Criterion 3

Insurer and National Fire Protection Association guidelines were used for the design of the plant.

Noncombustible and fire resistant materials are used wherever practical throughout the facility, particularly in areas containing critical portions of the plant such as containment structure, control room, and components of systems important to safety. These systems are designed and located to minimize the effects of fires or explosions on their redundant components. Facilities for the storage of combustible materials such as fuel oil are located, designed, and protected to minimize both the probability and the effects of a fire.

Equipment and facilities for detecting, annunciating, and extinguishing fires are provided to protect both plant and personnel from fire or explosion.

Administrative controls are utilized where applicable throughout the facility to minimize the probability and consequences of fires and explosions.

The fire protection system is designed such that a failure of any component of the system will not

a. Generate an accident resulting in significant release of radioactivity to the environment, or

b. Impair the ability of redundant equipment to safely shut down and isolate the reactor or limit the release of radioactivity to the environment in the event of a loss-of-coolant accident (LOCA).

For further discussion, see the following sections:

a.	Design of Seismic Category I structures	3.8
b.	Fire protection system	9.5.1, Appendix F

3.1.2.1.4 Criterion 4 - Environmental and Missile Design Bases

> Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit (GDC 4).

Evaluation Against Criterion 4

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. (See Section 3.11.)

These structures, systems, and components are appropriately protected against dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures.

The electrical equipment, instrumentation, and associated cables of protection and engineered safety features (ESF) system which are located inside the containment are discussed in the

sections listed below. The design requirements in terms of the time that each must survive the extreme environmental conditions following a LOCA are indicated.

The design of these structures, systems, and components meets the requirements of Criterion 4.

For further discussion, see the following sections:

a.	Classification of structures, components, and systems	3.2
b.	Missile protection	3.5
c.	Protection against dynamic effects associated with the postulated rupture of piping	3.6
d.	Design of Seismic Category I structures	3.8
e.	Mechanical system and components	3.9
f.	Environmental design of mechanical and electrical equipment	3.11
g.	Integrity of reactor coolant pressure boundary (RCPB)	5.2
h.	ESF	6
i.	Instrumentation and controls	7
j.	Electric power	8

3.1.2.1.5 Criterion 5 - Sharing of Structures, Systems, and Components

> 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 (GDC 5).

<u>Evaluation Against Criterion 5</u>

CGS is a single unit plant, and therefore Criterion 5 does not apply.

3.1.2.2 Group II - Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10 - Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (GDC 10).

Evaluation Against Criterion 10

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 maintain 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 occurrences.

The reactor protection system (RPS) is designed to monitor certain reactor parameters, sense abnormalities, and to scram the reactor thereby preventing fuel design limits from being exceeded. Scram trip setpoints are selected on operating experience and by the safety design basis. There is no case in which the scram trip setpoints allow the core to exceed the thermal hydraulic safety limits. Power for the RPS is supplied by two independent ride-through ac power supplies. An alternate power source is available for each bus.

An analysis and evaluation has been made of the effects on 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 ensure that the specified fuel design limits are not exceeded during the conditions of normal or abnormal operation and therefore meet the requirements of Criterion 10.

For further discussion, see the following sections:

a. Principal design criteria 1.2.1

b. Plant description 1.2.2

c. Fuel system design 4.2

d. Nuclear design 4.3

e. Thermal and hydraulic design 4.4

f. Reactor materials 4.5

g. Functional design of reactivity control systems 4.6

h. Reactor coolant system and connected systems 5

i. Reactor trip system 7.2

j. Accident analyses 15

3.1.2.2.2 Criterion 11 - Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity (GDC 11).

Evaluation Against Criterion 11

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

The inherent dynamic behavior of the core is characterized in terms of the fuel temperature or doppler coefficient, the moderator void coefficient, and the 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; this 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 which permits the use of the 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 coefficient of reactivity, the BWR has a number of inherent advantages, such as the use of coolant flow as opposed to control rods for

load following, inherent self-flattening of the radial power distribution, ease of control, and 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 -0.3 $\Delta k/k/\Delta P/P$ 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:

a. Principal design criteria 1.2

b. Nuclear design 4.3

c. Thermal and hydraulic design 4.4

3.1.2.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to 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 (GDC 12).

Evaluation Against Criterion 12

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or doppler coefficient, moderator void coefficient, and moderator temperature coefficient to a 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 disturbance 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

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

b. Load following with recirculation flow control, and

c. Strong damping of spatial power disturbances.

The RPS design provides protection from excessive fuel cladding temperatures and protects the RCPB from excessive pressures that 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. 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 which could result in exceeding fuel design limits. These systems ensure that Criterion 12 is met.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Reactivity control systems	4.1
c.	Nuclear design	4.3
d.	Thermal and hydraulic design	4.4
e.	Functional design of reactivity control systems	4.6
f.	Overpressurization protection	5.2.2
g.	Reactor trip system	7.2
h.	Reactor manual control system instrumentation and control	7.7
i.	Accident analyses	15

3.1.2.2.4 Criterion 13 - Instrumentation and Control

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

its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges (GDC 13).

Evaluation Against Criterion 13

The fission process is monitored and controlled for all conditions from source range through power operating range. The intermediate and power ranges of the neutron monitoring system detect core conditions that threaten the overall integrity of the fuel barrier due to excess power generation and provide a signal to the RPS. Fission detectors, located in the core, are used for neutron detection. The detectors are located to provide optimum monitoring in the intermediate and power ranges.

The intermediate range monitors (IRM) measure neutron flux from the upper portion of the source range monitors (SRM) to the lower portion of the local power range monitor (LPRM) subsystem. The IRM is capable of generating a trip signal to scram the reactor.

The LPRM subsystem consists of fission chambers located throughout the core, signal conditioning equipment, and trip functions. Local power range monitor signals are also used to block rod withdrawal and to generate the necessary trip signal for reactor scram [average power range monitor (APRM)].

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

Nuclear system leakage limits are established so that appropriate action can be taken to ensure the integrity of the RCPB. Nuclear system leakage rates are classified as identified and unidentified, which corresponds respectively to the flow to the equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based on the makeup capabilities of various reactor component systems. Flow integrator and recorders are used to determine the leakage flow pumped from the drain sumps. The unidentified leakage rate as established in Section 5.2.5 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 the integrity of the process barrier is threatened.

The process radiation monitoring system monitors radiation levels of various processes and provides trip signals to the containment and reactor vessel isolation control system whenever preestablished limits are exceeded.

As noted above, adequate instrumentation is provided to monitor system variables in the reactor core, RCPB, and reactor containment. Appropriate controls are 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:

a.	Principal design criteria	1.2
b.	Functional design of reactivity control systems	4.6
c.	Detection of leakage through RCPB	5.2.5
d.	Main steam line isolation system	5.4.5
e.	Containment systems	6.2
f.	Primary containment and reactor vessel isolation control system instrumentation and control	7.1
g.	Reactor trip system	7.2
h.	ESF systems	7.3
i.	Systems required for safe shutdown	7.4
j.	Safety-related display instrumentation	7.5
k.	All other instrumentation systems required for safety	7.6
l.	Control systems not required for safety	7.7

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

> The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture (GDC 14).

Evaluation Against Criterion 14

The piping and equipment pressure parts within the RCPB through the outer isolation valves are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies systems and components within the RCPB to Quality Group A Requirements. The design requirements and codes and standards applied to this quality group ensure a quality product in keeping with the safety functions to be performed.

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 used to control toughness properties. Materials are impact tested in accordance with ASME Code Section III, where applicable. 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 defects. All welding procedures, welders, and welding machine operators used in producing pressure-containing welds are qualified in accordance with the requirements of ASME Code Section IX, 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 materials 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 Criterion 30 of the GDC.

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

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Design of structures, components, equipment, and systems	3
c.	Integrity of RCPB	5.2
d.	Overpressurization protection	5.2.2
e.	Reactor vessel	5.3

f. Reactor coolant piping 5.4.3

g. Reactor trip system 7.2

h. Accident analyses 15

3.1.2.2.6 Criterion 15 - Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences (GDC 15).

Evaluation Against Criterion 15

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 that ensure the integrity of the RCPB throughout the plant lifetime. The reactor coolant system was designed and fabricated to meet the requirements of the ASME Code Section III, as indicated in Section 5.2.1.

The auxiliary, control, and protection systems associated with the reactor coolant system act to provide sufficient margin to ensure that the design conditions of the 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 ensure that the design conditions of the RCPB are not exceeded, is the automatic initiation of the nuclear system pressure relief system on 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 system (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 ensure that the requirements of Criterion 15 are satisfied.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Design of structures, components, equipment, and systems	3
c.	Integrity of RCPB	5.2
d.	Overpressurization protection	5.2.2
e.	Detection of leakage through the reactor coolant pressure boundary	5.2.5
f.	Reactor vessel	5.3
g.	Reactor coolant piping	5.4
h.	Reactor trip system	7.2
i.	Accident analyses	15

3.1.2.2.7 Criterion 16 - Containment Design

> Reactor containment and associated systems shall be provided to establish an essentially leak-tight 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 (GDC 16).

Evaluation Against Criterion 16

An essentially leaktight barrier is provided by a steel primary containment consisting of a drywell and suppression chamber and a containment isolation system. Release of radioactivity from the primary containment to the environment is further controlled by the reactor building, which provides secondary containment, and the standby gas treatment system (SGTS). These

systems are designed to protect the public from the consequences of a LOCA, based on a postulated break of the reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe.

The primary containment, reactor building, and the associated ESF systems are designed to safely sustain all internal and external environmental conditions that may be postulated to occur during the life of the plant, including both short- and long-term effects following a LOCA.

Containment temperature and pressure following an accident are limited by the suppression pool and the RHR system.

For further discussion, see the following sections:

a.	General plant description	1.2
b.	Steel containment	3.8
c.	Containment systems	6.2
d.	Accident analyses	15

3.1.2.2.8 Criterion 17 - Electric Power Systems

> An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function 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 reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.
>
> The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.
>
> Electric power from the transmission network 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 reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available to assure that core coolant, primary 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 (GDC 17).

Evaluation Against Criterion 17

Offsite power is transmitted to the plant transformer yard by two physically independent transmission lines which approach the plant along separate rights-of-way. Offsite power is available for plant startup or shutdown utilizing the 230-kV startup transformer. This transformer has sufficient capacity to carry all plant normal loads, in addition to those ESF loads required for plant shutdown or mitigation of the consequences of an accident.

A 115-kV transmission line, located on a separate right-of-way, provides power to a backup transformer which is capable of providing power for the plant safety-related systems to mitigate the effects of an accident. This power source is automatically transferred to the plant safety-related system on loss of all other offsite power sources.

The three independent onsite ac load groups provide independence and redundancy of equipment function. They meet the safety requirements assuming a single failure since the load groups are connected to offsite power by circuits having independent routes from the Class 1E switchgear.

For each of the three ac load groups there are independent batteries that furnish dc load and control power for the corresponding divisions.

The reactor protection instrumentation is powered from two independent ride-through ac power sources.

During normal plant operation, the station auxiliary power is supplied from the main generator through the plant auxiliary transformers. On loss of power from the normal auxiliary transformers, there will be an automatic transfer to a source of offsite power from the 230-kV startup or 115-kV backup transformers. In the event of a loss of the 230-kV and 115-kV offsite power sources, three onsite diesel generator sets and redundant sets of station batteries provide the necessary ac and dc power for safe shutdown or, in the event of an accident, to

restrict the consequences to within acceptable limits. The onsite emergency ac and dc power systems contain sufficient redundancy and independence such that a single failure does not prevent the systems from performing their safety function.

The power systems as designed meet the requirements of Criterion 17.

For further discussion, see the following sections:

a.	General plant description	1.2
b.	Seismic qualification of Seismic Category I instrumentation and electrical equipment	3.10
c.	Environmental design of mechanical and electrical equipment	3.11
d.	Electric power	8

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electric Power Systems

> Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system (GDC 18).

Evaluation Against Criterion 18

The engineered safety systems power supply buses and associated diesel generators are arranged for periodic inspection and testing of each diesel generator independently.

At periodic intervals when the reactor is not at power operation, tests simulating loss of power and accident signals will be initiated to demonstrate the capability of the power systems to meet the starting and loading requirements.

Periodically during plant operation each diesel generator will be manually started and loaded. The redundant (Division 1 and 2) units will be synchronized to the startup power source and

loaded. The high-pressure core spray (HPCS) (Division 3) unit will be loaded by synchronizing to the startup power source.

These tests are designed to prove the operability of the onsite power system under conditions as close to design as possible, to assess the continuity of the systems and condition of the components.

The electrical power systems are configured to provide access for appropriate periodic inspection. For further discussion, see the following sections:

a. Onsite power systems 8.3

b. Initial test program 14

3.1.2.2.10 Criterion 19 - Control Room

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.
>
> Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures (GDC 19).

Evaluation Against Criterion 19

A control room is provided from which the nuclear power plant can be safely operated under normal conditions and can be maintained in a safe condition following postulated accidents.

The control room contains the following equipment: controls and necessary surveillance equipment for operation of the plant functions, such as the reactor and its auxiliary systems, ESF, turbine generator, steam and power conversion systems, and station electrical distribution boards.

The control room is located in a Safety Class 3, Seismic Category I structure. Safe occupancy of the control room during abnormal conditions is ensured by the design. Adequate shielding

is provided to maintain tolerable radiation levels in the control room in the event of a design basis accident for the duration of the accident.

The control room ventilation system has redundant and spatially separated fresh air intakes and redundant equipment, radiation detectors, and smoke detectors with appropriate alarms and interlocks. Redundant systems are provided for the control room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters during isolation of the control room from outside air.

The control room can be continuously occupied under all operating and accident conditions. In the unlikely event that the control room must be vacated and access is restricted, instrumentation and controls are provided outside the control room from which safe shutdown of the reactor can be effected.

The above demonstrates that the control room design meets the requirements of Criterion 19.

For further discussion, see the following sections:

a.	General plant description	1.2
b.	Radwaste and control building	3.8
c.	Classification of structures, components, and systems	3.2
d.	Control room habitability	6.4
e.	Instrumentation and control	7
f.	Shutdown outside the control room	7.4
g.	Control room area ventilation system	9.4
h.	Fire protection	Appendix F
i.	Shielding	12.3
j.	Ventilation	12.3

3.1.2.3 Group III - Protection and Reactivity Control System

3.1.2.3.1 Criterion 20 - Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to 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 (GDC 20).

Evaluation Against Criterion 20

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB barrier. Fuel damage is prevented by the initiation of an automatic reactor shutdown if monitored system variables exceed 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 RPS includes the high-inertia motor generator power system, sensors, relays, bypass circuitry, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear system high pressure, turbine stop valve closure, turbine control valve fast closure, MSIV closure, and reactor vessel low water level will prevent fuel damage following abnormal operational transients. Specifically, the process parameters initiate a scram in time to prevent the core from exceeding thermal hydraulic safety limits during abnormal operational transients. In addition, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to safety.

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

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Fuel system design	4.2
c.	Functional design of reactivity control systems	4.6
d.	Main steam line isolation system	5.4.5
e.	Reactor trip system	7.2
f.	Accident analyses	15

3.1.2.3.2 Criterion 21 - Protection System Reliability and Testability

> The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to 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 (GDC 21).

Evaluation Against Criterion 21

The RPS 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 will impair the ability of the system to perform its intended safety function. Additionally, the system design ensures that when a scram trip point is exceeded there is a high scram probability. However, should a scram not occur, other monitored components will scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between trip logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

The RPS includes design features that permit inservice testing. This ensures the functional reliability of the system should the monitored reactor parameter exceed the corrective action setpoint.

The RPS 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 results when both trip systems have tripped. The logic scheme is a one-out-of-two twice arrangement. The RPS 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.

Control rod drive (CRD) operability can be tested during normal reactor operation. Drive position indicator 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. One control rod is tested at a time. Control rod drive mechanism over-travel tests demonstrate rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. More importantly, the hydraulic control unit scram accumulator and the scram discharge volume level are continuously monitored.

The MSIVs may be tested during full reactor operation. Individually, they can be closed to 90% of full open position without affecting the reactor operation. If reactor power is reduced sufficiently, the isolation valves on a single main steam supply line may be fully closed. Provisions are provided to evaluate valve stem leakage during reactor shutdown. During refueling operation, valve leakage rates can be determined.

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:

a. Principal design criteria 1.2

b. Functional design of reactivity control systems 4.6

c. Main steam line isolation system 5.4.5

d. Reactor trip system 7.2

3.1.2.3.3 Criterion 22 - Protection System Independence

> 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 (GDC 22).

<u>Evaluation Against Criterion 22</u>

The components of protection systems are designed so that the mechanical and thermal environmental conditions resulting from any emergency situation in which the components are

required to function will not interfere with the operation of that function. The safety-related wiring and cabling necessary to support the operation of the RPS, outside of enclosures, are routed within totally enclosed and grounded raceway systems. The RPS has four independent initiation channels providing input to two trip systems for redundancy. Cabling for the four channels is routed in separate raceway systems to preserve channel independence. The wires from duplicate sensors on a common process tap are run in separate wireways. The system sensors are electrically and physically separated. Only one trip actuator logic circuit from each trip system may be run in the same wireway.

The RPS is designed to permit maintenance, calibration, and testing activities while the reactor is operating without restricting plant operation or affecting system safety function completion. This is accomplished by limiting these activities to only one of the independent initiation sensors or associated logic channels at a time leaving at least two channels per monitored parameter available to respond to an accident or transient. Only a single channel per trip system is required to initiate a scram. The protection system meets the design requirements for functional and physical independence as specified in Criterion 22.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Functional design of reactivity control systems 4.6

c. Main steam line isolation system 5.4

d. Reactor trip system 7.2

3.1.2.3.4 Criterion 23 - Protection System Failure Modes

> The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced (GDC 23).

<u>Evaluation Against Criterion 23</u>

The RPS components/logic are designed to fail into a safe state (system actuation) when conditions such as loss of electrical power source or fire occur. This is accomplished by maintaining the input/output logic components and final actuated devices in a normally energized state with deenergization resulting in the completion of their intended safety functions. Additionally, RPS cables that are located outside of panels are enclosed within totally enclosed and grounded raceways providing additional fail safe design.

Reactor protection system components that perform safety-related functions to mitigate a design basis event are designed to complete their intended safety functions even when exposed to adverse environmental conditions (e.g., temperature, pressure, steam, water, and radiation) postulated to occur during those events.

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

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Environmental design of mechanical and electrical equipment	3.11
c.	Functional design of reactivity control systems	4.6
d.	Reactor trip system	7.2
e.	Electric power	8.3

3.1.2.3.5 Criterion 24 - Separation of Protection and Control Systems

> The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired (GDC 24).

Evaluation Against Criterion 24

There is separation between the RPS and the process control systems. Sensors, trip channels, and trip logics of the RPS 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 system. High scram reliability is designed into the RPS and hydraulic control unit for the CRD. The scram signal and mode of operation overrides all other signals.

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

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Functional design of reactivity control systems	4.6
c.	Reactor trip system	7.2
d.	Reactor manual control system instrumentation and controls	7.7

3.1.2.3.6 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

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 (GDC 25).

Evaluation Against Criterion 25

The RPS provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Any monitored variable exceeding the scram setpoint will initiate an automatic scram and not impair the remaining variables from being monitored. If one channel fails, the remaining portions of the RPS shall function.

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

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

The design of the protection system ensures that specified acceptable fuel 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:

a.	Principal design criteria	1.2
b.	Nuclear design	4.3
c.	Thermal and hydraulic design	4.4
d.	Functional design of reactivity control systems	4.6
e.	Reactor trip system	7.2
f.	Reactor manual control system instrumentation and controls	7.7
g.	Accident analyses	15

3.1.2.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to 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 acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions (GDC 26).

Evaluation Against Criterion 26

Two independent reactivity control systems using different design principles are provided. The normal method of reactivity control employs control rod assemblies. Positive insertion of these control rods is provided by means of the CRD 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 will not adversely affect the capability to maintain the core within fuel design limits.

The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. 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. 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 worth minimizer, 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 unlikely occurrence of a limited number of stuck rods will not hinder the capability of the control rod system to render the core subcritical.

The second independent reactivity control system is provided by the reactor coolant recirculation system. By varying reactor flow, it is possible to affect the type of reactivity changes necessary for planned, normal power changes (including xenon burnout). In the event that reactor flow is suddenly increased to its maximum value (pump runout), the core will not exceed fuel design limits because the power flow map defines the allowable initial operating states such that the pump runout will 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_2O_3) to control the high reactivity of fresh fuel. In addition, the standby liquid control system is available to add soluble boron to the core and render it subcritical, as discussed in Section 3.1.2.3.8.

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:

a. Principal design criteria 1.2

b. Reactivity control system 4.3

c. Functional design reactivity control system 4.6

d. Reactor trip system 7.2

e. Process computer system instrumentation and controls (rod worth minimizer) 7.7

f. Reactor manual control system instrumentation and controls 7.7

g. Reactor recirculation system 5.4

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained (GDC 27).

Evaluation Against Criterion 27

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 BWR design is capable of maintaining the reactor core subcritical, including allowance for a stuck rod, without addition of any poison to the reactor coolant. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed and, if protection system limits are reached, corrective action is initiated through 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 RPS 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 control rods alone as the reactor is cooled down subsequent to initial shutdown, the standby liquid control (SLC) system will be actuated to insert soluble boron into the reactor core. The SLC system has sufficient capacity to ensure that the reactor can always be maintained subcritical; and hence, only decay heat will be generated by the core which can be removed by the RHR system, thereby ensuring that the core will always be coolable.

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

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Reactivity control system	4.3
c.	Nuclear design	4.3
d.	Thermal and hydraulic design	4.4
e.	Functional design of reactivity control system	4.6
f.	Reactor trip system	7.2
g.	Reactor manual control system	7.7
h.	Process computer system instrumentation and controls (rod worth minimizer)	7.7
i.	Standby liquid control system	9.3.5
j.	Accident analyses	15

3.1.2.3.9 Criterion 28 - Reactivity Limits

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition (GDC 28).

Evaluation Against Criterion 28

The control rod system design incorporates appropriate limits on the amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve

optimum core performance and low individual rod worths. The rod worth minimizer prevents withdrawal other than by the preselected rod withdrawal pattern. The rod worth minimizer function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level operations control rod procedures.

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod which prevents rapid rod ejection. This protects against a high reactivity insertion rate by limiting the control rod velocity to less than 5 ft/sec. Normal rod movement is limited to 6 in. increments and the rod withdrawal rate is limited through the hydraulic valve to 3 in./sec.

The accident analyses (Chapter 15) evaluates 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 would result in damage to the RCPB. In addition, the integrity of the core, its support structures or other reactor pressure vessel (RPV) 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 analyses.

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:

a.	Principal design criteria	1.2
b.	CRD systems	3.9
c.	Reactor core support structures and internals mechanical design	4.2
d.	Reactivity control system	4.3
e.	Nuclear design	4.3
f.	CRD housing supports	4.5.3
g.	Overpressurization protection	5.2.2
h.	Reactor vessel and appurtenances	5.3

i.	Main steam line flow restrictor	5.4.4
j.	MSIV system	5.4.5
k.	Process computer system	7.7.1.9
l.	Accident analyses	15

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

> The protection and reactivity control system shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences (GDC 29).

Evaluation Against Criterion 29

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

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Components important to safety such as CRDs and MSIVs are tested during normal operation. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action setpoint.

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

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Reactivity control system	4.6
c.	MSIV system	5.4.5
d.	Reactor trip system	7.2

3.1.2.4 Group IV - Fluid Systems

3.1.2.4.1 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

> Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage (GDC 30).

Evaluation Against Criterion 30

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 that compose the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Table 3.2-1 and Chapter 5. Further, product and process quality planning is provided to ensure conformance with the applicable codes and standards and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Criterion 14, Reactor Coolant Pressure Boundary.

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, 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 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 with loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the leak detection system provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

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

For further discussion, see the following sections:

a. Principal design criteria 1.2

b.	Design of structures, components, equipment, and systems	3
c.	Overpressurization protection	5.2.2
d.	Integrity of RCPB	5.2
e.	Detection of leakage through the RCPB	5.2.5
f.	Reactor vessel	5.3
g.	Component and subsystem design	5.4
h.	Reactor recirculation system	5.4
i.	Instrumentation and control systems	7
j.	Primary containment and reactor vessel isolation control system	7.3
k.	Leak detection system - instrumentation and control	7.6
l.	Quality assurance	17
m.	Inservice inspection and testing of the RCPB	5.2.4

3.1.2.4.2 Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to 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 (GDC 31).

Evaluation Against Criterion 31

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the RPV, the RPV was designed to meet the requirements of ASME Code Section III.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1 x 10^{17} nvt with neutron 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 up on the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming plant operation at rated power and availability of 100% for the plant life time, the neutron fluence at the inner surface of the vessel causes a slight shift in the transition temperature. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assume that NDT temperature shifts are accounted for in the reactor operation. The Technical Specifications describe the mechanism for incorporating temperature shifts caused by the neutron flux.

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

For further discussion, see the following sections:

a. Design of structures, components, equipment, and systems 3

b. Integrity of RCPB 5.2

c. Reactor vessel 5.3

3.1.2.4.3 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) appropriate material surveillance program for the reactor vessel (GDC 32).

Evaluation Against Criterion 32

The RPV design includes provisions for inservice inspection. Removable plugs in the sacrificial 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 feedwater systems extending out to and including the first isolation valve outside containment. Inspection of the

RCPB is in accordance with the ASME Code Section XI. Section 5.2.4 defines the Inservice Inspection Program Plan, access provisions, and areas of restricted access.

Vessel material surveillance capsules are located at the 30° and 120°azimuth locations in the vessel adjacent to the wall. Specimens of the base metal, weld metal, and heat-affected zone metal are located in each capsule.

The plant testing and inspection program ensures that the requirements of Criterion 32 will be met.

For further discussion, see the following sections:

a.	Design of structures, components, equipment, and systems	3
b.	Inservice inspection and testing of RCPB	5.2.4
c.	Reactor vessel	5.3
d.	Component and subsystem design	5.4

3.1.2.4.4 Criterion 33 - Reactor Coolant Makeup

> A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to 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 (GDC 33).

Evaluation Against Criterion 33

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines and changes in reactor water level. The allowable leakage rates have been based on predicted and experimentally determined behavior

of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that in the absence of normal ac power concomitant with a loss of feedwater supply, makeup capabilities are provided by the RCIC system.

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:

a.	Detection of leakage through RCPB	5.2.5
b.	RCIC system	5.4.6
c.	ECCS	6.3
d.	Instrumentation and controls	7
e.	Electric power	8

3.1.2.4.5 Criterion 34 - Residual Heat Removal

> A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to 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 (GDC 34).

Evaluation Against Criterion 34

The RHR system provides the means to remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed.

The major equipment of the RHR system consists of heat exchangers cooled by the service water system 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 RHR system consists of two modes of operation that provide the shutdown cooling function. One mode, the normal Shutdown Cooling Mode, is the preferred operational mode. Although preferred, this mode of RHR does not meet the redundancy/single failure requirements of IEEE 279 and 10 CFR 50 Appendix A, GDC 34. As a result, a second shutdown cooling mode, the Alternate Shutdown Cooling Mode, is available and is the shutdown cooling mode credited to meet the requirements of IEEE 279 and GDC 34. This mode is safety related, Quality Class 1, Seismic Category 1, redundant and single failure proof. Since the normal Shutdown Cooling Mode of RHR is preferred for CGS, the components required for the operation of this mode are maintained as safety related, Quality Class 1. Refer to Section 5.4.7 for a complete discussion of the shutdown cooling modes of RHR.

Both normal ac power and the auxiliary onsite power system 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 which supply power to ESF, RPS, and auxiliaries required for safe shutdown are connected by appropriate switching to the standby diesel generators.

Each power source, up to the 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 physically separate essential bus sections, standby diesel generators, switchgear, interconnections, feeders, power centers, motor control centers, and other system components.

Standby diesel generators are provided to supply electrical power from within the plant that is not dependent on external sources. 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.

The RHR system is adequate to remove residual heat from the reactor core to ensure that fuel and RCPB design limits are not exceeded. Redundant reactor coolant circulation paths are available to and from the vessel and RHR system. 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:

a.	RHR system	5.4
b.	ECCS	6.3
c.	ESF systems	7.3
d.	Auxiliary power system	8.3
e.	Water systems	9.2
f.	Accident analyses	15

3.1.2.4.6 Criterion 35 - Emergency Core Cooling

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

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to 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 (GDC 35).

Evaluation Against Criterion 35

The ECCS, consists of the following:

a. High-pressure core spray system (HPCS),

b. Automatic depressurization system (ADS),

c. Low-pressure core spray (LPCS) system, and

d. Low-pressure coolant injection (LPCI) (an operating mode of the RHR system).

The ECCS 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 HPCS system consists of a single motor-driven pump, system piping, valves, controls, and instrumentation. The HPCS system is provided to ensure 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 operation maintains core cooling. A source of water is available from either the condensate storage tank or the suppression pool.

The automatic depressurization system 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 several of the main steam safety/relief valves (MSRVs) to relieve the high pressure steam to the suppression pool.

The LPCS system consists of (a) a centrifugal pump that can be powered by normal auxiliary power or the standby ac power system, (b) a spray sparger in the reactor vessel, above the core (separate from the HPCS sparger), (c) piping and valves to convey water from the suppression pool to the sparger, and (d) associated controls and instrumentation. In case of low water level in the reactor vessel or high pressure in the drywell, the LPCS 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 operate 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.

Results of the performance of the ECCS for the entire spectrum of liquid line breaks are discussed in Section 6.3. Peak cladding temperatures are well below the 2200°F design basis.

Also provided in Section 6.3.3 is an analysis to show that the ECCS conform to 10 CFR Part 50, Appendix K. This analysis shows complete compliance with the Final Acceptance Criteria with the following results:

a. Peak clad temperatures are well below the 2200°F NRC acceptability limit,

b. The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1% acceptability limit,

c. The clad temperature transient is terminated while core geometry is still amenable to cooling, and

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

For further discussion, see the following sections:

a.	RHR system	5.4.7
b.	ECCS	6.3
c.	Onsite power systems	8.3
d.	Water systems	9.2
e.	Accident analyses	15

3.1.2.4.7 Criterion 36 - Inspection of Emergency Core Cooling System

> The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system (GDC 36).

Evaluation Against Criterion 36

The ECCS discussed in Criterion 36 include inservice inspection considerations. The spray spargers within the vessel are accessible for inspection during each refueling outage. Removable plugs in the reactor 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 the drywell. Inspection of the ECCS is in accordance

with the intent of Section XI of the ASME Code. Sections 5.2.4 and 6.6 define the Inservice Inspection Program Plan, access provisions, and areas of restricted access.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time. Components inside the primary containment can be inspected when the primary containment 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 which are part of the RCPB are designed to specifications for inservice inspection to detect defects which might affect the cooling performance. Particular attention will be given to the reactor nozzles, core spray, and feedwater spargers. The design of the reactor vessel and internals and the plant testing and inspection program ensure that the requirements of Criterion 36 are met.

For further discussion, see the following sections:

a. Reactor pressure vessel internals 3.9.5

b. Inservice inspection and testing of RCPB 5.2.4

c. Reactor vessel 5.3

d. ECCS 6.3

e. Inservice inspection of Class 2 and 3 components 6.6

3.1.2.4.8 Criterion 37 - Testing of Emergency Core Cooling System

> The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to 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 (GDC 37).

Evaluation Against Criterion 37

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

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

The pumps and valves of these systems are tested and flow rate tests conducted periodically to verify operability in accordance with the CGS Inservice Testing Program Plan and Technical Specifications. Flow rate tests are conducted on HPCS, LPCS, and LPCI systems.

The ECCS are tested under conditions as close to design as practical to verify the performance of the full operational sequence that brings each system into operation, in accordance with the Technical Specifications surveillance requirements. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46. The design of the ECCS meets the requirements of Criterion 37.

For further discussion, see the following sections:

a.	Overpressurization protection	5.2.2
b.	Tests and inspections	6.3.4
c.	ESF systems	7.3
d.	Electric power	8

3.1.2.4.9 Criterion 38 - Containment Heat Removal

> A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident 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 electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure (GDC 38).

Evaluation Against Criterion 38

In the event of a LOCA within the drywell, the pressure suppression system will rapidly condense the steam to prevent overpressure. An increase in pressure in the drywell from a leak in the RCPB is relieved below the surface of the suppression pool and the steam portion of the gases so vented are condensed by contact with the suppression pool water. Cooling

systems remove heat from the core and from the water in the suppression pool during accident conditions and thus provide continuous cooling of the drywell.

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

The containment heat removal function is accomplished by the RHR system. Following a LOCA, one or both of the following operating modes of the RHR system would be initiated:

a. Containment spray - condenses steam within the containment, and

b. Suppression pool cooling - limits the temperature within the containment by removing heat from the suppression pool water by way of the RHR heat exchangers. Either or both redundant RHR heat exchangers can be manually activated.

The redundancy and capability of the offsite and onsite electrical power systems for the RHR system are presented in the evaluation against Criterion 34.

For further discussion, see the following sections:

a.	RHR system	5.4.7
b.	Containment systems	6.2
c.	ECCS	6.3
d.	ESF systems	7.3
e.	Electric power	8
f.	Standby service water system	9.2.7
g.	Accident analyses	15

3.1.2.4.10 Criterion 39 - Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system (GDC 39).

Evaluation Against Criterion 39

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

The suppression pool is designed to permit periodic inspection. 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:

a. RHR system 5.4.7

b. Containment systems 6.2

c. ECCS 6.3

d. ESF systems 7.3

e. Standby service water system 9.2.7

3.1.2.4.11 Criterion 40 - Testing of Containment Heat Removal System

> The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems, 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 (GDC 40).

Evaluation Against Criterion 40

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 testing. The pumps and valves of the RHR system are operated periodically to verify operability in accordance with the Inservice Testing Program Plan. The suppression pool cooling mode is not automatically initiated, and the operation of the components of this system is periodically verified. The operation of associated cooling water systems is discussed in the response to Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion see the following sections:

a. RHR system 5.4.7

b. ECCS 6.3

c. Electric power 8

3.1.2.4.12 Criterion 41 - Containment Atmosphere Cleanup

> Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to 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 electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety can be accomplished, assuming a single failure (GDC 41).

Evaluation Against Criterion 41

A primary containment that provides a barrier to control the release of fission products, hydrogen, oxygen, and other substances to the atmosphere is totally enclosed within the reactor building. Following an accident the reactor building is maintained at a negative pressure with respect to atmosphere to ensure that any leakage from the primary containment does not leak to atmosphere but is retained in the reactor building. The air exhausted from the reactor building to maintain it at a negative pressure is processed through the SGTS, which contains both HEPA and charcoal filters to minimize the release of radioactivity to the environs. The primary containment atmosphere can also be purged through the standby gas treatment for

cleanup prior to entry of personnel. The SGTS is composed of two fully redundant trains of filters and fans, with all active components powered from the emergency diesel buses.

The above described systems meet the requirements of Criterion 41.

For further discussion see the following sections:

a.	Principal design criteria	1.2
b.	Containment functional design	6.2.1
c.	Secondary containment functional design	6.2.3
d.	Combustible gas control in containment	6.2.5
e.	Fission product removal and control systems	6.5
f.	Instrumentation and controls	7

3.1.2.4.13 Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

> The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems (GDC 42).

Evaluation Against Criterion 42

All components of the SGTS are located in the reactor building and are accessible for inspection during normal plant operation. The design of the system therefore meets the requirements of Criterion 42 (see Section 6.5).

3.1.2.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

> The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems (GDC 43).

Evaluation Against Criterion 43

As discussed under the evaluation of Criterion 42, all components of the containment atmosphere cleanup system are accessible for tests and inspections during normal plant operation thereby meeting the requirements of Criterion 43. A detailed discussion of the periodic tests that will be performed to verify system operability is given in Section 6.5.

3.1.2.4.15 Criterion 44 - Cooling Water

> A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal 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 electrical power system operation (assuming offsite electrical 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 (GDC 44).

Evaluation Against Criterion 44

The safety-related cooling water system is the standby service water (SW) system, which supplies cooling for the RHR, HPCS, LPCS, fuel pool cooling and cleanup (FPC) system, and the essential HVAC systems.

The redundant SW systems are open loop systems which transfer heat from structures, systems, and safety-related components to the ultimate heat sink.

The ultimate heat sink consists of two man-made Seismic Category I spray ponds and is designed to withstand extreme natural phenomena.

The piping, valves, pumps, and heat exchangers of the SW system are designed and arranged so that the system safety function can be performed assuming a single failure. Electrical power is supplied to the system from offsite or onsite emergency power sources, with distribution arranged such that a single failure will not prevent the system from performing its safety function.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b.	Classification of structures, components, and systems	3.2
c.	Wind and tornado loadings	3.3
d.	Water level (flood) design	3.4
e.	Missile protection	3.5
f.	Design of Seismic Category I structures	3.8
g.	Ultimate heat sink	9.2.5
h.	Standby service water system	9.2.7
i.	Electrical power	8

3.1.2.4.16 Criterion 45 - Inspection of Cooling Water System

> The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system (GDC 45).

Evaluation Against Criterion 45

The SW system is designed to permit periodic inspection of important components, including pumps, strainers, heat exchangers, and isolation valves, to ensure the integrity and capability of the system.

All important components are located in accessible locations to facilitate periodic inspection during normal plant operation. Suitable manholes, handholes, inspection ports, or other design and layout features are provided for this purpose.

These features meet the requirements of Criterion 45.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Inservice inspection of Class 2 and 3 components	6.6
c.	Standby service water system	9.2.7
d.	Initial test program	14

3.1.2.4.17 Criterion 46 - Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources (GDC 46).

Evaluation Against Criterion 46

The SW system is designed to permit system operability testing with simulation of emergency reactor shutdown or LOCA conditions and transfer between normal and emergency power sources.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Standby service water system	9.2.7
c.	Electric power	8
d.	Initial test program	14

3.1.2.5 Group V - Containment (Criteria 50-57)

3.1.2.5.1 Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and 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 loss-of-coolant accident. 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 (GDC 50).

Evaluation Against Criterion 50

The containment structure, including access openings and penetrations, are designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a LOCA up to and including a double-ended rupture of the largest reactor coolant pipe.

The containment structure and ESF systems have been evaluated for various combinations of energy release. The analysis accounts for system thermal energy, chemical energy, and nuclear decay heat energy.

The maximum temperature and pressure reached in the drywell and containment during the worst case accident are shown in Chapters 6 and 15 to be below the design temperature and pressure of this structure.

The cooling capacity of the containment heat removal systems is adequate to prevent overpressurization of the structure and to return the containment to near atmospheric pressure.

For further discussion, see the following sections:

a.	Classification of structures, components, and systems	3.2
b.	Wind and tornado loadings	3.3
c.	Missile protection	3.5
d.	Protection against dynamic effects associated with the postulated rupture of piping	3.6
e.	Seismic design	3.7
f.	Steel containment	3.8
g.	Containment systems	6.2
h.	Containment heat removal systems	6.2.2
i.	Accident analyses	15

3.1.2.5.2 Criterion 51 - Fracture Prevention of Containment Pressure Boundary

> 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 (GDC 51).

Evaluation Against Criterion 51

The containment vessel was designed, fabricated, inspected, and tested to meet the requirements of ASME Code Section III Subsection NE (1971 edition through Summer 1972 Addenda). Material for the containment was qualified by impact testing at a temperature which is at least 30°F below the minimum service temperature of +30°F. Means are provided by an auxiliary boiler to maintain the reactor building and consequently the containment temperature at a suitable level during a shutdown of the unit during cold weather.

To demonstrate that the primary containment pressure boundary meets the requirements of Criterion 51, the Class 1, Class 2, and Class MC components of the CGS containment pressure boundary were reviewed according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing of PWR Steam Generator and Reactor Coolant Pump Supports." Based on review of the available fracture toughness data and material fabrication histories, and the use of correlations between metallurgical characteristics and material fracture toughness, it was concluded that the ferritic materials in the CGS containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with these code requirements provide reasonable assurance that the CGS reactor containment pressure boundary materials will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of Criterion 51 are satisfied.

The preoperational test program and the quality assurance program ensure the integrity of the containment and its ability to meet all normal operating and accident requirements.

The containment design thus meets the requirements of Criterion 51.

For further discussion, see the following sections:

a.	Steel containment	3.8.2
b.	Reactor building ventilation system including spent fuel pool area ventilation system	9.4
c.	Initial test program	14
d.	Quality assurance	17

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure (GDC 52).

Evaluation Against Criterion 52

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leak rate tests during the plant lifetime. The testing program will be conducted in accordance with Appendix J to 10 CFR Part 50.

For further discussion, see the following sections:

a.	Leak rate tests	3.8.2
b.	Containment leakage testing	6.2.6

3.1.2.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (c) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansions bellows (GDC 53).

Evaluation Against Criterion 53

The reactor containment design permits periodic inspection of the exposed interior surfaces. It also includes provisions for periodic testing of the leaktightness of all penetrations and inserts in the reactor containment pressure boundary.

The containment design therefore meets the requirements of Criterion 53.

For further discussion, see the following sections:

a.	Steel containment	3.8.2
b.	Containment functional design	6.2.1
c.	Containment heat removal systems	6.2.2
d.	Containment leakage testing	6.2.6
e.	Inservice inspection of Class 2 and 3 components	6.6

3.1.2.5.5 Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits (GDC 54).

Evaluation Against Criterion 54

Piping systems penetrating containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection tests to be performed.

The ESF actuation system test circuitry provides the means for testing isolation valve operability.

Conformance with Criterion 54 is further discussed in Sections 3.1.2.5.6 (Criterion 55), 3.1.2.5.7 (Criterion 56), and 3.1.2.5.8 (Criterion 57).

3.1.2.5.6 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

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

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

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

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as 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 (GDC 55).

Evaluation Against Criterion 55

The RCPB [as defined in 10 CFR Part 50, Section 50.2 (v)] consists of the RPV, pressure retaining appurtenances attached to the vessel, valves, and pipes which extend from the RPV up to and including the outermost isolation valve. The RCPB lines which penetrate the containment have isolation valves capable of isolating the containment to preclude any significant releases of radioactivity. Lines which do not penetrate the containment but which form a portion of the RCPB, can be isolated from the RCPB.

The design of the isolation systems described meet the requirements of Criterion 55.

For further discussion, see the following sections:

a. Integrity of RCPB 5.2

b.	Containment isolation system	6.2.4
c.	Instrumentation and controls	7
d.	Accident analyses	15

3.1.2.5.7 Criterion 56 - Primary Containment Isolation

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

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

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

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

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

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety (GDC 56).

Evaluation Against Criterion 56

Criterion 56 requires that lines which penetrate the containment and communicate with the containment interior must have two isolation valves; one inside the containment, the other outside. It should be noted that this criterion does not reflect consideration of the BWR suppression pool design. For instance, those lines which connect to the suppression pool do not have an isolation valve located inside the containment as this would necessitate placement of the valve underwater. In effect, this would result in introducing a potentially unreliable valve in a highly reliable system thereby compromising design.

The manner in which the containment isolation system meets this requirement is described in the following sections:

a.	Containment isolation system	6.2.4
b.	Instrumentation and control systems	7
c.	Primary containment and reactor vessel isolation control system instrumentation and controls	7.3
d.	Accident analyses	15

3.1.2.5.8 Criterion 57 - Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve (GDC 57).

Evaluation Against Criterion 57

Each line that penetrates the reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere has at least one containment isolation valve which is automatic, locked closed or capable of remote manual operation, located outside the containment as close to the containment as practical. Simple check valves are not used as automatic isolation valves on these lines.

Details demonstrating conformance with Criterion 57 are provided in the following sections:

a.	Containment isolation system	6.2.4
b.	Instrumentation and controls	7

3.1.2.6 Group VI - Fuel and Reactivity Control

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

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated

operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment (GDC 60).

Evaluation Against Criterion 60

Waste handling systems have been incorporated in the plant design for processing and/or retention of radioactive wastes from normal plant operations to ensure that the effluent releases to the environment are as low as reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. The plant is also designed to prevent radioactivity releases during accidents from exceeding the limits of 10 CFR Part 100.

The principal gaseous effluents from the plant during normal operation are the noncondensable gases from the air ejectors. These gases are exhausted through a hold up system and offgas treatment system including charcoal adsorbers. The effluent from this system is continuously monitored and controlled and the system will be shutdown and isolated in the event of abnormally high radiation levels.

Ventilation air from the various plant areas is continuously monitored and controlled. In the event of an accident inside containment, noncondensable gases are held within the leaktight containment vessel. Release of these effluents will be by controlled purging of the containment.

Liquid radioactive wastes are collected in waste collector tanks, treated on a batch basis through demineralizers, and then either returned to the plant systems or released in a controlled manner to the environment. All discharges to the environment are routed through a monitoring station that continuously monitors and records the activity of the waste and provides an alarm to the operator in the unlikely event of high activity level.

Solid wastes including spent resins, filter sludges, filter cartridges, evaporator bottoms, contaminated tools, equipment, and clothing are collected, packaged, and shipped offsite in approved shipping containers.

The design of the waste disposal systems meets the requirements of Criterion 60.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Detection of leakage through RCPB 5.2.5

c.	Containment functional design	6.2.1
d.	Heating, ventilating, and air conditioning systems	9.4
e.	Liquid waste management systems	11.2
f.	Gaseous waste management systems	11.3
g.	Solid waste management system	11.4
h.	Process and effluent radiological monitoring and sampling systems	11.5
i.	Radiation protection	12
j.	Accident analyses	15

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to 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 residual heat removal 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 (GDC 61).

Evaluation Against Criterion 61

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

3.1.2.6.2.2 Spent Fuel Handling and Storage. Irradiated fuel is stored in the spent fuel pool in the reactor building. Fuel pool water is circulated through the FPC system to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental critically (see evaluation against Criterion 62).

Reliable decay heat removal is provided by the closed loop FPC system. It consists of two circulating pumps, two heat exchangers, two filter-demineralizers, two skimmer surge tanks, and the associated piping, valves, and instrumentation. The pool water is circulated through the system, suction is taken from fuel pool skimmer surge tanks, flow passes in series through the heat exchangers and filters, and it is discharged through diffusers at the bottom of the fuel pool. Normal source of cooling water to the heat exchangers is from the reactor closed cooling water (RCC) system. When the preferred RCC source is unavailable, the SW system provides cooling water and makeup. The FPC system can also be interconnected with the RHR system to increase the cooling capacity of the FPC system. See Section 9.1.3.

The portion of the system required for cooling of spent fuel after refueling is designed to Seismic Category I requirements and can be isolated by automatic redundant, Seismic Category I isolation valves from the Seismic Category II cleanup portion of the system. Safety-grade cooling water from the SW system is available to the shell side of the FPC heat exchangers by remote-manual operation from the control room. Safety-grade makeup water to the fuel pool is also available from the SW system by remote-manual operation from the control room.

High- and low-level switches indicate pool water level changes in the main control room. Fission product concentration in the pool water is minimized by use of the filter-demineralizer. This minimizes the release of radioactivity from the pool to the reactor building environment.

No special tests are required to ensure system operability, except as noted below, because at least one pump, with a heat exchanger, and filter-demineralizer is routinely in operation while fuel is stored in the pool. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system operability.

Service water flow to the shell side of the fuel pool cooling heat exchangers, including operation of the valves that interconnect the service water and RCC systems to the FPC system, are tested in conjunction with testing of the service water system. The service water makeup isolation valves are also tested in conjunction with testing of the service water system.

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

Liquid radwastes are segregated and treated as equipment drain, floor drain, chemical, detergent, sludges, or concentrated wastes. Processing methods include filtration, ion exchange, neutralization, concentration, solidification, analysis, and dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. Wet solid wastes and concentrates are normally dewatered and packaged in steel or polyethylene containers as required for disposal. Dry solid radwastes are packaged in steel boxes or other suitable containers. 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 refueling and radwaste buildings have sufficient shielding to maintain dose rates within the limits set forth in 10 CFR Part 20 and 10 CFR Part 50. The radwaste building is designed to preclude accidental release of radioactive materials to the environs.

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

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

For further discussion, see the following sections:

a.	RHR system	5.4.7
b.	Containment systems	6.2
c.	New fuel storage	9.1.1
d.	Spent fuel storage	9.1.2
e.	Spent fuel pool cooling and cleanup system	9.1.3
f.	Heating, ventilating, and air conditioning systems	9.4
g.	Radioactive waste management	11
h.	Radiation protection	12

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

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations (GDC 62).

Evaluation Against Criterion 62

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new and spent fuel storage is prevented by the

geometrically safe configuration of the new fuel storage racks and by the geometrically safe configuration of the spent fuel racks using fixed poison. There is sufficient spacing between assemblies to ensure that the array when fully loaded is substantially subcritical. Fuel elements are limited to top loading and fuel assembly positions by rack design. The new and spent fuel racks are Seismic Category I components.

Spent fuel and new fuel is stored under water in the spent fuel pool. The racks in which spent fuel assemblies are placed are designed to ensure subcriticality in the storage pool. Fuel is maintained at a subcritical multiplication factor k_{eff} of less than 0.95 under both normal and abnormal conditions. Abnormal criticality conditions may result for abnormal conditions. Abnormal criticality conditions may result from accidental dropping of a fuel assembly to a location adjacent to or on top of a loaded fuel storage rack. Various abnormal loading conditions such as loadings from the operating basis earthquake (OBE) and safe shutdown earthquake (SSE), the dropping of a fuel assembly, or a stuck assembly exerting an upward force on the racks were also considered in the design of the racks.

Refueling interlocks include circuitry which senses conditions of the refueling equipment and the control rods. These interlocks reinforce operational procedures that prohibit making the reactor critical. The fuel handling system is designed to provide a safe, effective means of transporting and handling fuel and to minimize the possibility of mishandling or maloperation.

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

For further discussion, see the following sections:

a.	Refueling interlocks system instrumentation and controls	7.7.1.13
b.	New fuel storage	9.1.1
c.	Spent fuel storage	9.1.2
d.	Fuel handling system	9.1.4

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas, (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions (GDC 63).

Evaluation Against Criterion 63

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the FPC system which could result in loss of RHR capability and excessive radiation levels is alarmed in the control room. Alarmed conditions include high/low level in the fuel pool, cooling water pump discharge pressure low, and high/low level in the fuel storage pool skimmer surge tanks. System temperature is also continuously monitored and alarmed in the control room. The area radiation monitors sense radioactivity in this area and initiate an alarm in the control room on abnormal radiation levels.

Area radiation, tank, and sump levels are monitored and alarmed to give indication of conditions that may result in excessive radiation levels in radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

For further discussion, see the following sections:

a.	Area radiation monitoring system instrumentation	12.3.4
b.	Fuel storage and handling	9.1
c.	Radioactive waste management	11
d.	Radiation protection	12

3.1.2.6.5 Criterion 64 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident 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 (GDC 64).

Evaluation Against Criterion 64

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

a. Main plant vent monitor,

b. Turbine building ventilation exhaust,

c. Radwaste building ventilation exhaust,

d. Liquid radwaste system effluent,

e. Standby service water system,

f. Plant service water,

g. Containment monitoring system radiation monitors, and

h. Turbine building sumps.

In addition, offsite monitors are provided.

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

a.	Detection of leakage through RCPB	5.2.5
b.	Process radiation monitoring system instrumentation and controls	7.6
c.	Process and effluent radiological monitoring systems and sampling systems	11.5
d.	Area radiation and airborne radioactivity monitoring instrumentation	12.3.4

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Appendix A of 10 CFR Part 50 requires that all nuclear power plants be designed so that, if the safe shutdown earthquake (SSE) occurs, all structures, systems, and components (SSCs) important to safety remain functional.

General Design Criterion 1 (GDC 1) of Appendix A to 10 CFR Part 50 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

3.2.1 SEISMIC CLASSIFICATION

Plant structures, systems, and components important to safety are designed to withstand the effects of an SSE and remain functional if they are necessary to ensure

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

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

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposure comparable to the exposure limits of 10 CFR Part 50.67.

Plant structures, systems, equipment, and components, including their foundations and supports, designed to remain functional in the event of an SSE are designated as Seismic Category I, as indicated in Table 3.2-1.

All Seismic Category I SSCs are analyzed under the loading conditions of the SSE and the operating basis earthquake (OBE). Since the two earthquakes vary in intensity, the design of Seismic Category I structures, components, equipment, and systems to resist each earthquake and other loads are based on levels of material stress or load factors, whichever is applicable, with margins of safety appropriate for each earthquake. The margin of safety provided for Seismic Category I structures, components, equipment, and systems for the SSE are sufficient to ensure that their design functions are not jeopardized.

For further details of seismic design criteria refer to

Seismic design	3.7
Design of Category I structures	3.8
Systems and components	3.9

Seismic qualification of Seismic Category I instrumentation and electrical equipment	3.10

All SSCs not analyzed under the loading conditions of the SSE and OBE are classified Seismic Category II. Where applicable, the seismic loading conditions as determined from the Uniform Building Code are used for the design of Seismic Category II SSCs.

The OBE, as defined in 10 CFR Part 100, Appendix A, is not incorporated as part of the seismic classification scheme.

The seismic classification indicated in Table 3.2-1 meets the requirements of NRC Regulatory Guide 1.29, Revision 3, except as otherwise noted in the table.

Seismic Category 1M classification denotes systems, structures or components that are designed and constructed to comply with position C.2 of Regulatory Guide 1.29.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS

System quality group classifications, as defined in NRC Regulatory Guide 1.26, Revision 3, are determined for each water, steam, or radioactive waste containing component of those applicable fluid systems relied on to ensure

a. The integrity of the RCPB,

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

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposure comparable to the exposure limits of 10 CFR Part 50.67.

The quality group classifications for systems and components which meet these definitions are shown in Table 3.2-1.

System quality group classifications, as indicated in Tables 3.2-1 and 3.2-2, meet the requirements of 10 CFR 50.55a and Regulatory Guide 1.26, Revision 3.

Note: Classification of the RWCU system in the Radwaste Building takes exception to Reg. Guide 1.26. Piping, pumps, valves, and supports of the RWCU system in the Radwaste Building have been reclassified as Quality Group D, taking exception to the Reg. Guide 1.26 recommended classification of Quality Group C.

3.2.3 SAFETY CLASSIFICATIONS

Structures, systems, and components were classified as Safety Class 1, Safety Class 2, Safety Class 3, or General Class G in accordance with the ANS-22 definition of their importance to nuclear safety (ANS-22 became ANSI/ANS-52.1-1983). Recognizing that components within a system may be of differing safety importance, a single system may have components in more than one safety class. Supports are designed in accordance with ASME III-Winter 1973 Addenda-Subsection NF and are classified appropriately based on the component supported. Jurisdictional boundaries between NF and other codes were established by the owner as required by the ASME Code. Load combinations and allowable stresses used in support design are in accordance with ASME III-NF as defined in Section 3.9 and are applied to both ASME and the American Institute of Steel Construction (AISC) support members. Modifications and construction practices are controlled by the quality classification of the individual support and by the additional requirements of each applicable code. Table 3.2-1 provides a summary of the safety classes for the principal SSCs of the plant.

Design requirements for components of safety classes are also described in this section. Reference is made to accepted industry codes and standards which define design requirements commensurate with the safety function(s) performed.

3.2.3.1 Safety Class 1

3.2.3.1.1 Definition of Safety Class 1

Safety Class 1 applies to components of the RCPB or core support structure whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system or to equipment in which a single failure could cause major fuel damage.

3.2.3.1.2 Design Requirements for Safety Class 1

The design requirements for Safety Class 1 mechanical equipment (i.e., vessel, pipes, pumps, and valves) are described in Section 3.9. The design requirements for the reactor vessel are described in Section 5.3.

The design requirements for Safety Class 1 structures (i.e., reactor pedestal and supports of the RCPB as defined in Section 3.2.3.1.1) are specified in Sections 3.8 and 3.9.

Safety Class 1 SSCs are identified in Table 3.2-1.

3.2.3.2 Safety Class 2

3.2.3.2.1 Definition of Safety Class 2

Safety Class 2 applies to those SSCs, other than service water systems, that are not Safety Class 1 but are necessary to accomplish the safety function of

a. Inserting negative reactivity to shut down the reactor,
b. Preventing rapid insertion of positive reactivity,
c. Maintaining core geometry appropriate to all plant process conditions,
d. Providing emergency core cooling,
e. Providing and maintaining containment, and
f. Removing residual heat from the reactor and reactor core.

3.2.3.2.2 Design Requirements for Safety Class 2

The design requirements for Safety Class 2 mechanical equipment (i.e., vessels including heat exchangers, pipes, pumps, valves, and tanks) are described in Section 3.9.

The design requirements for Safety Class 2 structures (i.e., reactor building including the primary containment) are specified in Section 3.8.

The seismic and environmental design requirements for Safety Class 2 instrumentation and electrical equipment are specified in Sections 3.10 and 3.11.

Safety Class 2 SSCs are identified in Table 3.2-1.

3.2.3.3 Safety Class 3

3.2.3.3.1 Definition of Safety Class 3

Safety Class 3 applies to those SSCs that are not Safety Class 1 nor Safety Class 2 that is relied upon to accomplish a nuclear safety function.

3.2.3.3.2 Design Requirement for Safety Class 3

The design requirements for Safety Class 3 structures, (i.e., standby service water pump houses, diesel generator buildings, radwaste/control building, and spray ponds) are specified in Section 3.8.

The design requirements for Safety Class 3 mechanical equipment (i.e., vessels including heat exchangers, pipes, pumps, valves, and tanks) are described in Section 3.9.

The seismic and environmental design requirements for Safety Class 3 instrumentation and electrical equipment are specified in Sections 3.10 and 3.11.

3.2.3.4 General Class G, Structures, Systems, and Components

3.2.3.4.1 Definition of General Class Structures, Systems, and Components

A boiling water reactor (BWR) has a number of SSCs in the power conversion or other portions of the facility which have no direct safety function but which may be connected to or

influenced by the equipment within the safety classes defined above. Such SSCs are designated as General Class G. For example, portions of the service water systems, the turbine generator auxiliaries, and portions of the heating, ventilating, and air conditioning (HVAC) systems are designated as having no safety classification.

3.2.3.4.2 Design Requirements for General Class G Structures, Systems, and Components

The design requirements for equipment classified as General Class G are specified with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Design requirements are based on applicable industry codes and standards.

3.2.4 QUALITY ASSURANCE CLASSIFICATION

Structures, systems, and components are classified in Table 3.2-1 using the following quality class designations

a. Quality Class I - Any nuclear system, structure, subassembly, component, or design characteristics that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. All engineered safety features fall within this category. All Quality Class I items meet the applicable provisions of 10 CFR Part 50 Appendix B.

b. Quality Class II+ - Assigned to SSCs having no safety-related function but requiring quality augmentation either as a result of NRC requirements or as committed by CGS. Quality augmentation may include such requirements as environmental qualification, seismic qualification, or other quality affecting activities specifically committed. Augmented quality applies to the following categories of SSCs:

   1. Essential fire protection SSCs,

   2. Structures, systems, and components that do not perform a safety-related function but must be seismically supported/mounted (seismic 2-over-1) per Regulatory Guide 1.29,

   3. Structures, systems, and components required for radwaste management that are subject to Regulatory Guide 1.143,

   4. Structures, systems, and components required to cope with a station blackout per Regulatory Guide 1.155 (see Section 8A),

5. Structures, systems, and components required to respond to or mitigate anticipated transients without scram (ATWS) per the requirements of 10 CFR 50.62,

6. Structures, systems, and components required to respond to an electrical separation safe shutdown event (see Section 8.3.1.4.4.1.4),

7. Postaccident monitoring instruments subject to Regulatory Guide 1.97, Category 2, requirements,

8. Remote shutdown items required in response to control room evacuation (10 CFR 50 Appendix A, GDC 19) and,

9. Beyond Design Basis External Event (BDBEE) structures, systems, and components associated with Spent Fuel Level Instrumentation per requirements of NRC Order EA-12-051 and portions of the HCV system per the requirements of NRC Order EA-13-109.

c. Quality Class II - Any system, structure, subassembly, component, or design characteristic that could cause a safety hazard to plant personnel, an extended reduction in unit output, an unscheduled unit trip, or equipment damage. Appropriate quality assurance requirements for these items are assigned in the purchase specifications.

d. Quality Class G - Any nonnuclear system, structure, subassembly, component, or design characteristic to which quality assurance requirements are assigned in accordance with the consequences of failure, operating costs, or procurement costs.

3.2.5 CORRELATION OF SAFETY CLASSES WITH OTHER DESIGN REQUIREMENTS

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 Table 3.2-3.

3.2.6 IDENTIFICATION OF SAFETY-RELATED SYSTEMS AND COMPONENTS ON FLOW DIAGRAMS AND IN THE MASTER EQUIPMENT LIST

The system classification are shown on the flow diagrams using symbols code group A, B, C, D, D+; Seismic Category I, II; and Quality Class I, II, II+, and G. The Master Equipment List (MEL) uses arabic number designators for quality class and seismic category. The following is a comparison of the classification designators:

Flow Diagram	Table 3.2-1	MEL
Code Group A	Quality Group A	
Code Group B	Quality Group B	
Code Group C	Quality Group C	
Code Group D	Quality Group D	
Code Group D+	Quality Group D+	
Seismic Category I	Seismic Category I*	Seismic Category 1*
		Seismic Category 1M*
Seismic Category II	Seismic Category II*	Seismic Category 2*
Quality Class I	Quality Class I	Quality Class 1
Quality Class II	Quality Class II	Quality Class 2
Quality Class II+	Quality Class II+	Quality Class A
Quality Class G	Quality Class G	Quality Class G

Flow diagrams which include the seismic and code classifications assigned to each piping system are identified in Table 3.2-1. The flow diagrams delineate the boundary of seismic and code classifications for each system and present the as-built classifications. For the appropriate quality class of specific components, refer to the MEL. In some instances, these classifications reflect voluntary upgrades which may exceed Regulatory Guide 1.26, Revision 3, requirements. For inservice inspection requirements, the appropriate levels of inspection are given in Sections 5.2.4 and 6.6.

Note: Classification of the RWCU system in the Radwaste Building takes exception to Reg. Guide 1.26. Piping, pumps, valves, and supports of the RWCU system in the Radwaste Building have been reclassified as Quality Group D, taking exception to the Reg. Guide 1.26 recommended classification of Quality Group C.

* Table 3.2-1 may indicate Seismic Category I, 1M or II. Clarification of the Seismic 2 over 1 (i.e., 1M) support/mounting requirements is specified in the notes and/or MEL (see Table 3.2-1).

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Table 3.2-1

Equipment Classification

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
1. Reactor system							
Reactor vessel	GE	1	C	A	I	I	
Reactor vessel support skirt	GE	1	C	N/A	I	I	
Reactor vessel appurtenances pressure retaining portions	GE	1	C	A	I	I	
Control rod drive housing supports	GE	2	C	N/A	I	I	
Reactor internal structures, engineered safety features	GE	2	C	N/A	I	I	
Reactor internal structures, other	GE	G	C	N/A	N/A	N/A	
Control rods	GE	2	C	N/A	I	I	
Control rod drives	GE	2	C	N/A	I	I	
Core support structure	GE	2	C	N/A	I	I	
Power range detector hardware	GE	2	C	B	I	I	
Fuel assemblies	P	1	C	N/A	I	I	
Reactor vessel stabilizer	GE	2	C	N/A	I	I	
2. Nuclear boiler system (Figure 10.3-2)							
Vessels, level instrumentation condensing chambers	GE	1	C	A	I	I	
Vessels	P	2	C	B	I	I	
Piping, relief valve discharge from relief valve to suppression pool	P	3	C	C	I	I	
Piping, relief valve discharge within suppression chamber and suppression pool	P	2	C	B	I	I	1
Piping, main steam and feedwater within outermost isolation valve	GE/P	1	C,R	A	I	I	
Pipe supports, main steam	GE	1	C,R	A	I	I	
Pipe restraints, main steam	P	2	C,R	N/A	I	I	
Piping, other within outermost isolation valves	P	1	C,R	A	I	I	2
Safety/relief valves	GE	1	C	A	I	I	
Valves, main steam isolation valves	GE		C,R	A	I	I	
Valves, other, isolation valves and within containment	P	1	C	A	I	I	

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Piping and valves outboard of outside isolation valve to and including first valve capable of timely actuation	P	2	R	B	I	I	2
Mechanical modules, instrumentation (with safety function)	GE	2	C,R	N/A	I	I	
Electrical modules (with safety function)	GE	2	C,R	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
3. Reactor recirculation system (Figure 5.4-7)							
Piping	GE	1	C	A	I	I	2
Pipe suspension, recirculation line	GE	1	C	N/A	I	I	
Pipe restraints, recirculation line	GE	2	C	N/A	I	I	
Pumps	GE	1	C	A	I	I	
Valves	GE	1	C	A	I	I	2
Motor, pump	GE	3	C	N/A	II+	I/IM	37
Electrical modules (with safety function)	GE	2	C,R	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
4. Control rod drive hydraulic system (Figure 4.6-5)							
Valves, scram discharge volume lines	GE/P	2	R	B	I	I	2
Valves, insert and withdraw lines	GE	2	R	B	I	I	3
Valves, other	P	G	R	D	II	II	2,4
Piping discharge volume lines	P	2	R	B	I	I	
Piping insert and withdraw lines	GE/P	2	C,R	B	I	I	
Piping, other	P	G	R	D	II	II	2,4
Hydraulic control unit	GE	2	R	Special	I	I	5
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cables (with safety function)	P	2	C,R,W	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
5. Standby liquid control system (Figure 9.3-14)							6
Standby liquid control tank	GE	2	R	B	II+	I	
Pump	GE	2	R	B	II+	I	
Pump motor	GE	2	R	N/A	II+	I	
Valves, explosive	GE	1	R	A	I	I	
Valves, isolation and within containment	P	1	C	A	I	I	2
Valves, beyond isolation valves	P	2	R	B	II+	I	2
Piping, within isolation valves to reactor vessel	P	1	C,R	A	I	I	2
Piping, beyond isolation valves	P	2	R	B	II+	I	2
Electrical modules (with safety function)	GE	2	C	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
Electrical modules, with boron injection function	GE	2	C	N/A	II+	I	
Cable, with boron injection function	P	2	C,R,W	N/A	II+	I	
6. Neutron monitoring system							
Tubing TIP	GE/P	2/G	C	B/D	I/II	I/II	
Electrical modules, IRM and APRM	GE	2	C,R	N/A	I	I	
Cable, IRM and APRM	P	2	C,R,W	N/A	I	I	
Valves, isolation TIP subsystem	GE/P	2	C,R	B	I	I	
Power range detector hardware	GE	2	R	B	I	I	
7. Reactor protection							
Electrical modules	GE	2	C,R,T	N/A	I	I	
Cable	P	2	C,R,T,W	N/A	I	I	
8. Leak detection system							7
Temperature element	GE	2	C,R	N/A	I	I	
Differential temperature switch	GE	2	C,R	NA/	I	I	
Differential flow indicator	GE	2	C,R	N/A	I	I	
Pressure switch	GE	2	C,R	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Differential pressure indicator switch	GE	2	C,R	N/A	I	I	
Differential flow summer	GE	2	C,R	N/A	I	I	
9. Process radiation monitors							
Electrical modules, main steam line and building ventilation monitors	GE	2	C,R,T,W	N/A	I	I	
Cable, main steam line and reactor building ventilation monitors	P	2	C,R,T,W	N/A	I	I	
10. Residual heat removal system (Figure 5.4-15)							
Heat exchangers, primary side	GE	2	R	B	I	I	
Heat exchanger, secondary side	GE	3	R	C	I	I	
Piping, within outermost isolation valves, reactor coolant pressure boundary	P	1	C,R	A	I	I	
Piping, other	P	2	R	B	I	I	8
Pumps	GE	2	R	B	I	I	
Water leg pumps	P	2	R	B	I	I	
Main Pump motors	GE/P	2	R	N/A	I	I	
Valves, isolation reactor coolant pressure boundary	P	1	C,R	A	I	I	
Valves, other	P	2	R	B	I	I	2
Mechanical modules	GE	2	R	B	I	I	
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
11. Low-pressure core spray (Figure 6.3-4)							
Piping, within outermost isolation valves to reactor vessel	P	1	C,R	A	I	I	2
Piping, beyond outermost isolation valves	P	2	R	B	I	I	2
Pumps	GE	2	R	B	I	I	
Water leg pumps	P	2	R	B	I	I	
Main Pump motors	GE	2	R	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Valves, isolation, reactor coolant pressure boundary	P	1	C	A	I	I	2
Valves, other	P	2	C,R	B	I	I	2
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	R,W	N/A	I	I	
12. High-pressure core spray (Figure 6.3-4)							
Piping, within outermost isolation valve	P	1	C,R	A	I	I	2
Piping, return test line to condensate storage tank beyond second isolation valve	P	G	R,O	D	II	II	4
Piping, beyond outermost isolation valve, other	P	2	R	B	I	I	2
Pump	GE/P	2	R	B	I	I	
Water leg pumps	P	2	R	B	I	I	
Main Pump motor	GE	2	R	N/A	I	I	
Valves, beyond diesel shutoff valves	P	3	P	C	I	I	
Valves, isolation, reactor coolant pressure boundary	P	1	C	A	I	I	
Valves, beyond isolation valves, motor operated	GE	2	R	B	I	I	2
Valves, other	P	2	R,P	B	I	I	
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Electrical auxiliary equipment	GE	3	DG	N/A	I	I	
Cable (with safety function)	P	2	W,R	N/A	I	I	
(See 38a - high-pressure core spray standby power supply)							
13. Reactor core isolation cooling system (Figure 5.4-11)							
Piping, within outermost isolation valves reactor coolant pressure boundary	P	1	C,R	A	I	I	2
Piping, beyond outermost isolation valves	P	2	R	B	I	I	2,10
Piping, drip pot discharge valve to condenser	P	G	R	D	II	II	4

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Piping, condenser to vacuum tank and to the condensate pump discharge; and vacuum pump discharge to the outboard check valve break flange	P	G	R	D	I	I	2
Pump	GE	2	R	B/D	I	I	
Water leg pumps	P	2	R	B	I	I	2
Valves, isolation reactor coolant pressure boundary	P	1	C, R	A	I	I	2
Valves, reactor coolant pressure boundary	P	1	R	A	I	I	2
Valves, containment isolation	P	2	R	B	I	I	
Valves, other (with safety function)	P	2	R	B/D	I	I	
Turbine	GE	2	R	N/A	I	I	9
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	R,W	N/A	I	I	
Reactor core isolation cooling electrohydraulic system	GE	2	R	N/A	I	I	
14. Fuel service equipment							
Fuel preparation machine	GE	3	R	N/A	I	I	
General purpose grapple	GE	3	R	N/A	I	I	
New fuel inspection stand	GE	G	R	N/A	II+	1M	
15. Refueling equipment							
Refueling platform	GE	G	R	N/A	II+	1M	
Refueling bellows	P	G	C,R	D	II+	I	11
16. Storage equipment							
Fuel storage racks	GE/P	3	R	N/A	I	I	
17. Radwaste system (Figures 11.2-2, 9.3-9, 9.3-12, 11.2-3, and 11.2-4)							2,4,13, 14,36
Tanks, atmospheric	GE/P	G	W	D+	II+	II	
Heat exchangers	GE/P	G	W	D+	II+	II	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Piping and valves forming part of containment boundary	P	2	C,R	B	I	I	
Piping and valves forming part of secondary containment	P	3	R	C,D	I,II+	I	36
Piping, other	P	G	W	D+	II+	II	
Pumps	GE/P	G	W	D+	II+	II	
Valves, flow control and filter system	GE/P	G	W	D+	II+	II	
Valves, other	P	G	W	D+	II+	II	
Mechanical modules	GE/P	G	W	D+	II+	II	
Radioactive equipment and floor drains and other radwaste piping and valves upstream of collector tanks	P	G	R,T,W	D+	II+	II	4
Instrumentation and control boards	GE/P	G	W	N/A	II+	II	
Concentrator	GE	G	W	D+	II+	II	
Plant discharge line	GE/P	G	W	D	II	II	15
18. Reactor water cleanup system (Figure 5.4-22)							
Vessels, filter/demineralizer	GE	G	W	C	II	II	
Regenerative Heat exchangers	P	G	R	C	II	II	
Non-Regenerative Heat exchangers	P	G	R	C	II+	1M	
Piping, within outermost isolation valves	P	1	C	A	I	I	2
Piping, beyond outermost containment isolation valves	P	G	R,W	C,D,D+	II,II+	II	2,4,38
Pumps	GE	G	R,W	C,D,D+	II,II+	II	2,4
Valves, isolation valves reactor coolant pressure boundary	P/GE	1	C,R	A	I	I	2
Valves, beyond outermost containment isolation valves	GE/P	G	R,W	C,D,D+	II,II+	II	2,4
Mechanical modules	GE	G	R,W	C,D,D+	II,II+	II	4
19. Fuel pool cooling and cleanup system (Figures 9.1-6.1 and 9.1-6.2)							
a. Cooling							
Vessels	P	3	R	C	II	I	
Heat exchangers	P	3	R	C	II	I	16
Piping	P	3	R	C	II	I	17
Pumps	P	3	R	C	II	I	18
Makeup system (normal)	P	G	R	C	II	II	4,19

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
RHR connection	P	3	R	C	I	I	19
Makeup system (emergency)	P	3	R	C	I	I	19
b. Cleanup							
Vessels, filter/demineralizer	P	G	W	C	II	II	
Piping	P	G	R,W	C,D+	II,II+	II	4
Pumps	P	G	W	C	II	II	
Piping, suppression pool to outer isolation valves	P	2	R	B	I	I	
20. Control room panels							
Electrical modules (with safety function)	GE	2	W	N/A	I	I	
Cable (with safety function)	GE/P	2	W	N/A	I	I	
21. Local panels and racks							
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	R	N/A	I	I	
22. Offgas system (Figure 11.3-2)							4
Tanks	GE	G	T,W	D+	II+	II	
Heat exchangers	GE	G	T,W	D+	II+	II	
Piping	P	G	T,W,O	D+	II+	II	
Pumps	GE	G	T,W	D+	II+	II	
Valves	P	G	T,W	D+	II+	II	
Mechanical modules (with safety function)	GE	G	T,W	C	II	II	
Pressure vessels	GE	G	T,W	D+	II+	II	
23. Standby service water system (Figure 9.2-12)							
Piping	P	3	P,R,DG,O,W	C	I	I	
Pumps	GE	3	P	C	I	I	
Pump motors	GE	3	P	N/A	I	I	
Valves	P	3	P,R,DG,O,W	C	I	I	
Electrical modules (with safety function)	P	3	P,R,DG,O,W	N/A	I	I	
Cable (with safety function)	P	3	P,R,DG,O,W	N/A	N/A	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
24. Turbine plant service water (Figure 9.2-1)							
Piping and valves	P	G	A,T,R,O,P, W,S	D	II,II+	II	4
Pumps	P	G	P	D	N/A	II	
25. Reactor building closed cooling water system (Figure 9.2-2)							4
Heat exchangers	P	G	R	D	II	II	
Pumps	P	G	R	D	II	II	
Tanks	P	G	R	D	II	II	
Piping and valves inside containment	P	G	C	C	II	II	
Containment isolation valves and associated piping	P	2	C,R	B	I	I	
Piping and valves in reactor building	P	G	R	D	II	II	
Piping and valves, other	P	G	W	D	II	II	
26. Primary containment cooling system (Figure 9.4-8)							
Piping and valves up to outermost isolation valves, containment purge and exhaust	P	2	C,R	B	I	I	
27. Standby gas treatment system (Figure 3.2-2)							
Filter units	P	2	R	B	I	I	
Fans	P	2	R	B	I	I	
Piping and valves	P	2	R	B	I	I	20
28. Primary containment atmospheric control system (DEACTIVATED) (Figure 3.2-3)							
Piping and valves	P	2	C,R	B	I	I	
Equipment	P	2	R	B	I	I	
29. Other heating, ventilating, and air conditioning (Figures 9.4-1, 9.4-2, and 9.4-7)							
Reactor building (nonessential)	P	G	R	N/A	II	II	4
Reactor building (essential)	P	3	R	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Turbine building	P	G	G	N/A	II	II	21
Radwaste building	P	G	W	N/A	II	II	21
Control room, critical switchgear area, cable spreading area (nonessential)	P	G	W	N/A	II	II	4
Control room, critical switchgear area, cable spreading area (essential)	P	3	W	N/A	I	I	
Diesel generator building	P	3	DG	N/A	I	I	22
Standby service water pump house	P	3	P	N/A	I	I	22
30. Condensate storage and transfer (Figure 9.2-11)							
Condensate storage tank	P	G	O	C	II	II	23
Piping and valves	P	G	O,T,R,W	D	II	II	4,24
Pumps	P	G	O	D	II	II	
31. Instrument and sample lines							2
32. Fuel storage facilities							
Fuel pool/dryer separator liner	P	3	R	N/A	II	I	
Storage racks and supports	GE	3	R	N/A	I	I	
33. Building cranes							
Reactor building	P	3	R	N/A	I	I	
Turbine building	P	G	T	N/A	II	II	
Radwaste building	P	G	W	N/A	II	II	
Standby service water pump house	P	G	P	N/A	II+	1M	
Miscellaneous	P	G	P,W,T,S	N/A	II	II	
34. Instrument and service air (Figures 9.3-1.1, 9.3-1.2, 9.3-1.3, 9.3-1.4 and 9.3-1.5)							
Piping and valves	P	G	R,W,T,O	D	II	II	4
Compressors	P	G	T	D	II	II	
Vessels	P	G	T	D	II	II	
Compressor cooling system	P	G	T	D	II	II	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
35. Containment instrument air system (Figure 9.3-2)							
Piping and valves inside containment to and including outboard isolation valve	P	2	C,R	B	I	I	
Piping and valves to main steam relief valves	P	2	R	B	I	I	
Other piping and valves	P	G	R	D	II	II	4
Receiver	P	G	R	D	II	II	4
Piping and valves outside containment isolation valves to and including solenoid pilot valves controlling supply of nitrogen from the nitrogen bottles	P	3	R	C	I	I	
36. High-pressure core spray standby power systems (Division 3)							
Day tanks	P	3	DG	C	I	I	
Piping and valves, fuel oil system	P	3	DG	C	I	I	
Pumps, fuel oil system	P	3	DG	C	I	I	
Diesel generators	GE	2	DG	N/A	I	I	25
Electrical modules (with safety functions)	GE	2	DG	N/A	I	I	
Cable (with safety functions)	P	3	DG	N/A	I	I	
Diesel fuel storage tanks	P	3	DG	C	I	I	
Diesel generators service water supply	P	3	P	C	I	I	
Diesel starting air	P	3	DG	D	I,II	I	26,27
Diesel intake exhaust piping	P	3	DG	D	I	I	26
Diesel jacket water cooling	GE	3	DG	D	I	I	26
37. Standby ac power systems (Divisions 1 and 2)							
Storage and day tanks	P	3	DG	C	I	I	
Piping and valves, diesel oil	P	3	DG	C	I	I	
Pumps, diesel oil	P	3	DG	C	I	I	
Diesel generators	P	2	DG	N/A	I	I	25
Electrical modules (with safety function)	P	3	DG	N/A	I	I	
Diesel cooling water supply	P	3	DG	C	I	I	
Cable (with safety function)	P	3	DG	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Diesel intake/exhaust air piping	P	3	DG	D	I	I	26
Diesel starting air	P	3	DG	D	I,II	I	26,27
Diesel jacket water cooling	P	3	DG	D	I	I	26
38. Auxiliary ac power system							
Essential components	P	2	W,R,DG	N/A	I	I	
Nonessential components	P	G	W,R,T,O	N/A	II	II	4
39. Auxiliary 125/250-V dc power system							
Batteries	P	2	W	N/A	I	I	
Battery Chargers	P	3	W	N/A	I	I	
Cables	P	2	W,R	N/A	I	I	
Modules	P	2	W,R	N/A	I	I	
40. 24-V dc power system							
Batteries	P	2	W	N/A	I	I	
Battery chargers	P	2	W	N/A	I	I	
Cables	P	2	W,R	N/A	I	I	
Modules	P	2	W,R	N/A	I	I	
41. 120-V critical power supply system							
Equipment	P	2	W,R	N/A	I	I	
42. Power conversion system (Figures 10.3-1 and 3.2-4)							
Main steam piping from outermost isolation valves up to turbine stop valves	P	2	R,T	B	I	I	28
Main steam branch piping to first valve capable of timely closure	P	2	T	B	I	I	28
Main steam piping downstream of MS-V-146 including turbine bypass piping, and steam piping down stream of first valve capable of timely closure	P	G	T	D	II	II	

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Turbine stop valves, turbine control valves, and turbine bypass valves	P	G	T	D	II	II	29,30
Main steam leads from turbine control valve to turbine casing	P	G	T	D	II	II	30
Feedwater and condensate system beyond outermost isolation valve	P	G	R,T	D	II	II	4
Turbine generator	P	G	T	D	II	II	
Condenser	P	G	T	D	II	II	
Air ejection equipment	P	G	T	D	II	II	
Feedwater treatment system	P	G	T	D	II	II	
Turbine bypass system beyond turbine bypass valve	P	G	T	D	II	II	
Turbine gland sealing system components	P	G	T	D	II	II	
Piping, valves, other	P	G	T	D	II	II	
Equipment, other	P	G	T	D	II	II	
43. Circulating water and cooling tower makeup water system(s) (Figure 10.4-4)							
Piping and valves	P	G	P	D	II	II	31
Pumps	P	G	P	D	II	II	
Cooling tower fans	P	G	P	D	II	II	
44. Main steam isolation valves leakage control system (DEACTIVATED) (Figure 3.2-5)							
Piping and valves within primary containment and out through the outermost isolation valves	*P*	*1*	*R,C*	*A*	*I*	*I*	
Piping and valves beyond the outermost isolation valves	*P*	*2*	*R*	*B*	*I*	*I*	
Blowers	*P*	*2*	*R*	*N/A*	*I*	*I*	
45. Containment vessel	P	2	R	B	I	I	

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
46. Buildings							
Reactor building	N/A	2	R	N/A	I	I	
Turbine building	N/A	G	T	N/A	II	II	32
Radwaste control building	N/A	3/G	W	N/A	I,II	I/II	33
Diesel generator building	N/A	3	DG	N/A	I	I	
Spray ponds and standby service water pump house	N/A	3	P	N/A	I	I	
Service building	N/A	G	S	N/A	II	II	
Cooling towers	N/A	G	O	N/A	II	II	
Makeup water pump house	N/A	G	O	N/A	I	II	31
Circulation water pump house	N/A	G	O	N/A	II	II	
Air intake structures No. 1 and No. 2	N/A	3	O	N/A	I	I	
47 Containment/drywell atmosphere monitoring system	P	3	R	A	I	I	
48. Drywell insulation							
Insulation on piping which is within the drywell	P	G	R	N/A	I	I	
49. Instrumentation and control equipment							
Safety-related instrumentation and control systems	P	1	C,R,T,DG	A	I	I	34
50. Postaccident sampling system (Figure 3.2-6)							
Piping within outermost reactor coolant boundary isolation valves	P	1	C,R	A	I	I	
Piping within outermost containment isolation valves	P	2	C,R	B	I	I	
Piping within the outermost RHR system isolation valves	P	2	R	B	I	I	
Piping beyond the outermost reactor coolant boundary, containment, or RHR system isolation valves	P	G	R,W	D	II	II	
Sample station	GE	G	R,W	D	G	II	
All other	P	G	R,W	D	II	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
51. <u>Hydrogen Water Chemistry System</u>							
Hydrogen and air injection piping and valves near and inside TGB (Figure 10.4-9.1)	GE/P	G	O,T	D	II	II	
52. <u>Hydrogen Storage and Supply Facility (HSSF)</u> (Figures 10.4-9.2, 10.4-9.3)							
Tubing and valves	P	G	H	D	II	II	
Buried piping from HSSF to TGB	P	G	H,O	D	II	II	
Liquid Hydrogen Storage Vessel	P	G	H	D	II+	1M	35
Vessels, other	P	G	H	D	II	II	
53. <u>Hardened Containment Vent (HCV)</u>							
Piping and valves forming part of containment boundary from the containment penetration up to and including Rupture Discs HCV-RD-54 and HCV-RD-60	P	2	R	B	I	I	
Piping and valves forming part of secondary containment from but not including Rupture Discs HCV-RD-54 and HCV-RD-60 to the secondary containment penetrations including the PCIV pneumatic lines between the valves and secondary containment	P	2	R	D	I	I	
Piping and valves from the secondary containment penetration to the exhaust point	P	G	R	D	II+	I	

Table 3.2-1

Equipment Classification (Continued)

[a] A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electric modules include sensors, power supplies, and signal processors; and mechanical modules include turbines, strainers, and orifices.

[b] GE - General Electric; P - Plant Owner (Energy Northwest)

[c] 1, 2, 3, G - Safety classes defined in Section 3.2.3.

[d]

A - Auxiliary building	M - Any other location
C - Part of or within primary containment	O - Outdoors onsite
L - Offsite locale	P - Pump house
R - Reactor building	W - Radwaste and control building
S - Service building	DG - Diesel generator building
T - Turbine building	H - Hydrogen Storage and Supply Facility

[e] A, B, C, D - NRC quality groups defined in Regulatory Guide 1.26, Revision 3. The equipment is constructed in accordance with the codes listed in Table 3.2-2, as minimum requirements.

N/A - Quality Group classification not applicable to this equipment.

[f] Quality Classes defined in Section 3.2.4.

Table 3.2-1

Equipment Classification (Continued)

[g] I - constructed in accordance with the seismic requirements of Seismic Category I structures and equipment as described in Section 3.7.
II - constructed in accordance with the requirements of Seismic Category II structures and equipment as described in Section 3.7. The approaches outlined in the Uniform Building Code will be followed where applicable.

N/A - seismic requirements as described for Seismic Category I and II structures and equipment are not applicable to this structure or equipment.

1M - non-safety-related components required to be seismically supported/mounted. This was alternatively called Seismic II⊕ which is Seismic II piping supported on hangers designed to Seismic I loads.

[h] The notes clarify system/component boundaries, design requirements, and/or alternate quality classification applications.

Reactor pressure boundary components (RPBC) and their original applicable code cases are listed in the attachments to Table 3.2-1. Current practices regarding code cases that are adopted for use at CGS (and are approved by Regulatory Guides 1.147, 1.84, 1.85, or other approval authority) requires that they are specified in the component's design specification as required by ASME Section III, NA-3250.

Table 3.2-1

Equipment Classification (Continued)

NOTES

1. This piping has been upgraded from Safety Class 3 to Safety Class 2 and from Quality Group Classification C to Quality Group Classification B.

2. a. Lines 0.75-in. and smaller which are part of the RCPB are Quality Group B or higher and Seismic Category I.

   b. All instrument lines which are connected to the RCPB and are used to actuate and monitor safety systems are Quality Group B from the outer isolation valve or the process shutoff valve (root valve) to the bulkhead of the instrument rack, if rack mounted, or the sensing instrumentation, if locally mounted.

   c. All instrument lines which are connected to the RCPB and are not used to actuate and monitor safety systems are Quality Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrument

   d. All other instrument lines:

      1. Through the root valve are of the same classification or higher as the system to which they are attached;

      2. Beyond the root valve to the instrument rack bulkhead, if rack mounted, or to the sensing instrumentation, if locally mounted, if used to actuate a safety system, are of the same classification as the system to which they are attached;

      3. Beyond the root valve, if not used to actuate a safety system are Quality Group D.

      4. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system are Quality Group D.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

3. The control rod drive insert and withdraw lines from the drive flange up to and including the first valve on the hydraulic control unit (HCU) is in Safety Class 2.

4. Supports for Quality Class II (nonessential) piping systems, HVAC, cable trays, and system components in the reactor building, primary containment, the control building, diesel generator building, the standby service water (SW) pump houses, and the radwaste building corridor are designed and constructed to withstand an SSE per position C.2 of Regulatory Guide 1.29. These supports are constructed to Quality Class II requirements, as a minimum.

5. The HCU is a GE factory assembled engineered module of valves, tubing, piping, and stored water which controls a single control rod drive by the application of precisely timed sequences of pressure and flows to accomplish slow insertion or withdrawal of the control rods for power control and rapid insert ion for reactor scram.

   Although the HCU is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Quality Groups A, B, C, D pressure integrity quality levels clearly apply to all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

   The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, (a) all welds are LP inspected, (b) all socket welds are inspected for gap between pipe and socket bottom, (c) all welding is performed by qualified welders, and (d) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Groups A, B, or C. This is supplemented by the quality control techniques described above.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

6. The standby liquid control (SLC) system is not a safety-related backup to a scram function (RPS/CRD). However, these components of the SLC system are designed to engineering standard greater than those normally applied to Safety Class 2 systems.

7. Only equipment associated with a safety action (e.g., isolation) need conform to a safety function.

8. These lines meet the requirements of Quality Group B except that hydrostatic testing of the containment spray piping is not required.

9. The RCIC turbine does not fall within the applicable design codes. To ensure that the turbine is fabricated to the standards commensurate with its safety and performance requirements, GE has established specific design requirements for this component which are as follows:

   a. All welding is qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code;

   b. All pressure containing castings and fabrications are hydrotested to 1.5 times design pressure;

   c. All high-pressure castings are radiographed according to:

      1. ASTM E-94
      2. E-142 maximum feasible volume
      3. E-71, 186 or 280 severity level 3;

   d. All-cast surfaces are magnetic particulate or liquid penetrant tested according to ASME, Section III, Paragraph NB-2575 or NB-2576;

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

e. Wheel and shaft forgings are ultrasonically tested according to ASTM-A-388;

f. Butt-welds are radiographed according to ASME Section III, IX-3300, and magnetic particle or liquid penetrant tested according to ASME Section III, IX-3500, or IX-3600;

g. Notification is made on major repairs and records maintained thereof;

h. Record system and traceability according to ASME Section III, NA-4900;

i. Control and identification according to ASME Section III, NA-4400;

j. Procedures conform to ASME Section III, NA-4400;

k. Inspection personnel are qualified according to ASME Section III, Appendix IX, paragraph IX-400.

10. The RCIC turbine exhaust line from the isolation valve to the suppression pool meets all the requirements of Quality Group B except that hydrostatic testing of this portion of piping is not required.

11. Although the refueling bellows are designed to withstand the SSE without rupture, they may be plastically deformed.

12. DELETED.

13. Equipment, piping, and valves that are part of the radwaste system but not used for processing radioactive fluids are designed to Quality Group D standards.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

14. Equipment, piping, and valves that are part of the radwaste solids handling system are designed to Quality Group D standards.

15. Up to and including the last stop valve, this line meets requirements of Quality Group C.

16. The fuel pool cooling heat exchangers have been upgraded to 270 psig (shell side) and Seismic Category I through several modifications. These heat exchangers are Quality Class II (with the modifications installed as Quality Class I) and ASME Section III Class 3. See Note 13 in Figure 9.1-6.1.

17. Piping and valves of the cooling portion of the fuel pool cooling system have been upgraded to Seismic Category I and are ASME Section III Class 3, Code Group C (with exception of valves listed in Note 16 in Figure 9.1-6.1). See Note 13 in Figure 9.1-6.1 for Quality Classification.

18. The fuel pool cooling circulation pumps have been upgraded to Seismic Category I. Their respective motors have been upgraded to Quality Class I Seismic Category I. The fuel pool cooling pumps are ASME Section III Class 3. See Note 13 in Figure 9.1-6.1 for Quality Classification.

19. The fuel pool cooling system (see Section 9.1.3) normally receives makeup from the Seismic Category II condensate storage tank. Should this normal supply be unavailable, makeup is available from the Seismic Category I SW system. Likewise, the normal source of cooling water to the FPC heat exchangers is from the Seismic Category II RCC system. When this normal supply is unavailable, the Seismic Category I SW system provides the cooling water and makeup. In addition, by means of removable spool-pieces, the RHR system is also available as the supplementary source of cooling during cold shutdown under core offload conditions. The above complies with Regulatory Guides 1.26, Revision 3, and 1.29, Revision 3.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

The cleanup portion is automatically isolated from the cooling portion of the system by Seismic Category I valves on low fuel pool level (see Section 9.1.3).

20. Piping and valves downstream of valves SGT-V-5A1, SGT-V-5A2, SGT-V-5B1, and SGT-V-5B2 meet the requirements of Quality Group B except that a pneumatic test is not performed.

21. Lavatory exhaust systems are designed to Quality Assurance Class G.

22. Part of the HVAC components are non-safety-related. Non-safety-related equipment and components required to be seismically supported/mounted according to Regulatory Guide 1.29 are designated Seismic Category 1M and Quality Assurance Class II+. Non-safety-related equipment and components not required to be seismically supported/mounted according to Regulatory Guide 1.29 are designated Seismic Category II and Quality Assurance Class II.

23. The condensate storage tanks are designed, fabricated, and tested to meet ASME Code, Section III, Subsection ND-3800. In addition, the specification for this tank requires 100% surface examination of the side wall to bottom joint, and 100% volumetric examination of the side-wall weld joints.

24. The high-pressure core spray (HPCS) suction piping from the condensate storage tank provides the initial source of makeup water to the HPCS system for safety injection. Consequently, this piping has been upgraded by full volumetric examination of every weld.

25. The auxiliary piping systems on the engines are built to the guidelines of ANSI B31.1.

26. These piping systems are liquid penetrant or magnetic particle examined to the acceptance standards of ASME Section III, Class 3, if they are over 4 in. IPS as required by the 1973 Winter Addenda except for the diesel cooling water piping that is on engine skid.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

27. Piping upstream from the check valves at the inlet of the air receivers is Quality Class II, ANSI B31.1, Seismic Category I. All other piping is Quality Class 1.

28. The piping is supported to Seismic Category I requirements, but is not housed in a Seismic Category I structure. The power conversion system structures are constructed in accordance with applicable codes for steam power plants. The turbine building, interacting with main steam lines and branch lines, is designed as a modified Seismic Category II structure as described in Section 3.8.4.1.3.

29. All cast pressure retaining parts of a size and configuration for which volumetric examination methods are required are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternative to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examination in ANSI B31.1.0 Code, Paragraph 136.4.3.

30. The following qualification is met with respect to the certification requirements:

    a. The manufacturer of the turbine stop valves, turbine governor valves, turbine bypass valves, and mainsteam leads from turbine control valve to turbine casting utilized quality control procedures, and

    b. A certification has been obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

31. The makeup water pump house is designed to withstand the design-basis tornado. The design also considers the possible effects of tornado-generated missiles. The tower makeup water piping, valves, and cabling located underground are provided with adequate earth cover to be resistant to tornado-generated missiles or are protected by tornado-resistant structures. The circulating water system also supports the post-tornado operation of the standby service water system. In the event of the loss of the spray headers in the SW spray ponds, tornado-protected underground lines can provide a flow path to provide makeup water from TMU, and to return water to the CW basin. See FSAR Section 9.2.5.3 for additional discussion.

32. Portions of the turbine building that support or interact with main steam piping are designed to Seismic Category I.

33. Those portions of the radwaste and control building that house systems or components necessary for safe shutdown of the reactor are designed to Quality Class I and Seismic Category I requirements. Those portions of the radwaste building housing equipment containing significant quantities of radioactive material are designed to Seismic Category I requirements.

34. Safety-related instrument and control systems are identified in Table 7.1-1.

35. The HSSF liquid hydrogen storage tank, foundations, anchorage (i.e., anchor bolts, slide plates, and the baseplate welding) and the underlying soil are not safety-related, and are designated as Quality Class II+. However, these are designed for Seismic Category I loads and ground motion as defined by Regulatory Guide 1.60. In addition, they were designed to remain in place for both design basis tornado characteristics and maximum probable flood.

36. Piping located between secondary containment isolation valves EDR-V-394, EDR-V-395, FDR-V-219, FDR-V-220, FDR-V-221 and FDR-V-222 is classified as Seismic Category I, Quality Class II+ and Quality Group D.

37. Seismic Category I applies to the motor rotor, bearings and bearing load path parts. The motor is mounted Seismic Category IM.

38. Large bore piping is analyzed to Seismic Category I loading.

Table 3.2-1 Attachment 1

Nuclear Boiler System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-X70d			PI(1)-4S-X114		
PI(1)-4S-X71a			PI(1)-ST-X114		
PI(1)-ST-X71a			PI(1)-4S-X115		
PI(1)-4S-X71b			PI(1)-ST-X115		
PI(1)-ST-X71b			PI(1)-ST-(IR-64)-4		
PI(1)-4S-X72a			PI(1)-ST-(IR-64)-5		
PI(1)-ST-X72a			PI(1)-ST-MS-PT-2		
PI(1)-4S-X75c			RFW(1)-4A	1 thru 11, K,L,M,S,T	
PI(1)-ST-X75c			RFW(1)-4B	1 thru 11, K,L,M,S,T	
PI(1)-4S-X75d					
PI(1)-ST-X75d					
PI(1)-4S-X106					
PI(1)-ST-X106					
PI(1)-4S-X107					
PI(1)-ST-X107					
PI(1)-4S-X108					
PI(1)-ST-X108					
PI(1)-4S-X109					
PI(1)-ST-X109					
PI(1)-4S-X110					
PI(1)-ST-X110					
PI(1)-4S-X111					
PI(1)-ST-X111					
PI(1)-4S-X112					
PI(1)-ST-X112					
PI(1)-4S-X113					
PI(1)-ST-X113					

Table 3.2-1 Attachment 1

Nuclear Boiler System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
MS(9)-4	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X39a		
MS(18)-2-1	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X39a		
MS(18)-2-2	1 thru 11 K,L,M,S,T		PI(1)-4S-X39b		
MS(18)-2-3	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X39b		
MS(18)-2-4	1, 4 thru 11, K,L,M,S,T		PI(1)-4S-X42a		
MS(18)-2-5	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X42a		
MS(18)-2-6	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X42b		
MS(18)-2-7	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X42b		
MS(18)-2-8	1, 4 thru 11,K,L,M,S,T		PI(1)-4S-X42e		
MS(18)-2-9	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X42e		
MS(18)-2-10	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X61c		
MS(18)-2-11	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X61c		
MS(18)-2-12	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X62b		
MS(18)-2-13	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X62b		
MS(18)-2-14	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X69a		
MS(18)-2-15	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X69a		
MS(18)-2-16	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X69b		
MS(18)-2-17	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X69b		
MS(18)-2-18	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X69f		
B22-G001A	1 thru 3		PI(1)-ST-X69f		
B22-G001B	1 thru 3		PI(1)-4S-X70a		
B22-G001C	1 thru 3		PI(1)-ST-X70a		
B22-G001D	1 thru 3		PI(1)-4S-X70b		
PI(1)-4S-X38a			PI(1)-ST-X70b		
PI(1)-ST-X38a			PI(1)-4S-X70c		
PI(1)-4S-X38b			PI(1)-ST-X70c		
PI(1)-ST-X38b			PI(1)-4S-70d		

Table 3.2-1 Attachment 2

Reactor Recirculation System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X40c			PI(1)-ST-X44Ah		
PI(1)-ST-X40c			PI(1)-4S-X44Aj		
PI(1)-4S-X40d			PI(1)-ST-X44Aj		
PI(1)-ST-X40d			PI(1)-4S-X44Ak		
PI(1)-4S-X40e			PI(1)-ST-X44Ak		
PI(1)-4S-X40f			PI(1)-4S-X44A1		
PI(1)-4S-X41c			PI(1)-ST-X44A1		
PI(1)-ST-X41c			PI(1)-4S-X44Am		
PI(1)-4S-X41d			PI(1)-ST-X44Am		
PI(1)-ST-X41d			PI(1)-4S-X44Ba		
PI(1)-4S-X41e			PI(1)-ST-X44Ba		
PI(1)-4S-X41f			PI(1)-4S-X44Bb		
PI(1)-4S-X44Aa			PI(1)-ST-X44Bb		
PI(1)-ST-X44Aa			PI(1)-4S-X44Bc		
PI(1)-4S-X44Ab			PI(1)-ST-X44Bc		
PI(1)-ST-X44Ab			PI(1)-4S-X44Bd		
PI(1)-4S-X44Ac			PI(1)-ST-X44Bd		
PI(1)-ST-X44Ac			PI(1)-4S-X44Be		
PI(1)-4S-X44Ad			PI(1)-ST-X44Be		
PI(1)-ST-X44Ad			PI(1)-4S-X44Bf		
PI(1)-4S-X44Ae			PI(1)-ST-X44Bf		
PI(1)-ST-X44Ae			PI(1)-4S-X44Bg		
PI(1)-4S-X44Af			PI(1)-ST-X44Bg		
PI(1)-ST-X44Af			PI(1)-4S-X44Bh		
PI(1)-ST-X44Ag			PI(1)-ST-X44Bh		
PI(1)-ST-X44Ag			PI(1)-4S-X44Bj		
PI(I)-4S-X44Ah			PI(1)-ST-X44Bj		

Table 3.2-1 Attachment 3

Control Rod Drive Hydraulic System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X44Bk			PI(1)-ST-X75b		
PI(1)-ST-X44Bk			PI(1)-4S-X75e		
PI(1)-4S-X44Bl			PI(1)-ST-X75e		
PI(1)-ST-X44B1			PI(1)-4S-X75f		
PI(1)-4S-X44Bm			PI(1)-ST-X75f		
PI(1)-ST-X44Bm			PI(1)-4S-X78f		
PI(1)-4S-X61a			PI(1)-ST-X78f		
PI(1)-ST-X61a			PI(1)-1-RRC-SPV-85A		
PI(1)-4S-X61b			PI(1)-1-RRC-SPV-85B		
PI(1)-ST-X61b			RRC(5)-4S-A	1, 3 thru 11 K,L,M,S,T	
PI(1)-4S-X62c			RRC(5)-4S-B	1, 3 thru 11 K,L,M,	
PI(1)-ST-X62c			RRC(51)-1	1, 3 thru 11 K,L,M	
PI(1)-4S-X62d			RRC(51)-4	1, 3 thru 11 K,L,M,S,T	
PI(1)-ST-X62d			B35-G001A	1, 3	
PI(1)-4S-X69e			B35-G001B	1, 3	
PI(1)-ST-X69e					
PI(1)-4S-X70e					
PI(1)-4S-X70f					
PI(1)-4S-X74a					
PI(1)-ST-X74a					
PI(1)-4S-X74e					
PI(1)-ST-X74e					
PI(1)-4S-X74f					
PI(1)-ST-X74f					
PI(1)-4S-X75a					
PI(1)-ST-X75a					
PI(1)-4S-X75b					

Table 3.2-1 Attachment 3

Control Rod Drive Hydraulic System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
CRD-I/W	G				
CRD-SCRAM/INST	5, 13, G				
Hy(1)-65-A	1, 3 thru 11 K,L,M,S				
Hy(1)-65-B	4 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 4

Standby Liquid Control System

Component	Code Case	Comments	Component	Code Case	Comments
SLC(1)-1S	1, 3 thru 11 K,L,M				
SLC(2)-3S	1, 3 thru 11 K,L,M,S,T				
SLC(2)-4S	1, 3 thru 11 K,L,M,S,T				
PI(1)-ST-SLC-FT-1					
PI(1)-ST-SLC-PT-4					

Table 3.2-1 Attachment 5

Residual Heat Removal System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X37e			PI(1)-ST-(H22-PO21)-A10		
PI(1)-ST-X37e			PI(1)-ST-(H22-PO21)-B1		
PI(1)-4S-X37f			PI(1)-ST-(H22-PO21)-B3		
PI(1)-ST-X37f			PI(1)-ST-(H22-PO21)-B4		
PI(1)-4S-X39d			PI(1)-ST-(H22-PO21)-B5		
PI(1)-ST-X39d			PI(1)-ST-(H22-PO21)-B6		
PI(1)-4S-X39e			PI(1)-ST-(IR-69)-2		
PI(1)-ST-X39e			PI(1)-ST-(IR-69)-3		
PI(1)-4S-X42d			PI(1)-ST-(IR-69)-7		
PI(1)-ST-X42d			PI(1)-ST-(IR-71)-1		
PI(1)-4S-X54Bf			PI(1)-ST-(IR-71)-2		
PI(1)-ST-X54Bf			PI(1)-ST-RHR-DPIS-9A		
PI(1)-4S-X61f			PI(1)-ST-RHR-DPIS-9B		
PI(1)-ST-X61f			PI(1)-ST-RHR-DPIS-9C		
PI(1)-1-X62f			PI(1)-ST-RHR-FT-1		
PI(1)-1-X69c			PI(1)-ST-RHR-FT-13		
PI(1)-4S-X74b			PI(1)-ST-RHR-LT-8A		
PI(1)-ST-X74b			PI(1)-ST-RHR-LT-8B		
PI(1)-ST-(H22-P018)-A5			PI(1)-ST-RHR-PI-2A		
PI(1)-ST-(H22-P018)-A6			PI(1)-ST-RHR-PI-2B		
PI(1)-ST-(H22-P018)-A7			PI(1)-ST-RHR-PI-2C		
PI(1)-ST-(H22-P018)-A9			PI(1)-ST-RHR-PS-18		
PI(1)-ST-(H22-P018)-A10			RHR(1)-2A	1, 3 thru 12 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A5			RHR(1)-2B	1 thru 12 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A6			RHR(1)-2C	1, 3 thru 11 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A7			RHR(1)-4A	1, 3 thru 11 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A9			RHR(1)-4B	1, 3 thru 11 K,L,M,S,T	

Table 3.2-1 Attachment 5

Residual Heat Removal System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
RHR(1)-4C	1, 3 thru 11 K,L,M,S,T				
RHR(3)-1C	1, 3 thru 12 K,L,M,S,T				
RHR(3)-2A	1, 3 thru 11 K,L,M,S,T				
RHR(3)-2B	1, 3 thru 11 K,L,M,S,T				
RHR(4)-1A	1, 3 thru 12 K,L,M,S,T				
RHR(4)-1B	1, 3 thru 12 K,L,M,S,T				
RHR(4)-1C	1, 3 thru 11 K,L,M,S,T				
RHR(9)-1	1, 3 thru 11 K,L,M,S,T				
RHR(1)-4B1	1, 4 thru 11 K,L,M,S,T				
RHR(1)-4A1	1, 3 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 6

Low-Pressure Core Spray

Component	Code Case	Comments	Component	Code Case	Comments
LPCS(1)-2	1 thru 11 K,L,M,P,S,T				
LPCS(1)-4	1, 4 thru 11 K,L,M,S,T				
LPCS(2)-1	1, 3 thru 11 K,L,M,S,T				
LPCS(3)-1	1, 3 thru 11 K,L,M,S,T				
PI(1)-4S-X78b					
PI(1)-ST-X78b					
PI(1)-1-X78d					
PI(1)-ST-X78d					
PI(1)-ST-(H22-POO1)-A2					
PI(1)-ST-(H22-POO1)-A3					
PI(1)-ST-(H22-POO1)-A4					
PI(1)-ST-(H22-POO1)-A5					
PI(1)-ST-(H22-POO1)-A6					
PI(1)-ST-(H22-POO1)-A7					
PI(1)-ST-LPCS-PI-1					

Table 3.2-1 Attachment 7

High-Pressure Core Spray

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X73a					
PI(1)-ST-X73a					
PI(1)-1-X78e					
PI(1)-ST-X78e					
PI(1)-ST-(H22-PO24)-A1					
PI(1)-ST-(H22-PO24)-A2					
PI(1)-ST-(H22-PO24)-A5					
PI(1)-ST-(H22-PO24)-A6					
PI(1)-ST-(H22-PO24)-A8					
PI(1)-ST-HPCS-PS-3					
HPCS(1)-4CL1	1, 3 thru 11 K,L,M,S,T				
HPCS(1)-4CL2	1, 3 thru 11 K,L,M,S,T				
HPCS(2)-1	1, 3 thru 11 K,L,M,S,T				
HPCS(3)-1	1, 4 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 8

Reactor Core Isolation Cooling System

Component	Code Case	Comments	Component	Code Case	Comments
RCIC(1)-4CL1	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A5		
RCIC(1)-4CL2	1, 4 thru 12 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A6		
RCIC(2)-1	1 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A7		
RCIC(12)-4CL1	1, 2, 4 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A8		
RCIC(12)-4CL2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B1		
RCIC(13)-4CL2	1 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B2		
RCIC(16)-1	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B3		
RCIC(19)-1	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B6		
RCIC(50)-1	1 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B7		
PI(1)-4S-X38c			PI(1)-ST-(H22-PO18)-A13		
PI(1)-ST-X38c			PI(1)-ST-(H22-PO21)-A13		
PI(1)-4S-X38d			PI(1)-ST-(H22-PO29)-A5		
PI(1)-ST-X38d			PI(1)-ST-(H22-PO29)-A6		
PI(1)-4S-X38e			PI(1)-ST-(IR-62)-2		
PI(1)-ST-X38e			PI(1)-ST-(IR-62)-4		
PI(1)-4S-X38f			PI(1)-ST-(IR-62)-5		
PI(1)-ST-X38f			PI(1)-ST-(IR-63)-15		
PI(1)-1-X54Aa			PI(1)-ST-(IR-63)-16		
PI(1)-ST-X54Aa			PI(1)-ST-(IR-67)-4		
PI(1)-4S-X71c			PI(1)-ST-RCIC-PCV-15		
PI(1)-ST-X71c			PI(1)-ST-RCIC-PS-1		
PI(1)-4S-X71d			PI(1)-ST-RCIC-PS-9A		
PI(1)-ST-X71d			PI(1)-ST-RCIC-PS-9B		
PI(1)-4S-X71e			PI(1)-ST-RCIC-PS-34		
PI(1)-ST-X71e					
PI(1)-4S-X71f					
PI(1)-ST-X71f					

Table 3.2-1 Attachment 9

Radwaste System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-61)-6					
PI(1)-ST-(IR-65)-3					
PI(1)-ST-(IR-61)-4					
PI(1)-ST-(IR-65)-4					
PI(1)-ST-(IR-62)-17					
PI(1)-ST-(IR-62)-18					
EDR(48)-1	1, 3 thru 11 K,L,M,S,T				
FDR(48)-1	1, 4 thru 11 K,L,M,S,T				
MWR(62)-15	1, 3				

Table 3.2-1 Attachment 10

Reactor Water Cleanup System

Component	Code Case	Comments	Component	Code Case	Comments
RWCU(1)-3A	1 thru 11 K,L,M,S,T				
RWCU(1)-3B	1, 3 thru 12 K,L,M,S,T				
RCWU(1)-4	1, 3 thru 11 K,L,M,S,T				
RWCU(3)-4	1 thru 11 K,L,M,S,T				
PI(1)-4S-X78c					
PI(1)-4S-X79a					
PI(1)-ST-X79a					
PI(1)-4S-X79b					
PI(1)-ST-X79b					
PI(1)-ST-(H22-P002)-A8					
PI(1)-ST-(H22-P002)-A9					
PI(1)-ST-(H22-P002)-A12					
PI(1)-ST-(H22-P002)-A13					

Table 3.2-1 Attachment 11

Standby Service Water System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-69)-1					
PI(1)-ST-(IR-69)-8					
PI(1)-ST-(IR-69)-10					
PI(1)-ST-(IR-62)-3					
PI(1)-ST-(IR-71)-4					
PI(1)-ST-(IR-71)-5					
PI(1)-ST-FPC-FT-16					
PI(1)-ST-FPC-FT-17					
PI(1)-ST-FPC-LIS-1A					
PI(1)-ST-FPC-LIS-1B					
PI(1)-ST-FPC-SPV-1					
FPC(1)-1	1 thru 11 K,L,M				
FPC(2)-1A	1, 3 thru 11 K,L,M				
FPC(2)-1B	1, 3 thru 11 K,L,M,S,T				
FPC(5)-2	1, 3 thru 11 K,L,M,S,T				
FPC(7)-1	1				
FPC(12)-1	1, 4 thru 11 K,L,M,N,S,T				

Table 3.2-1 Attachment 12

Standby Service Water System

Component	Code Case	Comments	Component	Code Case	Comments
SW(1)-2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-SW-PS-11B		
SW(1)-2-UG	1, 3 thru 11 E,K,L,M		PI(1)-ST-SW-PS-11A		
SW(2)-2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(CMS-SR-14)-C2		
SW(2)-2-UG	1, 4 thru 11 E,K,L,M		PI(1)-ST-(IR-22)-1		
SW(21)-2	1 thru 11 K,L,M,S,T		PI(1)-ST-(IR-22)-2		
SW(21)-2-UG	1, 3 thru 11 E,K,L,M,S,T		PI(1)-ST-(IR-22)-3		
SW(22)-2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(IR-69)-15		
SW(22)-2-UG	1, 3 thru 11 E,K,L,M,S,T		PI(1)-ST-(SW-SR-43)-A		
SW(70)-1-HPCS	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(SW-SR-43)-B		
SW(71)-1-HPCS	1, 3 thru 11 K,L,M		P1(1)-ST-SW-FIS-9		
PI(1)-ST-(CMS-SR-13)-C1			PI(1)-ST-SW-FIS-12		
PI(1)-ST-(CMS-SR-13)-C2			PI(1)-ST-SW-FIS-15		
PI(1)-ST-(H22-PO18)-A11					
PI(1)-ST-(H22-PO18)-A12					
PI(1)-ST-(H22-PO21)-A11					
PI(1)-ST-(H22-PO21)-A12					
PI(1)-ST-(IR-21)-1					
PI(1)-ST-(IR-21)-2					
PI(1)-ST-(IR-21)-3					
PI(1)-ST-(IR-24)-1					
PI(1)-ST-(IR-24)-2					
PI(1)-ST-(IR-71)-9					
PI(1)-ST-(SW-SR-42)-A					
PI(1)-ST-(SW-SR-42)-B					
PI(1)-ST-(CMS-SR-14)-C1					
PI(1)-ST-SW-FT-8A					
PI(1)-ST-SW-FT-8B					

Table 3.2-1 Attachment 13

Reactor Building Closed Cooling Water System

Component	Code Case	Comments	Component	Code Case	Comments
RCC(3)-1	1 thru 11 K,L,M,S,T				
RCC(4)-2	3 thru 11 K,L,M,S,T				
RCC(5)-2	1, 3 thru 11 K,L,M,S,T				
RCC(36)-1	1 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 14

Primary Containment Cooling System

Component	Code Case	Comments	Component	Code Case	Comments
CSP(1)-1B	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X73e		
CEP(1)-1A	1, 2, 4 thru 11 K,L,M,S,T		PI(1)-ST-X73e		
CEP(1)-1B	1 thru 11 K,L,M,S,T		PI(1)-4S-X82b		
PI(1)-4S-X29c			PI(1)-ST-X82b		
PI(1)-4S-X29d			PI(1)-4S-X82c		
PI(1)-ST-X29d			PI(1)-ST-X82c		
PI(1)-4S-X29e			PI(1)-4S-X84a		
PI(1)-ST-X29e			PI(1)-ST-X84a		
PI(1)-4S-X30a			PI(1)-4S-X84b		
PI(1)-ST-X30a			PI(1)-ST-X84b		
PI(1)-4S-X30d			PI(1)-4S-X85c		
PI(1)-ST-X30d			PI(1)-ST-X85c		
PI(1)-4S-X72b			PI(1)-4S-X85d		
PI(1)-ST-X72b			PI(1)-ST-X85d		
PI(1)-4S-X72c			PI(1)-4S-X85e		
PI(1)-ST-X72c			PI(1)-ST-X85e		
PI(1)-4S-X72d			PI(1)-4S-X86a		
PI(1)-ST-X72d			PI(1)-ST-X86a		
PI(1)-4S-X72e			PI(1)-4S-X86b		
PI(1)-ST-X72e			PI(1)-ST-X86b		
PI(1)-4S-X72f			PI(1)-4S-X87a		
PI(1)-ST-X72f			PI(1)-ST-X87a		
PI(1)-4S-X73c			PI(1)-4S-X87b		
PI(1)-ST-X73c			PI(1)-ST-X87b		
PI(1)-4S-X73d			PI(1)-ST-(IR-62)-14		
PI(1)-ST-X73d			PI(1)-ST-(IR-62)-15		
			PI(1)-ST-(IR-62)-16		

Table 3.2-1 Attachment 14

Primary Containment Cooling System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-62)-19					
PI(1)-ST-(IR-63)-1A					
PI(1)-ST-(IR-63)-1B					
PI(1)-ST-(IR-63)-2					
PI(1)-ST-(IR-63)-3					
PI(1)-ST-(IR-63)-4					
PI(1)-ST-(IR-63)-10					
PI(1)-ST-(IR-64)-1A					
PI(1)-ST-(IR-64)-1B					
PI(1)-ST-(IR-64)-2					
PI(1)-ST-(IR-64)-38					
PI(1)-ST-(IR-64)-7					
PI(1)-ST-(IR-64)-9					
PI(1)-ST-(IR-65)-1					
PI(1)-ST-(IR-65)-2					
PI(1)-ST-(IR-65)-7					
PI(1)-ST-(IR-65)-8					
PI(1)-ST-(IR-65)-9					
PI(1)-ST-(IR-66)-3					
PI(1)-ST-(IR-67)-2					
PI(1)-ST-(IR-67)-3					
PI(1)-ST-(IR-68)-1					
PI(1)-ST-(IR-68)-2					
PI(1)-ST-(CMS-SR-13)-DT					
PI(1)-ST-(CMS-SR-14)-DT					

Table 3.2-1 Attachment 15

Standby Gas Treatment System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-SGT-FT-1A1					
PI(1)-ST-SGT-FT-1A2					
PI(1)-ST-SGT-FT-1B1					
PI(1)-ST-SGT-FT-1B2					

Table 3.2-1 Attachment 16

Primary Containment Atmospheric Control System (DEACTIVATED)

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-CAC-FT-1A					
PI(1)-ST-CAC-FT-1B					
PI(1)-ST-CAC-FT-2A					
PI(1)-ST-CAC-FT-2B					
PI(1)-ST-CAC-FT-3A					
PI(1)-ST-CAC-FT-3B					
PI(1)-ST-CAC-FT-4A					
PI(1)-ST-CAC-FT-4B					
CAC(1)-1	1, 3 thru 11 K,L,M,S,T				
CAC(2)-1	1, 3 thru 11 K,L,M,S,T				
CAC(11)-1	1 thru 11 K,L,M,S,T				
CAC(12)-1	1, 2, 3, 7				
CAC(21)-1A	1				
CAC(21)-1B	4 thru 11 K,L,M				

Table 3.2-1 Attachment 17

Other Heating, Ventilating, Air-Conditioning

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-67)-10					
PI(1)-ST-(IR-67)-13					
PI(1)-ST-(IR-68)-15					
PI(1)-ST-(IR-69)-16					
PI(1)-ST-(IR-69)-17					
PI(1)-ST-(IR-71)-12					
PI(1)-ST-(IR-71)-13					

Table 3.2-1 Attachment 18

Condensate Storage and Transfer

Component	Code Case	Comments	Component	Code Case	Comments
COND(98)-1	1, 4 thru 11 K,L,M,N				

Table 3.2-1 Attachment 19

Instrument and Service Air

Component	Code Case	Comments	Component	Code Case	Comments
CAS(5)-1	1, 3 thru 11 K,L,M				
SA(1)-1	1, 4 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 20

Containment Instrument Air

Component	Code Case	Comments	Component	Code Case	Comments
CIA(3)-2	1, 3 thru 11 K,L,M				
CIA(5)-1A	1, 3 thru 11 K,L,M				
CIA(5)-1B	1 thru 11 K,L,M				
CIA(5)-2A	1, 3 thru 11 K,L,M				
CIA(5)-2B	1, 3 thru 11 K,L,M				
PI(1)-1-X56					
PI(1)-ST-(IR-67)-8					
PI(1)-ST-(IR-67)-9					
PI(1)-ST-(IR-68)-8					
PI(1)-ST-(IR-68)-9					
PI(1)-ST-(IR-68)-10					
PI(1)-ST-(IR-71)-17					
PI(1)-ST-(IR-71)-18					
PI(1)-ST-(IR-72)-1					
PI(1)-ST-(IR-74)-4					
PI(1)-ST-CIA-PIS-5					
PI(1)-ST-CIA-PIS-6					
PI(1)-ST-CIA-PIS-7					
PI(1)-ST-CIA-PIS-8					

Table 3.2-1 Attachment 21

Diesel Generator System

Component	Code Case	Comments	Component	Code Case	Comments
DO(1)-1A	1, 3 thru 11 K,L,M				
DO(1)-1B	1, 4 thru 11 K,L,M				
DO(1)-1-HPCS	1, 3 thru 11 J,K,L,M				
DO(9)-1A	1				
DO(9)-1B	1				
DO(9)-1-HPCS	1				
PI(1)-ST-DO-LS-11A					
PI(1)-ST-DO-LS-11B					
PI(1)-ST-DO-LS-13					

Table 3.2-1 Attachment 22

Power Conversion System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-81)-1					
PI(1)-ST-(IR-81)-3					
PI(1)-ST-(IR-82)-1					
PI(1)-ST-(IR-82)-3					
PI(1)-ST-(IR-83)-1					
PI(1)-ST-(IR-83)-3					
PI(1)-ST-(IR-84)-1					
PI(1)-ST-(IR-84)-3					
PI(1)-ST-MS-PTD-1A					
PI(1)-ST-MS-PTD-1B					
PI(1)-ST-MS-PT-54A					
PI(1)-ST-MS-PT-54B					
PI(1)-ST-MS-PS-56A					
PI(1)-ST-MS-PS-56B					
PI(1)-ST-MS-PS-56C					
PI(1)-ST-MS-PS-56D					
MS(1)-4A	1, 3 thru 12 K,L,M,S,T				
MS(1)-4B	1, 3 thru 11 K,L,M				
MS(1)-4C	1, 3 thru 12 K,L,M,S,T				
MS(1)-4D	1, 3 thru 12 K,L,M,S,T				

Table 3.2-1 Attachment 23

Main Steam Isolation Valves Leakage Control System (DEACTIVATED)

Component	Code Case	Comments	Component	Code Case	Comments
MSLC(2)-1	1 thru 11 K,L,M,S,T				
MSLC(4)-1	1, 3 thru 11 K,L,M				
PI(1)-ST-(IR-21)-7					
PI(1)-ST-(IR-21)-9					

Table 3.2-1 Attachment 24

Containment Vessel

Component	Code Case	Comments	Component	Code Case	Comments
Containment Vessel	F				
Containment System	2, 14, H	Pen. Assem. & Stiffener			

Table 3.2-1 Attachment 25

Postaccident Sampling System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X73f					
PI(1)-4S-X77Ac					
PI(1)-4S-X77Ad					
PI(1)-4S-X80b					
PI(1)-4S-X82d					
PI(1)-4S-X82f					
PI(1)-4S-X83a					
PI(1)-4S-X83f					
PI(1)-4S-X84e					
PI(1)-4S-X84f					
PI(1)-4S-X88					

Table 3.2-1 Attachment 26

Key to Code Cases

The Following Code Cases Were Used as Noted on the Attachments

Regulatory Guide 1.85		Regulatory Guide 1.84	
1.	N-242-1	A.	N-154
2.	1567	B.	N-122
3.	1713	C.	N-318
4.	1644 thru 1644-6	D.	N-316
5.	N-71-7 (1644-7)	E.	N-328
6.	N-71-8 (1644-8)	F.	N-362
7.	N-71-9 (1644-9)	G.	1606-1
8.	N-249	H.	N-58 (1614)
9.	N-249-1	J.	N-192
10.	1728	K.	1718
11.	N-225	L.	N-111 (1729)
12.	N-224-1	M.	N-247
13.	1571	N.	N-240
14.	N-274	P.	N-272
		R.	N-252
		S.	1683-1
		T.	N-175 (1818)
		U.	N-411-1

Table 3.2-2

Code Group Designations - Industry Codes and Standards
for Mechanical Components[a,b]

		ASME Section III Code Applicable Subsection				
Quality Group Classification	ASME Section III Code Classes	Pressure Vessels and Heat Exchangers	Pumps, Valves, and Piping	Metal Containment Components	Storage Tanks 0-15	Storage Tanks Atmospheric
A	1	NA and NB TEMA C	NA and NB[c]	-	-	-
B	2 or MC	NA and NC TEMA C	NA and NC[c]	- NA and NE	NA and NC API-620[d]	NA and NC or D100, B96.1 or API-620[e]
C	3	NA and ND TEMA C	NA and ND[c]	- -	NA and ND or API-620[f]	NA and ND or D100, B96.1 or API-650[f]
D		ASME Section VIII Division 1 TEMA C	Piping and valves B31.1[g] Pumps[h]	-	API-620 or equivalent[i]	API-650 AWWA-D100 ANSI B96.1 or equivalent[i]

[a] With options and additions as necessary for service conditions and environmental requirements.

Table 3.2-2

Code Group Designations - Industry Codes and Standards
for Mechanical Components[a,b] (Continued)

[b] Components of the reactor coolant pressure boundary meet the requirements of 10 CFR 50, Section 50.55a, "Codes and Standards," except as shown in Table 5.2-1 and discussed in Section 5.2.1.1. All components satisfy codes and addenda in effect at the time of component order or later.

[c] For pumps classified A, B, or C; applicable subsections NB, NC, or ND; respectively, in ASME Code Section III is used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting.

[d] 100% volumetric examination of the sidewall and roof weld joints for plates over 3/16 in. thick and 100% surface examination of weld joints for plates 3/16 in. thick or less of the sidewall to bottom and sidewall to roof joints. These examination requirements are performed in accordance with the rules of ASME Section III, Class 2.

[e] 100% volumetric examination of the sidewall weld joints for plates over 3/16 in. thick and 100% surface examination sidewall to bottom joints. These examination requirements are performed in accordance with the rules of ASME Code Section III, Class 2.

[f] Nondestructive tests examination requirements per ASME Code, Section VIII, Division 1.

[g] Welds not totally conforming to B31.1 are evaluated and dispositioned on a case-by-case basis considering (a) the function of the systems, (b) the risk of failure, and (c) the consequences of failure for safety and plant availability.

[h] For pumps classified Group D, and operating above 150 psig and 212°F Section VIII, Division 1 is used as a guide in calculating the wall thickness for pressure-retaining parts and in sizing the cover bolting.

[i] Tanks are designed to meet the intent of API, AWWA, and/or ANSI 96.1 standards as applicable.

Table 3.2-3

Summary of Safety Class Design Requirements (Minimum)

Design Requirements	Safety Class 1	Safety Class 2	Safety Class 3	Safety Class G
Quality group classification[a]	A	B	C	D
Quality class[b]	I	I, II+	I, II+	II, G
Seismic category[c]	I	I, 1M	I, 1M	II

[a] The equipment is constructed in accordance with the indicated code group listed in Table 3.2-1 and defined in Table 3.2-2.

[b] I (1 in MEL) - the equipment is constructed in accordance with the quality assurance requirements of 10 CFR 50, Appendix B, and Nuclear Energy Division Boiling Water Reactor Quality System Summary.

II+ (A in MEL) - Non-safety-related equipment for which Columbia Generating Station has made specified commitments to the NRC or others relating to the quality. Some 10 CFR 50, Appendix B, criteria are applied.

II (2 in MEL), G - The equipment is constructed in accordance with the quality assurance requirements as specified in contract documents as described in Section 3.2.3.

[c] I - the equipment of these safety classes is constructed in accordance with the seismic requirements for the safe shutdown earthquake as described in Section 3.7.

1M - The equipment does not perform a safety-related function but must be seismically supported/mounted per Regulatory Guide 1.29.

II - The seismic requirements for the safe shutdown earthquake are not applicable to the equipment of this classification. The approaches outlined in the Uniform Building Code are followed where applicable.

Deleted

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 a basic wind velocity (fastest mile), including gusts, of 100 mph at an elevation of 30 ft above the site grade. This wind velocity exceeds the basic wind velocity having a statistically derived probable period of recurrence of 100 years in this geographical area, as specified in the American Society of Civil Engineers Task Committee Report (Reference 3.3-1).

The Hanford region experiences high wind speeds due to squall lines, frontal passages, strong pressure gradients, and thunderstorms. The Hanford Reservation has experienced only one recorded tornado (June 1948) and has not been known to be affected by typhoons. No complete statistics are readily available which present frequency of occurrence of high winds produced or accompanied by a particular meteorological event. However, the highest winds produced by any cause are tabulated for the Hanford Meteorological Station (HMS) in Tables 2.3-5 and 2.3-6. Figure 2.3-4 indicates the return probability of any peak wind gust again due to any cause. The 100-mph design wind speed is conservative for the CGS site for the following reasons:

a. Peak wind gusts measured at the 50-ft HMS tower level, as reported in Tables 2.3-5 and 2.3-6, have never exceeded 80 mph, and

b. The statistically derived 100-year return period peak gust (Figure 2.3-4) at an elevation of 50 ft is 86 mph based on HMS records.

Recurrence intervals, data sources, and the history of occurrence of high winds, hurricanes, and tornadoes in the vicinity of the site are discussed in the CGS Environmental Report.

3.3.1.2 Determination of Applied Forces

The basic wind velocity of 100 mph is applied in accordance with Table 1(a) of Reference 3.3-1, including the variation in wind velocity with height and drag coefficients. Considering a gust factor of 1.0 and drag coefficient of C_D and V as the specific wind velocity at a particular height above grade given in Table 1(a) of Reference 3.3-1, the total combined average wind pressure p is given by

$$p = C_D \times 0.002558 V^2$$

For example, at a height of 30 ft with V = 100 mph, C_D = 1.3, p = 33 psf

The above chosen gust factor is adequate for the CGS site for the following reasons. In Reference 3.3-1, the wind gust factor is defined as gust velocity divided by the velocity of the fastest mile of wind. Whereas the gust factor can not be less than 1.0, it is implied in Section 3.3.1.1 that the factor is unity since the 100-mph fastest mile wind accounts for wind gusts. A factor greater than 1.0 is not warranted at CGS since the statistically derived peak wind gust considered is less than the assumed site basic wind velocity. Additional considerations regarding gust factors and fastest mile velocities are presented in Section 2.3.1.2.1.4.

The following wind velocities are used in the design of structures at various elevations:

Height Above Grade (ft)	Wind Speed, V (mph)*
Less than 50	100
50 to 149	120
150 to 400	140

The wind pressures are applied as static forces. The translation of wind velocities into applied static forces, the wind force distribution, and the drag coefficients are in accordance with published values such as Reference 3.3-1. *The italicized information is historical and was provided to support the application for an operating license. The magnitude and distribution of the applied static forces originally calculated for structures are as follows:*

Height Above Grade	*Windward Side 0.8q (psf)*	*Leeward Side 0.5q (psf)*
Less than 50 ft	*20*	*13*
50 to 150 ft	*30*	*18*
150 to 400 ft	*0*	*25*

The term q, as described in Reference 3.3-1, is designated as the dynamic pressure of a free wind stream at a point on the surface of a structure immersed in the wind stream and when multiplied by the pressure coefficient, Cp, characteristic of the building types defined in Reference 3.3-1, gives the pressure on the structure's surfaces.

The original reactor building membrane roofing system has been replaced with a modern elastic sheet membrane system in accordance with the Factory Mutual system for Class I Fire and I-90 windstorm rating. Local wind uplift forces at the roof perimeter and corners would result in local failure of the existing controlled release fasteners during a gale or squall windstorm.

* *Table 1(a), Reference 3.1-1.*

3.3.2 TORNADO LOADINGS

3.3.2.1 Applicable Design Parameters

The original tornado design criteria provided for wind speeds of 300 mph rotational and 60 mph translational velocity, with a pressure drop of 3 psi to occur in 3 sec as discussed in Section 2.3.1.2.1.3, Tornados. The tornado design criteria were updated in January 1996 based on the design basis tornado characteristics in NUREG-1503 (Reference 3.3-2), which were found acceptable by the NRC (References 3.3-3 and 3.3-4).

Since January 1996, the updated design basis tornado is one having a maximum horizontal component of tangential (or peripheral rotational) wind velocity of 160 mph and a constant translational velocity of 40 mph. The resultant wind velocity (the sum of the maximum horizontal component of tangential velocity and the translational velocity) is 200 mph. The atmospheric pressure at the center of the tornado is 0.9 psi below ambient. The 0.9 psi external pressure drop is assumed to occur at a rate of 0.3 psi/sec. The nonventing structures are designed for the worst combination of wind velocity and associated atmospheric pressure drop in accordance with the load combinations contained in Section 3.3.2.2. The venting structures are designed for the worst combination of wind velocity and associated difference of the pressure within and the atmospheric pressure outside. The effects of a design basis tornado are considered in combination with other loads, including tornado-generated missiles, in Section 3.5.

3.3.2.2 Determination of Forces on Structures

Design static pressures, drag coefficients, and wind pressures are selected in accordance with published values such as Reference 3.3-1. The provisions for gust factors and variation of wind velocity, noted therein, are not applied for the following reasons.

The wind velocity may vary with the height of the structures but, for conservatism, the wind force due to tornado loadings is applied as a uniform static load invariant with the height above grade.

The total wind velocity occurs only in a localized area but is used in the design over the full height of the projected area of the structure. The total wind velocity is in effect the gust wind velocity, since by definition a tornadic gust wind velocity is a high localized wind velocity of very short duration. Therefore, no additional gust factor is applied.

The procedure used to transform the design-basis tornado wind velocity into an effective pressure applied to exposed surfaces of structures is described in Reference 3.3-1.

The following is an example of how Reference 3.3-1 was applied in the original calculations:

The same procedure as that used to transform the basic design-basis wind velocity in Section 3.3.1.2 is used with the exception that the velocity and velocity pressure are assumed not to vary with height.

The equivalent uniform tornado wind velocity used on the structure due to a tangential component of 300 mph and a transitional component of 60 mph is 360 mph. The pressure loads are calculated on the basis of a uniform 360 mph wind velocity.

a. *The dynamic pressure on the structure is:*

   $q = 0.002558 \times (360)^2 = 331.5$ *psf*

b. *The applied static pressures are:*

   1. *Windward pressure on walls:*

      $p = 0.8 \times 331.5 = 265$ *psf*

   2. *Leeward suction on walls:*

      $p = 0.5 \times 331.5 = 166$ *psf*

   3. *Total design pressure on the structure is the sum of 265 psf and 166 psf, or 431 psf.*

The differential pressure loading is calculated using the following pressure-time function: The differential pressure is assumed to vary from zero to 0.9 psi at a rate of 0.3 psi/sec and then return to zero at 0.3 psi/sec.

The procedure used for transforming the tornado-generated missile loadings into effective static loads is described in Section 3.5.3.

Mathematical models for the design-basis tornado take into consideration the phase relationship between the wind load and the differential pressure effects.

The tornado load, W′, in the load combinations in Tables 3.8-9 and 3.8-10 constitutes the combined effect of the three separate loads a, b, and c generated by the design basis tornado. The three loads are combined in the following manner to obtain the combined effect:

a. $W' = W_w$

b. $W' = W_p$

c. $W' = W_m$

d. $W' = W_w + 0.5W_p$

e. $W' = W_w + W_m$

f. $W' = W_w + 0.5W_p + W_m$

where:

W' = Total tornado load

W_w = Tornado wind load

W_p = Tornado differential pressure load

W_m = Tornado-generated missile load

3.3.2.3 Additional Design Features

Except for the steel superstructure atop the refueling floor, the reactor building remains sealed through the tornado event and a differential pressure of 0.9 psi across the exterior and interior is bounded by the design. All other Seismic Category I structures are provided with adequate openings to relieve a differential pressure of 0.9 psi in 3 sec or are designed to withstand an external pressure drop of 0.9 psi.

The structural steel frame superstructure atop the refueling floor of the reactor building is designed to withstand the design basis tornado. However, all the siding and roof decking enclosing the steel superstructure is designed for a maximum differential pressure of approximately 0.5 psi. The siding and girts are designed to blow off the steel frame when a differential pressure of approximately 0.5 psi is exceeded. The roof decking and roof purlins are designed to blow off the steel frame when a differential pressure of approximately 0.5 psi is exceeded. This value considers the dead weight loading from the roof membrane, roofing insulation, roof decking, and roof purlins. This is ensured by the use of controlled release type fasteners connecting the girts to the columns and roof purlins to the roof trusses. The release of the girts, siding, roof purlins, and roof decking from the structural steel frame will not affect the ability to shut down the reactor, the integrity of the primary containment or other Seismic Category I structures, or the capability of the essential heat removal systems to perform their intended design functions.

The design of the reactor building crane and its support system considers tornado effects in addition to normal loads to eliminate the possibility of generating internal missiles which may endanger the primary and secondary containment structures. The trolley is provided with

latches to engage tornado racks attached to the bridge girder to prevent horizontal movement of the trolley due to tornado loadings. The bridge trucks are provided with latches to engage tornado racks attached to the crane runway girders to prevent horizontal movement of the bridge due to tornado loadings. See Figure 3.8-44.

Based on the General Electric publication, (Reference 3.3-5), there is no credible mechanism by which a significant amount of water could be sucked from the fuel pool by a tornado.

The design of hardened containment vent system considers tornado effects in addition to normal loads to eliminate the possibility of creating a tornado generated missile.

The design considers the possible effects of tornado-generated missiles discussed in Section 3.5. Primary containment and components, equipment, and systems essential to a safe shutdown are protected from tornado-generated missiles by enclosing structures.

The diesel generator building and the radwaste and control building are designed to withstand the effects of tornado-generated missiles that might be released, such as girts and roof purlins. The results of analyses determining the effects of missiles are discussed in Section 3.5.

Piping and cabling required for safe shutdown and which penetrate exterior walls of the tornado-resistant structures are located below grade or are protected, as in the case of the standby service water (SW) piping at the service water pump houses, by tornado-resistant structures. Piping and cabling required for safe shutdown, which is provided with less than adequate earth cover for tornado protection, is protected with a tornado-resistant concrete slab or other structure. For information concerning protection provided for the SW system see Section 9.2.5.

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

The two reinforced-concrete spray pond structures have the capability to tolerate tornado generated missiles. Each pond is made up of four perimetral walls, 1 ft 3 in. at the top and 2 ft 0 in. at the bottom, and a floor slab of 7-in. thickness placed on top of a 2-in.-unreinforced-concrete subgrade leveling slab, as described in Section 3.8.4.1.5. Finish grade around the perimetral walls of the spray ponds is 1 ft 0 in. below the top of the walls, and the normal pond water level is at 6 in. below the top of the walls. The walls and slab are bounded by Quality Class I high relative density backfill. On this basis, missile protection is provided for the pond structures. A direct hit by a tornado-generated design basis missile resulting in localized floor and wall penetration is unlikely because of the protection provided by the backfill and the water in the pond.

Damage to the spray pond concrete structure due to tornado-generated missiles would be localized cracking in the area of impact. The structure will remain intact and any leakage will be made up by the cooling tower makeup system which pumps water directly from the river.

The makeup water pump house is a non-Seismic Category I structure impervious to tornado damage. The pump house and structures associated with the cooling tower makeup system, such as valve boxes, are designed to withstand the design basis tornado, including tornado-generated missiles. Piping, valves, and electrical equipment are protected in a similar way. Exterior wall and roof penetrations are designed and anchored to withstand the design basis tornado and the effects of tornado-generated missiles. Tornado-generated missiles are discussed in Section 3.5. Section 9.2.5.3 provides a discussion on the makeup water system and ultimate heat sink interaction if tornado damage to the spray headers require a feed-and-bleed mode of operation.

The availability of essential electrical power to the makeup water pump house systems is ensured. The electrical lines are underground with sufficient earth cover to resist tornado-generated missiles.

The electrical lines are installed in such manner as to provide two redundant electrical systems from the power source to the makeup water pump house. The two electrical systems are physically separated to provide adequate missile protection of one system from the other. At one end of each system, redundant power source transformers, associated switchgear, and cabling are provided on the ground floor of the turbine building where, for the trajectories required to cause damage to this equipment, they are protected against missile impact and spalling by the exterior walls of the turbine building and the floor slabs overhead. The terminal ends and transformers at the makeup water pump house are enclosed within the tornado-resistant pump house. Manholes within each system are also designed to withstand tornado-generated missiles.

The spray pond piping and supports are designed to withstand the effects of the design basis tornado. The piping system cannot be protected from the impact of tornado-generated missiles. In the event of missile damage to one of the pond spray headers, the alternate spray system which is 100% redundant is placed in operation. (In the event that both spray systems are rendered inoperative, the cooling tower makeup water system is placed into operation to provide continuous water supply to the spray ponds using Columbia River water which based on historical data has not exceeded 70°F.) The cooling tower makeup water system is provided with sufficient protection to prevent its loss of function in the event of a design-basis tornado passing over the project site. Since the makeup water flow rate exceeds that of the SW systems, and since the makeup water temperature is lower than the SW system design temperature of 85°F, the continuous availability of cooling water is ensured. Procedures for alternate spray pond use is described in Section 9.2.5.

Failure of nontornado-resistant cooling towers due to tornado loads does not endanger Seismic Category I structures since the plant arrangement provides sufficient distance between the cooling towers and Seismic Category I structures.

The liquid nitrogen storage tank located at the corner of the diesel generator building will not fail due to tornado wind loads. However, if the design-basis tornado missile were to strike the tank straight on near the top of the tank, it could be toppled. Toppling of the liquid nitrogen storage tank due to the impact of a tornado missile can cause the entire contents of the tank to be rapidly emptied in the vicinity of the inerting system skid. There is no safety-related equipment in the vicinity of the tank that would be affected by the cryogenic temperatures associated with liquid nitrogen. In addition, due to the turbulent mixing produced in close proximity to a tornado, no oxygen deficiency condition could be sustained outdoors at the diesel generator intake structures.

Failure of nontornado-resistant structures and components will not affect the ability to shut down the reactor, the integrity of the primary containment or other Seismic Category I structures, the capability of the essential heat removal systems to perform their intended design functions, nor result in the release of radioactivity.

3.3.2.5 Conformance with Regulatory Guide 1.76, Revision 0

The discussion of conformance with Regulatory Guide 1.76, Revision 0, is in Section 1.8.3.

3.3.3 REFERENCES

3.3-1 Task Committee Report, Wind Forces on Structures, Transactions of the American Society of Civil Engineers, Paper No. 3269, Vol. 126, Part II, 1961.

3.3-2 Nuclear Regulatory Commission, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, NUREG-1503, Vol. 1, July 1994.

3.3-3 Letter from J. V. Parrish, Supply System, to Document Control Desk, NRC, Subject: WNP-2 Request for Approval to Revise Tornado Design Criteria, October 10, 1995 (G02-95-212).

3.3-4 Letter from J. W. Clifford, NRC, to J. V. Parrish, Supply System, Subject: Revision of Tornado Design Criteria for the Supply System WNP-2, dated January 24, 1996 (G12-96-032).

3.3-5 Miller, D. R., and Williams, W. A., Tornado Protection for the Spent Fuel Storage Pool, General Electric Company, APED-5696, November 1968.

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

3.4.1.1 External Flood Levels

3.4.1.1.1 Design Basis Flood

From the flooding conditions considered in Section 2.4, the design basis flood (DBF) condition arrived at for use in the protection of Seismic Category I structures and safety-related systems and components is that flood which results from the probable maximum precipitation (PMP) event. The PMP event results in a flood elevation of 433.3 ft mean sea level (msl). This event includes an additional 2.2 ft to account for wind wave action.

3.4.1.1.2 Breach of the Grand Coulee Dam

Floods associated with breaches of the Grand Coulee Dam are discussed in Section 2.4.4. The associated highest flood level is at el. 424 ft msl with wave action included.

3.4.1.1.3 Acceptance Criteria

The facility design and equipment locations are in accordance with General Design Criterion 2, as related to systems and components withstanding flood conditions, and Regulatory Guide 1.59, Revision 1, dated April 1976.

3.4.1.2 Groundwater Levels

3.4.1.2.1 Design Basis Groundwater

The design basis groundwater conditions are defined in Section 2.4.13. From the groundwater conditions considered in Section 2.4.13, the design-basis groundwater condition for use in the protection of Seismic Category I structures and safety-related systems and components is a rise of the present groundwater table with the possible construction of Ben Franklin Dam as discussed in Section 2.4.13.

The design-basis groundwater is el. of 420 ft msl. The normal groundwater table is at approximately el. 380 ft msl.

3.4.1.2.2 Breach of the Grand Coulee Dam

The effects on the groundwater table due to a breach of the Grand Coulee Dam is discussed in Section 2.4.4. This flood would peak for a short time and would not affect the design-basis groundwater level of 420 ft msl due to its short duration.

3.4.1.2.3 Design Basis Flood Probable Maximum Precipitation

As stated in Section 3.4.1.1.1, the worst DBF condition results from the PMP event. Due to the short duration of this flood and its confines, groundwater level at the site would not be affected.

3.4.1.3 Identification of Structures, Systems, and Components

Seismic Category I structures and safety-related systems and components are identified in Section 3.2. Figures 1.2-1 through 1.2-24 show locations and elevations for systems and components. See Section 2.4.2 for discussion of flood protection of safety-related systems and components.

3.4.1.4 Description of Structures, Systems, and Components

3.4.1.4.1 Flood Protection Requirements

3.4.1.4.1.1 External Flood Protection Requirements. The plant site elevation at Seismic Category I structures and safety-related systems and components is approximately 441 ft msl, except at the spray ponds where the finish grade elevation is 434 ft msl and the spray pond overflow weirs are at 434.5 ft msl. These elevations are sufficient to protect the plant site and, therefore, Seismic Category I structures and the safety-related systems and components housed therein against the DBF. Exterior and access openings to all Seismic Category I structures are located above the plant site grade and, therefore, above the DBF level. Flood protection measures are not provided since they are not required.

3.4.1.4.1.2 Internal Flood Protection Requirements. Section 3.4.1.5.2 discusses internal flood protection measures provided for safety-related systems, equipment, and components.

Figures 1.2-2, 1.2-7 through 1.2-17, and 1.2-22 illustrate plant arrangement and layout and show that safety-related equipment is located within individual rooms or compartments. The pump rooms located on the 422 ft 3 in. elevation are enclosed by reinforced-concrete walls. Penetrations and doors in these walls are provided with seals that minimize the effects of flooding between rooms should a break occur in one of the rooms.

The potential flooding and environmental effect attributable to postulated through-wall leakage cracks in moderate-energy fluid piping systems and postulated rupture of high-energy fluid piping systems are evaluated and discussed in Section 3.6.1.

Section 6.3 addresses single failure of the emergency core cooling systems (ECCS) piping, including leak detection requirements for ECCS passive failures, ECCS passive failures during long-term cooling, and potential flooding attributable thereto.

Section 9.3.3 discusses the design bases, system descriptions, safety evaluation testing and inspection requirements, and the instrumentation requirements relative to equipment and floor drainage systems. The design bases used ensure equipment and floor drainage systems integrity during normal plant operation and preclude any danger to health and safety of plant personnel, the environs and the general public. Five independent sumps are provided in the reactor building at floor el. 422.25 ft. These sumps serve pump rooms as shown on Figures 1.2-6 and 1.2-7. There is a single equipment drain sump for the reactor building (see Figure 9.3-9) located in the CRD/Condensate pump room. This sump also connects to the RCIC pump room through an unisolable drain header. The other four sumps are floor drain sumps serving the ECCS pump rooms. A single floor drain sump serves no more than two rooms. A sump located in one room is connected to a second room by a drain header containing a single isolation valve. Each sump is equipped with a level monitoring system, which automatically actuates the sump pump when high water level is reached in the sump. Each sump pump discharges to the Radwaste System. The isolation valve located in the header between connected rooms automatically closes on high water level in the sump to isolate the room with the leak.

The sumps collect water from such typical sources as equipment drains from the drywell and from other equipment carrying low purity water and from the drywell floor drain and other floor and pit drains. In the event of a pipe break in one of the pump rooms of sufficient size to exceed the capacity of the sump pump and overflow a sump in one room, the effects of common mode flooding between pump rooms are minimized by the following:

a. The reactor building equipment drain sump, located in the CRD/Condensate pump room, serves only the RCIC and the CRD/Condensate pump rooms. Although both rooms will be affected by a flood occurring in one room, analysis ensures that sufficient safe shutdown equipment remains unaffected by the flood, preserving the ability to safely shutdown the plant. Existing equipment drains located in the RHR pump rooms A and B are capped and, thus, do not connect to each other or to a common equipment drain sump. See Section 9.3.3.2.1.

b. The floor drain sumps serving more than one pump room have a single isolation valve in the drain header between pump rooms. This valve is fail-safe and automatically closes on a high water level in the sump. Accordingly, flooding

in any pump room, exceeding the capacity of the sump pump, will be confined to the room in which the leakage occurs with only minimal leakage through wall penetrations and door seals into other rooms or areas. Analysis is performed to ensure that sufficient safe shutdown equipment remains unaffected by the flood, preserving the ability to safely shutdown the plant. The high sump level also annunciates in the main control room. See Section 9.3.3.2.2.1.

c. Wall-mounted Class 1E level switches are also located in each of the ECCS and RCIC pump rooms and are mounted just above floor level. These level switches ensure that if a failure of the sump pump, the sump header isolation valve, or sump alarm system should occur in any of these rooms during a flood event, prompt operator notification of the event would be received allowing sufficient time for mitigating actions and safe plant shutdown. See Sections 6.3.2.5 and 9.3.3.2.2.1.

Administrative controls ensure that separation criteria is maintained by ensuring the appropriate doors and hatches are closed.

3.4.1.4.2 Groundwater Protection Requirements

Seismic Category I structures house safety-related systems and components and Seismic Category I components. The elevation of the lowest floor surface of these structures is as follows:

Structures	Elevation of Top of Floor
Reactor building	422 ft 3 in.
Radwaste and control room areas of radwaste control building	437 ft 0 in.
Diesel-generator building	441 ft 0 in.
Spray ponds 1A and 1B	420 ft 0 in.
Standby service water pump houses 1A and 1B	408 ft 3 in.
Retaining area for the condensate storage tanks	441 ft 0 in.

Seismic Category I structures and safety-related systems and components are located above the present groundwater el. 380 ft msl and are not subject to any force effects of buoyancy and static water from this groundwater elevation. Uplift and increased lateral hydrostatic pressure are considered in the design of all Seismic Category I structures and safety-related systems and components, to ensure their safety in the event of a rise in the groundwater table to 420 ft msl. Standby service water pump houses 1A and 1B are designed to resist the increased hydrostatic pressure which would result from the rise in the groundwater to el. 420 ft msl. The lowest floor surface in the reactor building is the top of the foundation mat at el. 422 ft 3 in. msl. Since this is above the design basis groundwater level, the structure is unaffected by the force effects of buoyancy and static water due to groundwater at el. 420 ft msl. Groundwater el. 420 ft msl was compared with foundation levels of Seismic Category I structures and it was determined that waterproofing is not required. Seismic Category I piping and electric conduit penetrations that are below grade are above the design basis groundwater table, and sealing against groundwater pressure is therefore not required. However, all pipes penetrating exterior walls are waterproofed sealed by boots installed on both sides of the wall penetration; all electrical conduit penetrations are through-wall waterproof sealed using silicon foam.

The only materials underlying the site that might exhibit unfavorable response to seismic or other events under saturated soil conditions are the loose to medium dense, fine to coarse sand with scattered gravel, in the upper approximate 40 ft of the soil profile. These were removed and recompacted as structural fill, as described in Sections 2.5.4.8 and 2.5.4.12. Structural fill supports the Seismic Category I structures, including the turbine generator building and service building, in the central plant complex. Structural fill, as required, is also utilized below the other Seismic Category I and safety-related structures including underground piping and electrical duct banks. The structural fill is compacted to a minimum of 75% relative density and an average relative density of not less than 85%. The compacted backfill will eliminate the possibility of liquefaction and provide satisfactory foundation performance should the groundwater level at the Columbia Generating Station site rise.

To evaluate the possible effects of a gross rise in groundwater levels, Shannon and Wilson, soil consultants, performed a series of repetitive triaxial tests in identical soils in the dry and saturated states and concluded that saturation would not necessitate changes in allowable bearing pressures or settlement calculations as discussed in Section 2.5.4.10.

To provide conservatism in the design of structures, the seismic dynamic response of the structures and components was examined over a range of soil shear moduli, as discussed in Section 2.5.4.7.

The possibility of soil liquefaction under the effects of the safe shutdown earthquake was evaluated for the conditions found to exist at the Columbia Generating Station site. Soils having high (75% plus) relative densities have been found in the past to be safe against liquefaction. In the soil underlying the foundations at Columbia Generating Station, liquefaction potential can only be assessed on the basis of soil type and general characteristics.

The foundations of all critical structures lie on compacted fill of fine to coarse sand with gravels. The phenomenon of the liquefaction has never been observed in such soils as this, and therefore, liquefaction was not postulated at the Columbia Generating Station site.

3.4.1.5 Flood Protection Measures

3.4.1.5.1 External Flood Protection Measures

As discussed in Section 3.4.1.4.1, external flood protection measures are not provided since they are not required. Equipment is located so that it is not vulnerable. Protection is not required to cope with potential leakage from such phenomena as cracks in structure walls. Since Seismic Category I structures are located at sufficient grade and distance from the Columbia River, as described in Section 2.4, the effects of wind wave action, including spray, do not require flood protection measures.

3.4.1.5.2 Internal Flood Protection Measures

The primary or safety-related functions fulfilled by the reactor building include the capability to withstand the effects of flooding from internal sources. Measures provided are discussed in Sections 3.8.4.1.1, 3.8.4.3.3, and 3.8.4.4.1.1. ECCS and RCIC pump rooms located in the reactor building basement at elevation 422 ft 3 in. are designed to withstand the effects of flooding between and including the top of the foundation mat at el. 422 ft 3 in. and el. 466 ft. Although not watertight, the doors and penetrations in these pump room walls are provided with seals that will minimize flooding between rooms (except for the RCIC and CRD pump rooms that are connected by an unisolable sump pipe) even with significant hydrostatic pressure generated from flooding water levels up to 466 ft. These pump rooms, comprised of exterior walls, interior walls, biological shield wall, isolation valves (when provided), penetrations, and doors are designed so that any one compartment at any one time, flooded to an elevation of 466 ft, can withstand the effects of seismically induced water sloshing loads. Section 3.8.4.3.3 discusses the CRD/Condensate pump room and railroad bay south exterior wall design hydrostatic pressure. Section 3.8.4.1.1 discusses the equipment hatch in the vehicle air lock (railroad bay) floor at el. 441 ft. See Section 3.4.1.4.1.2 for additional nonstructural related internal flood protection measures.

3.4.1.6 Emergency Flood Protection

Emergency flood protection procedures for bringing the reactor to a safe shutdown are unnecessary because flood conditions do not impact safe shutdown operation.

3.4.2 ANALYSIS PROCEDURES

Seismic Category I structures are located at sufficient grade and distance from the Columbia River so that dynamic water forces are precluded. Seismic Category I structures are designed

for the force effects of buoyancy and hydrostatic pressure due to the design basis groundwater in conjunction with loading combinations in Section 3.8.4. Seismic Category I structures are checked for stability and foundation pressures. In all cases, the dead loads of structures maintain stability and a positive soil bearing pressure. Design loads due to groundwater force application are applied to all Seismic Category I structures as follows:

a. A vertical hydrostatic pressure due to the water head below el. 420 ft msl on subgrade surfaces of structures and a lateral hydrostatic pressure on subgrade surfaces of structures. The lateral pressure is treated as an additional triangular loading increasing at the rate of 62.4 lb/ft^2 per vertical foot from el. 420 ft to the bottom of the structure, and

b. A buoyant force equal to the weight of water displaced by the structure.

The loading combinations used in conjunction with hydrostatic pressure due to groundwater, taken from Table 3.8-9, are

a. $U = 1.4D + 1.7L + 1.7\,P_o + 1.4F + 1.7Q$

b. $U = 1.4D + 1.7L + 1.4\,T_o + 1.7\,P_o + 1.4F + 1.7Q$

c. $U = 0.9D + 1.4F + 1.7Q$

d. $U = 0.9D + 1.4\,T_o + 1.4F + 1.7Q$

e. $U = 1.1D + 1.3L + 1.3\,P_o + 1.1F + 1.3Q + 1.3W$

f. $U = 1.1D + 1.3L + 1.1\,T_o + 1.3\,P_o + 1.1F + 1.3Q + 1.3W$

g. $U = 1.4D + 1.7L + 1.7\,P_o + 1.9E + 1.4F^* + 1.7Q$

h. $U = 1.4D + 1.4L + 1.4\,T_o + 1.4\,P_o + 1.4E + 1.4F^* + 1.4Q^*$

In the above load combinations, Q denotes lateral pressure on subgrade surfaces of structures due to either dry or saturated soils; and Q* denotes lateral pressure, including seismic effects, on subgrade surfaces of structures due to either dry or saturated soils. In applying the above loading combinations in conjunction with lateral hydrostatic pressure due to groundwater, the value for saturated soil is used.

In the above load combinations, the vertical hydrostatic pressure due to the water head below el. 420 ft msl and acting on subgrade surfaces of structures is included in D, dead loads, with consideration given to the direction of the hydrostatic pressure. The loading combinations

used in conjunction with hydrostatic pressure due to internal flooding are given in Table 3.8-9. The definitions of load terms used in the load combinations are given in Section 3.8.4.3.3.

3.5 MISSILE PROTECTION

The CGS missile protection design basis conforms to 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 4, Environmental and Missile Design Bases. The objectives of missile protection design are to ensure that the plant can be brought to and kept in a safe shutdown mode and to prevent offsite radiological consequences assuming an additional single component failure.

The primary provisions incorporated into the design of the CGS facility to preclude damage to the systems necessary to achieve and maintain a safe plant shutdown are separation and redundancy. These provisions provide that in the event of an accident plus an additional active component failure, where a system required for safe (cool) shutdown is rendered unavailable, enough systems are left available to bring the plant to a safe (cool) shutdown without allowing any offsite radiological consequences. This redundancy and separation is obtained by the deliberate routing of systems, by the presence of structural floors, walls, structural steel members, and adjacent equipment which serve as barriers.

Design against generated missiles involves an initial selection process to define postulated missiles, an evaluation of postulated missile credibility, then a damage assessment to evaluate the effects of credible missiles, and finally, if necessary, to ensure safe shutdown, the provision of barriers or physical modifications of systems and components to preclude damage.

Structures housing systems and components essential for safe shutdown are designed to withstand externally generated missiles so that essential systems and components are not damaged by such missiles or by the secondary effects of such missiles.

3.5.1 MISSILE SELECTION AND DESCRIPTIONS

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Systems Available for Safe Shutdown

Systems available outside containment to facilitate safe shutdown include: high-pressure core spray (HPCS), low-pressure core spray (LPCS), residual heat removal (RHR), standby service water (SW), reactor core isolation cooling (RCIC), control rod drives (CRD), and the reactor feedwater system (RFW). These systems and their function are described in Sections 4.6, 5.4.9, 7.3, and 7.4.

Figures 3.5-1 through 3.5-15 illustrate the location of these systems. Seismic categories, quality group classifications, and reference sections are provided in Table 3.5-1.

3.5.1.1.2 Missiles Due to Rotating Equipment Failure

The systems located outside the primary containment have been reviewed to identify potential rotating equipment missiles. The design objective is to prevent the generation of missiles and their effects.

All rotating equipment (e.g., pumps, turbines, fans, and compressors) outside the primary containment have been evaluated to determine missile generation potential (postulated missiles), missile credibility, and an analysis of credible missile effects was completed. Credible missiles outside containment, missile sources, safety-related systems requiring protection (if any), and the extent of damage to safety-related systems (if any) are listed in Table 3.5-2. All emergency core cooling systems (ECCS) rotating equipment outside the primary containment are grouped by division in different rooms or areas of the plant, separated by walls or barriers, so that a single missile cannot damage redundant systems. The walls or barriers are designed to contain all missiles.

The RCIC turbine is prevented from reaching a runaway speed, where component failure could occur, by overspeed tripping devices. However, the RCIC turbine, similar to all plant rotating equipment, is also evaluated for credible missile generation at normal full speed operation. In addition, as with the ECCS systems, the RCIC turbine is located in a separate compartment.

3.5.1.1.3 Missiles Due to Pressurized Component Failure

The potential of the following equipment to generate missiles was investigated:

a. High energy piping

   Pressurized piping in systems where the service temperature exceeded 200°F and/or the service pressure exceeded 275 psig was evaluated for potential generation of missiles. High energy piping pipe whip is discussed in Section 3.6.

b. Valve bonnets

   1. Pressurized seal bonnets

      Valves with an American National Standards Institute (ANSI) rating of 900 psig and above are pressurized seal bonnet type valves, constructed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III. Valve bonnets, on pressure seal bonnet type valves, are prevented from becoming missiles by the retaining ring, which would have to fail in

shear, and by the yoke, which would capture the bonnet or reduce bonnet energy.

The bonnet bolts preload the pressure seal gasket to seal the valve initially. When pressurized, the valve is sealed by process fluid pressure and the bonnet bolts are under no load. All ASME III, Class I, 900# bonnet seal type valves were analyzed per ASME B&PV Code Section III requirements. Valve design pressures used in these analyses were given by the ASME B&PV Code Section III, Division 1, Subsection NB, Figure NB-3545.1-2, for weld-end valves. Using a typical pressure seal valve, the total thrust load on the retaining ring and valve body was calculated. The results demonstrated that both the retaining ring and valve body meet the ASME B&PV Code Section III, Division 1, NB-3227, requirements for pressures much higher than the normal operating pressure of the valve.

The majority of valves inside containment have massive valve operators which are supported by the yoke. For these valves, the valve operators act as an additional limitation to the yoke becoming a missile.

For a yoke clamp to fail, it must be assumed that the retaining ring fails completely and instantaneously so that the bonnet could strike the yoke. The yoke is normally under no load and complete failure of the yoke clamp is not considered credible.

Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable, and hence bonnets are not considered credible missiles.

2. Bolted bonnets

Most valves of ANSI rating 600 psig and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material by requirements set forth in the ASME B&PV Code Section III and by designing flanges in accordance with applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing simultaneous complete severance failure is remote. A study of bolted valve bonnets was made in which 25% of the connective bolts in the circular pattern were assumed missing. The stresses occurring under operating conditions with these bolts missing were found to be within acceptable limits. The widespread use of valves with bolted bonnets and the lack of

historical incidence of complete severance failures of bonnets confirms that bolted valve bonnets need not be considered credible missiles.

3. Screwed-typed bonnets

Some valves in the 1 in. to 1.5 in. size range have coarse threaded bonnets which screw into the valve body. These valves were analyzed and found to have low stress intensities in the bonnet retaining threads. The valve design stress intensities were found to be a minimum of 4.5 times the stress intensities that will be experienced by the valves. Because of the highly conservative design of these valves, they are not considered credible missiles.

c. Valve stems

Valve stems are not considered credible missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection; for example, valves with backseats.

d. Thermowells and sample probes

Temperature or other detectors installed on piping or in wells are evaluated as potential missiles if failure of a single circumferential weld would cause their ejection. This is highly improbable since a complete and sudden failure of a circumferential weld is needed for a detector to become a missile. These circumferential welds were analyzed and found to have yield stress values from six to 20 times the stress intensities that will be experienced in service. Because of their highly conservative design, thermowells and sample probes are not considered credible missiles.

e. Nuts and bolts

Nuts, bolts, nut and bolt combinations, and nut and stud combinations are unlikely to fail because of the low stress intensities for these parts. The ASME and ANSI Codes limit the allowable stresses in bolts and studs to 20% to 30% of yield. These low stress intensities are ensured by measuring the torque of all bolts, studs, and nuts during installation. Because of their highly conservative design, nuts and bolts are not considered credible missiles.

f. Blind flanges

Bolted blind flanges are not considered credible missiles because of the extremely unlikely occurrence of all bolts experiencing simultaneous complete severance failure as discussed in Item b.2 above.

g. Nitrogen tanks and bottles

Nitrogen tanks and bottles in the reactor building provide nitrogen for CRDs, charging of main steam safety/relief valve (SRV), isolation valve accumulator tanks, and instrument nitrogen inside containment. These tanks and bottles have design pressures considerably in excess of their operating pressures. Because of their highly conservative design, installed nitrogen tanks and bottles are not considered credible missiles.

No credible missiles are in a position to impact any of the nitrogen tanks or bottles.

3.5.1.1.4 Evaluation of Postulated Missiles

a. Assessment of postulated missile credibility

Postulated missiles are analyzed to determine if a credible failure mode resulting in a missile exists. Failure modes determined to be credible are then assessed for impact on plant safe shutdown.

b. Assessment of potential credible missile damage

The ability of the plant to achieve safe shutdown is ensured by physical separation and redundancy of safety-related systems. The adequacy of the physical separation and redundancy of safety-related systems was evaluated using the following procedure:

1. Target determination

Based on the missile location and orientation, the target areas are predicted. Trajectories are selected to encompass the most adverse conditions. The essential systems within that region are assumed damaged and not available for a safe shutdown;

2. Evaluation of system damage

The essential systems which are available after the worst postulated missile accident and the most critical additional single failure are determined. An evaluation is then made to determine whether these remaining systems are sufficient to achieve safe reactor shutdown; and

3. Protection of systems

When the separation and redundancy of the essential systems is not adequate, or when a redundant system is not available, one or more of the following measures are taken to ensure safe shutdown:

(a) The orientation of the credible missile is changed so that systems necessary for safe shutdown are not damaged,

(b) Missile barriers are provided, and

(c) It is shown that the essential components will not be damaged by the credible missile.

c. Determination of missile energies

One of the following methods is used to calculate the extent of the damage caused by a credible missile:

1. Piston-type missiles

The velocity of a piston-type missile is calculated by assuming that the work done will be converted into kinetic energy of the missile with no losses of energy due to friction or air resistance.

Work is the integral of force times displacement, while the kinetic energy of the missile is one-half the product of the missile mass times the square of the missile velocity. Assuming the applied force constant (PA_o), the kinetic energy is equated to the work done (Reference 3.5-1). Subsequently, the missile velocity is obtained by the expression:

$$V = \left[\frac{2\,PA_o\,L}{m}\right]^{1/2}$$ (Reference 3.5-1)

where:

V = the initial velocity at the end of a piston stroke (ft/sec)

P = pressure of the fluid (psi)

A_o = cross-sectional area of the piston (in.^{2})

L = length of the stroke in ft

m = mass of missile (lb-sec^2/ft)

2. Jet-propelled missiles

Postulated jet-propelled missiles are propelled by fluid escaping from a pressurized system in which there is essentially no lateral containment of the fluid. The escaping jet will not only impinge on the missile, but will also flow around and past the missile.

The velocity of this type of postulated missile is estimated by (Reference 3.5-1):

$$\left(1-\frac{V}{V_f}\right) - \mathrm{Log}_e\left(1-\frac{V}{V_f}\right) = K_1 - \frac{K_2}{N_o + X\ \mathrm{Tan}\ B}$$

where:

$$K_1 = \left(1-\frac{V_o}{V_f}\right) - \mathrm{Log}_e\left(1-\frac{V_o}{V_f}\right) + \frac{K_2}{N_o}$$

$$K_2 = \frac{A_o A_m P_b}{m\pi(\mathrm{Tan}\ B)}$$

V = missile velocity at distance X (fps)

V_f = jet velocity = (fps)

N_o = radius of throat (ft)

P_f = density of the jet fluid (lb-sec^2/ft^4)

X = distance traveled (ft)

B = angle of jet expansion, degrees from normal

V_o = initial velocity of missiles

A_o = throat area (ft^2)

A_m = cross-sectional area of missile (ft^2)

m = mass of missile ($lb\text{-}sec^2/sec$)

3. Stored strain energy missiles

Stored strain energy missiles are assumed to convert all the strain energy at which they fail into kinetic energy. The velocity is calculated from the following formula (Reference 3.5-1):

$$V = \left[\frac{g}{EW}\right]^{1/2} S$$

where:

V = missile velocity (ft/sec)

E = modulus of elasticity (lb/ft^2)

W = specific weight of missile (lb/ft^3)

S = ultimate stress in the missile before failure (lb/ft^2)

g = acceleration of gravity (ft/sec^2)

4. Rotating Machinery

A variety of missiles from rotating machinery can be treated by considering each as a rotating block. Because it is part of a rotating structure, the block is considered to be initially rotating about its axis of revolution at a speed, ω, radians per second. The kinetic energy (KE) of the block is given by (Reference 3.5-4):

$$KE = \frac{1}{2}\left[Rcg^2 + K^2\right]\left(\frac{w}{g}\right)(\omega)^2, \text{ ft} - \text{lb}$$

where:

Rcg = radius to the center of gravity (CG) of the rotating block, measured from the initial axis of rotation in the machinery, ft

K = radius of gyration of the rotating block, about the cg axis of the rotating block

w = block weight, lb

g = acceleration of gravity, ft/sec^2

ω = angular velocity, rad/sec

In this expression the Rcg term gives the kinetic energy due to translation, while the K term gives the kinetic energy due to rotation of the block about its cg axis.

3.5.1.1.5 Example of Postulated Missile Evaluation

a. Assessment of postulated missile credibility

The reactor protection system motor generator sets in the critical dc switchgear rooms in the reactor building (el. 467 ft 0 in.) were analyzed to determine their credibility as missiles. A structural failure of the 1800 rpm flywheel during the normal operation of the motor generator would produce high energy missiles with the potential to damage systems, components, or structures in their paths. Motor generator set modification to eliminate or contain the flywheel missiles was evaluated, but a feasible modification was not practical. The flywheels were, therefore, credible missiles.

b. Assessment of potential credible missile damage

The flywheel missiles were postulated to exit the motor generator sets along a plane perpendicular to the motor generator set, with the missile exiting a maximum of 10 degrees from the perpendicular plane.

1. Target determination

   The potential targets for the flywheel missiles were determined by reviewing the applicable drawings and visual inspection of the target area in the switchgear rooms. It was determined that safety-related cables in these rooms could potentially be damaged.

2. Evaluation of system damage

   The safety-related cables which could be damaged by the flywheel provide dc power to instrument panels in the control room and to isolation valves inside containment. Damage to these cables was determined to be unacceptable.

3. Protection of systems

   It was determined that there was not a feasible method of providing that the cables would not be damaged by the flywheel missiles. To preclude damage to the safety-related cables, the following alternatives were investigated:

   (a) The motor generator sets were analyzed to determine if a change in orientation was feasible. This was not a feasible alternative.

   (b) The feasibility of constructing a barrier around the flywheel was investigated. This was a feasible alternative.

c. Barrier design

A missile barrier proved to be the only feasible alternative. The barrier was designed to contain the highest energy missile that could be produced by the flywheel. The barrier was constructed of steel and energy-absorbing aluminum honeycomb material and firmly anchored to the concrete floor. This eliminated the effects of the credible missile. A tabulation of plant systems protected by missile barriers is provided in Table 3.5-3.

3.5.1.2 <u>Internally Generated Missiles (Inside Containment)</u>

3.5.1.2.1 Systems Available for Safe Shutdown

Figures *3.5-16* through 3.5-32 describe the mechanical and instrumentation locations of systems available for a safe shutdown. Each system (LPCS, HPCS, RHR, ADS, CRD, and primary containment) is color coded to specify the location of structures, systems, or

components. In addition, the reactor protection system and containment isolation valves inside containment are available for safe shutdown of the plant and to prevent offsite radiological consequences. Information pertaining to applicable seismic category, quality group classification, and reference sections where these systems are described is provided in Table 3.5-4. The evaluation of credible missile kinetic energies and missile target determinations is discussed in Section 3.5.1.1.4. Target and barrier damage evaluations are discussed in Section 3.5.3.

3.5.1.2.2 Missiles Due to Rotating Equipment

Rotating equipment inside containment consists of the following:

a. Recirculation pump and motor

The most substantial piece of nuclear steam supply system (NSSS) rotating machinery is the recirculation pump and motor. This potential missile source is discussed in Reference 3.5-4.

It is concluded in Reference 3.5-4 that destructive pump overspeed is highly improbable. If it occurred, it could result in failure of certain pump and motor components having the potential to become missiles. A careful examination of the pump and motor structure shows that rotor or shaft failure will not result in ejection of motor-generated missiles, and impeller missiles cannot penetrate the pump case. Reference 3.5-4 concludes that in the unlikely event of impeller failure resulting in ejection of missiles through ruptured pipe, penetration of containment by missile fragments is highly improbable. Evaluation of the effects on safety-related systems of impeller fragments that might be ejected from openings in ruptured pipe is not evaluated because of the extreme improbability of this event and because the effects would not be more severe than the assumed consequences of jet impingement due to pipe inside containment as discussed in Section 3.6. The recirculation pump and motor are, therefore, not considered to be credible missile sources.

b. Fans as potential missiles

The fans inside primary containment are designed such that the casing will restrain any possible missile. Therefore, fans and parts thereof are not considered as possible sources of missiles.

3.5.1.2.3 Missiles Due to Pressurized Component Failure

A discussion of the potential for missile generation from the failure of pressurized components, e.g., valve stems, valve bonnets, and temperature element assemblies, is presented in

Section 3.5.1.1.3. That discussion is also applicable to pressurized components inside containment. In addition, SRV and main steam isolation valve (MSIV) accumulators are particular to inside containment.

Pressurized ASME III vessels, such as SRV and MSIV accumulators, are not considered credible missiles. These vessels have low stresses and operate in the "moderate energy" range and, therefore, any failures would be a crack-type and not of concern for missile generation.

All potential sources of postulated missiles inside the primary containment were analyzed to determine missile credibility utilizing the criteria discussed above and in Section 3.5.1.1.3, as required by General Design Criterion 4, "Environmental and Missile Design Basis." It was determined that all postulated missiles inside the primary containment incorporated design features that eliminated their credibility as potential sources of missiles.

3.5.1.2.4 Falling Objects

Structural elements, equipment, and components inside containment which could be considered as potential falling objects are evaluated in accordance with Section 3.7.2.8.

3.5.1.2.5 Secondary Missiles Generated by Postulated Credible Primary Missiles

Secondary missiles are not considered credible missiles due to their low probability of occurrence and their low kinetic energy levels. The probability of damage due to a secondary missile is the probability of occurrence of a primary missile times the probability of hitting a part that can become a secondary missile times the probability that the part will actually become a missile. This probability is very low.

The level of stored kinetic energy in a secondary missile will be low because of the large energy required to produce a secondary missile. In addition, no reliable method to predict secondary missile characteristics is known, other than those characteristics in common with primary missiles.

3.5.1.3 <u>Turbine Missiles</u>

In April 1992, the original Westinghouse shrunk-on-disc type LP rotors were removed and new Westinghouse Fully Integral LP rotors were installed. The LP turbine missile analysis is based on Fully Integral rotors using a probabilistic method (Reference 3.5-27).

It is concluded that the probability of damage to safety-related systems by turbine missiles is acceptably low, due to (a) the protection provided by reinforced-concrete structural barriers, (b) the calculated probability of turbine missile generation, and (c) periodic testing and inspection of turbine overspeed protection systems with associated corrective action as required.

3.5.1.3.1 Safety-Related Targets

Target areas which are evaluated for capability to protect safety-related equipment, components, and systems from postulated turbine missiles consist of the following:

a. Vertical targets

   1. Reactor building north exterior wall,

   2. Control room north wall, and

   3. North wall of vertical cable chase, between reactor building and control room.

b. Horizontal targets

   1. Reactor building refueling floor,

   2. Roof over vertical cables chase, and

   3. Floor slab above control room.

3.5.1.3.2 Turbine Placement and Orientation

Figure 3.5-33 shows the turbine generator layout relative to safety-related plant structures and turbine missile target areas. Also shown on this drawing is the reinforced-concrete shield wall which acts as a barrier for protection of some safety-related targets from postulated low trajectory turbine missiles. A cross-sectional view through the turbine building and reactor building is shown in Figure 3.5-34 to indicate relative elevations of the turbine and target areas. See Figures 1.2-4, 1.2-8 and 1.2-16 for a general arrangement drawing of the turbine building, reactor building, and control building at the turbine operating floor elevation. CGS has an “unfavorable oriented” turbine generator in relation to the identified safety-related targets.

3.5.1.3.3 Missile Identification and Characteristics

3.5.1.3.3.1 High Pressure Turbine. Postulated missiles from the high pressure turbine (HPT) are shown in Reference 3.5-22 to have insufficient energy to penetrate the casing at normal operating speed. At 20% overspeed (120% of normal, or rated speed), HPT missiles are postulated to penetrate the casing, but at velocities too low to reach safety-related targets. The minimum bursting speed of the high pressure turbine rotor, based on minimum specified mechanical properties of the rotor material, is 300% of the rated speed (Reference 3.5-6).

The maximum speed at which the unit may rotate is 3200 rpm, which is 178% of rated speed 1800 rpm (Reference 3.5-27). At this speed the highest stressed low pressure turbine disc would fracture, damaging the turbine to the extent that additional overspeed would not be possible. Therefore, high pressure turbine missiles are not considered.

3.5.1.3.3.2 Low Pressure Turbine. Each low pressure turbine consists of a double-flow rotor assembly, an outer cylinder, two inner cylinders, and stationary blade rings. The rotor is a fully integral, single forging consisting of the shaft, disks, and couplings without any shrink fits or keyways and is made with low stress corrosion cracking (SCC) susceptible materials.

Westinghouse Report "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors" (Reference 3.5-27) states

> To assess the probability of missile generation resulting from a burst of a fully integral nuclear low pressure rotor, five potential failure mechanisms are considered:
>
> a. Ductile burst,
>
> b. Fracture resulting from high cycle fatigue cracking,
>
> c. Fracture resulting from low cycle fatigue cracking-startup/shutdown cycles,
>
> d. Fracture resulting from stress corrosion cracking, and
>
> e. Destructive overspeed.
>
> All the listed failure mechanisms demonstrate that the new Fully Integral rotor design significantly reduces the likelihood of missile generation, except for the destructive overspeed mechanism.
>
> The probability of reaching a destructive overspeed for the Fully Integral LPT rotor is primarily dependent on the DEH control system, which functions to avoid this condition.

3.5.1.3.4 Strike and Damage Probability

A probabilistic approach is adopted to assess the possibility of damage to systems required for safe shutdown or of accidents which could result in potential offsite exposure due to high trajectory missiles. The probability of this occurring is represented by combined probabilities of

$$P_4 = P_1 \times P_2 \times P_3$$

where:

P_1 = missile generation probability

P_2 = probability of a missile striking a structure or component required for safe shutdown or whose failure could result in release of radioactivity

P_3 = probability of significant damage to the structure or component

P_4 = combined overall probability

The terms and assumptions applicable to this analysis follow the procedures outlined by S. H. Bush in Reference 3.5-7.

3.5.1.3.4.1 Missile Generation Probability. Missile generation probability is based on the failure probability of the Fully Integral rotor design (installed 1992) and the turbine overspeed protection system. The dominant failure mechanisms of the rotor are SCC and turbine overspeed. The P1 values for SCC were provided by Westinghouse (References 3.5-27 and 3.5-28) and meet the requirements of the NRC Safety Evaluation Report (SER) for specific turbine inspection and testing intervals. However, the SER states the NRC position that the LP rotor operation, maintenance, and inspection programs be based on the P1 values being less than the 1 x 10^{-5} probability requirement for loading the turbine without inspection or inservice restriction. This value is applicable to the CGS unfavorably oriented turbines.

The P1 values for the CGS low pressure turbine Fully Integral rotors for stress corrosion cracking are

Inspection Interval (years)	P1 at Rated Running Speed	P1 at 20% Overspeed
4	4.0×10^{-14}	1.4×10^{-14}
8	8.8×10^{-11}	2.0×10^{-11}
12	4.3×10^{-9}	7.7×10^{-10}
16	5.1×10^{-8}	7.9×10^{-9}
32	8.3×10^{-6}	8.9×10^{-7}

The probability for SCC at 20% overspeed (120% of normal or rated speed) is lower than the SCC at rated speed condition, and therefore not considered in determining the overall strike and damage probability of turbine-generated missiles (Reference 3.5-28).

As with previous rotor designs (including the CGS original built-up rotors and current Fully Integral rotors), the potential exists for SCC to be the dominant failure mechanism depending on inspection interval (Reference 3.5-29). However, in Fully Integral rotor designs, the probability of failure by the SCC mechanism has been reduced drastically and the analysis shows that at the normal inspection interval established by the preventive maintenance (PM) program of 80,000 to 100,000 running hours (translates to 10-12 years), the contribution from SCC to P1 would be approximately 4.3×10^{-9}. Considering typical use factors for nuclear turbines, and considering that the crack locations are readily observable during routine turbine maintenance and inspection per the PM program, it is concluded that the NRC safety criteria of $P1 \leq 1 \times 10^{-5}$ will not be exceeded.

The methodology developed to calculate the probability of missile generation due to overspeed events identifies three turbine overspeed events that can result from the failure of the turbine valves to close following a system separation or a total loss of load (References 3.5-30 and 3.5-31). These overspeed events are design overspeed (approximately 120% of rated turbine speed), intermediate overspeed (approximately 130%), and destructive overspeed (runaway speed in excess of approximately 180%). Design overspeed is assumed when a system separation occurs and a turbine trip does not occur at event initiation, one or more governor valves or two or more interceptor valves fail to close immediately, and a successful overspeed trip closes the throttle valves and reheat stop valves. Intermediate overspeed is assumed to occur when there is a system separation and one or more alignment of reheat stop valves and interceptor valves fail to close. Destructive overspeed is assumed to occur when a system separation occurs and at least one governor valve and one throttle valve in the same steam chest fail to close.

Missile generation probability analyses (References 3.5-30 and 3.5-31), indicate that the design and intermediate overspeed failure probabilities are not major contributors to turbine missile ejection probability for plants with fully integral low pressure rotors.

The P1 value for turbine overspeed is derived from the probability of failure of the turbine overspeed protection system in a manner that results in destructive overspeed and includes consideration for periodic turbine valve testing. Additionally, the P1 value includes conservatism to account for design and intermediate overspeed missile generation. The overspeed protection system is a fault tolerant, redundant, digital electro-hydraulic turbine control and overspeed protection system (installed-May 2007). The system has the capability of failure detection and on-line repair of critical components. The missile generation probability analysis for potential turbine overspeed yields a missile generation probability of 3.4×10^{-6} for P1 based on a twelve month turbine valve testing interval (Reference 3.5-32).

Combining the SCC mechanism (assuming inspection at 12 year intervals) and the destructive overspeed mechanism to establish a combined P1 value for missile generation, yields a probability of turbine missile generation of 3.4×10^{-6}.

3.5.1.3.4.2 Strike Probability (P2) and Damage Probability (P3). Rather than performing elaborate calculations for site-specific strike and damage probabilities as previously performed for the disc type rotors, generally accepted industry typical values were used for P2 and P3. For CGS, the product of P2 * P3 is 1×10^{-2} for unfavorably oriented turbine generators (Reference 3.5-28).

3.5.1.3.4.3 Combined Overall Probability (P4). Assuming the longest turbine inspection interval of 12 years and a destructive overspeed probability assuming a twelve month turbine valve test interval, the highest overall damage probability for postulated missiles is then calculated by

$$\begin{aligned} P4 &= P1 * P2 * P3 \\ &= (3.4 \times 10^{-6}) * (1 \times 10^{-2}) \\ &= 3.4 \times 10^{-8} \text{ (which is acceptably less than the NRC safety criteria of } P4 \leq 1 \times 10^{-7}\text{)} \end{aligned}$$

3.5.1.3.5 Turbine Overspeed Protection System and Testing

A single failure in the overspeed sensing and turbine trip systems will not prevent overspeed protection from operating. The turbine generator is equipped with a fault tolerant and redundant DEH control system.

The DEH control system, with its overspeed protection features and inspection and testing requirements, is described in Sections 7.7.1.5 and 10.2.

3.5.1.3.6 Turbine Valve Testing

Turbine valve testing is discussed in Section 10.2.

3.5.1.3.7 Turbine Characteristics

For information characterizing the Westinghouse turbines used at CGS, refer to Westinghouse report covering the effects of a high pressure turbine rotor fracture (Reference 3.5-6) and low pressure turbine forged integral rotor fractures, (Reference 3.5-27). Therein, the low-pressure disc materials, manufacturing processes, and operating conditions are stated.

3.5.1.4 Missiles Generated by Natural Phenomena

The consideration of potential missiles injected or suspended in a tornado wind stream is based on References 3.5-8 and 3.5-12.

All Seismic Category I and safety-related structures and components are designed to include the effects of missiles generated by the design basis tornado described in Section 3.3.2 except as noted in Sections 3.5.1.4.1 and 3.5.2. Missiles are categorized as either external or internal missiles. External missiles are materials and/or items usually found outside and in the immediate vicinity of the buildings, whereas internal missiles are materials and/or items found inside the buildings. Descriptions, properties, and impact velocities of the design-basis tornado-generated external missiles are shown in Table 3.5-5. Tornado missile protection provided by plant structures is described in Section 3.5.1.4.1. Those structures have adequate thickness to prevent penetration, perforations, and backface spalling. The basis for evaluating missile penetration is discussed in Section 3.5.3. The protection provided for external and internal missiles is discussed in Sections 3.5.1.4.1 and 3.5.1.4.2, respectively.

Figures 1.2-1 through 1.2-24 indicate the location of structures, equipment, and components protected against tornado-generated missiles.

3.5.1.4.1 Tornado-Generated External Missiles

Structures that house systems, equipment, and components essential to safe shutdown are designed to withstand the effects of design-basis tornado-generated missiles described in Section 3.5.1.4. These structures provide protection by the following means:

a. Reactor building

The location of the reactor building with respect to the other plant structures is illustrated in Figure 1.2-1. Portions of the reactor building exterior walls are protected by adjacent structures against direct impact of tornado generated missiles, as indicated in Figure 3.5-35. The exterior walls of the reactor

building, up to the refueling floor at el. 606 ft 10.5 in., are capable of withstanding the impact of the design basis tornado generated missiles. The exterior walls are constructed of 4 ft thick reinforced concrete to el. 471 ft 0 in. which is 30 ft above plant finish grade. From el. 471 ft 0 in. to the refueling floor at el. 606 ft 10.5 in., the exposed exterior walls are constructed of reinforced concrete, 18 in. minimum in thickness. The reactor building exterior wall thickness from plant grade to the refueling floor at el. 606 ft 10.5 in. is adequate to prevent design basis missile penetration and spalling of concrete. (See Figures 3.5-36 through 3.5-38.)

The refueling floor at el. 606 ft 10.5 in. comprises the spent fuel storage pool, dryer/separator pool, and various other items of refueling equipment. The reactor building walls and roof above this floor are constructed of insulated metal siding and insulated metal roof decking erected on a superstructure consisting of a structural steel frame as indicated in Figures 1.2-11 and 1.2-12. The superstructure also supports the overhead bridge crane. The refueling floor at el. 606 ft 10.5 in. is constructed of reinforced concrete of various thicknesses, with a minimum thickness of 18 in. The walls and floor of the pools have a minimum thickness of 5 ft. The equipment on the refueling floor is not required for safe shutdown. The refueling floor and the walls and floor of the pools are sufficiently thick to withstand the effects of the design basis missiles, and to prevent secondary missile effects caused by spalling of concrete. A missile impacting the spent fuel pool does not have sufficient energy to damage the equipment and spent fuel located in the pool, as discussed in Reference 3.5-12.

b. Diesel generator building

The exposed exterior walls of the structure are constructed of reinforced concrete with a minimum thickness of 2 ft 8 in. The roof has a minimum thickness of 1 ft 6 in. The thicknesses of walls and roof are sufficient to withstand the effects of the design-basis tornado-generated missiles. Figures 1.2-22 and 3.5-42 through 3.5-44 illustrate this structure.

c. Radwaste and control building

The exposed exterior concrete walls and roofs, housing safety-related systems, equipment, and components are designed to withstand the effects of the design-basis tornado-generated missiles. Figures 1.2-13 through 1.2-17 illustrate the radwaste and control building and their relative location in the plant complex. The exterior walls that house safety-related equipment have a minimum thickness of 2 ft. The roof of the portion of the building that houses safety-related equipment is 1 ft 6 in. thick (see Figure 3.5-45).

d. Standby service water pump houses and spray ponds

The exterior walls of both pump houses are constructed of reinforced concrete and are 2 ft 4 in. thick, minimum. The roofs of the pump houses are 1 ft thick. These thicknesses are adequate to withstand design-basis tornado-generated missiles. In addition, the two pump houses are redundant to each other. In the event that one pump house is inoperable, the other is capable of providing sufficient service water for safe shutdown. See Figures 3.5-46 and 3.5-47.

The ability of the spray ponds to tolerate the design-basis tornado-generated missiles is discussed in Section 3.3.2.3.

Figure 1.2-20 illustrates the pump houses and spray ponds.

e. Makeup water pump house

The exterior walls and roof of the makeup water pump house are of reinforced concrete and are sufficiently thick to withstand the effects of the design-basis tornado-generated missiles, as discussed in Section 3.3.2.3. The exterior walls, with the exception of the equipment access opening, are 2 ft 4 in. thick and the roof slab is 1 ft 4 in. thick. Figures 1.2-1, 1.2-23 and 1.2-24 furnish its location and arrangement. The 8 ft 0 in. by 10 ft 0 in. equipment access opening is protected by a 14-in.-thick concrete door.

f. Turbine building

Safety-related components in the turbine building are located in areas where tornado missile protection is provided by reinforced-concrete exterior walls, a minimum of 18 in. in thickness, and two reinforced-concrete slabs overhead, at el. 471 ft and 501 ft. Figures 1.2-2 through 1.2-6 illustrate the turbine building.

g. Safety-related systems outside of tornado-hardened structures

The protection provided to the pipe lines and electrical lines located underground between the spray ponds, SW pump houses, makeup water pump houses, reactor building, and the diesel generator building is described in Sections 3.5.2 and 3.5.3. The protection provided to the critical electrical trays in the corridor between the turbine generator building and the radwaste and control building is also described in Sections 3.5.2 and 3.5.3.

3.5.1.4.2 Tornado-Generated Internal Missiles

The tornado-generated internal missiles as mentioned in Section 3.5.1.4 are materials and/or items attached to or found inside a building, but subjected to the design basis tornado described in Section 3.3.2 as a result of a loss of a building exterior wall or roof. The materials and/or items considered as potential tornado-generated internal missiles are discussed below.

a. The reactor building steel framed superstructure uses girts and roof purlins fastened to the building frame by means of controlled release fasteners. The steel girts and purlins are considered to become free falling tornado-generated internal missiles which can strike the roof of the diesel generator building, the radwaste and control building, and main steam corridor slabs, in the event a tornado blows the roofing and/or siding off of the building frame. Structures housing safety-related systems, equipment, and components are designed to withstand the effects of these missiles.

b. In the event that a tornado blows the roof purlins, roof decking, girts, and siding panels off the reactor building frame, the reactor building crane is then exposed to the design basis tornado. The reactor building crane is designed with provisions which preclude it, or any part thereof, from becoming a missile (see Section 3.3.2.3).

3.5.1.4.3 Flood Generated Missiles

The design basis flood el. discussed in Section 3.4 and defined in Section 2.4, exceeds the flood levels associated with breaches of the Grand Coulee Dam. The final plant grade level is higher than the design basis flood. Therefore, flood-generated missiles are not considered in the design of the Seismic Category I safety-related structures and installations.

3.5.1.4.4 Protection and Design

Systems protected from missiles generated by natural phenomena, and barrier design are described in Sections 3.5.2 and 3.5.3 respectively.

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

Hazards due to missiles postulated in the design basis explosions or accidents at nearby industrial plants, military facilities, pipe lines, or storage facilities can be discounted as discussed in Section 2.2.

The Hydrogen Storage and Supply Facility (HSSF) contains a liquid hydrogen storage tank, ASME tubes (gaseous hydrogen), trailer tubes (gaseous hydrogen) and a hydrogen pipeline to the plant. An analysis shows that an explosion and subsequent missile generation from a

random tank rupture at normal pressure would not affect the plant due to the remote distance of the facility. Another analysis shows that an overpressurization event and subsequent rupture of the liquid hydrogen tank is a credible event. However, the total annual probability of impact for any of the missiles generated is less than 10^{-7} and does not meet the threshold for consideration as a design basis event. The hydrogen storage containers have relief valves to prevent overpressurization.

3.5.1.6 Aircraft Hazards

NUREG-0800, Standard Review Plan, Section 3.5.1.6, "Aircraft Hazards," provides guidance to ensure that the risks from aircraft hazards are low enough for nuclear power plants. If the distance at which aircraft activity occurs meets all the criteria provided in the following, the probability of an aircraft accident resulting in radiological consequences greater than the exposure limits in 10 CFR 50.67 can be considered to be less than about 10^{-7} per year and no additional analysis is required (Reference 3.5-16):

a. The plant-to-airport distance (D) is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or D is greater than 10 statute miles and the number of operations is less than 1000 D^2,

b. The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a use greater than 1000 flights per year or where activities (such as practice bombing) may create an unusual stress situation, and

c. The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

Additionally, aviation crashes associated with general aviation (light planes, < 12,500 lb) may be excluded from this analysis, as tornado design-basis requirements establish structural requirements sufficient to protect safety-related structures, systems, and components against these types of events (Reference 3.5-17).

3.5.1.6.1 Airports

There are no active airports located within 10 statute miles of the CGS site, but there are two commercial airports and six private airports which are active within 20 miles of the site. The annual number of operations of flights from these airports of less than 1000 D^2 is satisfied for all aviation uses at the commercial airports, including general, airtaxi, freight, and commercial traffic.

3.5.1.6.2 Military Airspace Use

The airspace over the Hanford site is periodically used as a marshaling area for military aircraft participating in training missions at the Yakima Training Center. A low-level military training route associated with this training passes above the western edge of the Hanford site about 18 miles west of CGS (Reference 3.5-18). Thus Criterion 2 associated with military use is also satisfied.

3.5.1.6.3 Federal Airways and Airport Approaches

CGS does not satisfy the third criterion identified in Section 3.5.1.6 of being at least 2 statute miles beyond the nearest federal airway, holding pattern, or approach pattern. Therefore, the probability of an aircraft accident resulting in an exposure greater than 10 CFR 50.67 limits has been evaluated.

Figure 2.2-3 shows the commercial airports, low-level federal airways, and airport instrument approaches in the vicinity of the CGS site. Airway V187 (with a minimum altitude of 3500 ft mean sea level) passes over the CGS site. The 14 nautical mile (NM) instrumented approach pattern is also important to the CGS site. Aircraft tend to maintain significant altitude over the Hanford site because of a request by the Federal Aviation Administration (FAA) (Reference 3.5-18).

The Tri-City Airport traffic supervisor provided a conservative estimate of 7500 flights per year that can be expected through airway V187 and the 14 NM approach (Reference 3.5-19). The number obtained is for flights by aircraft that are under instrument flight rule (IFR) flight plans. Aircraft that leave the airport in the general direction of V187 and that are under visual flight rules do not have to file a flight plan and are not required to follow the airway. For the purpose of this evaluation, the total number of aircraft flying through the airspace over CGS is assumed to be those constrained to the airway V187 and those using the 14 NM instrument-approach path (i.e., 7500 per year). Other flight paths are either sufficiently far away (more than 2 miles) with sufficiently low volume or are not currently active.

The aircraft using the Richland Airport may also pass over the CGS airspace. However, the only aircraft over 12,500 lb expected to use this approach would be ambulance (Life-Flight) aircraft using the Richland Airport under adverse weather conditions. Other aircraft activities possibly impacting the CGS site include activities conducted by/for the Department of Energy or their contractors on the Hanford site. These typically consist of less than 10 low-level flights per year over the entire Hanford site airspace (Reference 3.5-19). The additional risk posed by these operations is expected to be very low and is sufficiently bound by the conservatisms used in the above estimates.

The probability per year (P_{PY}) of an aircraft crashing into the plant can be expressed as (Reference 3.5-16) equation 3.5.1.6-1:

$$P_{PY} = C \times N \times A_{eff}/w$$

where:

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

N = Number of flights per year along the airway

w = Width of airway in miles

A_{eff} = Effective area of the plant in square miles.

The effective area of the facility is the ground surface area covered by all flight crash trajectories that could impact the surface structure. This area is larger than the facility itself because of the possibility of the aircraft skidding across the ground before hitting the facility as well as the possibility of a direct hit on the structure before striking the ground. The effective area is the sum of the structure's area, the shadow area (behind the structure), and the aircraft skid area (in front of the structure).

3.5.1.6.4 Summary

The probability per year, estimated per Reference 3.5-16, is 9.07×10^{-8}. Thus, the probability of an aircraft accident resulting in radiological consequences exceeding 10 CFR 50.67 limits is below 10^{-7}/year. Therefore, aircraft are discounted as credible missiles and accidents involving aircraft are not considered design basis events.

3.5.2 SYSTEMS TO BE PROTECTED

The structures, systems, and components necessary for bringing the plant to a safe shutdown and the protection provided for these structures, systems, and components from missiles is discussed in Sections 3.5.1 and 3.5.2.

Protection provided for the safety-related structures located outdoors against tornado-generated missiles is described in Section 3.5.1.4 and by turbine missiles in Section 3.5.1.3.

The plant structures, systems, equipment, and components that are required to bring the plant to a safe shutdown condition, or whose failure could lead to offsite radiological consequences under accident conditions, are protected from external (outdoor) missiles by barrier structures or redundant systems as follows:

a. The exposed exterior walls of safety-related structures are designed to protect internal structures, systems, and components. Structures with such exposed exterior walls are described in Section 3.8.4 and shown in figures referred to therein. Figure 3.5-35 illustrates exterior walls subject to tornado-generated missile impact.

b. All openings for heating, ventilation, and air conditioning system fresh air intakes (FAI) and exhausts (EXH), in buildings housing safety-related equipment, are protected against externally generated missiles as indicated in Table 3.5-6. Examples are the louvered openings above the floor at el. 572 ft 0 in. in the north and south walls of the reactor building. These openings are protected by a labyrinth of missile shield walls inside the opening.

c. The SW and the TMU pipelines and electrical lines between the SW pump houses, the TMU pump house, the reactor building, and the diesel generator building are located below grade and are protected from external missiles by sufficient Quality Class I earth cover of high relative density (described in Section 3.5.3). The TMU system is required for safe shutdown only when both spray ring headers are lost to tornado missiles (see Section 3.3.2).

d. The two SW pump houses, the tower makeup water pump house, and the valve boxes along the cooling tower makeup water lines are tornado-hardened (see Section 3.5.3).

e. As described in Sections 3.8.4.1.5 and 3.3.2.3, the spray ponds (except spray headers) are below grade and are protected from external missiles by sufficient Quality Class I earth backfill of high relative density. The spray headers are not required to be protected from tornado missiles.

f. The critical electrical trays in the corridor between the turbine generator building and the radwaste and control building are protected from possible missiles by the cantilever wall on column line 17 (Figure 3.5-48) and the additional 10 ft 0 in. high cantilever barrier wall immediately east of it. The 10 ft 0 in. high barrier wall provides missile protection for the door at el. 441 ft 0 in. on column line 17. Although the trays are above the door level and do not require missile protection provided by the 10 ft 0 in. high wall, the protection is provided, as shown in Figure 3.5-48.

g. The 18 in. CW piping used to provide service water return flow to the CW basin following a tornado has been analyzed to withstand the effects of all postulated tornado missiles, assuming that it is embedded in soil with the surface of the soil flush with the top of the pipe.

h. The tower makeup water (TMU) pipe lines and electrical lines between the TMU pump house and the SW pump houses are located below grade and are protected from external missiles by sufficient Quality Class I earth cover.

i. Portions of the following underground and aboveground piping associated with the diesel oil system are safety related. However, tornado missile protection is not provided to that piping. In the event of damage to the fill and vent connections due to tornado missiles, there are tank pump out connections and unused flanged connections on the storage tank which can be used as fill and vent openings.

1. Diesel oil day tank overflow and drain to storage tanks,
2. Diesel oil storage tank fill and vent lines, and
3. Filter-polisher unit recirculation piping.

3.5.3 BARRIER DESIGN

The barrier design objectives emphasize missile containment and structural integrity without secondary missile generation.

The overall response of barriers subject to impact are investigated by the use of general energy equations given in Reference 3.5-9. On determination of penetration depth and duration of impact, an effective dynamic force is computed. The additional calculation of the natural period of the target structure and the selection of a ductility ratio facilitates the determination of the required structural resistance. In this manner, missile impact is translated to an equivalent static load in an effort to quantify bending moments and shear. The detailed method used for predicting the overall response of missile barriers, including the forcing function method of determining ductility in structural elements and the basis for the ductility ratios used in the calculations, is provided in Appendix C of Reference 3.5-13 that was approved by the NRC.

3.5.3.1 Concrete Barriers

Concrete missile barriers are designed in accordance with penetration equations such as the modified Petry equation (Reference 3.5-2), the Ballistic Research Laboratory formula (Reference 3.5-1), or the modified NDRC formula (Reference 3.5-14). In all cases, except for barriers exposed to turbine missiles, a minimum concrete thickness of 2.2 times the penetration thickness determined for an infinitely thick slab is provided to prevent perforation, spalling, or scabbing. For discussion of turbine-generated missiles, see Section 3.5.1.3.

3.5.3.2 Steel Barriers

The Ballistic Research Laboratories Formula (Reference 3.5-1) is used to determine penetration depths of missiles into steel barriers.

3.5.3.3 Earth Barriers

When the protective barrier is of earthen origin, the soil penetration studies are based on alternate techniques. Buried safety-related piping and electrical systems required for a safe shutdown are ensured adequate protection from tornado-generated missiles. The original embedment depth of 5 ft was calculated to provide acceptable protection for the original design-basis tornado/missile parameters. The current design-basis tornado/missile parameters are now less severe and are used for any required evaluations for protection from postulated missiles. Based on the specific circumstances, the required embedment may be less than the original 5 ft.

3.5.3.4 Applications

Examples of barrier design are as follows:

Steel covers for manholes containing cabling for safety-related equipment required for safe shutdown are designed to withstand tornado-generated missile impact and associated wind pressure. These steel plates are designed using conventional elastic analysis and design methods for determining stress and strain. The design uses two plates of ASTM A 514 steel plate to prevent penetration and blowout.

The reactor building vehicle air lock (railroad bay) exterior doors and the SW pump house exterior equipment doors are designed and certified by the manufacturer to withstand the effects of tornado-generated exterior missiles as described in Section 3.5.1.4.

All other doors in Seismic Category I and safety-related structures are not designed to withstand the effects of the missiles described in Section 3.5.1. These doors are backed up, wherever missile protection is required, with reinforced-concrete walls forming a labyrinth behind the door. Similarly, louvers in exterior walls, which are vulnerable to missile penetration, are backed up by reinforced-concrete plenums or walls.

Based on the selection and description of missiles cited in Section 3.5.1, the interaction of missiles with structural elements is determined and the results are given in Section 3.5.1.4.1. The tabulations assume the missiles to impact at the most vulnerable point of a structure or component (e.g., at the center of a slab).

The reactor protection system motor generator sets flywheels located in the critical dc switchgear rooms at el. 467 ft 0 in. in the radwaste building were analyzed and determined to be credible missile sources, with the potential consequences affecting the safe shutdown of the

plant. Barriers were constructed around these flywheels of steel and aluminum honeycomb material, which were designed to contain the credible missiles (see Section 3.5.1.1.5).

The SW piping between the SW pump houses, the reactor building, and the diesel generator building is provided with sufficient cover for protection from tornado-generated missiles (see Section 3.5.3.3). The SW piping exits the pump houses at a centerline el. of 435 ft 3 in. and immediately turns down at a 45 degree angle to el. 432 ft, where the piping is routed to the reactor building in high relative density Quality Class I backfill. Grade level is at 440 ft 6 in., providing an embedment depth of over 7 ft from the top of the pipe. Where the pipe exits the pump houses, a 1.5-in. asphaltic concrete road with a 6-in. base coarse and 2-in. leveling coarse bed provides additional protection from tornado-generated missiles. Additionally, the two SW loops are separated by at least 20 ft to preclude loss of redundancy.

The tower makeup water piping to the river is embedded in sufficient Quality Class I backfill for protection of tornado-generated missiles (see Section 3.5.3.3).

The control room remote air intake structure piping is embedded a minimum of 5 ft for protection from tornado-generated missiles. The remote air intake structures are also missile-hardened.

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3.5-28 WESS-90-008, June 26, 1990. Westinghouse Letter with FI Rotor Missile Analysis Information with NRC's SER response.

3.5-29 Westinghouse Letter, May 4, 1991, "Requested Information on Turbine Missile Probabilities" with Fully Integral Rotor Overspeed Analysis Summary.

3.5-30 WCAP-11525, June 1987. Westinghouse "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency."

3.5-31 WCAP-14732, Revision 1, June 1997. Westinghouse "Probabilistic Analysis of Reduction in Turbine Valve Test Frequency for Nuclear Plants with Westinghouse BB-296 Turbines with Steam Chests."

3.5-32 ME-02-06-16, Revision 1, December 2007. DEH/Pressure Control Upgrade Frequency.

Table 3.5-1

Systems Description Outside Containment

Systems Available for a Safe Shutdown	Function	Section	Figures	Seismic/Quality Class
RCIC	Maintain RPV water inventory	5.4.6, 7.4.1.1	3.5-9 through 3.5-14	I/I
HPCS	Maintain RPV water inventory	6.3, 7.3.1.1.1.1	3.5-9 through 3.5-14	I/I
SW	Heat rejection	7.3.1.1.6	3.5-9 through 3.5-14	I/I
RHR A B C	Maintain water inventory and decay heat removal	5.2, 7.3.1.1.1.4, 6.3, 5.4.7	3.5-1 through 3.5-8	I/I
CRD	Reactivity control	7.7.1.2	3.5-15	I/I
RFW	Maintain RPV water inventory	5.4.9	3.5-9 through 3.5-14	--
LPCS	Maintain RPV water inventory	6.3, 7.3.1.1.1.3	3.5-1 through 3.5-8	I/I

Note: Identification of missiles to be protected against, their sources, and bases for selection are discussed in Section 3.5.1.1.3. The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles is discussed in Section 3.5.1.1.2.

Table 3.5-2

Internally Generated Missiles Outside Containment

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-7	RRA-FN-6	RCIC pump room fan coil fan	I/I	(a)
F-9	RRA-FN-2	MS fan coil fan	I/I	(a)
F-11	RRA-FN-7	RR lock fan coil fan	1M/II+	(b)
F-33	REA-FN-2A	Exhaust fan	1M/II+	(b)
F-35	REA-FN-2B	Exhaust fan	1M/II+	(b)
F-37	REA-FN-1A	Exhaust fan	1M/II+	(b)
F-39	REA-FN-1B	Exhaust fan	1M/II+	(b)
F-41	ROA-FN-1A	Supply fan	1M/II+	(b)
F-43	ROA-FN-1B	Supply fan	1M/II+	(b)
F-57	RRA-FN-3	RHR pump room fan coil fan	I/I	(b)
F-59	REA-FN-15	Exhaust fan	1M/II+	(b)
F-61	RRA-FC-19	FPC heat exchanger and pump room cooler	I/I	(b)
F-62	RRA-FC-20	FPC heat exchanger and pump room cooler	I/I	(b)
P-1	LPCS-P-2	LPCS water leg pump	I/I	(d)
P-3	HPCS-P-3	HPCS water leg pump	I/I	(d)
P-5	CRD-P-1A	CRD pump C12-C001A	1M/II+	(d)
P-6	CRD-P-1B	CRD pump C12-C001B	1M/II+	(d)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
P-7	RHR-P-3	RHR water leg pump	I/I	(d)
P-9	COND-P-3	Reactor building condensate supply pump	1M/II+	(d)
P-10	RCIC-P-4	RCIC condensate pump	I/I	(d)
P-11	RCIC-P-2	RCIC vacuum pump	I/I	(d)
P-12	RCIC-DT-1	RCIC turbine drive	I/I	(d)
P-13	RCIC-P-1	RCIC pump	I/I	(d)
P-14	COND-P-4	Radwaste building condensate supply pump	1M/II+	(c)
P-15	COND-P-5	Condensate filter demineralizer backwash pump	1M/II+	(c)
P-16	RCIC-P-3	RCIC water leg pump	I/I	(d)
P-17	FPC-P-3	Suppression pool cleanup pump	1M/II+	(b)
P-18	RHR-P-2A	RHR pump	I/I	(b)
P-19	RHR-P-2B	RHR pump	I/I	(b)
P-20	RWCU-P-1A	Cleanup circulation pump	1M/II+	(b)
P-21	RWCU-P-1B	Cleanup circulation pump	1M/II+	(b)
P-24	RCC-P-1A	RBCC water pump	1M/II+	(b)
P-25	RCC-P-1B	RBCC water pump	1M/II+	(b)
P-26	RCC-P-1C	RBCC water pump	1M/II+	(b)
P-27	RCC-P-2	RBCC chemical metering pump	1M/II+	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
P-28	FPC-P-1A	Fuel pool cooling pump	I/(II/I)	(b) (g)
P-29	FPC-P-1B	Fuel pool cooling pump	I/(II/I)	(b) (g)
P-30	PWC-P-4A	Potable water pump	1M/II+	(b)
P-31	PWC-P-4B	Potable water pump	1M/II+	(b)
P-32	EDR-P-5A EDR-P-5B	Reactor building equipment drain sump pumps	1M/II+	(b)
P-33	FDR-P-1A	Reactor building floor drain sump pump	1M/II+	(b)
P-34	FDR-P-1B	Reactor building floor drain sump pump	1M/II+	(b)
P-35	FDR-P-2	Reactor building floor drain sump pump	1M/II+	(b)
P-36	FDR-P-3	Reactor building floor drain sump pump	1M/II+	(b)
P-37	FDR-P-4A	Reactor building drywell floor sump pump	1M/II+	(b)
P-38	FDR-P-4B	Reactor building drywell floor sump pump	1M/II+	(b)
P-39	ROA-P-1A	Air washer pump	1M/II+	(b)
P-40	ROA-P-1B	Air washer pump	1M/II+	(b)
R-1	RPS-MG-1	RPS motor generator set	II/II	(e)
R-2	RPS-MG-2	RPS motor generator set	II/II	(e)
F-71	RRA-FN-8	MS tunnel fan coil fan	I/I	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-72	RRA-FN-9	MS tunnel fan coil fan	I/I	(b)
R-5	DSA-C-1C	Starting air compressor HPCS diesel generator	I/I	(b)
R-6	DSA-C-2C	Starting air compressor HPCS diesel generator	I/I	(b)
R-7	DSA-C-1A1	Starting air compressor diesel generator A1	I/I	(f)
R-8	DSA-C-1A2	Starting air compressor diesel generator A2	I/I	(f)
R-9	DSA-C-1B1	Starting air compressor diesel generator B1	I/I	(f)
R-10	DSA-C-1B2	Starting air compressor diesel generator B2	I/I	(f)
R-11	DLO-P-3A1	Motor driven lube oil pump diesel generator A1	I/I	(b)
R-12	DLO-P-3A2	Motor driven lube oil pump diesel generator A2	I/I	(b)
R-13	DLO-P-3B1	Motor driven lube oil pump diesel generator B1	I/I	(b)
R-14	DLO-P-3B2	Motor driven lube oil pump diesel generator B2	I/I	(b)
R-15	DO-P-3A1	Motor driven lube oil pump	I/I	(b)
R-16	DO-P-3A2	Motor driven lube oil pump	I/I	(b)
R-17	DO-P-3B1	Motor driven lube oil pump	I/I	(b)
R-18	DO-P-3B2	Motor driven lube oil pump	I/I	(b)
R-19	DO-P-6	dc motor driven HPCS fuel pump	I/I	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-73	DEA-FN-13	Exhaust fan loop A pump room	1M/II+	(b)
F-74	DEA-FN-23	Exhaust fan loop B pump room	1M/II+	(b)
F-75	DEA-FN-33	Exhaust fan HPCS oil pump room	1M/II+	(b)
F-76	DEA-FN-32	Exhaust fan day tank room	1M/II+	(b)
F-77	DEA-FN-12	Exhaust fan day tank room	1M/II+	(b)
F-78	DEA-FN-22	Exhaust fan day tank room	1M/II+	(b)
F-79	DRA-FN-34	Fire deluge equipment room recirculation fan	1M/II+	(b)
F-80	DRA-EUH-11	Electric unit heater loop A oil pump room	1M/II+	(b)
F-81	DRA-EUH-21	Electric unit heater loop B oil pump room	1M/II+	(b)
F-82	DRA-EUH-31	Electric unit heater HPCS oil pump room	1M/II+	(b)
F-83	DRA-EUH-32	Electric unit heater fire deluge equipment room	1M/II+	(b)
F-84	DRA-EUH-33	Electric unit heater fire deluge equipment room	1M/II+	(b)
F-85	DMA-AH-32	Air handling HPCS diesel generator room	I/I	(b)
F-86	DMA-AH-12	Air handling Division 1 diesel generator room	I/I	(b)
F-87	DMA-AH-22	Air handling Division 2 diesel generator room	I/I	(b)
F-88	DEA-FN-11	Exhaust fan Division 1 diesel generator room	I/I	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-89	DEA-FN-21	Exhaust fan Division 2 diesel generator room	I/I	(b)
F-90	DEA-FN-31	Exhaust fan HPCS diesel generator room	I/I	(b)

RESOLUTION CODE NOTES:

[a] These postulated missiles originate from air conditioning fan coil units. These fan coil units are contained in housings capable of containing any potential fan missile. The air outlets have grating installed to prevent potential missiles from exiting via this route. The air inlets are into the fan impeller eye and thus the fan inlets are not a missile exit path. Missiles originating from this equipment cannot get beyond the protective housing.

[b] These postulated missiles were determined to be credible. A safe shutdown analysis was performed, which determined that the failure of all equipment in the missile path envelope would have no effect on the ability of the plant to safely shut down in the event of an accident.

[c] These postulated missiles are impeller fragments originating from the radwaste building condensate supply pump and the condensate filter demineralizer backwater pump. These missiles were assumed to be credible. A safe shutdown analysis was performed, which determined that the failure of all equipment in the missile path would have no effect on the ability of the plant to shut down in the event of an accident, with the exception of impacts on one 20-in. standby service water pipe with standard (3/8-in.) wall thickness. A worst case missile, which would transmit the maximum energy to a point of contact with the nearest standby service water pipe was analyzed.

This worst-case missile and impact could not penetrate the standby service water pipes. The target pipe would sustain acceptable deformation in which the pressure integrity of the target pipe is not affected.

This analysis demonstrated that these pumps could not generate a missile capable of compromising the safe shutdown functions of the service water pipes.

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Resolution Code Notes (Continued):

[d] These postulated missiles are pump impeller and turbine blade fragments. Analysis has shown that the impeller and turbine casings have sufficient thickness to contain each postulated missile. Postulated missiles originating from this equipment cannot be ejected beyond the protective casings.

[e] These postulated missiles are flywheel fragments from the RPS motor-generator sets. A safe shutdown analysis was performed which showed critical Division 1 and 2 electrical power and control cables were in the potential missile path that were required to safely shut down the plant in the event of an accident. Missile barriers were designed and installed around both motor-generator flywheels. These barriers ensure that potential missiles originating from the motor generators will have no effect on the ability of the plant to shut down in the event of an accident.

[f] These postulated missiles originate from the air compressors that provide starting air for the diesel generator sets. These four compressors are of the reciprocating three-cylinder piston type. The three cylinder heads cover the upper portion of the crankshaft and preclude any missile exiting the compressor from the upper half of the crankcase. Credible missiles were postulated to exit out the bottom half of the crankcase. A safe shutdown analysis was performed, which determined that the failure of all equipment in the missile path envelope would have no effect on the ability of the plant to safely shut down in the event of an accident.

[g] Pumps purchased or installed prior to Jan 1, 1980 shall be Quality Class II. Pumps purchased or installed after Jan 1, 1980 shall be Quality Class 1.

Table 3.5-3

Plant Systems Protected By Missile Barriers

The missile barrier for RPS motor-generator set 1, located in the Division 1 essential switchgear room in the radwaste building, el. 467 ft 0 in., protects electric power and control cables servicing the following equipment:

Power distribution panel DP-S1-1A

Power panel PP-7A-F

Inverter IN-3

Motor control center MC-S1-1D

Motor control center MC-7A

Power panel PP-7A

24 V dc power supply system DP-SO-A

The missile barrier for RPS motor-generator set 2, located in the Division 2 essential switchgear room in the radwaste building, el. 467 ft 0 in., protects electric power and control cables servicing the following equipment:

24 V dc power supply system PP-DP-SO-8

125 V dc power distribution panel DP-S1-2D

125 V dc power distribution panel DP-S1-2E

125 V dc power distribution panel DP-S1-2A

125 V dc power distribution panel MC-S1-2D

Motor control center MC-8A

Motor control center MC-8F

Instrument and control power panel PP-8A-F

Instrument and control power panel PP-8A-E

Motor control center MC-8B

Table 3.5-4

Systems Description Inside Containment

Systems Available for a Safe Shutdown	Function	Section	Figures	Seismic/Quality Class
RPS	Reactor protection	7.2.1.1, 4.6	3.5-29	I/I
CRD	Reactivity control	7.7.1.2	*3.5-19* through *3.5-22*, *3.5-25*, *3.5-27*, *3.5-28*	I/I
Inboard isolation valves	Containment isolation	5.4.5, 6.2.4	--	I/I
HPCS	Maintain RPV water inventory	7.3.1.1.1.1, 6.3	*3.5-16* through *3.5-18*, *3.5-20*, *3.5-25*, *3.5-27*, 3.5-29, 3.5-31	I/I
LPCS	Maintain RPV water inventory	7.3.1.1.1.3, 6.3	*3.5-16* through *3.5-19*, *3.5-26*, *3.5-28*, 3.5-29, 3.5-31	I/I
ADS	Depressurize RPV	5.2, 7.3.1.1.1.2	*3.5-17*, 3.5-29	I/I
RHR A B C	Maintain RPV water inventory and decay, heat removal	6.3, 5.4.7, 7.3.1.1.1.4	*3.5-16* through *3.5-18*, *3.5-20* *3.5-26*	I/I
Containment	Containment integrity	3.8.2	*3.5-1* through 3.5-50, 3.8-49	I/I

Note: Identification of missiles to be protected against, their sources, and bases for selection are discussed in Section 3.5.1.2.3. The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles is discussed in Section 3.5.1.2.2.

Table 3.5-6

Location of and Missile Protection
Provided for Fresh Air Intakes (FAI) and Exhausts (EXH)

Building Location	Exterior Walls: Opening Type	Exterior Walls: Size	Opening Centerline Location: Elevation	Opening Centerline Location: Plan	Figure	Protection
Reactor						
South	FAI	6 ft 2-7/8 in. W x 17 ft 8 in. H	581 ft 8 in.	5 ft 8 in. west of col. line 6	3.5-36 3.5-37	Shielding wall
North	EXH	72 in. diameter	595 ft 8 in.	6 ft 0 in. west of col. line 6	3.5-36 3.5-38	Shielding wall
Radwaste and control						
West	FAI	4 ft 0 in. x 4 ft 0 in.	526 ft 6 in.	4 ft 6 in. north of col. line K.1	3.5-45 3.5-49 3.5-50	Shielding wall, slab and hood
Roof	EXH	3 ft 0 in. x 3 ft 0 in.	546 ft 0 in.	5 ft 0 in. north of col. line K.1 4 ft 0 in. east of col. line 15.1	3.5-39 3.5-40	Concrete shielding hood
Diesel Generator						
South	FAI	10 ft 11 in. W x 12 ft 8 in. H	449 ft 4 in.	5 ft 7.5 in. east of col. line 9.4	3.5-41 3.5-42	Shielding wall
South	FAI	10 ft 11 in. W x 12 ft 8 in. H	449 ft 4 in.	7 ft 5.5 in. east of col. line 7.4	3.5-41 3.5-42	Shielding wall
South	FAI	10 ft 11 in. W x 12 ft 8 in. H	449 ft 4 in.	6 ft 5.5 in. west of col. line 3.8	3.5-41 3.5-42	Shielding wall
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. east of col. line 8.4	3.5-43 3.5-42	Penthouse
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. west of col. line 7.4	3.5-42 3.5-43	Penthouse

Table 3.5-6

Location of and Missile Protection
Provided for Fresh Air Intakes (FAI)
and Exhausts (EXH) (Continued)

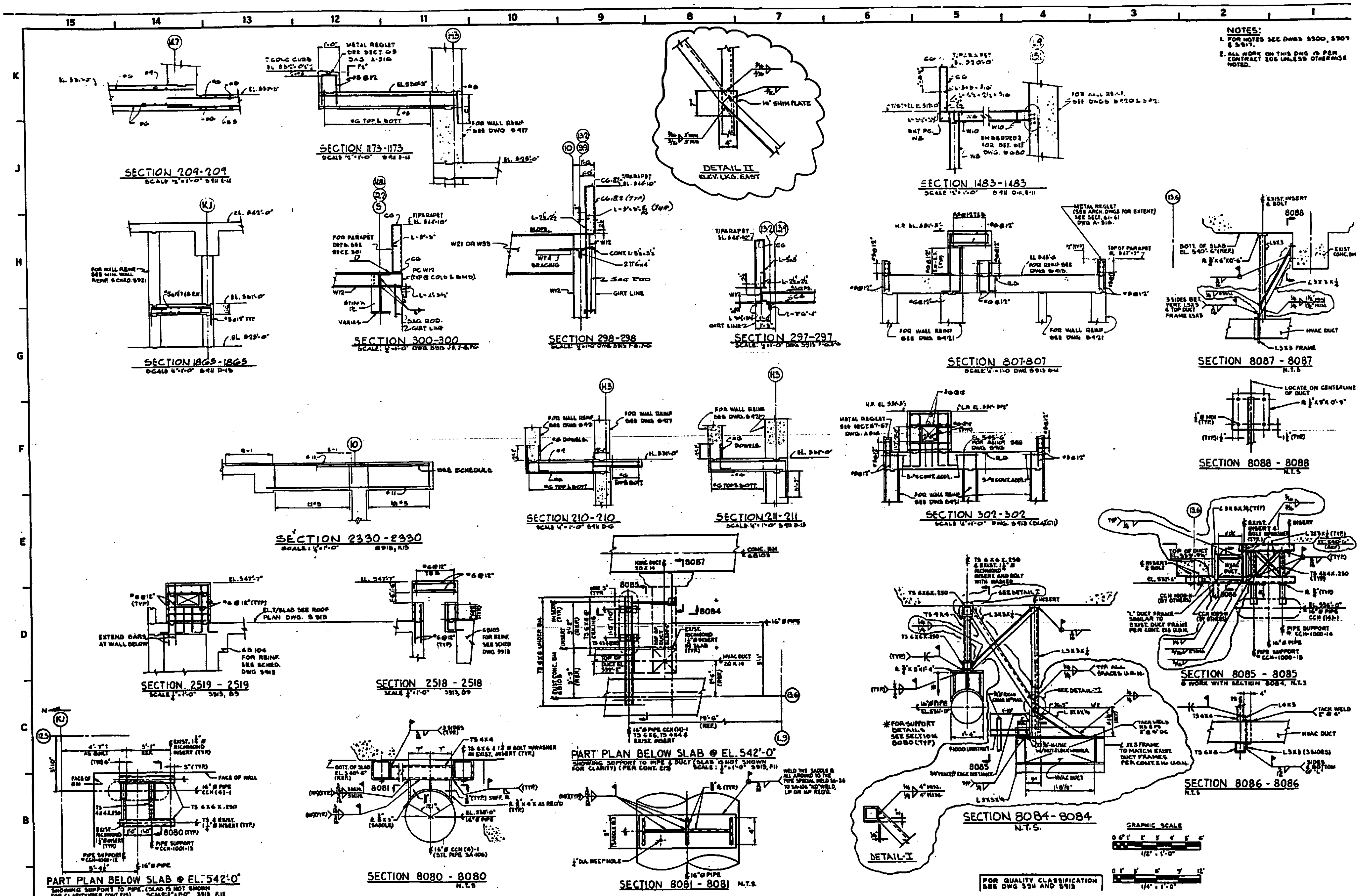
Building Location	Exterior Walls: Opening Type	Exterior Walls: Size	Opening Centerline Location: Elevation	Opening Centerline Location: Plan	Figure	Protection
Diesel Generator (Continued)						
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. east of col. line 6.6	3.5-42 3.5-43	Penthouse
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. west of col. line 5.5	3.5-42 3.5-43	Penthouse
South	EXH	36 in. diameter	480 ft 2.5 in.	2 ft 9 in. east of col. line 5.5	3.5-42 3.5-43	Penthouse
East	EXH	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	12 ft 8 in. south of col. line P.1	3.5-44 3.5-42	Shielding beam MB5
East	FAI	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	3 ft 2 in. north of col. line Q	3.5-44 3.5-42	Shielding beam MB5
East	EXH	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	12 ft 10 in. south of col. line Q	3.5-44 3.5-42	Shielding beam MB5
East	FAI	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	8 ft 2 in. north of col. line R	3.5-44 3.5-42	Shielding beam MB5
East	EXH	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	7 ft 10 in. south of col. line R	3.5-44 3.5-42	Shielding beam MB5
East	FAI	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	10 in. north of col. line P.1	3.5-44 3.5-42	Shielding beam MB5

Table 3.5-6

Location of and Missile Protection
Provided for Fresh Air Intakes (FAI)
and Exhausts (EXH) (Continued)

Building Location	Exterior Walls Opening Type	Size	Opening Centerline Location Elevation	Plan	Figure	Protection
SSWPH[a] 1A						
East	FAI	2 ft 6 in. x 2 ft 6 in.	437 ft 6 in.	3 ft 3 in. south of col. line A	3.5-46	Wall and floor
West	EXH	3 ft 5 in. x 3 ft 5 in.	451 ft 11 in.	3 ft 4.5 in. north of col. line E	3.5-46 3.5-47	Wall
SSWPH[a] 1B						
South	FAI	2 ft 6 in. x 2 ft 6 in.	437 ft 6 in.	3 ft 3 in. west of col. line A	3.5-46	Wall and floor
North	EXH	3 ft 5 in. x 3 ft 5 in.	451 ft 11 in.	3 ft 4.5 in. east of col. line E	3.5-46 3.5-47	Wall
FAIS[b]						
Roof	FAI	3 ft 3 in. x 3 ft 5 in.	441 ft 9 in.	Centerline of structure	3.8-49	Slabs, walls

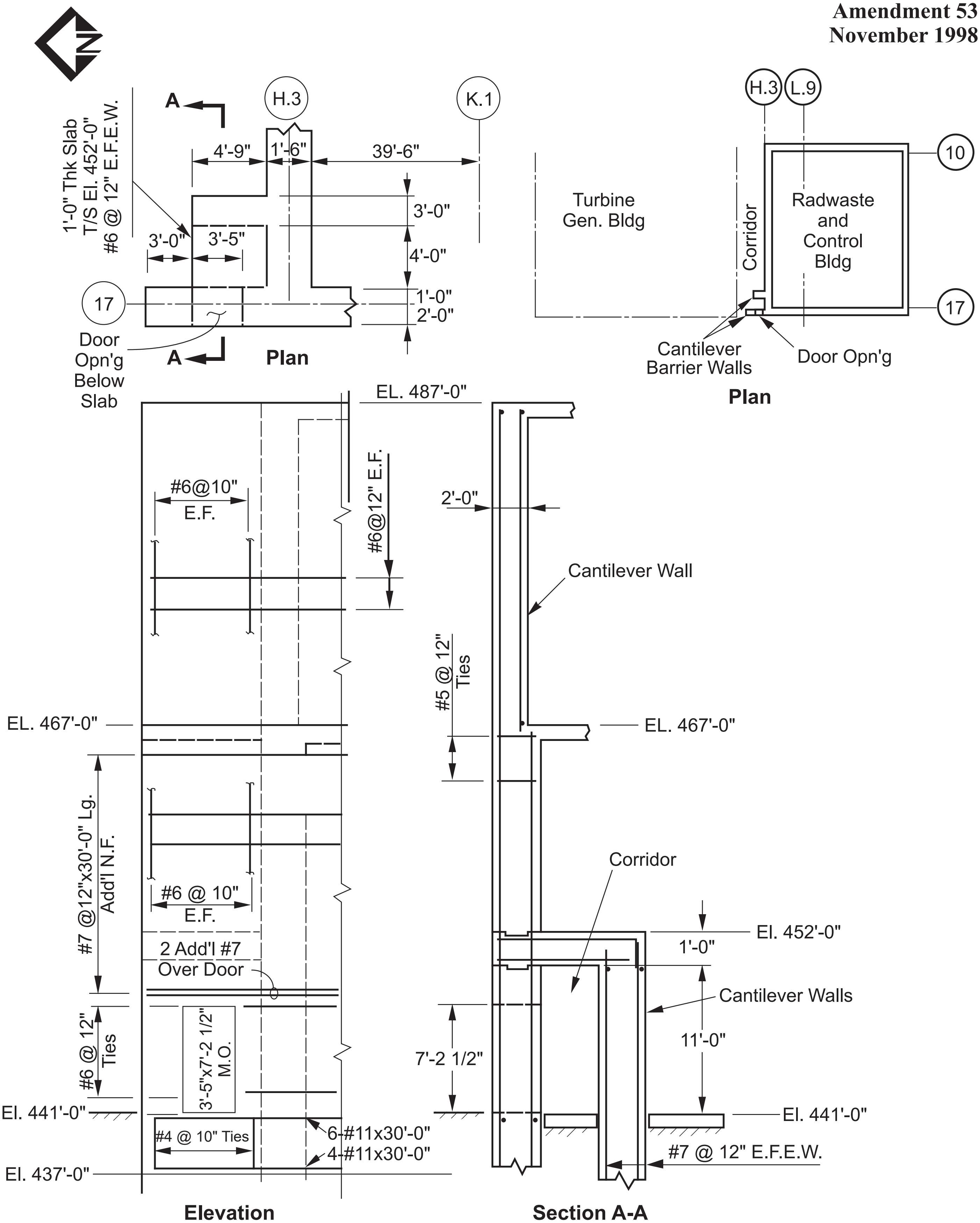
[a] Standby service water pump house
[b] Fresh air intake structures



Columbia Generating Station
Final Safety Analysis Report

Radwaste and Control Building

Draw. No.	S912	Rev.	8	Figure	3.5-40



Columbia Generating Station Final Safety Analysis Report	Cantilever Barrier Structure	
Draw. No. 910402.24	Rev.	Figure 3.5-48

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

This section describes the design bases and design measures that are implemented for Columbia Generating Station to ensure that the primary containment vessel and all essential equipment inside and outside primary containment, including components of the reactor coolant pressure boundary (RCPB), have been adequately protected against the effects of blowdown jet and reactive forces, and pipe whip resulting from postulated rupture of piping located either inside or outside of primary containment. Design measures have been implemented to ensure that any such postulated accident does not result in loss of required functions that are necessary to mitigate the consequences of that particular accident and place the reactor in a cold shutdown condition. The implementation of design measures considered a single, random active component failure.

The implementation of criteria for protection of safety-related systems from the effects of pipe rupture as discussed in this section is based not only on analytic evaluations but also on the walkdown of plant systems, equipment, and components which were performed. Verification of the protection of safety-related systems, equipment, and components from the effects of pipe rupture was provided by means of this final walkdown prior to fuel load.

The power uprate and Maximum Extended Load Line Limit Analysis (MELLLA) projects resulted in changes to the input parameters associated with the pipe break analysis. The evaluation of these changes (see Reference 3.6-10 and 3.6-24) supports the conclusion that the analysis and design presented in this section bounds the power uprate and MELLLA conditions.

3.6.1 PLANT DESIGN FOR PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE PRIMARY CONTAINMENT

The analysis of piping systems is performed in accordance with Branch Technical Position (BTP) ASB 3-1, Rev. 1, 1981. Systems or components important to plant safety or shutdown, which are located approximate to high- or moderate-energy piping systems and are susceptible to the consequences of failures of these piping systems, were considered during the evaluations described below. The identification is related to the predetermined piping failure locations as described in Section 3.6.2.

A listing of systems containing high- and moderate-energy lines is provided in Tables 3.6-1 and 3.6-2, and the physical arrangements of piping systems are shown in Figures 3.6-1 through 3.6-20B.

The continued habitability of the control room following a postulated fluid system piping failure is addressed in Section 3.6.1.12.

The results of failure mode and effect analyses are provided in accordance with BTP ASB 3-1 to verify that the consequences of failures of high- and moderate-energy lines do not affect the ability to safely shut the plant down. The analyses consider a single active component failure occurring in a required system concurrently with the postulated event. The results of these analyses are presented in Sections 3.6.1.11 and 3.6.1.18.

3.6.1.1 Energy Classification of Fluid Piping Systems Outside Containment

3.6.1.1.1 High and Moderate Energy Criteria

Definitions of high- and moderate-energy fluid piping systems are given in Section 3.6.2.1. Fluid systems which could be classified as high energy during operation, but are required to function only to mitigate the consequences of a loss-of-coolant accident (LOCA), are considered moderate energy as ruptures of these systems during operation would imply a second passive component failure.

Specifically, the low-pressure coolant injection (LPCI) modes of the residual heat removal (RHR) system, the low-pressure core spray (LPCS) system, the high-pressure core spray (HPCS) system, the suppression pool cooling modes of the RHR system and the standby liquid control (SLC) system are considered moderate energy.

Fluid piping systems which are classified as high energy by Section 3.6.2.1, whose temperature does not exceed 200°F, and whose pressure is derived solely by a centrifugal pump instead of a fluid reservoir, are considered moderate energy systems. In these cases the system is depressurized as a subcooled liquid after the break and the pump energy is simply converted to velocity head. The jet force induced by the pump at the break plane is insufficient to cause pipe whip and thus these cases are properly considered as moderate energy systems.

According to the criteria established in Section 3.6.2.1, the shutdown cooling mode of the RHR and the normally unpressurized portions of the reactor core isolation cooling (RCIC) system, are considered moderate energy.

3.6.1.1.2 Systems Subject to Analysis

According to the criteria established in Section 3.6.1.1.1, the high-energy fluid piping systems subject to analysis are listed in Table 3.6-1. The moderate-energy fluid piping systems subject to analysis are listed in Table 3.6-2.

3.6.1.2 Criteria for Establishing the Postulated Design Basis Break Locations

The criteria that define the postulated design basis break locations in high- and moderate-energy fluid piping systems are presented in Sections 3.6.2.1.1 through 3.6.2.1.3.

3.6.1.3 Criteria for Establishing the Postulated Design Basis Break Orientations

The criteria that define the postulated design basis break orientations in high- and moderate-energy fluid piping systems are presented in Section 3.6.2.1.4.

3.6.1.4 Summary of Dynamic Analysis of Category I Piping and Supports

3.6.1.4.1 Design Basis Breaks for the Dynamic Analysis Outside Primary Containment

Table 3.6-3 lists the number of design basis breaks on which the dynamic analysis is based as well as the particular piping system involved, the diameter of the pipe, the plan figure showing the piping system, the maximum blowdown thrust or the thrust versus time figure, and the room or area containing the postulated break.

3.6.1.4.2 Diagrams of Mathematical Models Used in the Dynamic Analysis

3.6.1.4.2.1 Models Used for Pipe Whip Restraint Design. A description of the mathematical models used to design the pipe whip restraints is presented in Section 3.6.2.2.2.

3.6.1.4.2.2 Models Used for Structural Analysis. Descriptions of the mathematical models and procedures used to ensure adequacy of Seismic Category I structures are presented in Sections 3.6.1.6 through 3.6.1.10.

3.6.1.4.2.3 Models Used to Represent Jet Stream Dynamics. Descriptions of the procedure used to model the dynamics of a jet stream caused by a postulated rupture of a high-energy fluid piping system are presented in Section 3.6.2.3.1.

3.6.1.4.3 Effect of Postulated Fluid Piping System Ruptures on Structures, Systems, or Components Necessary for Safe Reactor Operation

A discussion is presented in Section 3.6.1.11 concerning the effects of postulated high- and moderate-energy fluid piping system failures on structures, systems, and components necessary to shut down and maintain the reactor in a cold shutdown condition.

3.6.1.5 Methods to Protect Structures, Systems, or Components Necessary for Safe Reactor Operation From the Dynamic Effects of Postulated Fluid Piping System Ruptures

3.6.1.5.1 Pipe Whip Restraints

A description of the method used to design pipe whip restraints is presented in Section 3.6.2.3.3.

3.6.1.5.2 Protective Provisions for Structures, Systems, and Components Important to Safety

Separation of redundant features is the basic design criteria utilized to protect structures, systems, or components important to safety from the dynamic effects of postulated pipe ruptures including pipe whip impact and blowdown jet impingement.

The emergency core cooling pumps and the RCIC pump are located in individual rooms. The CRD/Condensate pumps are also located in an individual room, but the equipment drain sump located in that room is connected to the RCIC pump room through an unisolable pipe; other rooms that are connected to a common sump are provided with a connecting line isolation valve. In the event of a flood in any pump room, the walls of each room, including penetrations and doors, allow only minimal leakage between rooms and can withstand the dynamic effects of any postulated fluid piping system rupture in their vicinity as well.

The remaining mechanical and electrical equipment, which must function to mitigate the consequences of a postulated fluid piping system rupture outside primary containment, is protected by providing adequate separation of redundant features per the requirements of BTP ASB 3-1, Rev. 1, 1981.

A more detailed discussion of the analysis employed to demonstrate that structures, systems, and components important to safety are adequately protected is provided in Section 3.6.1.11.

3.6.1.5.3 Physical Separation of Systems or Components Important to Safety

Physical separation of systems or components important to safety is provided by maintaining sufficient distance between redundant features, by enclosing high-energy fluid piping systems and systems or components important to safety within protective structures.

3.6.1.5.4 Description, Number, and Location of Pipe Whip Restraints Outside Primary Containment

Typical pipe whip restraints are seen in Figures 3.6-21 and 3.6-22.

A total of 12 restraints are utilized for the main steam and feedwater lines in the main steam tunnel outside of primary containment. These restraints are illustrated in Figure 3.6-24.

3.6.1.6 Procedures to Evaluate the Structural Adequacy of Seismic Category I Structures Under Pipe Break Effects Outside Containment

3.6.1.6.1 General Approach

Structures and structural components important to safety are designed with sufficient strength to resist the effects of postulated pipe breaks in high energy fluid piping systems, such as pipe whip and jet impingement. High-energy fluid piping systems are defined in Section 3.6.1.1. Section 3.6.1.6 discusses the effects of postulated pipe breaks on structures and structural components. Environmental effects of postulated pipe breaks are addressed in Sections 3.6.1.13, 3.6.1.15, and 3.11.

The main component effects of a postulated pipe break include the following:

a. Pipe whip with its impacting energy,
b. Jet impingement and accompanying jet reaction, and
c. Pressurization and temperature effects which accompany pipe break.

Pipe whip effects from circumferential breaks are seen in Figure 3.6-25. Jet impingement effects from circumferential breaks and longitudinal splits are seen in Figure 3.6-26. Circumferential breaks and longitudinal splits are defined in Section 3.6.2.1.4.

In making a structural evaluation of the effects of pipe break accidents, the loads resulting from these pipe break accidents are used in combination with other prevailing loads that occur at the time of the break. For load information and combinations see Sections 3.6.1.6.5 and 3.6.1.6.6.

To make a structural evaluation of the effects of a postulated pipe break, the local damage to the structural element is predicted and the overall structural response is assessed. Local damage is the damage done to a structural element in the immediate vicinity of the pipe whip impact or the jet impingement. Overall structural response concerns the overall response of the entire structural element to the effects of a postulated pipe break. In the following discussion, whipping pipe is described as missiles.

3.6.1.6.2 Local Damage Prediction

Local damage prediction due to whip or jet impingement in the immediate vicinity of the impacted area includes estimation of the depth of penetration and whether, in the case of concrete targets, secondary missiles might be generated by spalling. In general, a whipping pipe is a blunt missile and penetration and spalling are not appreciable for the structural component (e.g., walls) thickness of interest. Such a condition is seen in Figure 3.6-25. Jet impingement local damage is not considered significant because the fluid mass does not have the mass concentration of a solid and because of the divergence of a jet which spreads the load

over a wide area (see Figure 3.6-26). Missile penetration is predicted for reinforced-concrete targets and for steel targets.

3.6.1.6.2.1 Reinforced Concrete Targets.

Penetration

The depth to which a rigid missile penetrates a reinforced-concrete target of infinite thickness is estimated by the following Modified Petry Formula (References 3.6-15 and 3.6-17):

$$X = K_p A_p \quad \log_{10}\left(1 + \frac{V_s^{\,2}}{215{,}000}\right) \tag{3.6-1}$$

where

X = depth of missile penetration into concrete element of infinite thickness (ft)

K_p = penetration coefficient for reinforced concrete (4.76 x 10^{-3} ft^3 /lb. for normal reinforced concrete with a crushing strength of 3200 psi and 1.4% of reinforcement. See Reference 3.6-15.)

$$Ap = \frac{W}{A} = \frac{\text{Missile Weight (lb)}}{\text{Projected Frontal Missile Area } (ft^2)}$$

V_s = striking velocity of missile (ft/sec)

When the element has a finite thickness, the depth of penetration is (References 3.6-15 and 3.6-17):

$$X_1 = \left[1 + e^{-4\frac{(t-2)}{x}}\right] x, (t = 2x) \tag{3.6-2}$$

where

X_1 = depth of penetration of missile into a concrete element of finite thickness (ft)

e = base of Napierian Logarithms

t = thickness of concrete element (ft)

Perforation

The thickness of a concrete element that will be just perforated by a missile is given as (References 3.6-15 and 3.6-17):

$$T = 2X \text{ (ft)} \tag{3.6-3}$$

Spalling

Spalling of concrete from the side opposite the impact surface of the structural element may occur even if the missile does not perforate the element. The estimate of the thickness that will just start spalling is given as (Reference 3.6-20):

$$T_s = 2.2X \text{ (ft)} \tag{3.6-4}$$

3.6.1.6.2.2 Steel Targets. The Ballistic Research Laboratories formula is used to determine perforation of a steel target. The thickness, T, of a steel target that will be just perforated by a missile is given as (Reference 3.6-17):

$$T^{3/2} = \frac{0.5MV^2}{17{,}400\ K^2 D^{3/2}} \tag{3.6-5}$$

where

T = steel wall thickness to just perforate (in.)

M = mass of the missile (weight/g in lb-sec^2/ft)

V = velocity of missile (ft/sec)

K = constant depending on grade of steel and is usually ≅1

D = diameter of missile (in.). For irregularly shaped missiles, an equivalent diameter is used, taken as the diameter of a circle with the same area as the projected frontal area of the irregularly shaped missile

The recommendation in Reference 3.6-13 to increase the perforation thickness, T, obtained by the Ballistic Research Laboratories Formula by 25% to prevent perforation is observed; that is:

$$t_p = 1.25T \tag{3.6-6}$$

where

t_p = thickness of steel barrier required to prevent penetration (in.)

3.6.1.6.3 Overall Structural Response

3.6.1.6.3.1 General. In general, pipe break loads are considered in combination with other loads (see Section 3.6.1.6.6). Dead loads, live loads, operating thermal loads, and earthquake loads may or may not be significant compared to the pipe break load, depending on the severity of the pipe break load. Thermal loadings due to pipe break have only skin effect and are not considered.

Pressure loads due to pipe break do not necessarily peak with pipe whip and jet impingement loads; however, in the analysis, they are considered to act simultaneously.

With regard to pipe break, when high energy pipes under pressure fail, a fluid jet is created. The associated jet impingement force on a target as well as the reaction force exerted on the piping by the fluid jet force have a time history qualitatively presented in Figure 3.6-27. This force is conservatively idealized as a step function load. For the fluid forces associated with these pipe failures, see Table 3.6-3.

To obtain a solution for the actual complex system, the structure is idealized by an equivalent single degree of freedom system (see Figure 3.6-28) following the procedures described by J. M. Biggs in Chapter 5 of Reference 3.6-1. The response of this mathematical idealization to a step function load (jet impingement) or to a step function load concurrently with an impact loading (due to whipping pipe) involves an energy transfer from the impacting object to the impacted structure. The following exposition on how this energy transfer is addressed makes use of procedures that have been presented by the Bechtel Corporation in its report on missile impact, Topical Report BC-TOP-9A, Revision 2 (Reference 3.6-13).

3.6.1.6.3.2 Structural Response to Whipping Pipe Missile Impact Load.

Discussion

A method of energy-balance procedures is used to evaluate the structural response when a missile impacts a target. The method utilizes the strain energy of the target at maximum response to counteract the residual kinetic energy of the target or target missile combination that results from the missile impact.

A missile of mass M_m is postulated to strike a spring-backed target mass, M_e, with a velocity, V_s. Since the actual coupled mass during impact varies, an estimated average effective target mass, M_e, is used to evaluate the inertia effects during impact. The impact of the missile is

considered plastic. This assumes that the missile remains in contact with the target after impact.

Velocity After Impact

The velocities of the missile and target after impact are calculated from the following relationships (Reference 3.6-19):

$$V_m = \frac{V_s(M_m - eM_e)}{M_m + M_e} \qquad (3.6\text{-}7)$$

$$V_t = \frac{V_s M_m (1 + e)}{M_m + M_e} \qquad (3.6\text{-}8)$$

where

V_m = missile velocity after impact (ft/sec)

V_t = target velocity after impact (ft/sec)

V_s = missile striking velocity (obtained by using basic velocity formulas, knowing the initial thrust force, missile mass, and missile travel before impact) (ft/sec)

M_m = mass of missile (lb-sec^2/ft)

M_e = effective mass of target during impact (lb-sec^2/ft)

e = coefficient of restitution

Plastic Impact

In a plastic impact, the coefficient of restitution becomes zero, and the velocity of the missile and target masses become equal following impact. The strain energy, E_s, required to stop the missile/target combination is the summation of the missile mass kinetic energy and the target mass kinetic energy at the end of the impact duration, as follows.

$$E_s = \frac{M_m V_m^2}{2} + \frac{M_e V_T^2}{2} \qquad (3.6\text{-}9)$$

From Equations 3.6-7 and 3.6-8:

$$V_m = V_T = \frac{M_m V_s}{M_m + M_e} \tag{3.6-10}$$

Substituting the value for V_m and V_T from Equation 3.6-10 into Equation 3.6-9, the required target strain energy is:

$$E_s = \frac{M_m{}^2 V_s{}^2}{2(M_m + M_e)} \tag{3.6-11}$$

Target Effective Mass

Due to the complexity of missile-target impact, a determination of an effective coupled mass on a continuous time basis by means of a general analytical solution is not available. However, an estimate of the average effective mass can be approximated from the results of impact tests on reinforced-concrete beams (Reference 3.6-10) in which the measured structural response is used to back-calculate the average mass during impact. Based on these data, the following formulas are used for estimating the target effective mass.

For concrete beams:

$$M_e = (D_x + 2T)\frac{B\gamma_c T}{g}, \text{ if } B \leq (D_y + 2T) \tag{3.6-12}$$

$$M_e = (D_x + 2T)(D_y + 2T)\frac{\gamma_c T}{g}, \text{ if } B \geq (D_y + 2T) \tag{3.6-13}$$

For concrete slabs:

$$M_e = (D_x + T)(D_y + T)\frac{\gamma_c T}{g} \tag{3.6-14}$$

For steel beams:

$$M_e = (D_x + 2d)M_x \tag{3.6-15}$$

For steel plates:

$$M_e = D_x D_y \frac{\gamma_s t}{g} \tag{3.6-16}$$

where

M_e = average effective mass of target during impact $\frac{(lb - sec^2)}{ft}$

M_x = mass per unit length of steel beam $\frac{(lb - sec^2)}{ft}$

D_x = maximum missile contact dimension in the direction (longitudinal axis for beams or slabs) (ft)

D_y = maximum missile contact dimension in the y direction (transverse to longitudinal axis for beams or slabs) (ft)

T = thickness or depth of concrete element (ft)

t = thickness of steel plate (ft)

d = depth of steel beam (ft)

B = width of concrete beam (not to exceed D + 2T)(ft)

γ_c = weight per unit volume of concrete (lb/ft^3)

γ_s = weight per unit volume of steel (lb/ft^3)

g = acceleration of gravity (32.2 ft/sec^2)

<u>Structural Response by Energy Balance Method</u>

a. General Procedures

The strain energy, E_s, required to stop the target (or missile-target combination) is determined from the relationships in Section 3.6.1.6.3.2.

The resistance-displacement function, R(x), for a concentrated load at the area of impact is determined from the target structure physical configuration and material properties.

The estimated maximum target response is determined by equating the available target strain energy to the required strain energy and solving for the maximum displacement, x_m, (see Figure 3.6-29).

b. Elasto-Plastic Target Response

For elasto-plastic target response with no other concurrent loads acting:

$$R(x) = Kx, (0 < x \leq x_e)$$

$$R(x) = Kx_e = R_m, (x_e < x \leq x_m)$$

where

R = resisting force of target (lb)

x = displacement of target (ft)

k = elastic Spring constant for target (lb/ft)

x_e = yield displacement (ft) (See Tables 3.6-4, 3.6-5, and Figure 3.6-29)

R_m = plastic resistance (lb) (See Tables 3.6-4, 3.6-5, and Figure 3.6-29)

x_m = maximum displacement of target (ft)

then

$$E_s = R_m\left(x_m - \frac{x_e}{2}\right)$$

or

$$x_m = \frac{E_s}{R_m} + \frac{x_e}{2} \qquad (3.6\text{-}17)$$

The required ductility ratio, μ_r, is obtained from Equation 3.6-17 by dividing both sides of the equation by x_e.

$$\mu_r = \frac{x_m}{x_e}$$

$$\mu_r = \frac{E_s}{x_e R_m} + 1/2 \qquad (3.6\text{-}18)$$

If other loads are present on the target structure which act concurrent with missile impact loads, (see Sections 3.6.1.6.5 and 3.6.1.6.6 and Table 3.6-6), the maximum combined displacement is determined as follows:

Let

x' = $x_e - x_o$ (see Figure 3.6-29) (ft)

x_o = displacement due to other loads (ft)

x_e = yield displacement (ft)

x_m = maximum combined displacement (ft)

R_m = plastic resisting force (lbs)

k = elastic spring constant (lb/ft)

Then

$$E_s = \frac{k(x')^2}{2} + kx'(x_m - x_e)$$

(See Figure 3.6-29)

or

$$x_m = \frac{E_s}{kx'} - \frac{x'}{2} + x_e$$

Substituting $x' = x_e - x_o$ in the above equation gives

$$x_m = \frac{E_s}{k(x_e - x_o)} + \frac{x_e + x_o}{2} \tag{3.6-19}$$

The required ductility ratio, μr, is obtained by dividing both sides of Equation 3.6-19 by x_e.

$$\mu_r = \frac{E_s}{R_m(x_e - x_o)} + \frac{1 + x_o / x_e}{2} \tag{3.6-20}$$

The values of M_r should be less than the allowable ductility ratios, M, given in Table 3.6-7.

3.6.1.6.3.3 Jet Impingement. Jet impingement loads are loads that emanate from a break in a high energy line. It is postulated that the characteristics of the jet are such that the jet exits from a break opening in the pipe equal in area to the cross sectional area of the pipe itself (see Figure 3.6-26). The jet is postulated to travel conforming to the configuration of the cross sectional area of the pipe for a distance of five pipe diameters and then to diverge at an angle of divergence of 10°. Where piping is restrained and break separation is limited to one-half pipe diameter or less a fan jet is postulated. A fan jet is perpendicular to the pipe centerline and extends 360° around the break at a 10° half angle as shown in Figure 3.6-30. For the jet thrust forces at the postulated breaks, see Table 3.6-3. Jet loads impacting structures are treated as equivalent static loads. A dynamic load factor is applied to the jet force emanating from the pipe and the resulting load is modified by an appropriate load factor according to its use in combination with other loads. The structure impacted is then evaluated for structural capability.

3.6.1.6.4 Allowable Design Stresses and Strains

For allowable design stresses and strains for reinforced concrete and structural steel, see Section 3.8.4.5 and Tables 3.8-11 and 3.8-12, except as modified in Sections 3.6.1.6.4.1 and 3.6.1.4.2.

3.6.1.6.4.1 Pipe Whip Loading With or Without Other Loads. The acceptability of pipe whip loading with or without other loads is considered from two aspects:

a. The overall structural response of the impacted structural element, and
b. The local damage sustained by the impacted structural element.

The overall structural response is considered acceptable if the ductility ratio resulting from the loading does not exceed the maximum allowable ductility ratios as given in Table 3.6-7. The

determination of ductility ratios utilizes the procedures set forth in Section 3.6.1.6.3 and the loading combinations in Section 3.6.1.6.6. In using these procedures, the allowable limit on section strength, M, used in the determination of yield displacement X_e, (Section 3.6.1.6.3.2, Tables 3.6-4 and 3.6-5 and Figure 3.6-29) is computed in accordance with the strength design methods described in ACI 318-71 (Reference 3.6-12) and in the general practices of Part 2 of the AISC specifications (Reference 3.6-11), modified by the dynamic strength increase factors of Table 3.6-8.

The local damage is considered acceptable if the pipe whip impact does not case spalling and excessive penetration in concrete, or perforation in steel, as determined by the procedures described in Section 3.6.1.6.2.

3.6.1.6.4.2 Pipe Break Loads (Excluding Pipe Whip) With or Without Other Loads. Pipe break loads (excluding pipe whip) with or without other loads are considered acceptable if the loading from the loading combinations in Section 3.6.1.6.6 does not result in stresses that exceed the allowable limits on section strength as given in Tables 3.8-11 and 3.8-12, modified by the dynamic strength increase factors in Table 3.6-8.

3.6.1.6.5 Loads, Definition of Terms, and Nomenclature

For loads, definition of terms, and nomenclature, see Section 3.8.4.3.

3.6.1.6.6 Load Combinations

3.6.1.6.6.1 Seismic Category I Concrete Structures. For load combinations for Seismic Category I concrete structures, see Table 3.8-9, load combinations 6, 7, and 8.

3.6.1.6.6.2 Seismic Category I Steel Structures. For load combinations for Seismic Category I steel structures, see Table 3.8-10, load combinations 6, 7, and 8.

3.6.1.7 Structural Design Loads

Structural elements are designed to withstand the loads generated by piping failures outside of primary containment in combination with other loads given in Section 3.6.1.6.6. Table 3.6-6 furnishes the design loads considered in the areas where piping failures occur.

3.6.1.8 Analysis of Load Reversal

Structural elements such as floors, interior walls, exterior walls, and the building as a whole are analyzed for the effects of reversal of load due to the postulated pipe failure accident. They are also analyzed for rebound loads that accompany pipe break accidents. The analysis approach for rebound is seen in Figure 3.6-31.

3.6.1.9 Modified Structures

The capabilities of structures to carry the design loads will be demonstrated for modifications as part of design change process controls.

3.6.1.10 Verification That Failure of Any Structure Does Not Preclude Safe Reactor Shutdown

Structures subject to pipe whip and/or jet impingement loads are investigated and found not to fail under these loads in conjunction with the applicable load combinations, so that there are no cases of structural barriers failing and causing additional structural failures which would adversely affect the mitigation of the consequences of accidents and the capability to bring the plant to a cold shutdown condition.

3.6.1.11 Verification That Adequate Redundancy Exists for All Postulated Fluid Piping System Ruptures

3.6.1.11.1 Approach

The purpose of the study is to ensure that for all postulated ruptures of fluid piping systems, safe reactor operation and shutdown is not precluded. The basis of this approach is that adequate separation of redundant systems or components required to shut down and maintain the reactor in a cold condition provides the level of protection required to ensure safe reactor operation and shutdown.

The input used for this study includes the routing of all cables, cable trays, and conduit necessary to shut down and maintain the reactor in a cold condition. The locations of all motor control centers; instrument racks; sensors; and heating, ventilating, and air conditioning (HVAC) equipment necessary to shut down and maintain the reactor in a cold condition are also included in the input of this study.

The locations of all postulated high- and moderate-energy fluid piping system ruptures dictate where this study is to be performed.

The input described above is coded to indicate the location of the system or component by elevations; the electrical division to which the component belongs; what the function of the component is; the various references, such as the drawings, in which the component is found; devices interconnecting the component and another system; and additional information of this type. This coding facilitates storage of the input for retrieval at any time.

Table 3.6-3 lists the high-energy design-basis break locations outside containment, the piping systems involved, the pipe diameter, the plan figure showing the piping system, and the maximum blowdown thrust or the thrust versus time figure.

Figures 3.6-32 through 3.6-56 illustrate and list the high-energy break locations inside containment and inside the main steam tunnel.

Moderate-energy crack locations are postulated in accordance with Standard Review Plan Sections 3.6.1 and 3.6.2.

3.6.1.11.2 Method of Analysis for Postulated High-Energy Fluid System Ruptures

3.6.1.11.2.1 Effects of Postulated Passive Component Failures. Postulated pipe breaks in high-energy fluid systems are investigated to determine their effects on the ability to bring the plant to a safe shutdown and to limit the offsite radiological consequences to an acceptable level as stated in 10 CFR 50.

On a case-by-case basis, the effects of pipe whip, jet impingement, and the resulting environmental conditions on safety-related equipment necessary for safe shutdown are evaluated. The effects of the postulated pipe break are dependent on the fluid properties of the system, the location and orientation of the pipe break, the proximity to safe shutdown systems, components, and structures, and the individual design limits of the safe shutdown systems, components, and structures.

Pipe breaks in high energy systems are postulated according to the criteria in Section 3.6.2.1. After identifying what equipment becomes inoperable, a worst case single random active component failure is postulated in a system not affected by the postulated high-energy fluid system rupture. Additionally, if the direct consequences of the postulated rupture results in a reactor or turbine trip, or could otherwise cause a loss of offsite power, offsite power is assumed unavailable.

3.6.1.11.2.2 Analytical Procedure. After all the consequences of the postulated pipebreak, passive and active component failures are evaluated, an analysis determines if safe shutdown can be accomplished. The following guidelines are used in this analysis:

a. For postulated ruptures of fluid piping systems, ensure that core cooling and reactivity control is maintained,

b. Demonstrate that redundant components or systems necessary to safely shut down and cool the reactor are not involved in the postulated passive component failure, and

c. Demonstrate that offsite radiological consequences do not exceed relevant standards.

3.6.1.11.3 Method of Analysis for Postulated Moderate-Energy Fluid System Ruptures

3.6.1.11.3.1 Approach. The analysis of moderate energy piping is performed in accordance with BTP ASB 3-1, Rev. 1, 1981. Postulated ruptures in moderate-energy fluid systems do not generate pipe whip. The analysis investigates the effects of the environment that results from such a postulated rupture on safety-related equipment, including the effects of water spray.

The effects of the postulated moderate-energy pipe cracks are dependent on the fluid properties, available fluid reservoir, drain systems, and the location and the individual design limits of the safety-related equipment, components, and structures necessary for safe shutdown.

Where moderate-energy pipe cracks are postulated in close proximity to high energy systems, the environmental analysis compares the effects of both high- and moderate-energy pipe ruptures. The most limiting case is evaluated for safe cold shutdown.

Moderate-energy pipe cracks are postulated according to the criteria in Section 3.6.2.1.

3.6.1.11.3.2 Method of Analysis. The locations of all postulated ruptures, resulting in through wall leakage cracks, are identified for later retrieval. The analysis assumes that the spray resulting from a postulated moderate-energy rupture causes the malfunction of all equipment not enclosed by compartments that minimize the effects of flooding.

Additionally, the most damaging single random active component failure in a system not affected by the postulated moderate energy pipe rupture is assumed. If the direct consequences of the passive component failure results in a turbine or reactor trip or could otherwise cause a loss of offsite power, then offsite power is assumed unavailable.

3.6.1.11.4 Summary of Analysis

In those cases where analysis discussed in Sections 3.6.1.11.2 and 3.6.1.11.3 identified a location where a postulated pipe rupture in a high- or moderate-energy system had impact on a safety-related component, which precluded the safe shutdown and cooling of the reactor, the component was relocated or protected.

This analysis by actual examination of the plant is undertaken to provide results based on as-built conditions.

Piping layouts for areas containing high- and moderate-energy lines, whose failure can affect the performance of safety-related equipment, are presented as Figures 3.6-1 through 3.6-20B.

Section 3.6.1.11 discusses in detail the methods used to demonstrate that no fluid system piping rupture, in conjunction with a single active component failure, precludes safe shutdown of the plant.

The following should serve to further clarify the method of analysis:

a. Analysis is performed to section B.3 criteria (pages 3.6.1-12 and 3.6.1-13) of BTP ASB 3-1, Rev. 1, 1981;

b. The forces developed at each postulated high-energy pipe break are determined by the methods of Section 3.6.2.2. The effects of the resultant pipe whip and jet impingement are evaluated. Credit is taken for automatic isolation and/or operator action to mitigate the consequences of the postulated pipe break, if the equipment required for this function is not affected by the break or included in item d below;

c. As a first step, all equipment with impact from the whipping pipe or jet is assumed to fail. If the equipment is required for safe cold shutdown or accident mitigation, a detailed analysis is performed to determine if the equipment will actually fail. Structures contacted by the whipping pipe or jet are evaluated for structural adequacy by the methods described in Section 3.6.2.2;

   Impact to pipes of smaller nominal diameter than the impacting pipe are assumed to fail, regardless of wall thickness of impacted pipe. Impact to pipe of both larger nominal diameter and thinner wall thickness than the impacting pipe are assumed to develop through wall leakage cracks;

d. The following comprises the minimum set of equipment necessary for safe shutdown following any high-energy or moderate-energy pipe break/crack assuming a worst-case single active failure:

   1. Pipe break/crack detection instrumentation
   2. Automatic, high-energy line break isolation equipment
   3. RPS (scram)
   4. MSIVs
   5. Five (5) SRVs for reactor vessel depressurization
   6. A single RHR loop with a heat exchanger (loop A or B), in the alternate shutdown cooling mode providing reactor vessel inventory makeup (short term cooling) and reactor vessel and suppression pool inventory cooling (long term cooling)
   7. Supporting Service Water
   8. Supporting HVAC
   9. Supporting electrical power sources;

e. After items b, c, and d have been evaluated, the ability to safely shut down is evaluated;

f. If analysis indicates that safe shutdown cannot be accomplished, then

   1. Reroute or relocate cable, pipe, or equipment to prevent loss of function, or

   2. If this is not feasible, shield the appropriate affected component(s) to prevent loss of function;

g. The flooding and environmental effects of moderate-energy failure are evaluated to determine whether they are more severe than the high-energy breaks. The high-energy breaks are addressed in Sections 3.6.1.11 and 3.6.1.15.

The area temperature is evaluated by determining the limiting postulated pipe break and using RELAP4/MOD5 (Reference 3.6-21) or GOTHIC (References 3.6-25 and 3.6-26). GOTHIC analyses described within Section 3.6 are performed in accordance with the constraints and limitations provided in the River Bend Unit 1 Safety Evaluation (Reference 3.6-27), or have demonstrated results which are essentially the same or conservative. The limiting pipe break for temperature analysis is that pipe break giving the highest energy release rate over the longest blowdown period.

The effects of flooding are evaluated by determining the limiting pipe break and calculating the effects of the fluid release. The limiting pipe break for flooding analysis is that pipe break with the highest mass flow rate over the longest blowdown period.

Peak differential pressure analysis results are provided in Table 3.6-9 and discussed in Section 3.6.1.20.

See Section 3.6.1.13 for electrical equipment environmental qualifications.

3.6.1.12 Control Room Habitability

A postulated rupture of main steam, feedwater, or RWCU piping has no effect on the continued habitability of the control room, since the radiation dose that control room personnel receive as a result of a postulated rupture is below the allowable limits.

The nuclear steam supply system (NSSS) piping outside of primary containment within the reactor building is enclosed by the main steam tunnel. The main steam tunnel, provided with pressure-relieving blowout panels, is designed to withstand the worst postulated piping system rupture attributable to the NSSS within the steam tunnel.

The high energy piping in the main steam tunnel is provided with pipe whip restraints as described in Section 3.6.1.5. These restraints limit the motion of the free ends of the ruptured NSSS piping to preclude the impact of the NSSS piping with the main steam tunnel structure.

The remaining high energy piping outside the primary containment is not routed in the vicinity of the control room, or does not possess sufficient energy to adversely affect the structural integrity of the control room wall.

Additionally, a remote shutdown panel is provided to permit safe reactor shutdown to a cold condition in the event the control room must be evacuated.

3.6.1.13 Electrical Equipment Environmental Qualifications

All electrical systems necessary for safe shutdown and necessary to maintain the plant in a safe shutdown condition are designed to remain functional in the general area environment resulting from a high-energy line break or from leakage cracks in moderate-energy piping. Specific equipment is either of the following:

a. Designed to remain functional as long as necessary in the general area environment, or

b. Isolated from the general area environment in compartments capable of maintaining normal equipment operating conditions.

Certain rotating equipment cannot be designed to function in the more severe, local steam environment. However, due to physical separation, rotating equipment of not more than one system is exposed to the local conditions which exceed the general area accident environment. Required redundancy is thus maintained for safety equipment.

See Section 3.11 for a more complete description of environmental design of electrical equipment.

3.6.1.13.1 Identification of Equipment

Safety equipment required to mitigate the consequences of an accident and place the reactor in a cold shutdown condition is listed in Table 3.11-2. The table also indicates the required duration following an accident that equipment is required to operate.

3.6.1.13.2 Environmental Design

See Section 3.11 for a discussion of environmental design and an analysis of safety-related electrical components. The section identifies the safety-related equipment that must operate in a hostile environment, and Table 3.11-2 indicates the postulated environmental enveloping conditions for both the general and local accident areas.

3.6.1.13.3 Jet Impingement Barriers

For results of the steam system study, see Section 3.6.1.11.4. Jet impingement barriers have been provided where analysis indicates they are needed to protect components required for reactor safe shutdown. In addition, some room wall, floors, and ceilings act as jet impingement barriers.

3.6.1.13.4 Control Room Equipment

Control room environmental effects, resulting from pipe break accidents, are discussed in Section 3.6.1.12. The postulated pipe breaks have no effect on the control room environment. All control room equipment, therefore, remains functional following a break.

3.6.1.13.5 Onsite Power Distribution System Equipment

See Section 3.11.

3.6.1.14 Design Diagrams of Nuclear Steam Supply System Piping

Figures 3.6-58 to 3.6-60 show the routing of NSSS piping from the outboard end of the containment penetrations to the turbine building.

3.6.1.15 Flooding Analysis

A study investigating the potential flooding attributable to the postulated rupture of high-energy fluid piping systems outside primary containment is provided below.

3.6.1.15.1 Postulated Rupture of the Reactor Feedwater Piping

The reactor feedwater piping outside primary containment, inside the reactor building, is completely enclosed by the main steam tunnel to provide protection against the dynamic effects of postulated fluid piping system ruptures. The main steam tunnel is provided with a blowout panel to discharge steam, to the atmosphere, above the turbine building. A second blowout panel provides for water drainage from the main steam tunnel into the turbine building. A flooding evaluation of the turbine building from any feedwater line break has been completed which shows that no safety-related equipment located within the turbine building would be adversely affected.

3.6.1.15.1.1 Consequences. The postulated rupture, of the feedwater piping in the main steam tunnel, would scram the reactor on low water level.

The isolation valves on the reactor pressure boundary would close to prevent loss of reactor coolant. A postulated active component failure could prevent the inboard isolation valve from

closing, while the dynamic effects of the passive component failure could prevent the outboard isolation valve from closing. The inboard check valve would close to prevent coolant loss, effectively stopping the reactor feedwater flow into the main steam tunnel.

Although a loss of offsite power is assumed, sufficient equipment remains available to safely shut down the reactor to a cold condition.

3.6.1.15.2 Postulated Ruptures of Reactor Water Cleanup System

Since the reactor water cleanup (RWCU) system has no safety function, postulated ruptures of this system have no effect on safe reactor operation, except the effect these postulated ruptures have on structures, systems, or components important to safety. As the analyses discussed in Sections 3.6.1.11 and 3.6.1.20 indicate, postulated ruptures of the RWCU system have no effect on the ability to safely shut down the reactor to a cold condition.

3.6.1.15.3 Postulated Ruptures of the Auxiliary Steam, Heating Steam, Auxiliary Condensate, and Heating Steam Condensate Systems

Since these systems have no safety function, postulated ruptures of these systems, as described in Sections 3.6.1.11 and 3.6.1.20, have no unacceptable consequences, except the effect these postulated ruptures have on structures, systems, or components important to safety. Ruptures in the auxiliary steam (AS), heating steam (HS), auxiliary condensate (CO), or HS condensate (HCO) systems do not prevent safe reactor shutdown to a cold condition. In the case of an AS line break in the reactor building, redundant isolation valves are automatically closed by temperature elements. There are no dynamic effects from the rupture, but the equipment environment (temperature and humidity) would exceed the qualified limits if the AS line to the reactor building was not isolated.

3.6.1.15.4 Additional Considerations

Postulated ruptures of high-energy fluid piping systems are not considered to cause flooding if the contained fluid is in the vapor phase. The main steam and reactor core isolation coolant systems are not considered sources of flooding incidents for this reason.

3.6.1.16 Quality Control and Inspection Programs

The quality control and inspection programs, required for the design and construction of piping systems outside containment, were given in Section 17.1. For the operational phase, the Inservice Inspection (ISI) Program and the Operational Quality Assurance Program Description (OQAPD) are applicable.

3.6.1.17 Leak Detection System Capabilities

The floor and equipment drain headers drain into five sumps in the reactor building at el. 422 ft 3 in. Four sumps are provided for floor drains and one for equipment drains. These sumps contain level switches which activate the sump pumps on high water level. This allows the operator to determine the severity of the leak and take corrective action.

In addition, wall-mounted Class 1E level switches are provided in each pump room in the reactor building at the basement (422 ft 3 in.) level. See Sections 3.4.1.4, 6.3.2.5, and 9.3.3 for more details.

3.6.1.18 Emergency Procedures to Mitigate the Consequences of a Postulated Fluid Piping System Rupture Outside Primary Containment

Shutdown modes available to the operator are determined by whether or not the reactor is isolated.

3.6.1.18.1 Reactor Not Isolated

For those classes of postulated ruptures which do not result in isolation of the RCPB, a normal shutdown is accomplished by using the main condenser and reactor feedwater system.

3.6.1.18.2 Reactor Isolated

With the reactor isolated from the main condenser, a variety of shutdown modes are available to the operator. A single postulated pipe rupture does not preclude the availability of all of them.

3.6.1.18.2.1 Reactivity Control. Before a shutdown can be accomplished, the core must be prevented from generating additional power. Two methods of reactivity control are utilized on the Columbia Generating Station reactor.

The CRD system is the normal means of controlling core reactivity.

The SLC system can be used to shut down the reactor by means of injected sodium pentaborate poison. Both of these systems are redundant and driven by either offsite or plant standby power supplies.

No single accident can render both power supplies unavailable.

3.6.1.18.2.2 Core Coolant Level Maintenance. The systems available for core cooling during reactor isolation are RHR, automatic depressurization system (ADS), RCIC, LPCS, and HPCS. The RHR system is comprised of four modes: shutdown cooling, hot standby (see

Section 5.4.7), low pressure cooling injection (see Section 6.3.2.2.4), and suppression pool cooling (see Section 6.2.2). See Section 6.3.1.2.4 for a description of the ADS. See Section 5.4.6 for a description of the RCIC system. See Section 6.3.2.2.3 for a description of the LPCS. See Section 6.3.2.2.1 for a description of the HPCS.

3.6.1.18.3 Methods of Shutdown Following a Postulated Rupture of a Fluid Piping System

3.6.1.18.3.1 Postulated Rupture of Main Steam Piping in the Main Steam Tunnel Vicinity. A postulated rupture of the main steam system does not generate any pipe whip impact on structures in the reactor building, since the main steam lines are supplied with pipe whip restraints.

High pressure differential signals from the main steam flow restrictors (indicating decrease in steam pressure at the turbine inlet) or high temperature indication in the vicinity of the main steam piping initiates automatic isolation of the main steam lines which, in turn, causes a reactor scram. Automatic relief valve operation maintains the reactor vessel pressure within allowable limits.

With the reactor isolated and scrammed, the HPCS or manual initiation of the ADS would be used to depressurize the reactor while either of the two redundant trains of the shutdown cooling mode of RHR systems would be employed to remove decay and sensible heat from the reactor pressure vessel (RPV).

3.6.1.18.3.2 Postulated Rupture of Reactor Feedwater Piping. A postulated rupture of a reactor feedwater pipe does not generate pipe whip impacts on structures outside primary containment within the reactor building since the feedwater piping is restrained.

The resulting blowdown from the feedwater break scrams the reactor when the low water level setpoint is reached. The operator can also initiate manual scram as the instrumentation will indicate a mismatch in main steam flow rate, reactor feedwater flow rate, and RPV inventory.

The scram, due to low water level, initiates emergency core cooling and RCIC operation. Decay heat removal is manually initiated by the operator.

The shutdown modes available after an operator initiated scram are the same as listed in Section 3.6.1.18.3.1.

3.6.1.18.3.3 Postulated Rupture of Reactor Core Isolation Cooling System Piping. Ruptures are postulated in the RCIC system in the turbine steam supply line or the condensate removal line, from the drip pot. A rupture of either line precludes operation of the RCIC system. A rupture of either line does not generate a scram and, therefore, normal shutdown procedures are used. Automatic isolation of the RCIC system occurs if a high pressure drop across the flow restrictor is sensed, high area temperature is sensed, or low line pressure is sensed.

3.6.1.18.3.4 Postulated Rupture of Residual Heat Removal System Piping. The RHR system, except in the hot standby mode, operates at pressures less than 135 psig steam dome pressure. Postulated failures of the RHR system piping at these pressures does not require any particular safety action. Rather shutdown cooling is reestablished using the redundant train of the RHR system.

In the event that the suction line from the RPV to the RHR pumps becomes inoperable, reactor depressurization can be accomplished by using the HPCS or the ADS. Heat can be removed from the suppression pool by use of the suppression pool cooling mode of the RHR system. For a detailed discussion of the possible shutdown modes see Section 15.2.9.

3.6.1.18.3.5 Postulated Rupture of the Control Rod Drive System Piping. Rupture of the CRD piping does not generate any significant pipe whip loads. While rupture of the CRD piping could prevent system operation, reactor scram is possible using either the nitrogen accumulators or reactor pressure. The SLC system could also be used to bring the reactor to a subcritical condition if the CRDs failed to function.

3.6.1.18.3.6 Postulated Ruptures of the Auxiliary Steam, Heating Steam, Auxiliary Condensate, and Heating Steam Condensate Piping. The consequences of a rupture in any of the above, including the dynamic effects of pipe whip and the resulting environmental conditions, are investigated as described in Section 3.6.1.11. In no instance does a postulated rupture of these systems preclude reactor shutdown to a cold condition.

These systems provide no emergency function that would be required to mitigate the consequences of a postulated piping failure. Therefore, normal reactor shutdown as well as the emergency methods described would not be simultaneously impaired.

3.6.1.18.3.7 Postulated Rupture of the Reactor Water Cleanup System Piping. The consequences of a RWCU system piping rupture are investigated as discussed in Section 3.6.1.11. In no circumstance, does the postulated failure of RWCU system piping preclude the availability of all shutdown modes. Since the RWCU system does not fulfill any safety function, nonoperability has no impact on the safe shutdown of the reactor.

3.6.1.19 Seismic and Quality Classifications of Piping Used in the Dynamic Analysis

See Table 3.2-1 for the seismic and quality classifications of piping systems identified in Table 3.6-3. See Section 3.2 for the definitions of the various seismic and quality classifications.

3.6.1.20 Method Used to Predict Blowdown Rates and Subcompartment Pressure Transient After a Postulated Pipe Break

3.6.1.20.1 Blowdown Analysis for a Postulated Pipe Break Outside the Primary Containment

The analytical approach used to determine the blowdown mass and energy rates from a postulated pipe break outside the primary containment are described in Sections 3.6.1.20.1.1 through 3.6.1.20.1.3.

3.6.1.20.1.1 Method of Analysis for a Postulated Pipe Break in Larger Pipes. For larger pipes, the blowdown mass and energy release are predicted by using the computer programs RELAP3 (Reference 3.6-9) or RELAP4/MOD5 (Reference 3.6-21) or GOTHIC (References 3.6-25 and 3.6-26). In the computer model, the piping system is nodalized into a number of volumes connected by flow junctions (or flow paths in GOTHIC). A multiplier of 1.0 is used with the choked flow correlation. Except in special cases, all breaks are assumed to be the double-ended circumferential type which open instantaneously. Initial conditions and other assumptions necessary for the analysis are such that the result is on the conservative side.

3.6.1.20.1.2 Method of Analysis for a Postulated Pipe Break in Smaller Pipes. For smaller pipes, a constant blowdown profile with an applicable choked flow correlation is used. The initial conditions are chosen to maximize the blowdown mass and energy release rates.

3.6.1.20.1.3 Blowdown Mass and Energy Release Rates for a Postulated Pipe Break in the Main Steam Line and the Reactor Feedwater Line in the Main Steam Tunnel. See Section 3.6.1.20.3.2 for a description of the arrangement and features of the main steam tunnel. For subcompartment analysis in the main steam tunnel, the postulated break in the main steam line and in the feedwater line is assumed to be a crack with the flow area equivalent to the flow area of a single-ended pipe. The blowdown mass and energy release rates are computed by the RELAP3 Program or GOTHIC (References 3.6-25 and 3.6-26).

Figures 3.6-61 and 3.6-62 show the mass and energy release rates after a postulated crack in the main steam line in the main steam tunnel. Figures 3.6-63 and 3.6-64 show the mass and energy release rates after a postulated crack in the reactor feedwater line in the main steam tunnel.

3.6.1.20.2 Subcompartment Analysis for Postulated Pipe Break Outside the Primary Containment Excluding the Main Steam Tunnel, Ventway, and Tunnel Extension

3.6.1.20.2.1 Method of Analysis. The pressure transient in the reactor building after a postulated pipe break is analyzed with the computer programs RELAP3 (Reference 3.6-9) or RELAP4/MOD5 (Reference 3.6-21) or GOTHIC (References 3.6-25 and 3.6-26). In the computer model, subcompartments are represented by nodes (or volumes in GOTHIC), and flow paths between two nodes are represented by flow junctions (or flow paths in GOTHIC). Volumes, vent areas, flow resistances, initial atmospheric conditions, as well as the blowdown mass and energy release rates from the pipe breaks, are input to the computer program. Since

the absolute pressure within the subcompartments after a pipe break outside the primary containment is low in all cases, no significant pressure gradient exists within a subcompartment itself. Therefore, a subcompartment is not nodalized in the analysis and a sensitivity study is not performed.

Table 7.6-2 lists leak detection system instrumentation. Table 6.2-16 lists valve closure times for automatic isolation functions tied into the leak detection system. Credit is taken for automatic isolation if the system capability is not affected by the postulated pipe break or assumed as a single active component failure. Check valves close on reversal of flow in a fraction of a second. In all cases, the blowdown terminates when the inventory of fluid in the line is exhausted.

Where blowdown flow is not automatically terminated by isolation valves or check valves as described above, the duration of the blowdown event as the inventory of fluid in a line is exhausted is not considered in the analysis of peak compartmental pressure and temperature. To evaluate the peak pressures and temperatures in compartments and structures following a postulated break of the high energy pipes inside the structures, the blowdown analysis is extended far beyond the initial transient until the blowdown flow becomes steady or decreases continuously. The duration of the analysis is therefore sufficient to correctly predict the peak pressures and temperatures in these compartments and structures.

For a postulated pipe break or leakage crack in the main steam lines outside the primary containment, the flow from the reactor side of the break is terminated by the closing of the main steam isolation valves (MSIVs) located in each of the four main steam lines. The MSIVs start to close at 0.5 sec after the break and are fully closed at or prior to 5.5 sec after the break, as given in Table 15.6-5.

For a postulated break or leakage crack in the reactor feedwater lines outside primary containment, the flow from the reactor side of the break is terminated by closing of the check valves in each of the two reactor feedwater lines. The check valves start to close when the direction of flow reverses, and the flow from the reactor side of the break is therefore terminated within a fraction of a second.

3.6.1.20.2.2 Initial Atmospheric Conditions. The initial atmospheric conditions within the subcompartments used for the analysis are

Reactor Building:

a. Pressure = 14.7 psia,
b. Temperature = 110°F, and
c. Relative humidity = 0.0%.

Radwaste Building:

a. Pressure = 14.45 psia,
b. Temperature = 80° F, and
c. Relative humidity = 60%.

These conditions are simulated in the RELAP computer analysis as a homogeneous saturated steam-water mixture at 14.7 psia with an average density equivalent to the density of air at the above conditions. When using GOTHIC, the initial conditions are as described using air and water vapor.

3.6.1.20.2.3 Vent Flow. In the RELAP analyses, the vent flow between the subcompartments is assumed to be a homogeneous steam-water mixture with 100% water entrainment. For choked flow, a multiplier of 0.6 is used for Moody two-phase flow correlations. In the GOTHIC analyses, vent flow is modeled as air, steam and water with choked flow as described in References 3.6-25 and 3.6-26. For unchoked flow, the flow resistance consists of an entrance loss, an exit loss, and frictional losses. For conservatism, an entrance loss of approximately 0.5 and an exit loss of 1.0 are assumed for most of the vents.

3.6.1.20.2.4 Results of Subcompartment Analyses. Subcompartment analyses are performed for all subcompartments containing high energy piping. The results are summarized in Table 3.6-9.

3.6.1.20.2.5 Verification of Structural Adequacy. Verification of structural adequacy of compartments or of structural elements thereof subject to pressure generated by a postulated pipe break and to the local effects in the structure generated by the postulated pipe break; namely, a broken pipe reaction, jet impingement, and pipe whipping impact are discussed in Sections 3.6.1.6 through 3.6.1.10.

3.6.1.20.3 Subcompartment Analysis for a Postulated Pipe Break in the Main Steam Tunnel

See Section 3.6.1.20.3.2 for a description of the arrangement and features of the main steam tunnel. Subcompartment analysis in the main steam tunnel, ventway, and tunnel extension is performed for a postulated crack in the main steam line (see Section 3.6.1.20.1.3). Comparison of mass and energy release rates for the postulated crack in the main steam line to that of a postulated crack in the reactor feedwater line (Figures 3.6-61 through 3.6-64) shows that the main steam line crack is the limiting case. Other lines inside the main steam tunnel or its extension are of smaller sizes and a break in those lines is less severe.

3.6.1.20.3.1 General Approaches. The pressure and temperature transients in the main steam tunnel, ventway, and tunnel extension after a postulated crack in the main steam line are computed by the RELAP4/MOD5 program (Reference 3.6-21) or GOTHIC (References 3.6-25 and 3.6-26). The general approaches discussed in Sections 3.6.1.20.2.1 through 3.6.1.20.2.3 also apply to this case.

3.6.1.20.3.2 Description of the Main Steam Tunnel, Ventway, and Tunnel Extension. Descriptive information of the main steam tunnel, ventway, and tunnel extension is provided in Section 3.8.4.1.1.4. Figures 3.6-65 and 3.6-66 show a sectional plan view and a sectional elevation view, respectively, of the main steam tunnel, ventway, and tunnel extension.

In plan, the main steam tunnel is located at 0° azimuth of the north side of the reactor building; in elevation, it extends from el. 501 ft 0 in. to el. 522 ft 0 in. At the interface of the reactor building and the turbine generator building, the main steam tunnel continues for a short distance into the turbine generator building; the portion in the turbine generator building is referred to as the tunnel extension. The ventway starts at the same level as the main steam tunnel and extends horizontally from the main steam tunnel in the easterly direction and continues upward to the underside of the corridor floor above at el. 548 ft 0 in., where a blowout panel in the north wall of the ventway provides a ventilating path to the atmosphere.

Four blowout panels are used, as shown in Figures 3.6-65 and 3.6-66:

a. Panel A, vertical, part of secondary containment, located between the north end of the main steam tunnel (in the reactor building) and the tunnel extension (in the turbine generator building), bolted in place, of sheet steel,

b. Panel B, vertical, part of secondary containment, located in the east wall of the main steam tunnel, bolted in place, of sheet steel,

c. Panel C, horizontal, part of secondary containment, located at the top of the main steam tunnel, the north edge of panel is hinged, other edges are free and not bolted in place, of sheet steel, and

d. Panel D, vertical, not part of secondary containment, located in the north exterior wall of the ventway, bolted in place, of insulated metal siding.

The fasteners of the blowout panels which are bolted in place (namely, panels A, B, and D) are designed to fail in single shear, and all blow-off panels (namely, panels A, B, C, and D) are designed to blow-off and permit venting when the pressure generated in the main steam tunnel, ventway, and tunnel extension by a postulated pipe break within the main steam tunnel or tunnel extension exceeds 0.5 psi.

3.6.1.20.3.3 Analysis for a Postulated Pipe Break in the Main Steam Tunnel. The analysis for a postulated pipe break in the tunnel extension is discussed in Section 3.6.1.20.3.4.

Figure 3.6-67 shows the nodalization scheme for a postulated pipe break in the main steam tunnel. For conservatism blow-out panels A and B are assumed to remain in place during the pressure transient. Therefore, the tunnel extension and the turbine generator building are not modeled in this case. Nodes 1 and 2 represent the main steam tunnel. Nodes 3, 4, 5, and 6 represent the ventway. Tables 3.6-10 and 3.6-11 provide the volume and flow junction data, respectively.

The hinged panel C is modeled as an inertia valve in the RELAP4/MOD5 analysis (see Reference 3.6-21). The differential equation of motion for the valve gate is:

$$I \ddot{\theta} (t) = \overline{A} P (t) - K \dot{\theta} (t) \qquad (3.6\text{-}21)$$

where

θ = opening angle (radians)

t = time (sec.)

$$\dot{\theta} = \frac{d\theta}{dt}$$

$$\ddot{\theta} = \frac{d_2\theta}{dt^2}$$

I = moment of inertia (lb_{mass} - ft^2)

A = area X moment arm (ft^3)

P = differential pressure (lb/ft^2)

K = damping constant $\left(\frac{lb_{mass} - sec^2}{ft}\right)$

Let ω= angular velocity in radians/sec. and substituting $\omega = \theta$ and $\omega = \theta$ in Equation 3.6-21.

$$I\dot{\omega} + K\omega = \overline{A}P$$

It has the solution

$$\theta = \theta_o + \omega_o t + \left(\frac{AP}{K} + \omega_o\right)\left[t\frac{I}{K}\left(1 - e^{\frac{-Kt}{I}}\right)\right] \tag{3.6-22}$$

where θ_o and ω_o are values for θ and ω at t = O.

For panel D, the differential equation of motion is

$$M\ddot{s}(t) + AP(t) \tag{3.6-23}$$

where

M = mass of panel $\left(\frac{lb_{mass} - sec^2}{ft}\right)$

s = displacement (ft)

v = velocity (ft/sec)

s = linear acceleration (ft/sec^2)

A = area of panel (ft^2)

P = average pressure (lb/ft^2)

t = time (sec)

The frictional force is neglected. The solution of the above equation is:

$$s = s_o + (v_o + \frac{F}{2m} t)\, t \qquad (3.6\text{-}24)$$

where s_o and v_o are values for s and v at t = O.

The displacement, s, of the panel and the opening area as functions of time are determined by iterative procedures.

Pertinent properties of blow-out panels C and D are seen in Table 3.6-12.

Figures 3.6-68 and 3.6-69 are plots of the pressure transients and Figures 3.6-70 and 3.6-71 are plots of the temperature transients for a postulated pipe break in Node 1.

Figures 3.6-72 and 3.6-73 are plots of the pressure transients and Figures 3.6-74 and 3.6-75 are plots of the temperature transients for a postulated pipe break in Node 2.

Blow-out panels C and D are assumed to blow off at the differential pressure noted in Section 3.6.1.20.3.2.

3.6.1.20.3.4 Analysis for a Postulated Pipe Break in the Tunnel Extension. Figure 3.6-76 shows the nodalization scheme for a postulated pipe break in the tunnel extension. Nodes 1 and 2 represent the tunnel extension. The vertical pipe restraint (see Figure 3.6-24) divides

Node 1 and 2. Nodes 3, 4, and 5 represent the following portions of the turbine generator building:

a. Node 3 represents the portion between the mezzanine floor at el. 471 ft 0 in. and the operating floor at el. 501 ft 0 in.,

b. Node 4 represents the portion between the ground floor at el. 441 ft 0 in. and the mezzanine floor at el. 471 ft 0 in., and

c. Node 5 represents the portion between the operating floor el. 501 ft 0 in. and the roof of the turbine generator building.

For conservatism, panel A (in Figures 3.6-65 and 3.6-66) is assumed to remain in place during the pressure transient, and only 10% of the insulated metal siding comprising the exterior walls above the operating floor (see Figure 1.2-5) of the turbine generator building is assumed to blow off the structural steel frame at a differential pressure of 0.5 psi. Tables 3.6-13 and 3.6-14 provide the volume and flow junction data, respectively.

Figures 3.6-77 and 3.6-78 are plots of the pressure transients and Figures 3.6-79 and 3.6-80 are plots of the temperature transients for a postulated pipe break in Node 1.

Figures 3.6-81 and 3.6-82 are plots of the pressure transients and Figures 3.6-83 and 3.6-84 are plots of the pressure transients for a postulated pipe break in Node 2.

3.6.1.20.3.5 Verification of Structural Adequacy. Verification of structural adequacy of the main steam tunnel ventway and tunnel extension, or of structural elements thereof, subject to load combinations involving pressure generated by a postulated pipe break and local effects in the structure generated by the postulated pipe break; namely, broken pipe reaction, jet impingement, and pipe whip impact are discussed in Sections 3.6.1.6 through 3.6.1.10.

3.6.1.20.3.6 Turbine Building Consequences of a Postulated Pipe Break in the Steam Tunnel Extension. The only items with safety-related functions in the turbine building are some reactor protection system (RPS) sensor inputs from the main steam system, MSIV isolation logic inputs from the main steam system, and the tower makeup transformers located on the turbine ground floor which are required to function only for the design basis tornado event. This last item is remote from the steam and feedwater lines (being located on the turbine ground floor) and has been evaluated to have adequate protection from tornado missiles and internal flooding (see Section 3.6.1.15.1). The RPS and MSIV isolation logic sensor inputs due to their nature cannot be made immune from pipe break effects. However, loss of this equipment during a postulated event would not result in loss of capability to bring the plant to a cold shutdown or mitigate the radiological consequences of such an incident even assuming a single failure among the safety systems that remain unaffected.

A pipe break in a main steam or feedwater line in the turbine building or steam tunnel extension could result in transitory pressurization of the corridors between the turbine building, reactor building, radwaste control building, and the diesel generator building. Air and steam would be forced into these corridors through openings in the south wall of the turbine-generator building, and through the seismic gap between the turbine building, reactor building, and radwaste control building. The large volume of the turbine building and because the turbine building metal siding and exterior doors are not leaktight and are not designed to withstand more than a minimal pressure differential, would result in the peak pressures seen by the reinforced-concrete walls of the reactor building and radwaste control building not exceeding the structural capacity of the walls. The doors to the control room are low-range blast doors designed to withstand a pressure differential of 3 lb/in.^{2}, which is considered adequate to maintain control room habitability as discussed in Section 3.6.1.12.

3.6.1.21 Description of Methods of Analyses to Ensure That Primary or Secondary Containment Integrity Is Not Compromised by a Postulated Passive Component Failure

The previous 20 sections present the results of analyses that indicate that postulated piping failures do not adversely affect safe reactor operation. Implicit in these analyses is the necessity of conforming to relevant standards with regard to offsite radiological consequences.

3.6.2 DETERMINATION OF BREAK LOCATIONS FOR DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Information concerning postulated break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping inside and outside of primary containment is presented in this section. The information presented in this section and in Section 3.6.1 confirms that the requirements for the protection of structures, systems, and components relied on for safe reactor shutdown or to mitigate the consequences of a postulated pipe break have been met.

3.6.2.1 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 in high-energy and moderate-energy piping systems both inside and outside of primary containment.

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

a. Maximum temperature exceeds 200°F
b. Maximum pressure exceeds 275 psig.

Moderate energy fluid systems are defined as those systems, or portions of systems, that during normal plant conditions are pressurized under both of the following conditions:

a. Maximum temperature is 200°F or less, and
b. 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 periods in performing their system function. For the major operational period they 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 approximately 2% of the time period that the system operates as a moderate-energy fluid system.

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 crack (longitudinal split). These are postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe rupture is confined to postulation of leakage cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only and do not cause jet impingement or uncontrolled whipping of the pipe.

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

Postulated pipe break locations are selected as described herein and are based on the guidelines provided in Regulatory Guide 1.46, Revision 0; the BTP ASB 3-1, and as expanded in BTP MEB 3-1 for piping inside and outside primary containment.

* Normal plant conditions are defined as the plant operating conditions during reactor startup, operation at power, hot standby, or reactor cool down to cold shutdown, but excluding test modes.

3.6.2.1.1 Postulated Pipe Break Locations in High-Energy Fluid System Piping Not in the Containment Penetration Area

Pipe breaks (not including leakage cracks) are postulated at locations as indicated below:

3.6.2.1.1.1 Postulated Pipe Break Locations in ASME Section III Class I Piping Runs.

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

Intermediate locations of postulated pipe breaks are selected by application of one of the following sets of rules:

a. Pipe break is postulated at each location of significant change in flexibility, such as pipe fittings (elbows, tees, and reducers), and circumferential connections to valves and flanges;

b. Based on stress and fatigue analysis, as calculated according to ASME Code Section III Subarticle NB-3600, no break is postulated if any of the following applies:

1. S_n† does not exceed $2.4S_m$‡,

* Terminal ends are extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of the main run in the stress analysis and is shown to have a significant effect on the main run behavior. In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve), a terminal end of such runs is the piping connection to this closed valve.

† S_n is the primary plus secondary stress intensity range, as calculated by use of Equation (10) of ASME Code Section III Subsection NB, Paragraph NB 3653.1, between any two load sets (including the zero load set) for normal and upset plant conditions, including an OBE event transient.

‡ S_m is the design stress intensity, as described in ASME Code Section III Subsection NB, Paragraph NB 3229.

2. S_n exceeds $2.4S_m$ but does not exceed $3S_m$, and the Cumulative Usage Factor (U)* does not exceed 0.1, and

3. S_n exceeds $3S_m$, but S_e† and S_r‡ are each less than $2.4S_m$, and U does not exceed 0.1.

3.6.2.1.1.2 Postulated Pipe Break Locations in ASME Section III Class 2 and 3 Piping Runs.

The terminal ends of the pressurized portions of the run.

Intermediate locations of postulated pipe breaks are selected by applications of one of the following sets of rules:

a. Pipe break is postulated at each location of significant change in flexibility, such as pipe fittings (elbows, tees, and reducers), and circumferential connections for valves and flanges,

b. At each location here the stresses under the loadings resulting from upset plant conditions, including an operating basis earthquake (OBE) event as calculated by the summation of Equations (9) and (10) of ASME Code Section III Subsection NC, Paragraph NC 3652, exceed 0.8 ($1.2S_h + S_A$) where S_h and S_A), are as defined in Paragraph NC 3611.2, or

c. Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, valves, or flanges.

3.6.2.1.1.3 Break Locations in Other Piping Runs. Postulated pipe break locations for piping other than ASME Code Section III Class 1, 2, and 3, are postulated in accordance with pipe whip criteria which conforms to the criteria set forth for ASME Code Section III Class 2 and 3 piping.

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* U is the Cumulative Usage Factor that indicates the total fatigue damage as calculated by the procedure in ASME Code Section III Subsection NB, Paragraph NB 3653.

† S_e is the nominal value of expansion stress as calculated by use of Equation (12) of ASME Code Section III Subsection NB, Paragraph NB 3653.6(a).

‡ S_r is the range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending and thermal expansion stresses as calculated by use of Equation (13) of ASME Code Section III Subsection NB.

3.6.2.1.2 Postulated Pipe Break Locations in High-Energy Fluid System Piping Between Primary Containment Isolation Valves

Pipe breaks (not including leakage cracks) are postulated in locations as indicated below:

3.6.2.1.2.1 Postulated Pipe Break Locations in ASME Section III Class 1 Piping Between Primary Containment Isolation Valves. No pipe breaks are postulated in the portion of piping between primary containment isolation valves, if any of the following apply:

a. S_n does not exceed $2.4S_m$,

b. S_n exceeds $2.4S_m$ but does not exceed $3S_m$, and the Cumulative Usage Factor (U) does not exceed 0.1, or

c. S_n exceeds $3S_m$, but S_e and S_r are each less than $2.4S_m$, and U does not exceed 0.1.

The stress levels in the ASME Section III Class 1 containment penetration high-energy piping are maintained at or below these limits and therefore, breaks are not postulated.* See Section 3.6.2.1.2.3 for further discussion of containment penetration piping.

Piping systems which may have break exclusion areas between primary containment isolation valves are those determined by examining the list of high-energy piping systems (see Section 3.6.2.1 and Table 3.6-15). Systems which do not pass through primary containment are excluded. In addition, systems which are not pressurized between the isolation valves during normal plant operation (see Table 3.6-15) are excluded. The remaining systems, those which may have break exclusion areas between primary containment isolation valves are listed in Table 3.6-16. Break exclusion areas for these systems are shown in Figures 3.6-85 through 3.6-88.

3.6.2.1.2.2 Postulated Pipe Break Locations in ASME Section III Class 2 and 3 Piping Between Primary Containment Isolation Valves. See Section 3.6.2.1.1.2, for stress criteria applicable to ASME Section III Class 2 and 3 piping between containment isolation valves.

The stress levels are maintained at or below these limits and therefore, breaks are not postulated. See Section 3.6.2.1.2.3 for further discussion of containment penetration piping.

* A program for augmented inservice inspection for high energy line breaks is included in the Inservice Inspection Program Plan.

3.6.2.1.2.3 Primary Containment Penetration Piping. To maintain containment integrity, primary containment penetrations are designed with the following characteristics:

a. They are capable of withstanding the forces caused by impingement of the fluid from the rupture of the largest local pipe without failure, and

b. They are capable of withstanding the maximum reactions that the pipes to which they are attached are capable of exerting.

Piping and electrical penetration details are discussed and shown in Section 3.8.6.

The stress criteria for postulating breaks in containment penetration piping between isolation valves is given in Sections 3.6.2.1.2.1 and 3.6.2.1.2.2.

Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses or tests are performed to demonstrate compliance with the limits of Section 3.6.2.1.2. In addition, the number of circumferential and longitudinal piping welds and branch connections are minimized.

Any pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) are designed such that they are not welded directly to the outer surface of the piping except where such welds are 100% volumetrically examinable while in service, and a detailed stress analysis is performed to demonstrate compliance with the limits of Section 3.6.2.1.2.

Tunnel structures surrounding the primary containment penetration piping are designed for the thermal and pressure loads of a through wall leakage crack regardless of crack postulation requirements. See Section 3.6.1.20 for further discussion.

Access for ISI of welds in high-energy (hot type) containment penetration assemblies is described in Section 3.8.6.1.1. All required ISI locations are accessible.

3.6.2.1.3 Postulated Through Wall Leakage Crack Locations in High- and Moderate-Energy Fluid Systems

Cracks in high energy systems, regardless of ASME Code classification, were evaluated at Columbia Generating Station both inside and outside of containment. Inside containment it was found that the circumferential or longitudinal high energy line breaks bounded the structural and environmental consequences produced by a leakage crack at any location in a high energy line. Outside containment analyses were completed or protection was provided to ensure that critical plant systems are not harmed by leakage cracks in adjacent high energy systems.

In moderate-energy piping systems consisting of ASME Code Section III Class 2 and 3 piping and moderate-energy nonnuclear piping, including fluid system piping between primary containment isolation valves, cracks are not postulated provided the stress range of 0.4 ($1.2S_h$[*] + S_A[†]) is not exceeded for the load combination which includes the effects of pressure, weight, other sustained loads, and occasional loads such as the OBE and thermal expansion loads. Since all piping in structures housing safety-related systems are supported and controlled as Seismic Category I systems regardless of service, the criteria for postulated cracks is the same as above for all systems.

3.6.2.1.4 Types of Breaks and Cracks Postulated in High-Energy and Moderate-Energy Fluid System Piping

3.6.2.1.4.1 Breaks in High-Energy Fluid System Piping. The following types of breaks are postulated in high-energy fluid system piping:

No breaks need be postulated in piping having a nominal diameter less than, or equal to 1 in.

Circumferential breaks are postulated only in piping exceeding a 1 in. nominal pipe diameter.

Longitudinal splits are postulated only in piping having a nominal pipe diameter equal to or greater than 4 in.

Longitudinal splits are not postulated at terminal ends.

At each of the postulated break locations, consideration is given to the occurrence of either a longitudinal split or circumferential break. Both types of breaks are considered if the maximum stress ranges in the circumferential and axial directions are not significantly different. Only one type break is considered as follows:

a. If the result of a detailed stress analysis indicates that the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, only a circumferential break is postulated. Where usage factor is a determinant in establishing a postulated break location, the fatigue dominant stresses are

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[*] S_h is the allowable stress at maximum (hot) temperatures defined in ASME Code Section III, Article NC 3611.2.

[†] S_A is the allowable stress range for thermal expansion, as defined in ASME Code Section III, Article NC 3611.2.

examined as indicated above to determine whether longitudinal, circumferential, or both are postulated; or

b. If this type of analysis indicates that the maximum stress range in the circumferential direction is at least 1.5 times that in the axial direction, only a longitudinal split is postulated.

Where break locations are selected without the benefit of stress calculations, circumferential breaks are postulated at the piping welds to each fitting, valve, or welded attachment.

For a longitudinal split, the break area is assumed to be equal to cross-sectional flow area of the pipe.

For circumferential breaks, pipe whipping is assumed to occur in the plane defined by the piping configuration and is assumed to cause pipe movement in the direction of the jet reaction.

A longitudinal break is assumed to result in an axial split without severance and to be oriented at any point about the circumference of the pipe or alternatively at the point(s) of highest stress as indicated by a detailed stress analysis. If a postulated break location is at a nonaxisymmetric fitting, such as a tee or elbow, the split is assumed to be oriented (but not concurrently) on each side of the fitting at its center, perpendicular to the plane of the fitting and is assumed to cause pipe movement in the direction of the jet reaction.

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.
A circumferential break is assumed to result in pipe severance with full separation amounting to at least a one-pipe-diameter lateral displacement of the ruptured piping sections, except as limited by structural design features. The break is assumed to be oriented perpendicular to the longitudinal axis of the pipe. Line restrictions, flow limiters, and the absence of energy reservoirs are accounted for in the calculation of the design jet discharge.

3.6.2.1.4.2 <u>Cracks in High-Energy and Moderate-Energy Fluid System Piping</u>. The following controlled through wall leakage cracks are postulated in high-energy and moderate-energy fluid systems (or portion of systems):

Cracks are postulated in fluid systems or portions of systems whose size exceeds a nominal pipe diameter of 1 in.

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

The flow from the postulated crack is assumed to result in an environment that wets all unprotected components within the compartment, with subsequent flooding in the compartment and communicative compartments. Flooding effects are determined on the basis of a conservatively estimated time period required to affect corrective action.

3.6.2.1.5 Protection Criteria for the Effects of Pipe Break

Protection from the effects of a whipping pipe due to a pipe break is provided where necessary. Protection from pipe whip need not be provided if any one of the following conditions exists: The piping is classified as moderate energy piping. Following a single postulated pipe break, piping for which the unrestrained movement of either end of the ruptured pipe, in the direction of the jet reaction about a plastic hinge, formed within the piping, cannot impact any structure, system, or component important to safety. Piping for which the internal energy level associated with the 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 will be considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and of equal or heavier wall thickness.

For further discussion of pipe whip protection, see Section 3.6.2.3.3.

Protection of essential systems from the effects of jet impingement is provided where necessary to ensure reactor shutdown to a safe cold condition and to limit the release of radioactivity to within 10 CFR Part 100 limits. For further discussion of criteria for protection against jet impingement, see Section 3.6.2.3.2.

3.6.2.2 <u>Analytic Methods to Define Blowdown Forcing Functions and Response Models</u>

3.6.2.2.1 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 on 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.

A rise time, not exceeding 1 msec, is assumed for the initial pulse of the fluid blowdown forcing function, unless longer crack propagation times or rupture opening times are substantiated by experimental data or analytical theory.

Blowdown forcing functions are determined by either of two general methods given below:

a. 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 References 3.6-3 and 3.6-22. These may be simply described as follows:

   1. The transient forcing functions occur at points along the pipe from the propagation of waves (wave thrust) along the pipe and at the broken end from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust);

   2. 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 and the pressure vessel end until a steady flow condition is established. Free-space and vessel conditions are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the plane of the pipe break, reaching a final steady state value;

   3. 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);

   4. Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the flowdown force on the pipe using the following equation:

$$F = \{(P - P_a) + \frac{\rho u^2}{g}\}A$$

where

F = blowdown force

P = pressure at exit plane

P_a = ambient pressure

u = velocity at exit plane

ρ = density at exit plane

A = area of break

g = gravitational constant

5. Following the transient period, a steady-state period is assumed to exist. Steady-state blowdown forces are calculated, considering frictional effects. For saturated steam, these effects reduce the blowdown forces from the theoretical maximum of 1.26 P_oA. The method of accounting for these effects is presented in References 3.6-3 and 3.6-22. For subcooled water, a reduction from the theoretical maximum of 2.0 P_oA is found through the use of Bernoulli's and other standard equations, such as Darcy's equation, which account for friction.

b. The following is an alternative method for calculating blowdown forcing functions.

Computer code simulation either by RELAP3 (Reference 3.6-9) or GOTHIC (References 3.6-25 or 3.6-26) is used to obtain exit plane thermodynamic states for postulated ruptures (see Section 3.12.11 for further discussion of the software). Specifically, the software calculates exit pressure, specific volume, and mass rate. From these data the blowdown reaction load is calculated using the following relation:

$$\frac{T}{A_E} = (P_E - P_\infty) + \frac{G_E{}^2 \overline{V}_E}{g_C}$$

$$R = -\frac{T}{A_E} \text{ x } A_E$$

where

$\frac{T}{A_E}$ = thrust per unit break area

P_E = exit pressure

P_∞ = receiver pressure

G_E = exit mass flux

$\overline{V}_E$ = exit specific volume

g_c = gravitational constant

R = reaction force on the pipe

3.6.2.2.2 Analytical Methods to Define Response Models

3.6.2.2.2.1 General Description of Analytical Methods. 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 the design of piping systems and in the evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads are given in Section 3.6.2.2.1. The analytical methods used to account for this loading are discussed below.

3.6.2.2.2.2 Dynamic Analysis of the Effects of Pipe Rupture.

Criteria

a. Analysis is performed for each postulated pipe break;

b. The analysis includes the dynamic response of all components of the system including the pipe, pipe whip restraints, and all structures required to transmit loading to foundation. The structures are analyzed for a suddenly applied force in conjunction with impact and rebound effects due to gaps between piping and pipe whip restraints;

c. The analytical model adequately represents the mass/inertia and stiffness properties of the system;

d. 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;

e. Piping contained 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 similar to pipe whip material (see Section 3.6.2.2.3.2) restraint. Piping systems are designed so that plastic

instability would not occur in the pipe at the design dynamic and static loads, unless damage studies are performed which show that the consequences could not result in the direct damage of any essential system or component; and

f. 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 accidents, are not designed to meet ASME Code requirements for essential components under faulted loading. However, if these components are required for safe shutdown or if they serve a safety function to protect the structural integrity of an essential component, then these components are designed to code limits for faulted conditions and to the limits necessary to ensure operability.

Analytical Models

a. Lumped-Parameter Analysis Model: Lumped mass points are interconnected by springs to take into account for the effects of inertia and stiffness inherent in the system, and time histories of the responses are computed by numerical integration to account for gaps and inelastic effects. This analytical method is discussed in detail in Reference 3.6-4;

b. Energy-Balance Analysis Model: Kinetic energy, generated during the first quarter cycle movement of the ruptured pipe as imparted to the piping/restraint system through impact, is converted into equivalent strain energy. Deformations of the pipe and the restraint are compatible with the level of absorbed energy; and

c. Pipe whip restraints for the reactor recirculation system are designed by the NSSS supplier. The analytical method utilized for this design is the computer program PDA which is described in Reference 3.6-4 and further discussed in Section 3.12.33.

   Pipe whip restraints for all other piping systems requiring such protection are designed by the architect/engineer; the method described below is used for this pipe whip restraint design.

Simplified Dynamic Analysis

a. To simplify dynamic analysis the following conservative assumptions are used:

   1. The entire structure including pipe, restraint linkage, support beams, and major structure to foundation connections absorb energy by elastic, elastoplastic, or plastic deformation. To provide a simplified dynamic

mathematical model, one member is generally considered to absorb all the energy. This member is classified as an energy absorbing member and is designed as described in Section 3.6.2.3.3.2. The other components of the structure are assumed as infinitely rigid. These are classified as load transmitting members and are designed as described in Section 3.6.2.3.3.2;

2. The time history of unbalanced forces on the ruptured pipe is reduced to a suddenly applied constantly maintained force, which envelopes the actual force at any particular time, or is based on a more sophisticated blowdown calculation; and

3. Dynamic loading on the pipe whip restraint is reduced to a suddenly applied, constantly maintained force described above, in conjunction with a kinetic energy of impact.

b. Simplified analytical models such as simple beams, structural frames, and ring girders on assumed rigid supports are modeled as single springs. One example of such a typical analytical model is shown in Figure 3.6-89. For these, the required member resistance (R_r) is determined by application of the formula shown in Figure 3.6-90. This derived equation is based on References 3.6-1 and 3.6-2. The following is a description, and discussion of the parameters used in this derivation:

1. The term (R_r) represents the required member resistance (strength) when loaded by a suddenly applied constantly maintained force (F_1), in conjunction with a kinetic energy of impact (K), due to collision of a moving body (i.e., ruptured pipe) with the member in question;

2. The impulse (i) is represented by the area under any load time history having a time of duration (t_d), which is small compared with the natural period of the impacted member;

3. The kinetic energy is represented by $i^2/2M$, where (i) is the impulse and (M) is the mass of the moving body;

4. The kinetic energy of impact (K) is also represented by the product of the Force (F_1) which accelerates the ruptured pipe, and a distance (d) the total distance traveled by the moving body from time zero to time of collision with the member in question. Note that when the resisting member is in direct contact with the ruptured pipe, the distance (d) is zero and the kinetic energy (K) reduces to zero. Likewise, when no

resisting member is required, the ruptured pipe does not collide with anything and therefore no kinetic energy of impact exists. In these events the equation shown in Figure 3.6-90 is applicable with (K) equal to zero;

5. The energy absorbing member is permitted to deform into the plastic region. Thus the member resistance is bilinear. (Y_e) is the deflection of the member at the end of elasticity of the member. (Y_m) is the maximum deflection of the member;

6. The elastic spring constant (k) is the ratio of load on the member divided by the deflection due to this load, where the deflection is equal to or less than the value (Y_e) and the load is compatible with this concept. Thus (k) can be expressed as (R_r/Y_e);

7. For inelastic response the maximum deflection (Y_m) is always larger than the elastic deflection (Y_e). For this case, the ratio (Y_m/Y_e) is defined as the ductility ratio (μ);

8. The maximum deformation of the energy absorbing member is controlled by limiting the ductility ratio (μ).

   Reference 3.6-6 provides the ductility ratio that corresponds to collapse (μ_c). For structural steel members, these values vary with upper limits in the order of 20 to 30 and up (for very ductile structures). For Columbia Generating Station, the maximum permissible ductility ratio is limited to 50% of (μ_c), except that energy absorbing members in direct contact with primary containment are limited to 5% of (μ_c). For Columbia Generating Station, only steel members are utilized as energy absorbing members, as defined in Section 3.6.2.3.3.2. The maximum values of (μ_c), for various structural components, are given in Table 3.6-7; and

9. The equation derived in Figure 3.6-90 accounts for a suddenly applied constantly maintained force in conjunction with a kinetic energy of impact on the resisting member. Total transfer of energy is implied. This is combined with the constantly maintained force (from ruptured piping blowdown) on the restraint structure. This assumption is consistent with a zero coefficient of restitution (full plasticity), and is a conservative assumption.

With regard to rebound, it should be noted that if a coefficient of restitution of unity is assumed (full rebound), there is zero kinetic energy transfer to the restraint structure.

If a coefficient of restitution less than unity is assumed (partial rebound), there is a partial amount of kinetic energy transfer to the restraint structure.

A coefficient of restitution of zero conservatively assumed in the application of the equation mentioned above gives zero rebound with 100% kinetic energy transfer to the restraint structure.

It should also be noted that the assumption of a suddenly applied constantly maintained force as used in the equation mentioned above is conservative with respect to rebound. Rebound implies a finite time of short duration contact with the restraint structure, in contrast to the infinite time assumed.

c. Actual structural resistance for the above structures is determined by methods of limit analysis using a dynamic yield strength, as defined in Section 3.6.2.2.3.1.

3.6.2.2.3 Material Properties Under Dynamic Loads

3.6.2.2.3.1 Dynamic Yield Strength. To account for the rapid strain rate effects, dynamic yield strength is utilized. This phenomenon is documented in References 3.6-6 and 3.6-7. Material tests have shown a consistent increase in yield strength under rapid loading. Under rapid strain rate, carbon steel yield strength consistently improves by more than 40%. High strength alloy steel displays a somewhat smaller improvement. For Columbia Generating Station, a conservative dynamic yield strength of 110% of minimum static yield strength, at the specified operating temperature, is used.

3.6.2.2.3.2 Maximum Strain of Tension Members. Pure tension members, such as U-bars shown in Figure 3.6-21 which act to limit pipe whip are permitted to deform during energy absorption: (a) a maximum of 50% of the minimum uniform strain (at the maximum stress on an engineering stress-strain curve) based on actual restraint material tests or (b) one-half of minimum percent elongation as specified in the applicable ASME Code Section II or ASTM Specifications if demonstrated to be less than 50% of the minimum uniform strain based on representative test results.

The dynamic tensile and impact properties are specified to be not less than (a) 70% of the static percent elongation or (b) 80% of the statically determined minimum total energy absorption.

3.6.2.2.3.3 Maximum Deformation of Flexural Members. Deformations of energy absorbing flexural support members are generally limited to 50% of that deformation which corresponds to structural collapse, except that deformation of energy absorbing members in direct contact with the primary containment vessel is limited to 5% of that deformation which corresponds to structural collapse.

3.6.2.2.3.4 Materials and Proportions of Structural Shapes. The materials and proportions of structural shapes for energy absorbing members are in accordance with recommendations for dynamic member design as documented in References 3.6-6 and 3.6-7.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Dynamic Analysis Methods for Jet Impingement Effects

The procedure for analyzing the dynamic effects of jet impingement has been extracted from Reference 3.6-8.

Computer simulation using either RELAP3 (Reference 3.6-9) or GOTHIC (References 3.6-25 and 3.6-26) with required geometric input data is run to obtain the exit plane thermodynamic state and mass flow rate. Specifically, the quantities output the computer simulations are

M	= mass flow rate	lb_m/sec
$\overline{V}_E$	= specific volume	ft^3/lb_m
P_E	= exit pressure	lb_f/ft^2

The reaction load on the piping system is given by

$$\frac{T}{A_E} = (P_E - P_\infty) + \frac{G_E{}^2 \overline{V}_E}{g_c} \qquad (3.6\text{-}25)$$

where

$\frac{T}{A_E}$	= thrust per unit area	lb_f/ft^2
P_∞	= receiver pressure	lb_f/ft^2
G_E	= mass flux	$lb_m/sec\ ft^2$

P_E	= exit pressure	lb_f/ft^2
$\overline{V}_E$	= specific volume at exit	ft^3/lb
g_c	= Newton's constant	$32.174 \frac{ft - lb_m}{lb_f \sec^2}$

(T/A_E) yields the thrust reaction load on the piping system subject to the assumption that the vapor to liquid velocity ratio is unity. By conservation of forward momentum, the jet force per unit area (F_j/A_E) equals the thrust force per unit area (T/A_E). To determine the effect of the fluid jet on targets located some distance LT from the break, the following procedure is used.

Classify target distance L_T as less than, equal to, or greater than the distance required for full jet expansion L_∞. See Figure 3.6-91.

where

$$L_\infty = \frac{D_E}{2}\left[\left(\frac{A_\infty}{A_E}\right)^{1/2} - 1\right] \tag{3.6-26}$$

$$\frac{A_\infty}{A_E} = \left[\frac{G_E^{\,2}}{g_c}\right]\left[\frac{\bar{v}_\infty}{T / A_E}\right] \tag{3.6-27}$$

where

A_∞	= area of jet at full expansion	ft^2
A_E	= area of exit	ft^2
D_E	= diameter of exit	ft
G_E	= mass flux	$lb_m/sec\ ft^2$
g_c	= Newton's Constant	$32.174 \frac{ft - lb_m}{lb_f \sec^2}$
h_E	= exit plant enthalpy	Btu/lb
L_∞	= distance to full expansion	ft

P_∞ = receiver pressure lbf/ft^2

T = thrust

$\overline{V}_\infty$ = specific volume at full expansion

Assuming the kinetic energy of the fully expanded jet to be insignificant, the following relation holds between vessel stagnation enthalpy and fully expanded jet enthalpy:

$$h_o = h_\infty \tag{3.6-28}$$

Therefore,

$$x_\infty = \frac{h_o - h_{f_\infty}}{h_{fg_\infty}} \tag{3.6-29}$$

where

h_o = stagnation enthalpy at full expansion, Btu/lb

h_{f_∞} = liquid enthalpy at full expansion, Btu/lb

h_{fg_∞} = vaporization enthalpy at full expansion, Btu/lb

X_∞ = quality at full expansion

then,

$$\overline{V}_\infty = \overline{V}_{f_\infty} + X_\infty \overline{V}_{fg_\infty} \tag{3.6-30}$$

where

$\overline{V}_{f_\infty}$ = specific volume of liquid at full expansion, ft^3/lbm

$\overline{V}_{fg_\infty}$ = specific volume of vaporization at full expansion ft^3/lbm

$\overline{V}_\infty$ = specific volume at full expansion, ft^3/lbm

From Equations 3.6-29 and 3.6-30, subject to the assumption of Equation 3.6-28, Equations 3.6-26 and 3.6-27 can be solved for L_∞.

For $L_T < L_\infty$ property variations are assumed linear from A_E to A_∞ for area and from P_E to P_∞ for pressure.

The jet impingement load F_j is as follows with F_j a constant:

$$F_{jT} = F_j \text{ x } \frac{A_T}{A_L} \tag{3.6-31}$$

where

A_T = area of target which is intercepted by the jet

A_L = jet area at the target distance calculated as

in Region 1: for $O < L < L_\infty$

$$A_L = \left(A_\infty - A_E\right)\frac{L_T}{L_\infty} + A_E \tag{3.6-32}$$

in Region 2: for $L_\infty \le L \le 1/2\left[\sqrt{\frac{4A_\infty}{\pi}} - D_E\right] \cot\ 10^\circ$

$A_L = A_\infty$

where

$$L_3 = 1/2\left[\left(\frac{4A_\infty}{\pi}\right)^{1/2} - D_E\right] \cot\ 10^\circ \tag{3.6-33}$$

in Region 3: for $1/2\left[\sqrt{\frac{4A_\infty}{\pi}} - D_E\right] \cot\ 10^\circ < L < \infty$

$$A_L = A_E\left(1 + \frac{2_L}{D_E} \tan\ 10^\circ\right)^2 \tag{3.6-34}$$

for nonflashing/nonexpanding fluids:

$$A_L = A_E \tag{3.6-35}$$

3.6.2.3.2 Jet Impingement Effect

3.6.2.3.2.1 Physical Separation. The physical separation of different essential systems and components is used to ensure that the plant retains function of sufficient essential systems to ensure safe shutdown in the event of a postulated LOCA and subsequent generation of a jet stream together with an additional single random active component failure and the loss of offsite power.

Where physical separation cannot be used to protect systems, a detailed analysis is performed to determine the effects of jet impingement on their operability. If necessary, barriers are provided to protect structures, systems, and components required for a safe shutdown to prevent offsite radiological consequences and to mitigate the effects of a LOCA.

3.6.2.3.2.2 Jet Impingement Evaluation. The evaluation of the adequacy of physical separation included the inspection of all essential systems and their components that are necessary to start, operate, and control the essential systems required for safe shutdown. The evaluation included the following:

a. Review pipe break locations to provide conservative jet stream orientation and geometry,

b. Review effected equipment by both design drawing examination and plant walkdown,

c. Review signals that result in the actuation of essential systems,

d. Review signals that are necessary to be returned to inside primary containment, to activate the shutdown systems,

e. Review availability of power that is required inside primary containment to operate the essential systems, and

f. Review mechanical engineered safety systems required for safe shutdown.

3.6.2.3.2.3 Postulated Pipe Rupture Locations Inside Containment. The criteria used to define pipe rupture locations is described in Section 3.6.2.1. Figures 3.6-32 through 3.6-56 show the inside containment break locations resulting from the application of the criteria provided in Section 3.6.2.1. Pipe whip restraint locations are also shown in these figures.

3.6.2.3.2.4 Signals From Primary Containment. For instrumentation located inside primary containment, sufficient redundancy is provided such that all signals necessary to cause actuation of essential systems remain functional. Each system that is required to bring the plant to a safe shutdown condition is furnished with two or more sets of redundant instrumentation lines.

In this review, it is conservatively assumed that a jet stream or whipping pipe may damage one of these sets. The redundant system is shown to remain operational by physical separation and barriers, such as the RPV. An example of the above is the location of sets A and B instrumentation lines for the HPCS. Set A and its redundant set B are located at opposite sides of the RPV. Therefore, a jet stream or whipping pipe cannot damage both sets of instrumentation. Function of instrumentation inside primary containment necessary to result in the actuation of the HPCS system is thereby ensured. These conditions as discussed for the HPCS instrumentation lines are typical for all instrumentation lines that support essential systems. The capabilities of redundant instrumentation is discussed in Section 7.3.

3.6.2.3.2.5 Signals to the Primary Containment. No instrumentation signal is necessary to return inside primary containment to operate any of the essential systems. Signals to the ADS valves are provided through their power supply as described in the following section.

3.6.2.3.2.6 Power Requirement Inside Primary Containment. The only essential system that requires power inside primary containment is the ADS.

There are 18 main steam relief valves inside primary containment, seven of which are in the ADS mode. These seven valves will quickly depressurize the RPV in the event of a small break and allow the LPCS or the LPCI to keep the core covered with water, thus permitting the reactor to be brought to a safe shutdown.

There are two independent and redundant electrical systems (divisions) inside primary containment which operate the ADS valves. The divisions are run separately in main paths of conduits, and individual conduits are branched to corresponding ADS valves.

Each ADS valve is equipped with three solenoids and two divisions as noted above. Power to the ADS valves is required to open a solenoid valve connected to the air accumulator, thus allowing the pressurized air to open the ADS valves.

The physical arrangement of the electrical conduits is such that one division runs next to the primary containment vessel wall and the other division runs around the sacrificial shield wall. The conduits for Division 1 (A solenoids) are routed above the platform at el. 541 ft in a horizontal configuration, running toward penetration X-105C. The conduits for Division 2 (B solenoids) are routed below the platform at el. 522 ft in a horizontal configuration, running toward penetration X-105B. The conduits are routed such as to take advantage of the piping

and structure for protection against a possible jet impingement. The ADS supply lines are discussed in Section 7.3.

Six out of the seven furnished ADS valves have to be operational to enable adequate depressurization of the RPV. Jet impingement cannot incapacitate more than three electrical lines. Since four lines would have to be incapacitated to incapacitate more than one valve, safety is ensured.

This condition is typical for all conduits leading to the ADS valves.

3.6.2.3.2.7 Mechanical Engineered Safety Systems. Under conditions involving a LOCA, a loss of offsite power, and the failure of any one diesel generator, the availability of redundant essential systems is determined by study of design drawings and by field walkdown of the physical location of these systems to verify that sufficient separation is provided between two redundant systems. Specifically, two situations are investigated:

a. The possibility of a jet from an essential system destroying or in any way damaging its redundant system. For example, in the case of failure of Division 2 diesel generator, the possibility of a pipe whip or jet from the HPCS damaging the LPCS and ADS is investigated; and

b. The possibility of a jet, from a high energy line, being capable of damaging an essential system and its redundant system. For example, in the event of Division 2 diesel generator failing, in conjunction with a possible jet or pipe whip from the RCIC, which may be capable of damaging the LPCS, ADS, HPCS, or related systems, thus preventing the plant from being brought to a safe shutdown is investigated.

All essential systems are examined to ensure that they will be capable of performing their required function after a jet impingement.

3.6.2.3.2.8 Jet Impingement on Major Structures Inside Primary Containment. Jet impingement loading on the steel primary containment vessel is discussed in Section 3.8.2. Jet impingement loading on the concrete and steel internal structures of the steel primary containment vessel is discussed in Section 3.8.3. Jet impingement loading on primary containment penetrations is discussed in Section 3.8.6.

3.6.2.3.3 Pipe Whip Restraints

3.6.2.3.3.1 Definition of Function. 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 depend on the pipe whip restraints for any loading combination. If the piping integrity is compromised by a

pipe 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) will be subject to a once in a lifetime loading. The pipe break event is considered to be a faulted condition for the piping system, its restraints, and structure to which the restraint is attached. The design and analysis of these components for this event are described later in this section and in Section 3.6.2.2. Piping is no longer considered to be a part of the RCPB following the break. Plastic deformation of the pipe is considered as a potential energy absorber. Piping systems are designed so that plastic instability would not occur in the pipe under design dynamic and static loads if the consequences of such instability could result in the loss of the primary containment integrity or loss of required plant shutdown capability.

3.6.2.3.3.2 Pipe Whip Restraint Features. The restraints are close to the pipe to minimize the kinetic energy of impact and yet are sufficiently removed from the pipe to permit unrestricted thermal pipe movement.

To facilitate ISI of piping, the restraints are generally located a suitable distance away from all circumferential welds and are of bolted construction so as to be removable.

Pipe whip restraint structures fall into one of the following two categories:

a. Energy absorbing members: These are modeled as elastic, elastoplastic, or plastic springs in a dynamic analysis. The required resistant (strength) of these structures is derived by application of the principles of structural dynamics; or

b. Load transmitting members: These are relatively stiff components and are modeled as rigid members in the dynamic analysis. Their function is to transmit loading from the source to foundation. The load due to the postulated pipe rupture is in the form of an equivalent static load and is derived as a result of the dynamic analysis performed for the energy absorbing members.

Energy absorbing members are ductile structures such as simple beams, frames, and ring girders (including the piping system itself) having the capability to deflect significantly in absorbing the energy imparted to them by a postulated broken pipe. For loading conditions including the effects of postulated pipe rupture, these members are designed within the limits for inelastic systems as stated in Table F1322.2-1 of ASME Boiler and Pressure Vessel (B&PV) Code Section III, 1974 Editions, Summer 1974 Addenda, Appendix F, “Rules for Evaluation of Faulted Conditions” adjusted to account for rapid strain rate effects as discussed in Section 3.6.2.2.3. These members are constructed to meet the requirements of Quality Class I structures.

The U-bar straps, as shown in Figure 3.6-21 and described in Section 3.6.2.2.3.2, act as nonlinear energy absorbing, nonrebounding, plastic springs. The U-bar straps are justified by empirical data, as described in Sections 3.6.2.2.2.2 and 3.12.33.

Load transmitting members are rigid components such as clevises, brackets or pins, rigid pipe whip restraint weldments as shown in Figure 3.6-22, or similar components; as well as major structures such as the drywell diaphragm floor, primary containment vessel, reactor pedestal, reactor building and foundation. For loading conditions, including the effects of postulated pipe rupture, the members in Figure 3.6-22 are designed within the limits stated in Table F1322.2-1 of ASME Code Section III, Appendix F, "Rules for Evaluating Faulted Condition" for components and component supports; except that the members beyond those included in the dynamic analytical model (i.e., reactor pedestal, reactor building, as well as certain steel members assumed to be infinitely rigid) are designed to AISC, ACI, and other appropriate structural component criteria. All these members are constructed to the requirements of Quality Class I structures.

The design limits for connecting members such as clevises, brackets, and pins per Figure 3.6-21 are based on the following stress limits:

a. Primary stresses (in accordance with definitions in ASME Section III) are limited to the higher of

   1. 70% of Su, where Su = minimum ultimate strength by tests or ASTM specification, and

   2. Sy + 1/3 (Su - Sy), where Sy = minimum yield strength by test or ASTM specification; or

b. Recommended stress limits in accordance with ASME Code Section III, Subsection NF, for faulted conditions, if applicable. The design limits for welds of connecting members to steel structures are based on the following stress limits: the maximum primary weld stress intensity (two times shear stress) is limited to three times AWS or AISC building allowable weld shear stress.

The recirculation pump discharge and suction piping utilizes the U-bar strap pipe whip restraints (Figure 3.6-21), while all other systems listed in Table 3.6-15 use rigid types as shown in Figure 3.6-22 or similar configurations.

Typical installations of pipe whip restraints are shown in Figures 3.6-23, 3.6-24, and 3.6-92 through 3.6-95.

3.6.2.3.3.3 <u>Pipe Whip Restraint Loading</u>. For the purpose of predicting the pipe rupture forces associated with the reactor blowdown, the local line pressures are assumed to be those

normally associated with the reactor operating at ≥ 105 % of rated power and with a vessel dome pressure of 1040 psig.

In calculating pipe reaction, credit is taken for any line restriction and line friction between the break and the pressure reservoir. The following represent typical restrictions to flow which are specifically considered:

a. Jet pump nozzles,
b. Core spray nozzles (inside internals shroud),
c. Feedwater sparger, and
d. Steam line flow limiter.

The hydraulic bases and calculational techniques for predicting unbalanced forces on a pipe associated with a postulated instantaneous pipe rupture are as discussed in Section 3.6.2.

The dynamic loading on the pipe whip restraint commences at the effective time of impact of the pipe with the restraint. It includes the following:

a. Unbalanced force on the pipe associated with a postulated instantaneous pipe rupture in the form of a suddenly applied force; and

b. Dynamic inertia load of the moving section of pipe which is accelerated by the unbalanced force associated with the pipe rupture and collides with the restraint. This load is in the form of kinetic energy of impact.

3.6.2.3.4 Pipe Whip Effects on Safety-Related Components

Pipe whip (displacement) effects on safety-related structures, systems, and components can be placed in two categories: (a) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run in which the break occurred and (b) controlled pipe whip displacements as they apply to external components such as building structure, other piping systems, cable trays, and conduits.

3.6.2.3.4.1 Pipe Displacement Effects on Components in Same Piping Run. The criteria which is used for determining the effects of pipe displacements on inline components are as follows:

a. 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 requirements for essential components under faulted loading, and

b. If these components are required for safe shutdown, or serve a safety function to protect the structural integrity of an essential component, the code requirements for faulted conditions and limits to ensure operability, if required, are met.

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.

3.6.2.3.4.2 Pipe Displacement Effects on Structures, Other Systems, and Components. The criteria which are used for determining the effects of pipe displacements on structures, other systems, and components are as follows:

a. Systems and components which 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 requirements for essential components under faulted loading;

b. Systems and components that serve a safety function but are not required to mitigate the consequences of the postulated accident including consideration of random active component failure, are not required to be designed to meet the ASME Code Section III requirements for essential components under faulted conditions;

c. Systems and components, which do serve a safety function and which are required to mitigate the consequences of the particular postulated accident, are located such that the displaced pipe will not come in contact with these systems and components. In areas where this is not feasible, and contact is possible, the components are designed to meet the ASME Code requirements for faulted conditions; and

d. Pipe whip effects on structures are discussed in Section 3.6.2.3.3.2. Structural deflections resulting from pipe whip are covered in Section 3.6.2.2.2.2c.

The methods used to calculate the pipe whip loads are described in Section 3.6.2.2.2.

3.6.2.4 Guard Pipe Assembly Design Criteria for Dual Barrier Containment

The containment structure does not utilize dual barriers, guard pipes, or other protective devices that limit the pressurization of the space between the two barriers. The use of guard pipes is restricted to type 1 penetration assemblies (see Figure 3.8-54) as described in Section 3.8.6.1.1.

3.6.2.5 Implementation of Criteria for Pipe Whip and Jet Impingement Protection

The effects of jet impingement are discussed in Section 3.6.2.3.2. The implementation of criteria for pipe whip protection is discussed in the following.

3.6.2.5.1 Piping Systems Outside Primary Containment

Studies are performed as described in Section 3.6.1 to ensure that in the event of a postulated pipe rupture, sufficient equipment remains functional to mitigate the consequences of the particular pipe rupture, including considerations of failure of a single random active component. The study indicates that pipe whip supports are required within the main steam tunnel and to protect normally open isolation valves that are required to close. Pipe whip protection requirements in the main steam tunnel are discussed in Section 3.6.2.5.4.11.

Pipe whip protection requirements for isolation valves are discussed in Section 3.6.2.5.3.6.

3.6.2.5.2 Piping Systems Inside Primary Containment

High-energy piping systems inside primary containment subject to postulated pipe rupture are tabulated in Table 3.6-15. Specific criteria for determination of break locations and dynamic effects associated with the postulated rupture of piping are discussed in Section 3.6.2.1. The function and features of pipe whip restraints are discussed in Section 3.6.2.3.3. Equipment and system requirements subsequent to a postulated pipe rupture in a fluid system are discussed in Section 3.6.2.5.3.

Pipe whip restraints are furnished when it is necessary to limit pipe movements resulting from a postulated break, which could otherwise cause unacceptable damage to equipment necessary to mitigate the consequences of the particular postulated pipe rupture, including considerations of a single random active failure. For high-energy piping inside primary containment, this includes the following:

a. Assurance of primary containment leaktightness,

b. Assurance that potential for damage is such that the maximum pipe break areas and/or combinations of pipe break areas do not exceed the values described in Section 3.6.2.5.3.2 so that emergency core cooling systems (ECCS) capability is not impaired,

c. Assurance that the CRD system maintains sufficient function to ensure reactor shutdown,

d. Assurance that there is sufficient capability to maintain the reactor in a safe shutdown condition.

The criteria used to define pipe rupture locations for piping systems discussed in Section 3.6.2.5.4 follows Section 3.6.2.1.1.1.

Figures 3.6-32 through 3.6-56 show the piping configurations for each high energy system inside primary containment and include numerical identification of all significant points of interest in the piping system, locations of pipe whip supports, and postulated pipe break locations. The pipe whip supports are identified by the acronym PWS followed by an identification number on Figures 3.6-32 through 3.6-54 and as noted in Figure 3.6-55.

3.6.2.5.3 System Requirements Subsequent to Postulated Pipe Rupture

3.6.2.5.3.1 Control Rod Insertion Capability. To maintain the ability to insert the control rods in the event of a pipe break, no more than one in any array of nine CRD withdrawal lines may be completely crimped (totally blocked). Complete severance of withdrawal lines does not affect the rod insert function. Protection of the CRD insert lines is not required since a reactor pressure of 450 psig or higher can adequately insert the control rods, and no postulated pipe break resulted in severance or total crimping of the CRD insert or withdrawal lines. (See Reference 3.6-23.)

3.6.2.5.3.2 Core Cooling Requirements. The designed ECCS capability can be maintained provided that dynamic effects consequences do not exceed the following break area, break combination, and maintenance of minimum core cooling requirements.

3.6.2.5.3.3 Maximum Allowable Break Areas. For breaks involving recirculation piping, the total effective area of all broken pipes, including the effective area of the recirculation line break, does not exceed the total effective area of the design-basis double-ended recirculation line break. By limiting the total area of all broken pipes involving recirculation loops to an area less than or equal to that of the design basis accident (DBA) (circumferential break of recirculation loop), no accident can be more severe than the DBA.

For breaks not involving recirculation piping, the total effective area of all broken pipes for a given system shall not exceed the total effective area of the double-ended break of the maximum area pipe connected to the reactor boundary for that system.

3.6.2.5.3.4 Break Combinations. In addition to the pipe break area restrictions, breaks involving one recirculation loop do not result in loss of function or damage to the other recirculation loop or loss of coolant from the other loop in excess of that which can result from a break of the attached cleanup connection on the suction side of the loop.

3.6.2.5.3.5 Required Cooling Systems. To ensure compliance with Appendix A of 10 CFR Part 50, General Design Criteria, the cooling system requirements after an additional single active safety system failure are defined in Table 6.3-3. Cases which do not meet the

requirements in Table 6.3-3 must be assessed on an individual basis to determine compliance with core cooling requirements.

3.6.2.5.3.6 Containment System Integrity. The following were considered in addressing the LOCA dynamic effects with respect to containment system integrity.

Leaktightness of the containment fission product barrier is ensured throughout any LOCA.

For those lines which penetrate the containment and are closed during normal operation, the inboard isolation valves are as close as practicable to the RPV. This arrangement reduces the length of pipe subject to a pipe break.

Pipe whip supports are provided in the vicinity of normally open isolation valves inside and outside primary containment for high energy systems to ensure that operability of these valves remains unimpaired during a postulated pipe rupture event.

3.6.2.5.4 System by System Description of Pipe Whip Protection

3.6.2.5.4.1 Main Steam System.

System Arrangement

The main steam system consists of four 26-in. lines which are arranged inside primary containment with mirror image symmetry about the 0° and 180° north-south azimuth. The lines exit the RPV on opposite sides of primary containment and drop down vertically in two parallel pairs to the main steam relief valve platform at el. 541 ft where they are routed horizontally in parallel in the northeast and northwest quadrants to the 0° north azimuth. At this point, the four lines drop vertically in parallel to an elevation just above the diaphragm floor. The MSIVs are located here. The four lines exit the containment nearest the north azimuth at el. 500 ft (approximately). The two feedwater piping loops are described in Section 3.6.2.5.4.2 and are routed near the main steam lines.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the four main steam lines are shown in Figures 3.6-32 through 3.6-35. Where pipe breaks are postulated inside primary containment, the main steam lines are restrained to prevent the unacceptable motion of these pipes. These restraints are mounted on the side of the sacrificial shield wall structure, as well as on radial beams which extend from the sacrificial shield wall to the primary containment vessel wall. A sliding beam seat at the primary containment wall, permits the beam to grow axially and also permits the primary containment wall to move relative to the sacrificial shield wall.

A structural steel frame (see Figure 3.6-57) between the drywell diaphragm floor and the containment vessel, in the area of the MSIVs, is provided for mounting of pipe whip restraints. The structure is designed with vertically sliding connections at the containment vessel to allow for differential thermal expansion between the containment vessel and the diaphragm floor.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam system to ensure safety as defined in Section 3.6.2.5.2. The pipe whip restraints limit the pipe whip motion of the main steam lines to prevent impact and rupture of the adjacent feedwater lines which would otherwise result in a break area in excess of the ECCS capability. Impact with the CRD piping is prevented by pipe whip restraints at the main steam relief valve platform and separation. The CRD piping bundles are routed below the el. 541 ft main steam relief valve platform, a considerable distance away from where the main steam lines drop down to the diaphragm floor.

3.6.2.5.4.2 Reactor Feedwater System (Inside Primary Containment).

System Arrangement

The reactor feedwater system inside primary containment consists of two piping loops symmetrically arranged with respect to 0° and 180° north-south azimuth. The two piping loops emerge from each side of the reactor as three 12-in. vertical risers which drop down and join a header at the main steam relief valve platform. The header is routed parallel to and outside of the main steam lines, increasing in diameter from 12 in. to 18 in. and to 24 in. as it approaches the 0° north azimuth. At this location, the two 24-in. feedwater pipes drop down to 12 ft 6 in. above the diaphragm floor. The pipe is furnished with a check valve in each line in the short horizontal run near the primary containment vessel penetration.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for both reactor feedwater loops are shown in Figures 3.6-36 and 3.6-37. The feedwater lines are restrained to provide protection from the results of all postulated pipe breaks. Specifically, protection is provided where the resulting pipe motion would otherwise impact equipment necessary to mitigate the consequences of the break causing unacceptable damage to that equipment. The restraints are mounted on the side of the sacrificial shield wall, on radial beams at the el. 541 ft. main steam relief valve platform, and on a specially designed structure between the containment and diaphragm floor as shown in Figure 3.6-57. Special features of these structures are described in Section 3.6.2.5.4.1.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the reactor feedwater system to ensure safety as defined in Section 3.6.2.5.2.

In all cases the pipe is sufficiently restrained to prevent impact with containment or impact with other piping systems that would result in violation of pipe break area or pipe break combination limitations. Impact with the CRD piping is prevented by pipe whip restraints at the main steam relief valve platform and separation. The CRD piping bundles are routed below the el. 541 ft. main steam relief valve platform at a considerable distance away from the 0° north azimuth where the 24-in. feedwater lines drop to 12 ft 6 in. above the diaphragm floor. In two cases, portions of the 12-in. vertical risers are restrained in only one direction, allowing impact with one main steam relief valve. This constitutes acceptable damage because depressurization can be accomplished with one valve not functioning. Furthermore, the HPCS is available as a redundant system.

3.6.2.5.4.3 Reactor Water Cleanup System.

System Arrangement

The RWCU system consists of two 4-in. lines which branch from the two reactor recirculation cooling (RRC) pump suction lines located near the 0° and 180° north and south azimuths. The two lines are routed along the diaphragm floor at approximate el. 500 ft. to azimuth 67° where they join into one 6-in. pipe. This 6-in. pipe branches off into two segments. One branch rises to el. 538 ft just below the main steam relief valve platform. It is then routed to azimuth 150° where it exits primary containment. An isolation valve is located inside primary containment near the penetrations. The other 6-in. segment reduces to a 4-in. pipe and then rises to el. 514 ft and terminates at the 2-in. RPV drain system.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the RWCU system are shown in Figure 3.6-38. At all locations where pipe breaks are postulated inside primary containment, the RWCU system is restrained to prevent unacceptable motion of the pipe. Where the pipe is routed along the diaphragm floor, restraints are mounted on special structures built up from the floor. Where the pipe is routed below main steam relief valve platform, restraints are mounted on intermediate structures between radial beams.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RWCU system to ensure safety as defined in Section 3.6.2.5.2. Pipe whip restraints located above the diaphragm floor are designed to prevent impact with the floor, which might impair steam quenching capability of the

suppression pool. Pipe whip restraints located directly below the main steam relief valve platform prevent impact with CRD piping and also primary containment. Equipment necessary to mitigate RWCU pipe breaks, such as ADS system, core spray, LPCI, is protected by separation.

3.6.2.5.4.4 Standby Liquid Control Piping.

System Arrangement

The SLC system includes a section of 1.5-in. piping connected to a short length of 4-in. piping which is connected to HPCS-V-76. The 1.5-in. piping is routed through check valve SLC-V-7 and then downward to penetration X-13. SLC-V-7 limits the high energy portion of the system to approximately 18 in. of piping.

Pipe Whip Protection

The postulated pipe breaks for the SLC system are shown in Figures 3.6-39 and 3.6-48. Pipe whip restraints are not required for this system.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the SLC system to ensure safety as defined in Section 3.6.2.5.2. In the event of a pipe whip resulting from a pipe rupture at any postulated location, the piping system does not damage equipment or systems required for safe shutdown of the reactor. Therefore, pipe restraints are not required for this system.

3.6.2.5.4.5 Residual Heat Removal System - Shutdown Cooling Supply and Return Piping.

System Arrangement

The RHR shutdown cooling supply and return piping consists of two 12-in. piping loops and one 20-in. loop, with all three branching from the RRC piping at the el. 512 ft platform. All three loops are routed primarily in a horizontal plane below the 512 ft platform from the RRC pipe to its primary containment penetration. There is a normally closed valve in each loop located as close as possible to the high energy source, thereby limiting the portion of each loop considered high energy on the basis defined in Section 3.6.2.1.

Pipe Whip Protection

The pipe whip restraints for the RHR shutdown cooling supply and return system are shown in Figures 3.6-43, 3.6-44, and 3.6-45. Where pipe breaks are postulated inside primary containment, the lines are restrained to prevent the unacceptable motion of these pipes. For the two 12-in. shutdown cooling return loops restraints are mounted on intermediate structures

between the radial beams in the el. 512 ft platform. The radial beams extend from the reactor pedestal to the primary containment wall. A sliding beam seat at the primary containment wall permits differential thermal expansion between the containment vessel and reactor pedestal. Restraints for the 20-in. cooling return loop are mounted on a specially designed structure between the diaphragm floor and radial beams in the el. 512 ft platform as shown in Figure 3.6-95.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RHR shutdown cooling supply and return system to ensure safety as defined in Section 3.6.2.5.2.

For the two 12-in. shutdown cooling return loops, pipe whip restraints are provided to prevent impact with primary containment wall and the diaphragm floor. The pipe whip restraints also prevent impact with the CRD piping bundles located above the el. 512 ft. platform. The ECCS system and the ADS systems are protected by separation, being located at higher elevations.

For unrestrained sections of this system, analysis shows a plastic hinge does not develop at the recirculation pipe, and pipe whip does not occur.

For the 20-in. shutdown cooling supply loop, pipe whip restraints are provided to prevent impact with primary containment and the diaphragm floor. Impact with the CRD piping is precluded by a 90° separation from both CRD piping bundles.

3.6.2.5.4.6 Reactor Core Isolation Cooling Reactor Pressure Vessel Head Spray System.

System Arrangement

The RPV head spray system is a 6-in. line that originates at the top of the RPV dome. After a 2-ft vertical riser and a 2-ft horizontal run, there is a normally closed valve that limits the high energy portion of this system to a total length of 4 ft.

Pipe Whip Protection

The postulated pipe breaks for this system are shown in Figure 3.6-46. A pipe whip restraint is provided to prevent RCIC head spray line impact on the containment head as a result of the postulated break locations.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided to ensure safety as defined in Section 3.6.2.5.2. The location of the normally closed valve limits the high energy section of this system as steel

pads welded to the containment vessel head have been provided to distribute the impact load and prevent direct impact with the containment vessel head.

3.6.2.5.4.7 Low-Pressure and High-Pressure Core Spray Piping.

System Arrangement

The LPCS and HPCS are 12-in. piping systems with similar arrangements inside primary containment. They originate at el. 561 ft from the reactor at azimuths 120° and 240° respectively and drop vertically to an elevation just below the main steam relief valve platform where there is an expansion loop in a horizontal plane leading to a penetration through primary containment. In the vertical section, there is a normally closed check valve located as close as possible to the reactor, thereby limiting the portion of piping in both systems considered high energy under the definition given in Section 3.6.2.1.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the LPCS and HPCS systems are shown in Figures 3.6-47 and 3.6-48. Where pipe breaks are postulated inside primary containment the two lines are restrained to prevent the unacceptable motion of these pipes. These restraints are mounted directly onto the sacrificial shield wall.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the LPCS and HPCS systems to ensure safety as defined in Section 3.6.2.5.2. Pipe whip restraints are provided to limit pipe movement resulting from postulated pipe breaks to prevent impact with primary containment and adjacent RHR/LPCI piping. Impact on safety/relief valves (SRVs) resulting from postulated pipe breaks in the HPCS system, is precluded by sufficient separation between these two redundant depressurization methods. The CRD piping bundles are separated by sufficient distance from the high energy sections of the LPCS and HPCS systems.

3.6.2.5.4.8 Residual Heat Removal Condensing Mode and Reactor Core Isolation Cooling Turbine Steam Supply System.

System Arrangement

The RHR condensing mode has been deactivated. All system piping outboard of containment isolation valve RCIC-V-64 has been deactivated and is no longer in use. RCIC-V-64 has been locked closed and the motor operator disconnected. The piping inboard of RCIC-V-64 remains pressurized during plant operation.

The inboard system consists of a 10-in. piping loop which branches off a main steam line at el. 551 ft 2.25 in. and azimuth 105°. An expansion loop in the horizontal plane leads to a penetration through primary containment at el. 550 ft and azimuth 120°.

The RCIC turbine steam supply system consists of a 4-in. line which branches off the 10-in. inboard RHR condensing mode line at approximately azimuth 125° and drops down to el. 532 ft below the main steam relief valve platform. The line is then routed horizontally to a penetration through primary containment at azimuth 35°.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the inboard RHR condensing mode and RCIC turbine steam supply systems are shown in Figure 3.6-49. Where pipe breaks are postulated inside primary containment, this piping is restrained to prevent unacceptable motion of the piping. The restraints for these two systems are mounted on specially designed structures which tie into the sacrificial shield wall and/or radial beams of the main steam relief valve platform.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the inboard RHR condensing mode and RCIC turbine steam supply systems to ensure safety as defined in Section 3.6.2.5.2. For the 10-in. inboard RHR condensing mode system, pipe whip restraints are provided at all locations where pipe breaks are postulated. The pipe whip restraints limit pipe motion resulting from postulated break to prevent impact with primary containment vessel wall, the HPCS system, and the main steam SRVs. Protection is required since either the ADS or the HPCS are required to depressurize the reactor subsequent to a pipe break in a line with a cross-section area less than 0.7 ft^2 (see Section 3.6.2.5.3).

For the 4-in. RCIC turbine steam supply, restraints are provided for the portion above the main steam relief valve platform to protect containment and the HPCS and SRVs. For the section of this system below the main steam relief valve platform, the pipe movement resulting from postulated breaks will move radially inward impacting the sacrificial shield or vertically down impacting the el. 512 ft platform. The CRD piping bundle in this area is located above this line precluding impact.

3.6.2.5.4.9 Main Steam Valve Drainage Piping.

System Arrangement

The main steam valve drainage piping consists of four 2-in. pipe lines, each originating at the bottom of the four MSIVs inside primary containment. The four lines are routed above the

diaphragm floor joining into one 3-in. line which then exits containment. Isolation valves are located just inside and just outside of the primary containment penetration.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for this system are shown in Figure 3.6-50. Where pipe breaks are postulated the system is restrained to prevent unacceptable motion of the main steam valve drainage piping. A number of the pipe whip restraints for this system are mounted on specially designed structures built up from the diaphragm floor. The remaining restraints are attached to the structure between primary containment and the diaphragm floor (see Section 3.6.2.5.4.1), which has been designed to support the main steam and reactor feedwater pipe whip restraints.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam valve drainage piping to ensure safety as defined in Section 3.6.2.5.2. Pipe whip restraints are provided for this system to protect primary containment structure and the diaphragm floor. Other required safety systems are protected by separation by being located at considerably higher elevations.

3.6.2.5.4.10 Main Steam Reactor Pressure Vessel Head Vent System.

System Arrangement

The RPV head vent system consists of a 2-in. line which originates at the top of the RPV dome and is routed through the primary containment bulkhead plate at azimuth 237°. The line is then routed below the bulkhead plate, to azimuth 70° where it drops down to el. 570 ft and joins a 26-in. main steam line.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for this system are shown in Figure 3.6-51. For the piping section above the bulkhead plate, the pipe whip restraints are mounted onto a removable lattice framework. For the portion of this line below the primary containment bulkhead plate, the restraints are mounted on structures, which tie into the stiffening beams for the bulkhead plate.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RPV head vent piping to ensure safety as defined in Section 3.6.2.5.2. There are no safety-related systems in the vicinity of the RPV head vent piping and pipe whip restraints are provided to protect the primary containment structure.

3.6.2.5.4.11 Main Steam and Reactor Feedwater Piping Inside Main Steam Tunnel.

System Arrangement

The four 26-in. main steam and two 24-in. reactor feedwater lines inside the main steam tunnel originate at the primary containment penetrations and run horizontally to the end of the tunnel. At this point the six lines drop vertically and are then routed horizontally within the turbine generator building. An isolation valve is located in each line just beyond the penetration.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the main steam and reactor feedwater lines inside the main steam tunnel are shown in Figures 3.6-53 and 3.6-54. Where breaks are postulated, the six lines are restrained to prevent unacceptable motion. The restraints are mounted on steel structures which then tie into the concrete walls and floors.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam and reactor feedwater lines inside the main steam tunnel to ensure safety.

Pipe whip restraints are provided to prevent pipe whip impact with the main steam or feedwater isolation valves. In addition, impact with adjacent main steam or feedwater lines is prevented. See Figure 3.6-24.

3.6.2.5.4.12 Residual Heat Removal System - Low Pressure Core Injection.

System Arrangement

The RHR/LPCI piping consists of three 14-in. loops whose arrangement is the same for two loops with the third loop being the mirror image of the other two. The piping originates at the reactor vessel at el. 552 ft, rises vertically to el. 563 ft where there is a horizontal section with a check valve. This valve is normally closed, limiting the high energy portion of each loop. After the valve the normally unpressurized section of piping drops to an elevation just below the main steam relief valve platform where it is routed to a penetration through primary containment at el. 534 ft.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the three RHR/LPCI mode piping loops are shown in Figures 3.6-40, 3.6-41, and 3.6-42. Where pipe breaks are postulated, the three piping loops are restrained to prevent unacceptable motion. The restraints for this system are

mounted onto the sacrificial shield wall and also on structures which tie back to the sacrificial shield wall.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RHR/LPCI mode piping to ensure safety as defined in Section 3.6.2.5.2. The pipe whip restraints limit pipe motion resulting from postulated breaks to preclude impact with primary containment and adjacent feedwater or core spray piping. Impact with adjacent feedwater or core spray piping may result in pipe break escalation that can exceed limitations of pipe break area and pipe break combination. The CRD piping bundles are separated by a considerable distance from high energy sections of the RHR/LPCI mode piping.

3.6.2.5.4.13 Reactor Pressure Vessel Drain System.

System Arrangement

The RPV drain system is a 2-in. line that originates at the bottom of the RPV and is routed inside the pedestal to a sleeve which leads through the pedestal. Outside the reactor pedestal the line then joins the RWCU system.

Pipe Whip Protection

The postulated pipe breaks and pipe whip restraint for the RPV drain system are shown in Figure 3.6-52. At postulated pipe break locations inside primary containment, the RPV drain system is restrained to prevent unacceptable motion of the pipe. This system contains only one pipe whip restraint. Where the pipe is routed along the platform at el. 512 ft 8 in., the pipe whip restraint is mounted on a transverse beam which is welded to the top of the radial platform beams.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RPV drain system to ensure safety as defined in Section 3.6.2.5.2. The single pipe whip support for the RPV drain system serves the dual purpose of providing pipe whip protection and seismic restraint. The pipe whip restraint is located above the platform at el. 512 ft 8 in., and is designed to prevent impact with the Quality Class I electrical conduits in the immediate vicinity of the RPV drain line. Since an annular clearance of only 1/16-in. is maintained between the pipe and the pipe whip support, the pipe whip support is also utilized as a rigid three-way support.

3.6.2.5.4.14 Reactor Recirculation Cooling System.

System Arrangement

The two A and B loops of the recirculation piping system consist of the pump discharge and suction piping as shown in Figure 3.6-55. The recirculation pump A and B discharge lines are arranged in a diametrically opposed manner, in the northern and southern segments, respectively, of primary containment. The A lines exit the RPV in five, equally spaced, 12-in. diameter lines commencing at azimuth 30° and ending at azimuth 150° (for B lines azimuth 210° to 330°). These five lines drop vertically alongside the sacrificial shield wall, from el. 536 ft 0 in. to a 16-in. diameter header at centerline el. 528 ft 0 in. A single 24-in. diameter line then drops vertically from the center of the header to el. 506 ft 3-7/8 in. where it is routed into the discharge nozzle of the recirculation pump.

The recirculation pump B and A suction lines are oriented along the 0° and 180° azimuths, respectively, with respect to the RPV. Each suction line consists of a single 24-in. diameter line which exits the RPV at el. 535 ft 3/4 in. and drops vertically alongside the sacrificial shield wall to el. 502 ft 6-1/8 in. where it is routed to the suction nozzle of the recirculation pump.

Pipe Whip Protection

For the recirculation pump suction and discharge systems, the location of postulated pipe breaks and pipe whip restraints are shown in Figure 3.6-55 which is representative of both recirculation loops. Conformance of the postulated break locations with the criteria of Section 3.6.2.1.1.1 is demonstrated in Figure 3.6-56. Where pipe breaks are postulated inside primary containment, the recirculation system piping is restrained to prevent unacceptable motion. These restraints are generally mounted on the side of the sacrificial shield wall structure or the RPV pedestal immediately below. Four restraints, which are located near the diaphragm floor and are not near the sacrificial shield wall or the RPV pedestal, consist of saddle type structures mounted on the diaphragm floor.

Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RRC system piping to ensure safety as defined in Section 3.6.2.5.2. Pipe whip restraints are provided to prevent impact with the diaphragm floor as well as to mitigate the consequences of a pipe rupture with respect to surrounding piping systems, structures, and components required for safe shutdown.

The physical separation of the recirculation system from the containment vessel precludes any damage that could result as a result of postulated pipe break.

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Table 3.6-1

High Energy Fluid Systems Outside
Primary Containment

System	Portion
Reactor core isolation cooling	Turbine supply and condensate removal lines
Auxiliary steam	Entire system
Heating steam	Entire system
Auxiliary condensate	Entire system
Heating steam condensate	Entire system
Control rod drive	Exhaust water header
Reactor water cleanup	Supply and return piping between RPV and Demineralizers
Main steam	Portion inside reactor building
Reactor feedwater	Portion inside reactor building

Table 3.6-2

Moderate Energy Fluid Systems Outside
Primary Containment

System	Portion
Residual heat removal	Entire system
Condensate	Entire system
Control rod drive	Portion which is not high energy
Demineralized water	Entire system
Reactor closed cooling water	Entire system
Fuel pool cooling	Entire system
Service water	Entire system
Plant service water	Entire system
Low pressure core spray	Entire system
High pressure core spray	Entire system
Reactor core isolation cooling	Portion which is not high energy
Fire protection	Entire system
Standby liquid control	Entire system

Table 3.6-3

Design Basis Break Locations Outside
Primary Containment

System	Node	M-200 Isometric	Diameter (in.)	Maximum Force (kips) or Thrust vs. Time Figure	Plan Location Figure
RCIC	15	120	4	3.6-96	3.6-7
RCIC	38	120	4	3.6-97	3.6-6
RCIC	40	120	4	3.6-97	3.6-6
RCIC	8	120	4	14.59	3.6-7
RCIC	54	120	4	10.6	3.6-5
RWCU	75	126	4	17.64	3.6-8
RWCU	45	126	4	17.64	3.6-8
RWCU	23	126	6	3.6-100	3.6-9
RWCU	2	126	6	3.6-101	3.6-9
RWCU	94	128	6	3.6-102	3.6-9
RWCU	133	128	6	3.6-99	3.6-8
RWCU	38	129	4	3.6-98	3.6-8
RWCU	17	129	4	3.6-98	3.6-8
RWCU	1056	129	5	23.79	3.6-8
RWCU	1034	129	5	23.79	3.6-8
MS	195	134	2	2.85	3.6-2
MS	207	134	2	2.85	3.6-2
MS	219	134	2	2.85	3.6-2
MS	228	134	2	2.85	3.6-2
MS	171	134	3	6.88	3.6-2
AS	87/387	139	4	3.6-105	3.6-1
AS	42, 73, 44	139	4	3.6-104	3.6-1
AS	168	139	4	3.6-104	3.6-1
AS	12	141	6	7.28	3.6-1

Table 3.6-3

Design Basis Break Locations Outside
Primary Containment (Continued)

System	Node	M-200 Isometric	Diameter (in.)	Maximum Force (kips) or Thrust vs. Time Figure	Plan Location Figure
AS	87	141	8	12.61	3.6-1
AS	153	141	8	12.61	3.6-1
AS	42	141	8	12.61	3.6-1
RWCU	1	142	6	30.25	3.6-9
RWCU	13	142	6	30.25	3.6-9
RWCU	1	144	4	13.34	3.6-11
RWCU	31	144	4	13.34	3.6-9
RWCU	48	144	6	30.25	3.6-9
RWCU	105	144	6	13.25	3.6-9
RWCU	767-771	144	6	30.25	3.6-9
RWCU	1140	144	4	13.29	3.6-7
HS	1	148	3	1.02	3.6-18
HS	387/987	148	4	3.6-105	3.6-18
HS	143	148	2	1.02	3.6-18
HS	151	148	2	1.02	3.6-18
HS	161	148	2	1.02	3.6-18
HS	443	148	2	1.02	3.6-18
HS	424	148	2	1.02	3.6-18
HS	17	148	2	1.02	3.6-18
HS	192	148	2	1.02	3.6-18
HS	203	148	2	1.02	3.6-18
HS	1101	148	3	1.02	3.6-18
HCO	211	149	2.5	0.182	--
HCO	269	149	4	0.182	--

Table 3.6-3

Design Basis Break Locations Outside
Primary Containment (Continued)

System	Node	M-200 Isometric	Diameter (in.)	Maximum Force (kips) or Thrust vs. Time Figure	Plan Location Figure
HCO	59	149	3	0.182	3.6-16
HCO	923	149	3	0.182	3.6-20
RFW	159	128	4	18.33	3.6-7
RFW	160	128	4	18.33	3.6-7
RFW	161	128	4	18.33	3.6-7
RFW	166	128	4	18.33	3.6-7
RFW	876	335B	24	433.12	3.6-7
RFW	874	335B	24	433.12	3.6-7
RFW	839	335B	24	433.12	3.6-7
RFW	837	335B	24	433.12	3.6-7
RFW	875 (36)	335B (128)	4	18.33	3.6-7
RFW	838 (6)	335B (128)	4	18.33	3.6-7
HS	313	342	6	7.28	3.6-1
AS	34	342	8	12.6	3.6-1
SS	1000	342	8	12.6	--
MS	1190	315	26	432.2	3.6-2
MS	171	400	26	432.2	3.6-2
MS	4	401	26	432.2	3.6-2
MS	67	402	26	432.2	3.6-2
CO	--	440	2.5	1.63	--
CO	--	440	2.5	1.63	--
CO	--	440	2.5	1.63	--
HS	43	447	6	1.82	--
HS	--	447	6	1.82	--

Table 3.6-3

Design Basis Break Locations Outside
Primary Containment (Continued)

System	Node	M-200 Isometric	Diameter (in.)	Maximum Force (kips) or Thrust vs. Time Figure	Plan Location Figure
HS	1000	447	6	1.82	--
HS	193-194	447	6	1.82	--
HS	313	448	6	1.82	--
HS	47 (203-205)	448	6	1.82	--
HS	201-202	448	6	1.82	--
HS	199-200	448	6	1.82	--
HS	43 (198-984)	448	6	1.82	--
HS	206-207	448	6	1.82	--
HS	44 (208-209)	448	6	1.82	--
HS	209-210	448	6	1.82	--
HS	45 (211-213)	448	4	0.802	--
HS	18	448	4	0.802	--
HCO	0	449	3	0.093	--
HCO	1-2	449	3	0.093	--
HCO	3-4	449	3	0.093	--
HCO	5-6	449	3	0.093	--
HCO	170 (7-8)	449	3	0.093	--
HCO	9-10	449	3	0.093	--
HCO	11-12	449	3	0.093	--
HCO	145	449	3	0.093	--
HCO	160 (15-17)	449	3	0.093	--
HCO	18-19	449	3	0.093	--
HCO	21-22	450	3	0.093	--
HCO	26 (25-27)	450	3	0.093	--

Table 3.6-3

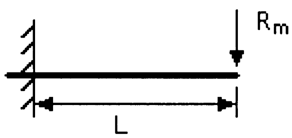
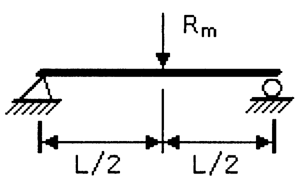
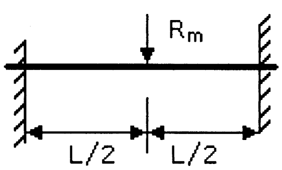
Design Basis Break Locations Outside
Primary Containment (Continued)

System	Node	M-200 Isometric	Diameter (in.)	Maximum Force (kips) or Thrust vs. Time Figure	Plan Location Figure
HCO	27-28	450	3	0.093	--
HCO	29-131	450	2.5	0.060	--
HCO	227-1227	450	2.5	0.060	--
HCO	1228-228	450	2.5	0.060	--
RWCU	600	500[a]	4	13.38	3.6-20a
RWCU	100a	500[a]	4	13.39	3.6-20b
RWCU	1670	500[a]	1.5	1.64	3.6-20a
RWCU	720	500[a]	4	13.39	3.6-20b
RWCU	6913	500[a]	2	2.61	3.6-20a
RWCU	680	500[a]	4	13.39	3.6-20a
RWCU	501	500[a]	4	13.39	3.6-20b
RWCU	4770	500[a]	1.5	1.64	3.6-20a
RWCU	310	500[a]	4	13.39	3.6-20b
RWCU	2898	500[a]	2	2.61	3.6-20a
RWCU	270	500[a]	4	13.39	3.6-20a
RWCU	100	500[a]	4	13.39	3.6-20a

[a] For Nodal Locations, Refer to FSAR Figures 3.6-38.4 and 3.6-38.5

Table 3.6-4

Resistance-Yield Displacement
Values for Beams

Description	Resistance	Yield Displacement
1. Cantilever  	$R_m = \dfrac{M_u}{L}$	$X_e = \dfrac{R_m L^3}{3EI}$
2. Simply supported  	$R_m = \dfrac{4M_u}{L}$	$X_e = \dfrac{R_m L^3}{48EI}$
3. Fixed supports  	$R_m = \dfrac{4\left(M_u^+ + M_u^-\right)}{L}$	$X_e = \dfrac{R_m L^3}{192EI}$

where:

M_u^+ = ultimate positive moment capacity (ft-lb)

M_u^- = ultimate negative moment capacity (ft-lb)

I = moment of inertia (in.^{4})

for reinforced-concrete I = I_a see notes accompanying table

NOTES

The resistance of typical structural elements, whose flexural strength defines the minimum capacity, and their yield displacement approximations are presented in Tables 3.6-4 and 3.6-5. It is preferable that the limiting capacity of an element be in the flexural mode, not in shear. In evaluating the yield displacement with the usual elastic analysis, the moment of inertia must account for cracking of concrete sections. The empirical relation for this type of loading is an average moment of inertia, I_a, calculated as follows.

Table 3.6-4

Resistance-Yield Displacement
Values for Beams (Continued)

NOTES (Continued)

$$I_a = \frac{1}{2}\left(I_g + I_c\right) = \frac{1}{2}\left(\frac{bt^3}{12} + Fbd^3\right)$$

where

I_g = moment of inertia of gross concrete cross section of thickness t about its centroid (neglecting steel areas) ($in.^4$)

I_c = moment of inertia of the cracked concrete section ($in.^4$)

b = width of concrete sections (in.)

F = coefficient for moment of inertia of cracked section with tension reinforcing only (see Figure 3.6-106)

t = concrete thickness (in.)

d = distance from extreme compression fiber to centroid of tension reinforcing (in.)

The moment of inertia, I_a, as calculated by the above equation must be used in the displacement equation in Tables 3.6-4 and 3.6-5 for all reinforced-concrete members. The ultimate moment capacity of a concrete section is considered as the moment strength:

$$M_u = 0.9 A_s f_{dy} \left(d - a/2\right) \quad \text{(in.-lb)}$$

where

A_s = area of tensile reinforcing steel ($in.^2$)

f_{dy} = allowable dynamic yield stress for reinforcing steel ($lb/in.^2$)

d = distance from extreme compression fiber to centroid of tension reinforcing (in.)

a = depth of equivalent rectangular stress block (in.)

Table 3.6-4

Resistance-Yield Displacement
Values for Beams (Continued)

NOTES: (Continued)

If the element has compression steel, the appropriate equation for compression steel applies.

The amount of reinforcing steel in concrete members satisfied the following criteria:

For members with tension steel only:

$$\frac{1.4\sqrt{f'_c}}{f_y}\left(\frac{t}{d}\right)^2 \leq \frac{A_s}{bd} \leq \frac{0.25\ f'_c}{f_y}$$

For members with tension and compression steel:

$$\frac{1.4\sqrt{f'_c}}{f_y}\left(\frac{t}{d}\right)^2 \leq \frac{A_s}{bd}$$

$$\frac{A_s - A'_s}{bd}\left(\frac{t}{d}\right)^2 \leq \frac{0.25\ f'_c}{f_y}$$

where

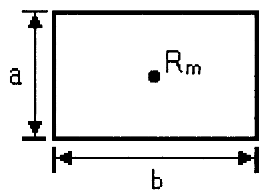
f'_c = compression strength of concrete (lb/in.^{2})
A'_s = area of compressive reinforcement of concrete (in.^{2})

Table 3.6-5

Resistance-Yield Displacement
Values for Slabs and Plates

Description	Resistance	Yield Displacement

1. Simply supported on all four sides with load at center

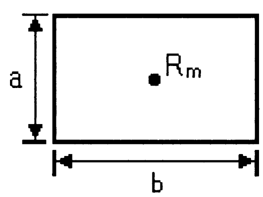


$$R_m = 2\pi M_u$$

$$X_e = \frac{\alpha R_m a^2}{12EI}\left(1 - \nu^2\right)$$

b/a	1.0	1.1	1.2	1.4	1.6	1.8	2.0	3.0	∞
α	0.1390	0.1518	0.1624	0.1781	0.1884	0.1944	0.1981	0.2029	0.2031

2. Fixed supports on all 4 sides with load at center



$$R_m = 2\pi\left(M_u^+ + M_u^-\right)$$

$$X_e = \frac{\alpha R_m a^2}{12EI}\left(1 - \nu^2\right)$$

b/a	1.0	1.2	1.4	1.6	1.8	2.0	∞
α	0.0671	0.0776	0.0830	0.0854	0.0864	0.0866	0.0871

where

ν = Poisson's ratio
t = thickness (in.)
E = modulus of elasticity (lb/in.^{2})
I = moment of inertia per unit width (in.^{4}/in.)
for reinforced-concrete section I = I_a see notes accompanying Table 3.6-4
M_u^+ = ultimate positive moment capacity (in.-lb/in.)
M_u^- = ultimate negative moment capacity (in.-lb/in.)

Table 3.6-6

Design Load in Areas Where Piping Failures Occur

Pipe Break	Room	El. (ft)	Differential Pressure (psid)	Differential Temperature (°F)		Live Load (psf)	Hung Loads (psf)		Equipment Loads (Kips)
				Int. to Int.	Int. to Ext.		From Floor	From Ceiling	
120-8	R15	422	3.9	0	40	--	--	59	1.4[k] pump
120-4	R113	441	4.1	0	40	250	59	68	None
120-5, 6, 7	R112	441	3.9	0	40	250	59	68	None
139-3, 4, 19, 20, 21	R206	471	1.2	0	40	250	32	34	None
120-1, 2	R313	510 ft 6 in.	2.9	0	40	250	40	30	None
128-11									
128-10	R408	522	2.3	0	--	250	41	88	None
126-3, 5	R406	522	15.0	0	--	250	126	0	1.5[k] pump
129-47, 48	R407								
126-1, 2	R409	535	14.6	0	--	250	40	80	None
129-39, 41, 42, 43, 44, 45									
128-48									
144-27, 28	R511	548	6.6	20	--	400	80	55	None
144-32									
126-6, 128-7, 8	R510	548	2.9	20	--	400	65	51	Heat exchangers 16.2 and 29.5
142-20, 21, 22, 23									
144-29, 31									

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Table 3.6-6

Design Load in Areas Where Piping Failures Occur (Continued)

Pipe Break	Room	El. (ft)	Differential Pressure (psid)	Differential Temperature (°F)		Live Load (psf)	Hung Loads (psf)		Equipment Loads (Kips)
				Int. to Int.	Int. to Ext.		From Floor	From Ceiling	
128-9	R509	548	1.00	20	--	400	88	50	None
139-1	R604	572	0.07	0	40	250	15	36	Heat and vent unit 51K
148-1, 2, 3, 5, 6, 7, 8, 9, 10, 11, 12									
120-6	R308	501	1.6	0	40	1000	63	55	None
Steam tunnel	R310	501	20.0	20	--	1000	277	41	None
(RWCU-100, 600)[a]	C230	467	0.58	0	--	150	(P)[b]	40	None
(RWCU-100a, 720)[a]	C333	487	1.00	0	--	150	120	40	37[k] Demin
(RWCU-1670, 6913, 680)[a]	C217	467	0.95	0	--	150	170	120	None
(RWCU-310, 501)[a]	C336	487	1.00	0	--	150	120	40	37[k] Demin
(RWCU-270, 2898, 4770)[a]	C219	467	0.95	0	--	150	170	120	None

Notes: For location of pipe break numbers, see Table 3.6-3.
For vertical and horizontal seismic factors, see Section 3.7.
[a] For Node Locations, Refer to FSAR Figures 3.6-38.4 and 3.6-38.5.
[b] Force given as multiple point loads.

Table 3.6-7

Maximum Ductility Ratios Steel
Structural Components

Steel beams (lateral load)	
(Note: To develop this ductility, the flanges must be thick enough to prevent local plastic buckling)	26
Steel beams (lateral and axial load)	8
Welded portal frams (vertical load)	6-16

REINFORCED-CONCRETE STRUCTURAL COMPONENTS

Tension reinforced-concrete beams and slabs, (flexure controls design)	$\frac{0.10}{p^{a}}$ _ 10
Doubly reinforced-concrete beams and slabs, (flexure controls design)	$\frac{0.10}{p-p'^{b}}$ _ 10
Reinforced-concrete columns, walls, and other elements exhibiting brittle fracture (compression controls design)	1.3

[a] p is the ratio of tensile reinforcement.

[b] p' is the ratio of compression reinforcement.

Table 3.6-8

Dynamic Strength of Materials

	Dynamic Increase Factor
Reinforced concrete	
Concrete	
Compression, axial or flexural	1.25
Shear as a measure of diagonal tension and punching shear	1.00
Bond	1.00
Reinforcing steel	
Tension	1.10
Compression	1.10
Shear reinforcement to resist shear as a measure of diagonal tension and punching shear	1.00
Structural Steel	
Flexure and tension	1.10
Compression	1.10
Shear	1.00

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Table 3.6-9

Summary of Subcompartment Pressure Analysis[a]

Compartment Where Break Occurs			Piping System	Differential Pressure	
Elevation (ft)	Room	Description	Line Designation	Maximum Differential (psi)	Differential Between the Rooms
422	R15/R112	RCIC pump room	4"RCIC (13)-4	2.8	R15, R112/R206
				2.8	R15, R112/R14, R113
				2.8	R15, R112/R6, R116
471	R206	El. 471 ft, SE corner	4"AS (11)-2	0.25	Reactor building and
		Open floor area	3"AS (11)-2	0.25	ambient
471	R206C	Pipe chase el. 471 ft	4"RCIC (13)-4	1.2	R308/R206
501	R308	TIP room	4"RCIC (13)-4	0.2	R308/R305, R206, R313
501	R308	TIP room	6"RWCU (2)-4	0.24	R308/R305, R206, R313
501	R313	El. 510 ft valve room	6"RWCU (2)-4	0.24	R313/R308, R408
522	R406/R407[b]	RWCU pump room	4"RWCU (2)-4	6.9	R406/R404, R305
				5.6	R406/R407, R409
				3.9	R409/R504, R404
				1.7	R405/R305, R404, R504
				2.2	R409/R405

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Table 3.6-9

Summary of Subcompartment Pressure Analysis[a]

Compartment Where Break Occurs			Piping System	Differential Pressure	
Elevation (ft)	Room	Description	Line Designation	Maximum Differential (psi)	Differential Between the Rooms
467	C230	Cable Chase Room	4"RWCU (1)-3	0.58 0.33	C230/C217, C212, Reactor building
467	C217	East Valve Room	2"RWCU (21)-3 1.5"RWCU (22)-3	0.95	C217/C230, C218
467	C219	West Valve Room	2"RWCU (21)-3 1.5"RWCU (22)-3	0.95	C219/C218, C220, C221
487	C333/C336	Demin. Vault 1A/1B	4"RWCU (1)-3	1.0	C333/C405, C336/C405
487	C344	Cable Chase Room	4"RWCU (1)-3	0.59 0.34	C344/C325, C326 Reactor building

Table 3.6-9

Summary of Subcompartment Pressure Analysis[a] (Continued)

Compartment Where Break Occurs			Piping System	Differential Pressure	
Elevation (ft)	Room	Description	Line Designation	Maximum Differential (psi)	Differential Between the Rooms
522	R408	Valve room	6"RWCU (2)-4	1.1	R408/R404
				1.1	R408/R305
				1.1	R408/R509
548	R509	Valve room	6"RWCU (2)-4	0.9	R509/R508, R408
				0.9	R509/R607
548	R510	RWCU HX room	6"RWCU (1)-5SX/SXL	2.7	R510/R504, R508
				2.7	R510/R404, R604
548	R511/ R511A	Valve room	6"RWCU (1)-4	6.3	R511/R404, R504
				2.4	R511/R604
572	R604	El. 572 ft open floor area	4"HS (1)-2	0.07	Reactor building and
			4"AS (11)-2	0.07	Ambient

[a] Table applies to reactor building secondary containment and portions of the radwaste building, exclusive of the main steam tunnel, tunnel ventway, and tunnel extension.

[b] Break could occur in either room; break assumed in R406.

Table 3.6-10

Subcompartment Analysis

Nodal Volume Data for a Postulated Pipe Break in the Main Steam Tunnel[a]

Node	Description	Volume (ft^3)	Elevation (ft)
1	Main steam tunnel, south	7427	501
2	Main steam tunnel, north	4345	501
3	Ventway, el. 501 ft 0 in. to el. 519 ft 0 in.	3629	501
4	Ventway, el. 519 ft 0 in. to el. 532 ft 0 in., west	3672	519
5	Ventway, el. 519 ft 0 in. to el. 532 ft 0 in., east	2340	519
6	Ventway, el. 532 ft 0 in. to el. 548 ft 0 in.	7855	532

[a] For nodalization scheme, see Figure 3.6-67.

Table 3.6-11

Subcompartment Analysis

Flow Junction Data for a Postulated Pipe Break in the Main Steam Tunnel[a]

From Node	To Node	Junction Flow Area (ft^2)	Junction Elevation (ft)	Junction Inertia (ft^{-1})	Form Loss Coefficient[b] Forward Flow	Form Loss Coefficient[b] Reverse Flow	Frictional Loss Coefficient[b]
1	2	438.7	509	0.02656	1.06	1.14	0.1
2	3	c					
2	4	218.4	519	0.06044	2.66	2.69	0.1
3	5	170.0	519	0.08014	0.163	0.116	0.1
4	5	84.6	525	0.378	0.6	0.6	0.1
4	6	310.4	532	0.368	0.6	0.6	0.1
5	6	170.0	532	0.0486	0.145	0.206	0.1

[a] For nodalization scheme, see Figure 3.6-67.

[b] These data are dimensionless.

[c] No data furnished since panel B between nodes 2 and 3 is assumed closed during postulated pipe break.

Table 3.6-12

Main Steam Tunnel Subcompartment Analysis
Information for Blowout Panels C and D

Panel C (horizontal, hinged, nonbolted, sheet steel blowout panel)

Total weight:	2060 lb
Area:	230.6 ft^2
Moment arm:	3.375 ft
Moment of inertia:	31,286 lb_{mass}-ft^2
Damping constant:	Neglected

Panel D (vertical, bolted, insulated metal blowout panel)

Total weight:	16,000 lb
Area:	1060.8 ft^2

Reference: Figures 3.6-65 and 3.6-66.

Table 3.6-13

Subcompartment Analysis

Nodal Volume Data for a Postulated Pipe Break in the Main Steam Tunnel Extension[a]

Node	Description	Volume (ft^3)	Elevation (ft)
1	Main steam tunnel extension, south	2320	501
2	Main steam tunnel extension, north	2799	501
3	Turbine generator building, el. 471 ft 0 in. floor	728610	471
4	Turbine generator building, el. 441 ft 0 in. floor	658938	441
5	Turbine generator building, el. 501 ft 0 in. floor	1270590	501

[a] For nodalization scheme, see Figure 3.6-76.

Table 3.6-14

Subcompartment Analysis

Flow Junction Data for a Postulated Pipe Break in the Main Steam Tunnel Extension[a]

From Node	To Node	Junction Flow Area (ft^2)	Junction Elevation (ft)	Junction Inertia (ft^{-1})	Form Loss Coefficient[b]		Frictional Loss Coefficient[b]
					Forward Flow	Reverse Flow	
1	2	379.0	509	0.01395	0.6	0.56	0.1
1	3	73.6	501	0.1722	1.29	1.07	0.1
2	3	114.6	501	0.1434	1.16	0.99	0.1
3	4	230.0	471	0.1091	1.28	1.49	0.1
3	5	507.0	501	0.01176	1.54	1.55	0.1

[a] For nodalization scheme, see Figure 3.6-67.

[b] These data are dimensionless.

Table 3.6-15

Piping Systems Inside Containment for Which Design Basis Pipe Breaks are Postulated

System	Portion
Low pressure core spray (LPCS)	RPV to first check valve.
High pressure core spray (HPCS)	RPV to first check valve.
RHR/LPCI mode (loop A)	RPV to first check valve.
RHR/LPCI mode (loop B)	RPV to first check valve.
RHR/LPCI mode (loop C)	RPV to first check valve.
RHR shutdown cooling return (loop A)	Recirculation pump discharge to first check valve.
RHR shutdown cooling return (loop B)	Recirculation pump discharge to first check valve.
RHR shutdown cooling supply	Recirculation pump suction to closed valve RHR-V-9 (MOF009).
Reactor feedwater (RFW) line A	Entire run within primary containment.
Reactor feedwater (RFW) line B	Entire run within primary containment.
RHR condensing mode/RCIC turbine steam	Entire run within primary containment.
Main steam (MS) loop A	Entire run within primary containment.
Main steam (MS) loop B	Entire run within primary containment.
Main steam (MS) loop C	Entire run within primary containment.
Main steam (MS) loop D	Entire run within primary containment.
Standby liquid control (SLC)	RPV to closed valve SLC-V-8, HPCS line to first check valve.
Reactor water cleanup (RWCU)	Entire run within primary containment.
RRC Recirculation pump A discharge	Entire run within primary containment.

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Table 3.6-15

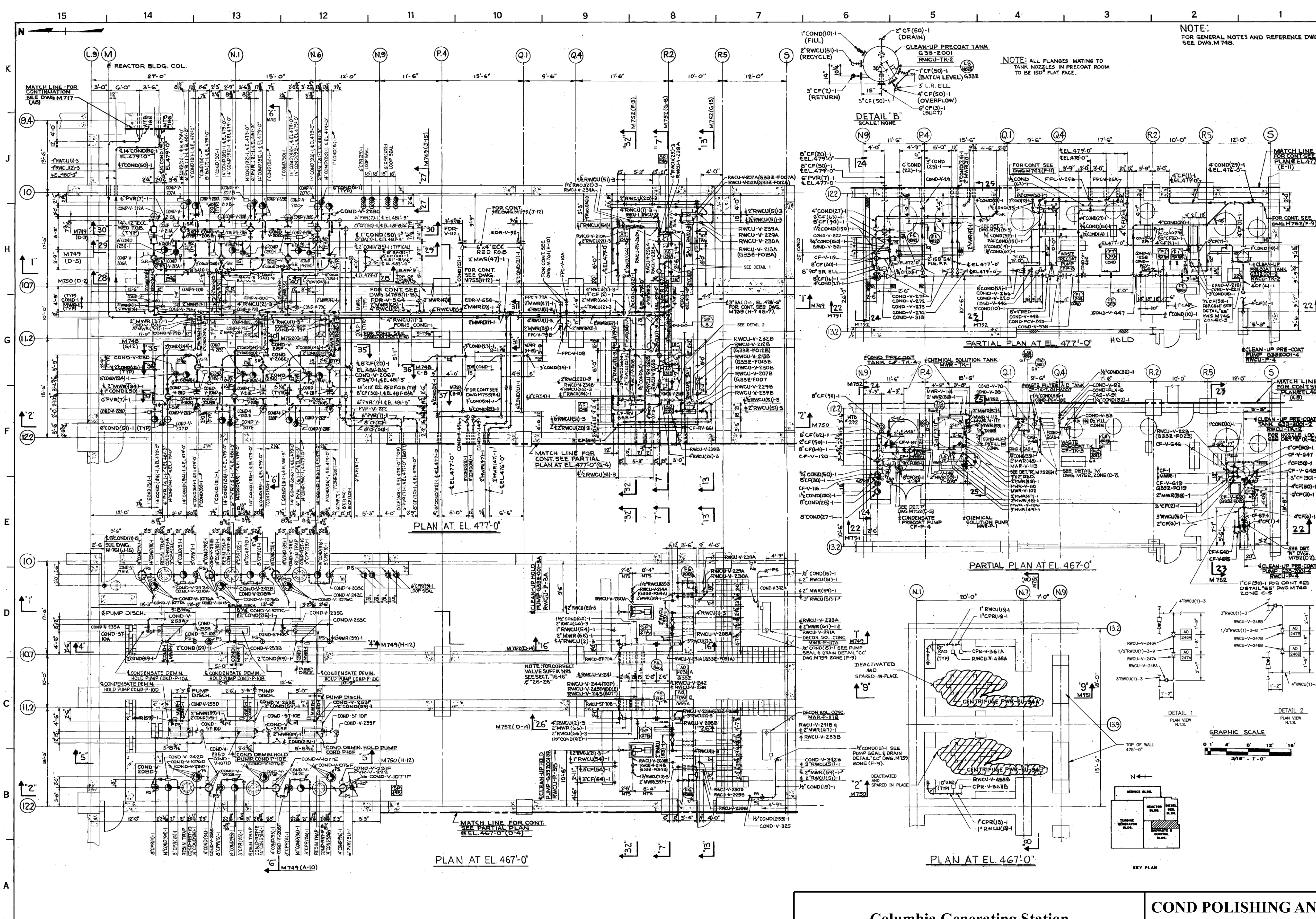
Piping Systems Inside Containment for Which Design Basis
Pipe Breaks are Postulated (Continued)

System	Portion
RRC recirculation pump B discharge	Entire run within primary containment.
RRC recirculation pump A suction	Entire run within primary containment.
RRC recirculation pump B suction	Entire run within primary containment.
RRC reactor pressure vessel drain	RPV to three (3) closed valves.
Main steam (MS) valves drainage piping	Entire run within primary containment.
Main steam (MS) RPV head vent	Entire run within primary containment.
RCIC/RPV head spray	RPV to first check valve.

Table 3.6-16

Piping Systems Containing Break Exclusion
Areas Between Primary Containment Isolation Valves

Piping System	Pipe Size (in.)
Main steam loop A	26
Main steam loop B	26
Main steam loop C	26
Main steam loop D	26
Reactor feedwater line A	24
Reactor feedwater line B	24
RHR condensing mode/RCIC turbine steam	10/4
Reactor water cleanup	6



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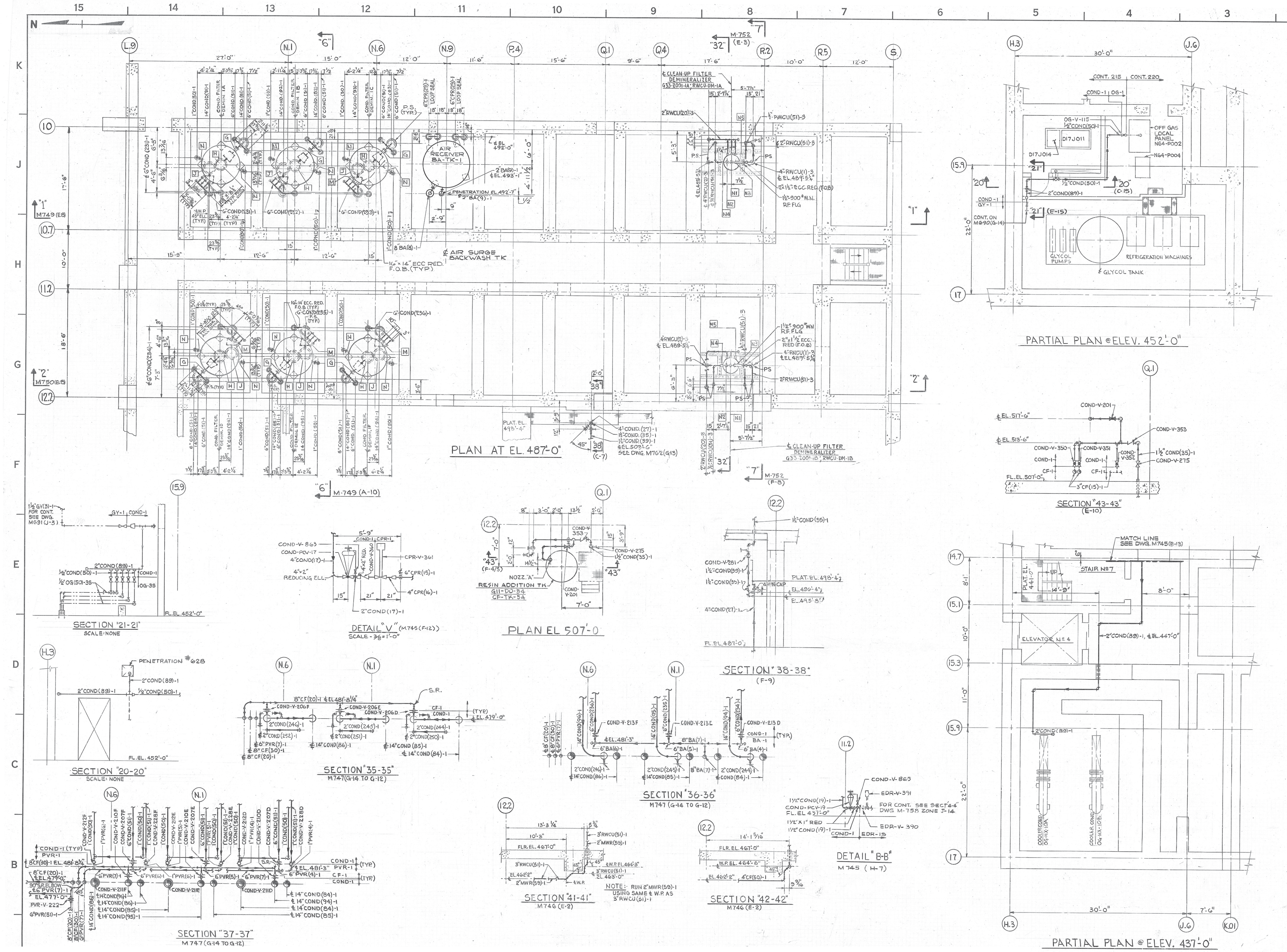
COND POLISHING AND RWCU PIPING PLAN
EL 467FT AND EL 477FT W

Draw. No. M747 | Rev. 46 | Figure 3.6-20A



GENERAL NOTES:

1. ALL DRAINS TO EQUIPMENT DRAINS TO TERMINATE 1" ABOVE FINISHED FLOOR UNLESS OTHERWISE NOTED.
2. W. T. B. - WATER TIGHT BOOT; FOR DETAIL SEE DWG. M858, SHT. 1 & 2.
3. P.S. - PIPE SLEEVE.
4. ALL PIPING & ASSOCIATED COMPONENTS SHOWN ON THESE DWGS SHALL BE CLASSIFIED AS INDICATED ON THEIR RESPECTIVE SYSTEM FLOW DIAGRAMS
5. WORK THIS DWG. WITH DWGS. M745 THRU M747 AND M749 THRU M752
6. FOR INSTALLATION OF THERMOWELLS AND TEMPERATURE TEST CONNECTIONS SEE DWG. M610.
7. ALL VALVE STEMS FOR VALVES 2½" AND LARGER NOT LOCATED SHALL BE IN A VERTICAL POSITION.

REFERENCE DRAWINGS:

FLOW DIAGRAM - REACTOR WATER CLEAN-UP SYSTEM. DWG. M-523

FLOW DIAGRAM - FUEL POOL COOLING AND CLEAN-UP SYSTEM. DWG. M-526

FLOW DIAGRAM - CONDENSATE SUPPLY SYSTEM ALL BUILDINGS. DWG. M-527

FLOW DIAGRAM - RADIOACTIVE WASTE SYSTEM FLOOR DRAIN PROCESSING. DWG. M-531

FLOW DIAGRAM - RADIOACTIVE WASTE SYSTEM EQUIPMENT DRAIN PROCESSING. DWG. M-532

FLOW DIAGRAM - CHEMICAL WASTE PROCESSING. DWG. M-533

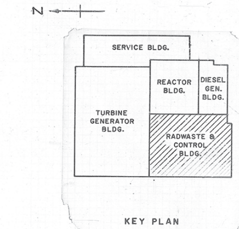
FLOW DIAGRAM - RADIOACTIVE WASTE SYSTEM CONDENSATE DEMINERALIZATION. DWG. M534.

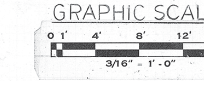
FLOW DIAGRAM - RADIOACTIVE WASTE DISPOSAL SOLIDS HANDLING SYSTEM DWG. M-536

INSTALLATION OF THERMOWELLS AND SAMPLE PROBES DWG. M610

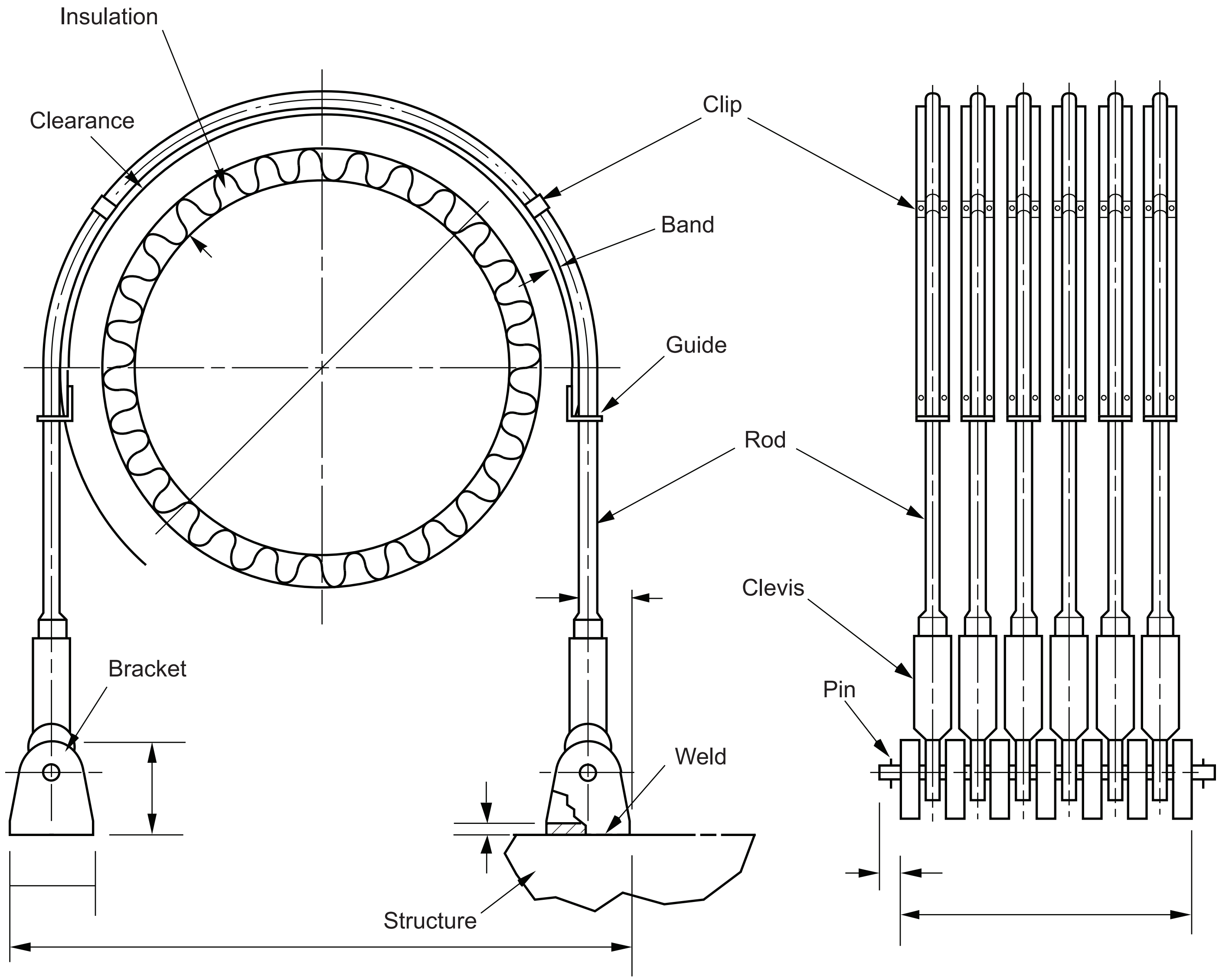
GENERAL NOTES: (CONT'D)

8. TANK FLUSHING CONTROL AIR CHECK VALVES AND FLEXIBLE HOSES ARE TO BE INSTALLED AS CLOSE AS POSSIBLE TO INSTRUMENT AIR BUBBLER.





Columbia Generating Station Final Safety Analysis Report	CONDENSATE POLISHING AND RWCU PIPING PLAN 487FT AND MISCELLANEOUS PARTIAL PLANS AND SECTIONS		
	Draw. No. M748	Rev. 15	Figure 3.6-20B



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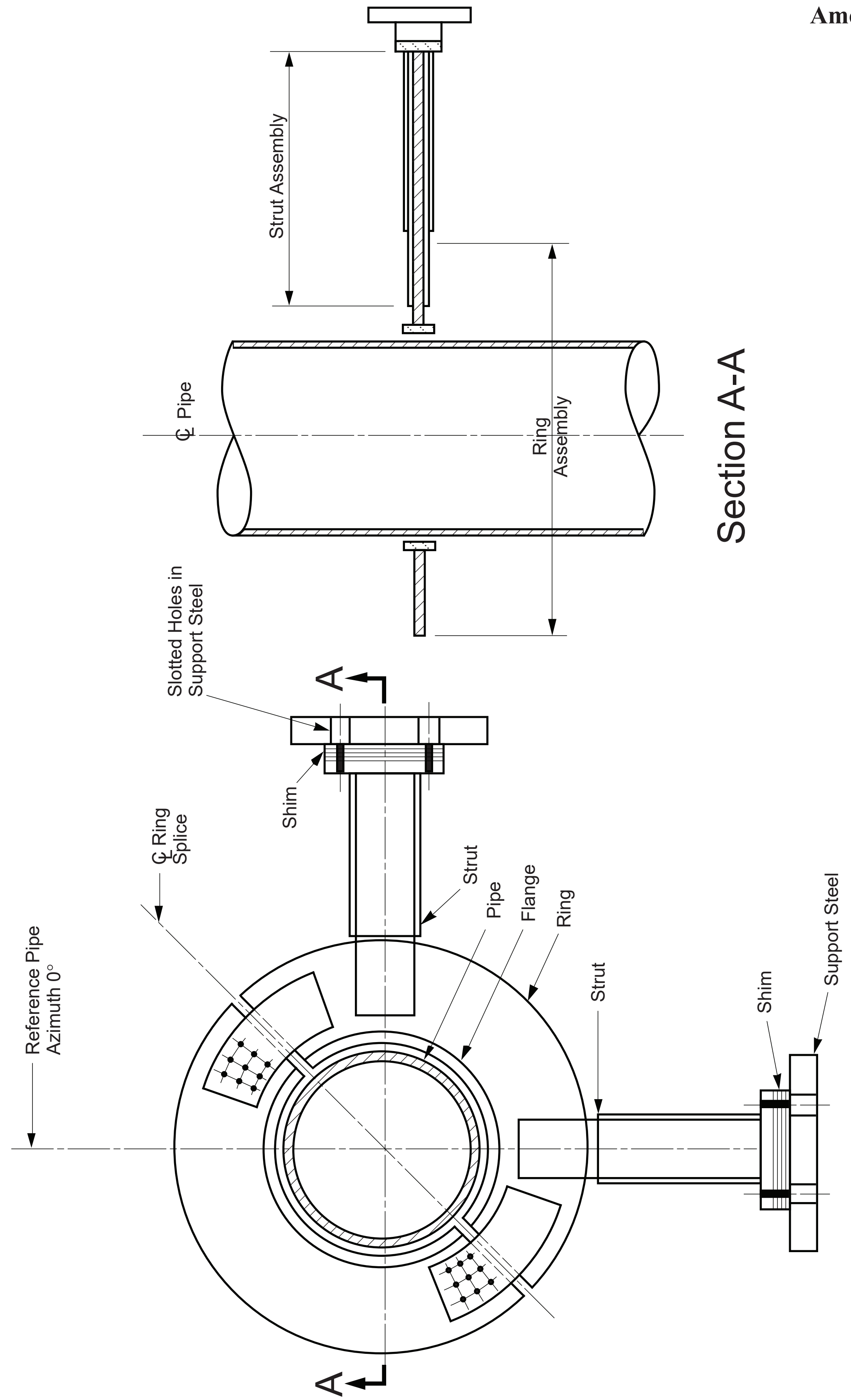
U-Bar Type Pipe Whip Restraint Configuration

Draw. No. 970187.78 | Rev. | Figure 3.6-21



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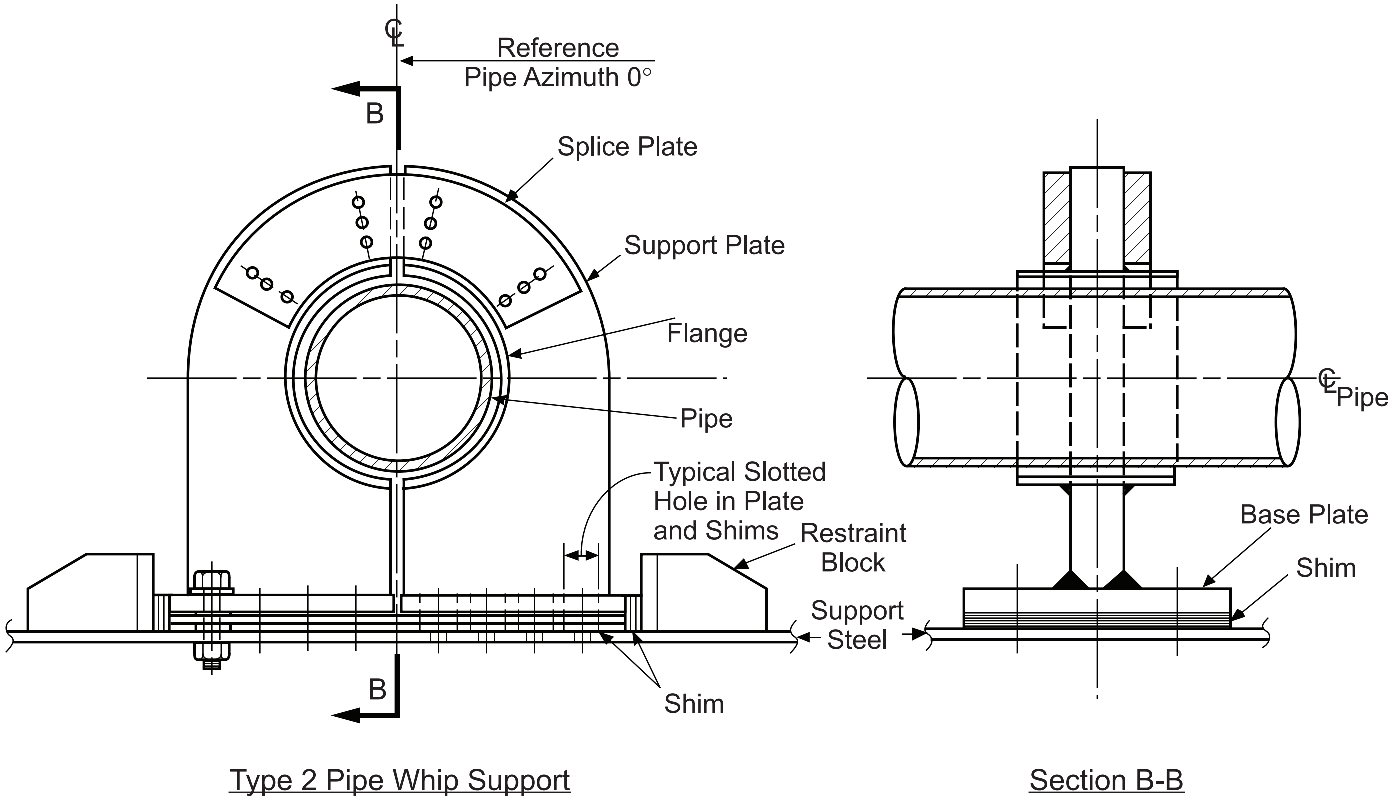
Rigid Type Pipe Whip Restraint Configuration

Draw. No. 990578.57	Rev.	Figure 3.6-22.1



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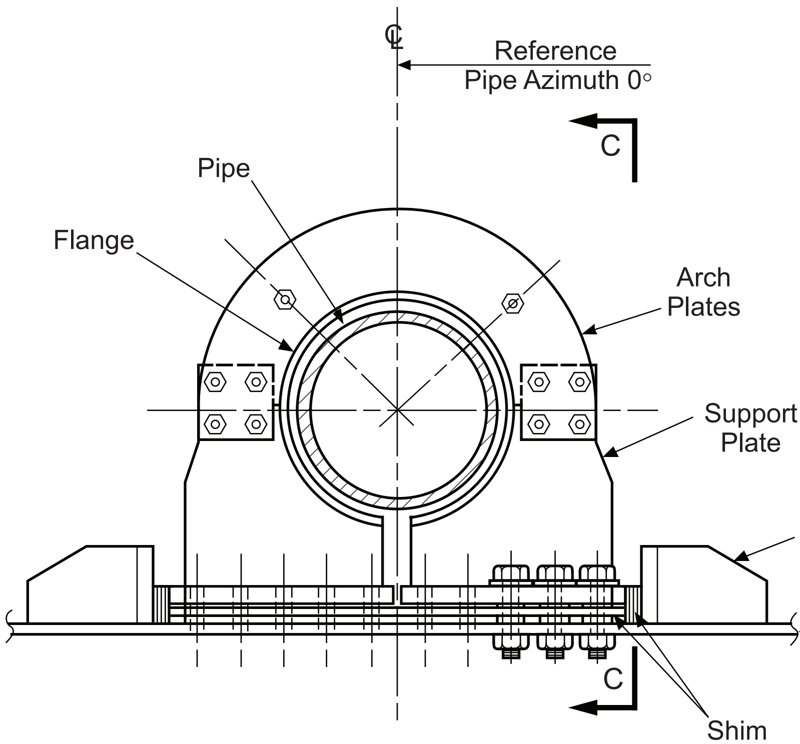
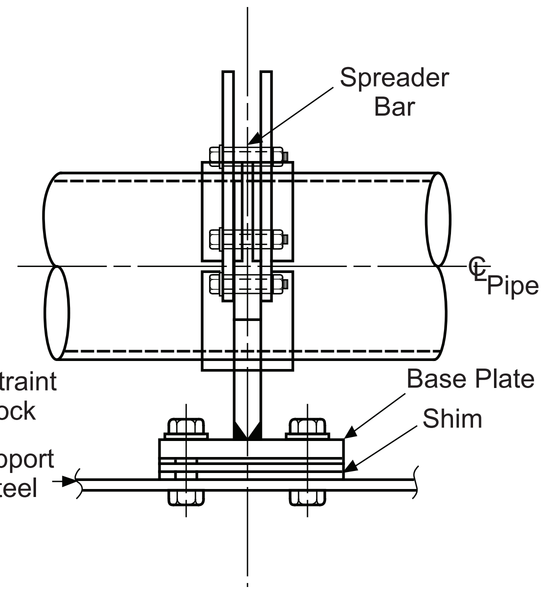
Rigid Type Pipe Whip Restraint Configuration

Draw. No. 970187.79 | Rev. | Figure 3.6-22.2



Type 3 Pipe Whip Support



Section C-C

Columbia Generating Station
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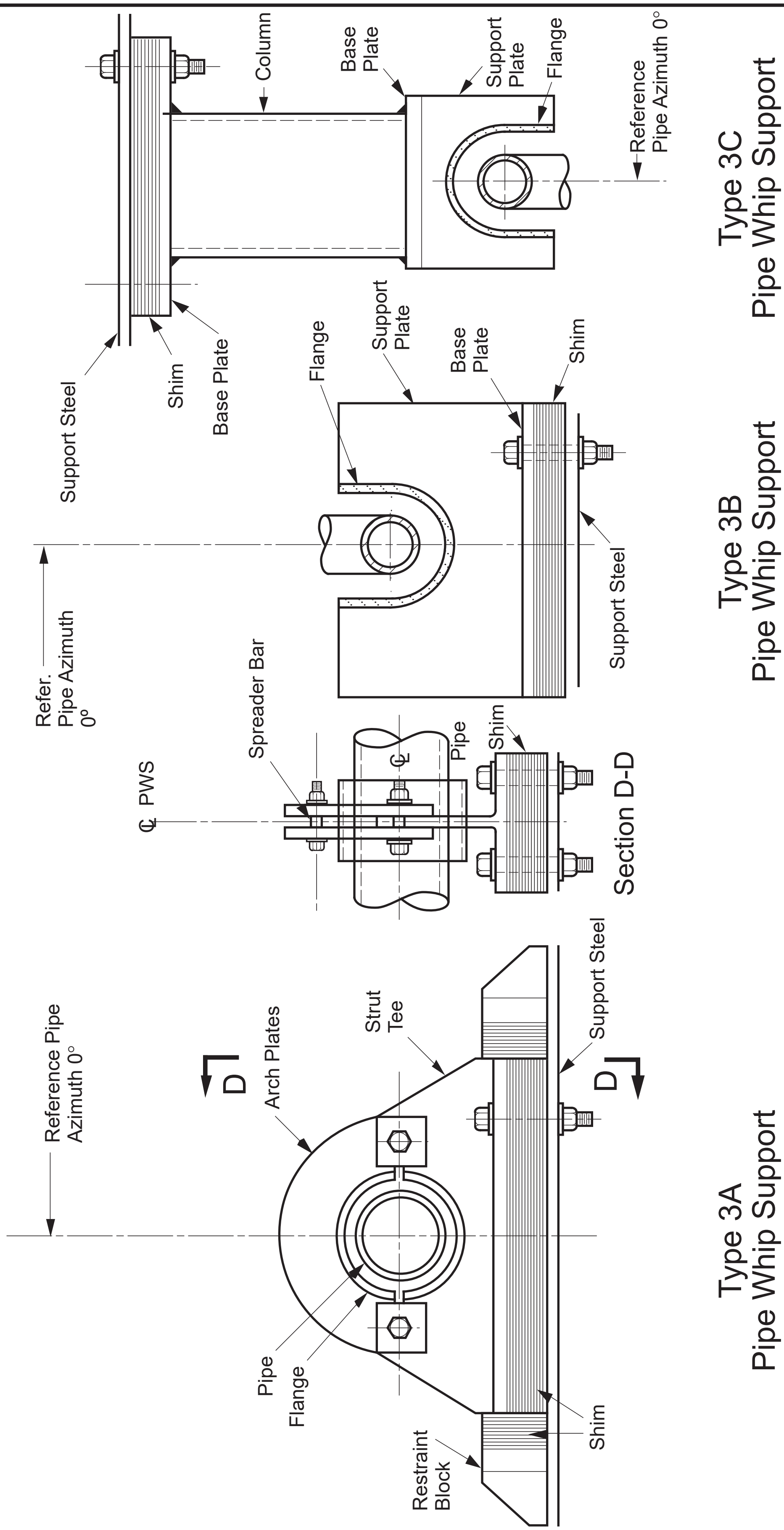
Rigid Type Pipe Whip Restraint Configuration

Draw. No. 970187.80 | Rev. | Figure 3.6-22.3

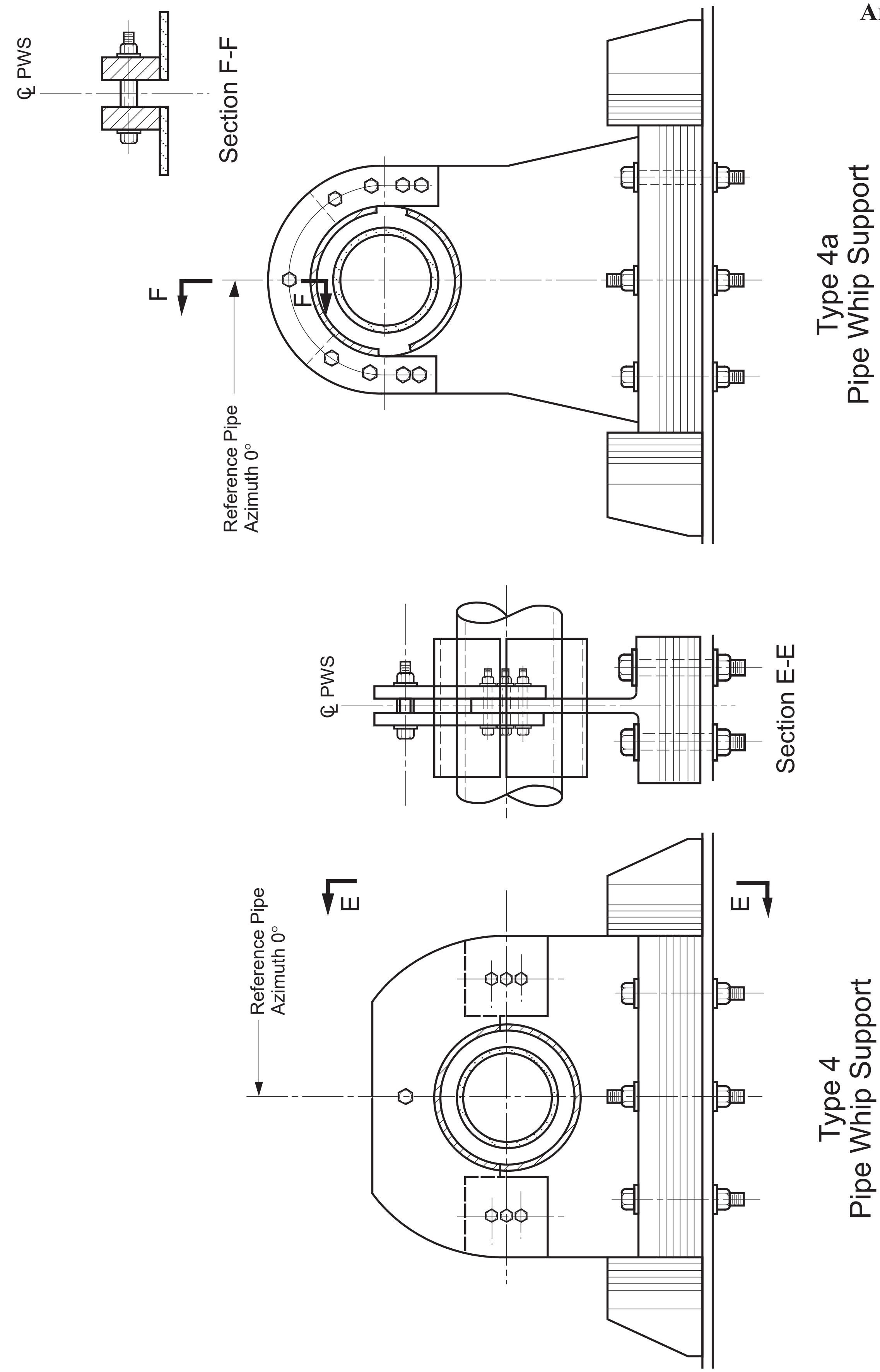


Columbia Generating Station
Final Safety Analysis Report

Rigid Type Pipe Whip Restraint Configuration

Draw. No. 990578.58	Rev.	Figure 3.6-22.4

Amendment 55
May 2001

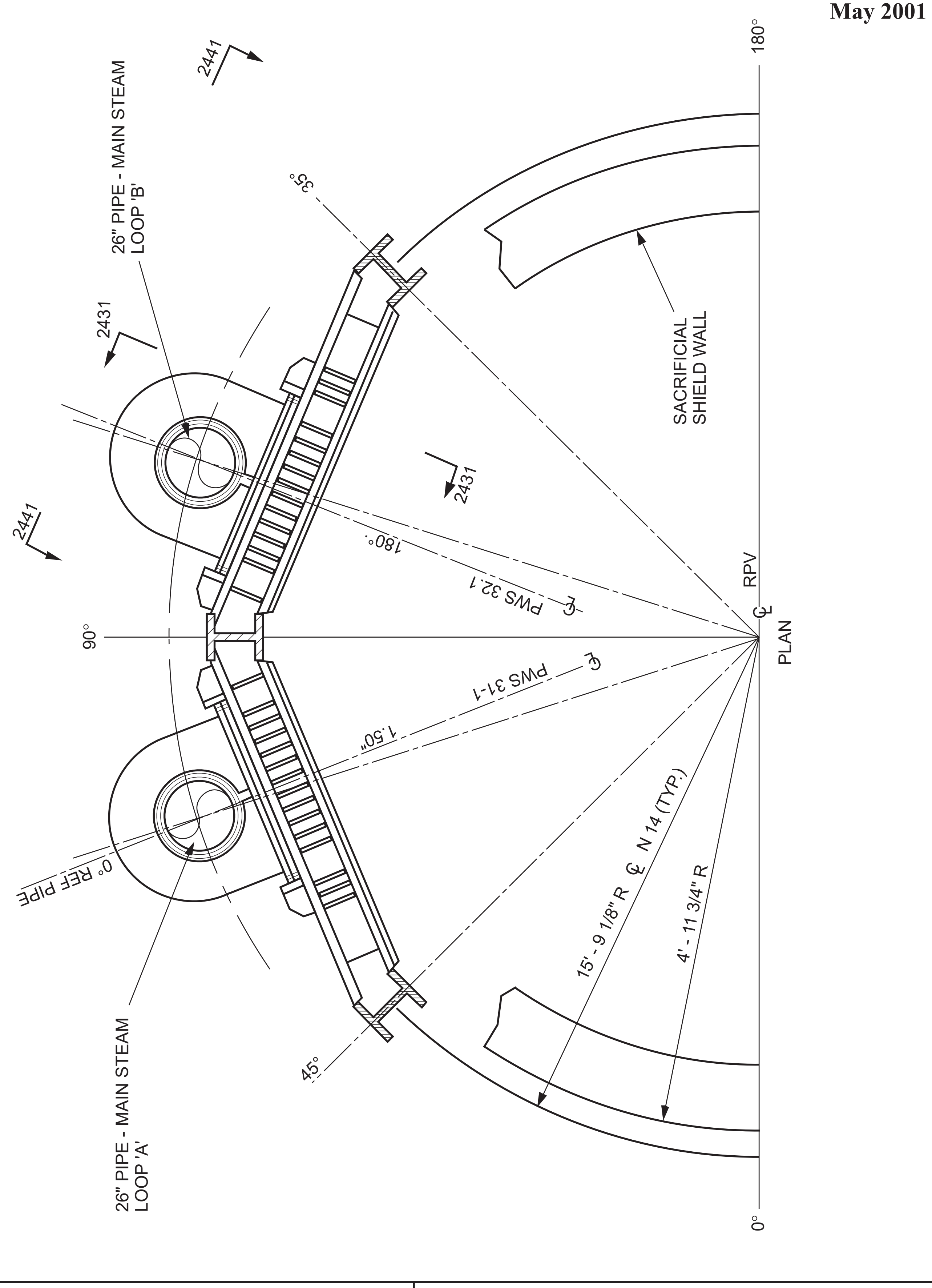


Columbia Generating Station
Final Safety Analysis Report

Rigid Type Pipe Whip Restraint Configuration

Draw. No. 990578.59	Rev.	Figure 3.6-22.5

Amendment 55
May 2001

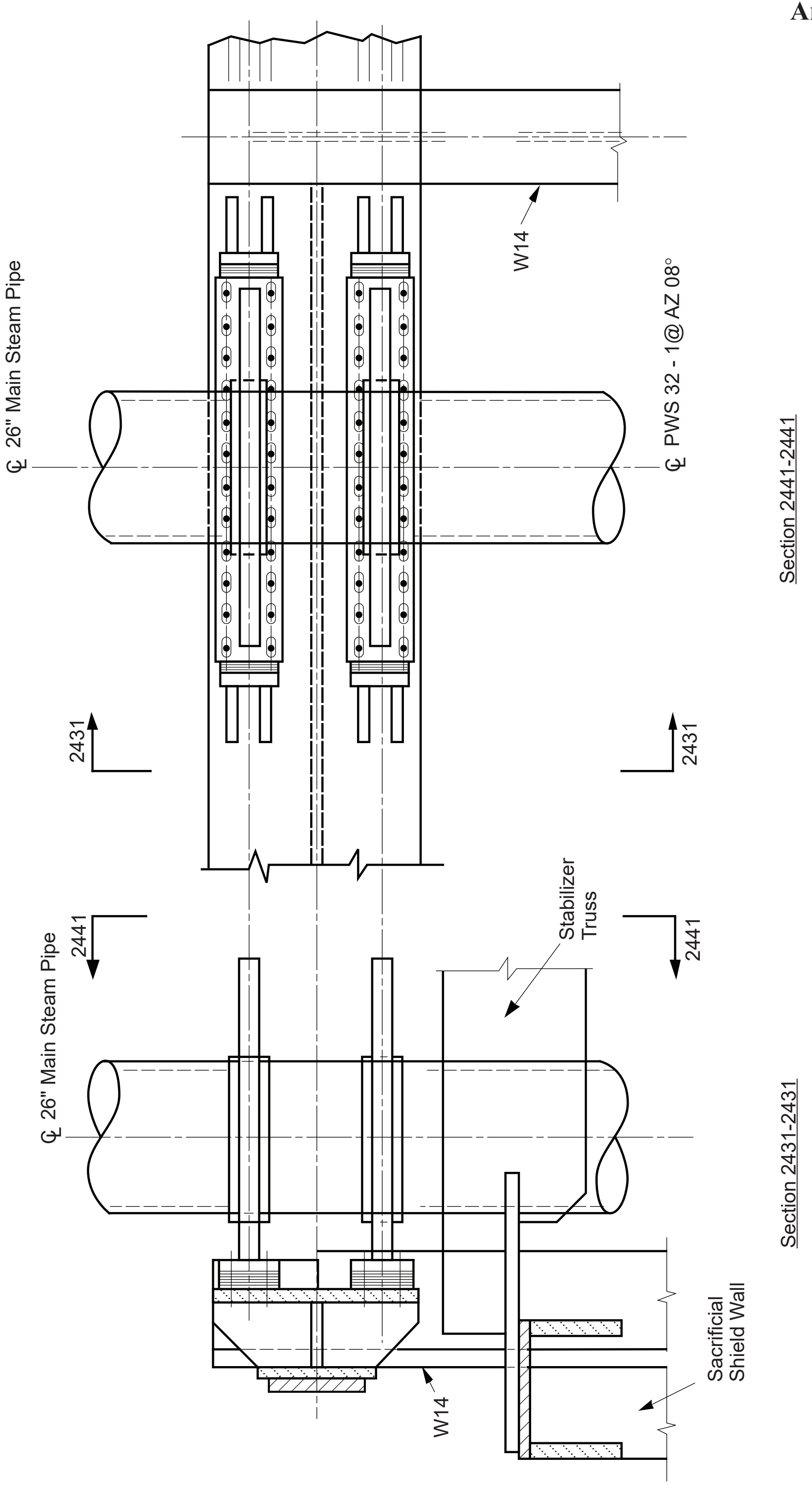


Columbia Generating Station
Final Safety Analysis Report

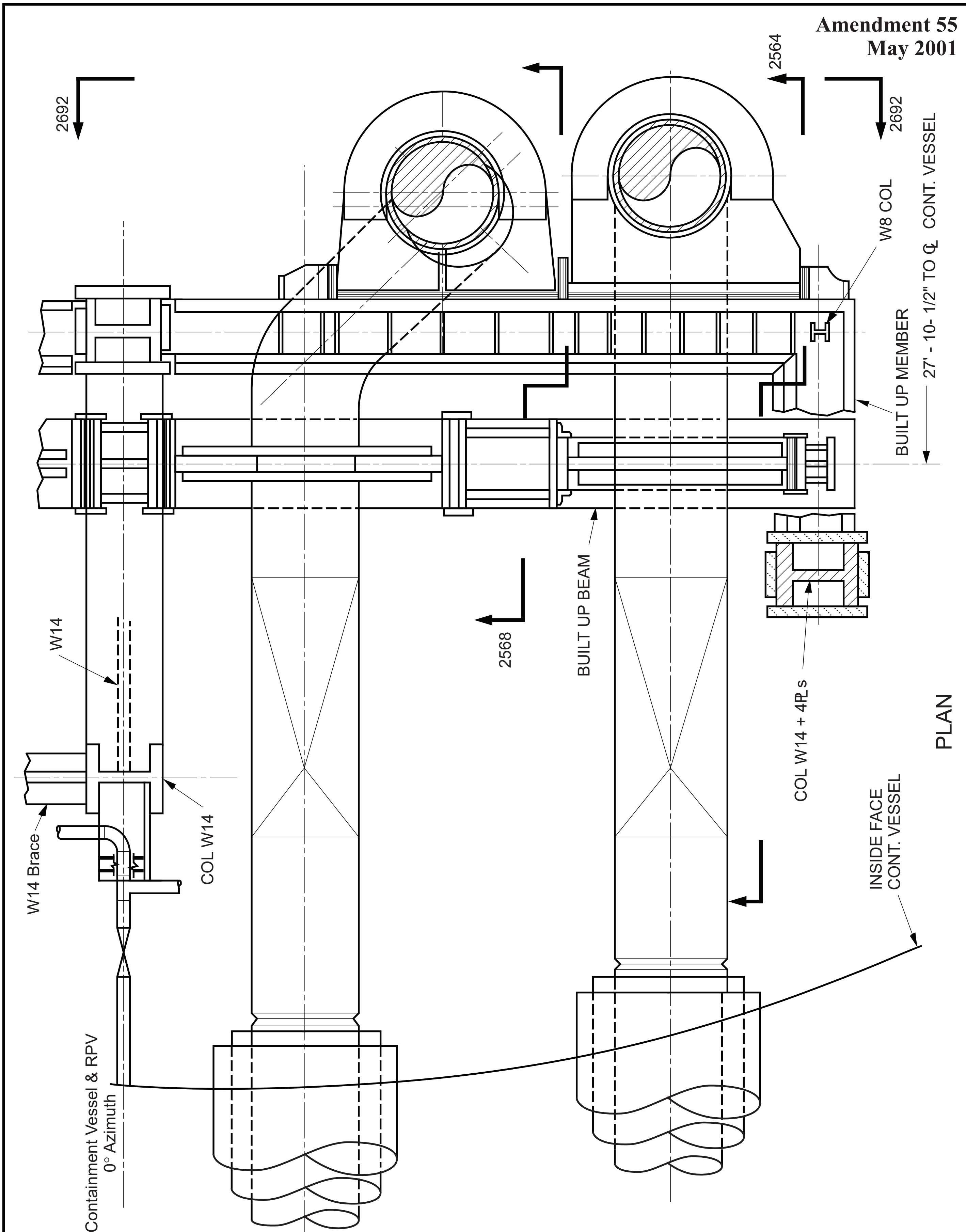
Pipe Whip Restraint Installation - Main Steam System

Draw. No.	990578.60	Rev.		Figure	3.6-23.1

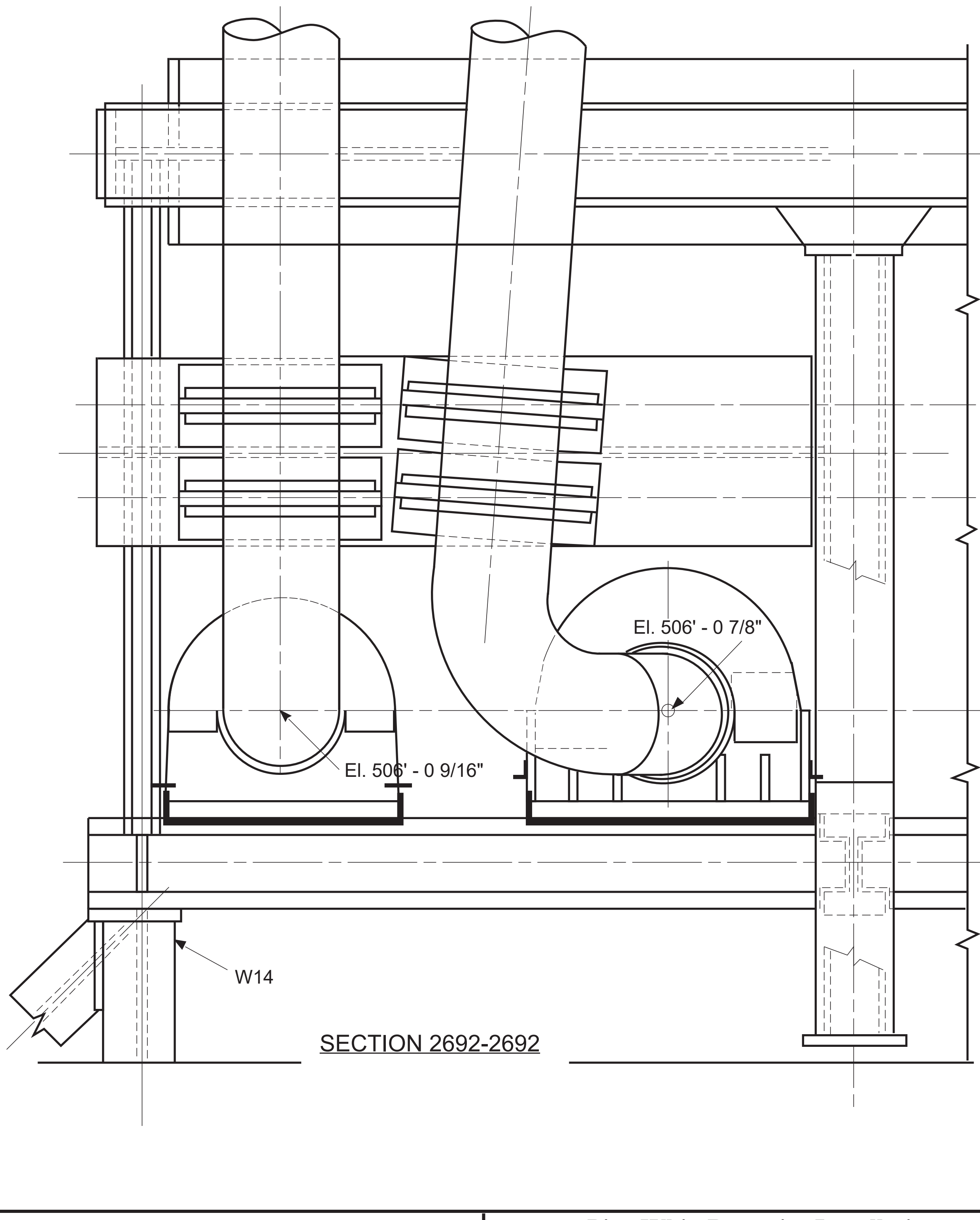
Form No. 960690



Columbia Generating Station **Final Safety Analysis Report**	**Pipe Whip Restraint Installation -** **Main Steam System**
	Draw. No. 990578.61 | Rev. | Figure 3.6-23.2



Columbia Generating Station Final Safety Analysis Report	Pipe Whip Restraint Installation - Main Steam System		
	Draw. No. 990578.69	Rev.	Figure 3.6-23.3

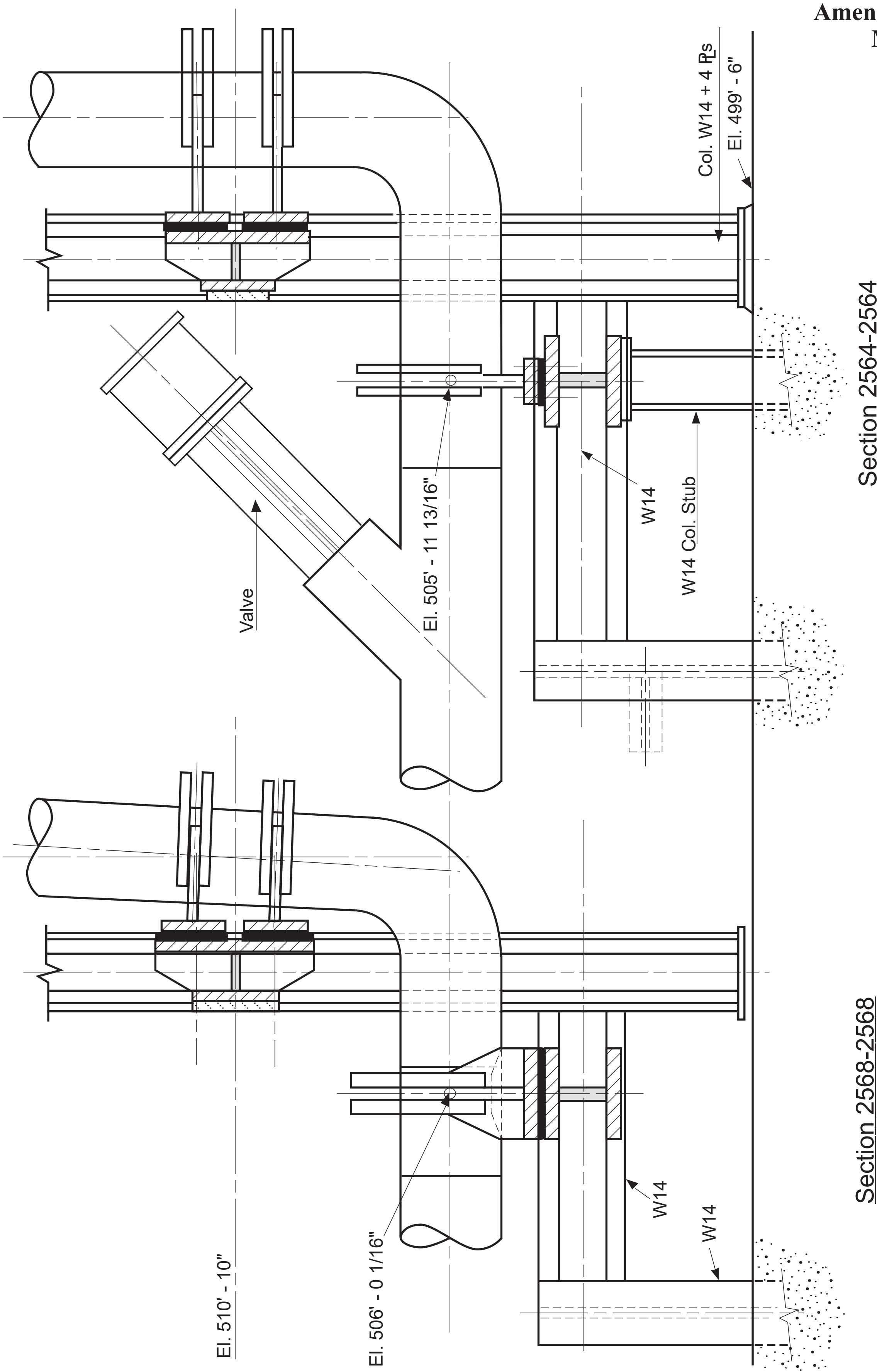


Columbia Generating Station
Final Safety Analysis Report

Pipe Whip Restraint Installation -
Main Steam System

Draw. No. 010126.47 | Rev. | Figure 3.6-23.4

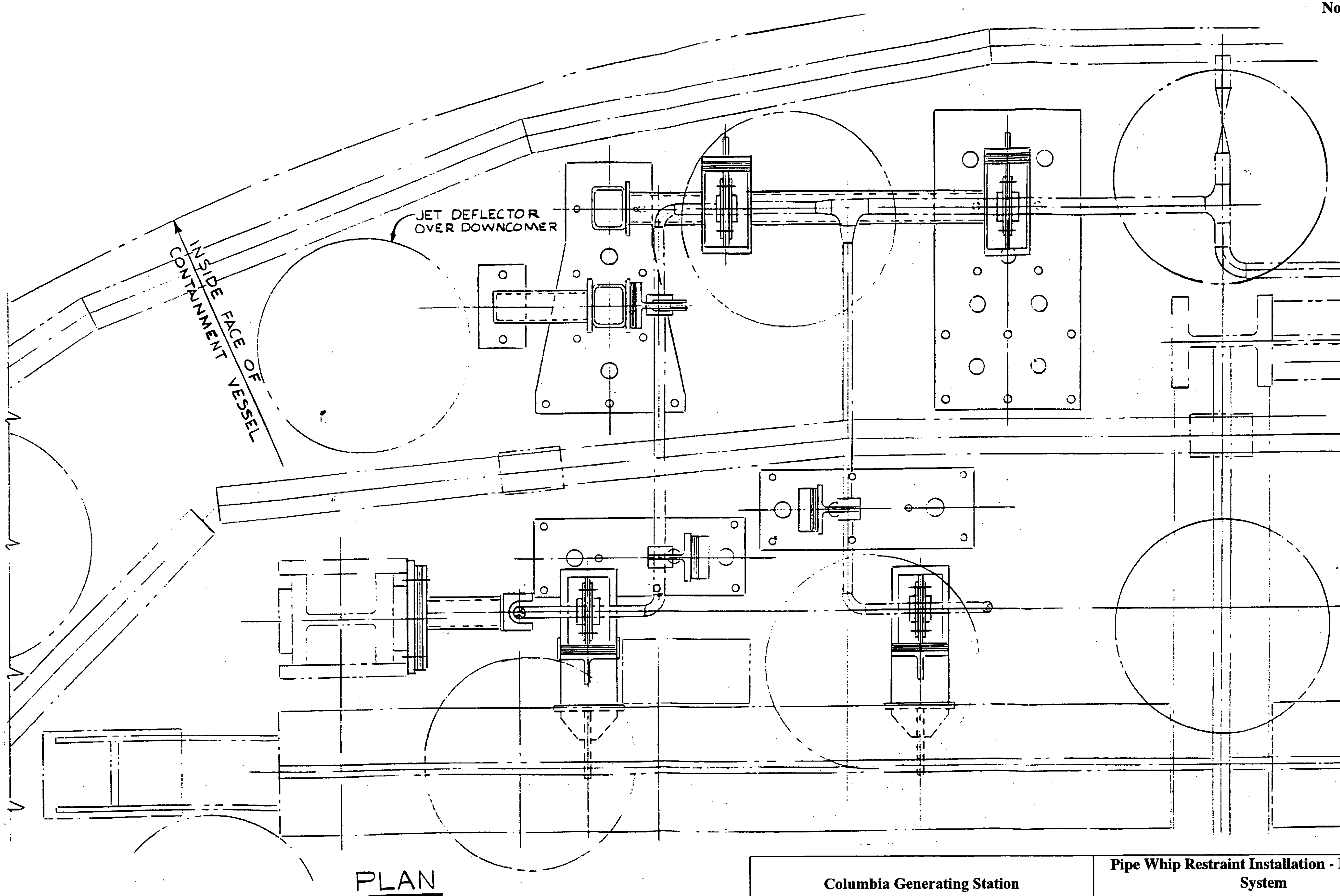
Amendment 55
May 2001



Columbia Generating Station
Final Safety Analysis Report

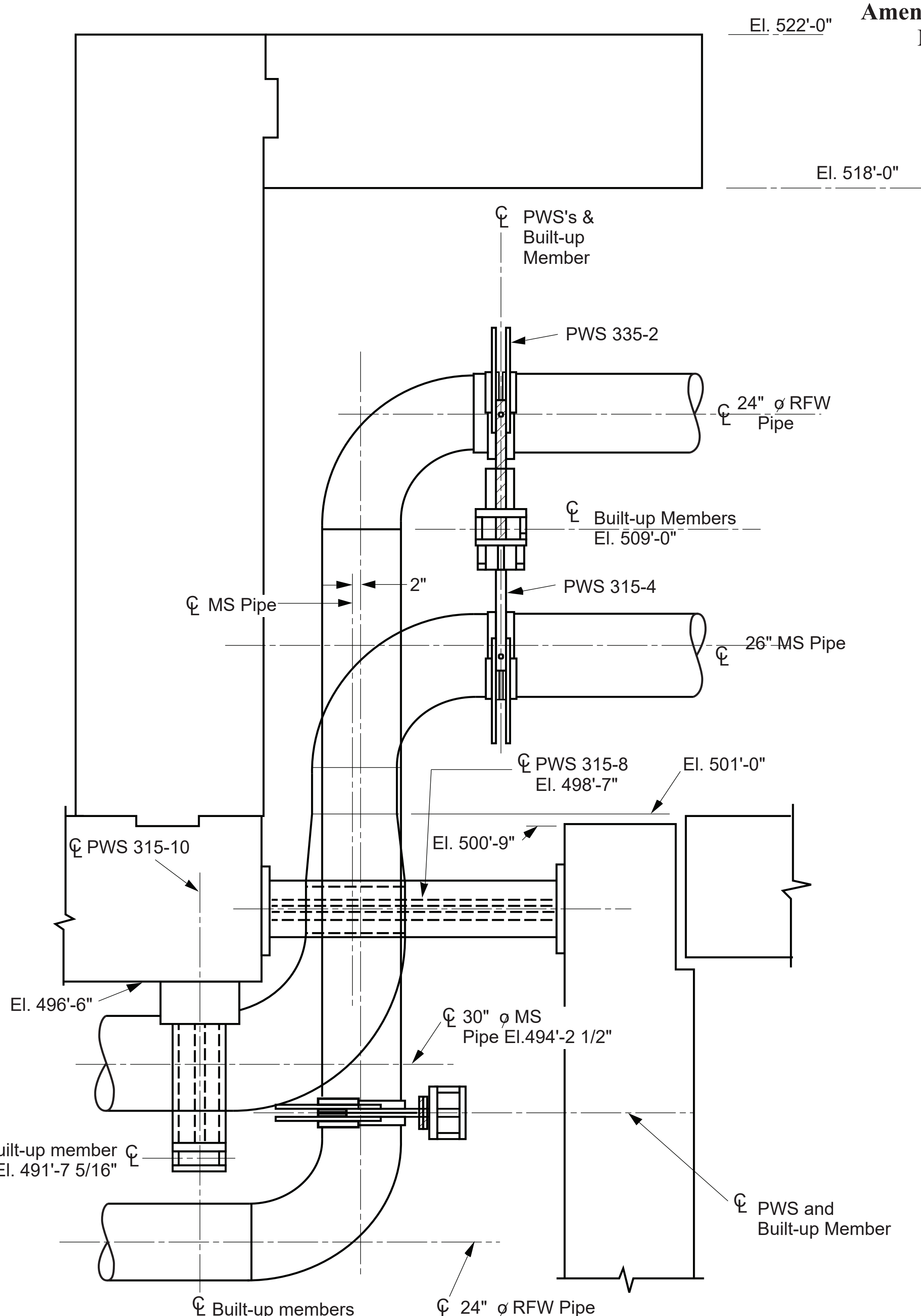
Pipe Whip Restraint Installation -
Main Steam System

Draw. No.	010126.50	Rev.		Figure	3.6-23.5



Columbia Generating Station Final Safety Analysis Report	Pipe Whip Restraint Installation - Main Steam System	
	Draw. No. 020552.04	Rev.
	Figure 3.6-23.6	

Amendment 55
May 2001

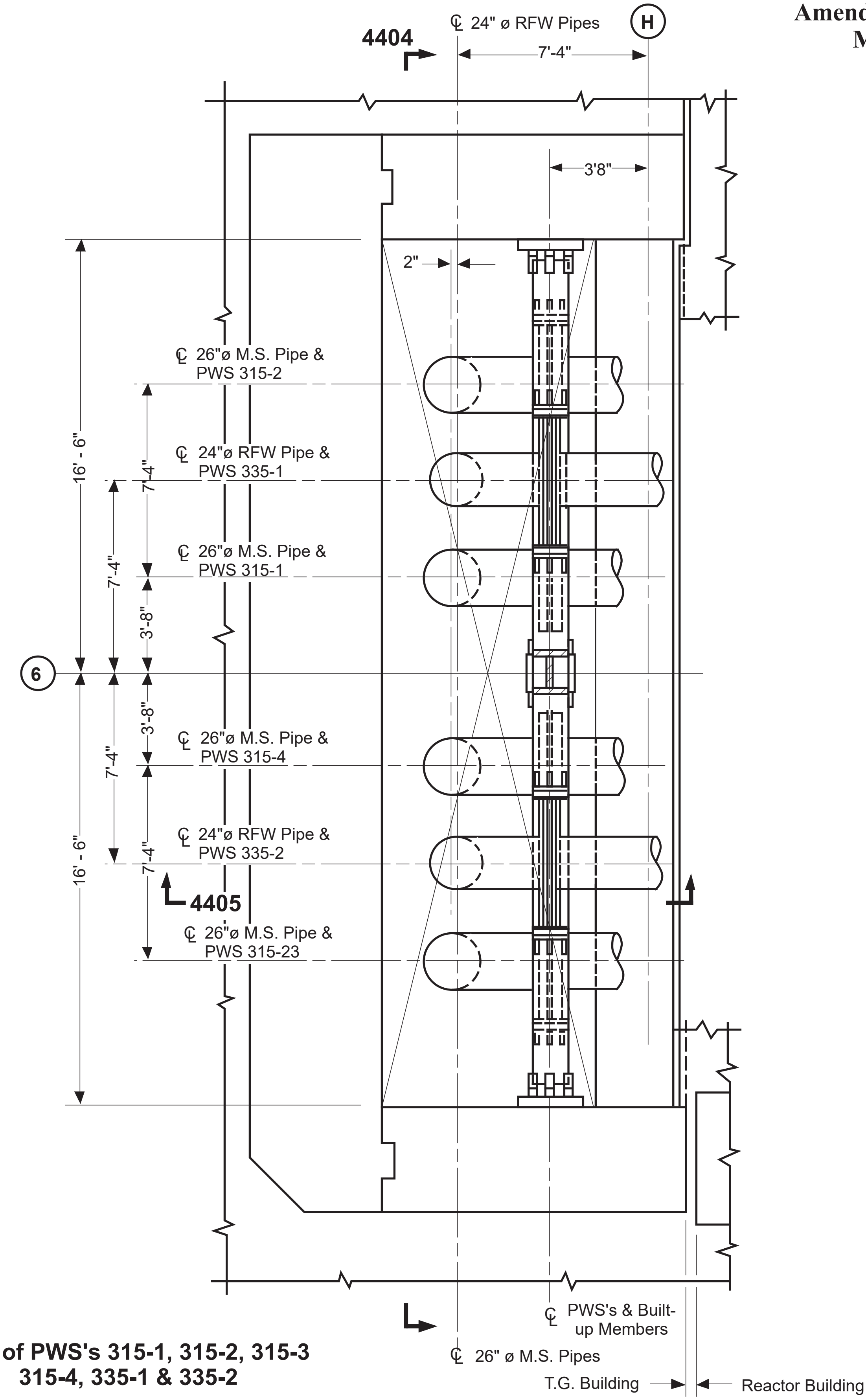


SECTION 4405-4405

Columbia Generating Station
Final Safety Analysis Report

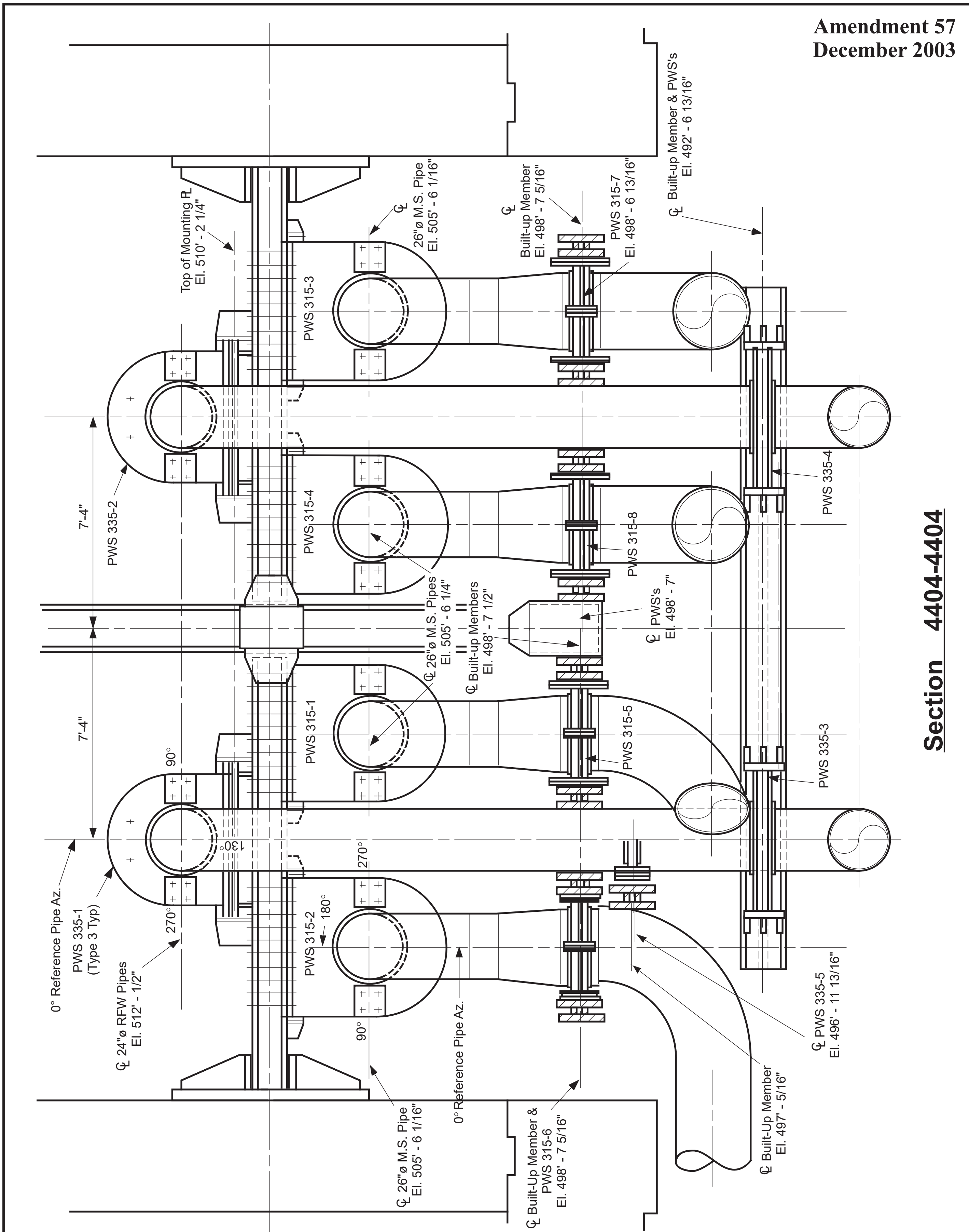
Pipe Whip Restraint Installation - Main Steam and Reactor Feedwater in Main Steam Tunnel

Draw. No. 990578.68	Rev.	Figure 3.6-24.1



**Plan of PWS's 315-1, 315-2, 315-3
315-4, 335-1 & 335-2**

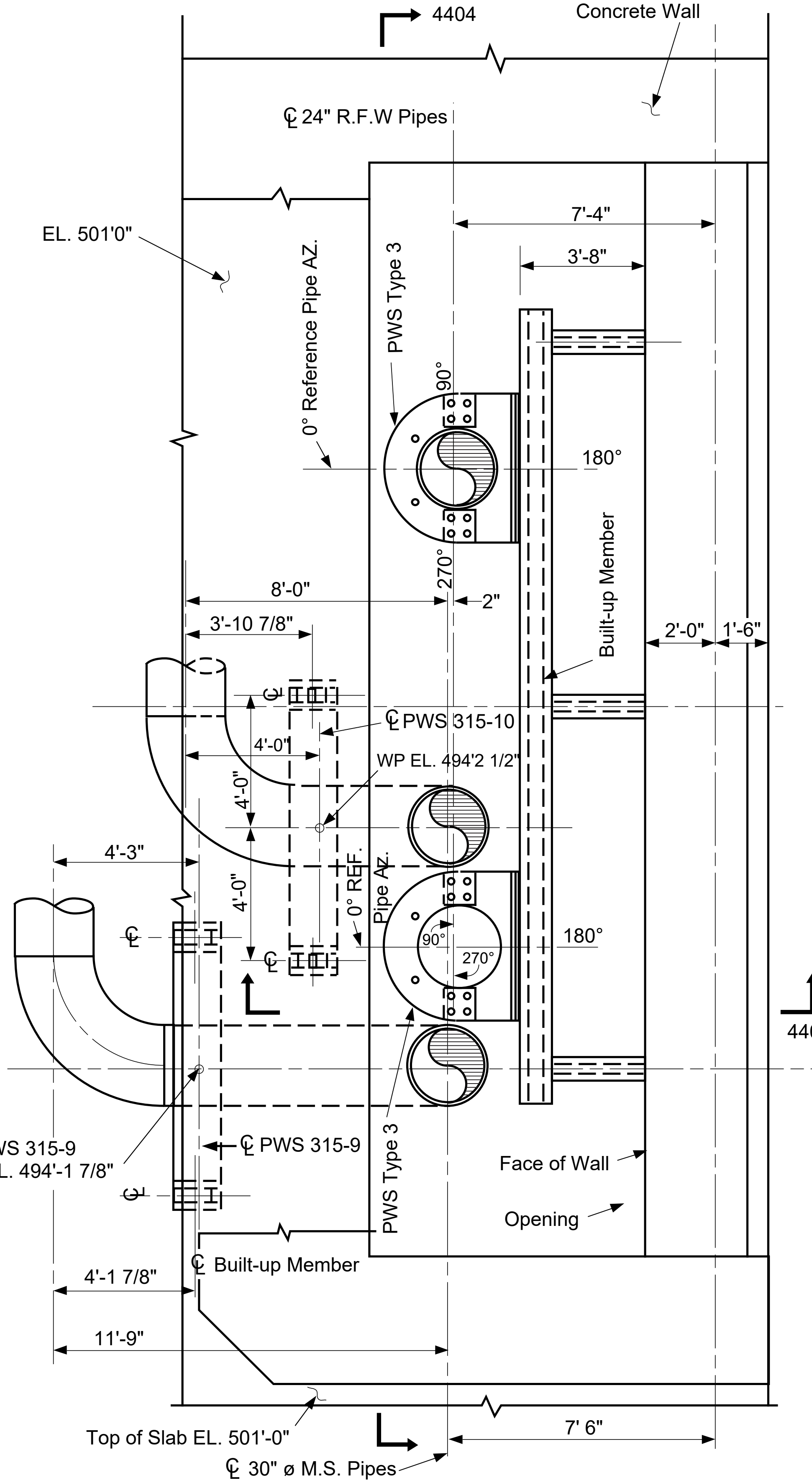
Columbia Generating Station Final Safety Analysis Report	Pipe Whip Restraint Installation - Main Steam and Reactor Feedwater in Main Steam Tunnel
	Draw. No. 010126.36 / Rev. / Figure 3.6-24.2



Columbia Generating Station
Final Safety Analysis Report

Pipe Whip Restraint Installation - Main Steam and Reactor Feedwater in Main Steam Tunnel

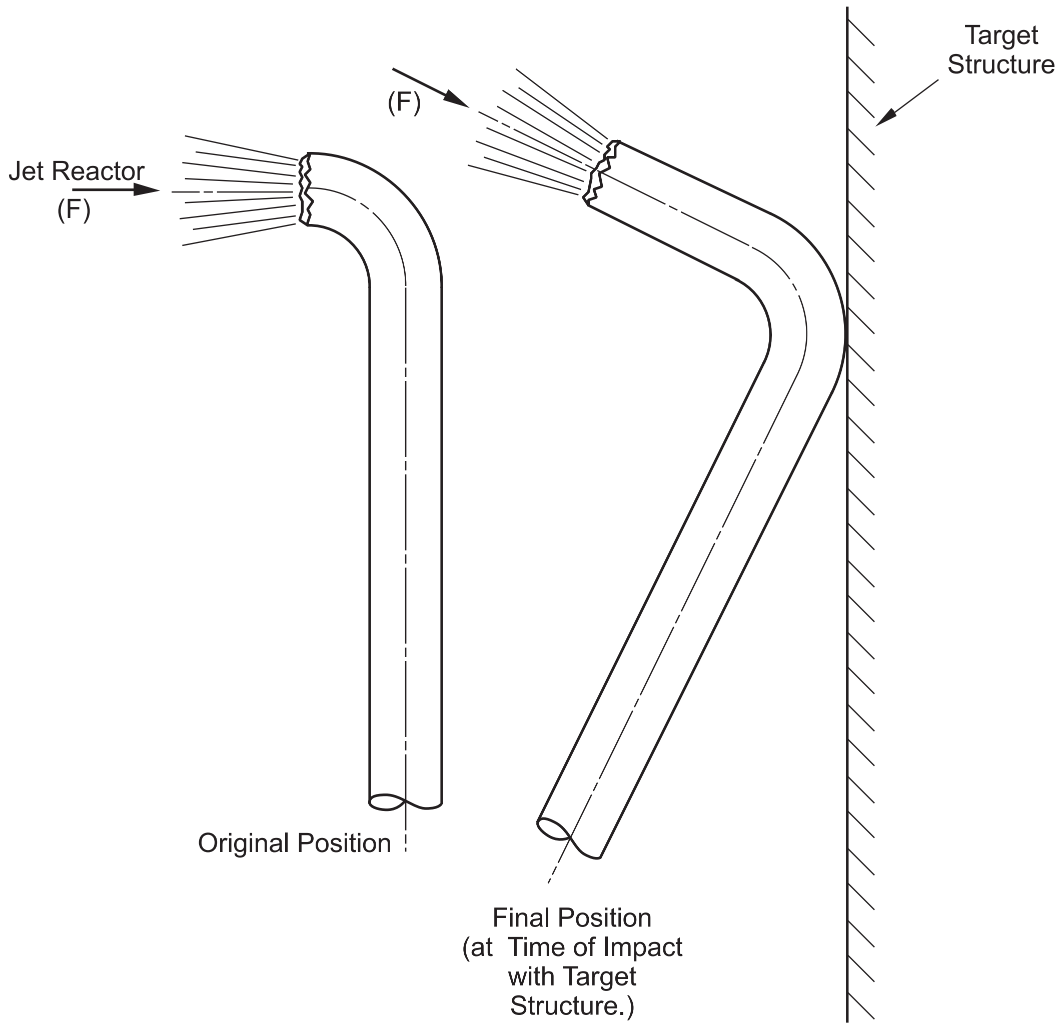
Draw. No. 010126.45 | Rev. | Figure 3.6-24.3



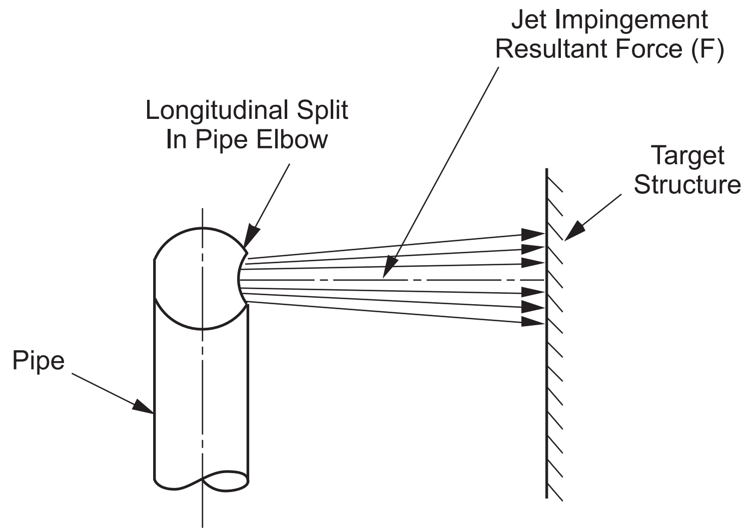
Columbia Generating Station
Final Safety Analysis Report

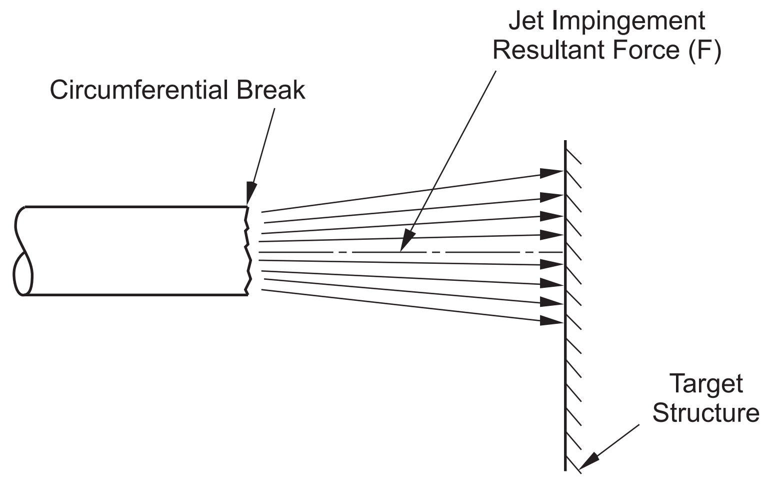
Pipe Whip Restraint Installation - Main Steam and Reactor Feedwater in Main Steam Tunnel

Draw. No. 010126.21	Rev.	Figure 3.6-24.4



Note: Effects on target structure are:
(1) A jet reaction force, F (for time history description, see Fig. 3.6-27, and
(2) Impact due to energy accumulated by pipe while being accelerated from original to final position.
(3) Circumferential break is shown.





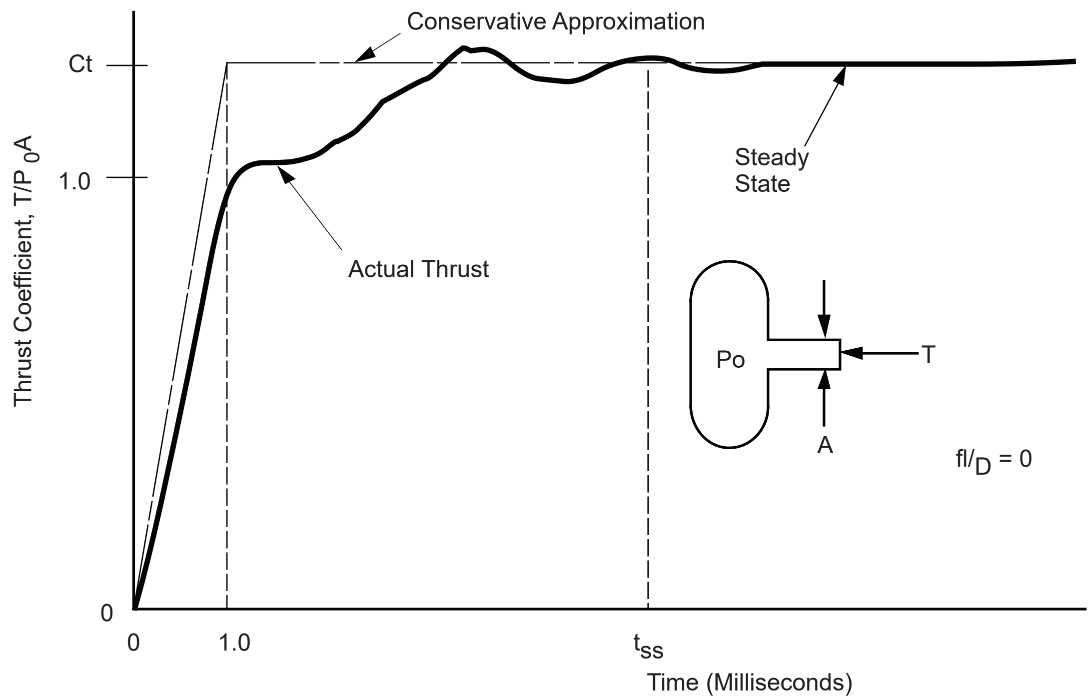
Note: For Time History Descriptions
See Fig. 3.6-27

Columbia Generating Station
Final Safety Analysis Report

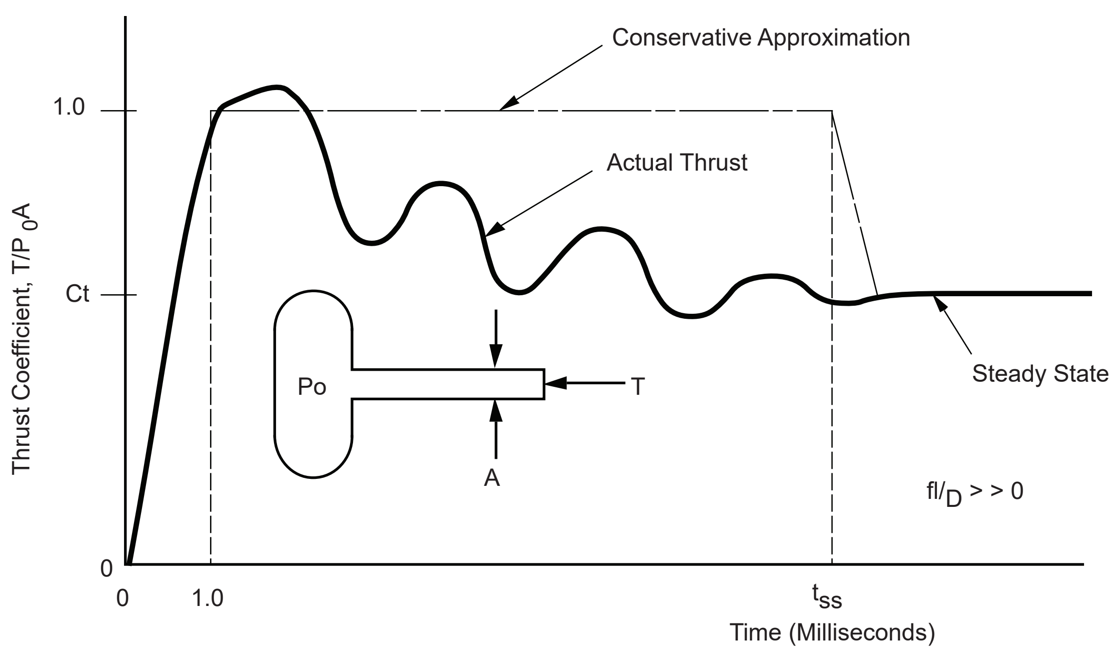
Jet Impingement on Target

Draw. No. 910402.23	Rev.	Figure 3.6-26

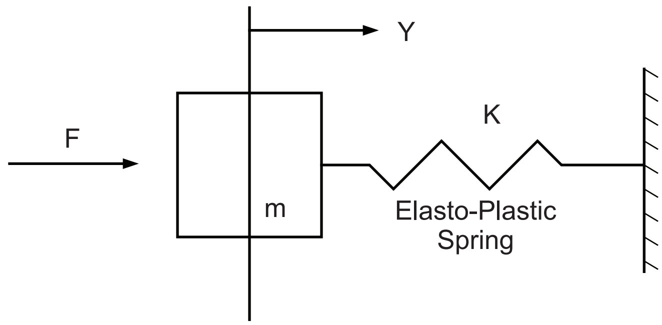
Trust Force Transient, Very Low Friction Flow



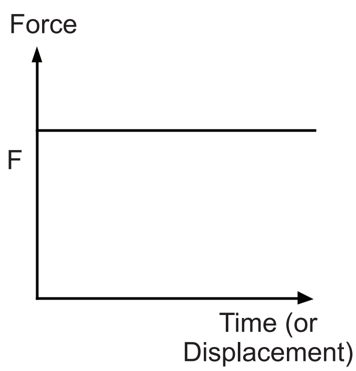
Thrust Force Transient, Friction Flow



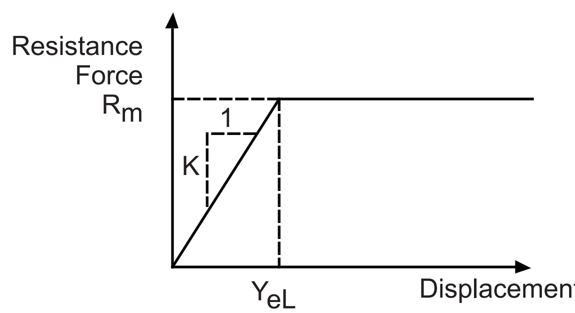
Columbia Generating Station Final Safety Analysis Report	Time History of Jet Impingement and Reaction Force		
	Draw. No. 990306.42	Rev.	Figure 3.6-27



(A) Single degree of freedom mathematical idealization for a structure.

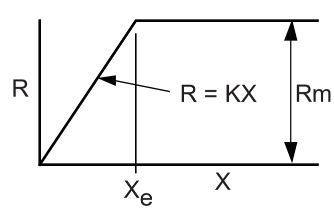
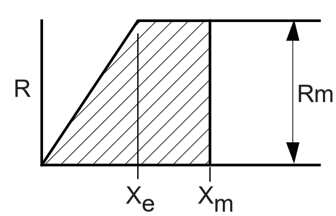
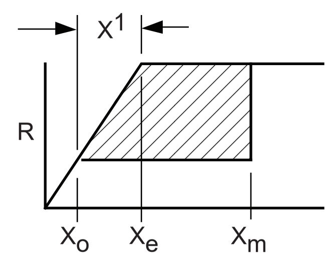


(B) Step Function Load



(C) Resistance Function

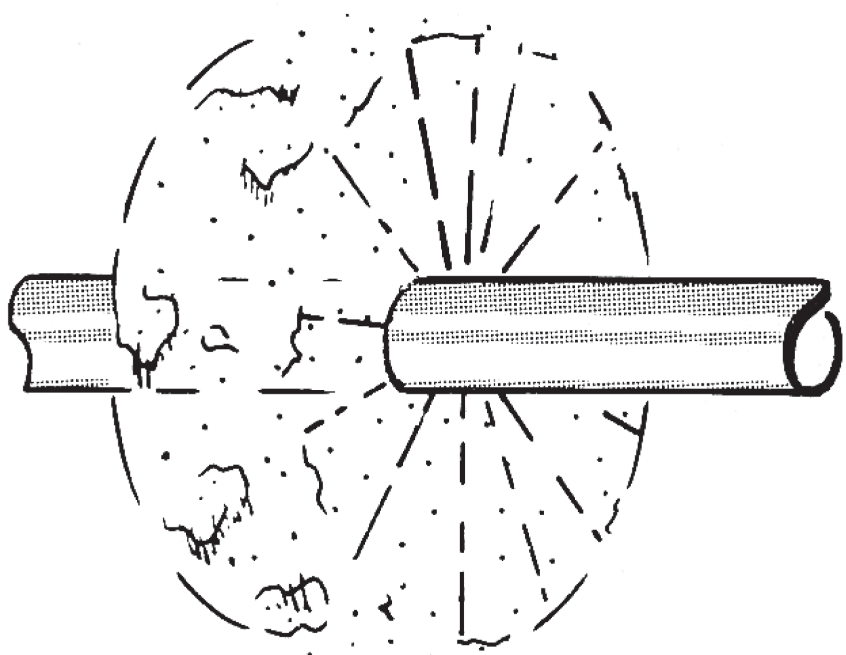
Columbia Generating Station **Final Safety Analysis Report**	**Structural Response to a Step Function Loading**
	Draw. No. 970187.81 | Rev. | Figure 3.6-28

Response	Resistance-Displacement Function	Available Strain Energy without other Loading	Available Strain Energy with other Loading
Elasto-Plastic			

Note: Shaded Area (Strain Energy)
Must Equal E_s (from 3.6.1.6.3.2)

Resistance-Displacement Functions with Associated Structural Response (with and without Effect of Other Loads)

Draw. No. 990306.43 | Rev. | Figure 3.6-29



Jet from Circumferential Break with Ends Restrained (Fan Jet)

Draw. No. 020361.14	Rev.	Figure 3.6-30

Note: The wall rebound force is used in a direction opposite to the pipe or pipe jet impact load.

To determine R_m :

$$\frac{Y_{eL_2}}{R_{m_2}} = \frac{Y_{eL_1}}{R_{m_1}}$$

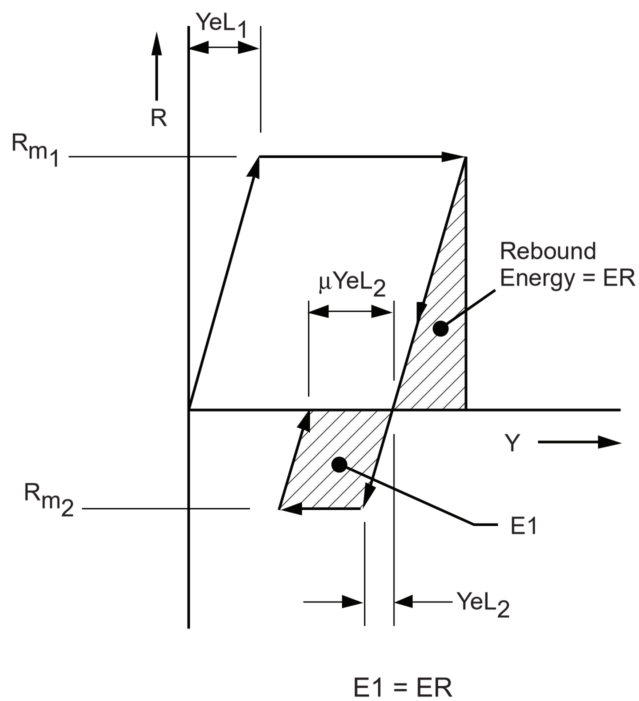
$$Y_{eL_2} = Y_{eL_1}\left(\frac{R_{m_2}}{R_{m_1}}\right)$$

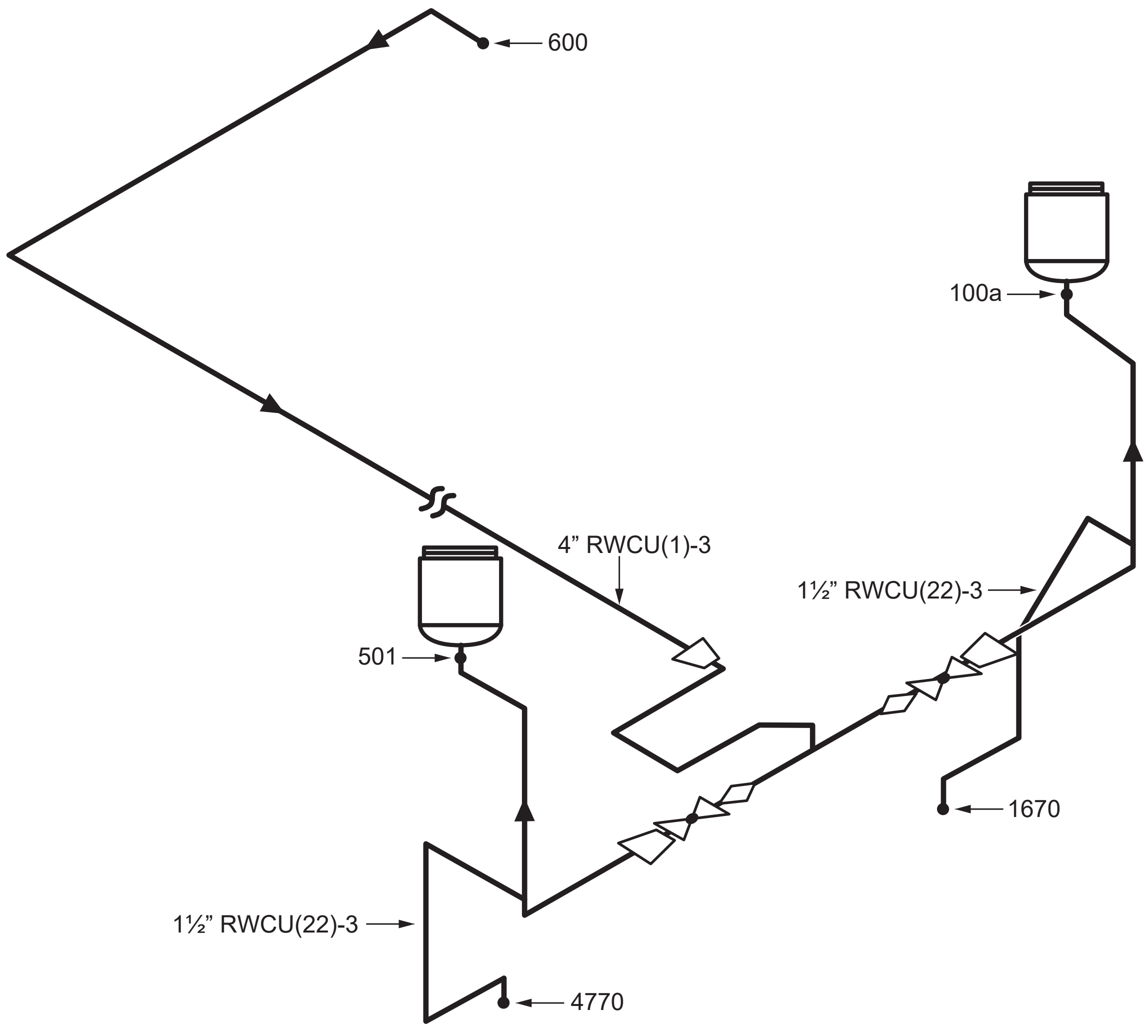
$E1 = \mu Yel_2\ Rm_2 = ER = 1/2\ yel._1\ Rm_1$

or $\mu YeL_1\left(\frac{R_{m_2}}{R_{m_1}}\right) Rm_2 = 1/2\ yel._1\ Rm_1$

Therefore, $Rm_2^{\,2} = 1/2\mu\ (RM_1^{\,2})$

or $Rm_2 = \sqrt{\frac{1}{2\mu}}\ Rm_1$



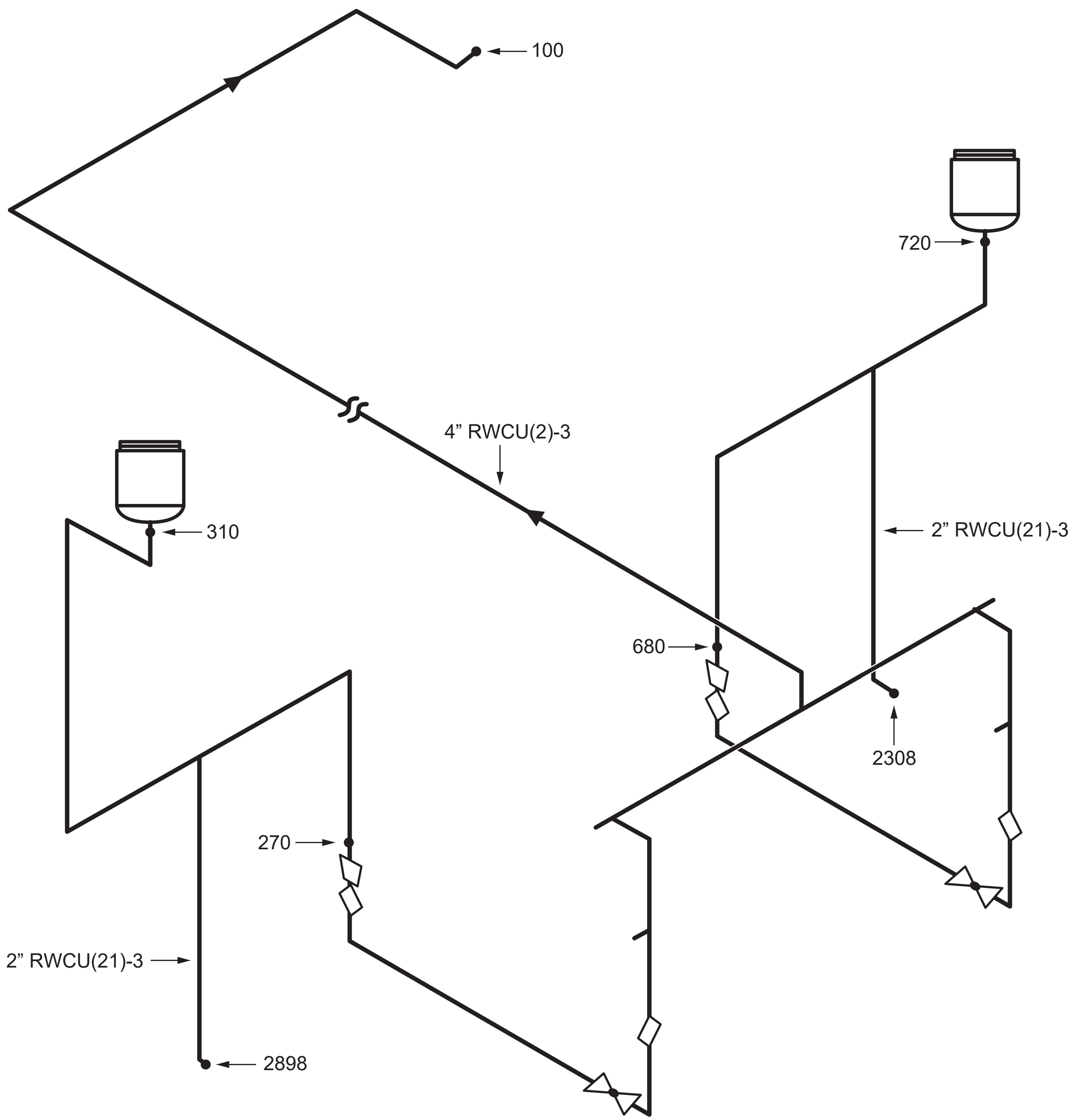


Based on M200-500,
ME-02-16-17, ME-02-16-18

Columbia Generating Station
Final Safety Analysis Report

Reactor Water Clean-Up
RWCU-DM-1A & 1B Influent

Draw. No. 160058.00 | Rev. | Figure 3.6-38.4

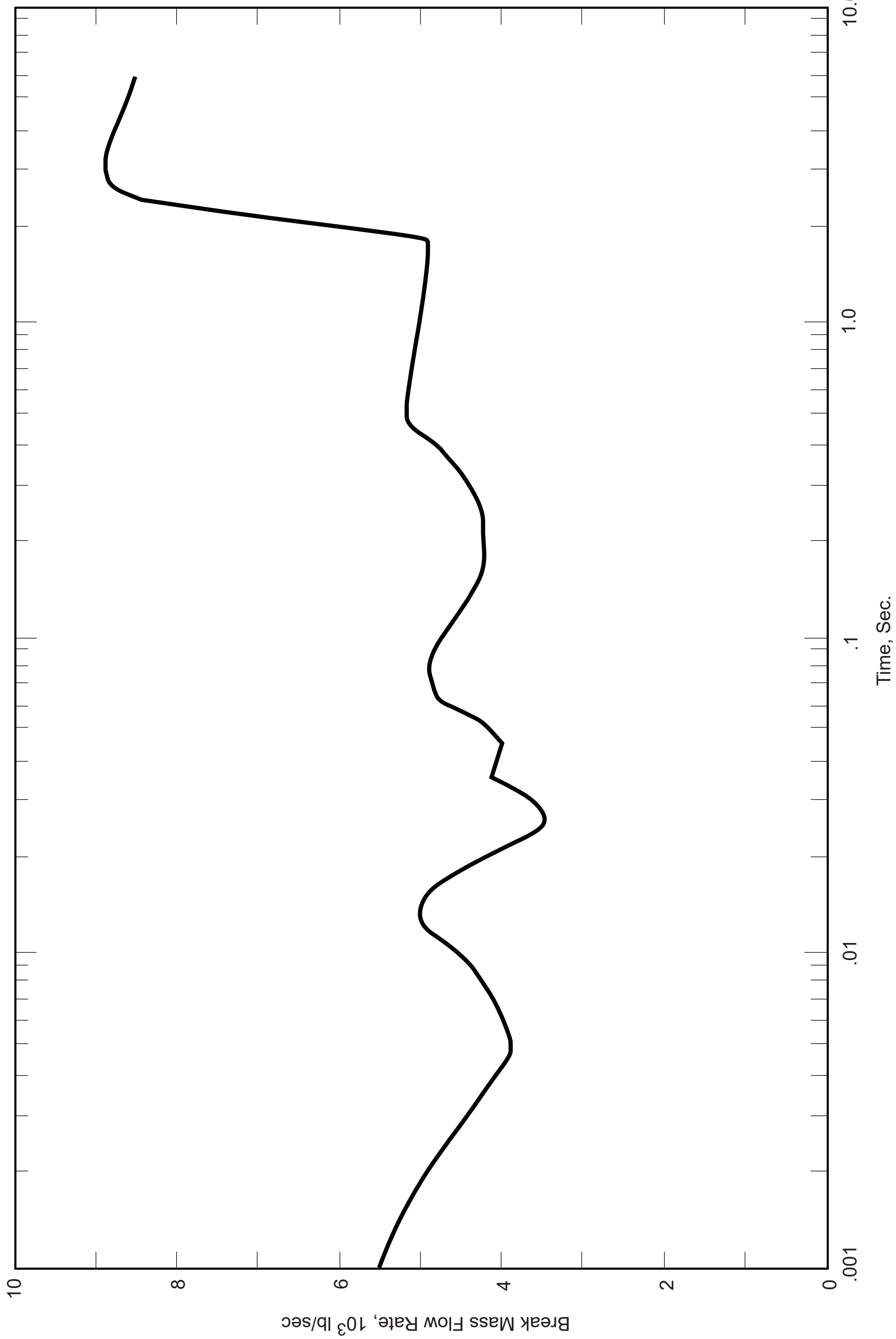


Based on M200-500,
ME-02-16-17, ME-02-16-18

Columbia Generating Station
Final Safety Analysis Report

Reactor Water Clean-Up
RWCU-DM-1A & 1B Effluent

Draw. No. 160058.01	Rev.	Figure 3.6-38.5

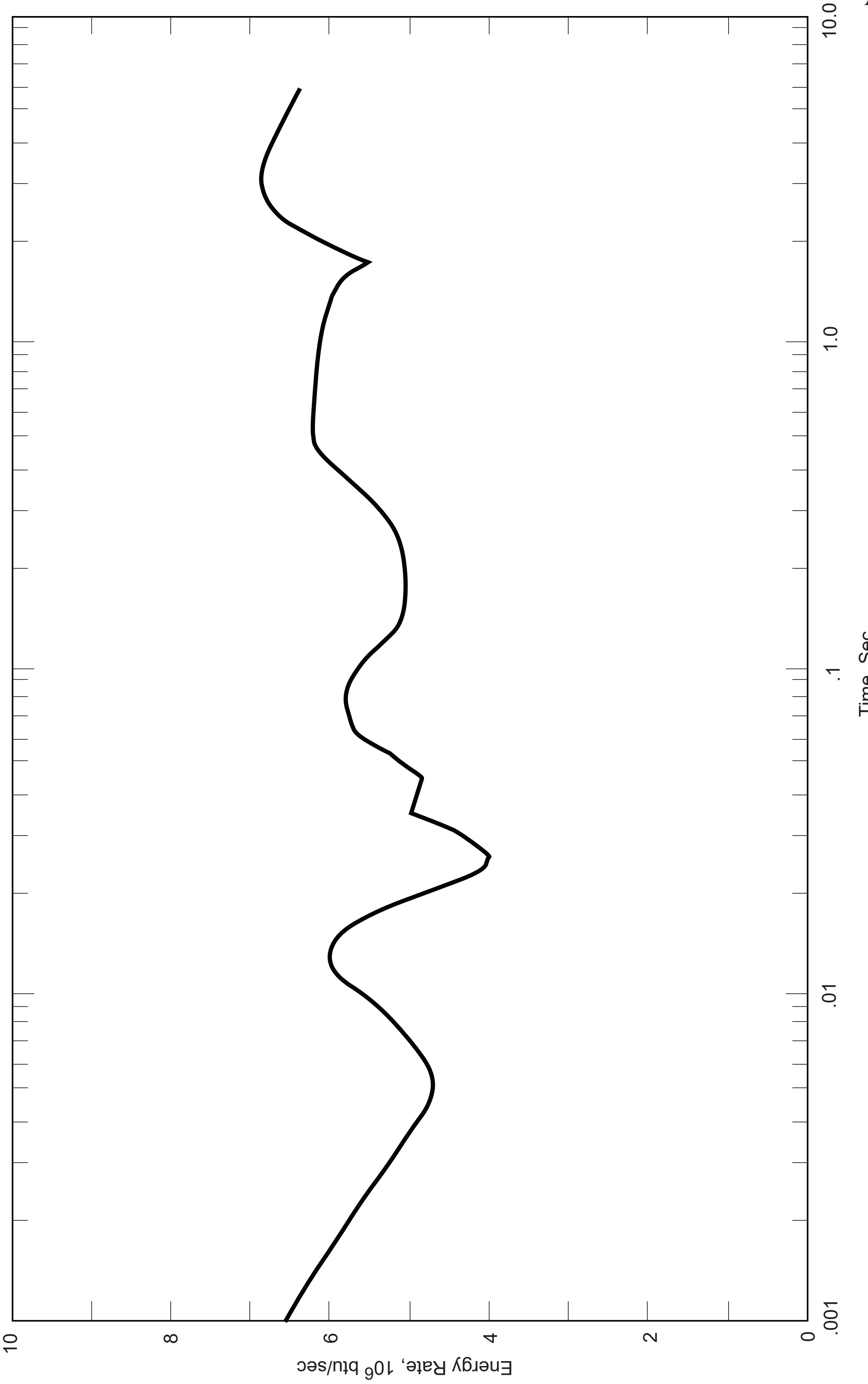


Columbia Generating Station
Final Safety Analysis Report

Blowdown Mass Flow Rate from Postulated Crack in 26 in. Main Steam Line - Outside Primary Containment in Main Steam Tunnel

Draw. No. 990306.49	Rev.	Figure 3.6-61

Amendment 55
May 2001

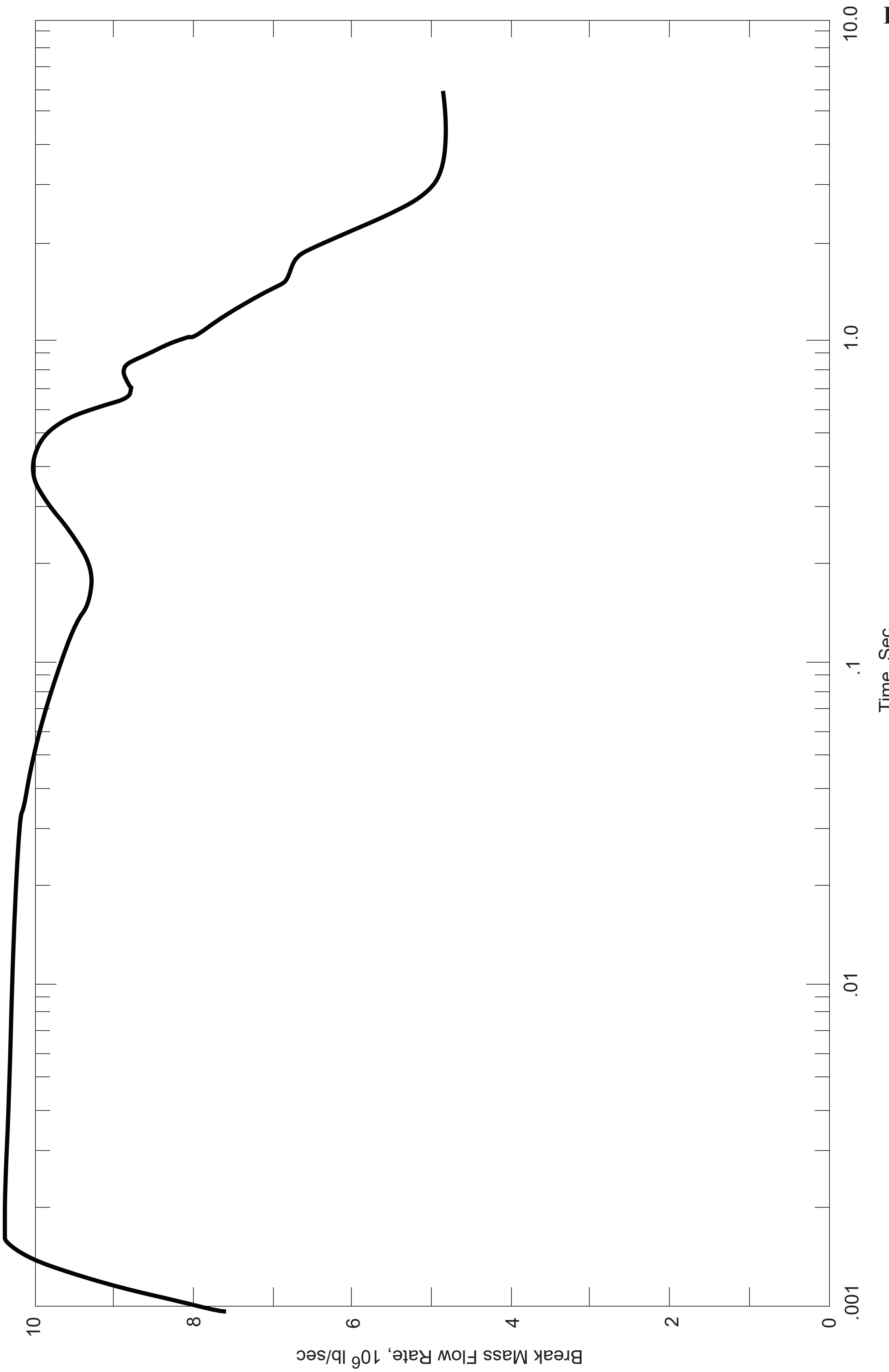


Columbia Generating Station
Final Safety Analysis Report

Energy Release Rate from Postulated Crack in 26 in. Main Steam Line - Outside Primary Containment in Main Steam Tunnel

Draw. No. 990306.51 | Rev. | Figure 3.6-62

Form No. 960690

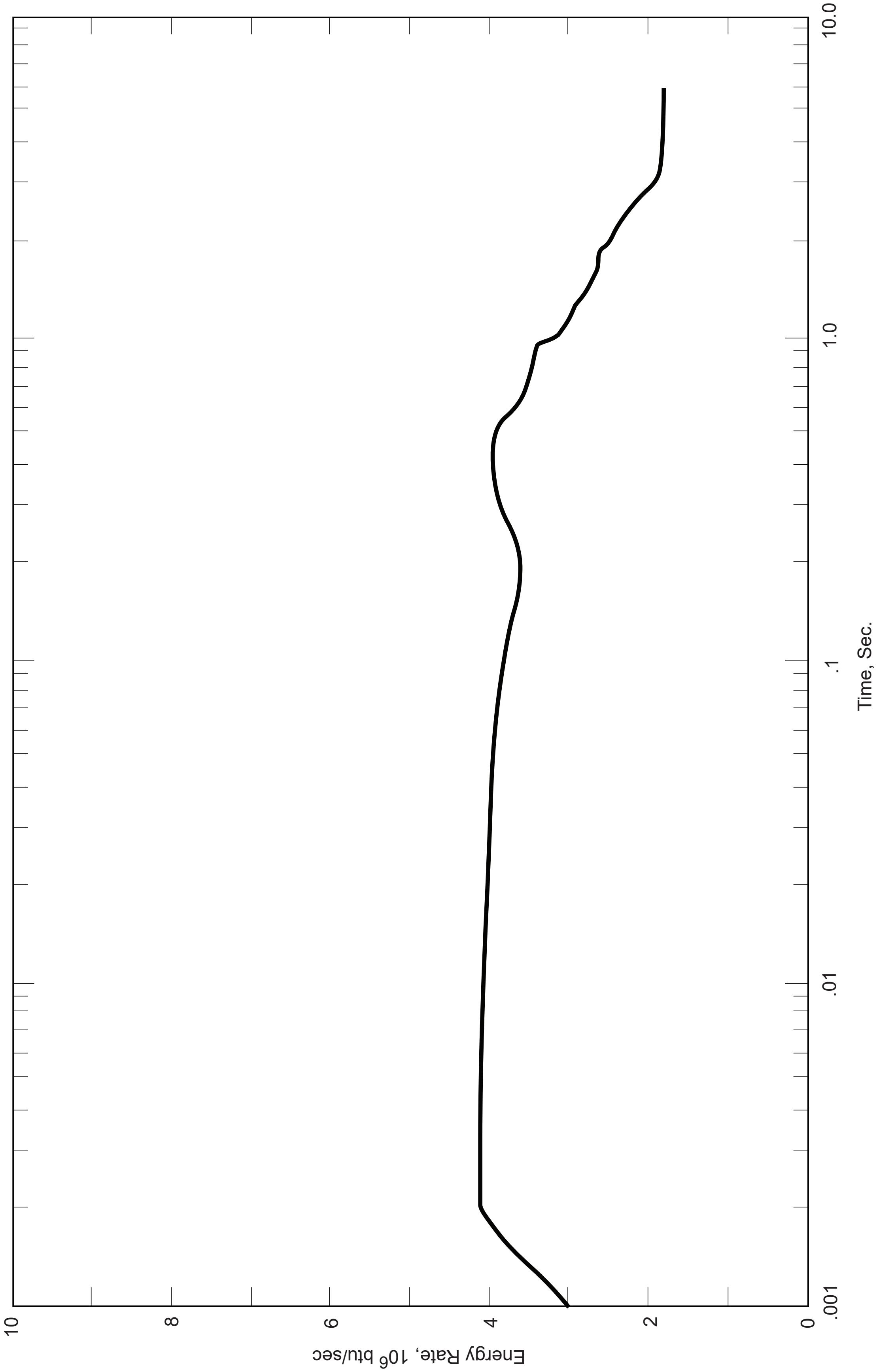


Columbia Generating Station
Final Safety Analysis Report

Blowdown Mass Flow Rate from Postulated Crack in 24 in. Reactor Feed Line - Outside Primary Containment in Main Steam Tunnel

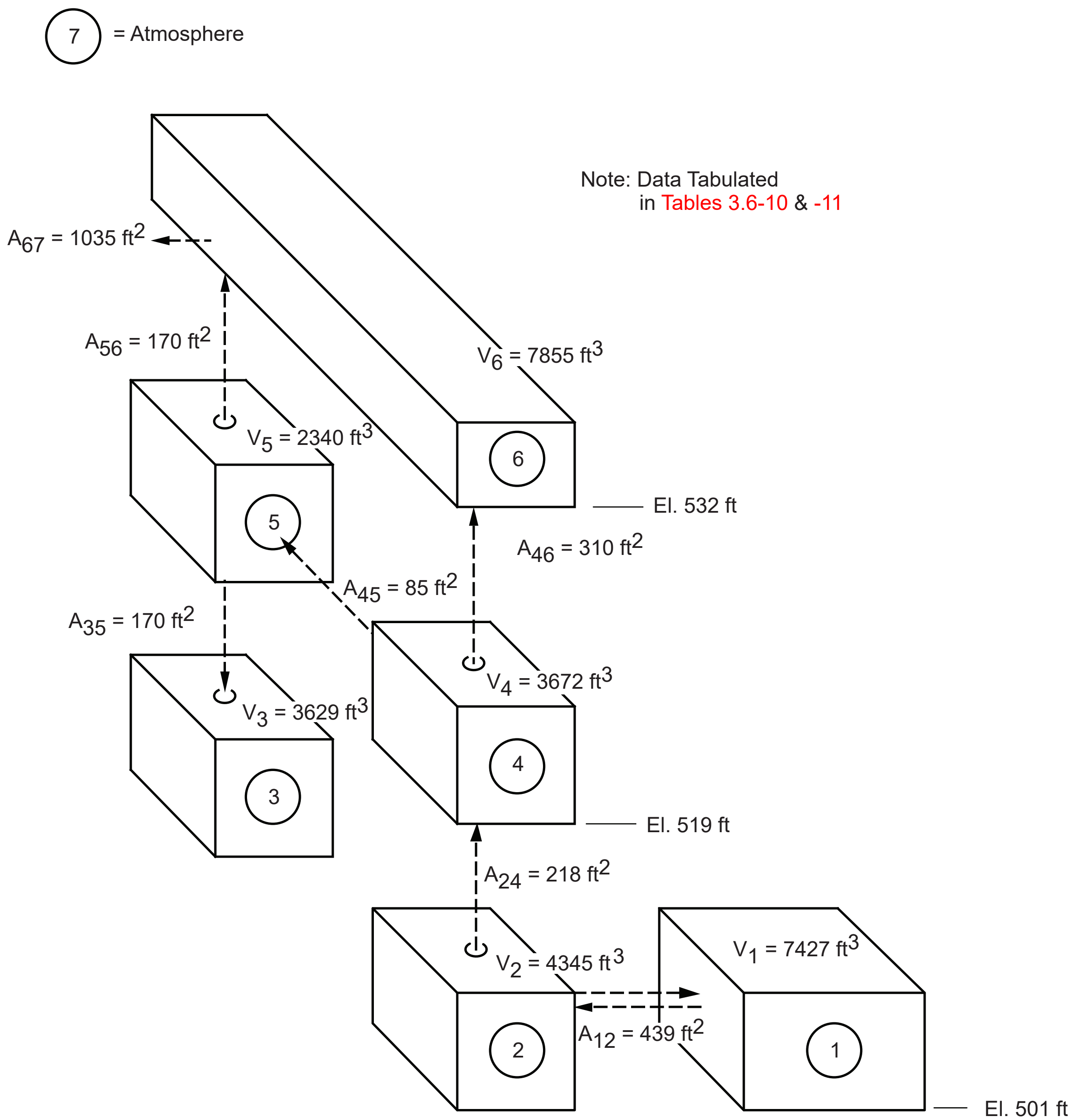
Draw. No. 990306.52	Rev.	Figure 3.6-63

Amendment 55
May 2001

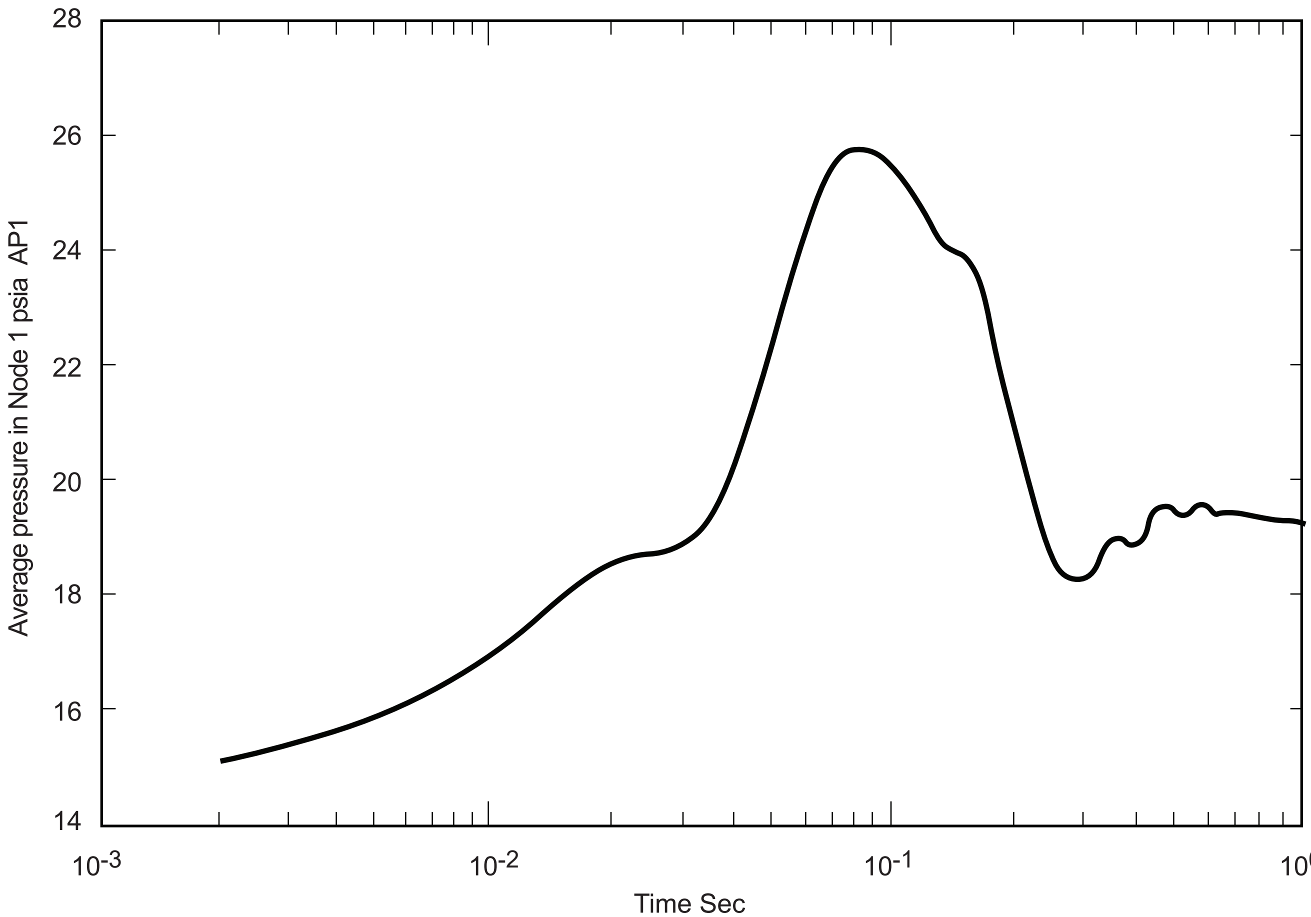


Columbia Generating Station **Final Safety Analysis Report**	**Energy Release Rate from Postulated Crack in 24 in. Reactor Feedwater Line - Outside Primary Containment in Main Steam Tunnel**
	Draw. No. **990306.53** Rev. Figure **3.6-64**

Form No. 960690



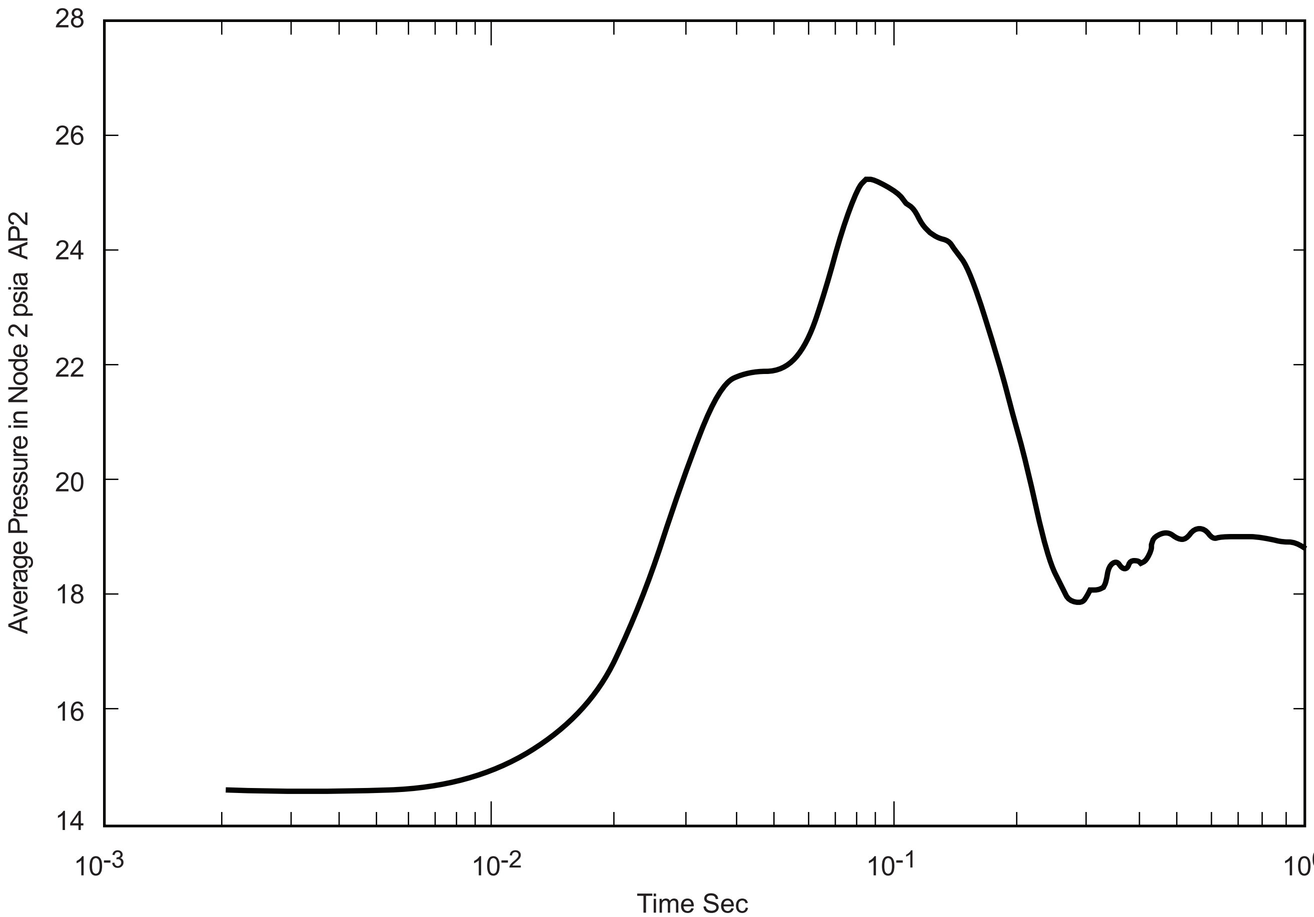
Nodalization Scheme for Postulated Pipe Break in Main Steam Tunnel



Columbia Generating Station
Final Safety Analysis Report

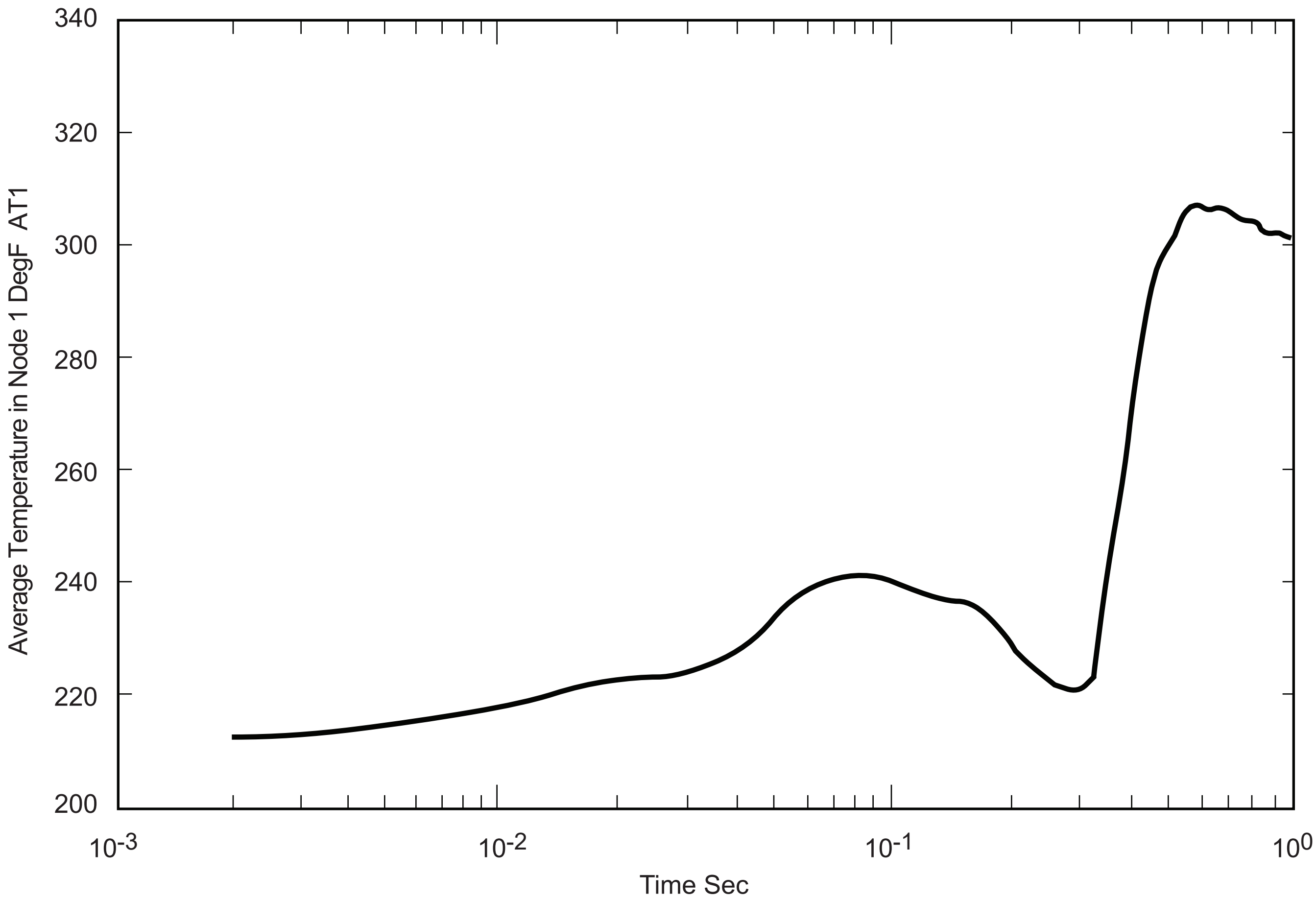
Pressure Transient in Node 1 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel

Draw. No. 970187.87	Rev.	Figure 3.6-68



Pressure Transient in Node 2 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel

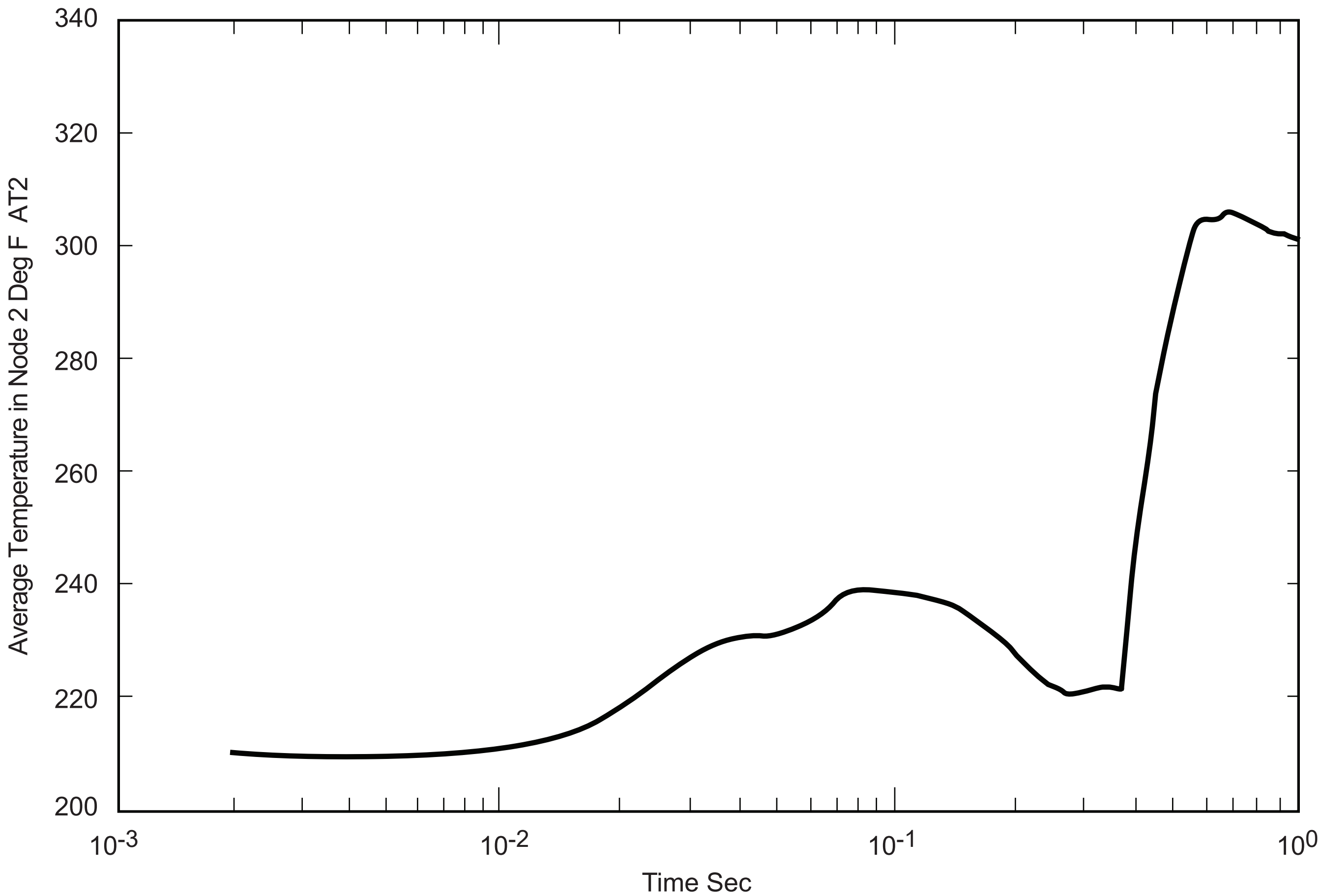
Draw. No. 970187.88 | Rev. | Figure 3.6-69



Columbia Generating Station
Final Safety Analysis Report

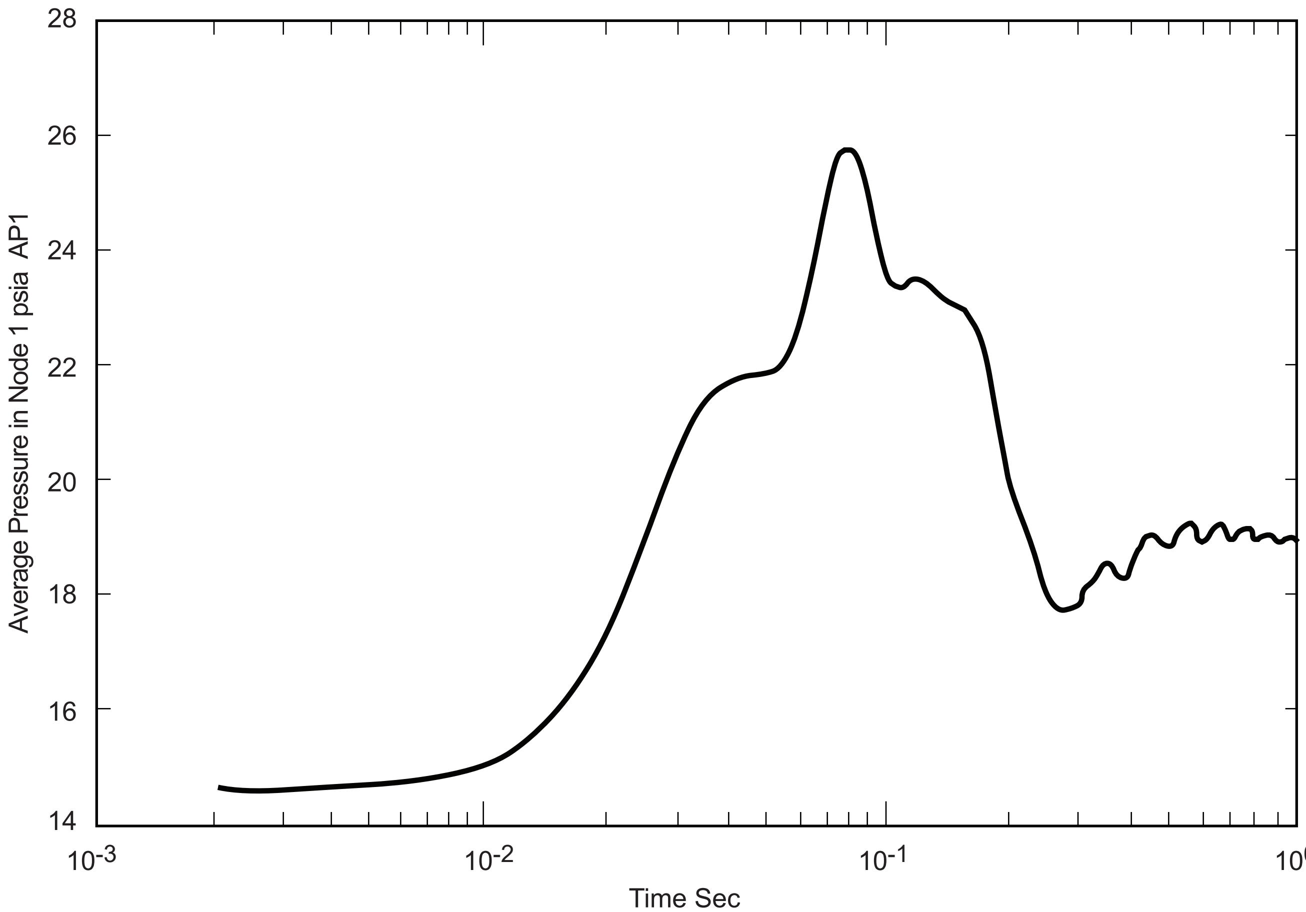
Temperature Transient in Node 1 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel

Draw. No. 970187.89 | Rev. | Figure 3.6-70



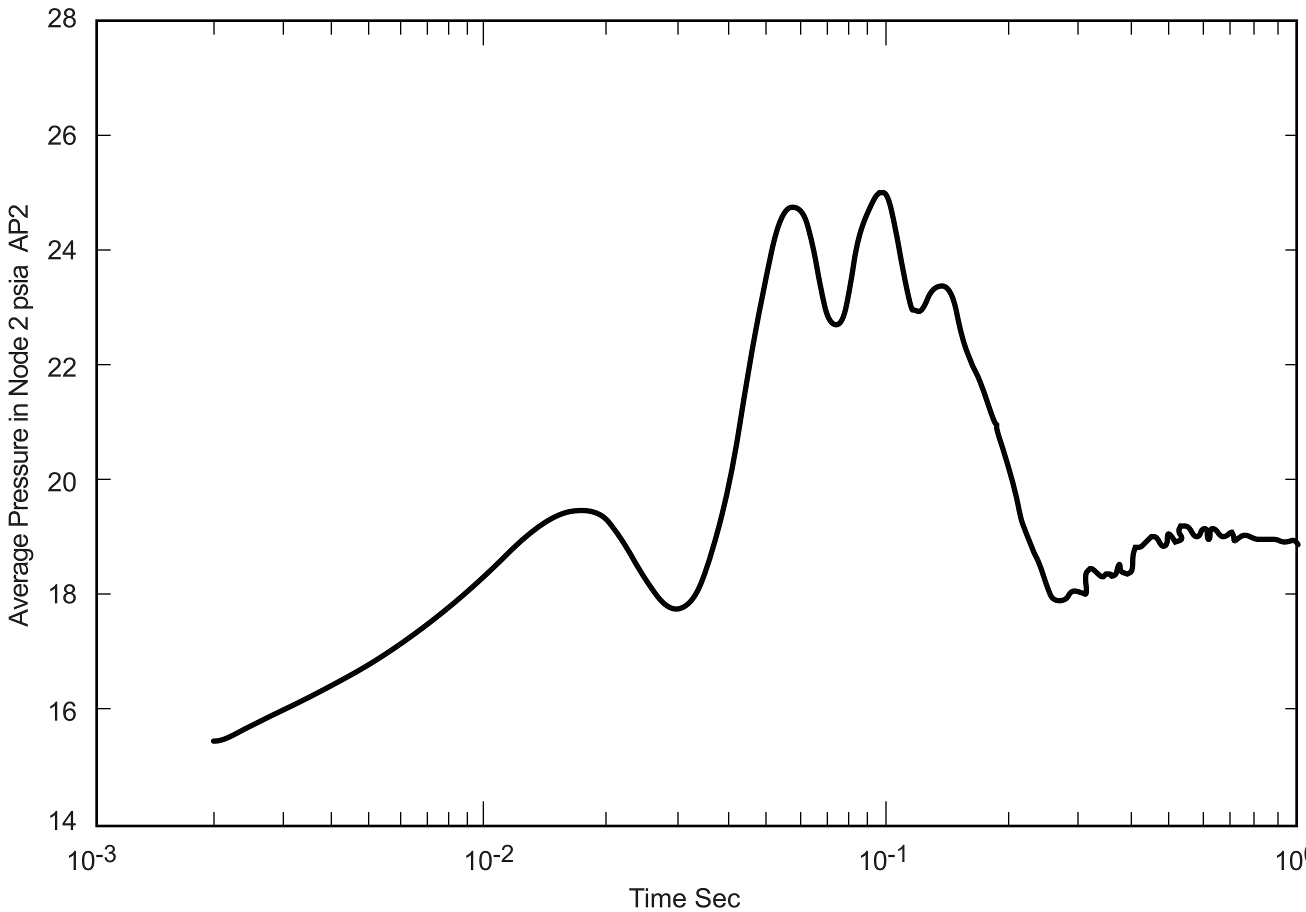
Temperature Transient in Node 2 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel

Draw. No. 970187.90 | Rev. | Figure 3.6-71

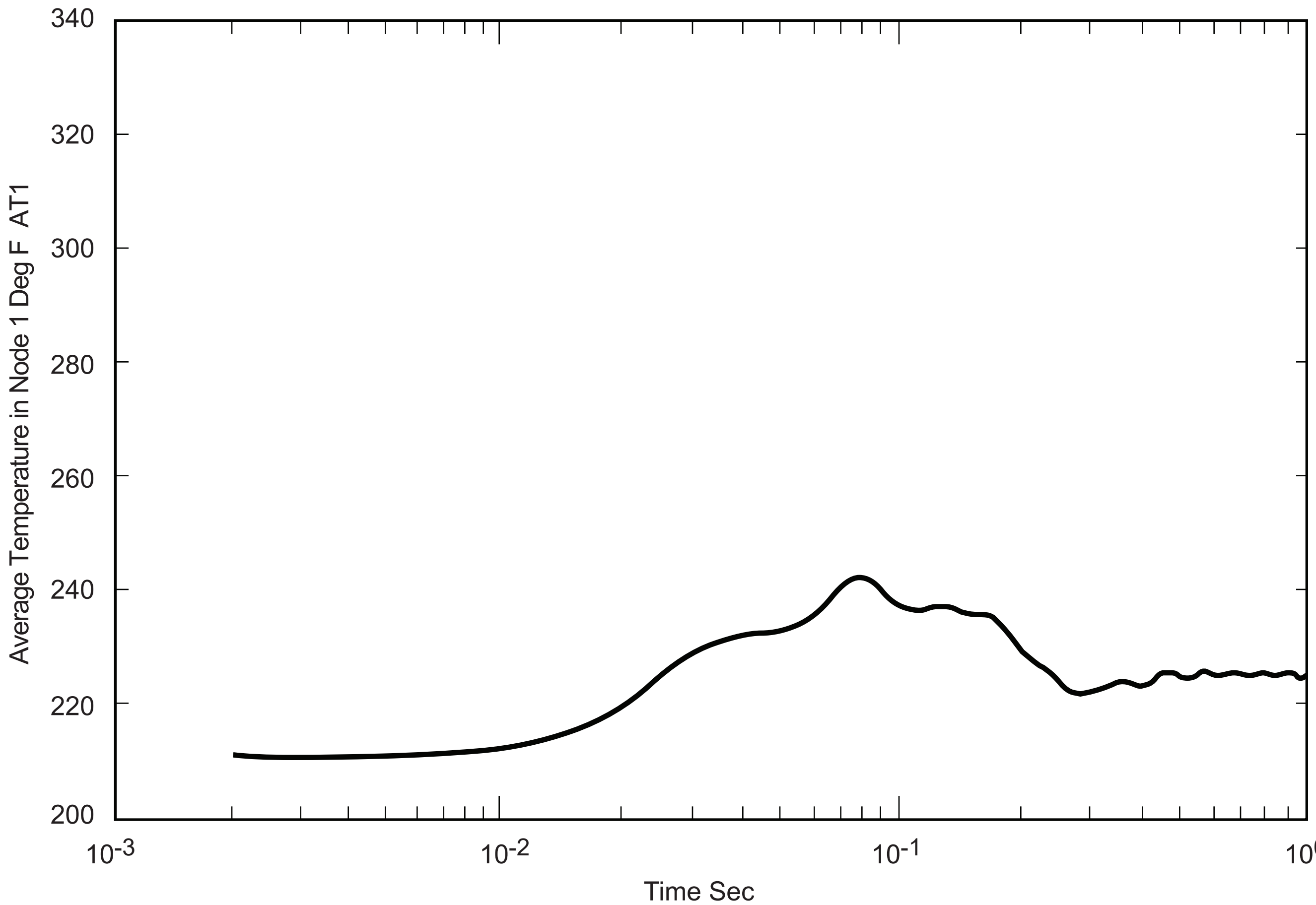


Pressure Transient in Node 1 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel

Draw. No. 970187.91 | Rev. | Figure 3.6-72

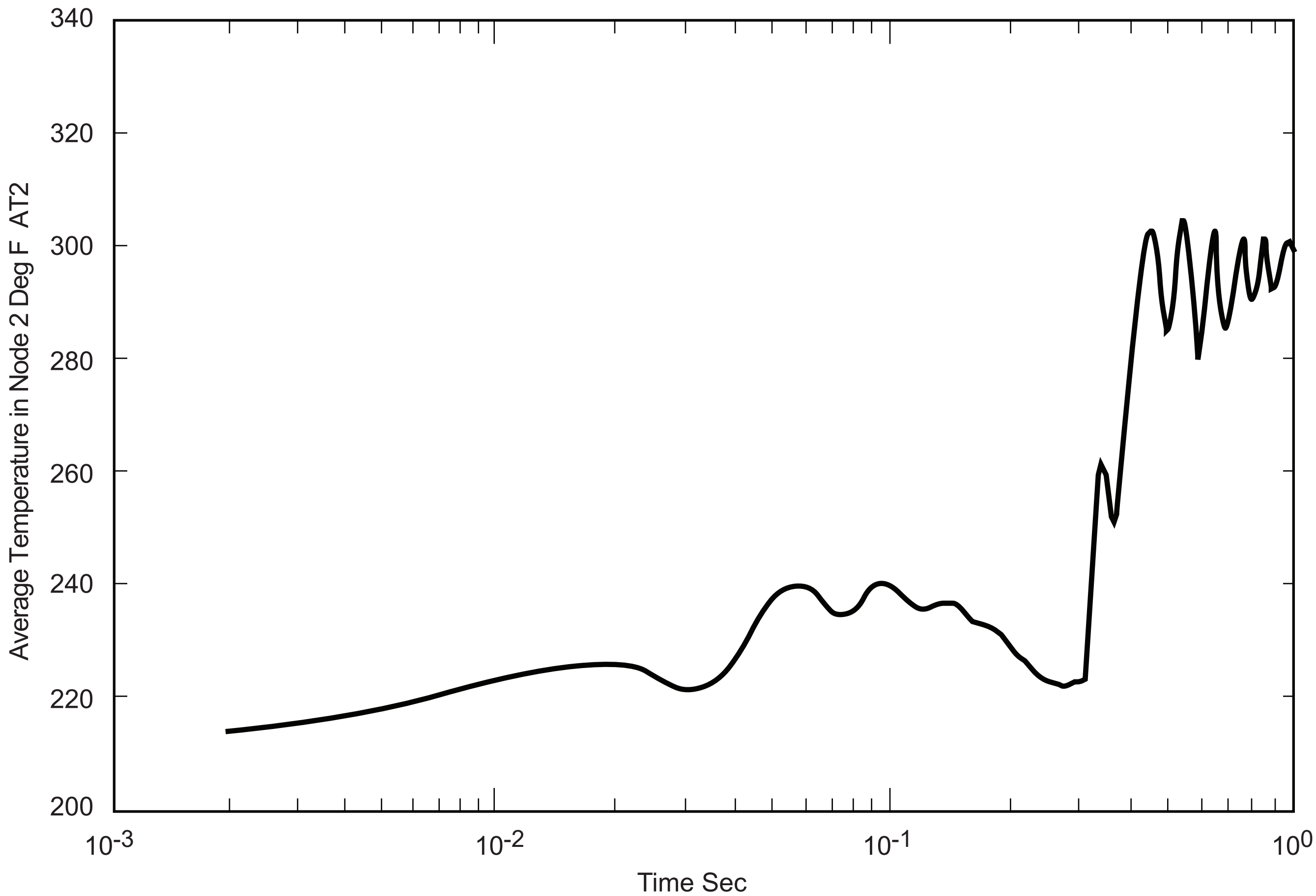


Columbia Generating Station Final Safety Analysis Report	Pressure Transient in Node 2 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel
	Draw. No. 970187.92 | Rev. | Figure 3.6-73



Temperature Transient in Node 1 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel

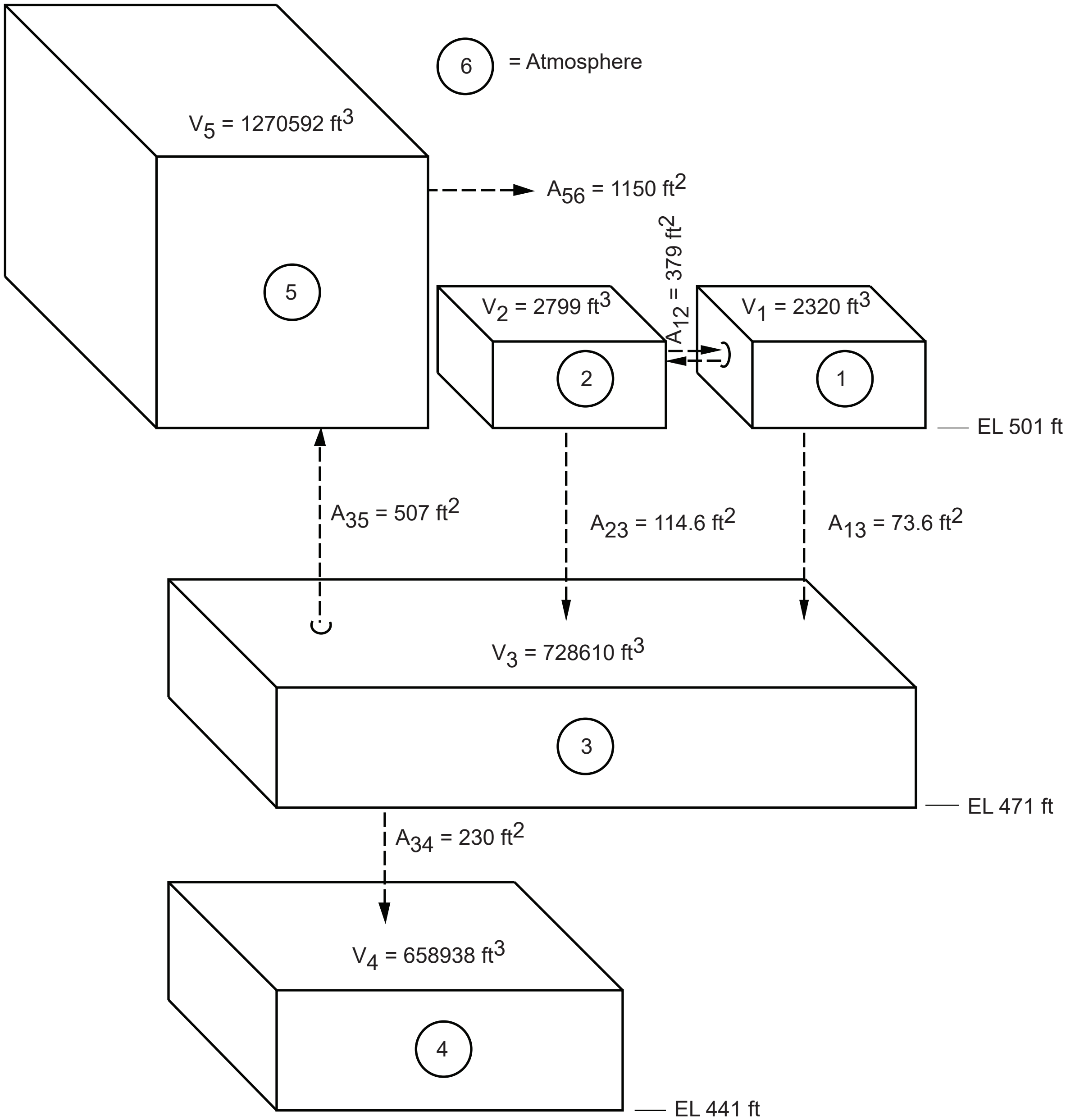
Draw. No. 970187.93 | Rev. | Figure 3.6-74



Columbia Generating Station
Final Safety Analysis Report

Temperature Transient in Node 2 of Main Steam Tunnel after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel

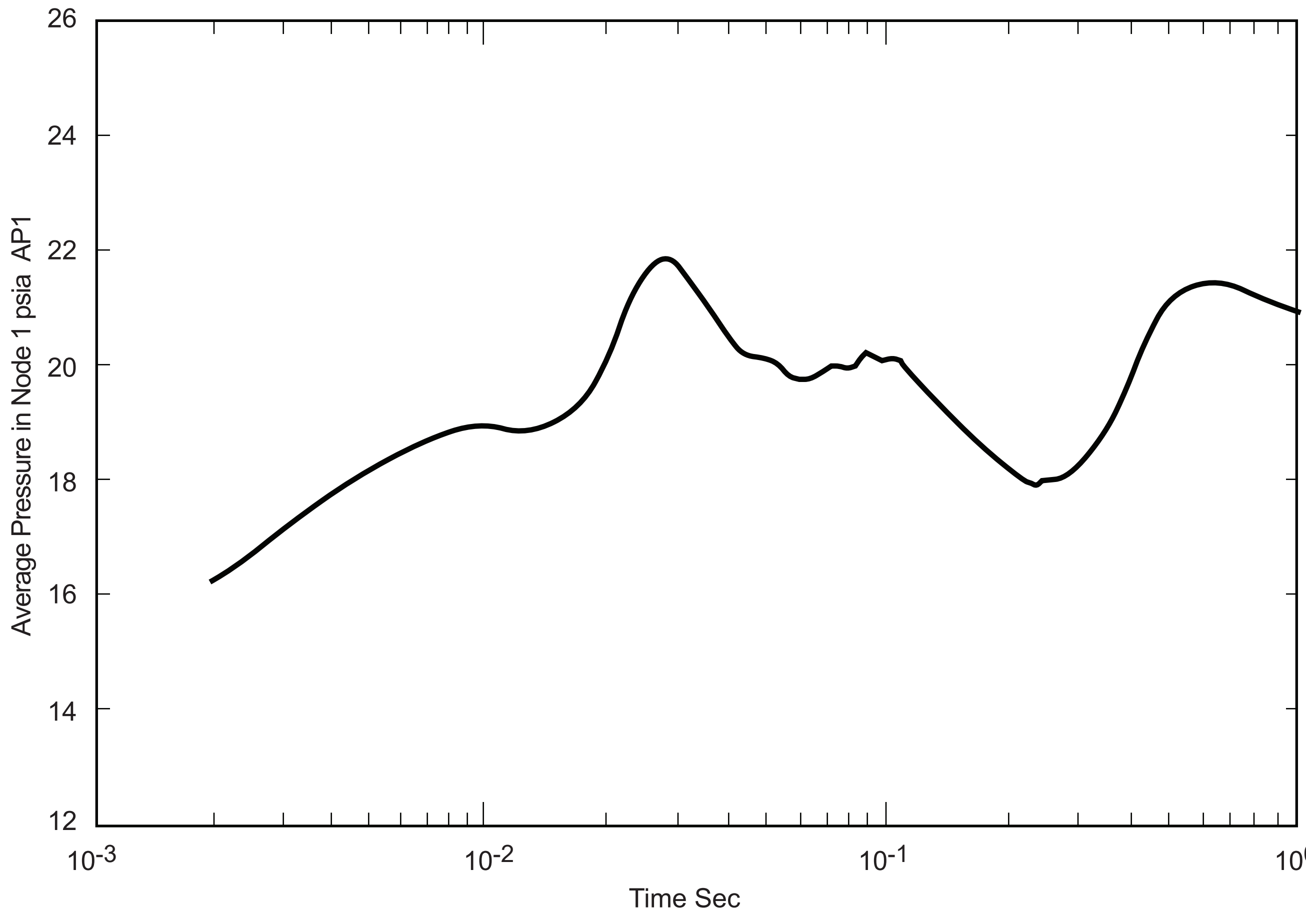
Draw. No. 970187.94	Rev.	Figure 3.6-75



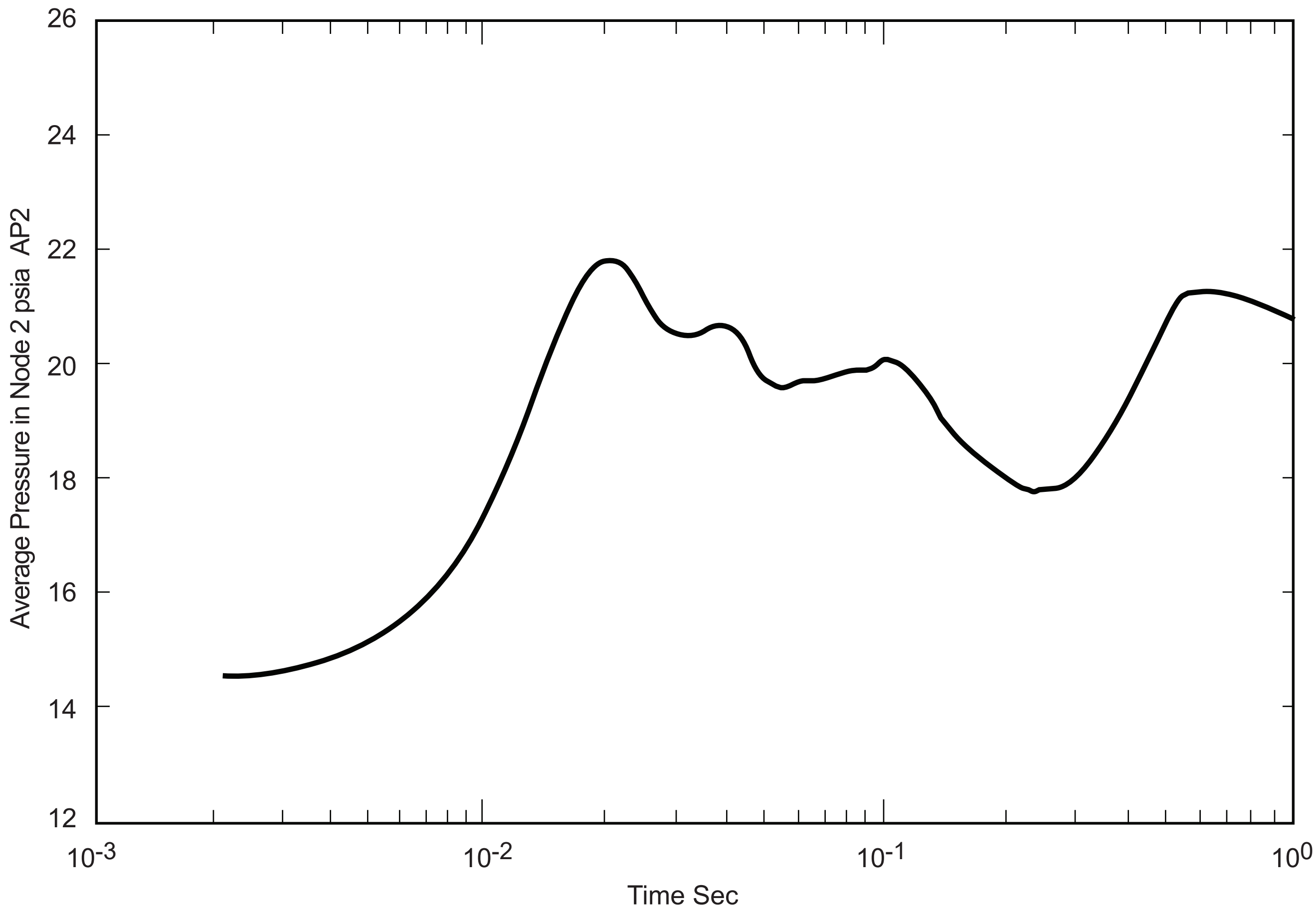
Note: Data Tabulated
in Tables 3.6-13 & -14

Nodalization Scheme for Postulated Pipe Break in Main Steam Tunnel Extension

Draw. No. 990306.57	Rev.	Figure 3.6-76

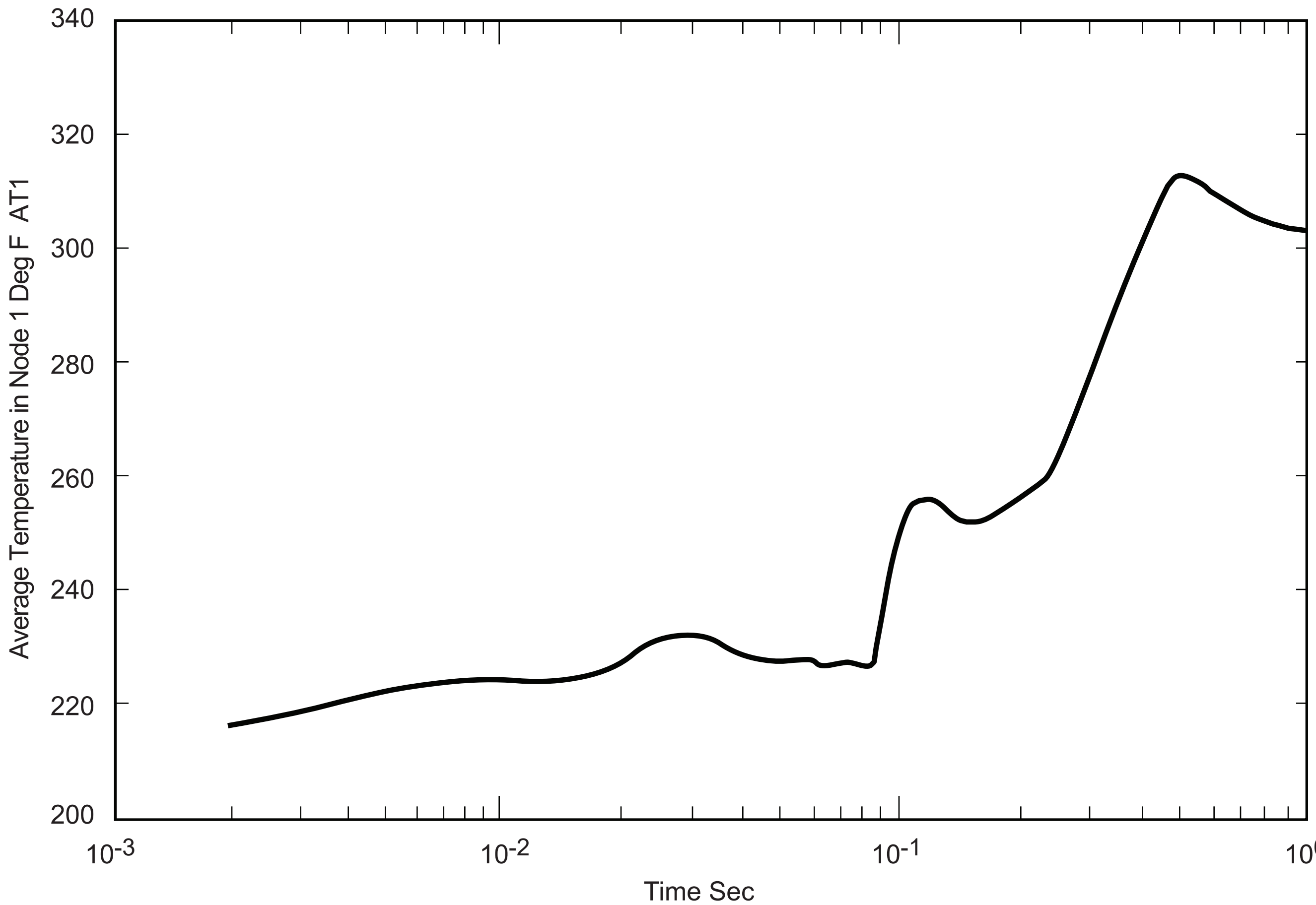


Columbia Generating Station **Final Safety Analysis Report**	**Pressure Transient in Node 1 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel Extension**
	Draw. No. **970187.95** Rev. Figure **3.6-77**



Pressure Transient in Node 2 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel Extension

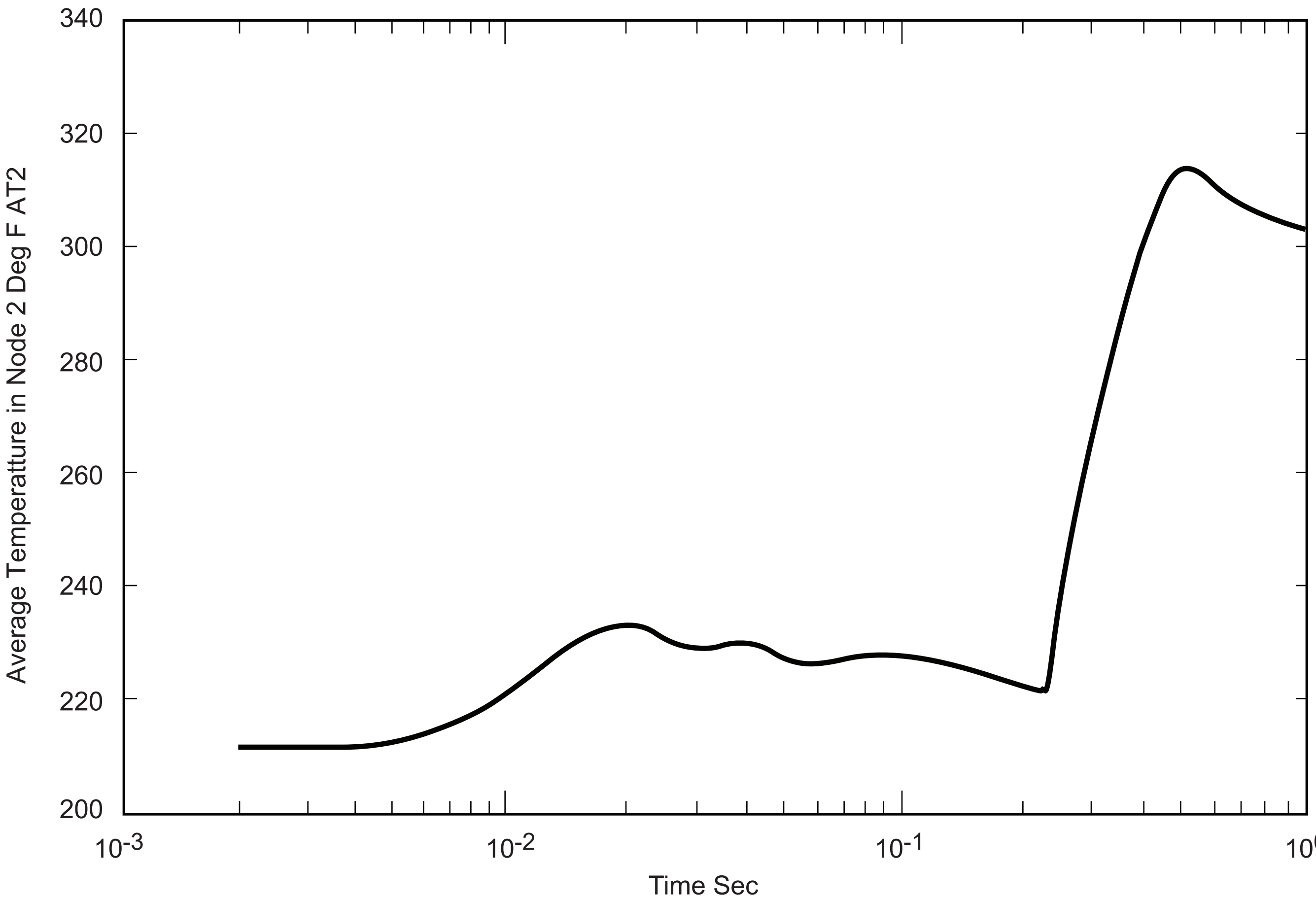
Draw. No. 970187.96	Rev.	Figure 3.6-78



Columbia Generating Station
Final Safety Analysis Report

Temperature Transient in Node 1 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel Exten.

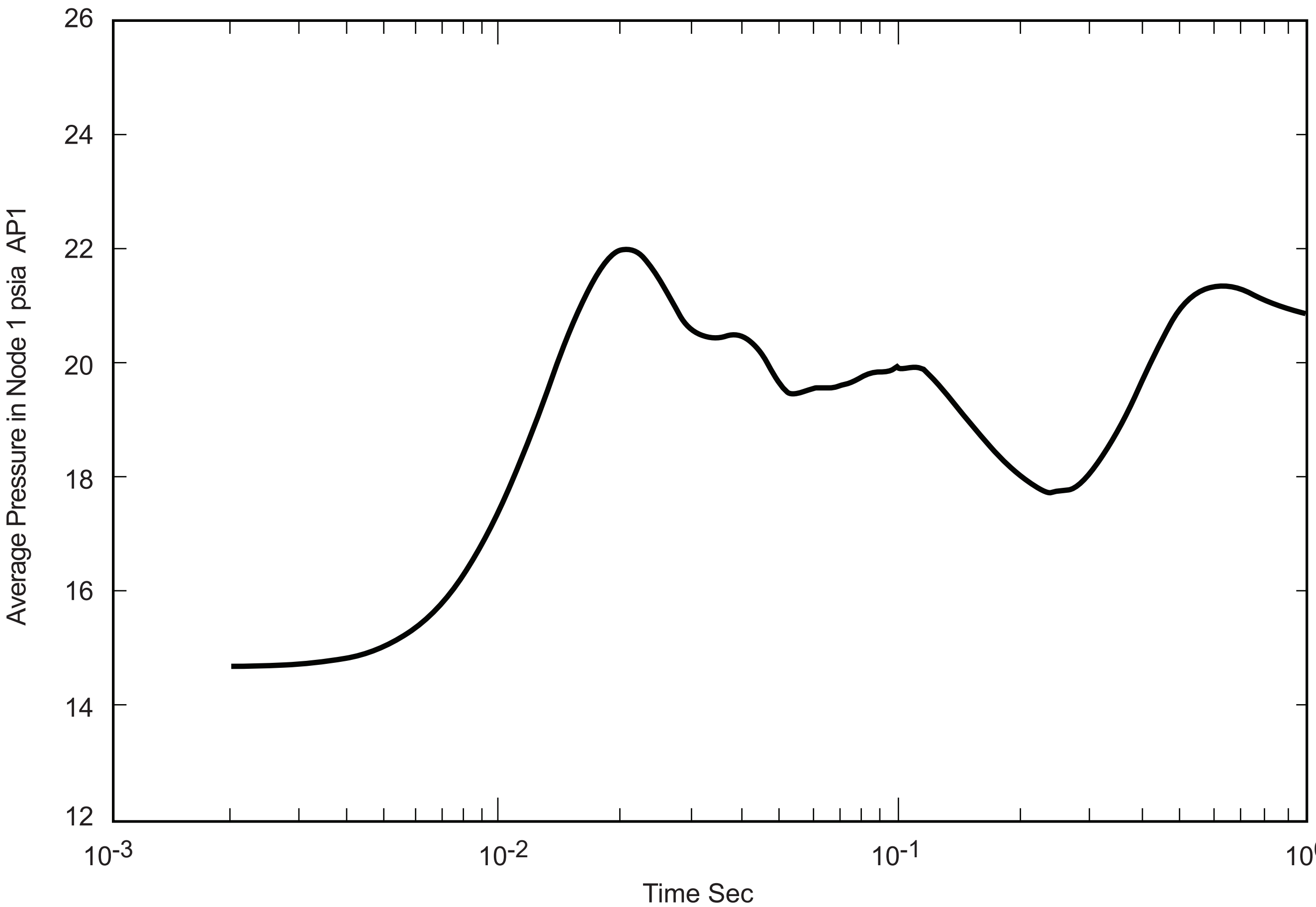
Draw. No. 970187.97	Rev.	Figure 3.6-79



Columbia Generating Station
Final Safety Analysis Report

Temperature Transient in Node 2 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 1 of Main Steam Tunnel Exten.

Draw. No. 970187.98	Rev.	Figure 3.6-80

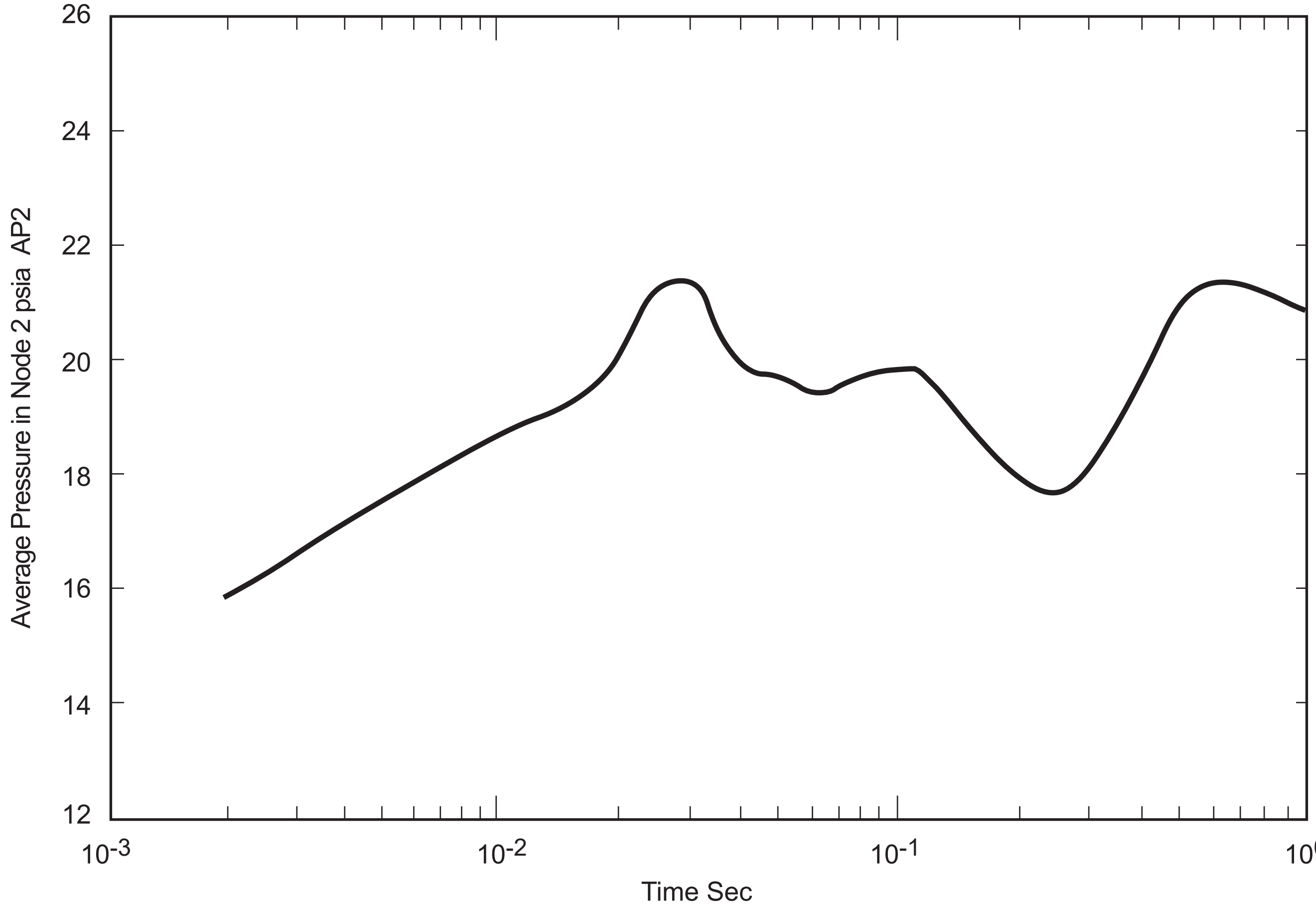


Columbia Generating Station
Final Safety Analysis Report

Pressure Transient in Node 1 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel Exten.

Draw. No. 970187.99	Rev.	Figure 3.6-81

Form No. 960690

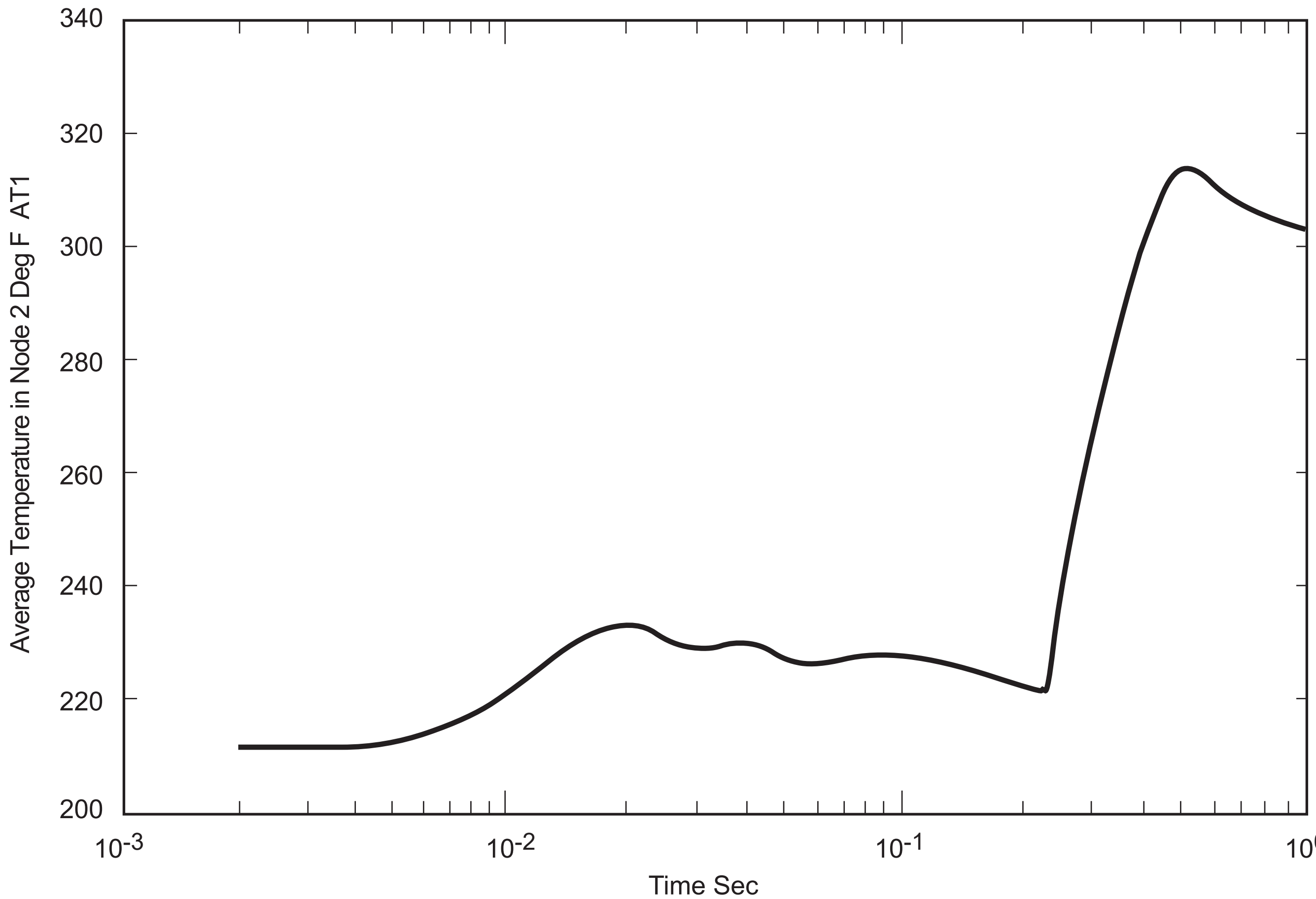


Columbia Generating Station
Final Safety Analysis Report

Pressure Transient in Node 2 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel Extension

Draw. No. 900547.97	Rev.	Figure 3.6-82

Amendment 53
November 1998

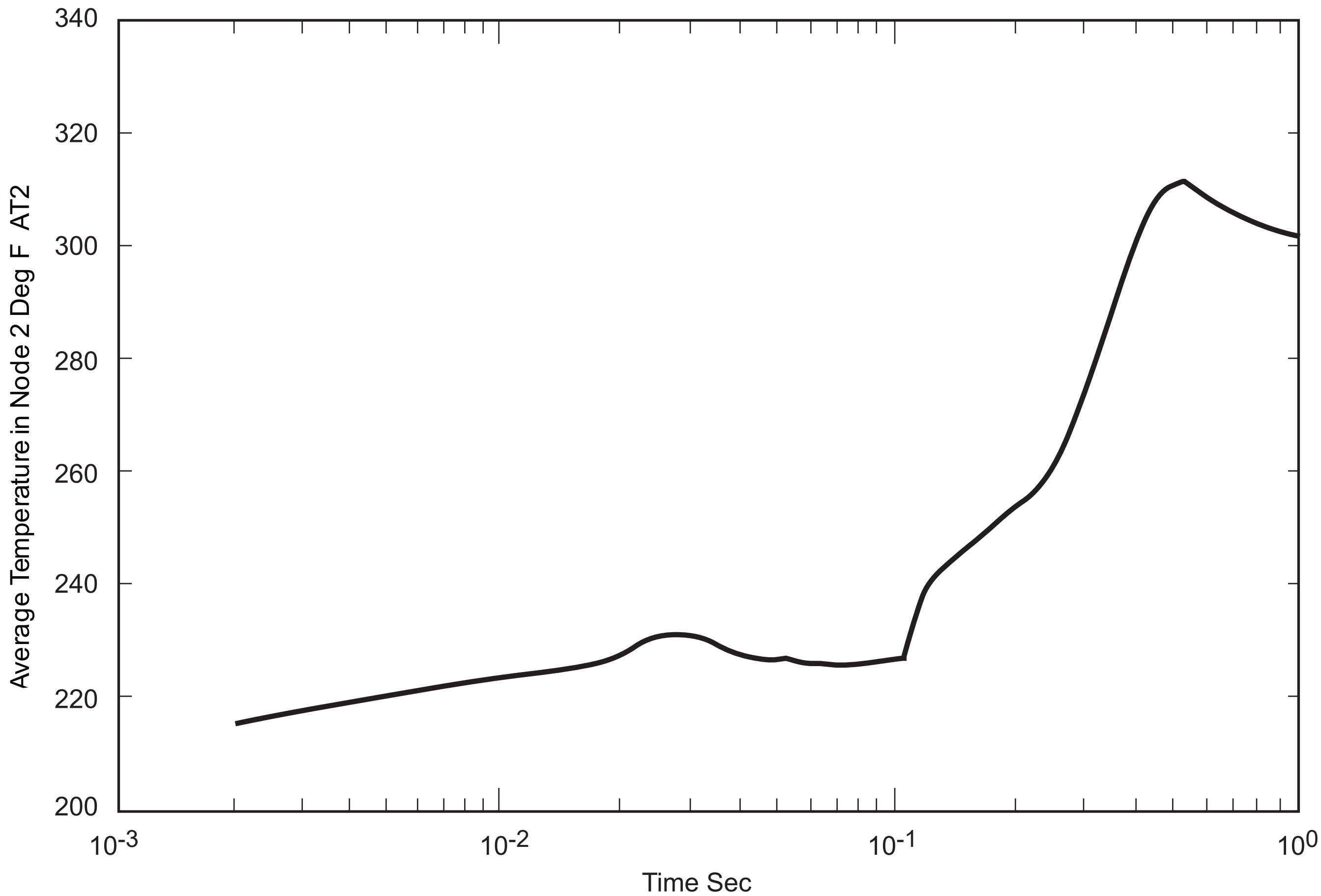


Columbia Generating Station
Final Safety Analysis Report

Temperature Transient in Node 2 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel Exten.

Draw. No. 910402.25 | Rev. | Figure 3.6-83

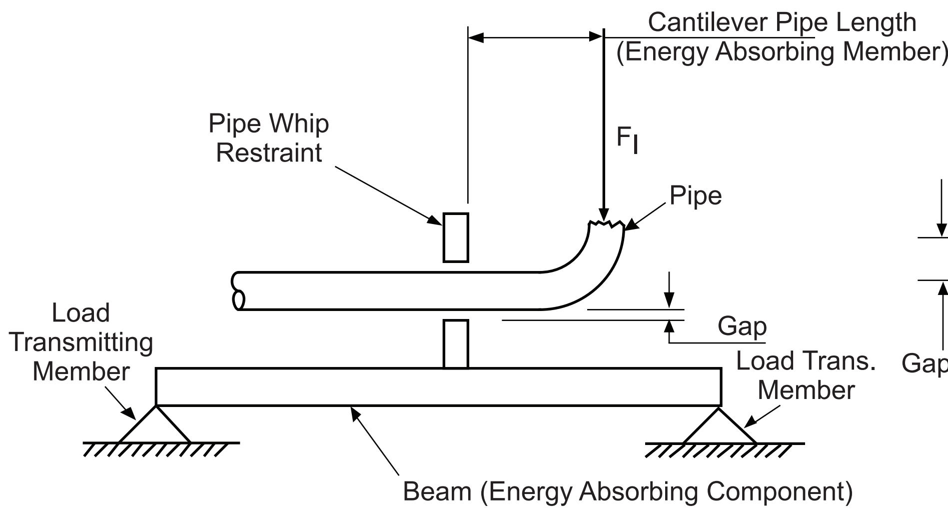
Form No. 960690



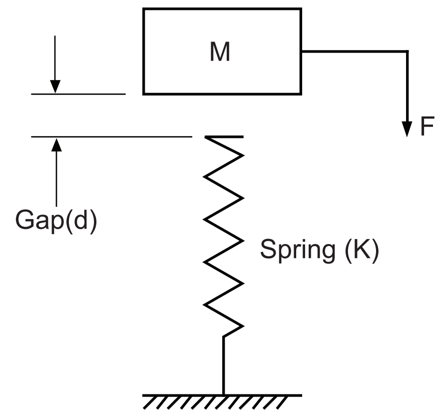
Columbia Generating Station
Final Safety Analysis Report

Temperature Transient in Node 2 of Main Steam Tunnel Extension after a Postulated Main Steam Pipe Break in Node 2 of Main Steam Tunnel Extension

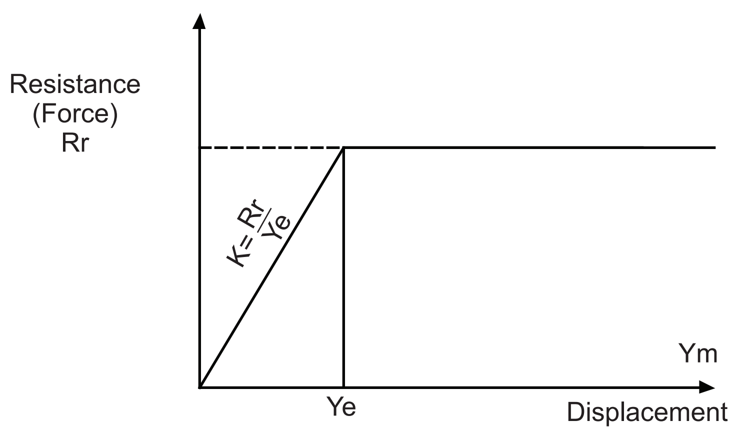
Draw. No. 900547.99	Rev.	Figure 3.6-84



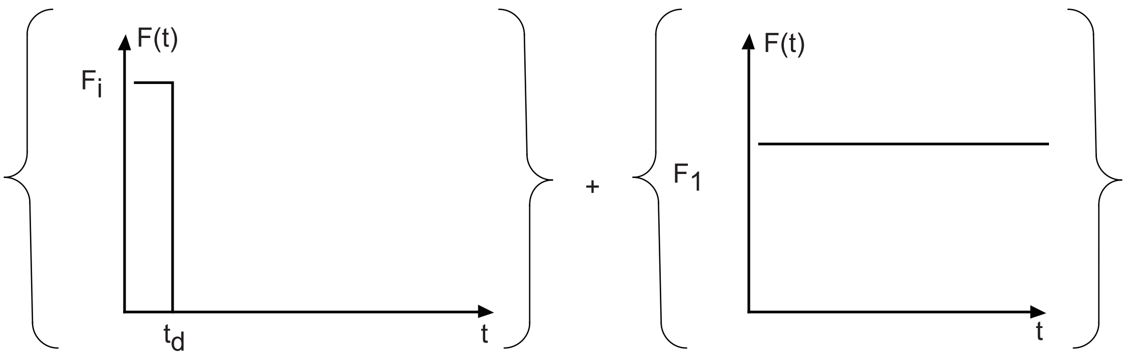
(A) Pipe Whip Support Configuration



(B) Single Degree of Freedom Mathematical Idealization for a Structure



(C) Resistance Function



Impulse (i) = F_i (t_d)

Kinetic Energy (K) = $\frac{i^2}{2M}$ = F_1 x distance (d)

Note: $\mu + \frac{Y_m}{Y_e}$; Elastic Spring Constant k = $\frac{R_r}{Y_e}$

Ref. 3.6-1 Chapter 5 paragraph 5.5b

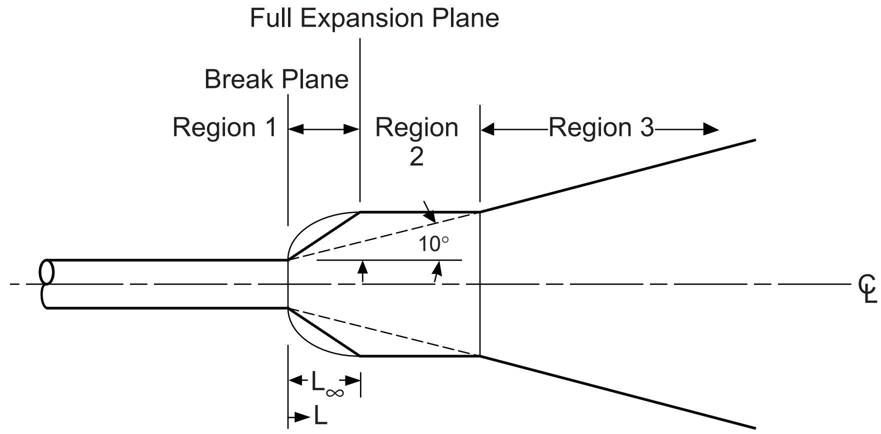
$F_1 Y_m + K = R_r [Y_m - (1/2) Y_e]$

Substituting $\mu = \frac{Y_m}{Y_e}$ & $\frac{R_r}{k}$ + Y_e

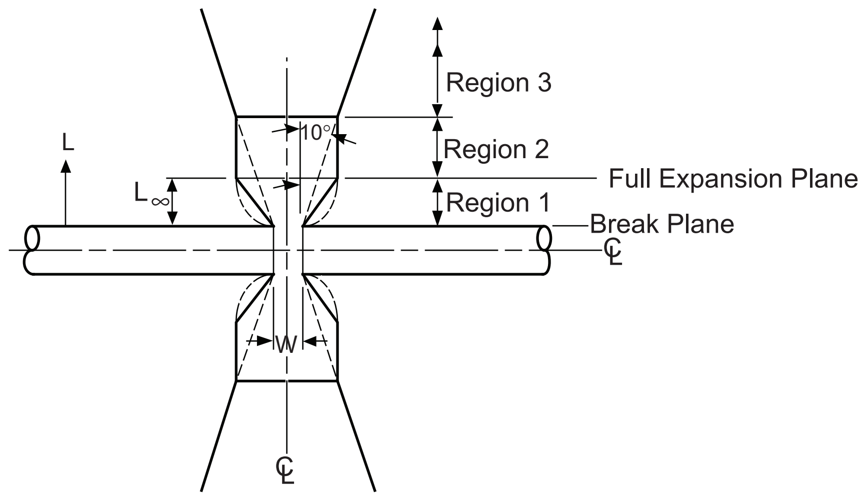
$R_r^2 (\mu - 1/2) - uF_1 R_r - (K)(k) = 0$

Solving Quadratically:

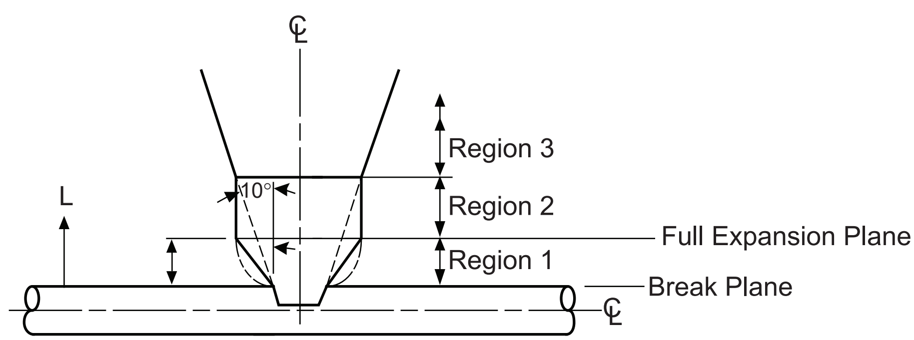
$$\boxed{R_r = \left[\frac{\mu F_1}{(2\mu-1)} \right] + \left[\left\{ \frac{u F_1}{(2\mu-1)} \right\}^2 + \frac{2 (K) (k)}{(2\mu-1)} \right]^{1/2}}$$



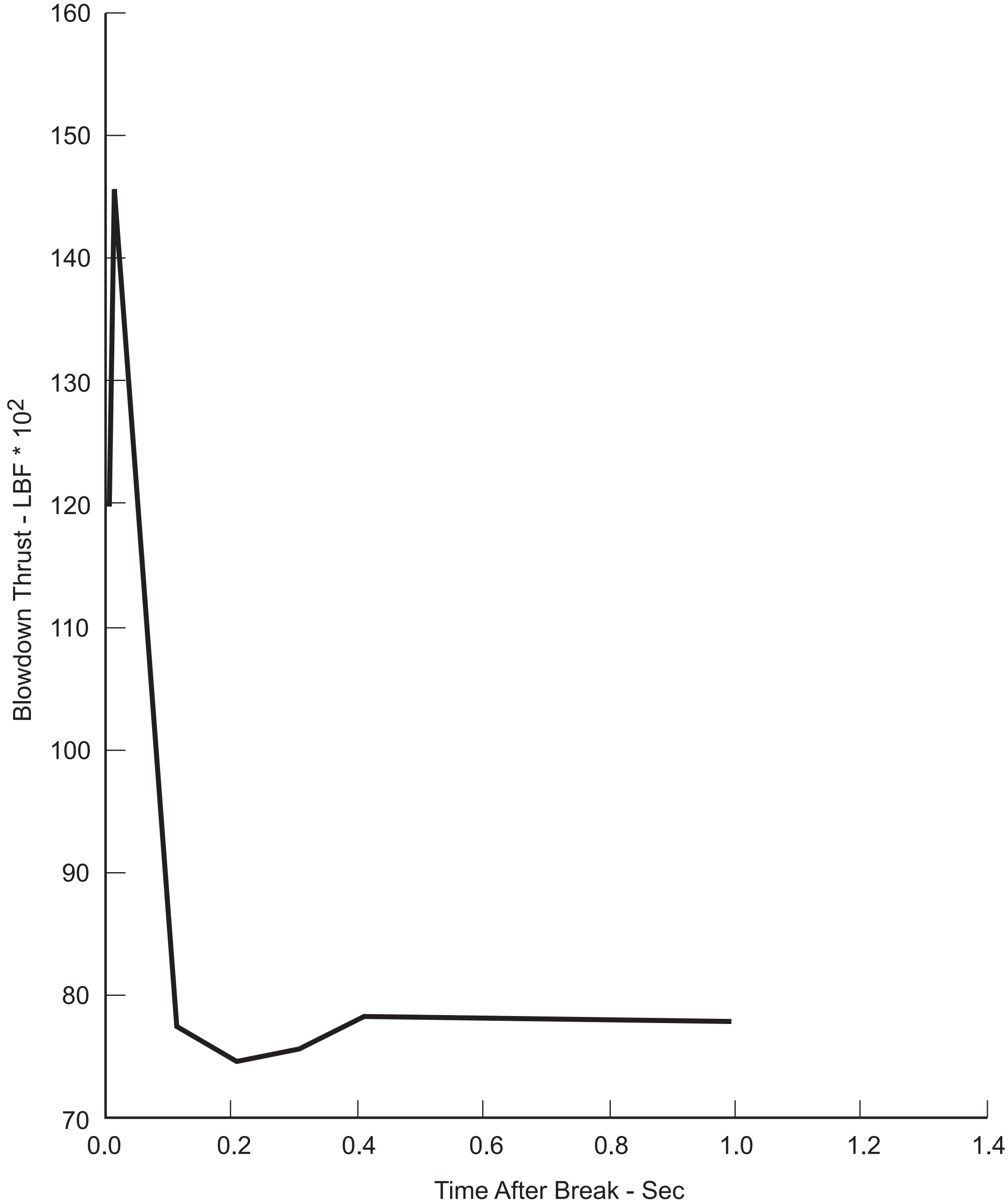
(A) Circumferential Break with Full Separation



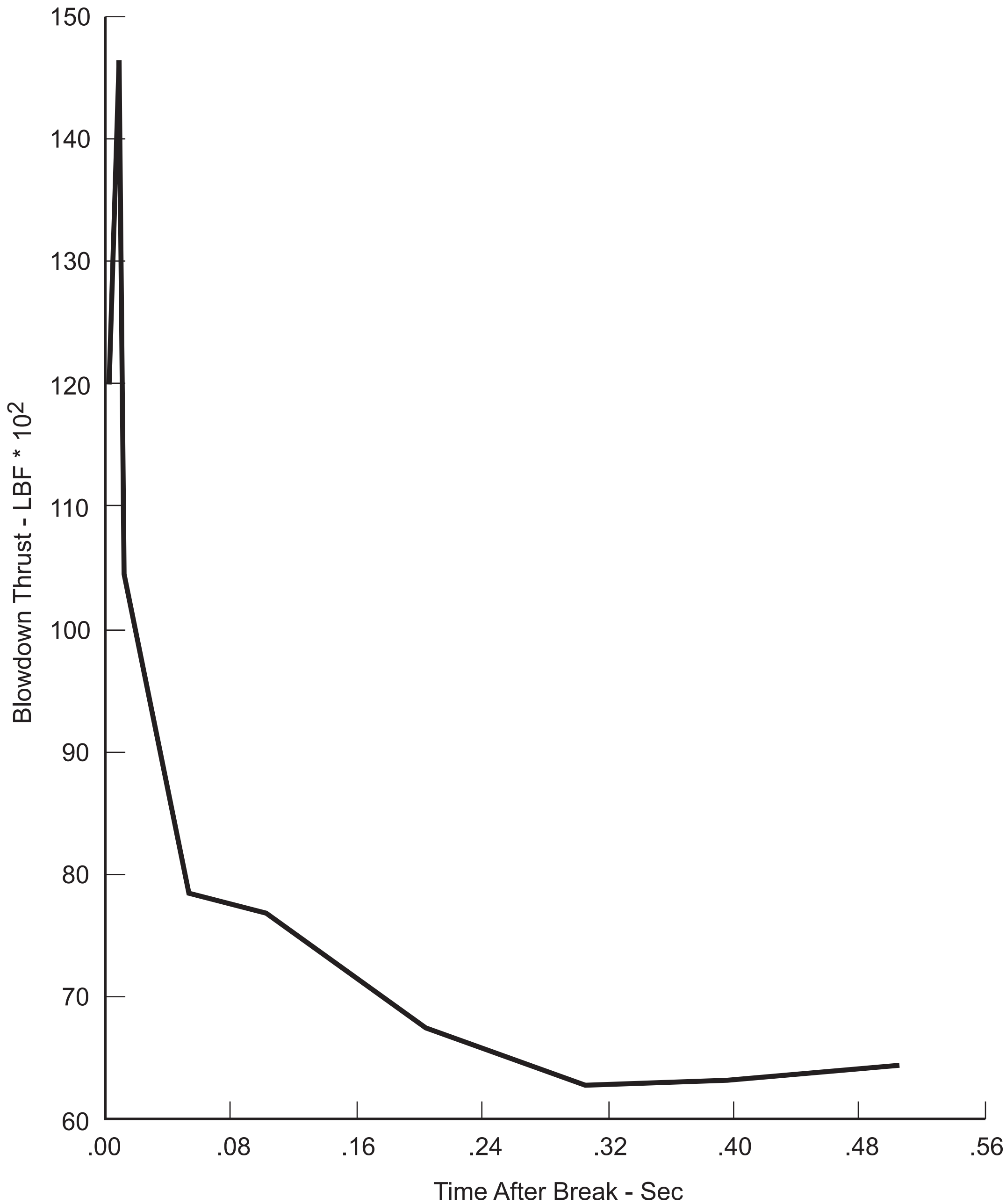
(B) Circumferential Break with Partial Separation



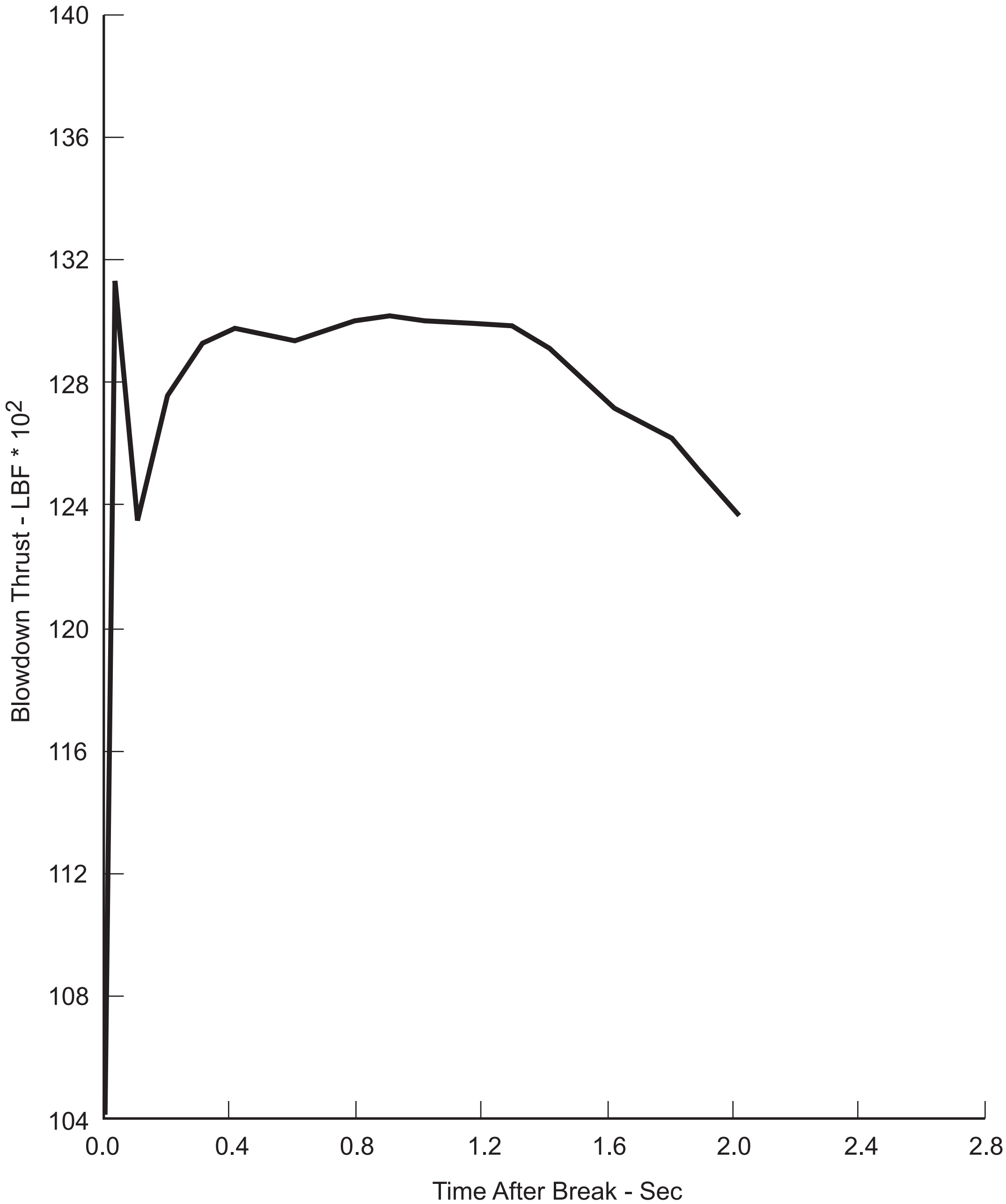
(C) Longitudinal Break



Columbia Generating Station Final Safety Analysis Report	Thrust Versus Time - Reactor Side of Break on 4 in. RCIC (13) - 4 in. Room R206		
	Draw. No. 910402.06	Rev.	Figure 3.6-96



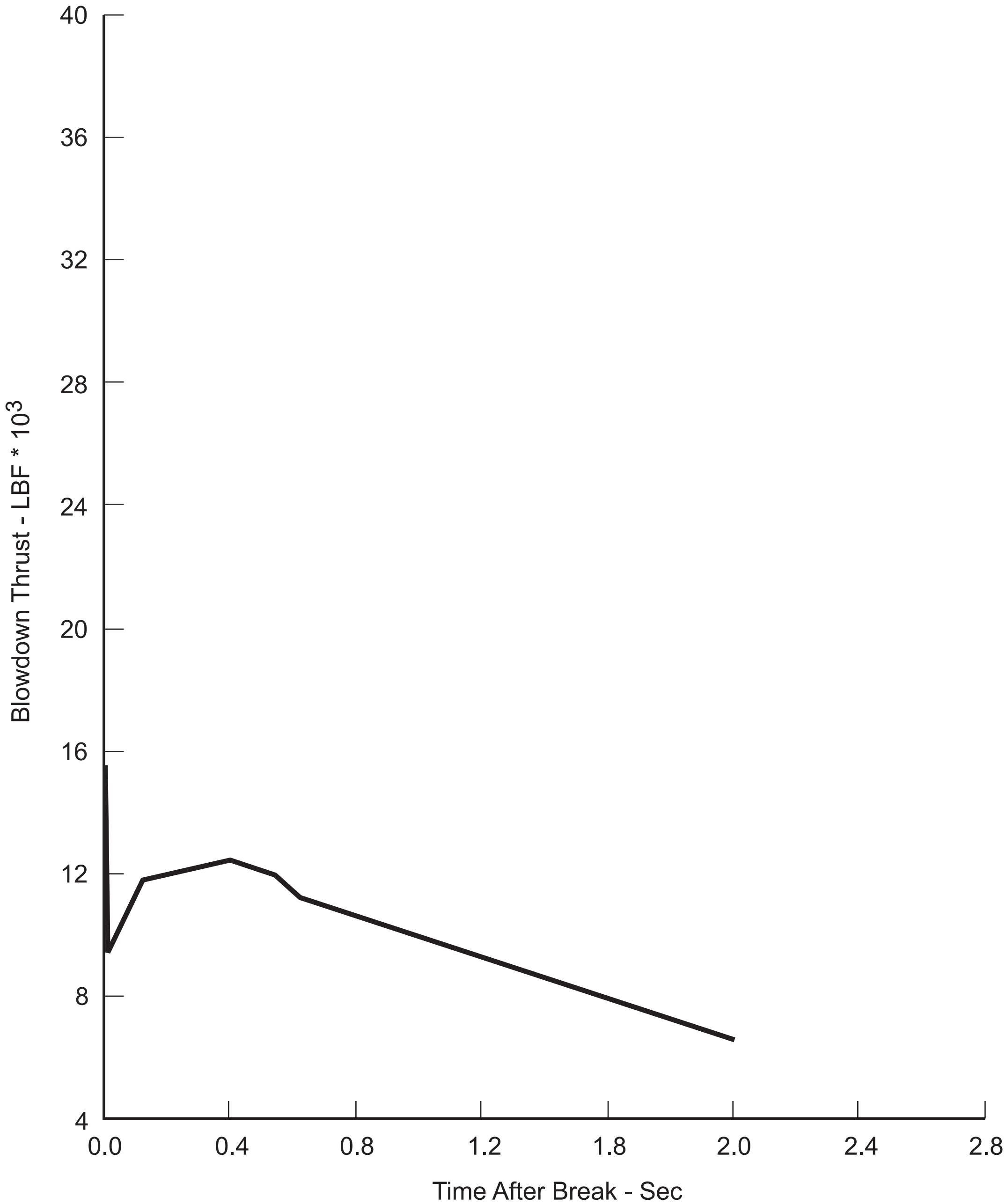
Columbia Generating Station **Final Safety Analysis Report**	**Thrust Versus Time - Reactor Side of Break on 4 in. RCIC (13) - 4 - El. 431.8 ft**
	Draw. No. **910402.07** | Rev. | Figure **3.6-97**



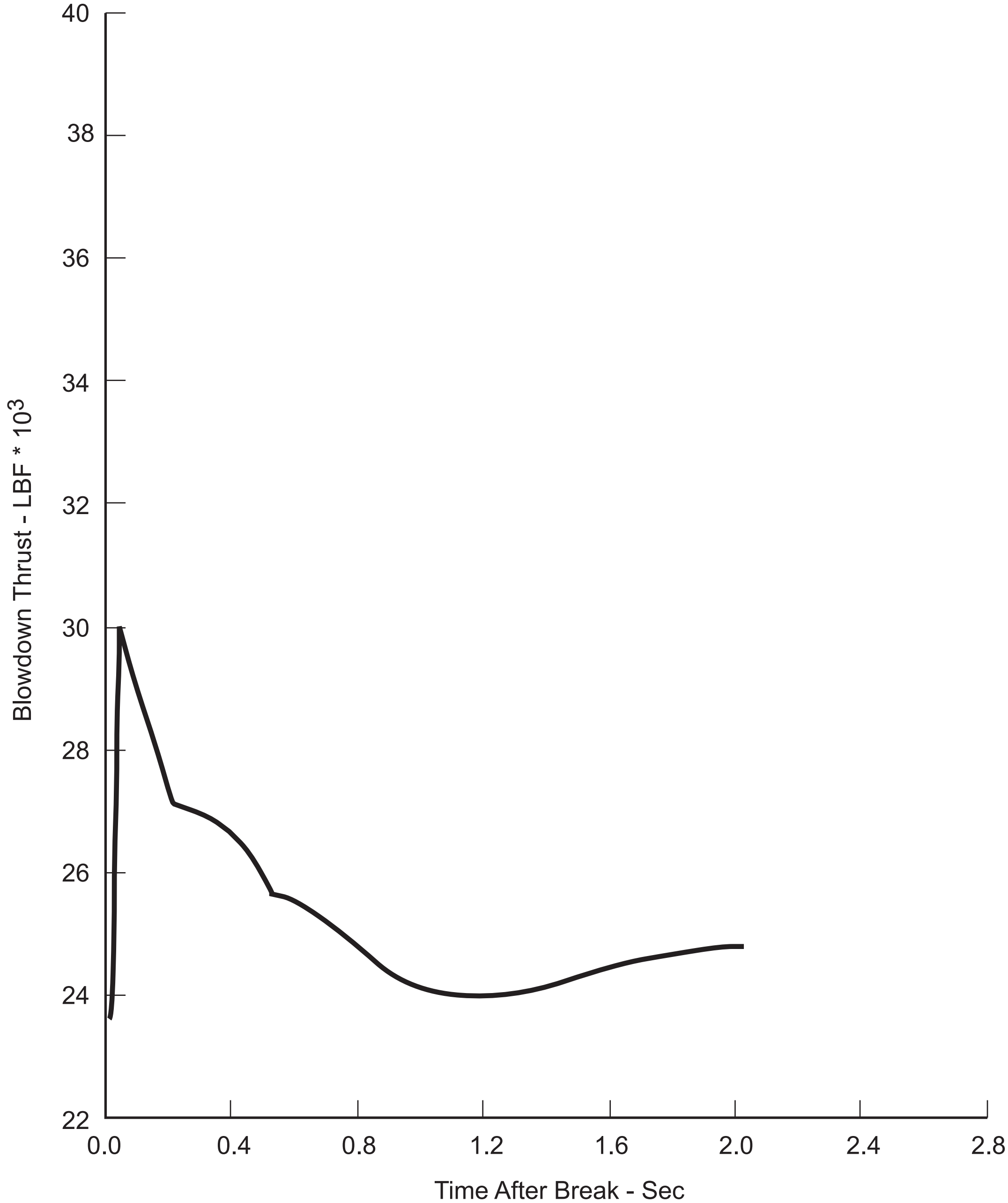
Columbia Generating Station
Final Safety Analysis Report

Thrust Versus Time - Upstream Side of Break on 4 in. RWCU (2) - 4 - El. 536 ft 0 in., Room R409

Draw. No. 910402.08 | Rev. | Figure 3.6-98

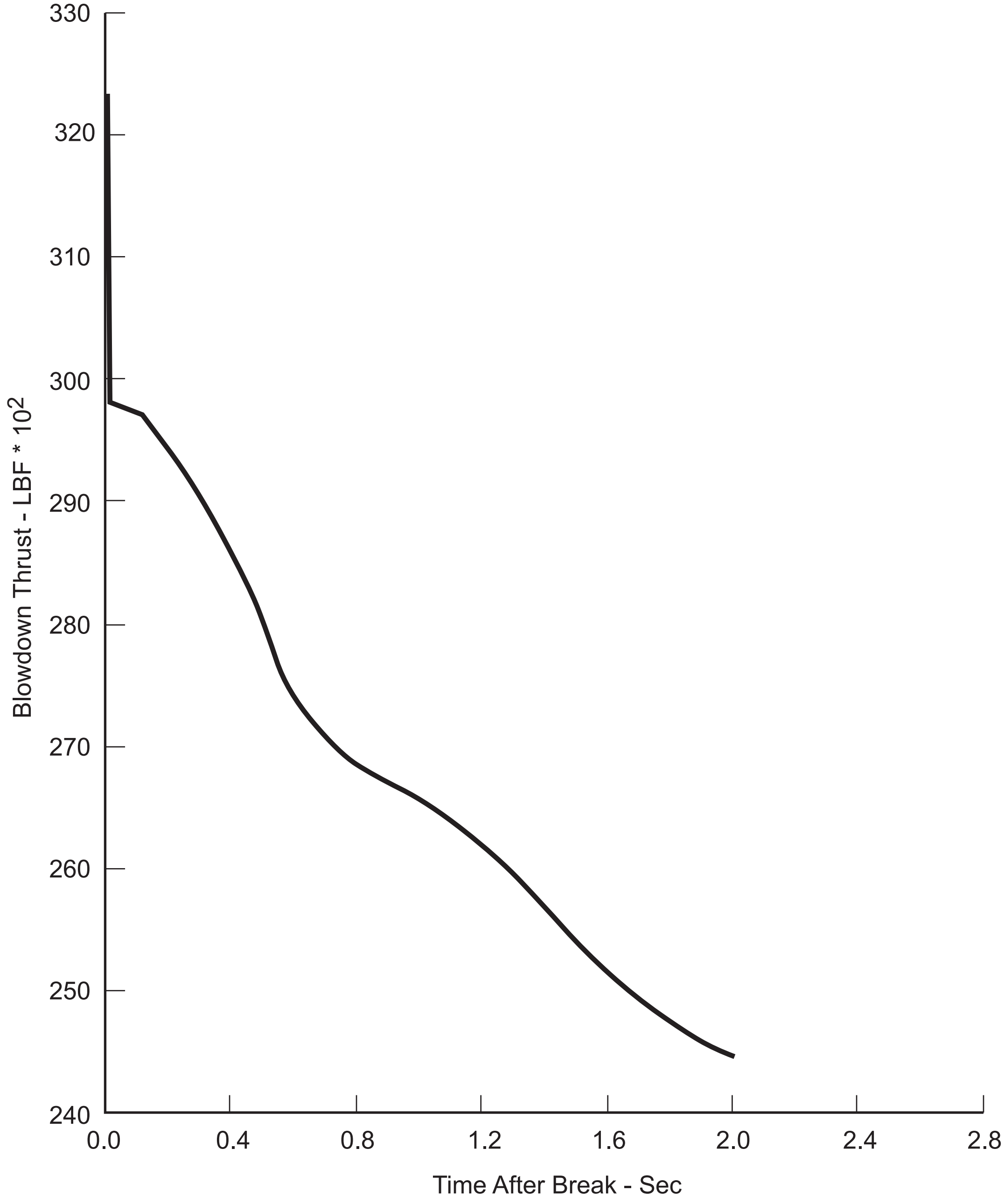


Columbia Generating Station Final Safety Analysis Report	Thrust Versus Time - Heat Exchanger Side of Break on 6 in. RWCU (2) - 4 - El. 514 ft 0 in., Room R308
	Draw. No. 910402.05 | Rev. | Figure 3.6-99



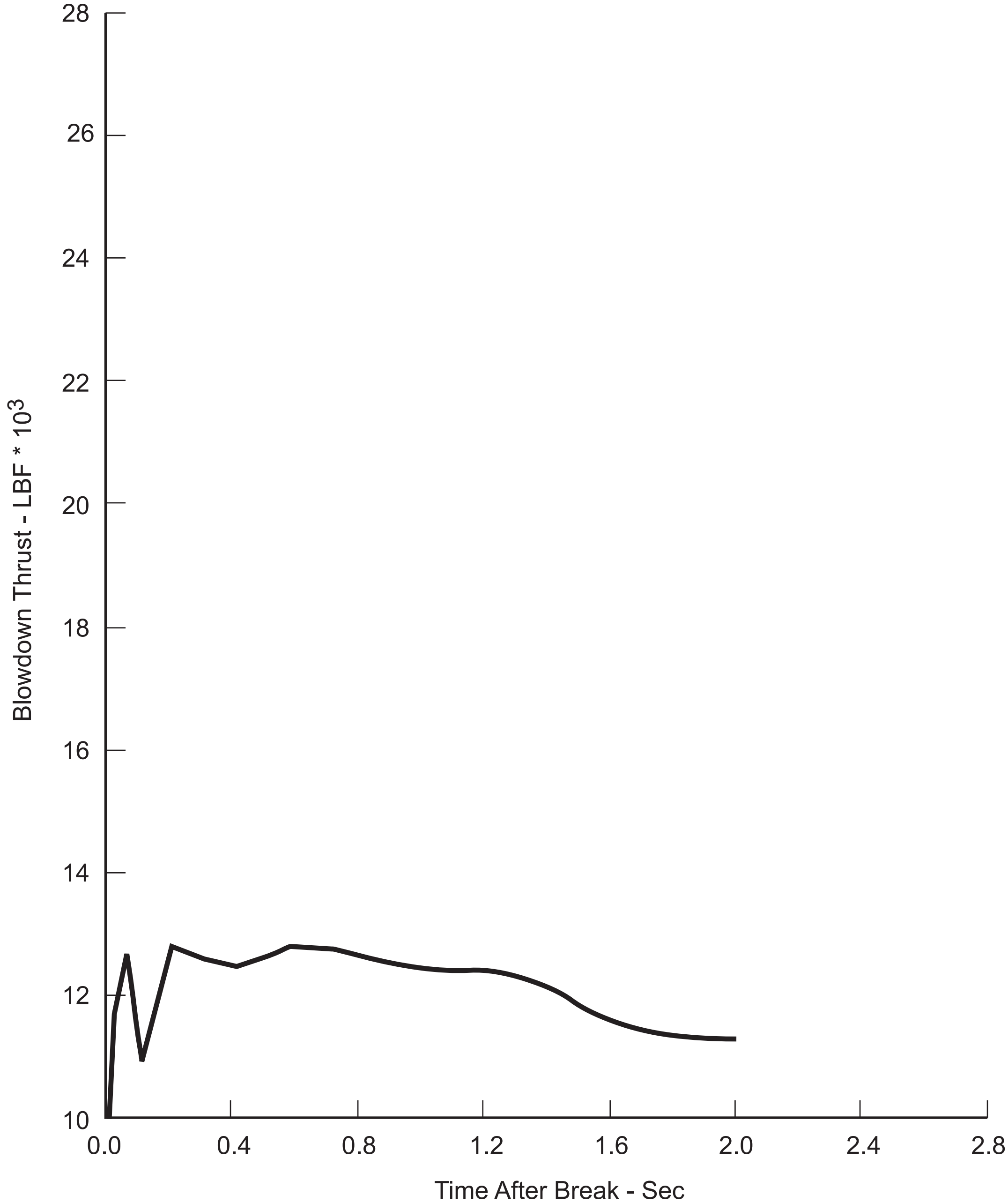
Thrust Versus Time - Pump Side of Break on 6 in. RWCU (1) - 4 - El. 548 ft 0 in., Room R409

Draw. No. 910402.09	Rev.	Figure 3.6-100



Thrust Versus Time - Pump Side of Break on 6 in. RWCU(1)-5SX/SXL - El. 557 ft 6 in., Room R510

Draw. No. 910402.11 | Rev. | Figure 3.6-101



Columbia Generating Station
Final Safety Analysis Report

Thrust Versus Time - Downstream Side of Break on 6 in. RWCU(2)-5SX/SXL - El.557 ft 6 in., Room R510

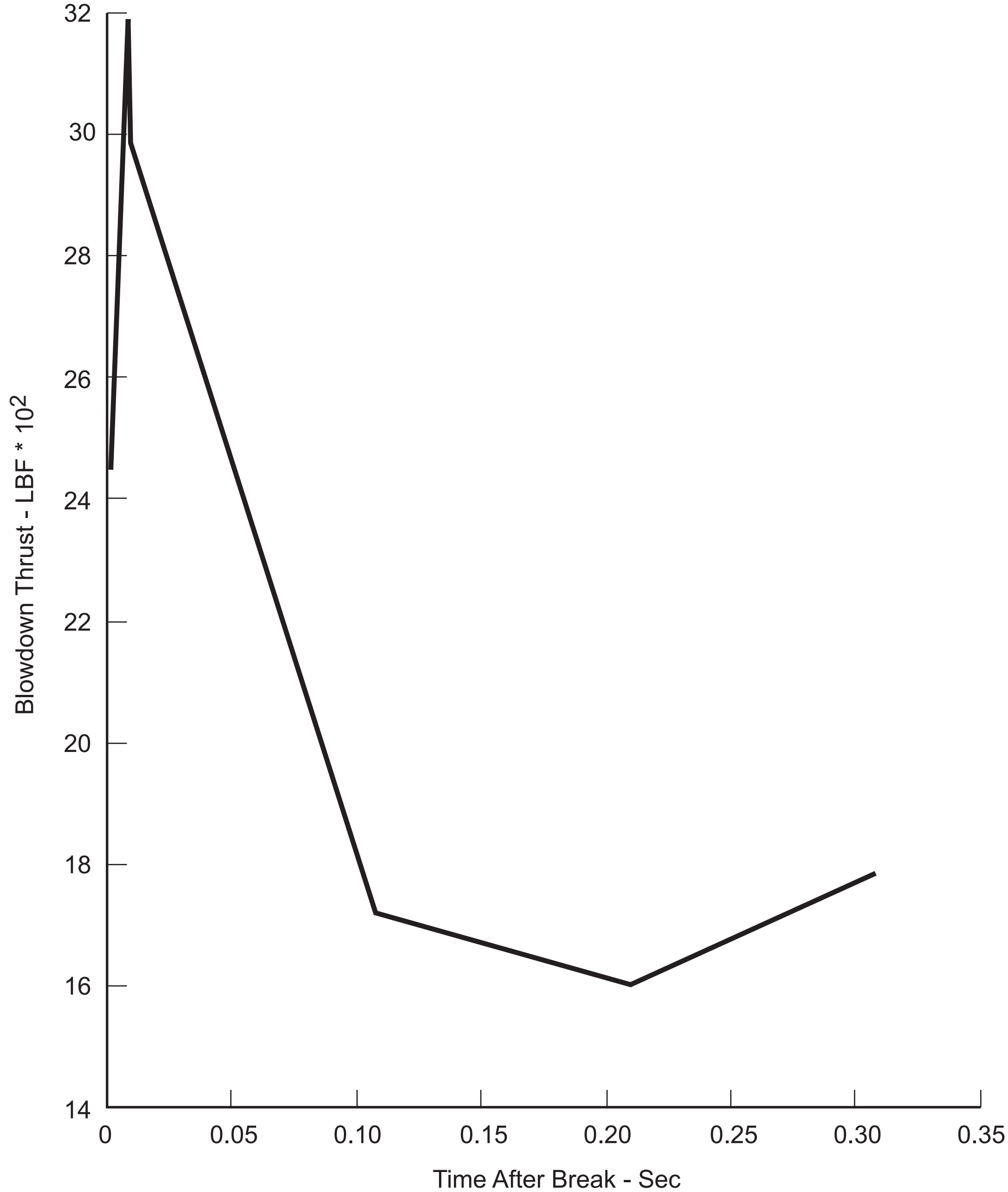
Draw. No. 910402.12	Rev.	Figure 3.6-102

DELETED

Columbia Generating Station
Final Safety Analysis Report

Thrust Versus Time - Both Sides of Break on 4 in. RWCU (1) - 4 El. 556.5 ft 0 in., Room R510

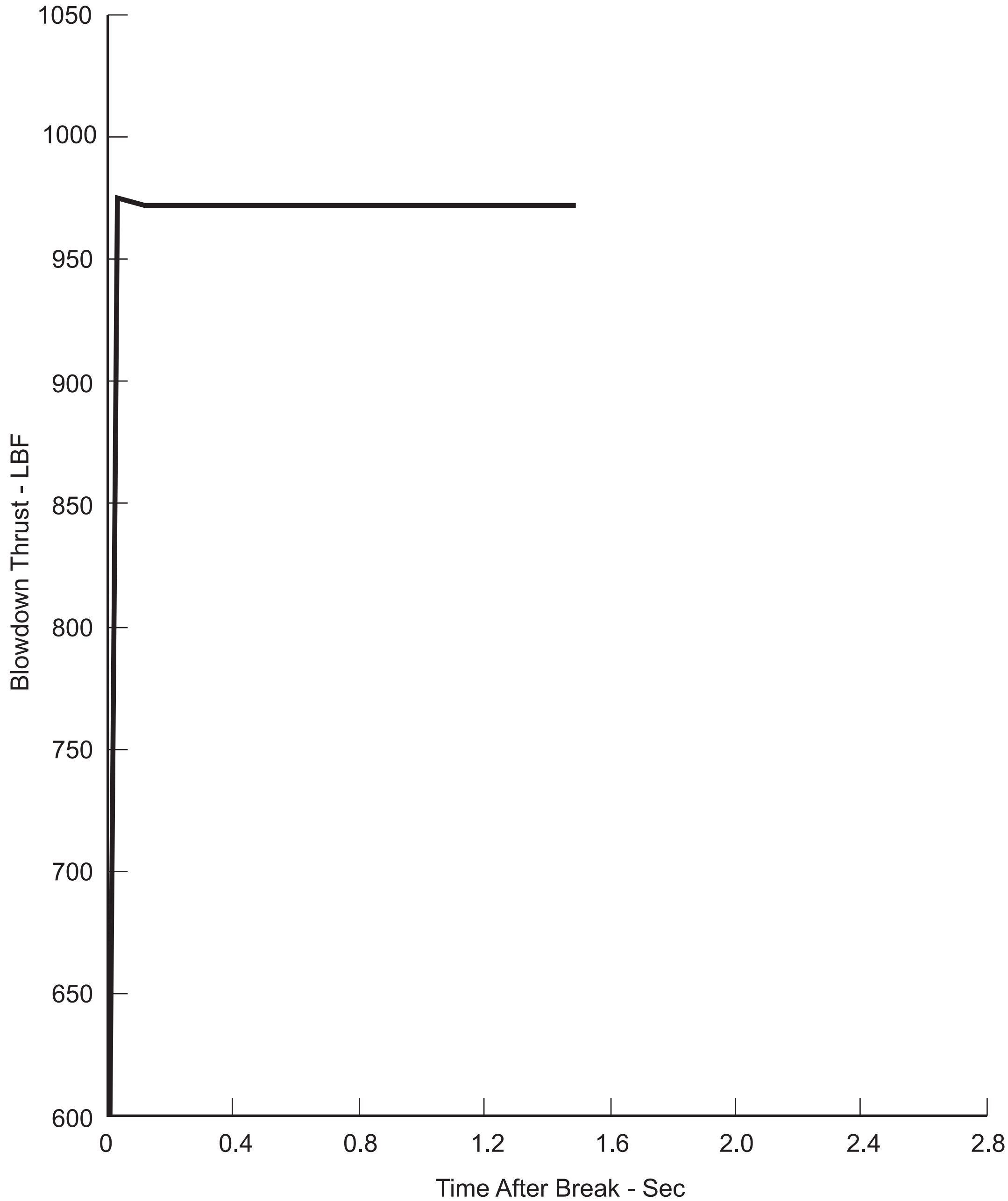
Draw. No. 910402.13 | Rev. | Figure 3.6-103



Columbia Generating Station
Final Safety Analysis Report

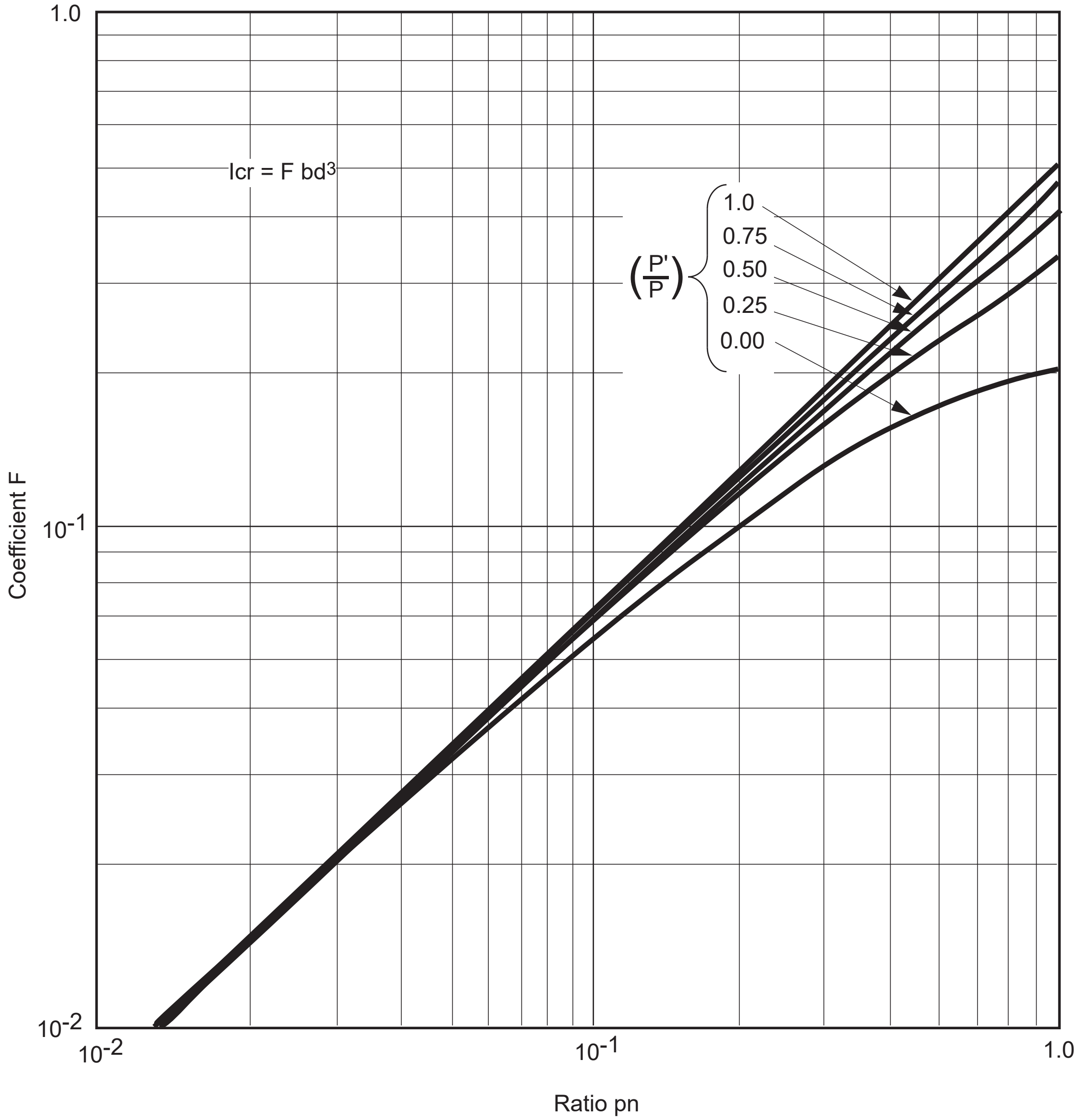
Thrust Versus Time - Upstream Side of Break on 4 in. AS (11) - 2 - El. 472 ft 0 in., Room R206

Draw. No. 910402.14 | Rev. | Figure 3.6-104



Thrust Versus Time - Upstream Side of Break on 4 in. HS (1) - 2 - El. 574.5 ft 0 in., Room R604

Draw. No. 910402.15 | Rev. | Figure 3.6-105



$p = \frac{As}{bd}$, $p' = \frac{A's}{bd}$, $n = \frac{Es}{Ec}$

$F = \frac{K^3}{bd}$, $+ pn\ (l\text{-}k)^2 + \left(\frac{2n\text{-}1}{n}\right)(pn)\frac{P'}{P}\left(K - \frac{d'}{d}\right)^2$

$\frac{2n\text{-}1}{n} \cong 1.9,\quad \frac{d'}{d} \cong 0.10,\quad k = -m + (m^2 + 2q)^{1/2}$

$m = pn\ (1 + 1.9\ \frac{p'}{p}),\quad q = pn\ (1 + 0.19\ \frac{p'}{p})$

Columbia Generating Station Final Safety Analysis Report	**Coefficients for Moment of Inertia of Cracked Sections**		
	Draw. No. 990306.64	Rev.	Figure 3.6-106

3.7 SEISMIC DESIGN

All structures, systems, and components (SSCs) of the facility are defined as either Seismic Category I or non-Category I. The non-Category I seismic features are also referred to in other sections of this report as Seismic Category II. The requirements for Seismic Category I qualification are given in Section 3.2 along with a list of SSCs that are so categorized.

All SSCs related to plant safety are designed to withstand the effects of the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE).

The SSE is that earthquake which is based on an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain SSCs are designed to remain functional. These SSCs are those necessary to ensure

a. The integrity of the reactor coolant pressure boundary,

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

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the exposure limits of 10 CFR Part 50.67.

The OBE is that earthquake which, considering the regional and local geology and 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. The OBE amplitude equals 50% the SSE amplitude.

Geological and seismic criteria related to the site are given in Section 2.5. Based on these criteria the characteristics and intensity of the postulated SSE are established.

3.7.1 SEISMIC INPUT

3.7.1.1 Design Response Spectra

The vibratory ground motion produced by the SSE is defined by design response spectra corresponding to the maximum vibratory accelerations at the elevations of the foundations of the nuclear power plant structures. The design response spectra are idealized, smooth curves relating the response of the foundations of the nuclear power plant structures to the vibratory

ground motion, considering such foundations to be single-degree-of-freedom damped oscillators and neglecting soil-structure interaction effects. The vibratory ground motion produced by the OBE is also defined by design response spectra.

Figures 3.7-1 and 3.7-2 show the design response spectra for the horizontal and vertical components, respectively, of ground motion associated with the SSE, for damping coefficients of 0.5, 2.0, 5.0, 7.0, and 10.0% of critical damping. The maximum horizontal ground acceleration for the SSE was selected to equal 0.25g, as described in Section 2.5, where g is acceleration of gravity. The peak ground acceleration in the vertical direction is taken as two-thirds of the horizontal value. The amplification values (and associated frequency ranges) in the design response spectra correspond to those of Newmark and Hall (Reference 3.7-1) with the exception that for 0.5, 2.0, and 5.0% of critical damping the amplification values were set at 4.8, 3.6, and 2.4, respectively. These response spectra correspond to design response spectra considered acceptable for soil sites (References 3.7-1 and 3.7-2).

These design response spectra are not identical to the design response spectra as defined in Regulatory Guide 1.60, Revision 1, scaled to 0.25g maximum horizontal ground acceleration. However, the latter are used with higher damping values as defined in Regulatory Guide 1.61, Revision 0. A response spectrum dynamic modal analysis was performed on the reactor building structure for an SSE input earthquake using (a) the design response spectra defined in Figure 3.7-1 and the damping values of Table 3.7-1, and (b) the design response spectra, scaled to 0.25g maximum horizontal ground acceleration, and damping values defined in Regulatory Guides 1.60, Revision 1, and 1.61, Revision 0, respectively. The structural responses of each of these modal analyses were within 10% of each other at almost all locations.

Figures 3.7-3 and 3.7-4 show the design response spectra for the horizontal and vertical components, respectively, of ground motion associated with the OBE, for damping coefficients of 0.5, 2.0, 5.0, 7.0, and 10.0% of critical damping. The ordinates of these spectra represent one-half of the ordinates of the design response spectra associated with the SSE.

Both earthquakes are of 15 sec duration. This is justified because (a) most records show a short period (10 to 20 sec) of high intensity acceleration, (b) the structural response analysis indicates that the low intensity build-up and phase-out periods preceding and following the high intensity acceleration period have no significant effect on structural response, and (c) the 15 sec duration is long enough to incorporate at least 7.5 cycles of motion at frequencies above 0.5 cps which is considered a representative limit for flexible structures.

3.7.1.2 Design Time History

A synthetic record of strong motion earthquake acceleration which reproduces the frequency content displayed in Figures 3.7-1 through 3.7-4 was developed (see Figure 3.7-5) by using a mathematical model described by Shinozuka (Reference 3.7-3). It consists of a duration T

(T was set at 15 sec) of a stationary Gaussian process with zero mean and a specified auto-correlation function which corresponds to the mean-square spectral density function.

The italicized information is historical and was provided to support the application for an operating license.

$$S_g(\omega) = \frac{S}{\left(\omega^2 - \omega_g^2\right)^2 + 4\zeta_g^2\omega^2\omega_g^2} \qquad \text{(Eq.3.7.1.2-1)}$$

where ω_g, ζ_g and S are positive parameters which will be determined such as to conservatively represent the frequency content of the ground acceleration displayed in Figures 3.7-1 through 3.7-4. This mathematical model with zero mean and the mean-square spectral density function defined by equation (3.7.1.2-1) was simulated by way of the following series:

$$\ddot{x}_g(t) = s\left(\frac{2}{N}\right)^{1/2} \sum_{k=1}^{N} \cos\left(w_k t + f_k\right) \qquad \text{(Eq. 3.7.1.2-2)}$$

where, $\ddot{x}_g(t)$ is the mathematical model of earthquake acceleration (a Gaussian random process defined above), and

$$\sigma = \left[\int_{-\infty}^{+\infty} S_g(\omega).\ d\omega\right]^{1/2} \qquad \text{(Eq. 3.7.1.2-3)}$$

is the standard deviation of the process $\ddot{x}_g(t)$; ω_k(=1,2,...,N) are independent random variables identically distributed with the density function $g(\omega) = g(\omega k)$ obtained by normalizing $S_g(\omega)$

$$g(\omega) = S_g(\omega) / \sigma^2 \qquad \text{(Eq. 3.7.1.2-4)}$$

and ϕ_K are independent random variables identically distributed with the uniform density $1/2^{\pi}$ between 0 and 2^{π}.

The sample function, $\ddot{x}_g(t)$, which was chosen to represent the random process $x_g(t)$ was corrected and optimized locally.

The response spectra derived from the synthetic record of earthquake acceleration envelope the design response spectra at all damping values used in the design. The comparisons of the synthetic and design response spectra are shown in Figures 3.7-6 through 3.7-10 for the

horizontal component of the OBE. *The spectra were calculated at a set of discrete values for circular frequency (ω_o, $\omega_o r$, $\omega_o r^2$,..., $\omega_o r^{n-1}$) forming a geometric progression. To ensure that the error due to the harmonic component of the simulated earthquake acceleration which contributes most to the response spectrum value is limited to 10%, the ratio of the geometric progression, r, was taken equal to 1.0196 for the damping coefficient of 2.0% of critical damping. This ratio corresponds to a period interval varying from 0.003 sec at a period of 0.03 sec to 0.010 sec at a period of 0.50 sec. The same intervals were used in computing the response spectra at other damping values.*

3.7.1.3 Critical Damping Values

The specific percentage of critical damping values used in dynamic analysis for Category I SSCs are shown in Table 3.7-1 and are based on the recommendations of Reference 3.7-1. Damping values for foundation materials (soils) are also shown in Table 3.7-1.

3.7.1.4 Supporting Media for Seismic Category I Structures

Table 3.7-2 provides a description of the foundation/supporting media for Seismic Category I structures.

All of the buildings/structures shown in Table 3.7-2 have independent foundations. Bedrock was encountered at approximately 525 ft beneath the plant grade (+440 ft 6 in. msl). See Section 2.5 for soil layering characteristics, shear wave velocity, shear modulus, and soil density.

3.7.2 SEISMIC SYSTEM ANALYSIS

Analysis of Seismic Category I SSCs is accomplished by using either the response spectrum method or the time-history method. The results obtained by the response spectrum method of dynamic analysis were used in the design of Seismic Category I structures. Seismic Category I structures were also analyzed by the time-history method of dynamic analysis, using as input a synthetic record of strong motion earthquake acceleration as defined in Section 3.7.1. The results of this analysis are used to generate seismic response spectra for the design and analysis of Seismic Category I systems and components housed in these structures. Alternately, the time histories of structural response at points of attachments/supports of components are used as input in the analysis of systems and components. In the case of Seismic Category I systems and components, the equivalent static load method and dynamic tests are used when conditions allow or require them as described below.

Analysis of Seismic Category I SSCs considers the following stress-producing earthquake effects:

a. Inertia forces determined by a dynamic analysis, and

b. Effects due to differential support displacement, where applicable.

The allowable stress, load combinations, and deformation limits are those set forth in the appropriate codes and design standards which are summarized in Sections 3.8, 3.9, and 3.10.

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 Introduction

Modeling procedures allow the equations of motion of a system to be written as a finite set of simultaneous ordinary differential equations. There are two approaches to the solution of the equations of dynamic equilibrium: the mode-superposition method and the direct integration method. The former was used in the original seismic analysis. It generally consists of two steps, the solution of the characteristic value problem represented by the free vibration response of the system and the transformation to normal coordinates utilizing the mode shapes of the system. This procedure uncouples the equations of motion so that the response of the system in each individual mode may be evaluated independently. Thus, the problem becomes one of solving independent differential equations rather than a set of simultaneous differential equations and, since the system is linear, the principle of superposition holds, and the total response of the system is determined by summing the responses of the individual modes.

3.7.2.1.2 The Equations of Dynamic Equilibrium

The equations of motion of a multi-degree-of-freedom discrete mass damped system subjected to an arbitrary ground motion assuming velocity proportional damping, are expressed in matrix form as follows:

$$\underline{m}\,\underline{\ddot{v}(t)} + \underline{c}\,\underline{\dot{v}(t)} + \underline{k}\,\underline{v(t)} = \underline{-m}\,\underline{I_O}\,\ddot{v}_g(t) \qquad \text{(Eq. 3.7.2.1-1)}$$

where

$\underline{m}$ = *Mass matrix*

$\underline{c}$ = *Damping matrix*

$\underline{k}$ = *Stiffness matrix*

$\underline{I_O}$ = *Unit vector*

$\ddot{v}_g(t)$ = *Ground acceleration*

$\underline{v(t)}$, $\underline{\dot{v}(t)}$, and $\underline{\ddot{v}(t)}$ = *Matrices of displacements, velocities. and accelerations, respectively.*

3.7.2.1.3 Solution of the Equations of Dynamic Equilibrium by Direct Integration

The direct integration method was not used.

3.7.2.1.4 Solution of the Equations of Dynamic Equilibrium by Mode-Superposition

The solutions to the dynamic equilibrium equations used *orthogonality relations and expressing the displacements, velocities, and accelerations in terms of generalized coordinates, (i.e.,* $\underline{v(t)} = \underline{\phi}\ \underline{\dot{Y}(t)},\ \underline{\ddot{v}(t)} = \underline{\phi}\ \underline{\dot{Y}(t)},\ \underline{\ddot{v}(t)} = \underline{\phi}\ \underline{\dot{Y}(t)},$ *) Equation 3.7.2.1-1 is rewritten as the following uncoupled, normal equations of motion:*

$$M_r \ddot{Y}_r(t) + 2 d_r \omega_r \dot{Y}_r(t) + K_r Y_r(t) = \psi_r M_r \ddot{v}_g(t) \qquad \textit{(Eq. 3.7.2.1-2)}$$

where

$M_r = \underline{\phi_r^T}\ \underline{m}\ \underline{\phi_r}$ = *Generalized mass for the r^{th} mode;*

$d_r = \dfrac{\underline{\phi_r^T}\ \underline{c}\ \underline{\phi_r}}{2 \omega_r \underline{\phi_r^T}\ \underline{m}\ \underline{\phi_r}}$ = *Damping ratio of the r^{th} mode (damping ratio for the r^{th} mode is obtained using a weighted average as described in Section 3.7.1)*

$\psi_r = \dfrac{\underline{\phi_r^T}\ \underline{m}\ \underline{I_\phi}}{\underline{\phi_r^T}\ \underline{m}\ \underline{\phi_r}}$ = *Participation factor for the r^{th} mode;*

ω_r = *undamped circular frequency of the r^{th} mode;*

$\underline{Y(t)}$ = *time dependent normal coordinate vector;*

$\underline{\phi r}$ = *mode shape matrix for the r^{th} mode;*

$\underline{\phi_r^T}$ = *transpose of $\underline{\phi_r}$.*

The undamped circular frequencies, ω, are calculated from

$$\left[\underline{k} - \omega^2 \underline{m}\right] = 0 \qquad \textit{(Eq. 3.7.2.1-3)}$$

and the mode shape matrix for the rth mode is obtained from

$$\left[\underline{k} - \omega_r^2 \underline{m}\right]\underline{\phi_r} = 0 \qquad \textit{(Eq. 3.7.2.1-4)}$$

The solution of the differential equation 3.7.2.1-2, for the case of at-rest initial conditions is

$$Y_r(t) = \frac{\psi_r}{\omega_r\sqrt{1-\lambda_r^2}\int_o^t \ddot{v}_g(\tau)e^{-\lambda_r\omega_r(t-\tau)}} \sin\left[\omega_r\sqrt{1-\lambda_r^2}(t-\tau)\right]d\tau \qquad \textit{(Eq. 3.7.2.1-5)}$$

For small damping ratios, λr, the above solution is approximated by:

$$Y_r(t) = \frac{\psi_r}{\omega_r}\int_o^t \ddot{v}_g(\tau)e^{-\lambda_r\omega_r(t-\tau)} \sin[\omega_r(t-\tau)]d\tau \qquad \textit{(Eq. 3.7.2.1-6)}$$

There are two methods of dynamic analysis that were used to solve the multi-degree-of-freedom problems: the response spectrum method and the time-history method.

3.7.2.1.5 Response Spectrum Method of Analysis

If the design earthquake is specified in terms of a response velocity spectrum, Equation 3.7.2.1-6 becomes

$$Y_r(t)\,max = \frac{\psi_r S_{vr}}{\omega_r} \qquad \textit{(Eq. 3.7.2.1-7)}$$

where: S_{vr} = Spectral velocity for the r^{th} mode

$$S_{vr} = \left|\int_o^t \ddot{v}_g(\tau)e^{-\lambda_r\omega_r(t-\tau)} \sin[\omega_r(t-\tau)]d\tau\right| max \qquad \textit{(Eq. 3.7.2.1-8)}$$

The maximum modal displacements, $\underline{v}_{r\,max}$, for the r^{th} mode is

$$\underline{v}_{r\,max} = \underline{\phi}_r\frac{\psi_r S_{vr}}{\omega_r} \qquad \textit{(Eq. 3.7.2.1-9)}$$

where S_{vr} = *spectral velocity for the rth mode.*

If the design earthquake is specified in terms of a response acceleration spectrum instead of a velocity spectrum, the maximum modal displacements, $\underline{v}_{r\,max}$, of the structure for the r^{th} mode are

$$\underline{v}_{r\,max} = \underline{\phi}_r \frac{\psi_r S_{ar}}{\omega_r^2} \qquad \text{(Eq. 3.7.2.1-10)}$$

where S_{ar} = *Spectral acceleration for the r^{th} mode.*

The maximum modal inertia forces, $\underline{F}_{r\,max}$*, for the r^{th} mode are computed from*

$$\underline{F}_{r\,max} = \underline{k}\ \underline{v}_{r\,max} \qquad \text{(Eq. 3.7.2.1-11)}$$

When the maximum modal displacements and modal inertia forces are known, the other modal quantities such as shears and moments are computed for each mode by conventional structural analysis procedures.

3.7.2.1.5.1 Combination of Modal Response. In a response spectrum modal dynamic analysis, if the modes were not closely spaced (i.e., if the frequencies differ from each other by more than 10% of the lower frequency), the modal responses were combined by the square-root-of-the-sum-of-the-squares (SRSS) method as described in Section 3.7.2.1.5.1.1. If two or more frequencies differ from each other by less than 10%, their modal responses were first combined by the absolute sum method and then combined with other individual modal responses by the SRSS method. For some nuclear steam supply system (NSSS) equipment, a double sum method, as described in Section 3.7.2.1.5.1.2 was used to evaluate the combined response. In a time-history method of dynamic analysis, the vector sum at every step was used to calculate the combined response. The use of the time-history analysis method precluded the need to consider closely spaced modes.

3.7.2.1.5.1.1 Square Root-of-the-Sum-of-the-Squares Method. Mathematically, this SRSS method is expressed as follows:

$$R = \left[\sum_{i=1}^{n} (R_i)^2 \right]^{1/2} \qquad \text{(Eq. 3.7.2.1-12)}$$

where:

R = *Combined response*

R_i = *Response in the i^{th} mode*

n = *Number of modes considered in the analysis.*

3.7.2.1.5.1.2 Double Sum Method. This method is defined mathematically as

$$R = \left[\sum_{k=1}^{N} \quad \sum_{s=1}^{N} \quad \left| R_k \ R_s \right| \varepsilon_{ks} \right]^{1/2} \qquad \textit{(Eq. 3.7.2.1-13)}$$

where

R = *Representative maximum value of a particular response of a given element to a given component of excitation*

R_k = *Peak value of the response of the element due to the k^{th} mode*

N = *Number of significant modes considered in the modal response combination*

R_s = *Peak value of the response of the element attributed to s^{th} mode*

where:

$$\varepsilon_{ks} = \left[1 + \left(\frac{(\omega_k - \omega_s)}{(\rho_k \omega_k + \beta_s \omega_s)}^2 \right)^{-1} \right]$$

in which:

$$\omega_k = \omega_k \left[1 - \beta_k^2 \right]^{1/2}$$

$$\beta_k = \beta_k + \frac{2}{t_d \omega_k}$$

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.

3.7.2.1.6 Time-History Method of Analysis

The time-history of ground acceleration, $\ddot{v}_g(t)$, is defined at discrete time intervals. *The acceleration is approximated by a segmentally linear function and the solution to Duhamel's Integral (Equation 3.7.2.1-5) is obtained by using a step-by-step integration procedure (Reference 3.7-7).*

$Y_r(t)$ is computed as a function of time for r = 1,2,3,…,n, where n is the number of significant modes of the system. The modal displacements, $\underline{v_r(t)}$, at the time t for the r^{th} mode, are then calculated from

$$\underline{v_r(t)} = \underline{\phi_r} Y_r(t) \qquad \textit{(Eq. 3.7.2.1-14)}$$

The total displacements, $\underline{v(t)}$, of the structure at any time, t, are obtained by adding the individual modal displacements at time t:

$$\underline{v(t)} = \underline{v_1(t)} + \underline{v_2(t)} + \cdots + \underline{v_n(t)} \qquad \textit{(Eq. 3.7.2.1-15)}$$

The inertia forces, $\underline{F_r(t)}$, at time t, for the r^{th} mode are determined from

$$\underline{F_r(t)} = \underline{k}\ \underline{v_r(t)} \qquad \textit{(Eq. 3.7.2.1-16)}$$

The total inertia forces, $\underline{F(t)}$, on the structure at any time t, are obtained by adding the individual modal inertia forces at time t.

$$\underline{F(t)} = \underline{F_1(t)} + \underline{F_2(t)} + \cdots + \underline{F_n(t)} \qquad \textit{(Eq. 3.7.2.1-17)}$$

Once the time-histories of the displacements and inertia forces have been determined, the time-histories of internal forces, such as shears and moments, for each mode are determined by conventional structural analysis procedures. The total internal forces are obtained by adding the internal forces from each mode at each increment of time. For example, the matrix of the desired moments, $\underline{M(t)}$, is calculated from

$$\underline{M(t)} = \underline{M_1(t)} + \underline{M_2(t)} + \cdots + \underline{M_n(t)} \qquad \textit{(Eq. 3.7.2.1-18)}$$

where $\underline{M_1(t), \cdots, M_n(t)}$ are the time-histories of moments in the individual modes. The maximum values of the internal forces are determined and used for design.

3.7.2.1.7 Analysis for Differential Support Displacements

Certain Seismic Category I systems (piping runs, electrical raceways and supports, duct runs, etc.), and particularly those spanning between different structures are subject to differential support displacements. Seismic Category I system components so effected are analyzed for such effects. The relative support displacements are obtained from the dynamic analysis of structures and are imposed on the systems analyzed thus determining through a static analysis the additional stresses due to relative support displacements.

Stresses due to relative displacements of supports for piping runs are combined with other stresses as described in Section 3.9.3.1.1.7.

For Seismic Category I raceways and cables spanning between different structures subject to differential movements, a flexible transition is made in the system. The transition includes a slack section in cables and a flexible section in the raceways. The slack in the cable sections and flexibility in the raceway sections are sufficient to accommodate the expected differential movements.

Conduit crossing expansion joints or vibration joints in concrete slabs are provided with suitable vibration or expansion fittings to compensate for the building vibration, expansion, and contraction.

For Seismic Category I ductwork spanning between different structures subject to differential movements, a flexible transition is made. The transition includes a slack section in the ductwork and sufficient flexibility in the system to accommodate the expected differential movements.

3.7.2.1.8 Dynamic Analysis of Seismic Category I Structures, Systems, and Components

Seismic Category I SSCs are analyzed for earthquake effects using either response spectrum or time-history methods of analysis.

3.7.2.1.8.1 Dynamic Analysis of Buildings. All Seismic Category I structures were analyzed by the response spectrum method of analysis and the results of the analyses were used in the design of these structures. Modal maxima were combined as described in Section 3.7.2.1.5. Seismic Category I structures for which floor response spectra are required were also analyzed by the time-history method of analysis. The analyses were performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses resulting from any one horizontal and the vertical excitations were considered to act simultaneously and were added using the absolute sum method.

3.7.2.1.8.2 Dynamic Analysis of Piping Systems. Each pipe line was idealized by a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system was determined using the elastic properties of the pipe. This includes the effects of torsion, bending, shear, and axial deformations as well as change in stiffness due to curved members. The mode shapes and the undamped natural frequencies were determined. The dynamic response of the system was calculated by using either the response spectrum or time-history method of analysis. When the piping system is anchored and supported at points with different excitations, the response spectrum analysis was performed using the envelope response spectrum of all attachment points. Alternatively, the multiple excitation analyses methods may be used where acceleration time histories or response spectra are applied to all piping system attachment points.

The analyses were performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South or East-West. Any one of the horizontal and vertical excitations were considered to act simultaneously. Moments and forces from each of the horizontal and vertical excitations considered, are added as described in C.3 discussion for Regulatory Guide 1.92.

The relative displacements between anchors were determined from the dynamic analysis of the structures. These relative displacements are then used in a static analysis to determine the additional stresses imposed on the piping system.

An alternate simplified method of dynamic analysis was used for cold and/or limber piping systems. This is the equivalent static load method for piping. This method consists of applying constant horizontal and vertical load factors conservatively derived from seismic floor response spectra.

The description of the method is as follows: Enveloped seismic building response spectra were derived from widened seismic floor response spectra. (The widening of the building response spectra is described in Section 3.7.2.5). The piping system was then supported seismically such that the minimum fundamental frequency was chosen to be above the spectral peak of the enveloped response spectrum for any given span of pipe between adjacent supports. Thus, the initial maximum seismic support spans were analytically determined from this model for the chosen fundamental frequency. The static "g" levels acting on the piping system were then obtained from the enveloped response spectra assuming that the frequencies of the piping system is at or above the chosen frequency. These maximum spans were modified, if required, so that the maximum stresses did not exceed a conservative value of maximum stress based on the American Society of Mechanical Engineers (ASME) Code allowables and a limiting piping deflection between supports.

In the application of the alternate simplified method, a conservative static "g" loading was chosen for all piping systems when this approach was used irrespective of the building or building elevation. This simplified the work and results in different amounts of conservatism for different piping systems. To confirm the adequacy of the alternate simplified method, a study was performed for several representative piping systems. Pipe stress and pipe support loads were calculated for these representative systems using response spectrum analysis method. Results were examined to confirm that both pipe stresses and pipe support loads were calculated using the equivalent static load method.

3.7.2.1.8.3 Dynamic Analysis of Equipment. Equipment is idealized by a mathematical model consisting of lumped masses connected by elastic members or springs. Results for selected Seismic Category I equipment are given in Table 3.9-2. The dynamic response of the system was originally calculated by using the response spectrum method of analysis. When the equipment is supported at two or more points at different elevations, the response spectrum

analysis was performed by using the response spectra at the elevation near the center of gravity of the equipment as the design spectra for the NSSS equipment, and for balance-of-plant using the envelope of response spectra for supports. Modal maxima were combined as described in Section 3.7.2.1.5. The analyses were performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses resulting from any one horizontal or vertical excitation are considered to act simultaneously and the absolute values are added directly, as described in Sections 3.7.2.6 and 3.7.2.7.

The relative displacements between anchors are determined from the dynamic analysis of the structures. All cases of relative displacement between anchors are considered. If significant, these relative displacements are then used in a static analysis to determine additional stresses imposed on equipment. Further details are given in Section 3.7.2.1.8.3.1 for the NSSS equipment and Section 3.7.3.9 for all other equipment. The cases where the relative displacements between anchors are insignificant and thus neglected in the analysis are those cases where the equipment is supported on a single structural element as a floor slab or wall. Typical examples where relative displacements are considered insignificant are a bank of electrical switchgear located on and anchored to a single floor slab, a diesel generator set located on and anchored to a single isolated foundation, and an air handling unit located on and anchored to a single wall.

3.7.2.1.8.3.1 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.

Relative displacement between the supporting points induces additional stresses in the supported equipment. These stresses can be evaluated by performing a static analysis where each supporting point is displaced a prescribed amount. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. The maximum relative displacements obtained from the time history were used to calculate stresses statistically.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the modal relative displacement was used. The mathematical model of the equipment was then subjected to the modal relative displacement at its supporting points. This procedure was repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the structure. The total stress due to relative displacement was obtained by combining the modal results using the method described in Section 3.7.2.1.5.1.

When a component is covered by the ASME Boiler and Pressure Vessel (B&PV) Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

3.7.2.1.9 Equivalent Static Load Method

This method of analysis is used for design of certain systems:

a. Unless otherwise justified, for systems which can be realistically represented by a simple model, the equivalent static acceleration corresponds to the response spectrum value at the system natural frequency times a factor of 1.5 providing the spectrum is single peaked. If the response spectrum has multiple peaks and the natural frequency lies between two peaks or below a peak, the equivalent static acceleration corresponds to the highest peak located above the natural frequency times a factor of 1.5. If the natural frequency of the system is not known, the equivalent static acceleration corresponds to the response spectrum peak (highest) times a factor of 1.5. If the system natural frequency is at or above the zero period acceleration (ZPA) of the response spectrum, the equivalent static acceleration is the ZPA with no multiplication factor required.

b. Equivalent static load method for piping is described in Section 3.7.2.1.8.2.

3.7.2.1.10 Dynamic Testing

When certain Seismic Category I equipment and components potential functional failure cannot be evaluated analytically (i.e., when structural integrity alone cannot ensure the design intended function), dynamic testing is used to ensure operability. For example, dynamic tests of electrical items are performed in accordance with the requirements of IEEE Standard 344 (Reference 3.7-8). Test performance data and results are obtained either from previously tested comparable equipment or from the actual testing of equipment supplied. When seismic testing is impractical, a combination of test and analysis is used. Other dynamic test procedures which conservatively simulate the seismic conditions for the equipment are also used when found acceptable by the engineer.

Seismic Category I equipment which is difficult to represent by a mathematical model or which is required to demonstrate its ability to remain operating without changing the mode of its operation (such as level switch which should not switch from “on” to “off” or vice versa during the earthquake) was subjected to actual vibration inputs on shake tables. These shake tests were performed by qualified laboratories.

The seismic qualification tests conducted in the laboratory generally consist of the following:

a. The equipment was mounted on the shake table in such a manner as to represent its installed condition;

b. Sine sweep tests were performed covering all practicable frequency ranges with constant or variable acceleration levels to determine the natural resonant

frequencies of the equipment. This procedure enables the determination of the predominant resonant frequencies, by monitoring the output response; and

c. Proof testing was then performed to establish the capability of equipment to function during the particular seismic event and withstand the effects of the particular seismic event, represented by the required response spectra at the appropriate damping level. This was accomplished by using one of the following methods.

   1. Sine dwell tests

      This test utilizes a sine wave function with one of the equipment natural frequencies and acceleration levels equal to or greater than the corresponding maximum floor acceleration as input. The test duration is generally 30 sec, during which time the behavior of the equipment is observed and recorded. This test is performed at all equipment resonances and at frequencies spaced apart throughout the frequency range. Alternately, the test may be performed only at the equipment resonances when justified.

   2. Sine beat tests

      A sine beat function with the number of beats and cycles per beat corresponding to the equipment natural frequency and with predetermined acceleration level equal to or greater than the corresponding maximum floor acceleration, is used as input motion to test and record the behavior of the equipment.

   3. Random motion tests

      A random waveform motion consisting of frequency bandwidths one-third octave apart over the practicable frequency range is used in this test. The amplitude of each frequency bandwidth is independently adjusted in each axis until the test response spectra exceeds the required response spectra. The behavior of the equipment was observed and recorded to ensure its capability to withstand the input vibrations.

3.7.2.2 Natural Frequencies and Response Loads

A summary of natural frequencies, natural mode shapes, and modal responses (displacements, accelerations, moments, and shears) were provided for the significant modes of the reactor building and are shown in Figure 3.7-11 and Tables 3.7-3 through 3.7-15. The modal responses are for the OBE and SSE, for one horizontal (North-South) and the vertical

directions. All Seismic Category I structures were also analyzed by the time-history method of analysis, using as input the synthetic motion obtained as described in Section 3.7.1.2, to develop floor response spectra to be used in design of systems, components, and equipment housed in these structures. Floor response spectrum curves were computed at all lumped-mass points of Seismic Category I structures as described in Section 3.7.1.1. These curves are for the SSE and the OBE, for two horizontal orthogonal (North-South and East-West) and the vertical directions, with equipment damping values of 0.5, 1, 2, and 5 of critical damping and for equipment natural periods ranging from 0.03 to 2.50 sec per cycle. Typical floor response spectra are shown in Figures 3.7-12 and 3.7-15 through 3.7-21 for the reactor building at refueling floor and mat elevations.

3.7.2.3 Procedure Used for Modeling

Seismic Category I SSCs were modeled as a system of lumped masses and springs suitable for mathematical analysis. Each system analyzed is thus replaced by a discrete set of lumped masses, springs, and dashpots, idealizing both the inertia and stiffness properties of the system. The details of the mathematical models are determined by the complexity of the actual structure and the information required for the analysis.

Seismic subsystems, such as equipment and piping [with the exception of the reactor pressure vessel (RPV)], were decoupled from the structure as described in Section 3.7.2.3.1 by lumping their mass contribution to the structural model.

The seismic subsystems were then analyzed separately using the seismic input from the analysis of the structure. Where a subsystem is comparatively rigid and rigidly connected to the primary system, only the mass of the subsystem at the support point is included in the primary system model. Where the subsystem is flexible, such as pipe supported by hangers or equipment mounted on nonrigid supports, a coupled dynamic analysis was performed for both the subsystem and primary system.

The criteria used for decoupling piping systems, other than the NSSS piping systems, to establish the analytical models for seismic analysis are discussed in Section 3.9.3.1.18.5, and have been demonstrated as equivalent to the decoupling criteria outlined by Paragraph II.36 of Reference 3.7-15. The criteria for the NSSS main steam and recirculation piping systems are discussed below.

For NSSS systems and components, the ASME B&PV Code Section III requires that piping systems be designed and analyzed as complete systems from anchor to anchor. A complete piping system must include the subject piping system, all major branch line piping, and all equipment reached to the pipe which influences stresses and movement of the pipe. The piping systems within the General Electric contractual scope for which seismic analysis is performed are as follows:

a. Main steam piping from the RPV in the first anchor at the penetration head fittings, and

b. Reactor recirculation piping bound by the RPV nozzles.

The criteria employed for decoupling the main steam and recirculation piping systems to establish the analytical models for seismic analysis are given below:

a. Small branch lines (6 in. diameter and less) are decoupled from the main steam and recirculation piping systems and analyzed separately; and

b. 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 analytic code jurisdiction boundary purposes. The RPV is very stiff compared to the piping system and, therefore, during normal operating conditions, the RPV is assumed to act as an anchor. Penetration assemblies (heat fitting) are also 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, reactor core isolation cooling (RCIC), residual heat removal (RHR) supply or RHR return] head fitting is sufficient to decouple the penetration assembly from the process pipe. General Electric analysis indicates that a satisfactory minimum stiffness for this attachment point is equivalent to the stiffness in bending and torsion of a cantilever equal to a pipe section of the same size as the process pipe and equal in length to three times the process pipe outer diameter.

Application of above criteria for analyses of the subject piping systems is as follows:

a. The main steam piping upstream of the outboard isolation valve (OBIV) is decoupled from the piping downstream of the OBIV at the first anchor at the penetration head fitting, and

b. The major branch lines which affect the stresses in the main steam and recirculation piping are incorporated in the analytical model for analysis. The system is not decoupled until it reaches the following virtual anchors:

    1. RCIC steam piping upstream of the OBIV is decoupled from the piping downstream of the OBIV at the first anchor at the penetration head fitting.

    2. Safety/relief valve discharge piping originating from the relief valve discharge flange is decoupled at each safety/relief discharge line first anchor at the suppression pool floor.

3. Residual heat removal supply and return piping (connected with the recirculation piping) upstream of the OBIV are each decoupled from the piping downstream of the OBIV at the first anchor at the penetration head fitting.

3.7.2.3.1 Modeling of Structures

In constructing the mathematical model of a structure, the locations for lumped masses were chosen at floor levels and points considered of critical interest such as supports/anchors for equipment and systems. The lumped mass comprises the weight of afferent walls, floors and other dead loads, including weight of systems supported on or hanging from the floor (pipes, ducts, raceways, etc.), and the weight of equipment mounted on the floor. It has been estimated that the equipment load constitutes, generally, less than 10% of the total weight associated with any lumped mass and is not expected to significantly effect the overall behavior of the structure. Between mass points, the structural properties were reduced to uniform segments of cross-sectional area, effective shear areas, and moments of inertia. Thus, the masses of the system were connected by weightless linear elastic springs which account for the axial (direct), flexural, and shear stress effects of the structure. Soil-structure dynamic interaction effects were considered by attaching basemat, assumed rigid, a set of equivalent springs, and dashpots as described in Section 3.7.2.4.

Typical mathematical models for the reactor building soil-structure lumped-mass system for horizontal and vertical input motions and the associated reactor building section are shown in Figures 3.7-13 and 3.8-1.

3.7.2.3.2 Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at the nodes and connected by a weightless elastic member, representing the physical properties of each segment. The pipe lengths between mass points must be sufficiently short, so as not to affect the accuracy of the dynamic analysis. The lengths utilized were determined by parametric studies. The resulting lengths are such that frequencies computed on the basis of a simply supported beam are no less than 33 Hz for all piping including the NSSS systems and components. All concentrated weights on the piping system such as main valves, relief valves, including valve operators with extended structures, and points of significant change in the geometry of the system, are modeled as lumped masses. The torsional effects of the valve operators and other equipment with offset centers of gravity, with respect to centerline of the pipe, are included in the analytical model. If the torsional stress is less than 500 psi, it is considered to be permissible to neglect this effect. Equipment nozzles are generally considered as boundaries for the piping systems. Inline spring-mounted equipment is modeled as a lumped mass.

3.7.2.3.3 Modeling of Equipment

For dynamic analysis, Seismic Category I equipment is represented by lumped mass systems that consist of discrete masses connected by weightless springs. The criteria used to lump masses are

a. 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 are estimated to amount to a significant percentage (approximately greater than 10%) of the total stresses calculated from lower modes;

b. Mass is lumped at any point where a significant concentrated weight is located. Examples are the motor in the analysis of pump motor stand, or the impeller in the analysis of pump shaft;

c. If the equipment has a free-end overhang span whose flexibility is significant compared to the center span, a mass is lumped at the overhang span; and

d. 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 conservatively 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 equipment frequencies are such that the floor spectra peak is in the lower frequency range. If such is not the case, the model is adjusted to give more conservative results.

3.7.2.3.4 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the RPV and internals were based on a dynamic analysis of an entire RPV-building complex with appropriate forcing function supplied at ground level. For this analysis, the seismic model of the RPV and internals, as shown in Figure 3.7-14, and the mathematical model of the building were coupled together.

This mathematical model consists of lumped masses connected by elastic (linear) members. Using the elastic properties of the structural components, the stiffness properties of the model are determined. This includes the effects of both bending and shear. To facilitate hydrodynamic mass calculations, several mass points (representing fuel, shroud, and vessel) are selected at the same elevation. Mass points are located at all points of critical interest such as anchors, supports, and points of discontinuity, etc. In addition, mass points are chosen such

that the total mass of the structure is generally uniformly distributed over all the mass points, and the full range of frequency of response of interest is adequately represented.

The various lengths of control rod drive (CRD) housings were grouped into two representative lengths. These lengths represent the longest and shortest housings to adequately represent the full range of frequency response of the housings. The high fundamental nature frequencies of the CRD housings result in very small seismic loads. 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 light components such as jet pumps, in-core guide tubes and housings, spargers, and their supply headers. This reduces the complexity of the dynamic model. If the seismic responses of these components are needed, they can be determined after the system response has been found.

The presence of a fluid or other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV were accounted for by a hydrodynamic mass matrix, which links the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in Reference 3.7-11. The seismic model of the RPV and intervals has two horizontal coordinates for each mass point. The remaining translational coordinate (vertical) is excluded because the vertical frequencies of the RPV and internals are well above the significant horizontal frequencies. All support structures, building, and containment walls are negligible. A separate generic and applicable vertical analysis is performed. Dynamic loads due to vertical motion are added to or subtracted from (whichever is more conservative), the static weight of components. The two rotational coordinates about each node point are excluded because of the moment contribution of rotary inertia from surrounding nodes. Since all deflections are assumed to be within the elastic range, the rigidity of some components is represented by equivalent linear springs.

The shroud support plate is loaded in its own plane during a seismic event and hence is extremely stiff. Therefore, the shroud support plate is 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 rotational spring.

3.7.2.4 Soil/Structure Interaction

Soil/structure interactions were taken into account by coupling the structural model with the foundation medium.

The lumped mass-spring method was used to represent soil/structure interactions; and was obtained from a simplified mechanical analog to the model of a rigid mat resting on the surface of an elastic half space. The resulting compliance functions were approximated by the

frequency independent springs and dashpots (Reference 3.7-5) and are listed in Table 3.7-16. The spring and dashpot constants depend on the geometry of the foundation and on the dynamic properties of the soil. The selected ranges of values for the equivalent dynamic shear modulus, G (which are used in the dynamic analysis), were derived (Reference 3.7-6) by interpreting data from laboratory tests and measurements of seismic velocities adjusted for calculated strains. The ranges of G values are as follows:

Mode	Lower Bound (ksi)	Average Value (ksi)	Upper Bound (ksi)
Horizontal translatory and rocking	50	75	100
Vertical translatory	80	120	160

Based on additional studies performed for CGS, it was found that the use of the elastic half-space/compliance function method and the finite element method for soil/structure interaction analysis yield very comparable results. Therefore, either method is considered acceptable for seismic soil/structure interaction analysis of the CGS plant.

3.7.2.5 Development of Floor Response Spectra

All Seismic Category I structures were analyzed by the time-history method of analysis to obtain time histories of the structural response at points within these structures. The acceleration histories are used to generate floor response spectra.

The floor response spectra are computed for the SSE and the OBE for the two horizontal orthogonal directions and the vertical direction.

Spectral values, (the maximum response of a single degree-of-freedom oscillator) are obtained using a step-by-step integration (Reference 3.7-7). The analytical solution assumed that the acceleration histories of structural response are linear within the time interval of 0.02 sec. The integration was performed at either the 0.02 sec time intervals or at 0.05 times the natural period of the single degree-of-freedom oscillator, whichever was smaller.

The discrete periods or frequencies used in the calculation of the floor response spectra are in compliance with the values suggested in Regulatory Guide 1.122, Revision 0, September 1976.

To account for variations in structural frequencies, the peaks of the computed floor response spectra associated with each of the structural frequencies were widened by no less than +15%

for all Seismic Category I structures analyzed using lumped-mass stick models. The only exceptions were the standby service water pump house (SSWP) spectra where peaks were widened by only +10%. In lieu of performing an analysis to justify the 10% peak broadening of the SSWP spectra, the effects of using a 15% peak broadening of the same response spectra were reviewed. Examination of representative Seismic Category I equipment demonstrates that the equipment meets the 15% broadened curves. In addition, these response spectra are conservative since they are developed using a lumped mass spring model and conservative soil damping values of Table 3.7-1.

For the primary metal containment, the RPV, the RPV pedestal, and the sacrificial shield wall, the response spectra were widened by no less than +10%. However, these spectral data are obtained from the seismic analysis of the finite element building model which included the soil-structure interaction effects. This analysis was performed in accordance with the provisions of Regulatory Guides 1.60, 1.61, 1.92, and 1.122, and Standard Review Plan Section 3.7.2.

3.7.2.6 Three Components of Earthquake Motion

The use of three components of earthquake motion, as described by Regulatory Guide 1.92, Revision 1, was not a requirement for the issuance of the CGS construction permit. The total seismic response was calculated by combining the response calculated from analyses due to one horizontal and one vertical seismic input.

Two sets of seismic results were obtained. First the maximum value of the horizontal component of the earthquake was assumed to act in one horizontal direction simultaneously with the vertical component, and the loads were computed for this combination.

The maximum value of the horizontal component of the earthquake was assumed to act perpendicular to the direction previously assumed and simultaneously with the vertical component, and loads were computed for this combination. The larger of these two loads, at each point in the system, was used for design.

This method of analysis was based on the fact that the maximum resultant value of the horizontal component of the earthquake is determined when the horizontal component of the SSE is specified. This method conservatively assumes that the maximum horizontal and vertical components of the earthquake response occur simultaneously.

In accordance with Regulatory Guide 1.92, Revision 1, an alternative procedure is also acceptable for combining seismic responses, when designing structures, systems, or components submitted to the simultaneous action of three orthogonal earthquake motions. In this case the combined three-dimensional earthquake effect can be obtained for any structural response as the SRSS of the codirectional maximum responses caused by each of the three

earthquake components acting separately. Results of either modal response spectrum or time-history dynamic analyses can be processed this way.

The SRSS method of superposition, as summarized herein, may be used in conjunction with any mathematical model of SSCs, provided a complete set of three earthquake components is utilized as input in the dynamic analysis of that particular model. It should be noted that a comparison of response spectra clearly demonstrates that the original design basis (2-D method ABS) is more conservative than the SRSS/3 component method for all frequencies larger than 1.25 Hz. In the frequency range of interest for the CGS plant (approximately greater than 5 Hz), the margin of conservatism is very significant.

3.7.2.7 Combination of Modal Responses

When the response spectrum method of modal analysis is used, modal maxima are combined as described in Section 3.7.2.1.5.

3.7.2.8 Interaction of Non-Category I Structures With Seismic Category I Structures

The interfaces between Seismic Category I and non-Category I structures and plant equipment have been designed for the dynamic loads and displacements of both Seismic Category I and non-Category I structures and plant equipment. In addition, all non-Category I structures meet one of the following requirements:

a. The collapse of any non-Category I structure does not cause the non-Category I structure to strike a Seismic Category I structure or component,

b. The collapse of any non-Category I structure does not impair the integrity of Seismic Category I structures or components,

c. The non-Category I structures are analyzed and designed to prevent their failure under SSE in a manner such that the margin of safety of these structures is equivalent to that of Seismic Category I structures, and

d. The collapse of non-Category I structures will not prevent the functioning of Seismic Category I structures or components.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Variations in structural properties, damping, soil properties, and soil/structure interaction parameters could result in the shifting of calculated periods to resonant peaks in the floor response spectra. To account for the effects of such variations on the shape of calculated floor response spectra, smooth design envelopes which incorporate a minimum shift of the periods corresponding to calculated resonant peaks were developed and used in seismic design of

systems, equipment and components. The shift in resonant peaks covers at least the calculated variations in natural periods due to the probable variation in soil properties of +33-1/3% (see Section 3.7.2.4) and is never less than 10%. The actual shift in resonant peaks exceeds in some cases 15% as may be seen from Figures 3.7-12 and 3.7-15 through 3.7-21. Less significant variations in peak response values are also expected as a result of variations in structural properties used in dynamic analysis. The use of conservative damping coefficients and of smooth design spectrum envelopes in seismic design of systems, equipment, and components is adequate to account for variations in peak response values.

3.7.2.10 Use of Constant Vertical Static Factors

Seismic Category I SSCs were subjected to a vertical dynamic analysis with the vertical response spectra defining the input, with the exception that a static analysis can be performed in lieu of dynamic analysis as described in Section 3.7.2.1.9.

3.7.2.11 Method Used to Account for Torsional Effects

For nonsymmetrical structures, a stiffness analysis was performed to determine torsional effects on vertical structural elements resisting lateral loads. Inertial forces, determined from dynamic analysis of structures, were applied at the center of mass for each floor. Thus, torsion effects were introduced in each story by applying a twisting moment about the center of rigidity of the story under consideration. This moment is calculated as the sum of the products of the inertial force applied at the center of mass of each floor above and a lever arm equal to the distance from the center of mass of the floor to the center of rigidity of the story. The lever arm was not less than the minimum eccentricity required by the Uniform Building Code (Reference 3.7-9). The torsional moment and story shear were distributed to the resisting elements in accordance with the provisions of the Uniform Building Code.

Symmetrical structures were analyzed in a similar manner for torsional effects using minimum eccentricities between the center of mass and the center of rigidity as defined by the Uniform Building Code.

Calculation of torsional effects using a dynamic analysis that considers coupled translational and torsional degrees of freedom was also performed. The new torsional effects were compared with those loads derived by the design methodology prescribed above. Structures subjected to the new torsional moments were investigated and found to be structurally adequate under the new torsional loads considered in conjunction with the applicable load combinations.

3.7.2.12 Comparison of Responses

Comparisons of structural responses (accelerations) of the reactor building were obtained using (a) the response spectrum method with the site design response spectra, and (b) the time-history method with the simulated time-history of earthquake acceleration described in

Section 3.7.1.2 are presented in Figure 3.7-22. These results demonstrate the conservatism inherent in the simulation process illustrated by Figures 3.7-6 through 3.7-10, and carried over in the calculation of floor response spectra used in seismic design of systems, components, and equipment. A more appropriate comparison was obtained between structural responses (acceleration) of the reactor building obtained using (a) the response spectrum method with response spectra calculated from the simulated earthquake acceleration, and (b) the time-history method. These are also in Figure 3.7-22 and show good agreement between the results obtained using the two methods of dynamic modal analysis.

3.7.2.13 Methods for Seismic Analysis of Dams

No Seismic Category I dams are utilized in this facility.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

Overturning and sliding effects of horizontal seismic loadings were considered in combination with the effects of vertical seismic loadings. The seismic loads used consist of two horizontal orthogonal and vertical components of earthquake motions. Each horizontal component was taken separately and is applied concurrently with the vertical component. For Seismic Category I structures, the results of the dynamic analysis were converted to equivalent static loads at the mass points.

Seismic Category I structures are located above the present groundwater elevation of 380 ft msl. However, uplift and lateral hydrostatic pressures are considered, taking into account the maximum groundwater elevation of 420 ft msl in the event the Ben Franklin Dam is constructed, as discussed in Section 3.4. The uplift and hydrostatic pressures, including seismic effects due to dry and saturated soils, as applicable, are applied concurrently.

To calculate the capability of safety-related structures to resist overturning, the following load combinations were considered:

a. D + E + Q* + Uplift
b. D + W + Q + Uplift
c. D + E′ + Q* + Uplift
d. D + W′ + Q + Uplift

Load combination in item c. above is used since the resulting horizontal and vertical forces produce the maximum overturning effects. The load terms in the load combinations are defined in Section 3.8.4.3.3, except as follows:

a. The dead load, D, also includes the weight of dry and saturated backfill, as applicable, and

b. The uplift force, not included in Section 3.8.4.3.3, is taken as the weight of the water displaced by the structure, acting vertically upward and applied to the bottom surface of the basemat.

The overturning moments and the stabilizing moments were calculated about the lower edge (toe) of the basemat. The safety factor against overturning was calculated by dividing the total stabilizing moment by the total overturning moment.

To calculate the capability of safety-related structures to resist sliding, the load combinations considered are the same as those listed above for calculating the capability of safety-related structures to resist overturning. Load combination c was used and produces the maximum sliding effects. The safety factor against sliding was calculated by dividing the frictional force resisting sliding between the basemat and the soil by the summation of horizontal forces causing sliding.

The factors of safety against overturning and sliding for safety-related structures are tabulated in Section 3.8.5.5.

3.7.2.15 Analysis Procedure for Damping

For structures and components, damping coefficients are selected as discussed in Section 3.7.1.3.

For the foundation materials, either internal or radiational damping or both are considered. The horizontal translation, vertical translation, and rocking motion damping values are determined as described in Reference 3.7-5.

The selected design values were significantly smaller than the calculated values. Table 3.7-1 presents the design values of the damping coefficients used. The formulas presented in Table 3.7-16 were used to calculate the realistic damping coefficients.

For composite structures made up from different materials, when the various components cannot be decoupled due to interaction effects, an approximate weighted average damping value was used for each mode of vibration of the structure. This is accomplished by breaking the mode shapes into their various components, then assigning a damping value to each component depending on the principal action of this component. A weighted average value is then determined for the particular mode under consideration. In this manner, a composite damping value is determined for each mode and the total response is calculated in the regular manner.

The method of obtaining weighted average damping values and how they were applied to the original design was obtained from the following relation:

$$D_n = \frac{D_s E_{sn} + D_h E_{hn} + D_r E_{rn}}{E_{sn} + E_{hn} + E_{rn}} \qquad (Eq.\ 3.7.2.15\text{-}1)$$

where

D_n = *Weighted average damping for the nth mode*

D_s = *Damping ratio for the superstructure*

D_h = *Damping ratio for the horizontal translation*

D_r = *Damping ratio for the rocking motion*

E_{sn} = *Energy stored in the superstructure*

E_{hn} = *Energy stored in the horizontal spring*

E_{rn} = *Energy stored in the rocking spring*

The basis for Equation 3.7.2.15-1 is presented as Equation (4) in Reference 3.7-15.

In a linear dynamic analysis for the NSSS Systems and components the procedure to be utilized to properly account for damping in different elements of a coupled system model is as follows:

a. *A structural damping of the various structural elements of the model are first specified. Each value is referred to as the damping ratio (B_j) of a particular component which contributes to the complete stiffness of the system;*

b. *Perform a modal analysis of the linear system model. This will result in a modal matrix (ϕ) normalized such that $\phi_i^T K \phi_i W_i^2$, where K is the stiffness matrix, W_i the circular natural frequency of mode i and ϕ_i^T is the transpose ϕ, which is a column vector of ϕ corresponding to the mode shape of mode i. Matrix ϕ contains all translational and rotational coordinates; and*

c. *Using the strain energy of the individual components as a weighting function, the following equation can be derived to obtain a suitable damping ratio (B_i) for the i^{th} mode.*

$$B = \frac{\sum_{j=1}^{N} \phi_{i_j}^{T} B_j K_j \phi_{ij}}{w_i^2} \qquad (Eq.\ 3.7.2.15\text{-}2)$$

where

> *N = Total number of structural elements*
>
> *f = Mode shape for mode i (f as transpose)*
>
> B_j *= Percent damping associated with element j*
>
> K_j *= Stiffness contribution of element j*
>
> W_i *= Circular natural frequency of mode i*

The original piping design calculations, as indicated in Table 3.7-1, have considered damping values lower than or equal to those permitted by Regulatory Guide 1.61. For piping reanalyses or for snubber support optimization, the following damping values, as stated in ASME Code Case N-411, may be used in both the OBE and SSE spectrum analyses of the ASME Class 1, 2, and 3 piping systems:

Frequency Range (Hz)	Critical Damping (%)
0-10	5
10-20	5 decreasing linearly to 2
Above 20	2

Subject to the requirements of NRC correspondence regarding Regulatory Guide 1.84 (Reference 3.7-13), the code case may be applied to any piping system including those located in the SSWPs. The SSWP spectral peaks shall be broadened by 15% when Code Case N-411 is applied. It may be noted that the seismic spectra for CGS were developed using either the ground response spectra as defined in Regulatory Guide 1.60, Revision 1, or were properly justified by studies to be similar to those which could have been obtained per Regulatory Guide 1.60. Also, the peak broading requirements of Regulatory Guide 1.122 is met in all cases except for SSWP spectra as noted.

The original design basis for CGS required that responses due to inertial loads be combined with seismic anchor motion loads by the absolute sum method. For snubber optimization or any reconciliation work, based on the recommendation of NUREG-1061, Volume 4, the SSRS methodology for the combination of inertial and seismic anchor motion loads is utilized.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

The general approach to the seismic subsystem analysis is identical to those procedures described in Section 3.7.2 for seismic system analysis, except for the soil/structure interaction effects.

3.7.3.1 Seismic Analysis Methods

The seismic analysis method used to analyze Seismic Category I subsystems is described in Section 3.7.2.1.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Number of Cycles for All Items Except Nuclear Steam Supply System Systems and Components

Fatigue evaluation due to an SSE is not required by ASME Code Section III, since it qualifies as a faulted condition. The OBE is an upset condition and therefore, must be included in fatigue evaluations according to ASME Code Section III.

As a minimum, 50 maximum stress cycles due to OBE are used for fatigue evaluations of BOP piping and components.

3.7.3.2.2 Number of Cycles for Nuclear Steam Supply System Systems and Components

3.7.3.2.2.1 Nuclear Steam Supply System Piping. Fifty peak OBE cycles are postulated for fatigue evaluation.

3.7.3.2.2.2 Other Nuclear Steam Supply System Equipment and Components. To evaluate the number of cycles that exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: (a) May 18, 1940, El Centro NS component 29.4 sec, (b) 1952, Taft N 69° W component, 30 sec, and (c) March 1957, Golden Gate S 80E component, 13.2 sec. The model 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 provide 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 3.7-17, was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% 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:

a. The fundamental frequency and peak seismic loads are found by a standard seismic analysis,

b. The number of cycles which the component experiences are found from Table 3.7-17 according to the frequency range within which the fundamental frequency lies, and

c. For fatigue evaluation, 0.005% of these cycles are conservatively assumed to be at the peak load and 4.5% are assumed to be at or above three-quarter peak. The remainder of the cycles has negligible contribution to fatigue usage.

The SSE has the highest level of response. However, the encounter probability of an SSE is so small that it is not necessary to postulate more than one SSE during the 40-year plant life. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Code Section III.

The OBE is an upset condition and, therefore, must be included in fatigue evaluations according to ASME Code Section III. An investigation of seismic histories for many plants shows that during a 40-year life, it is probable that five earthquakes with intensities 10% of the SSE intensity, and one earthquake approximately 20% of the proposed SSE intensity, will occur. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, 10 peak OBE cycles are postulated for fatigue evaluation.

Table 3.7-18 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.3 Procedure Used for Modeling

The procedure used for modeling for the subsystem dynamic analysis is described in Section 3.7.2.3.

The field location of seismic supports and restraints for Seismic Category I piping and piping system components is selected to satisfy the following two conditions:

a. Restraint locations are chosen sufficiently close to each other to limit the stress and strain of the piping system to acceptable values. Spring supports are not a

factor in seismic analysis. Seismic restraints are constructed sufficiently rigid so as to preclude interaction with the piping system; and

b. Structures are provided of sufficient capacity to support the seismic supports and restraints and to withstand the seismic and/or loss-of-coolant accident (LOCA) loads transferred to the supporting structure by the seismic support and/or restraint. The applicable load combinations in Tables 3.8-5, 3.8-6, 3.8-9, and 3.8-10 are used, depending on the loading conditions to which the structure could be subjected.

The final location of seismic supports and restraints for Quality Class 1, Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices is made to ensure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems.

3.7.3.4 Basis for Selection of Frequencies

All frequencies in the 0.25 to 33 Hz range are considered in the analysis and testing of SSCs. These frequencies cover the natural frequencies of most of the components and structures under consideration. If the fundamental frequency of a component is greater than or equal to 33 Hz, it is treated as rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not usually considered, as they represent very flexible structures and are not normally encountered in this plant.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

When the natural frequency of a structure, equipment, or component is unknown, the item may be analyzed by applying a static force at the center of mass. To account for the possibility of more than one significant dynamic mode, the static force is calculated as 1.5 times the mass times the maximum acceleration from the applicable design response spectra of the point of attachments as described in Section 3.7.2.1.9. For structures, equipment, or components which may be realistically represented by a single degree-of-freedom system, the peak spectral acceleration is used. Equivalent static load method for piping is described in Section 3.7.2.1.8.2.

3.7.3.6 Three Components of Earthquake Motion

The procedure used to consider the three components of earthquake motion for Seismic Category I subsystems is described in Section 3.7.2.6.

3.7.3.7 Procedure for Combining Modal Responses

The procedure used for combining modal responses for Seismic Category I subsystems is described in Section 3.7.2.1.5.

3.7.3.8 Analytical Procedures for Piping

A description of the modeling and analytical procedures applicable to piping systems is described in Sections 3.7.2.3.2 and 3.7.2.1.8.2, respectively.

3.7.3.9 Equipment Components Supported at Multiple Locations with Distinct Inputs

For seismic analysis of equipment and components supported at different elevations and between buildings, the envelope of response spectra, for the points of attachment, is used.

The procedure for considering differential seismic movement effects on equipment/system with interconnected components, supported at different floors of the same structure, or supported by different structures, is as follows:

a. Relative (differential) displacements between different floors of a structure and between different structures during a seismic event are obtained from the dynamic analysis of the structures, and

b. Maximum relative (differential) displacements are imposed on the equipment/system being analyzed and the induced stresses determined through a static analysis.

The allowable stress criteria are defined in Sections 3.8, 3.9, and 3.10.

3.7.3.10 Use of Constant Vertical Static Factors

The use of constant vertical static factors, as applied to Seismic Category I subsystems, is limited, as discussed in Section 3.7.2.10.

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are discussed in Section 3.7.2.3.2. When the torsional effect of an eccentric mass is likely to have a significant effect on the result of an analysis, the eccentric mass is included in the analytical mode. If the pipe stresses due to an eccentric mass are expected to be insignificant, the offset moment due to the eccentric mass is usually neglected.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

Seismic Category I piping penetrating exterior building foundation walls are furnished with oversized wall sleeves and flexible closure boots.

No buried Seismic Category I tunnels are utilized in this facility.

Underground nuclear safety related piping is designed to safely resist operating loads and loads due to accident conditions which include seismic waves passing through the soil media supporting these elements and relative seismic displacements between building and surrounding soil. Analysis of these underground pipes subject to ground motion is based on their configuration and boundary conditions and the elastic properties of the soil and piping.

For the stress analysis of the portions of the buried pipes penetrating the wall sleeves and connected to the buildings, the relative displacement between the building and soil is imposed on the buried pipe in addition to the pipe internal pressure, pipe dead weight, and seismic and thermal effects on the pipe. When the piping is enclosed in encapsulated sleeves, the supports inside the encapsulated sleeves are also modeled in the analysis of the piping system.

3.7.3.12.1 Procedures for Predicting the Stresses of Buried Pipes in the Free Field

3.7.3.12.1.1 Method of Analysis. The method of analysis developed is based on Reference 3.7-12.

3.7.3.12.1.2 Axial Stresses in Pipe. The method of analysis as suggested by N. M. Newmark in Reference 3.7-12 is used in the analysis of the axial stresses in buried pipes. The value of the maximum particle velocity, Vm, used in the analysis is calculated by following the recommendations of N. M. Newmark et al., in Reference 3.7-2.

3.7.3.12.1.3 Bending Stresses in Pipes. The method of analysis as suggested by N. M. Newmark in Reference 3.7-12 is used in the analysis of the bending stresses in buried pipes.

3.7.3.12.1.4 Buried Piping Encased in Oversized Culvert Sections. Certain portions of buried pipes are encased in oversized culverts. The encasement serves the dual purpose of providing protection against damage of piping under heavily loaded areas such as roads and of accommodating thermal expansion at changes in direction of the piping.

The encased piping does not come in contact with the soil and can thus be analyzed by the same methods used for piping in free space. The dead load, internal pressure, seismic, and thermal stresses are maintained below allowable limits.

3.7.3.12.2 Procedures for Predicting the Stresses of Buried Pipes at Connections to Various Buildings

The relative movement between the soil and buildings is accommodated by encasing the pipe for a sufficient length from the penetration to allow for elastic deformation thereby keeping stress levels below allowable limits. The encased pipe does not come in contact with the soil and can therefore be analyzed using the same methods that are used for piping in free space.

3.7.3.13 Interaction of Other Piping With Seismic Category I Piping

When non-Seismic Category I piping is attached to Seismic Category I piping, that portion of the other piping up to the nearest piping anchor or terminal point is also analyzed as Seismic Category I piping. The non-Seismic Category I piping is designed to withstand the SSE without failing in a manner that would cause the Seismic Category I piping to fail.

3.7.3.14 Seismic Analyses for Reactor Internals

The seismic analysis of the reactor is described in Section 3.7.2.3.4. A comparison of the maximum calculated seismic loads and the allowable seismic loads in the RPV and internals is given in Table 3.7-19. The damping values are given in Table 3.7-1.

3.7.3.15 Analysis Procedure for Damping

Damping values used for Seismic Category I subsystems are discussed in Section 3.7.2.15.

3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison With Regulatory Guide 1.12

The seismic instrumentation system for CGS complies with the requirements of Regulatory Guide 1.12, Revision 2.

3.7.4.2 Location and Description of Instrumentation

Triaxial strong-motion accelerographs are installed at appropriate locations to provide data on the seismic input to the free field area, containment, data on the frequency, amplitude, and phase relationship of the seismic response of the containment structure and to provide data on the seismic input to other Seismic Category I SSCs. The criteria for selection of Seismic

Category I structures, components, and equipment to be instrumented, and the location of instrumentation, is that which will enable the evaluation of the following:

a. To determine if the input design response spectra has been exceeded,

b. To determine if the calculated resultant vibratory responses used in the design of the representative Seismic Category I structures and equipment have been exceeded, and

c. The degree of applicability of the mathematical models used in the seismic analysis of the buildings and equipment.

In compliance with Regulatory Guide 1.12 Rev. 2, the seismic monitoring system at Columbia Generating Station is supported by six (6) strong motion recorders (SMR), one (1) Networked Control Center and a personal computer that provides and interface with the Networked Control Center (NCC).

The SMRs are triaxial time-history accelerographs which can operate on a stand-alone basis or be linked as part of the NCC. The accelerographs' internal sensors pickup vibration and transform into an electrical signal, which is proportional to the acceleration. The signal is compared to the trigger threshold. When the threshold is exceeded, an event is created and the data is recorded which is saved on the recorder that is equipped with a portable media drive. The media device can be removed and the data can be read with a standard personal computer (PC) independently.

The first SMR is located on the 422' elevation, which is the reactor building foundation. The second SMR is located on the 522' elevation of the reactor building. The third SMR is located on the 548' elevation of the reactor building. The SMRs located at the foundation and at elevation in the structures measure the seismic response spectra. This response spectra is used to determine the impact on plant equipment or piping in accordance with Regulatory Guide 1.167 (Ref. 6) prior to restart when the plant is shutdown due to OBE exceedance.

The free field SMR is located 1000 feet NE of the reactor building in a covered pit. The free field SMR data will be used to compare measured response to the design spectra to determine whether the OBE has been exceeded in accordance with Regulatory Guide 1.166.

Two additional SMRs are located in the Radwaste Building. The first SMR is located on the 437' elevation, which is the Radwaste Building foundation. The second SMR is located on the 487' elevation. All SMRs located in the free field, Reactor Building, and Radwaste Building are networked as part of the NCC interface and data gathering.

The NCC is a rack mounted unit that is located in the control room. The NCC connects the SMRs in a centralized location where the NCC acts as a primary control and access point for

the SMR. The NCC monitors the on-line operating status of each SMR on the network and performs common trigger and time synchronization for all connected SMRs. Additionally, when the trigger threshold has been exceeded through voting logic, the relay contacts on the NCC acts as a seismic switch to provide remote indication that a specified acceleration has been exceeded. The remote indication are notification to plant operator that the "Miminum Seismic Earthquake Exceeded." As part of the NCC, there are two power supplies with backup batteries, which provides power to the NCC and the SMRs that are connected to the network. The purpose of the batteries is to supply the NCC and SMRs with power to provide 25 minutes of system operation at any time over a 24-hour period without recharging, in combination with a battery charger in accordance with Regulatory Guide 1.12.

Note that when the NCC is connected to a PC, it performs the functions of an operator interface to interpret the collected data on each SMR, monitoring of on-line operating status of each SMR as well as configuring each of the SMR that is on the network. The computer utilizes proprietary software that is used in conjunction with the NCC to monitor, configure, gather and interpret the data from the SMRs. Interpreting the data would aid in the decision to shut down the plant in accordance with Regulatory Guide 1.166.

3.7.4.3 Control Room Operator Notification

The information, which the system makes available to the control room operator, is as described in Section 3.7.4.2.

The bases for establishing predetermined values for activating the readout of the seismic instrumentation to the control room operator are

a. To initiate the triaxial time-history recorders, at a very low level acceleration equal to 0.01g as recommended by Regulatory Guide 1.12, Rev. 2, Section 6.3, and

b. To provide immediate control room annunciation if the OBE has been exceeded.

3.7.4.4 Comparison of Measured and Predicted Responses

In accordance with Appendix A to 10 CFR 100, an orderly and sequential shutdown of the plant will be carried out according to detailed written station procedures if a seismic event with vibratory ground motion equal to or exceeding that of the OBE occurs. Prior to resuming operations, it will be demonstrated to the NRC that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public, or that the necessary repairs to those features have been completed.

As per Regulatory Guide 1.166, if the response spectrum check and the cumulative absolute velocity (CAV) check were exceeded, the OBE was exceeded and shutdown is required. If

either check does not exceed the criterion, the earthquake motion did not exceed the OBE. If only one check can be performed, the other check is assumed to be exceeded. In the event that data from the free-field instrumentation is unavailable, then the guidance of Regulatory Guide 1.166, Appendix A, will be followed.

The data from the free-field seismic instrumentation, coupled with information obtained from a plant walkdown, are used to make the initial determination for plant actions due to an earthquake. After an earthquake, a response spectrum check and the CAV will be calculated based on the recorded motions at the free-field instrument within 4 hours in accordance with Regulatory Guide 1.166 to determine if OBE was exceeded. If the OBE ground motion is exceeded or significant plant damage is identified during plant walkdowns, action will be take in accordance with Regulatory Guide 1.166 and plant procedures.

The Reactor Building and Radwaste Building foundation-level instrumentation are used for evaluation of the plant structures when the data from the free-field instrumentation is unavailable. In this case, the determination of OBE exceedance is based on a response spectrum check. A comparison is made between the foundation-level design response spectra and data obtained from the foundation-level instruments. If the response spectrum check at either foundation is exceeded, the OBE is exceeded and plant action is required for evaluation of plant operability. The CAV check is not applicable to the data recorded at the foundation-level instrumentation.

3.7.5 REFERENCES

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3.7-14 Whitman, R. V., "Soil-Structure Interaction," Seismic Design of Nuclear Power Plants, R. J. Hansen, editor, the M.I.T. Press, Cambridge, Massachusetts, and London, England, 1970, pp. 245-269.

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Table 3.7-1

Damping Coefficients[a]
(% of critical damping)

Structure or Component[b]	Operating Basis Earthquake	Safe Shutdown Earthquake
Welded steel plate assemblies	1.0	1.0
Welded steel frame structures	2.0	2.0
Bolted or riveted steel frame structures	2.5	2.5
Reinforced-concrete equipment supports	2.0	3.0
Reinforced-concrete structures	3.0	5.0
Vital piping[c]	0.5	1.0
Equipment[c]	1.0	2.0
Welded structural assemblies (equipment and supports)	1.0	2.0
Bolted or riveted structural assemblies	2.0	3.0
Reactor pressure vessel, support skirt, shroud head, separator, and guide tubes	2.0	2.0
Control rod drive housings	3.5	3.5
Fuel	7.0	7.0
Steel frame structures	2.0	3.0
Soil		
Rocking	5	7
Translation (horizontal and vertical)	10	10

[a] The tabulated damping values are used in the seismic analysis in conjunction with the ground response spectra shown in Figure 3.7-1, for the design of all Seismic Category I structures, systems, and components. The original Annulus Pressurization analysis used 3% for all locations except the main steam at RPV nozzle with a 2% damping ratio. The damping ratio for MELLLA is 4% as specified in R.G. 1.61 Rev. 1.

[b] For structures or components, combined stresses are considered below one-half yield for loading combinations including the OBE, and at or near yield for loading combinations including the SSE.

[c] In the event these damping values are found to be too restrictive, the higher damping coefficients cited by Regulatory Guide 1.61 Rev.1 (i.e., 1.0 and 2.0% of critical damping for OBE and SSE events, respectively) may be used.

NOTE: See also Section 3.7.2.15 for the ASME Code Case N-411 damping application to CGS piping systems.

Table 3.7-2

Foundation/Supporting Media for
Seismic Category I Structures

Structure	Average Foundation Embedment Depth (ft)	Width of Structural Foundation (ft)	Total Structural Height (ft)
Reactor building	21.5	147	265
Control room structure and portions of radwaste buildings[a]	15.5	163.5	120
Diesel generator building	4	79.5	40
Standby service water pump houses	12.4	37.5	62
Spray ponds	21	250	16
Turbine generator building[b]	11	192.5	159

[a] See Section 3.8.4.1.2 for a description of the portions of the radwaste and control building designed as Seismic Category I.

[b] The turbine generator building, classified as a modified non-Category I seismic structure, is dynamically analyzed and designed to withstand the effects of an SSE and maintain its structural integrity.

Table 3.7-3

Reactor Building - Seismic Analysis
Natural Frequency and Natural Period

Direction	Mode	Natural Frequency (cps)	Natural Period (sec/cycle)
Horizontal (N-S)	1	1.92	0.519
	2	3.42	0.292
	3	5.17	0.193
	4	5.88	0.170
	5	7.52	0.133
	6	10.37	0.096
	7	11.41	0.087
	8	12.54	0.079
	9	16.82	0.059
	10	18.80	0.053
Vertical	1	4.14	0.241
	2	10.27	0.097
	3	12.92	0.077
	4	19.29	0.051
	5	20.31	0.049
	6	23.37	0.042
	7	33.01	0.030
	8	37.89	0.026
	9	46.03	0.021
	10	49.22	0.020

Table 3.7-4

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Acceleration (units in g x 10^{-3})

	Mode Numbers										
Mass	1	2	3	4	5	6	7	8	9	10	Combined
1	681.65	770.36	147.14	10.00	0.58	1.13	0.61	1.74	0.02	0.05	1039.16
2	381.23	31.44	125.22	20.83	0.27	8.00	3.57	7.52	0.04	1.17	403.27
3	311.30	33.80	48.44	17.61	1.47	2.66	0.82	9.37	0.28	0.80	317.52
4	271.37	33.00	7.52	19.32	1.55	0.72	2.53	13.08	0.05	0.52	274.60
5	221.50	30.71	40.51	20.00	1.67	3.67	3.50	12.55	0.25	1.65	228.73
6	183.68	28.22	73.45	19.74	1.83	4.47	3.37	9.64	0.34	1.69	201.27
7	129.38	23.39	113.05	17.94	1.75	4.11	1.77	0.10	0.34	0.82	174.39
8	85.95	18.88	138.24	16.06	1.90	1.99	0.02	6.50	0.17	0.11	164.81
9	72.55	17.31	143.54	16.33	2.32	1.07	0.36	8.04	0.16	0.32	162.82
10	47.00	14.54	154.99	16.79	3.42	1.28	0.77	7.82	0.26	0.36	163.74
11	341.16	42.95	28.76	93.74	18.49	2.62	4.47	1.25	3.50	0.70	358.11
12	313.88	40.95	0.74	90.19	11.74	1.76	1.99	2.85	2.88	0.37	329.40
13	292.73	40.55	31.89	97.67	0.48	0.58	1.92	6.33	3.41	0.16	313.00
14	260.70	39.43	75.68	104.35	15.19	3.20	6.65	10.40	3.87	0.11	294.42
15	241.40	38.53	99.51	106.24	22.43	4.37	8.90	12.19	3.91	0.25	286.24
16	224.77	37.64	118.86	106.90	27.87	5.17	10.56	13.45	3.86	0.35	280.87
17	208.53	36.66	136.32	106.43	32.23	5.71	11.84	14.32	3.71	0.43	276.60
18	186.76	35.14	157.25	103.90	36.49	6.04	12.94	14.93	3.32	0.50	271.64
19	173.59	33.14	156.95	88.30	32.21	5.92	11.62	11.87	3.02	0.44	255.53
20	149.95	29.38	154.29	58.35	23.39	5.54	8.83	5.97	2.39	0.30	226.63
21	127.28	25.76	151.35	30.11	14.54	4.90	5.98	0.42	1.73	0.15	202.37
22	105.96	22.43	149.21	5.25	6.29	4.01	3.34	4.29	1.11	0.02	184.76
23	85.96	18.89	138.31	16.02	1.89	1.99	0.01	6.49	0.17	0.11	164.87
24	335.59	54.35	52.65	306.06	67.38	6.67	14.08	22.48	5.22	0.64	466.87
25	315.53	53.71	96.44	303.01	18.08	11.29	18.00	23.07	0.51	0.04	453.55
26	280.67	51.17	151.19	278.75	38.20	10.97	11.67	8.09	8.49	1.05	428.95
27	256.78	47.92	167.09	240.87	52.47	5.69	1.24	6.58	8.70	0.92	396.35
28	229.46	43.19	168.96	186.09	50.12	2.14	11.20	20.54	4.75	0.27	348.20
29	214.61	40.41	166.36	158.04	46.70	3.63	12.24	19.28	1.84	0.03	321.78
30	186.92	35.25	158.31	105.27	37.09	6.05	13.10	15.13	3.32	0.51	273.03
31	167.90	32.05	155.85	76.90	31.52	6.80	12.65	11.50	5.40	0.64	247.13
32	140.09	27.37	150.40	39.33	21.61	6.73	10.20	4.74	6.26	0.55	212.88
33	114.61	23.24	145.32	10.61	11.72	5.55	6.73	1.69	5.07	0.25	187.53
34	91.85	19.81	142.38	8.37	3.22	3.53	3.11	6.45	2.56	0.10	171.15
35	72.56	17.32	143.55	16.33	2.31	1.07	0.37	8.04	0.16	0.32	162.84

Table 3.7-5

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Displacement (units in ft x 10^{-4})

	Mode Numbers										
Mass	1	2	3	4	5	6	7	8	9	10	Combined
1	1502.05	-536.45	44.83	-2.36	0.08	-0.09	-0.04	-0.09	0.0	0.0	1595.60
2	840.07	21.90	-38.15	4.91	-0.04	0.61	0.22	0.39	0.0	0.03	841.24
3	685.97	23.54	-14.76	4.15	0.21	0.20	-0.05	-0.49	-0.01	0.02	686.54
4	597.97	22.98	-2.29	4.56	0.22	-0.05	-0.16	-0.68	0.0	-0.01	598.44
5	488.09	21.39	12.34	4.72	0.24	-0.28	-0.22	-0.65	0.01	-0.04	488.74
6	404.75	19.65	22.38	4.65	0.26	-0.34	-0.21	-0.50	0.01	-0.04	405.87
7	285.11	16.29	34.44	4.23	0.25	-0.31	-0.11	-0.01	0.01	-0.02	287.68
8	189.39	13.15	42.12	3.79	0.27	-0.15	0.0	0.34	0.0	0.0	194.50
9	159.88	12.06	43.73	3.85	0.34	-0.08	0.02	0.42	0.0	0.01	166.24
10	103.58	10.12	47.22	3.96	0.49	0.10	0.05	0.41	0.01	0.01	114.36
11	751.76	29.91	-8.76	-22.11	2.67	0.20	-0.28	-0.06	-0.01	0.02	752.74
12	691.65	28.51	-0.22	-21.27	1.70	0.13	-0.12	-0.15	-0.08	0.01	692.57
13	645.04	28.24	9.72	-23.03	-0.07	-0.04	0.12	-0.33	-0.10	0.0	646.14
14	574.46	27.46	23.06	-24.61	-2.19	-0.24	0.42	-0.54	-0.11	0.0	576.11
15	531.94	26.83	30.32	-25.05	-3.24	-0.33	0.56	-0.63	-0.11	-0.01	534.08
16	495.28	26.21	36.22	-25.21	-4.02	-0.39	0.66	-0.70	-0.11	-0.01	497.95
17	459.51	25.53	41.54	-25.10	-4.65	-0.43	0.74	-0.74	-0.11	-0.01	462.80
18	411.53	24.47	47.91	-24.50	-5.27	-0.46	0.81	-0.77	-0.10	-0.01	415.79
19	382.52	23.08	47.82	-20.82	-4.65	-0.45	0.73	-0.62	-0.09	-0.01	386.78
20	330.42	20.46	47.01	-13.76	-3.38	-0.42	0.55	-0.31	-0.07	-0.01	334.68
21	280.47	17.94	46.11	-7.10	-2.10	-0.37	0.37	-0.02	-0.05	0.0	284.90
22	233.48	15.62	45.46	-1.24	-0.91	-0.30	0.21	0.22	-0.03	0.0	238.38
23	189.42	13.15	42.14	3.78	0.27	-0.15	0.0	0.34	0.0	0.0	194.53
24	739.49	37.84	16.04	-72.17	9.73	0.15	-0.88	1.17	-0.15	-0.01	744.21
25	695.29	37.40	29.38	-71.45	2.61	0.86	-1.13	1.20	-0.01	0.0	700.58
26	618.47	35.64	46.06	-65.73	-5.52	0.83	-0.73	0.42	0.24	0.02	624.70
27	565.83	33.37	50.91	-56.80	-7.58	0.43	-0.08	-0.34	0.25	0.02	571.97
28	505.62	30.08	51.48	-43.88	-7.24	-0.16	0.70	-1.07	0.14	0.01	511.07
29	472.90	28.18	50.69	-37.27	-6.74	-0.28	0.76	-1.00	0.05	0.0	477.95
30	411.88	24.55	48.23	-24.82	-5.36	-0.46	0.82	-0.79	-0.10	-0.01	416.20
31	369.98	22.32	47.48	-18.13	-4.55	-0.52	0.79	-0.60	-016	-0.01	374.15
32	308.70	19.06	45.83	-9.27	-3.12	-0.51	0.64	-0.25	-0.18	-0.01	312.82
33	252.55	16.18	44.28	-2.50	-1.69	-0.42	0.42	0.09	-0.15	-0.01	256.93
34	202.40	13.80	43.38	1.97	-0.46	-0.27	0.19	0.33	-0.07	0.0	207.47
35	159.89	12.06	43.74	3.85	0.33	-0.08	0.02	0.42	0.00	0.01	166.25

Table 3.7-6

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Member Shears (units in kips)

Member	Mode Number 1	2	3	4	5	6	7	8	9	10	Combined
1	1778.0	-2009.4	383.8	-26.1	-1.6	-2.9	-1.6	-4.5	0.0	0.1	2710.5
2	11509.0	-1207.3	-2810.3	505.3	-5.3	201.2	89.4	187.1	1.0	30.0	11924.0
3	20658.0	27.1	-3380.6	-400.9	117.5	272.8	62.0	-60.0	-11.2	52.1	20939.0
4	25430.0	607.2	-3512.9	-61.1	144.7	260.3	17.6	-290.1	-12.0	43.0	25682.0
5	29256.0	1137.6	-2814.5	284.1	173.5	196.9	-42.9	-506.6	-7.8	14.6	29421.0
6	32468.0	1631.7	-1527.9	628.9	205.5	118.6	-102.0	-675.6	-1.8	-15.0	32461.0
7	35151.0	2116.5	821.4	1002.5	241.7	33.2	-138.8	-673.5	5.2	-32.0	35248.0
8	39200.0	2760.0	4264.8	126.2	15.3	-9.3	-88.7	-622.9	-1.0	-31.5	39535.0
9	42109.0	3325.2	8593.9	419.2	-52.9	-58.7	-39.8	-441.4	-17.4	-25.0	43110.0
10	45884.0	4484.0	20920.0	1753.8	218.5	43.1	21.4	179.6	3.3	3.7	50658.0
11	72.0	9.0	-6.0	-19.6	3.9	0.5	-0.9	-0.3	-0.7	0.1	75.5
12	915.4	-42.7	-429.3	183.7	110.9	13.3	-14.9	11.4	1.5	0.2	1034.9
13	935.0	-39.8	-426.9	176.7	110.9	13.2	-14.7	11.0	1.2	0.2	1049.9
14	987.4	-32.2	-412.6	156.9	108.0	12.6	-13.5	9.0	0.5	0.2	1087.7
15	1003.5	-29.3	-405.0	148.7	106.3	12.3	-12.8	8.1	0.2	0.2	1098.1
16	1031.5	-24.7	-390.3	135.3	102.8	11.6	-11.5	6.4	-0.3	0.1	1116.4
17	1081.3	-16.0	-358.0	110.5	95.2	10.3	-8.7	3.0	-1.2	0.0	1148.6
18	2357.1	264.8	796.3	-1041.1	-211.3	6.1	30.4	-55.0	-2.1	-0.7	2719.6
19	2380.2	269.2	817.5	-1053.1	-215.8	5.3	32.0	-56.6	-2.6	-0.8	2751.3
20	2405.4	274.5	845.6	-1063.7	-220.1	4.2	33.6	-57.7	-3.0	-0.8	2786.6
21	2428.1	279.0	872.7	-1069.2	-222.7	3.4	34.7	-57.8	-3.3	-0.9	2817.2
22	2925.0	384.6	1574.3	-1093.9	-252.3	-15.5	50.4	-37.6	-8.5	-1.0	3528.4
23	241.7	-76.2	-242.9	390.3	-11.7	8.7	-19.1	28.1	-12.1	-1.1	527.4
24	56.8	-44.7	-186.4	212.8	-1.1	15.3	-29.7	41.6	-12.4	-1.2	301.2
25	236.4	-8.9	-28.2	-79.0	-41.1	26.8	-41.9	50.1	-3.5	-0.1	271.8
26	374.8	34.4	60.7	-207.0	-69.0	29.9	-42.5	46.6	1.1	0.4	449.2
27	817.9	120.1	395.4	-575.8	-168.3	25.6	-20.4	5.9	10.5	1.0	1095.9
28	1033.6	157.7	550.6	-723.2	-211.9	22.2	-8.9	-12.1	12.3	1.0	1401.9
29	493.9	17.7	28.4	8.3	-53.1	2.3	4.2	-14.4	0.0	-0.3	498.3
30	718.0	59.1	228.5	-90.4	-93.7	-6.5	20.5	-29.2	-7.0	-1.1	768.6
31	833.8	81.8	354.2	-123.3	-111.7	-12.1	29.0	-33.2	-12.2	-1.6	926.9
32	940.1	103.3	488.4	-133.1	-122.5	-17.2	35.2	-31.6	-16.8	-1.8	1082.0
33	1075.2	132.0	693.5	-121.1	-127.1	-22.3	39.7	-22.3	-20.5	-1.7	1300.0
34	314.3	-275.1	-803.8	1405.7	-82.4	3.8	4.1	-18.7	4.1	0.6	1674.4
35	-1119.5	-218.3	-380.7	1191.1	-188.0	-8.7	17.7	-30.8	1.6	0.5	1703.6
36	-3776.0	-415.0	1859.0	1043.3	247.6	22.8	-50.5	15.5	8.0	0.7	4362.9
37	-995.0	-151.0	-450.0	119.2	127.0	23.2	-40.4	12.0	20.4	1.3	1117.7
38	-1225.0	-275.8	-1122.2	1132.0	300.4	2.9	-36.5	54.8	0.3	0.7	2053.3

Table 3.7-7

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Member Moments (units in ft-kips x 10^3)

	Mode Number							
	1		2		3		4	
Member	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
1	8.51	-92.52	-0.03	94.97	-9.02	-9.11	-0.1	1.25
2	146.58	-589.63	-95.16	141.64	-46.11	154.31	-1.32	-18.13
3	635.28	-1056.16	-140.82	140.27	-200.22	269.10	17.85	-9.68
4	1100.61	-1761.78	-138.92	123.13	-312.27	403.60	9.25	-7.66
5	1806.29	-2420.64	-121.11	97.22	-445.67	504.77	7.08	-13.05
6	2468.16	-3442.18	-94.44	45.49	-547.79	593.62	12.29	-31.19
7	3493.30	-4442.34	-41.67	-15.48	-638.79	616.56	29.85	-56.92
8	4690.34	-5032.50	33.19	-57.29	-615.24	578.02	3.70	-4.80
9	5157.28	-6000.32	68.62	-135.20	-583.26	411.20	-20.21	11.82
10	6382.04	6382.04	166.90	-166.90	-727.50	727.50	-27.03	27.03
11	0.03	-1.16	0.0	-0.14	-0.05	0.14	0.0	0.31
12	1.18	-10.19	0.15	0.27	-0.19	4.42	-0.31	-1.50
13	10.33	-25.52	-0.27	0.92	-4.46	11.39	1.49	-4.36
14	25.44	-35.56	-0.91	1.24	-11.52	15.75	4.35	-5.96
15	35.77	-44.90	-1.23	1.50	-15.80	19.48	5.94	-7.30
16	45.20	-54.66	-1.49	1.71	-19.57	23.14	7.27	-8.51
17	54.67	-68.45	-1.70	1.90	-23.35	27.91	8.45	-9.86
18	68.62	-85.18	-1.88	0.02	-28.12	22.52	9.78	-2.46
19	85.49	-115.57	0.0	-3.40	-22.62	12.28	2.40	10.91
20	115.86	-146.32	3.43	-6.90	-12.43	1.73	-10.99	24.46
21	146.30	-177.07	6.92	-10.46	-1.86	-9.20	-24.50	38.05
22	182.61	-217.71	10.93	-15.54	5.31	-24.20	-39.03	52.15
23	2.48	-0.09	0.24	0.51	-4.01	6.41	-3.03	-0.82
24	0.21	0.71	-0.50	1.22	-6.58	9.61	0.67	-4.13
25	-0.48	-1.94	-1.19	1.10	-9.81	10.10	3.82	-3.01
26	2.06	-6.36	-1.08	0.69	-10.14	9.45	2.83	-0.46
27	7.88	-13.45	-0.44	-0.38	-9.40	6.71	-2.03	5.95
28	13.52	-26.70	0.40	-2.41	-6.70	-0.32	-6.22	15.44
29	27.31	-31.90	2.51	-2.68	0.39	-0.66	-16.41	16.33
30	32.13	-42.32	2.71	-3.55	0.68	-3.92	-16.63	17.91
31	42.42	-54.16	3.57	-4.72	3.94	-8.93	-18.06	19.80
32	54.31	-67.56	4.74	-6.20	8.93	-15.81	-19.91	21.79
33	67.70	-82.82	6.22	-8.07	15.77	-25.52	-21.88	23.58
34	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
35	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
36	-214.67	214.67	-15.63	15.63	-19.68	19.68	52.36	52.36
37	-62.40	62.40	-7.84	7.84	-25.49	25.49	23.63	23.63
38	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 3.7-7

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Member Moments (units in ft-kips x 10^3) (Continued)

	Mode Number							
	5		6		7		8	
Member	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
1	-0.26	0.18	0.23	-0.09	0.43	-0.35	2.42	-2.21
2	-1.65	1.85	1.26	-9.00	2.40	-5.84	12.95	-20.15
3	-3.05	0.66	9.62	-15.18	7.23	-8.50	27.62	-26.40
4	-1.81	-1.96	15.34	-22.11	9.52	-9.98	32.32	-24.77
5	0.73	-4.38	21.48	-25.62	10.57	-9.67	29.30	-18.66
6	2.95	-9.12	24.05	27.61	9.89	-6.83	22.46	-2.20
7	7.18	-13.71	24.23	-25.12	6.52	-2.78	5.87	12.31
8	6.53	-6.66	23.10	-23.02	3.69	-2.91	-12.32	17.76
9	-0.67	1.72	19.21	-18.04	3.62	-2.82	-15.92	24.76
10	-20.57	20.57	-24.29	24.29	-2.54	2.54	15.97	-15.97
11	0.01	-0.07	0.0	-0.01	0.0	0.02	0.0	0.0
12	0.08	-1.18	0.01	-0.41	-0.02	0.17	0.0	-0.11
13	1.18	-2.98	0.14	-0.36	-0.17	0.41	0.12	-0.29
14	3.00	-4.11	0.36	-0.49	-0.42	0.56	0.30	-0.39
15	4.12	-5.09	0.49	-0.60	-0.56	0.67	0.40	-0.47
16	5.09	-6.04	0.60	-0.70	-0.68	0.78	0.47	-0.53
17	6.05	-7.26	0.70	-0.83	-0.79	0.90	0.53	-0.57
18	7.26	-5.78	0.82	-0.87	-0.90	0.69	0.57	-0.19
19	5.77	-3.04	0.8	-0.93	-0.69	0.28	0.19	0.53
20	3.03	-0.24	0.91	-0.97	-0.28	-0.14	-0.53	1.26
21	0.23	2.59	0.96	-1.00	0.15	-0.59	-1.26	1.99
22	-2.93	5.96	0.59	-0.41	0.59	-1.20	1.75	2.20
23	6.45	-6.33	-0.88	0.79	1.20	-1.01	-1.25	0.98
24	6.59	-6.57	-0.82	0.57	1.05	-0.57	-1.00	0.33
25	6.82	-6.40	-0.58	0.30	0.57	-0.14	-0.32	-0.19
26	6.46	-5.67	-0.30	-0.04	0.13	0.36	0.19	-0.73
27	5.53	-4.38	0.15	-0.33	-0.43	0.57	0.60	-0.64
28	4.36	-1.66	0.34	-0.62	-0.57	0.69	0.62	-0.47
29	1.51	-1.02	0.65	-0.67	-0.69	0.65	0.38	-0.24
30	0.96	0.37	0.67	-0.58	-0.64	0.35	0.21	0.21
31	-0.40	1.97	0.58	-0.41	-0.40	-0.07	-0.23	0.70
32	-2.01	3.74	0.41	-0.16	0.08	-0.58	-0.72	1.16
33	-3.77	5.56	0.15	0.16	0.59	-1.15	-1.17	1.49
34	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
35	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
36	6.11	-6.11	-0.09	0.09	-1.17	1.17	1.19	-1.90
37	5.57	-5.57	0.18	-0.18	-1.14	1.14	1.47	-1.47
38	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 3.7-7

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Member Moments (units in ft-kips x 10^3) (Continued)

	Mode number					
	9		10		Combined	
Member	Top	Bottom	Top	Bottom	Top	Bottom
1	0.10	-0.10	-0.96	0.95	12.73	132.93
2	0.38	-0.42	-2.77	1.61	181.40	626.60
3	0.58	-0.35	-2.64	1.58	682.01	1099.59
4	0.47	-0.15	-2.34	1.22	1153.35	1812.09
5	0.24	-0.08	-1.79	1.48	1864.96	2474.95
6	0.17	-0.11	-1.91	2.36	2530.33	3493.56
7	0.20	-0.34	-2.50	3.36	3551.69	4485.42
8	0.12	-0.11	-3.27	3.54	4730.71	5066.00
9	-0.17	0.52	-3.34	3.84	5190.72	6016.01
10	0.14	-0.14	0.81	-0.81	6425.70	6425.70
11	0.0	0.01	0.0	0.0	0.06	1.22
12	-0.01	0.0	0.0	0.0	1.24	11.28
13	0.0	-0.02	0.01	-0.01	11.41	28.47
14	0.02	-0.03	0.01	-0.01	28.45	39.60
15	0.03	-0.03	0.01	-0.01	39.80	49.78
16	0.03	-0.02	0.01	-0.01	50.09	60.31
17	0.02	-0.01	0.02	-0.02	60.39	74.98
18	0.01	0.01	0.02	-0.01	75.19	88.34
19	-0.01	0.0	0.01	0.0	88.66	116.83
20	-0.05	0.08	0.0	0.01	117.14	148.53
21	-0.08	0.13	-0.01	0.02	148.52	181.69
22	-0.15	0.25	0.0	0.01	187.16	225.81
23	0.19	-0.07	0.02	-0.01	8.94	9.31
24	0.08	0.12	0.01	0.01	9.61	12.48
25	-0.12	0.15	-0.01	0.01	12.65	12.54
26	-0.14	0.13	-0.01	0.01	12.58	12.80
27	0.12	-0.19	0.02	-0.03	13.66	16.81
28	0.22	-0.37	0.03	-0.04	16.95	31.02
29	0.45	-0.45	0.05	-0.04	32.02	35.98
30	0.47	-0.37	0.04	-0.03	36.32	46.27
31	0.38	-0.21	0.03	-0.01	46.42	59.59
32	0.20	0.03	0.01	0.02	58.76	73.10
33	-0.04	0.33	-0.02	0.04	73.26	90.39
34	0.0	0.0	0.0	0.0	0.0	0.0
35	0.0	0.0	0.0	0.0	0.0	0.0
36	0.25	-0.25	0.0	0.0	225.50	225.50
37	0.33	-0.33	0.04	-0.04	72.12	72.12
38	0.0	0.0	0.0	0.0	0.0	0.0

Table 3.7-8

Reactor Building - Seismic Analysis
Horizontal N-S Direction - SSE
Acceleration (units in g x 10^{-3})

	Mode Number										
Mass	1	2	3	4	5	6	7	8	9	10	Combine
1	1171.05	-1177.02	276.94	-15.31	0.87	-1.74	-0.92	-2.67	-0.03	0.07	1683.35
2	654.94	48.04	-235.69	31.90	-0.40	12.34	5.36	11.52	0.06	1.81	698.76
3	534.80	51.65	-91.18	26.96	2.15	4.10	-1.23	-14.37	-0.43	1.24	545.88
4	466.20	50.43	-14.14	29.59	2.33	-1.10	-3.80	-20.06	-0.07	-0.80	470.67
5	380.53	46.92	76.24	30.63	2.51	-5.66	-5.26	-19.23	0.37	-2.55	392.93
6	315.55	43.12	138.21	30.23	2.76	-6.89	-5.07	-14.78	0.52	-2.61	349.14
7	222.28	35.73	212.77	27.47	2.63	-6.34	-2.67	0.15	0.51	-1.26	311.07
8	147.65	28.85	260.19	24.60	2.87	-3.06	-0.03	9.96	0.25	0.16	301.76
9	124.64	26.45	270.16	25.01	3.51	-1.65	0.54	12.32	0.24	0.50	300.05
10	80.75	22.21	291.72	25.71	5.15	1.97	1.16	11.98	0.39	0.56	304.92
11	586.10	65.63	-54.12	-143.56	27.90	4.05	-6.72	1.91	-5.25	1.08	610.12
12	539.23	62.56	-1.39	-138.12	17.72	2.71	-2.99	-4.37	-4.32	0.57	560.50
13	502.89	61.95	60.02	-149.57	-0.73	-0.90	2.89	-9.70	-5.11	0.25	531.88
14	447.87	60.25	142.44	-159.81	-22.93	-4.93	9.99	-15.94	-5.81	-0.17	501.30
15	414.72	58.86	187.30	-162.70	-33.85	-6.73	13.38	-18.69	-5.87	-0.38	489.14
16	386.14	57.51	223.71	-163.71	-42.05	-7.97	15.87	-20.61	-5.79	-0.54	482.14
17	358.25	56.01	256.57	-162.99	-48.64	-8.81	17.79	-21.95	-5.57	-0.67	477.42
18	320.84	53.69	295.97	-159.12	-55.07	-9.31	19.44	-22.74	-4.98	-0.77	472.93
19	298.22	50.63	295.40	-135.23	-48.61	-9.13	17.46	-18.19	-4.53	-0.67	448.09
20	257.60	44.89	290.39	-89.37	-35.30	-8.55	13.26	-9.15	-3.59	-0.46	403.14
21	218.66	39.36	284.85	-46.11	-21.94	-7.56	8.99	-0.65	-2.60	-0.23	365.06
22	182.03	34.27	280.84	-8.04	-9.49	-6.18	5.01	6.57	-1.66	-0.04	336.91
23	147.68	28.86	260.31	24.53	2.85	-3.07	-0.02	9.95	0.25	0.16	301.86
24	576.53	83.03	99.10	-468.71	101.66	10.28	-21.16	34.47	-7.83	-1.00	763.14
25	542.07	82.06	181.51	-464.05	27.28	17.41	-27.05	35.37	-0.76	-0.07	744.18
26	482.18	78.19	284.55	-426.89	-57.65	16.91	-17.53	12.40	12.74	1.62	711.68
27	441.14	73.22	314.49	-368.87	-79.17	8.77	-1.87	-10.08	13.06	1.42	664.52
28	394.20	65.99	318.01	-284.98	-75.63	-3.30	16.83	-31.49	7.13	0.42	591.78
29	368.69	61.84	313.11	-242.03	-70.47	-5.60	18.40	-29.56	2.76	-0.04	551.07
30	321.12	53.86	297.95	-161.22	-55.96	-9.33	19.69	-23.20	-4.98	-0.78	475.25
31	288.45	48.96	293.32	-117.77	-47.57	-10.49	19.01	-17.62	-8.10	-0.99	435.07
32	240.68	41.82	283.08	-60.23	-32.60	-10.38	15.33	-7.27	-9.40	-0.85	381.06
33	196.89	35.51	273.50	-16.25	-17.68	-8.55	10.11	2.60	-7.60	-0.39	340.15
34	157.80	30.27	267.98	12.82	-4.86	-5.44	4.67	9.88	-3.85	0.15	313.17
35	124.65	26.46	270.19	25.01	-3.49	-1.66	0.55	12.32	0.23	0.50	300.07

Table 3.7-9

Reactor Building - Seismic Analysis
Horizontal N-S Direction - SSE
Displacement (units in ft x 10^{-4})

	Mode Number										
Mass	1	2	3	4	5	6	7	8	9	10	Combined
1	2580.46	-819.63	84.38	-3.16	0.13	-0.13	-0.06	-0.14	0.0	0.0	2708.4
2	1443.20	33.46	-71.81	7.52	-0.06	0.94	0.36	0.60	0.0	0.04	1445.2
3	1178.46	35.97	-27.78	6.36	0.31	0.31	-0.08	-0.75	-0.01	0.03	1178.9
4	1027.29	35.11	-4.31	6.98	0.34	-0.08	-0.24	-1.04	0.0	-0.02	1027.6
5	838.50	32.67	23.23	7.22	0.36	-0.43	-0.33	-1.00	0.01	-0.06	839.5
6	695.34	30.03	42.11	7.13	0.40	-0.52	-0.32	-0.77	0.01	-0.06	697.3
7	489.79	24.88	64.83	6.48	0.38	-0.48	-0.17	0.01	0.01	-0.03	494.7
8	325.35	20.09	79.28	5.80	0.41	-0.23	0.0	0.52	0.01	0.0	335.6
9	274.66	18.42	82.31	5.90	0.50	-0.13	0.03	0.64	0.01	0.01	287.4
10	177.94	15.47	88.88	6.06	0.74	0.15	0.07	0.62	0.01	0.01	199.6
11	1291.49	45.70	-16.49	-33.85	4.03	0.31	-0.42	-0.10	-0.15	0.02	1292.4
12	1188.23	43.57	-0.42	-32.57	2.56	0.21	-0.19	-0.23	-0.12	0.01	1189.2
13	1108.15	43.14	18.29	-35.27	-0.11	-0.07	0.18	-0.50	-0.15	0.01	1109.6
14	986.90	41.95	43.40	-37.68	-3.11	-0.37	0.63	-0.83	-0.17	0.0	989.5
15	913.86	40.99	57.07	-38.36	-4.89	-0.51	0.84	-0.97	-0.17	-0.01	917.4
16	850.87	40.05	68.16	38.60	-6.07	-0.60	0.99	-1.07	-0.17	-0.01	855.4
17	789.41	39.01	78.17	-38.43	-7.02	-0.67	1.11	-1.14	-0.16	-0.02	795.2
18	706.99	37.39	90.18	-37.52	-7.95	-0.71	1.22	1.18	-0.14	-0.02	714.7
19	657.15	35.26	90.00	-31.89	-7.02	-0.69	1.09	-0.94	-0.13	-0.02	665.1
20	567.64	31.26	88.48	-21.07	-5.10	-0.65	0.83	-0.47	-0.10	-0.01	575.7
21	481.83	27.41	86.79	-10.87	-3.17	-0.57	0.56	-0.03	-0.07	-0.01	490.5
22	401.11	23.87	85.57	-1.90	-1.37	-0.47	0.31	0.34	-0.05	0.0	410.8
23	325.42	20.10	79.31	5.79	0.41	-0.23	0.0	0.52	0.01	0.0	335.6
24	1270.42	57.82	30.19	-110.52	14.68	0.78	-1.33	1.79	-0.23	-0.02	1276.6
25	1194.48	57.14	55.30	-109.42	3.94	1.32	-1.70	1.84	-0.02	0.0	1201.6
26	1062.51	54.45	86.70	-100.66	-8.32	1.28	-1.10	0.64	0.37	0.04	1072.7
27	972.08	50.98	95.82	-86.98	-11.43	0.67	-0.12	-0.52	0.38	0.03	982.1
28	868.64	45.95	96.89	-67.20	-10.92	-0.25	1.06	-1.63	0.21	0.01	887.8
29	812.43	43.06	95.40	-57.07	-10.18	-0.42	1.15	-1.53	0.08	0.0	821.2
30	707.59	37.51	90.78	-38.02	-8.08	-0.71	1.23	-1.20	-0.14	-0.02	715.4
31	635.61	34.10	89.37	-27.77	-6.87	-0.80	1.19	-0.91	-0.23	-0.02	643.4
32	530.34	29.12	86.25	-14.20	-4.71	-0.79	0.96	-0.38	-0.27	-0.02	538.3
33	433.86	24.73	88.33	-3.83	-2.55	-0.65	0.63	0.13	-0.22	-0.01	442.5
34	347.72	21.08	81.65	3.02	-0.70	-0.41	0.29	0.51	-0.11	0.0	357.8
35	274.68	18.42	82.32	5.90	0.50	-0.13	0.03	0.64	0.01	0.01	287.4

Table 3.7-10

Reactor Building - Seismic Analysis
Horizontal N-S Direction - SSE
Member Shears (units in kips)

	Mode Number										
Member	1	2	3	4	5	6	7	8	9	10	Combined
1	3054.5	-3070.1	722.4	-39.9	2.3	-4.5	-2.4	-7.0	-0.1	0.2	4390.8
2	19772.	-1844.7	-5289.3	773.8	-7.9	310.1	134.3	286.9	1.4	46.4	20571.
3	35489.	41.4	-6362.8	-613.9	177.3	420.6	93.1	-92.0	-16.8	80.7	36064.
4	43688.	927.8	-6611.8	-93.6	218.3	401.2	26.4	-444.7	-18.1	66.6	44200.
5	50259.	1738.1	-5297.2	435.1	261.8	303.5	-64.4	-776.7	-11.7	22.5	50578.
6	55778.	2493.1	-2875.7	964.7	310.1	182.8	-153.2	-1035.7	-2.7	-23.2	55930.
7	60387.	3233.8	1545.9	1535.2	364.7	51.1	-208.6	-1032.6	7.9	-49.4	60527.
8	67346.	4217.0	8027.3	193.3	23.1	-14.3	-133.3	-954.9	-1.5	-48.7	67962.
9	72339.	5080.6	16175.	642.0	-79.8	-90.5	-59.8	-676.8	-26.1	-38.6	74305.
10	78827.	6851.0	39375.	2685.8	329.7	66.4	32.1	275.4	5.0	5.7	88422.
11	123.6	13.8	-11.3	-30.1	5.8	0.8	-1.4	-0.4	-1.1	0.2	128.6
12	1572.5	-65.2	-808.0	281.3	167.4	20.4	-22.3	17.5	2.2	0.3	1799.7
13	1606.2	-60.8	-803.6	270.6	167.3	20.4	-22.1	16.8	1.8	0.3	1825.5
14	1696.3	-49.2	-776.6	240.3	163.0	19.4	-20.2	13.8	0.7	0.3	1889.1
15	1724.0	-44.8	-762.2	227.7	160.4	18.9	-19.2	12.4	0.3	0.3	1906.3
16	1772.1	-37.7	-734.7	207.2	155.2	17.9	-17.2	9.8	-0.4	0.2	1936.4
17	1857.7	-24.4	-673.8	169.3	143.6	15.9	-13.0	4.6	-1.8	0.0	1988.8
18	4049.4	404.6	1498.8	-1594.4	-318.9	9.3	45.7	-84.4	-3.2	-1.1	4633.4
19	4089.0	411.3	1538.7	-1612.8	-325.6	8.1	48.1	-86.8	-3.8	-1.2	4688.5
20	4132.4	419.4	1591.6	-1620.1	-322.1	6.5	50.5	-88.5	-4.5	-1.3	4750.7
21	4171.5	426.2	1642.6	-1637.4	-336.0	5.2	52.1	-88.6	-5.0	-1.4	4805.7
22	5024.9	587.6	2962.9	-1675.2	-380.6	-23.9	75.7	-57.7	-12.8	-1.5	6111.0
23	-415.2	-116.5	-457.2	597.7	-17.6	13.5	-28.7	43.0	-18.1	-1.8	870.7
24	-97.5	-68.2	-350.8	325.8	-1.6	23.7	-44.6	63.8	-18.6	-1.8	506.0
25	406.2	13.6	-53.1	-121.0	-62.0	41.4	-62.9	76.8	-5.2	-0.1	455.7
26	643.9	52.5	114.3	-317.0	-104.1	46.0	-63.9	71.4	1.7	0.7	749.8
27	1404.6	183.5	744.2	-881.9	-253.9	39.5	-30.6	9.1	15.8	1.5	1845.5
28	1775.6	241.0	1036.4	-1107.5	-319.7	34.3	-13.4	-18.6	18.4	1.4	2370.0
29	848.5	27.1	53.5	12.7	-80.2	3.5	6.4	-22.1	0.0	-0.5	855.0
30	1233.3	90.2	430.0	-138.5	-141.4	-10.0	30.8	-44.8	-10.4	-1.8	1326.3
31	1432.4	125.1	666.6	-188.9	-168.5	-18.6	43.6	-50.8	-18.3	-2.5	1607.9
32	1615.0	157.9	919.1	-203.8	-184.8	-26.5	52.9	-48.4	-25.3	-2.8	1888.2
33	1847.1	201.7	1305.3	-185.5	-191.9	-34.3	59.7	-34.2	-30.8	-2.7	2288.7
34	-539.8	-420.3	-1512.9	2152.7	-124.3	5.8	6.1	-28.7	6.2	0.9	2721.7
35	-1923.2	-333.6	-716.4	1824.1	-283.7	-13.4	26.6	-47.1	2.4	0.8	2781.5
36	-6464.0	-635.0	-3498.0	1593.3	373.6	35.1	-75.9	23.8	12.0	1.0	7557.2
37	-1728.0	-231.0	-853.0	182.3	191.7	35.8	-60.6	18.4	30.6	2.0	1961.0
38	-2105.0	-421.4	-2112.1	1733.6	453.2	4.4	-54.8	83.9	0.4	1.0	3507.1

Table 3.7-11

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Member Moments (units in ft-kips x 10^3)

	Mode Number							
	1		2		3		4	
Member	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
1	14.62	158.94	-0.05	145.11	-16.98	-17.15	-0.02	1.91
2	251.79	-1012.98	-145.39	216.41	-86.79	290.43	-2.03	-27.76
3	1091.38	-1814.46	-215.15	214.31	-376.84	506.47	27.33	-14.83
4	1890.82	-3026.67	-212.25	188.13	-587.73	759.63	14.16	-11.73
5	3103.17	-4158.56	-185.04	148.54	-838.79	950.03	10.85	-19.68
6	4240.21	-5913.52	-144.29	69.50	-1031.02	1117.29	18.82	-47.76
7	6001.30	-7631.71	-63.67	-23.65	-1202.18	1160.43	45.71	-87.16
8	8057.97	-8645.82	50.72	-87.53	-1158.00	1087.92	5.67	-7.36
9	8860.00	-10308.3	104.85	-206.56	-1097.75	773.93	-30.96	18.10
10	10964.1	-10964.1	255.01	-255.01	-1369.25	-1369.25	-41.39	41.39
11	0.05	-2.00	0.0	-0.22	-0.09	0.27	0.0	0.48
12	2.02	-17.51	0.22	0.42	-0.36	8.32	-0.48	-2.29
13	17.74	-43.84	-0.41	1.40	-8.39	21.44	2.28	-6.68
14	43.71	-61.10	-1.39	1.89	-21.68	29.64	6.66	-9.12
15	61.45	-77.13	-1.88	2.29	-29.73	36.67	9.10	-11.17
16	77.66	-93.91	-2.27	2.62	-36.82	43.56	11.13	-13.03
17	93.92	-117.60	-2.59	2.91	-43.94	52.53	12.94	-15.10
18	117.88	-146.34	-2.88	0.03	-52.92	42.38	14.98	-3.78
19	146.87	-198.55	0.0	-5.20	-42.57	23.12	3.67	16.71
20	199.05	-251.37	5.24	-10.55	-23.40	3.25	-16.83	37.45
21	251.34	-304.20	10.58	-15.98	-3.50	-17.31	-37.53	58.27
22	313.71	-374.01	16.69	-23.74	9.99	-45.55	-59.76	79.87
23	4.26	-0.16	0.37	0.78	-7.56	12.06	-4.64	-1.26
24	0.37	1.22	-0.76	1.87	-12.39	18.09	1.03	-6.33
25	-0.82	-3.34	-1.82	1.68	-18.46	19.00	5.85	-4.61
26	3.54	-10.92	-1.66	1.06	-19.09	17.78	4.33	-0.70
27	13.54	-23.11	-0.68	-0.57	-17.70	12.63	-3.11	9.12
28	23.23	-45.87	0.61	-3.68	-12.61	-0.61	-9.53	23.65
29	46.91	-54.81	3.84	-4.09	0.74	-1.24	-25.12	25.01
30	55.21	-72.71	4.15	-5.43	1.28	-7.39	-25.46	27.42
31	72.88	-93.05	5.46	-7.22	7.42	-16.80	-27.65	30.31
32	93.30	-116.06	7.24	-9.47	16.80	-29.75	-30.49	33.36
33	116.31	-142.28	9.50	-12.33	29.67	-48.02	-33.51	36.12
34	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
35	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
36	-368.38	368.38	-23.88	23.88	-37.07	37.07	80.19	-80.19
37	-107.26	107.26	-11.98	11.98	-47.94	47.94	36.19	-36.19
38	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 3.7-11

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Member Moments (units in ft-kips x 10^3) (Continued)

	Mode Number							
	5		6		7		8	
Member	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
1	-0.39	0.28	0.35	-0.14	0.64	-0.53	3.71	-3.38
2	-2.49	2.80	1.94	-13.88	3.61	-8.77	19.85	-30.89
3	-4.61	0.99	14.84	-23.41	10.87	-12.77	42.34	-40.47
4	-2.73	-2.95	23.65	-34.08	14.31	-15.00	49.54	-37.98
5	1.11	-6.60	33.12	-39.49	15.89	-14.53	44.92	-28.61
6	4.46	-13.76	37.08	-42.56	14.86	-10.26	34.44	-3.37
7	10.84	-20.69	37.34	-38.72	9.80	-4.17	9.00	18.88
8	9.85	-10.05	35.61	-35.49	5.54	-4.38	-18.89	27.22
9	-1.00	2.60	29.62	-27.80	5.44	-4.24	-24.41	37.96
10	-31.04	31.04	-37.44	37.44	-3.82	3.82	24.49	-24.49
11	0.02	-0.11	0.0	-0.02	-0.01	0.03	0.0	0.0
12	0.12	-1.77	0.02	-0.22	-0.03	0.25	0.0	-0.18
13	1.78	-4.50	0.22	-0.55	-0.26	0.62	0.18	-0.45
14	4.53	-6.20	0.55	-0.75	-0.63	0.83	0.46	-0.60
15	6.22	-7.67	0.75	-0.92	-0.84	1.01	0.61	-0.72
16	7.69	-9.11	0.92	-1.09	-1.02	1.18	0.72	-0.81
17	9.13	-10.96	1.08	-1.28	-1.19	1.35	0.82	-0.88
18	10.96	-8.72	1.27	-1.33	-1.35	1.03	0.88	-0.29
19	8.71	-4.59	1.32	-1.43	-1.03	0.42	0.29	0.81
20	4.60	-0.37	1.41	-1.49	-0.42	-0.22	-0.81	1.93
21	0.35	3.91	1.47	-1.54	0.22	-0.88	-1.93	3.05
22	-4.43	8.99	0.92	-0.63	0.89	-1.80	-2.69	3.38
23	9.73	-9.55	-1.36	1.23	1.81	-1.52	-1.92	1.50
24	9.94	-9.92	-1.27	0.88	1.58	-0.85	-1.54	0.51
25	10.29	-9.66	-0.89	0.47	0.85	-0.21	-0.50	-0.29
26	9.75	-8.55	-0.46	-0.07	0.19	0.54	0.30	-1.12
27	8.34	-6.61	0.23	-0.50	-0.64	0.85	0.92	-0.98
28	6.58	-2.51	0.52	-0.95	-0.86	1.03	0.96	-0.72
29	2.28	-1.54	1.00	-1.03	-1.03	0.97	0.58	-0.37
30	1.46	0.55	1.04	-0.90	-0.96	0.52	0.32	0.32
31	-0.61	2.98	0.90	-0.63	-0.51	-0.10	-0.35	1.07
32	-3.03	5.64	0.63	-0.25	0.12	-0.87	-1.10	1.78
33	-5.69	8.39	0.23	0.25	0.88	-1.72	-1.80	2.28
34	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
35	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
36	9.22	-9.22	-0.14	0.14	-1.75	1.75	2.91	-2.91
37	8.41	-8.41	0.28	-0.28	-1.72	1.72	2.25	-2.25
38	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 3.7-11

Reactor Building - Seismic Analysis
Horizontal N-S Direction - OBE
Member Moments (units in ft-kips x 10^3) (Continued)

	Mode Number					
	9		10		Combined	
Member	Top	Bottom	Top	Bottom	Top	Bottom
1	0.15	-0.15	-1.48	1.47	22.83	215.95
2	0.57	-0.62	-4.28	2.50	304.36	1076.96
3	0.87	-0.53	-4.09	2.45	1176.1	1896.69
4	0.70	-0.23	-3.63	1.89	1992.61	3126.87
5	0.37	-0.12	-2.76	2.29	3220.62	4268.73
6	0.25	-0.17	-2.96	3.66	4366.61	6018.91
7	0.30	-0.51	-3.87	5.21	6121.17	7720.12
8	0.19	-0.17	-5.06	5.48	8141.03	8714.58
9	-0.25	0.77	-5.17	5.94	8928.51	10339.5
10	0.22	-0.22	1.26	-1.26	11052.4	11052.4
11	0.0	0.02	0.0	0.0	0.10	2.09
12	-0.02	0.0	0.0	-0.01	2.12	19.61
13	0.0	-0.03	0.01	-0.01	19.84	49.50
14	0.03	-0.04	0.01	-0.02	49.49	68.84
15	0.04	-0.04	0.02	-0.02	69.19	86.56
16	0.04	-0.04	0.02	-0.02	87.06	104.79
17	0.03	-0.01	0.02	-0.02	104.95	130.21
18	0.01	0.01	0.03	-0.02	130.60	152.66
19	-0.02	0.07	0.02	0.0	153.21	200.72
20	-0.07	0.12	0.0	0.01	201.25	254.39
21	-0.13	0.19	-0.01	0.03	254.39	310.68
22	-0.22	0.38	0.0	0.02	320.00	386.02
23	0.29	-0.11	0.03	-0.01	14.40	15.80
24	0.11	0.19	0.01	0.02	16.29	21.76
25	-0.17	0.23	-0.02	0.02	22.08	22.14
26	-0.21	0.19	-0.02	0.01	22.23	22.65
27	0.18	-0.29	0.03	-0.04	24.06	28.71
28	0.33	-0.56	0.04	-0.06	28.93	51.84
29	0.68	-0.68	0.07	-0.07	53.44	60.44
30	0.71	-0.56	0.07	-0.04	60.99	78.26
31	0.57	-0.31	0.04	-0.01	78.51	99.61
32	0.31	0.05	0.01	0.03	99.90	124.89
33	-0.06	0.49	-0.03	0.07	125.15	155.22
34	0.0	0.0	0.0	0.0	0.0	0.0
35	0.0	0.0	0.0	0.0	0.0	0.0
36	0.37	-0.37	0.0	0.0	379.72	379.72
37	0.49	-0.49	0.07	-0.07	123.87	123.87
38	0.0	0.0	0.0	0.0	0.0	0.0

Table 3.7-12

Reactor Building - Seismic Analysis
Vertical Direction - OBE
Acceleration (units in g x 10^{-3})

	Mode Number										
Mass	1	2	3	4	5	6	7	8	9	10	Combined
1	159.32	-0.14	-38.87	7.46	1.10	0.41	-0.53	-0.01	0.0	-0.06	164.22
2	151.73	-0.10	-20.81	-0.27	-0.16	-0.21	1.09	0.02	0.0	-0.33	153.15
3	149.49	-0.09	-17.58	-0.41	-0.15	-0.14	0.21	0.0	0.0	0.28	150.52
4	146.55	-0.08	-13.80	-0.42	-0.10	-0.05	-0.50	-0.02	0.0	0.30	147.20
5	142.40	-0.06	-8.79	-0.37	-0.04	0.06	-1.06	-0.02	0.0	0.04	142.68
6	138.52	-0.05	-4.54	-0.29	0.02	0.14	-1.15	-0.02	0.0	-0.19	138.60
7	131.84	-0.03	2.00	-0.12	0.09	0.21	-0.76	0.0	0.0	-0.32	131.85
8	126.72	-0.01	6.16	0.0	0.13	0.23	-0.23	0.01	0.0	-0.15	126.87
9	125.01	-0.01	7.41	0.04	0.14	0.22	-0.05	0.02	0.0	-0.08	125.23
10	122.55	0.0	8.78	0.08	0.14	0.20	0.19	0.02	0.0	0.04	122.86
11	129.23	0.0	15.72	0.45	1.12	-11.97	-1.27	2.68	0.0	0.06	130.77
12	129.00	0.0	15.46	0.43	1.07	-11.30	-1.13	2.29	0.0	0.05	130.45
13	128.80	0.0	15.22	0.42	1.03	-10.72	-1.01	1.96	0.0	0.04	130.17
14	128.53	0.0	14.90	0.40	0.98	-9.93	-0.84	1.52	0.0	0.02	129.79
15	128.31	0.0	14.65	0.38	0.93	-9.35	-0.73	1.23	0.0	0.01	129.50
16	128.11	0.0	14.42	0.37	0.90	-8.81	-0.63	0.96	0.0	0.0	129.23
17	127.90	0.0	14.17	0.35	0.86	-8.26	-0.53	0.70	0.0	-0.01	128.96
18	127.57	0.0	13.80	0.33	0.79	-7.44	-0.39	0.36	0.0	-0.02	128.54
19	127.38	0.0	13.59	0.32	0.76	-6.99	-0.31	0.19	0.0	-0.02	128.30
20	126.97	0.0	13.14	0.29	0.69	-6.06	-0.16	-0.15	0.0	-0.03	127.80
21	126.54	0.0	12.67	0.26	0.62	-5.09	-0.02	-0.46	0.0	-0.03	127.28
22	126.10	0.0	12.19	0.24	0.55	-4.16	0.11	-0.73	0.0	-0.03	126.76
23	124.57	0.0	10.70	0.17	0.36	-2.13	0.17	-0.43	0.0	0.01	125.05
24	156.00	1.22	-31.69	0.64	4.86	0.29	0.06	-0.01	-0.09	-0.03	159.28
25	153.67	1.10	-27.09	-0.43	3.12	0.15	0.03	0.0	0.07	0.03	156.09
26	149.00	0.88	-18.05	-0.39	-0.17	-0.10	-0.09	-0.01	0.29	0.11	150.09
27	145.12	0.71	-11.26	0.21	-2.22	-0.24	-0.12	-0.01	0.21	0.06	145.58
28	140.28	0.50	-3.28	0.48	-4.33	-0.37	-0.12	-0.01	0.02	-0.02	140.40
29	139.66	0.47	-2.50	0.49	-4.40	-0.37	-0.11	-0.01	0.0	-0.02	139.77
30	138.19	0.43	-0.83	0.50	-4.41	-0.35	-0.08	0.0	-0.04	-0.03	138.28
31	136.62	0.38	0.42	0.47	-4.05	-0.29	-0.04	0.0	-0.06	-0.02	136.69
32	133.97	0.30	2.32	0.41	-3.30	-0.20	0.02	0.01	-0.06	0.01	134.04
33	131.18	0.22	4.17	0.33	-2.47	-0.09	0.08	0.01	-0.06	0.03	131.29
34	128.22	0.14	5.98	0.25	-1.57	-0.02	0.14	0.01	-0.04	0.04	128.38
35	125.01	0.06	7.66	0.15	-0.58	0.13	0.17	0.02	-0.02	0.05	125.24

Table 3.7-13

Reactor Building - Seismic Analysis
Vertical Direction - OBE
Displacement (units in ft x 10^{-6})

	Mode Number										
Mass	1	2	3	4	5	6	7	8	9	10	Combined
1	7589.63	-1.03	-190.01	16.34	0.18	0.61	-0.40	0.0	0.0	0.02	7692.03
2	7227.78	-0.76	-101.69	-0.59	-0.32	-0.32	0.81	0.01	0.0	-0.11	7228.50
3	7121.02	-0.69	-85.94	-0.91	-0.29	-0.21	0.16	0.0	0.0	0.09	7121.54
4	6981.40	-0.60	-67.43	-0.92	-0.20	-0.07	-0.37	-0.01	0.0	0.10	6981.73
5	6783.59	-0.48	-42.96	-0.80	-0.07	0.09	-0.79	-0.01	0.0	0.01	6783.73
6	6598.67	-0.37	-22.17	-0.63	0.04	0.20	-0.86	-0.01	0.0	-0.06	6598.71
7	6280.28	-0.20	9.78	-0.27	0.18	0.32	-0.57	0.0	0.0	-0.11	6280.29
8	6036.62	-0.08	30.11	0.0	0.26	0.34	-0.17	0.01	0.0	-0.05	6036.70
9	5955.15	-0.04	36.20	0.08	0.28	0.33	-0.03	0.01	0.0	-0.03	5955.26
10	5837.88	0.01	42.90	0.18	0.28	0.30	0.14	0.01	0.0	0.01	5838.04
11	6256.11	0.01	76.86	0.98	2.22	-17.68	-0.95	1.53	0.0	0.02	6156.62
12	6145.35	0.01	75.55	0.94	2.13	-16.88	-0.84	1.30	0.0	0.02	6145.84
13	6135.73	0.01	74.39	0.91	2.04	-16.01	-0.75	1.11	0.0	0.01	6136.20
14	6122.58	0.01	72.81	0.87	1.93	-14.84	-0.63	0.87	0.0	0.01	6123.03
15	6112.33	0.01	71.59	0.83	1.85	-13.96	-0.55	0.70	0.0	0.0	6112.77
16	6102.83	0.01	70.47	0.80	1.77	-13.17	-0.47	0.55	0.0	0.0	6103.25
17	6092.76	0.01	69.28	0.77	1.69	-12.34	-0.40	0.40	0.0	0.0	6093.17
18	6077.03	0.01	67.46	0.72	1.57	-11.12	-0.29	0.20	0.0	-0.01	6077.42
19	6067.97	0.01	66.42	0.69	1.51	-10.44	-0.23	0.11	0.0	-0.01	6068.34
20	6048.69	0.01	64.23	0.64	1.37	-9.05	-0.12	-0.09	0.0	-0.01	6049.04
21	6028.00	0.01	61.90	0.58	1.22	-7.61	-0.02	-0.26	0.0	-0.01	6028.32
22	6007.02	0.01	59.58	0.52	1.08	-6.22	0.08	-0.42	0.0	-0.01	6007.32
23	5934.34	0.01	52.28	0.37	0.72	-3.19	0.13	-0.24	0.0	0.0	5934.57
24	7431.29	9.41	-154.90	-1.40	9.61	0.43	0.04	0.0	-0.03	-0.01	7432.91
25	7320.58	8.54	-132.40	-0.94	6.16	0.23	0.0	0.0	0.03	0.01	7321.78
26	7097.74	6.83	-88.21	-0.09	-0.35	-0.15	-0.07	-0.01	0.11	0.04	7098.29
27	6913.16	5.49	-55.01	0.47	-4.40	-0.36	-0.09	-0.01	0.08	0.02	6913.38
28	6682.37	3.87	-16.01	1.05	-8.56	-0.55	-0.09	-0.01	0.01	-0.01	6682.40
29	6653.12	3.69	-12.22	1.08	-8.69	-0.55	-0.08	0.0	0.0	-0.01	6653.14
30	6582.74	3.29	-4.05	1.10	-8.72	-0.52	-0.06	0.0	-0.02	-0.01	6582.75
31	6508.01	2.12	2.06	1.04	-8.00	-0.44	-0.03	0.0	-0.02	-0.01	6508.02
32	6381.94	2.33	11.33	0.90	-6.53	-0.29	0.02	0.0	-0.02	0.0	6381.95
33	6249.23	1.72	20.40	0.73	-4.89	-0.13	0.06	0.01	-0.02	0.01	6249.27
34	6108.22	1.01	29.41	0.55	-3.10	0.03	0.10	0.01	-0.02	0.01	6108.29
35	5954.94	0.46	37.42	0.34	-1.15	0.19	0.13	0.01	-0.01	0.02	5955.06

Table 3.7-14

Reactor Building - Seismic Analysis
Vertical Direction - SSE
Acceleration (units in g x 10^{-3})

	Mode Number										
Mass	1	2	3	4	5	6	7	8	9	10	Combined
1	313.63	-0.21	-60.27	11.68	1.78	0.72	-1.07	-0.02	0.0	0.12	319.65
2	298.67	-0.15	-32.26	-0.42	-0.27	-0.38	2.18	0.05	0.0	-0.66	300.42
3	294.26	-0.13	-27.26	-0.65	-0.24	-0.25	0.42	0.0	0.0	0.57	295.52
4	288.49	-0.12	-21.39	-0.66	-0.16	-0.09	-0.99	-0.04	0.0	0.59	289.29
5	280.32	-0.09	-13.63	-0.57	-0.06	0.11	-2.2	-0.05	0.0	0.07	280.66
6	272.68	-0.07	-7.03	-0.45	0.03	0.24	-2.31	-0.04	0.0	-0.38	272.78
7	259.52	-0.04	3.10	-0.19	0.15	0.37	-1.52	0.0	0.0	-0.64	259.54
8	249.45	-0.02	9.55	0.0	0.21	0.39	-0.45	0.02	0.0	-0.30	249.64
9	246.08	-0.01	11.48	0.06	0.23	0.39	-0.09	0.03	0.0	0.15	246.35
10	241.24	0.0	13.61	0.13	0.23	0.35	0.37	0.03	0.0	0.08	241.62
11	254.39	0.0	24.38	0.70	1.81	-20.94	-2.54	5.38	0.0	0.13	256.49
12	253.94	0.0	23.97	0.67	1.74	-19.78	-2.26	4.60	0.0	0.10	255.90
13	253.55	0.0	23.60	0.65	1.67	-18.76	-2.02	3.93	0.0	0.08	255.38
14	253.00	0.0	23.09	0.62	1.58	-17.38	-1.69	3.06	0.0	0.04	254.68
15	252.58	0.0	22.71	0.60	1.51	-16.36	-1.47	2.46	0.0	0.02	254.15
16	252.19	0.0	22.35	0.57	1.45	-15.42	-1.26	1.93	0.0	0.0	253.66
17	251.77	0.0	21.98	0.55	1.38	-14.46	-1.06	1.41	0.0	-0.01	253.15
18	251.12	0.0	21.40	0.52	1.28	-13.02	-0.78	0.72	0.0	-0.03	252.38
19	250.75	0.0	21.07	0.50	1.23	-12.24	-0.63	0.37	0.0	-0.04	251.93
20	249.95	0.0	20.37	0.46	1.12	-10.60	-0.33	-0.30	0.0	-0.05	251.01
21	249.10	0.0	19.64	0.41	1.00	-8.92	-0.04	-0.93	0.0	-0.06	250.03
22	248.23	0.0	18.90	0.37	0.88	-7.29	0.22	-1.47	0.0	-0.06	249.06
23	245.23	0.0	16.58	0.26	0.59	-3.73	0.34	-0.86	0.0	0.01	245.82
24	307.08	1.83	-49.13	-1.00	7.85	0.51	0.12	0.01	-0.18	-0.06	311.12
25	302.51	1.66	-42.00	-0.67	5.03	0.27	0.01	0.0	0.15	0.06	305.47
26	293.30	1.33	-27.98	-0.06	-0.28	-0.18	-0.18	-0.02	0.59	0.21	294.64
27	285.67	1.07	-17.45	0.33	-3.59	-0.42	-0.23	-0.02	0.41	0.12	286.23
28	276.14	0.75	-5.08	0.75	-6.99	-0.65	-0.23	-0.02	0.04	-0.04	276.30
29	274.93	0.72	-3.88	0.77	-7.10	-0.64	-0.21	-0.02	0.0	-0.05	275.07
30	272.02	0.64	-1.28	0.79	-7.12	-0.61	-0.16	-0.01	-0.09	-0.06	272.14
31	268.93	0.57	0.65	0.74	-6.53	-0.52	-0.08	0.0	-0.11	-0.03	269.03
32	263.72	0.45	3.59	0.64	-5.34	-0.34	0.04	0.01	-0.12	0.01	263.82
33	258.24	0.33	6.47	0.52	-4.00	-0.16	0.16	0.02	-0.11	0.05	258.37
34	252.41	0.21	9.27	0.39	-2.53	0.03	0.27	0.03	-0.09	0.09	252.60
35	246.08	0.09	11.87	0.24	-0.94	0.22	0.34	0.03	-0.04	0.09	246.37

Table 3.7-15

Reactor Building - Seismic Analysis
Vertical Direction - SSE
Displacement (units in ft x 10^{-6})

	Mode Number										
Mass	1	2	3	4	5	6	7	8	9	10	Combined
1	14940.2	-1.62	-294.60	25.60	3.52	1.08	-0.80	-0.01	0.0	0.04	14943.1
2	14227.9	-1.14	-157.67	-0.93	-0.52	-0.56	1.63	0.03	0.0	-0.22	14228.8
3	14017.8	-1.04	-133.25	-1.42	-0.47	-0.37	0.32	0.0	0.0	0.19	14018.4
4	13742.9	-0.90	-104.55	-1.44	-0.33	-0.13	-0.74	-0.02	0.0	0.20	13743.3
5	13353.5	-0.72	-66.61	-1.26	-0.12	0.16	-1.59	-0.03	0.0	0.02	13353.7
6	12989.5	-0.56	-34.38	-0.98	0.06	0.36	-1.73	-0.02	0.0	-0.13	12989.5
7	12362.8	-0.30	15.16	-0.42	0.30	0.56	-1.14	0.0	0.0	-0.22	12362.8
8	11883.1	-0.12	46.69	-0.01	0.42	0.59	-0.34	0.01	0.0	-0.10	11883.2
9	11722.7	-0.06	56.13	0.12	0.45	0.58	-0.07	0.02	0.0	-0.05	11722.8
10	11491.9	0.01	66.52	0.28	0.46	0.52	0.28	0.02	0.0	0.03	11492.1
11	12118.3	0.01	119.16	1.54	3.58	-31.28	-1.90	3.06	0.0	0.04	12118.9
12	12097.2	0.01	117.14	1.48	3.43	-29.54	-1.69	2.61	0.0	0.03	12097.8
13	12078.2	0.01	115.35	1.43	3.30	-28.02	-1.51	2.23	0.0	0.02	12078.8
14	12052.3	0.01	112.88	1.36	3.12	-25.96	-1.27	1.74	0.0	0.01	12052.9
15	12032.1	0.01	111.00	1.30	2.98	-24.44	-1.10	1.40	0.0	0.01	12032.6
16	12013.4	0.01	109.26	1.26	2.86	-23.04	-0.95	1.09	0.0	0.0	12013.9
17	11993.6	0.01	107.41	1.21	2.73	-21.59	-0.79	0.80	0.0	0.0	11994.1
18	11962.7	0.01	104.59	1.13	2.54	-19.45	-0.58	0.41	0.0	-0.01	11963.2
19	11944.8	0.01	102.99	1.09	2.43	-18.28	-0.47	0.21	0.0	-0.01	11945.2
20	11906.9	0.01	99.59	1.00	2.20	-15.83	-0.25	-0.17	0.0	-0.02	11907.3
21	11866.1	0.01	95.98	0.91	1.97	-13.32	-0.03	-0.53	0.0	-0.02	11866.5
22	11824.8	0.01	92.38	0.81	1.74	-10.89	0.17	-0.83	0.0	-0.02	11825.2
23	11681.8	0.01	81.06	0.57	1.16	-5.57	0.25	-0.49	0.0	0.0	11682.1
24	14628.5	14.12	-240.17	-2.19	15.53	0.76	0.09	0.01	-0.07	-0.02	14640.5
25	14410.6	12.82	-205.28	-1.48	9.95	0.40	0.0	0.0	0.06	0.02	14412.1
26	13971.9	10.25	-136.77	-0.13	-0.56	-0.27	-0.14	-0.01	0.23	0.07	13972.6
27	13608.6	8.24	-85.30	0.73	-7.10	-0.63	-0.17	-0.01	0.16	0.04	13608.9
28	13154.3	5.81	-24.82	1.64	-13.82	-0.97	-0.17	-0.01	0.01	-0.01	13154.3
29	13096.7	5.54	-18.95	1.69	-14.04	-0.96	-0.16	-0.01	0.0	-0.02	13096.7
30	12958.1	4.94	-6.28	1.73	-14.09	-0.91	-0.12	-0.01	-0.03	-0.02	12958.1
31	12811.0	4.38	3.19	1.62	-12.91	-0.77	-0.06	0.0	-0.04	-0.01	12811.0
32	12562.9	3.49	17.57	1.40	-10.55	-0.51	0.03	0.01	-0.05	0.0	12562.9
33	12301.6	2.58	31.63	1.15	-7.90	-0.23	0.12	0.01	-0.04	0.02	12301.6
34	12024.0	1.65	45.29	0.86	-5.00	0.05	0.20	0.02	-0.03	0.03	12024.1
35	11722.3	0.70	58.02	0.53	-1.86	0.33	0.26	0.02	-0.02	0.03	11722.4

Table 3.7-16

Lumped Representation of Soil-Structure Interaction

Motion	Equivalent Spring Constant[a]	Equivalent Damping Coefficient[a,b]
	A. For a Rectangular Foundation	
Vertical	$k_v = \frac{G}{1-\nu}\beta_v\sqrt{4cd}$	(c)
Horizontal	$k_h = 4(1+\nu)G\beta_h\sqrt{cd}$	(c)
Rocking	$k_r = \frac{G}{1-\nu}\beta_R \cdot 8cd^2$	(c)
Torsional	(c)	(c)
	B. For a Circular Foundation	
Vertical	$k_v = \frac{4GR}{1-\nu}$	$c_v = 0.85k_vR\sqrt{\rho/G}$
Horizontal	$k_h = \frac{32(1-\nu)GR}{7-8}$	$c_h = 0.575k_hR\sqrt{\rho/G}$
Rocking	$k_R = \frac{8GR^3}{3(1-\nu)}$	$c_R = \frac{0.30}{1+B_R}k_RR\sqrt{\rho/G}$
Torsional	$k_t = \frac{16GR^3}{3}$	$c_t = \frac{\sqrt{k_tI_t}}{1+2B_t}$

[a] In above formulas

$2c$ = width of the rectangular foundation (along axis of rotation for the case of rocking)

$2d$ = length of the rectangular foundation (in the plane of rotation for rocking)

β_v, β_h, and β_R = constants depending on the ratio d/c (See Figure 10-16, p. 351 of Reference 3.7-5)

R = radius of the circular foundation

$$B_R = \frac{3(1-\nu)I_R}{8\rho R^5}$$

I_R = total mass moment of inertia of structure and foundation about the rocking axis at the base

$$B_T = I_T/\rho R^5$$

I_T = polar mass moment of inertia of structure and foundation

G = shear modulus of soil material

ν = Poisson's ratio of soil material

ρ = density of soil material

Table 3.7-16

Lumped Representation of Soil-Structure Interaction (Continued)

[b] Conservative values not exceeding those defined in Table 3.7-1 are used for the soil damping in the analytical model for soil-structure interaction.

[c] Use formulas and diagrams for an equivalent circular base with a radius determined by the following:

For translation: $R = \sqrt{\dfrac{4cd}{\pi}}$

For rocking: $R = \sqrt[4]{\dfrac{16cd^3}{3\pi}}$

For torsion: $R = \sqrt[4]{\dfrac{16cd\left(c^2 + d^2\right)}{6\pi}}$

Table 3.7-17

Dynamic Response Cycles Expected During a
Seismic Event for Nuclear Steam Supply Systems
and Components

	Frequency Bandwidth (Hz)		
Number of Seismic Cycles	0-10	10-20	20-50
Total	168	359	643
Between 75% and 100% of peak loads (0.5% of total)	0.8	1.8	3.2
Between 50% 75% of peak loads (4.5% of total)	7.5	16.2	28.9

Table 3.7-18

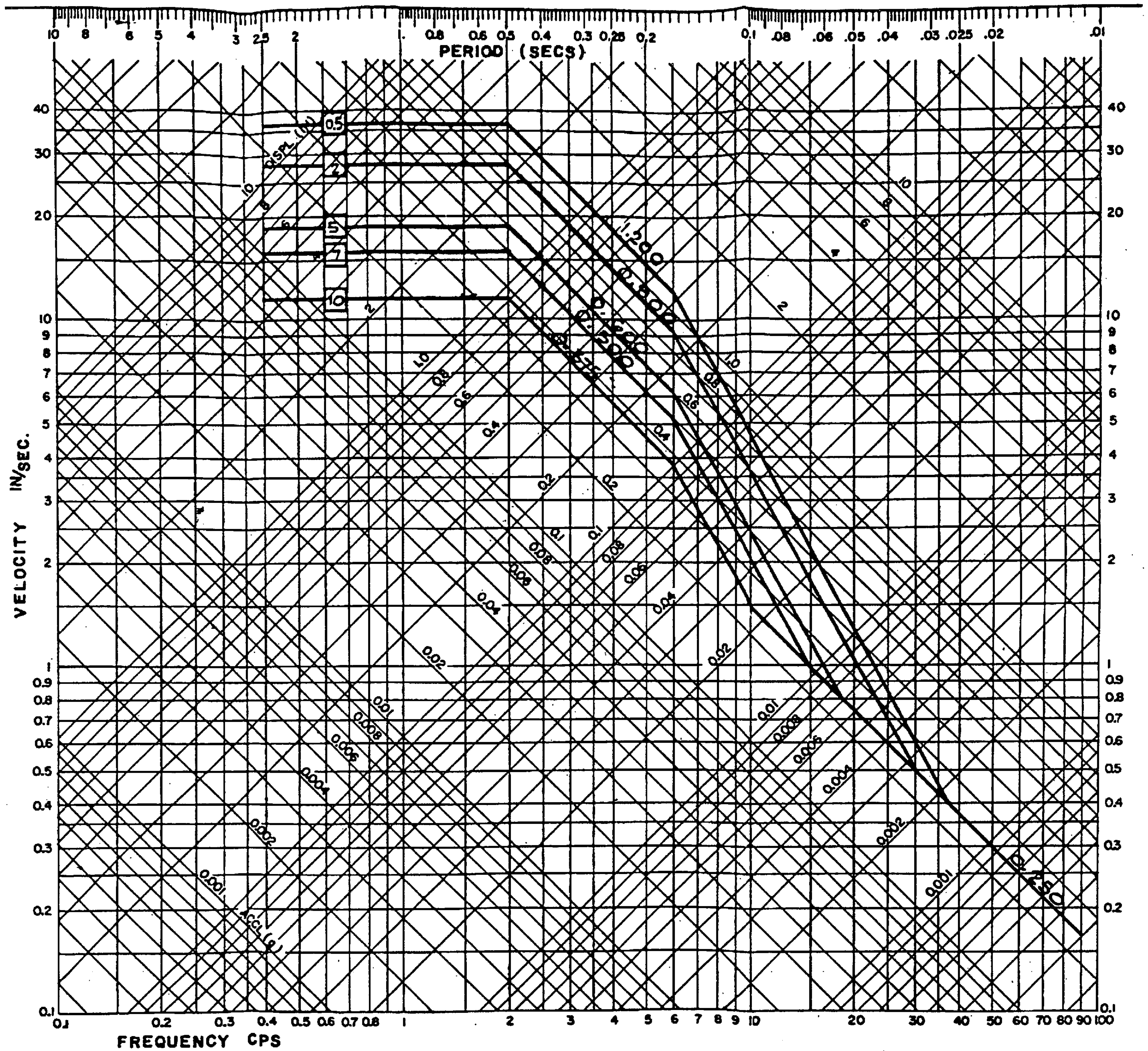
Fatigue Evaluation Due to Seismic Load

Component	Calculated Number of Cycles at Peak Stress	Design Number of OBE Cycles at Peak Stress
1. Reactor pressure vessel		
Vessel	3	10
Shroud support	3	10
Skirt	3	10
2. Category I piping		
Recirculation lines	3	50
Steam lines	3	50

Table 3.7-19

Comparison of the Maximum and Allowable Seismic Loads
of Reactor Pressure Vessel and Internals

	Seismic Loads		
Location	X-Excitation	Y-Excitation	Allowable Loads
Top guide shear (kip)	333	292	860
Core plate shear (kip)	333	290	860
Stabilizer force (total) (kip)	1250	1250	3600
Maximum fuel moment:			
Total (in.-kip)	16,960	14,760	40,300
Per bundle (in.-kip)	22.2	19.3	52.7
Maximum shroud moment (in.-kip)	220,500	218,500	450,900
Maximum shroud shear (kip)	1030	1030	2386
Maximum vessel shirt moment (in.-kip)	204,900	170,700	2,304,000
Vessel skirt shear (kip)	1130	1100	5200

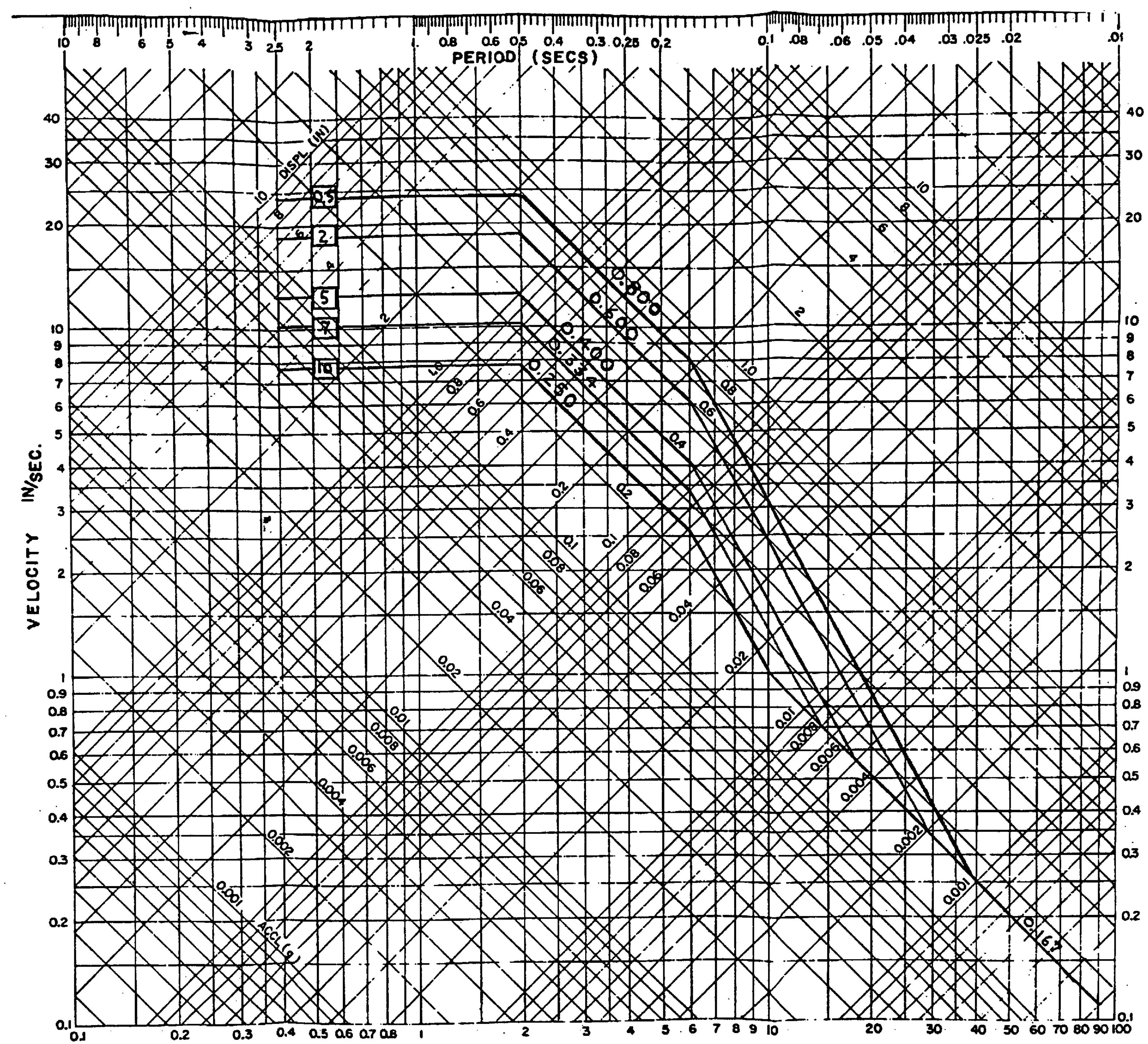


FOR 0.5%, 2%, 5%, 7% AND 10% DAMPING

Columbia Generating Station
Final Safety Analysis Report

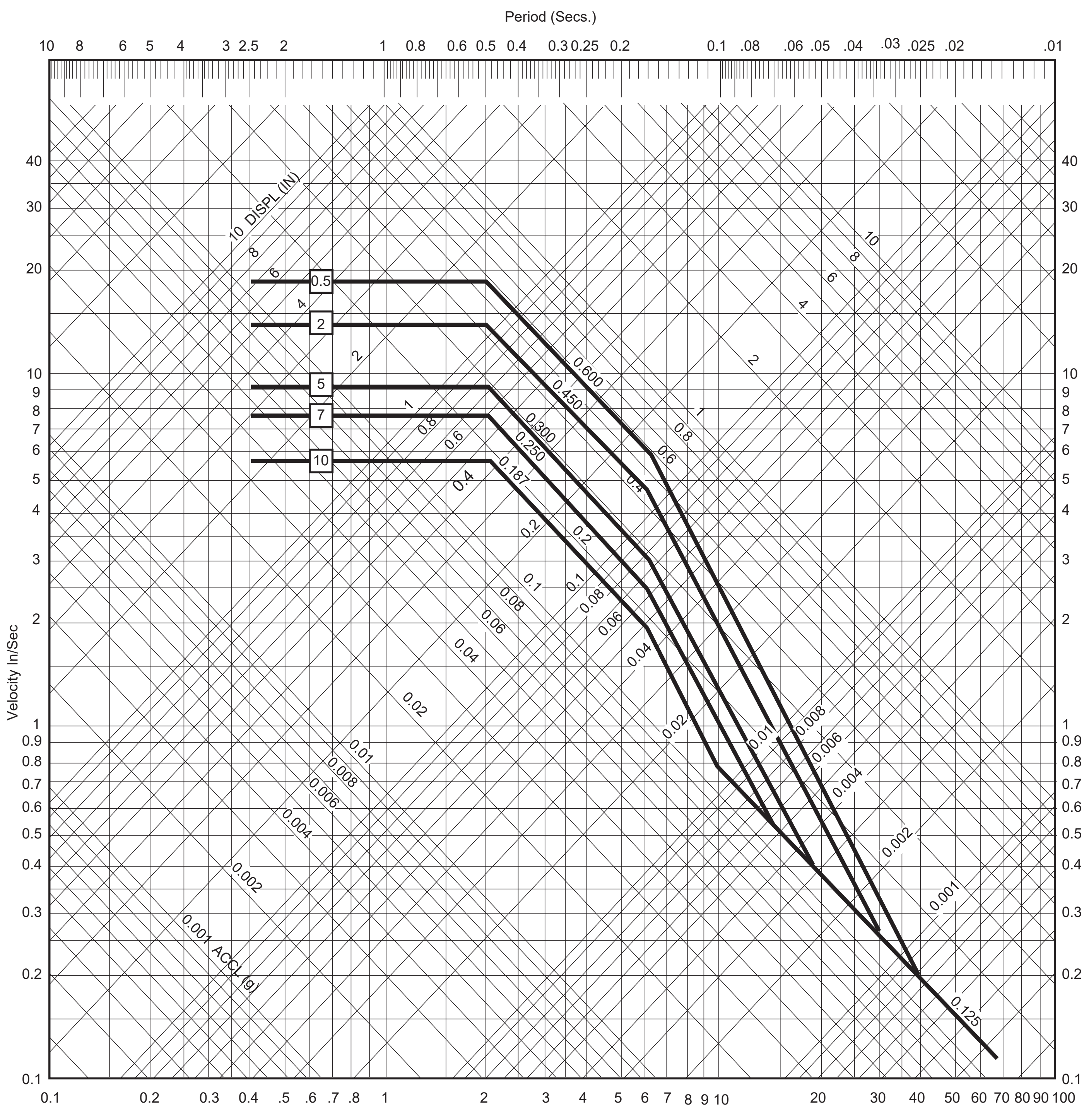
Response Spectra Safe Shutdown Earthquake - Horizontal Component

Draw. No. 020552.08	Rev.	Figure 3.7-1



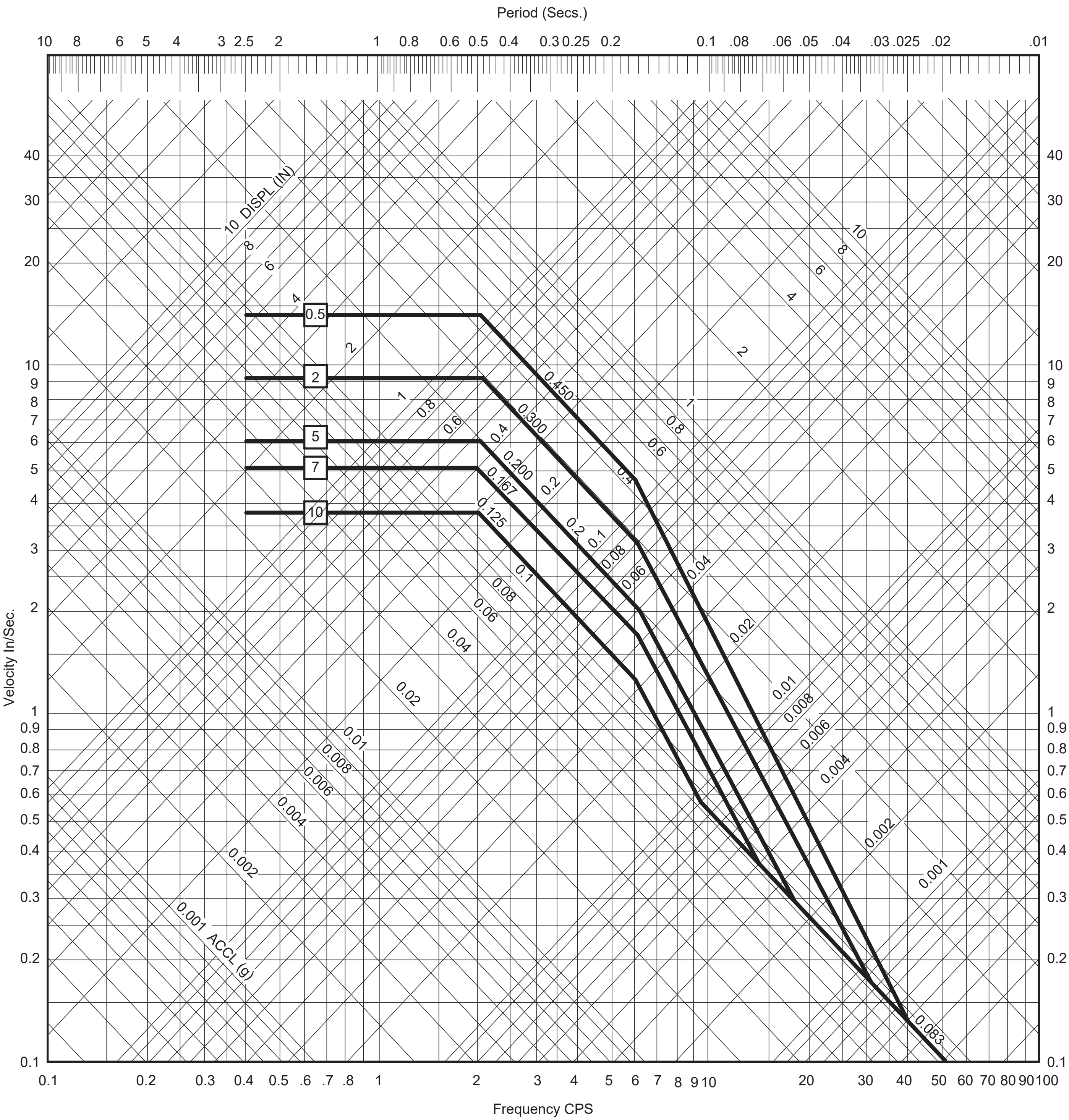
FOR 0.5%, 2%, 5%, 7% AND 10% DAMPING

Columbia Generating Station Final Safety Analysis Report	Response Spectra Safe Shutdown Earthquake - Vertical Component
	Draw. No. 020552.09 · Rev. · Figure 3.7-2



For 0.5%, 2%, 5%, 7% and 10% Damping

Columbia Generating Station Final Safety Analysis Report	Response Spectra Operating Basis Earthquake - Horizontal Component
	Draw. No. 990578.99	Rev.	Figure 3.7-3



For 0.5%, 2%, 5%, 7% and 10% Damping

Columbia Generating Station
Final Safety Analysis Report

Response Spectra Operating Basis Earthquake - Vertical Component

Draw. No. 990578.71	Rev.	Figure 3.7-4

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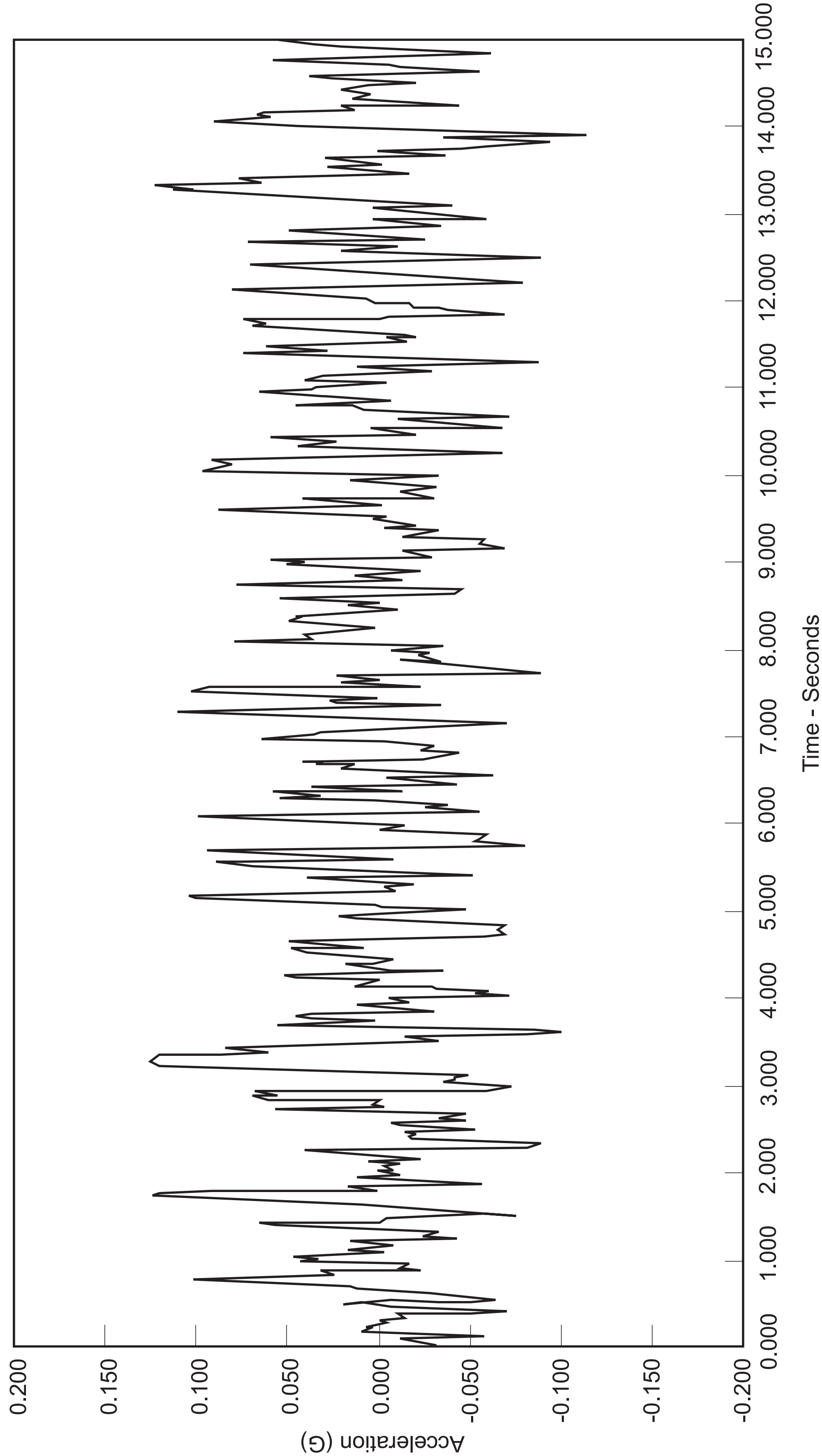
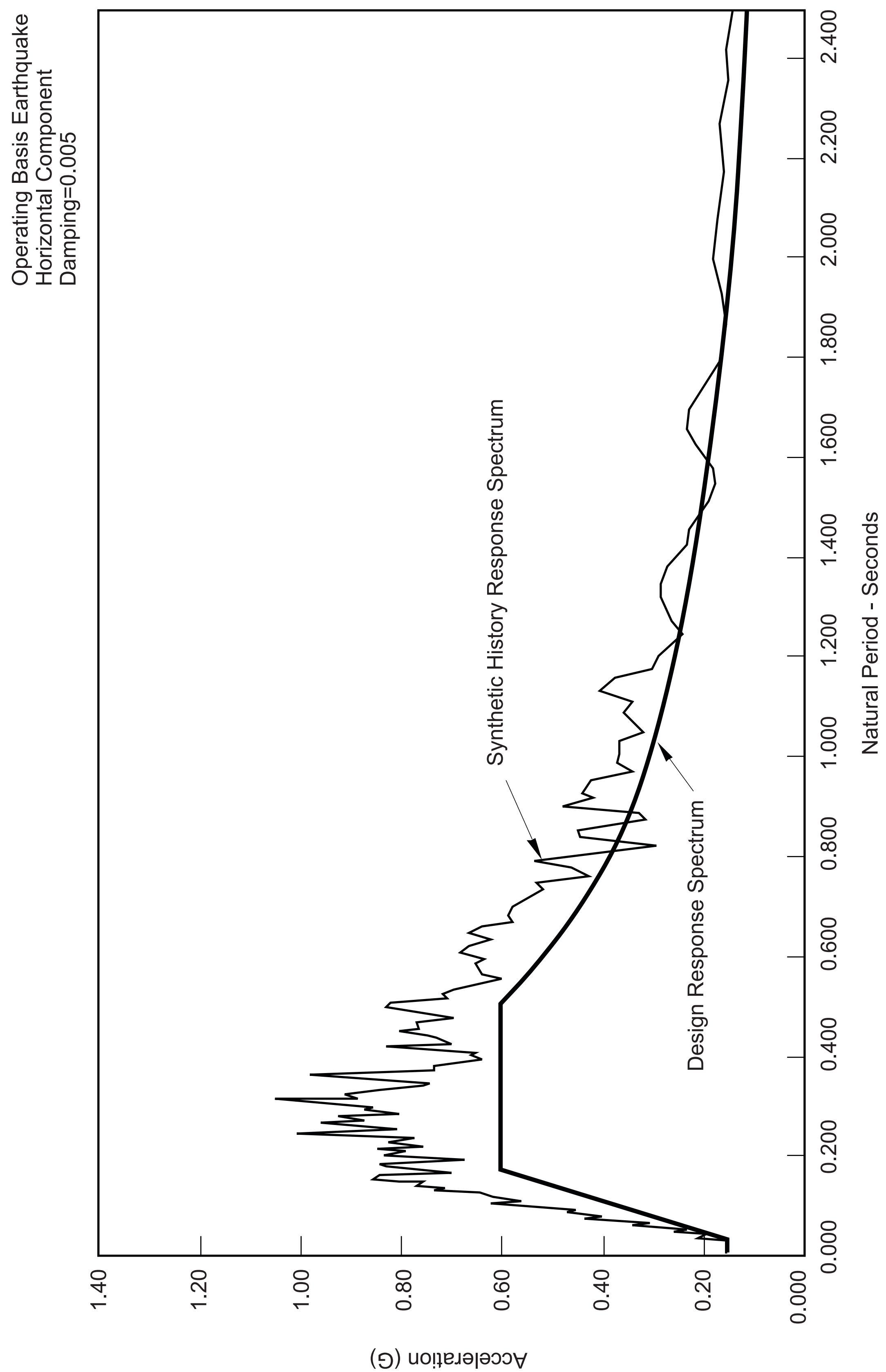
May 2001

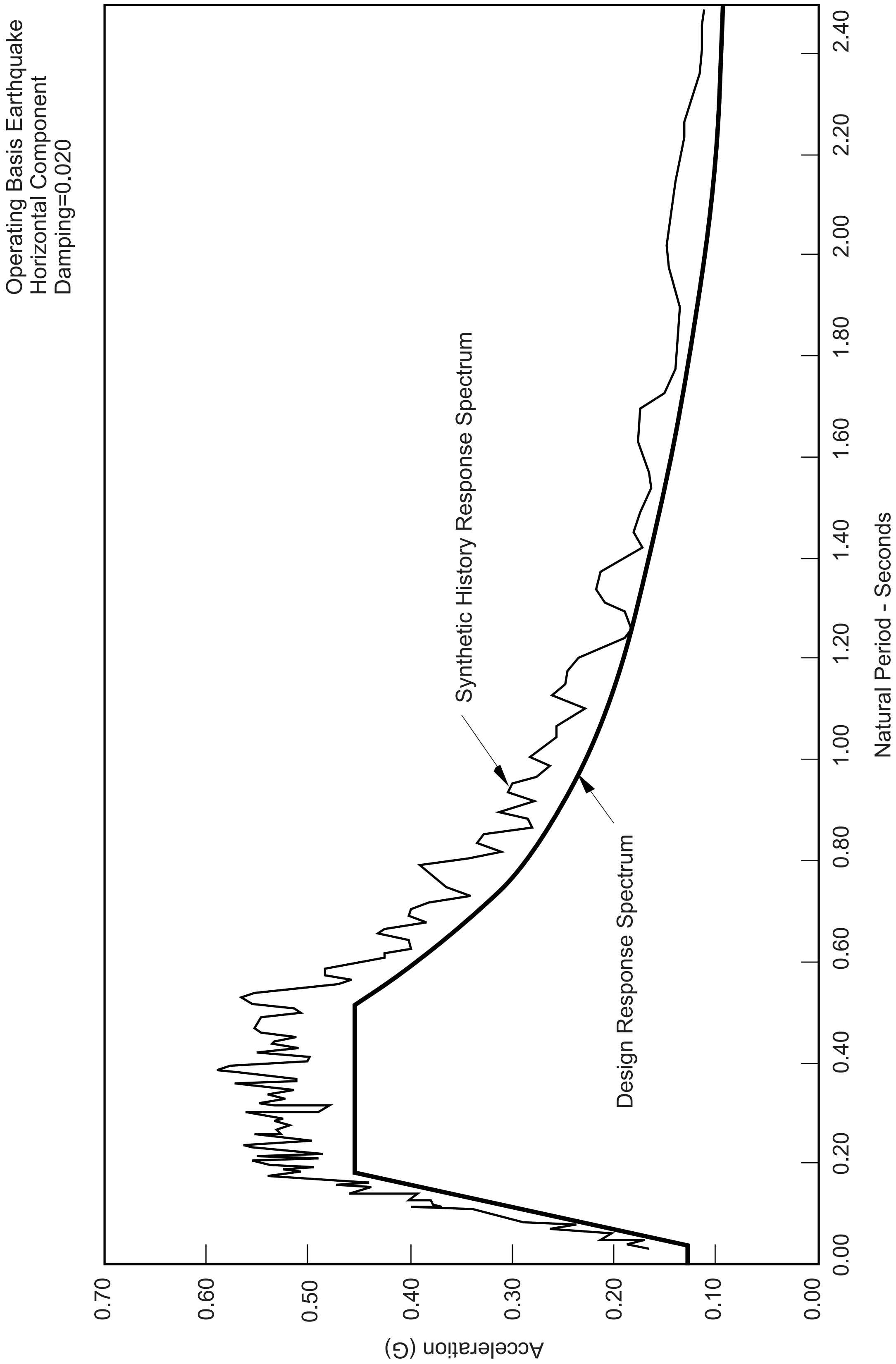


Columbia Generating Station Final Safety Analysis Report	Synthetic Time History of Ground Motion Acceleration - Duration = 15 Sec OBE - Horizontal Component		
	Draw. No. 010126.10	Rev.	Figure 3.7-5

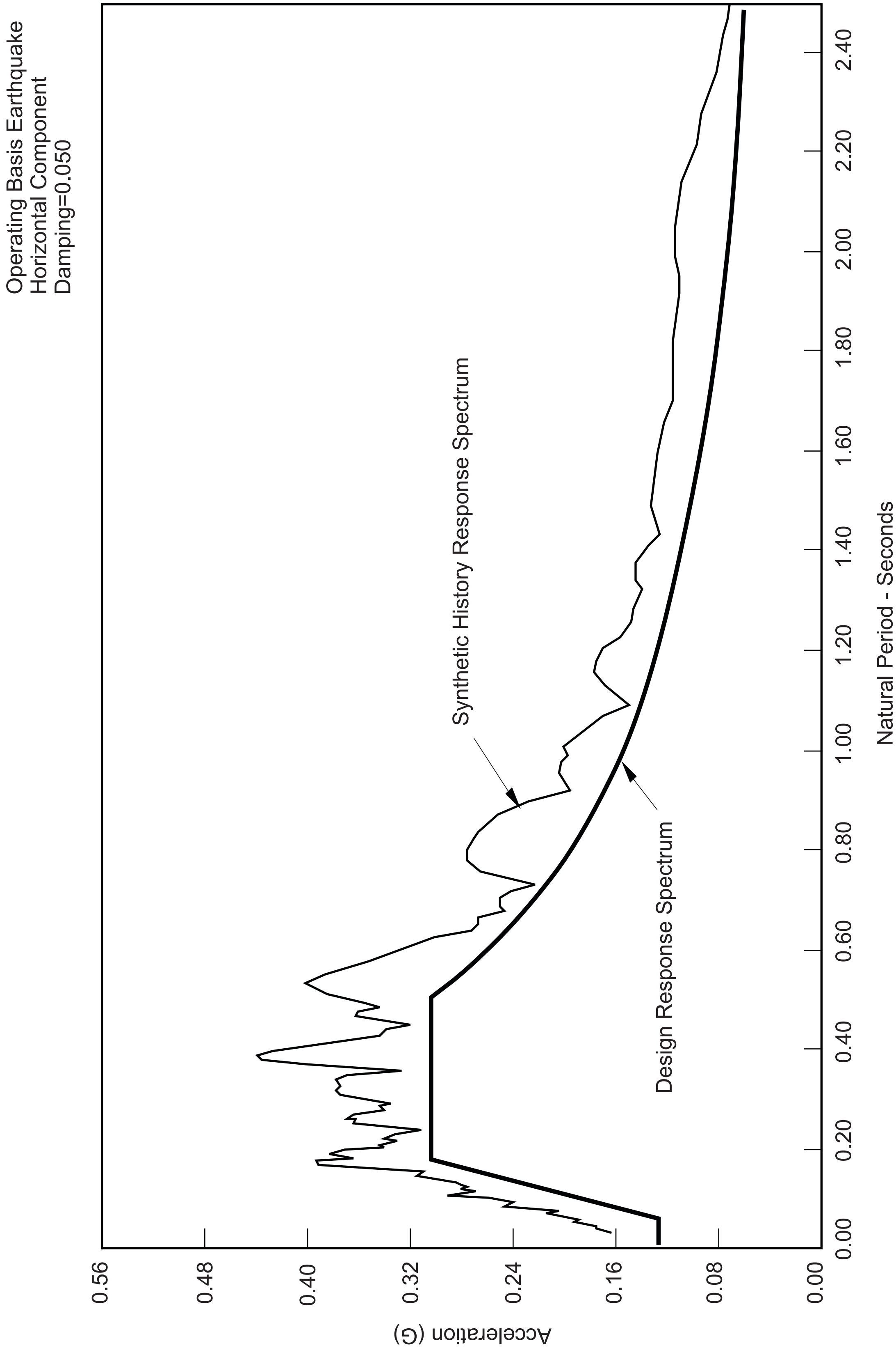
Form No. 960690



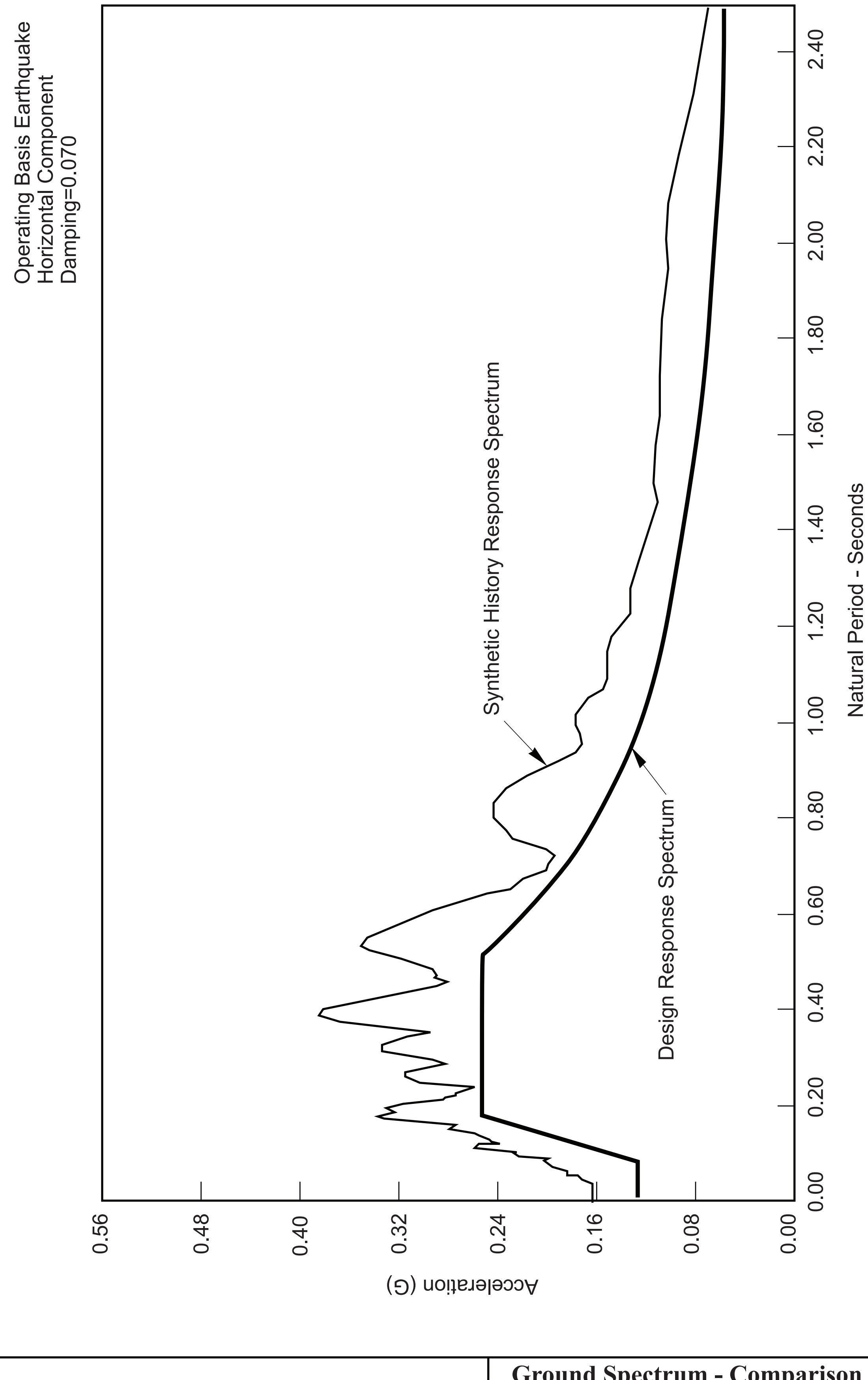
Columbia Generating Station Final Safety Analysis Report	Ground Spectrum - Comparison Between Design and Synthetic Time History Response Spectra (Damping 0.005)
	Draw. No. 010126.11 — Rev. 1 — Figure 3.7-6



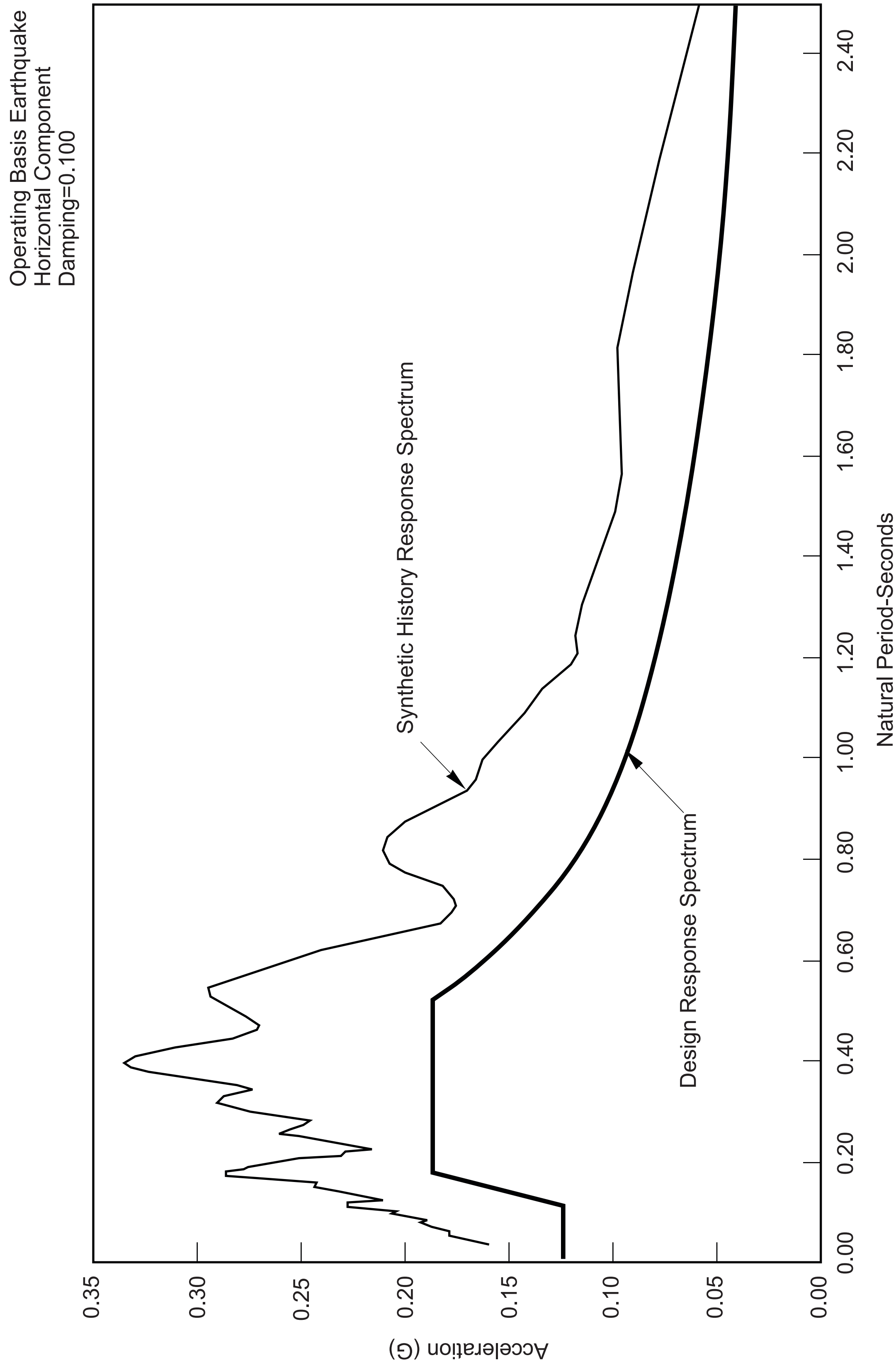
Columbia Generating Station **Final Safety Analysis Report**	**Ground Spectrum - Comparison Between Design and Synthetic Time History Response Spectra (Damping 0.020)**
	Draw. No. **010126.12** Rev. Figure **3.7-7**



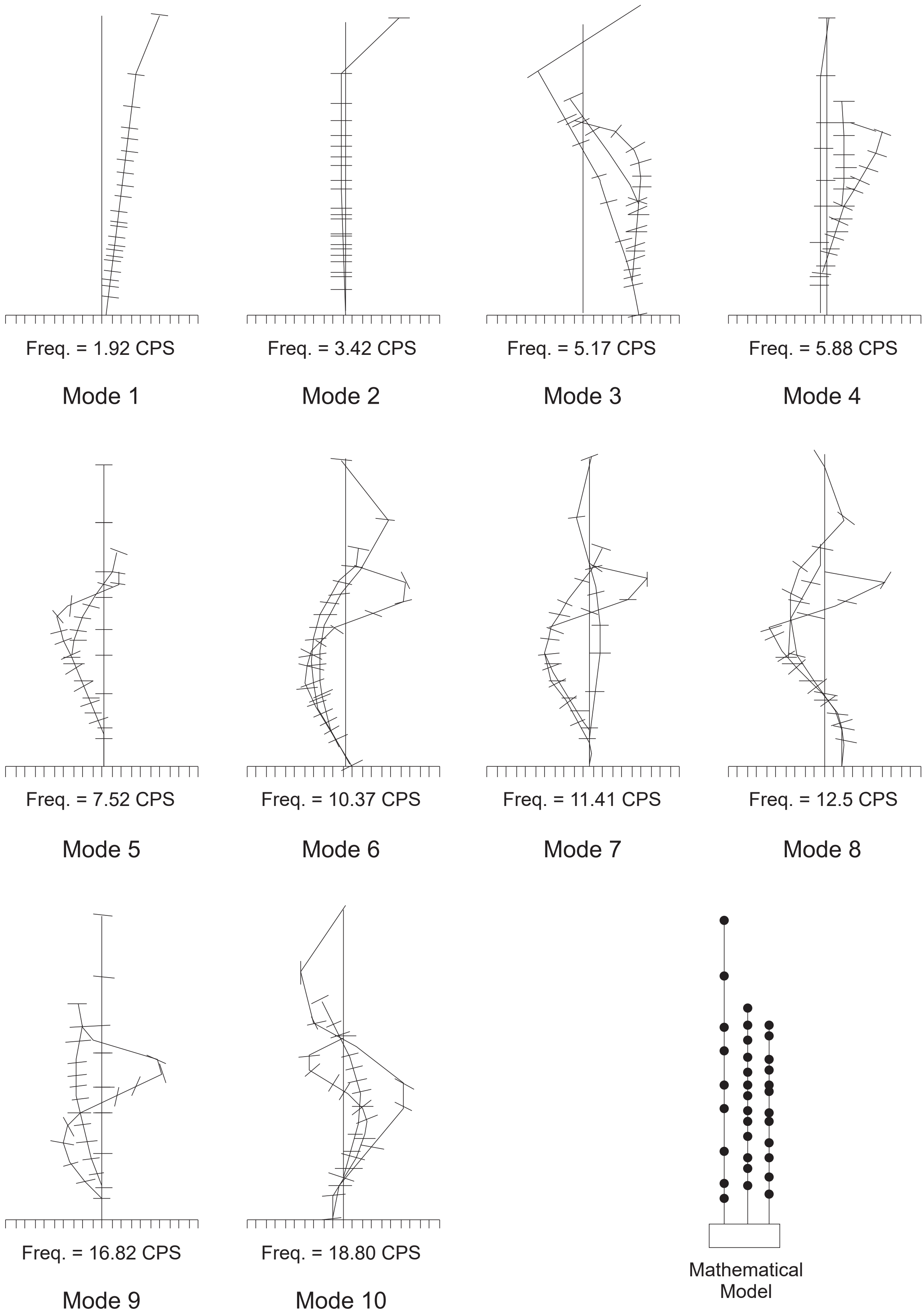
Columbia Generating Station Final Safety Analysis Report	Ground Spectrum - Comparison Between Design and Synthetic Time History Response Spectra (Damping 0.050)		
	Draw. No. 010126.13	Rev.	Figure 3.7-8



Columbia Generating Station **Final Safety Analysis Report**	**Ground Spectrum - Comparison Between Design and Synthetic Time History Response Spectra (Damping 0.070)**
	Draw. No. **010126.14** | Rev. | Figure **3.7-9**



Columbia Generating Station **Final Safety Analysis Report**	**Ground Spectrum - Comparison Between Design and Synthetic Time History Response Spectra (Damping 0.100)**
	Draw. No. **010126.15** Rev. Figure **3.7-10**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Mode 1
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Mode 2
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Mode 3
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Mode 4
Freq. = 7.52 CPS
Mode 5
Freq. = 10.37 CPS
Mode 6
Freq. = 11.41 CPS
Mode 7
Freq. = 12.5 CPS
Mode 8
Freq. = 16.82 CPS
Mode 9
Freq. = 18.80 CPS
Mode 10
Mathematical
Model

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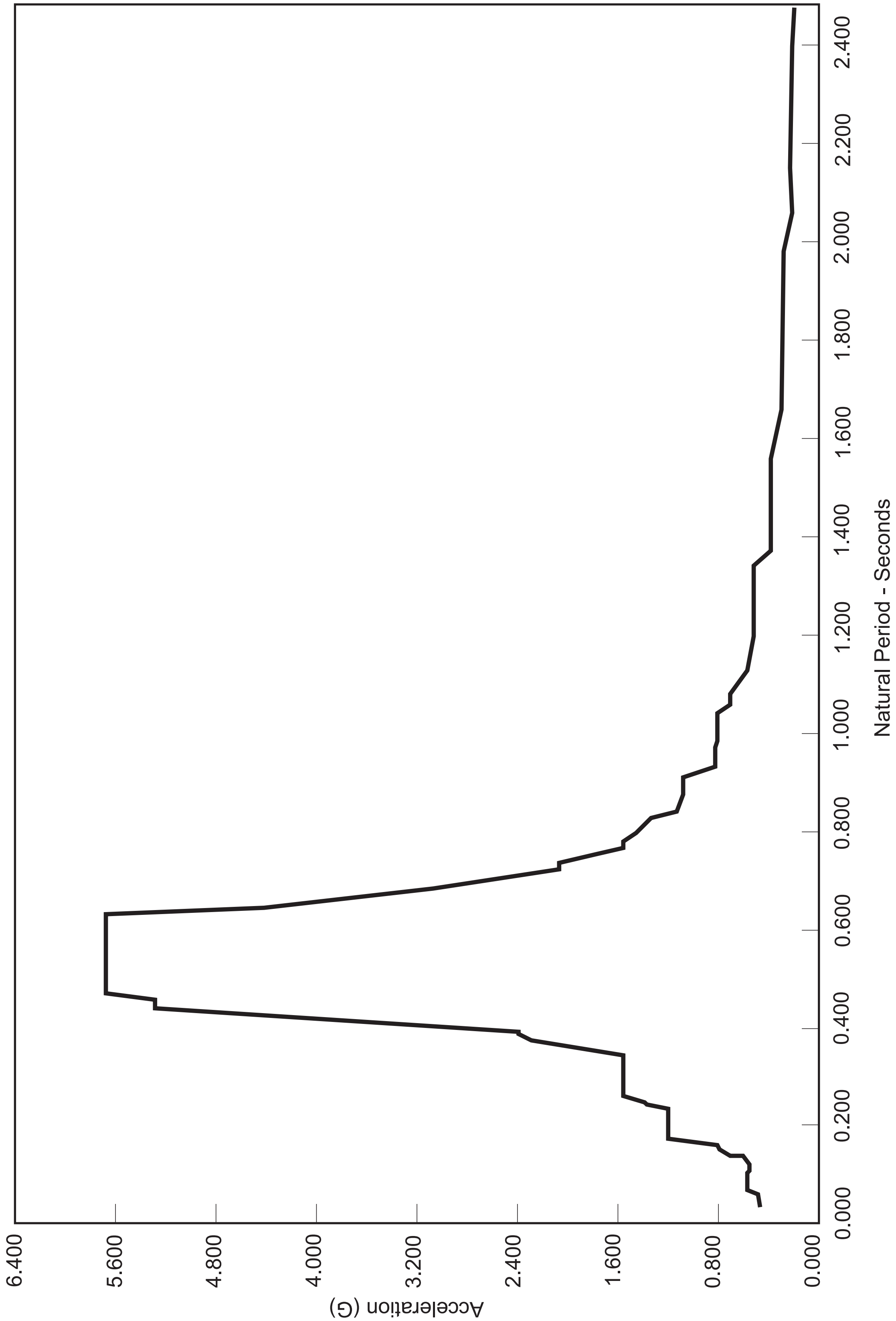
May 2001

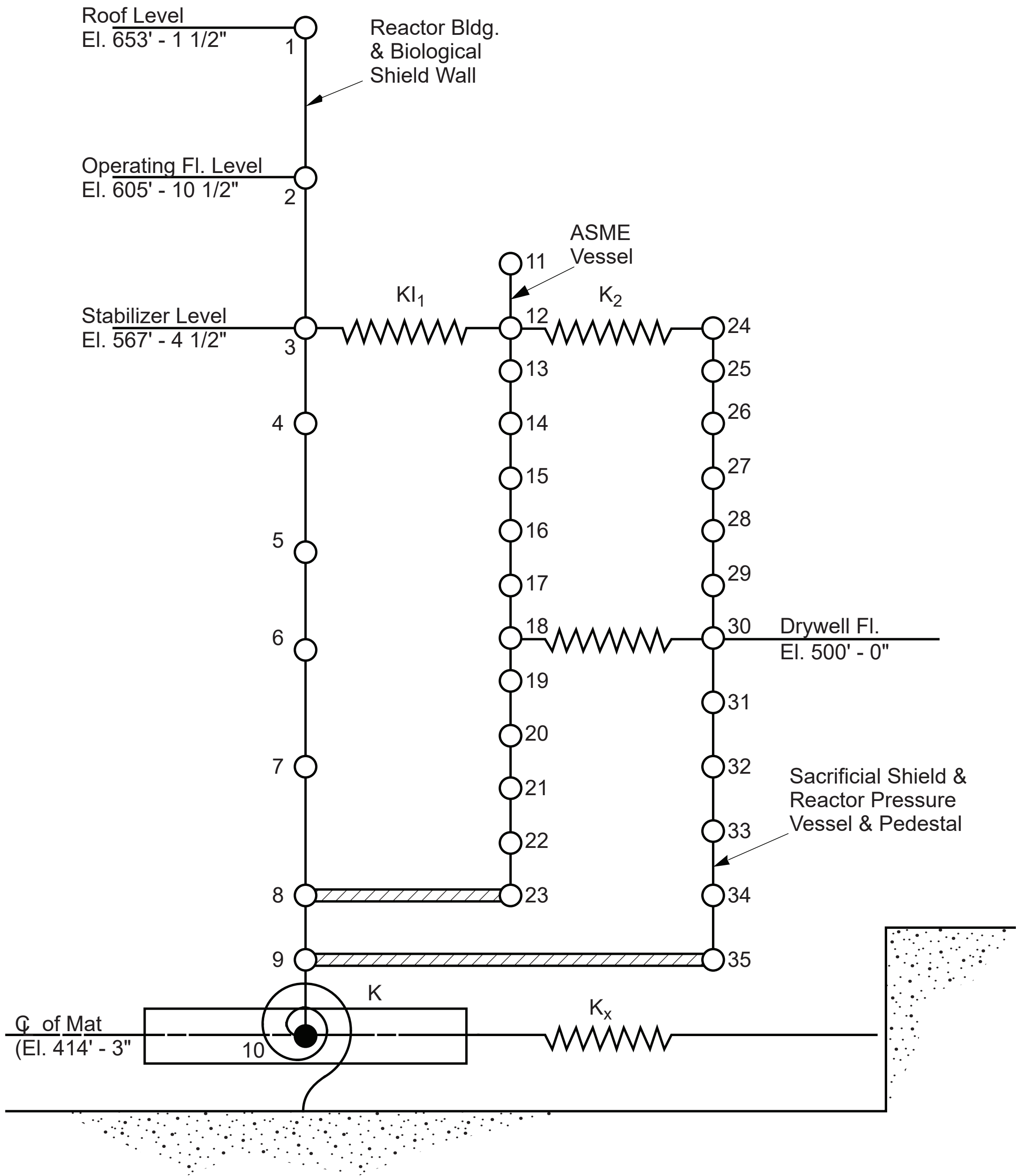


Columbia Generating Station
Final Safety Analysis Report

Reactor Building Refueling Floor Response Spectrum - Operating Basis Earthquake, 0.005 Damping, Horizontal NS and EW

Draw. No. 010126.02	Rev.	Figure 3.7-12

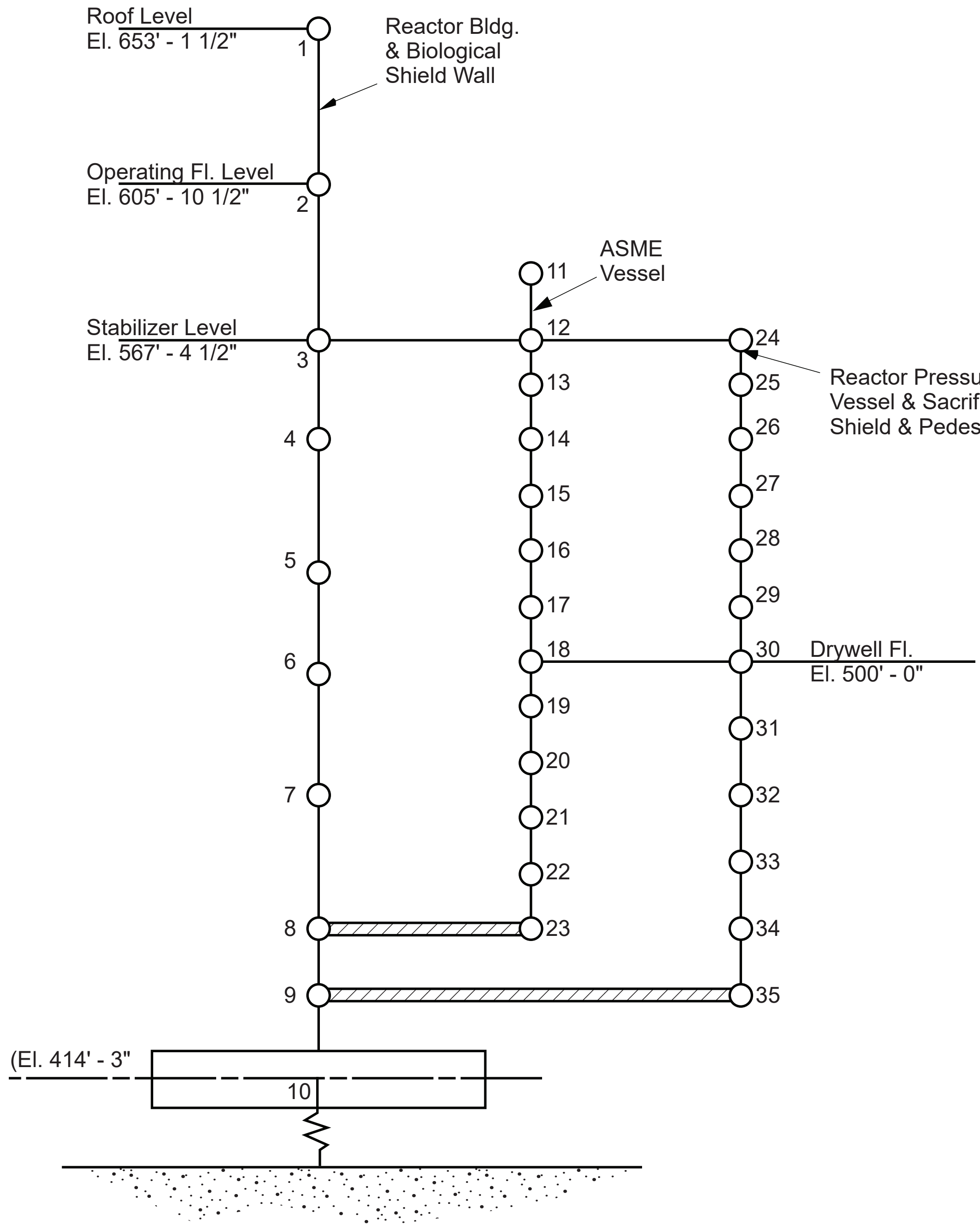
Form No. 960690



Columbia Generating Station
Final Safety Analysis Report

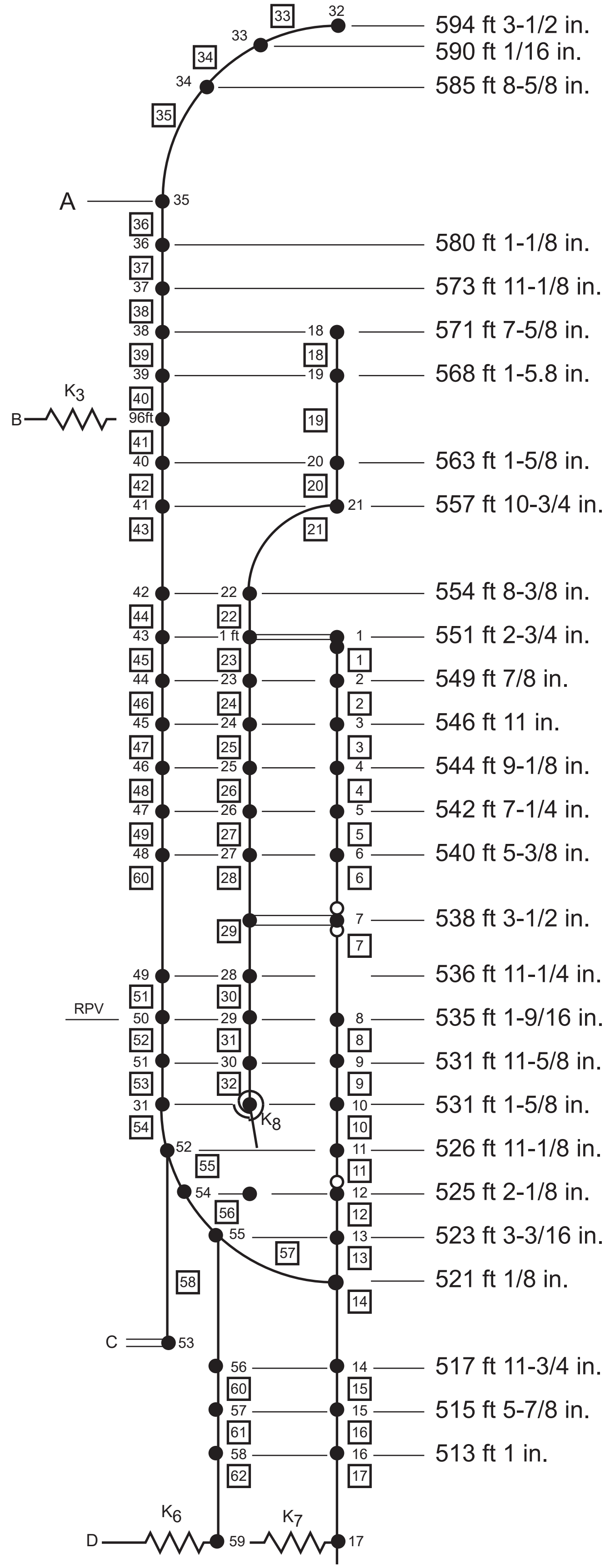
Mathematical Model of Reactor Building -
Horizontal Input Motion

Draw. No. 990306.65	Rev.	Figure 3.7-13.1

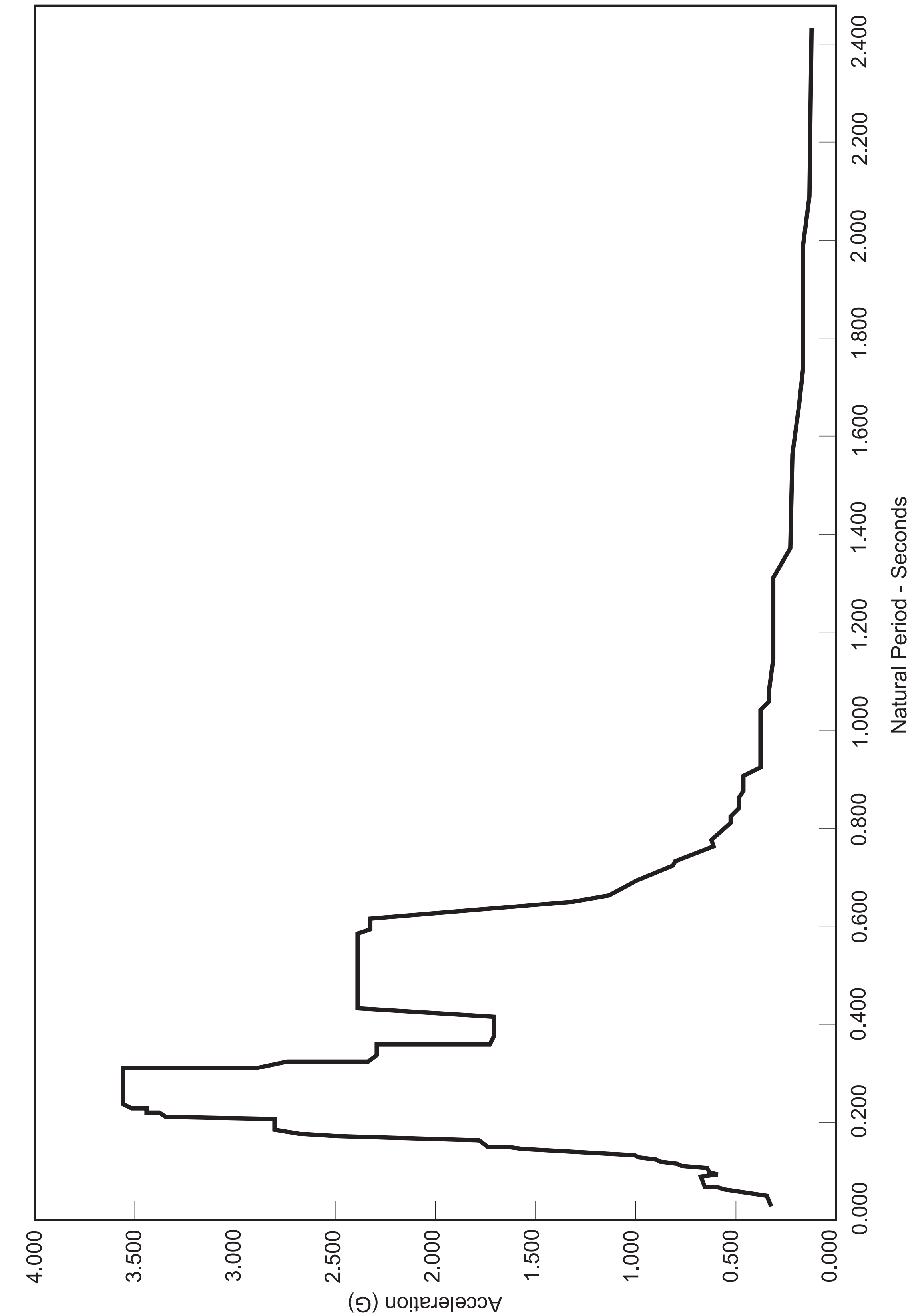


Mathematical Model of Reactor Building - Vertical Input Motion

Draw. No. 990306.66	Rev.	Figure 3.7-13.2



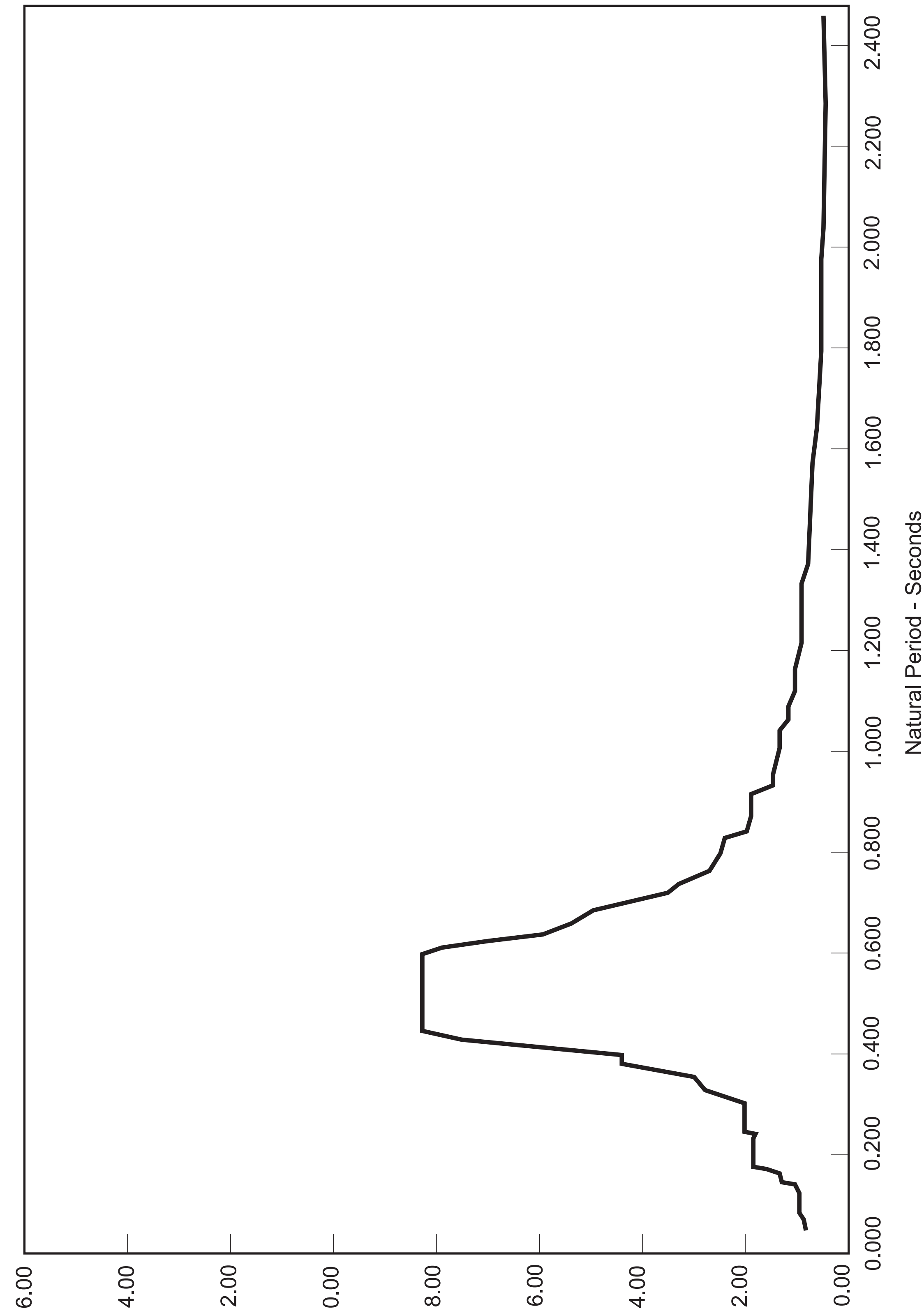
Reactor Pressure Vessel and Internals
Seismic Model



Columbia Generating Station Final Safety Analysis Report	Reactor Building Refueling Floor Response Spectrum - Operating Basis Earthquake, 0.005 Damping, Vertical		
	Draw. No. 010126.03	Rev.	Figure 3.7-15

Form No. 960690

Amendment 55
May 2001



Columbia Generating Station Final Safety Analysis Report	Reactor Building Refueling Floor Response Spectrum - Safe Shutdown Earthquake, 0.01 Damping, Horizontal NS and EW		
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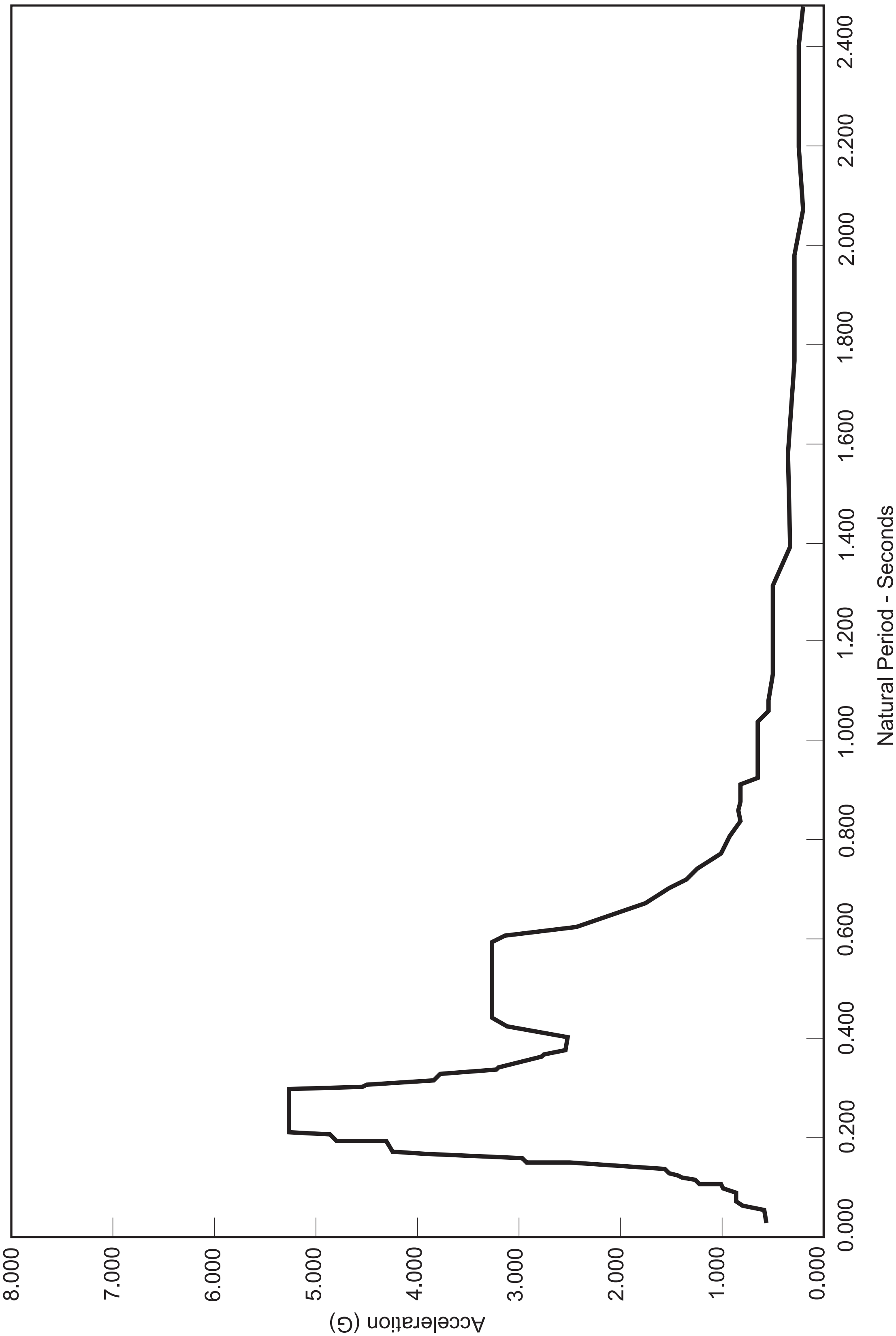
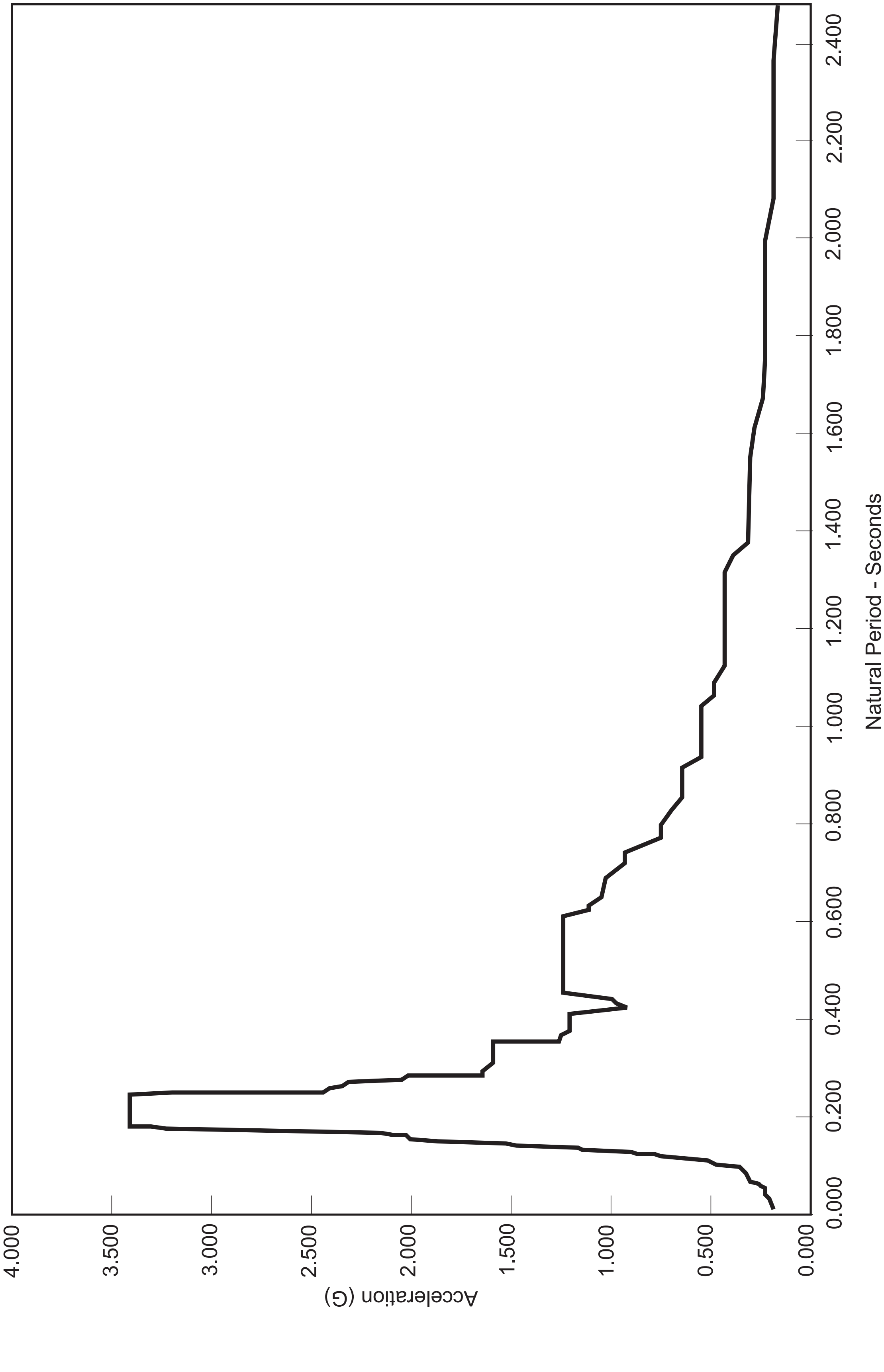


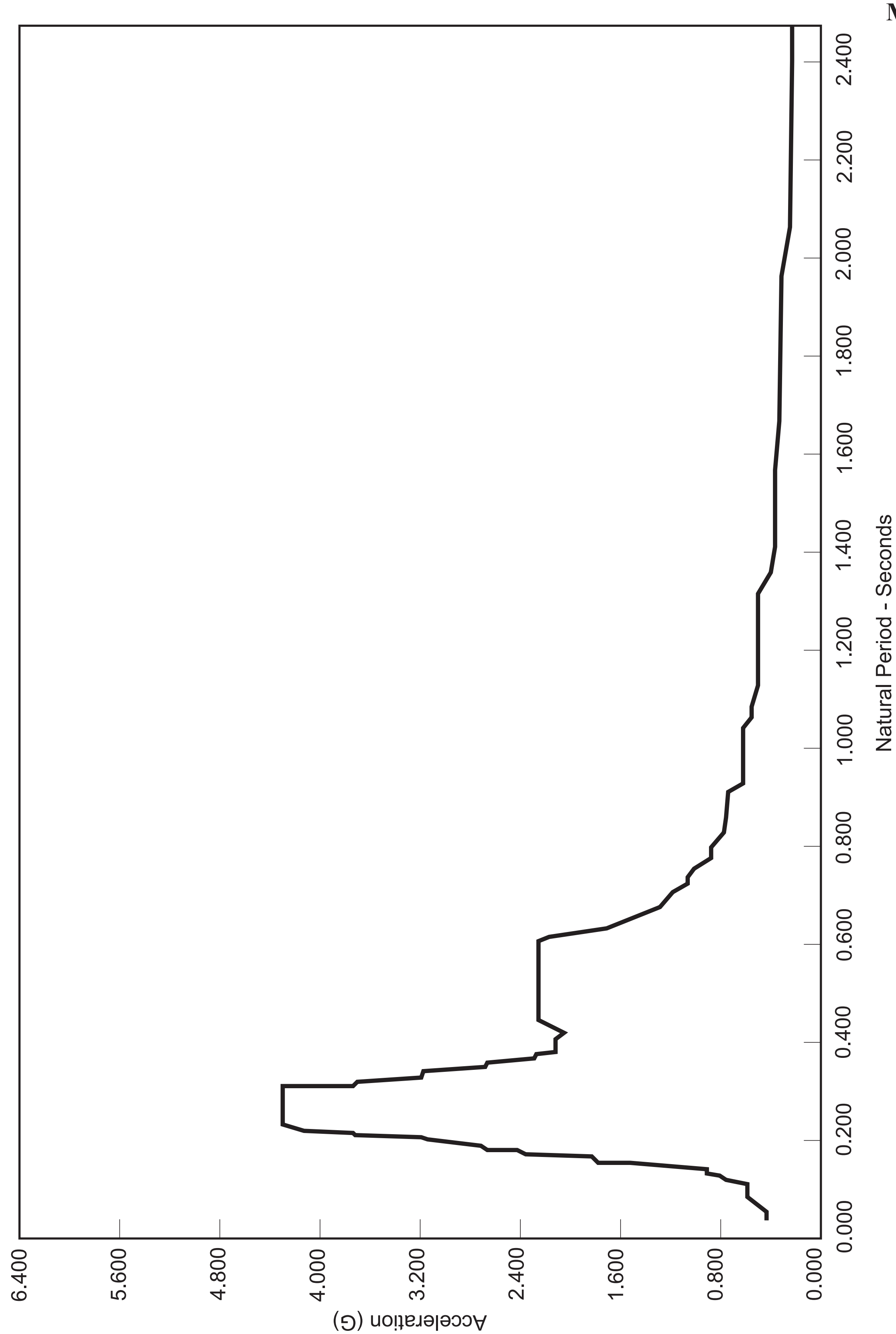
Columbia Generating Station Final Safety Analysis Report	Reactor Building Refueling Floor Response Spectrum - Safe Shutdown Earthquake, 0.01 Damping, Vertical		
	Draw. No. 010126.06	Rev.	Figure 3.7-17



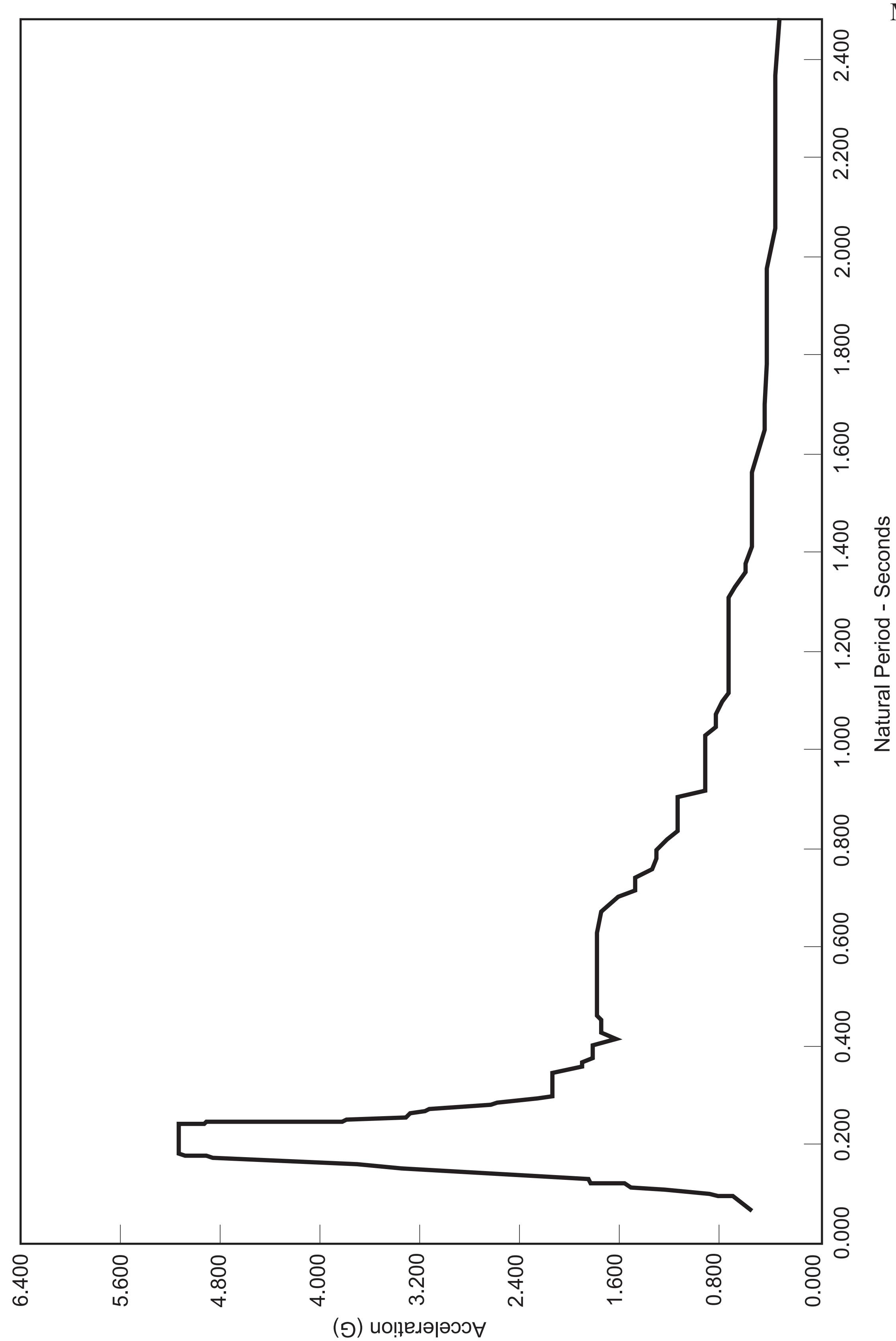
Columbia Generating Station
Final Safety Analysis Report

Reactor Building Mat Response Spectrum - Operating Basis Earthquake, 0.005 Damping, Horizontal NS and EW

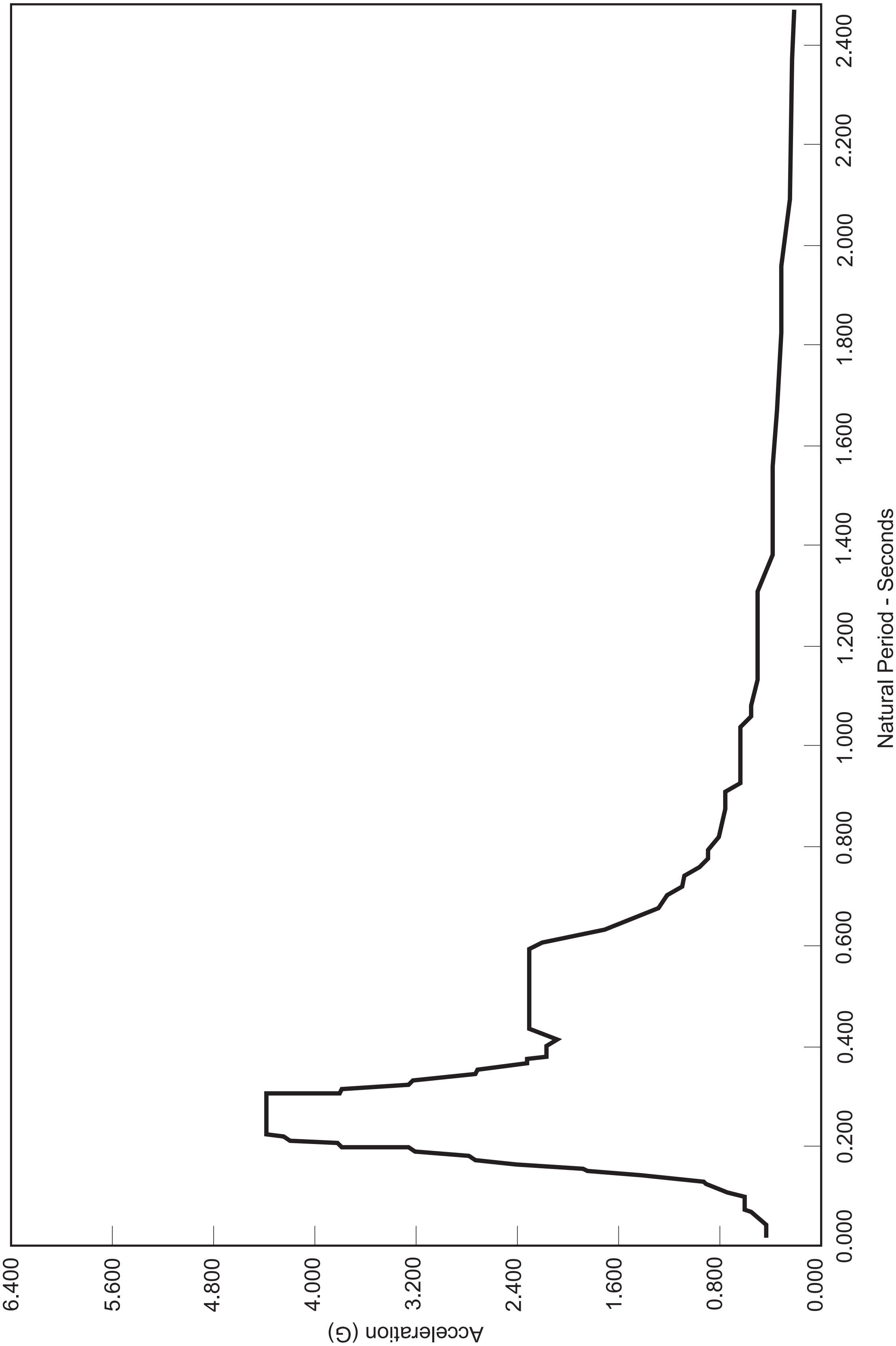
Draw. No. 010126.05	Rev.	Figure 3.7-18



Columbia Generating Station **Final Safety Analysis Report**	**Reactor Building Mat Response Spectrum - Operating Basis Earthquake, 0.005 Damping, Vertical**
	Draw. No. **010126.07** | Rev. | Figure **3.7-19**



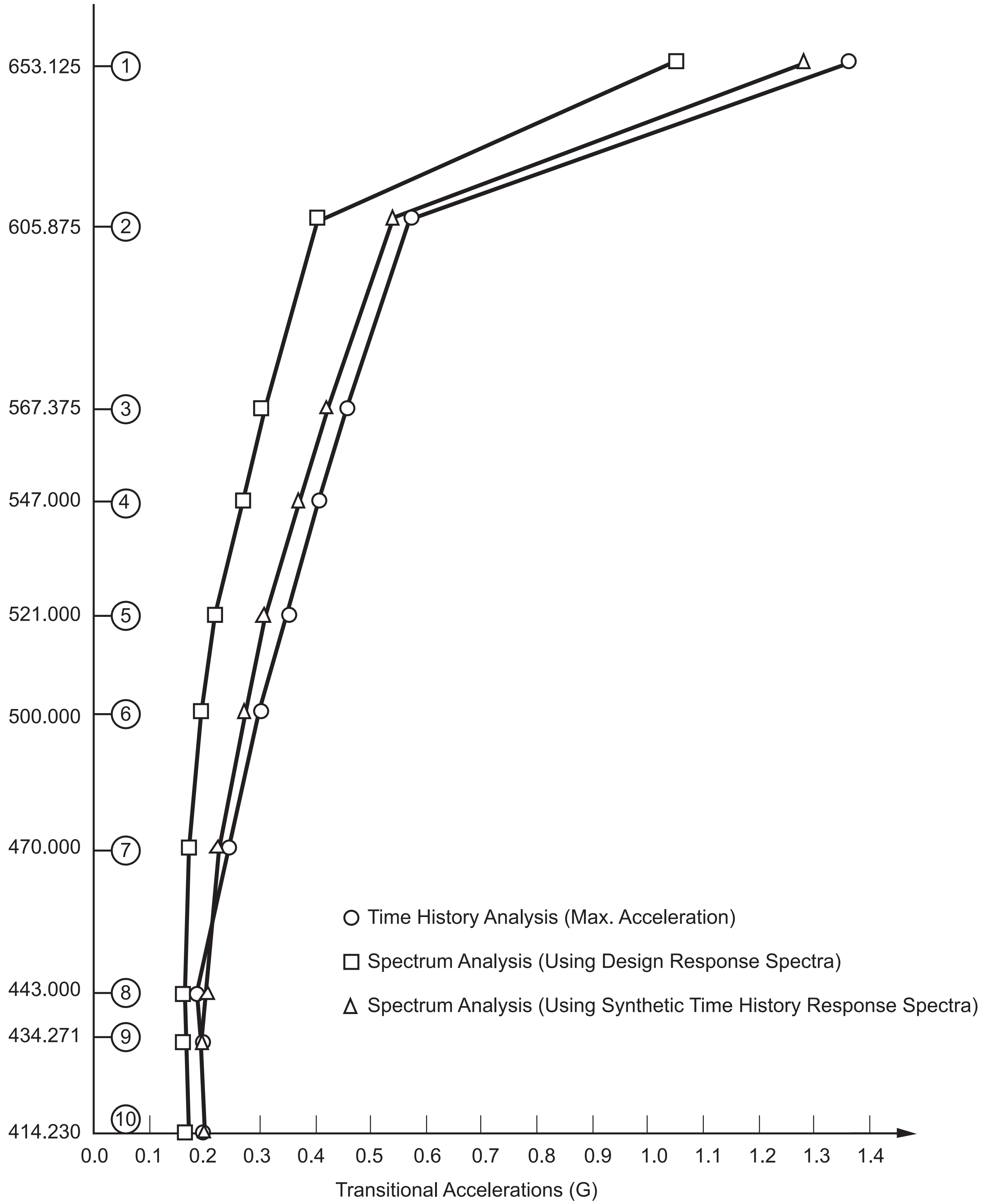
Columbia Generating Station **Final Safety Analysis Report**	**Reactor Building Mat Response Spectrum - Safe Shutdown Earthquake, 0.01 Damping, Horizontal NS and EW**
	Draw. No. **010126.08** | Rev. | Figure **3.7-20**



Columbia Generating Station
Final Safety Analysis Report

Reactor Building Mat Response Spectrum - Safe Shutdown Earthquake, 0.01 Damping, Vertical

Draw. No. 010126.09	Rev.	Figure 3.7-21



Columbia Generating Station **Final Safety Analysis Report**	**Reactor Building-Seismic Analysis** **Comparison of Responses**
	Draw. No. 910402.17 · Rev. · Figure 3.7-22

3.8 DESIGN OF SEISMIC CATEGORY I STRUCTURES

The nuclear steam supply system (NSSS) is housed in a steel primary containment vessel that has the ability to function as an independent structure. The primary containment vessel is housed in a reactor building also containing fuel storage and handling equipment and equipped with an overhead crane. The primary containment vessel for the NSSS is designed to confine the effects of a postulated nuclear accident and to limit the discharge of radioactive products to within levels specified in 10 CFR Part 50.67.

The Seismic Category I structures, which are part of a complex of buildings in close proximity to each other, are the reactor building, the diesel generator building, and portions of the radwaste and control building. The portions of the radwaste and control building which are Seismic Category I are the radwaste area and the control room tower. The radwaste area encompasses the foundation mat, the walls and all internal structures from the top of mat el. 437 ft to and including the reinforced-concrete slab at el. 467 ft. The control room tower consists of the vertical portion of the building encompassing the area of the control room.

The turbine generator building is a modified non-Category I seismic structure which is dynamically analyzed and designed to withstand the effects of an safe shutdown earthquake (SSE) and maintain its structural integrity thus providing adequate protection for the portions of the steam system designed as Seismic Category I, as defined in Regulatory Guide 1.29, Revision 3.

The reactor building consists of a dual barrier: the steel primary containment vessel and the reactor building which provides secondary containment. The primary containment vessel contains the drywell, suppression chamber, structural floor separating the drywell from the suppression chamber, sacrificial shield wall (SSW), and reactor pedestal. The reactor building secondary containment encloses the biological shield wall, spent fuel storage pool, dryer-separator pool, and the reactor well pool.

Seismic Category I structures that are not part of the building complex are two adjacent spray ponds, each provided with an integrally constructed standby service water pump house, and the condensate storage tank retaining area.

The major structures in the reactor building and principal dimensions are shown in Figures 3.8-1, 3.8-2, and 3.7-13. The seismic analysis methods and models used to obtain seismic loads and floor response spectra, soil/structure interaction, the damping values used, and the seismic input at the base of the buildings are discussed in Section 3.7.

Although the reactor building, radwaste and control building, diesel generator building, and turbine generator building are in close proximity to each other, they are supported on separate foundation mats. No interaction between these mats was considered. Each mat and its

superstructure was analyzed separately for soil-structure interaction due to horizontal and vertical earthquake motions.

The seismic design of the buildings was based on a parametric study for a wide range of soil properties. Thus, uncertainties in the soil properties and frequencies are adequately accounted for in the design.

Each of the Seismic Category I structures in the complex is physically separated from the other, above and below grade, in order to accommodate differential seismic displacements and thermal expansion. Abutting walls between buildings, which are separated by only a few inches, are designed to resist wind forces and tornado effects including tornado-generated missiles and pressure drop. The tornado and wind forces on roofs and external walls were computed as indicated in Section 3.3.

The seismic analysis of the reactor building included interaction between soil, foundation mat, and superstructures using lumped mass-spring, discrete models as described in Section 3.7. Conservative soil and structure damping parameters are included in the model. The response spectrum and time-history methods of analysis were used to determine the responses of the structures.

Time histories of displacements and accelerations at the top of the reactor building mat and at the lumped masses, which usually represent floor slabs, are the main output of the time-history analysis. This output was used as input for time-history analyses, as required, of the main components of the NSSS (the reactor vessel and internals) as well as to generate floor response spectra which were used in the seismic analyses and design of safety-related systems, equipment, and components housed in the Seismic Category I structures. In addition, moments and shears were obtained for comparison with results obtained from response spectrum analyses.

Response spectrum analyses were used for determining shears and moments for the final design of each of the structural components of the reactor building. These are discussed in Sections 3.8.2, 3.8.3, 3.8.4, and 3.8.5.

3.8.1 CONCRETE CONTAINMENT VESSEL (Not Applicable to CGS)

Columbia Generating Station (CGS) has a steel containment vessel (see Section 3.8.2).

3.8.2 STEEL CONTAINMENT VESSEL (ASME Class MC Components)

The capability of the primary steel containment vessel to withstand the hydrodynamic effects resulting from actuation of safety/relief valves (SRV) and specified loads associated with postulated loss-of-coolant accidents (LOCAs), and the applicable modifications were addressed in the Plant Design Assessment Report (DAR) for SRV and LOCA Loads. See Appendix 3A

which identifies the loads and effects which are most important to the design of the CGS plant and describes the CGS plant capability with respect to the hydrodynamic loading phenomena during SRV actuations and postulated LOCA events.

3.8.2.1 Description of the Primary Containment Vessel

The basic safety objective of the primary containment system is to provide the capability in the event of a postulated LOCA of limiting the release of fission products to the plant site environs so that offsite doses are in compliance with the limits specified in 10 CFR Part 50.67.

To meet the basic safety objective, several contributory objectives are achieved by the system or one or more of its components, including

a. Capability to withstand the peak transient pressures and temperatures which could occur due to postulated design basis LOCA; i.e., a mechanical failure of the reactor primary system equivalent to the circumferential rupture of one of the reactor coolant recirculation system pipes,

b. Capability to maintain the functional integrity of the primary containment indefinitely after the postulated design basis LOCA,

c. The primary containment design permits filling the primary containment system drywell with water to a level above the reactor core,

d. The primary containment system is protected against missiles from internal or external sources and excessive motion of pipes, which could directly or indirectly endanger the integrity of the primary containment,

e. Capability to withstand fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment,

f. Capability of limiting leakage during and following the postulated design basis accident to values less than leakage rates which would result in offsite doses greater than the reference doses in 10 CFR Part 50.67,

g. Means of conducting the flow from postulated pipe ruptures, including the design basis rupture of a recirculation line to the pressure suppression pool, to rapidly condense the steam portions of the flow so that the peak transient pressure is less than containment design pressure, to distribute such flow uniformly throughout the pool and to limit pressure differentials between the drywell and the pressure chamber during the various postaccident cooling modes, and

h. Capability for rapid isolation of the primary containment to provide a containment barrier sufficient to maintain leakage within permissible limits.

The primary containment vessel is a free-standing steel pressure vessel. It utilizes the pressure suppression technique through the Mark II over-under configuration. The primary containment vessel and its appurtenances comply with the requirement of the ASME Code, Section III, Subsection NE-Class MC Components, 1971 Edition through Summer 1972 Addenda. It is designed to resist all normal operating loads, loads resulting from the postulated design basis accident as well as those loads associated with the operating basis earthquake (OBE) and SSE. The design also accounts for stresses induced by thermal expansion. The drywell floor which divides the drywell and suppression chamber is a reinforced-concrete slab supported by steel beams and concrete columns. The drywell floor to primary containment vessel gap is closed off by means of a floor seal shown in Figure 3.8-3. This configuration permits unrestrained expansion of the containment shell under differential thermal expansion and pressure loadings. The containment vessel is enclosed in a reinforced-concrete biological shield wall for shielding purposes and is separated from the reinforced concrete by an annulus of compressible isolation material, approximately 2 in. thick. The concrete wall thickness is governed by shielding requirements but also serves as a support for the reactor building floors. Shielding over the top of the drywell is provided by removable, segmented, reinforced-concrete shield plugs. The drywell is located directly above the wetwell. The drywell configuration is basically a frustum of a cone with removable ellipsoidal top closure head. The suppression chamber (wetwell) is cylindrical with an ellipsoidal base. The primary containment vessel is anchored to the concrete mat foundation. The bottom of the suppression chamber is lined on the inside with reinforced concrete. The concrete mat foundation under the suppression chamber is a common foundation supporting the steel primary containment vessel, including all equipment and structures therein, and the reactor building of which the primary containment vessel is a part.

The drywell floor serves as a pressure barrier between the drywell and suppression chamber. The top closure head of the drywell is bolted to a steel flange attached to the top of the containment vessel. The drywell houses the reactor vessel and its associated primary system. The primary function of the drywell is to contain the effects (i.e., mass and radiation) of a LOCA, and to direct the steam released from a primary system pipe break into the suppression chamber pool to limit the total pressure rise during a LOCA.

Under normal operating condition (normal condition) a fatigue analysis is performed in accordance with the requirements of ASME Code Section III.

Under emergency condition, the jet impingement force of 534 kips as outlined in Section 3.8.2.3 might cause local yielding of the drywell shell. An analysis (plastic analysis in accordance with the requirements of the ASME Code Section III) demonstrates that rupture will not occur. Local deformation caused by the jet impingement force does not affect the leak tightness of the containment vessel.

The analysis of the jet impingement effects on the primary containment vessel (Reference 3.8-21) is summarized as follows:

a. Phase 1 - The conical region of the containment shell was modeled and a general shell of revolution analysis was performed using the HYBOS computer program (Reference 3.8-22).

Two critical locations were chosen for independent application of the jet force. One, located approximately in the middle of two box ring stiffeners, is a logical candidate for maximum deflection; the second, located on the thinnest nearly adjacent to a stiffener, is a location where largest curvatures could occur if the shell contacts the concrete biological shield wall spaced 2 in. from the shell.

Since the impinged area (429 in.^{2}) subtends only a small arc of the total periphery, a Fourier harmonic expansion of 11 terms is used to represent the jet forces of 534 kips.

The response to gravity, static seismic, and design pressure loads were also computed. The results of the most severe combinations of the loads (Appendixes C and D of Reference 3.8-21) show that the shell will contact the concrete for either candidate jet force location; consequently the elastic analysis is not valid in the immediate area of the jet load. The largest computer stresses were found for the second location and exceeded yield; therefore, an elastic-plastic analysis was next performed for that critical region;

b. Phase II - A local finite element elastic-plastic model was analyzed using the DYPLAS computer program which is capable of treating nonlinear inelastic materials.

The boundaries of this model as shown in Figure 3.8-4 are structurally remote from the jet-impinged area, indicated by cross-hatching. The displacements from the general shell analysis, therefore, were used as displacement boundary conditions. Inelastic deformation, strains, and stresses were computed for all finite elements during selected steps of load applications; and

c. Stress evaluation was based on the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NE, Class MC Components, NE-3131.2 and Appendix F, 1974 edition.

3.1 Code Requirements

$P_m < \sigma_m$

$P_L < 1.5\ \sigma_m$

where

$\sigma m = 0.85 \times 0.7 \times S_u = 0.595\ S_u$

From Table I-1.1, for SA-516, Grade 70 Steel:

S_u - 70,000 psi at 345°F

Therefore:

σm - 41,650 psi

To meet code requirements:

$P_m < 41,650$ psi

$P_L < 62,475$ psi

3.2 Stress Evaluation

From Appendix J of Reference 3.8-21

$(P_m)max = 29,943$ psi $< 41,650$ psi

which occurs on upper edge of model (Figure 3.8-4)

$(P_L)max = 36,305$ psi $< 62,475$ psi

which occurs on the lower portion of the impinged region (Figure 3.8-4)

Therefore code requirements are met.

The ductility ratio is defined as the maximum response of an elasto-plastic structure to a prescribed loading function divided by the response of the same structure to the load at incipient plasticity. The maximum radial displacement at incipient plasticity was computed and is shown in computer Appendix I to Reference 3.8-21 as 0.4335 in. The maximum radial deflection of 2.0 in. was the spacing between shell and shield.

Thus the ductility ratio might be considered to be

$$2.00/0.4335 = 4.6$$

However, since the maximum deflection of the shell is limited by contact with the concrete of the biological shield; the ductility ratio here may not be as meaningful a design parameter as it is in other cases.

A more significant measure of structural integrity against cracking and ultimate leakage may be the most severe local principal cumulative strain.

The most severely strained metal lies on the outside surface of the vessel shell near node 67 as shown in Figure 3.8-4. This strain is the maximum inelastic cumulative strain based on a computed nonlinear strain distribution through the vessel wall. Its value is 4%.

Figure 3.8-5 shows the minimum elongation to failure of ASTM A516 Grade 70 steel as a function of temperature. Over the range of 70°F to 350°F this has a least value of 14.8%.

An estimate of the factor of safety may then be said to be

$$\text{F. S.} = \frac{14.8}{4.0} = 3.7$$

Although failure strains in complex strain fields do not correlate precisely with results of one-dimensional ductility tests, the factor of safety, as computed above, is deemed ample.

The criteria used in the seismic design of the containment system is based on the responses of the containment system to earthquake excitation. The responses are derived from the analysis of a mathematical model developed to represent the containment vessel. Section 3.7 outlines methods of analysis, modeling techniques, the seismic input, and the soil-structure interaction effects.

All ferrous materials of plates, forgings, castings, and pipes for ASME Code, Class MC with thickness greater than 5/8 in. are Charpy V-notch impact tested at 0°F in accordance with the ASME Code, Section III, Paragraph NE-2300. Materials for ASME Code, Class 1 components are impact tested in accordance with NB-2300. The drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F.

The principal containment design parameters are listed in Table 3.8-1.

The physical dimensions of the steel primary containment vessel are

a. The diameter of the cylindrical portion at the base of the cone is approximately 86 ft,

b. The diameter at the top of the cone is approximately 39.5 ft and then narrows to 32 ft to carry the removable head,

c. Ellipsoidal bottom head with a ratio of 2:1 has an inside height of approximately 21.5 ft,

d. The removable ellipsoidal top closure head has an inside height of approximately 15.5 ft,

e. The drywell shell height is approximately 99 ft,

f. The suppression chamber shell height is approximately 72 ft, and

g. Overall shell height is approximately 171 ft.

The primary containment vessel shell plate thicknesses vary. Typical thicknesses are as follows:

a. Bottom ellipsoidal head: from 7/8 in. to 1.5 in.,
b. The suppression chamber cylinder: from 1-5/16 in. to 1.5 in.,
c. The drywell conical section: from 0.75 in. to 1.5 in., and
d. The removable top ellipsoidal head: 15/16 in.

Material thicknesses meet requirements of the ASME Code Section III, Paragraphs NE-3133 and NE-3324.

The primary containment vessel is reinforced with internal vertical and horizontal stiffeners to satisfy design requirements of the various loading combinations and conditions. Fully circumferential rings, attached to the inside face of the primary containment vessel are furnished at el. 516 ft 6 in. and 542 ft 7.25 in. The basic function of these rings is to support pipe whip protection framework and to adequately distribute pipe whip loading into the vessel.

The method used to ensure that there is an adequate clearance between the steel primary containment vessel and the concrete biological shield wall and which will account for differential thermal expansion is illustrated in Figure 3.8-2 and described as follows:

a. A system consisting of approximately 2-in.-thick polyurethane flexible foam sheets, butted at their joints and cemented directly to the containment shell, is

used. The exterior surface sealed with laminated fiberglass reinforced polyester, epoxy jointed, panels which are bonded to the polyurethane sheets;

b. Steel anchor fasteners are attached to the fiberglass reinforced polyester panels, and used for anchoring the inner end of the form ties which extend to the forms on the outside face of the concrete biological shield wall. These fasteners are mounted on the fiberglass reinforced plastic prior to bonding of the flexible foam;

c. The fiberglass panel joints are taped and filled with epoxy so that the entire assembly forms a shell around the containment vessel with the polyurethane material set against and bonded panels thereby are permanent inner forms for the pouring of the concrete structure;

d. Drywell penetrations which extend from the containment vessel shell through the concrete biological wall, are surrounded with concentric sleeves. These pipe sleeves are joined to the fiberglass shell using fiberglass tape and epoxy resins. This technique similarly provides a form for the concrete and maintains adequate clearance between the penetrations and the sleeves to accommodate thermal expansion;

e. The polyurethane foam material is chosen for its resistance to the environmental conditions likely to exist during its service life. Although normal inservice temperature ranges from 95°F to 135°F (average) and 150°F (local), the polyurethane which is used is capable of withstanding temperatures in excess of the LOCA temperature of 340°F. Furthermore, this material is self extinguishing in accordance with ASTM D-1692;

f. The sizing of the expansion gap in which the foam sheet is placed is based on an ultimate steel shell temperature of 340°F and an internal pressure of 45 psig following a postulated reactor LOCA. The external compressive stress applied to the containment vessel by the polyurethane foam, due to vessel thermal expansion from ambient conditions, ranges from 1.2 to 1.5 psi at normal operating and LOCA temperatures. At these temperatures the stress-strain curve for the polyurethane foam is nearly linear between 5% and 60% compression, ranging from approximately 1.2 psi at 5% compression to 1.8 psi at 60% compression. These properties are not significantly affected by the level of radiation exposure received over the plant life; and

g. The above design, materials, and construction of the containment vessel expansion gap provides sufficient space for thermal expansion of the steel containment vessel shell. Moreover, this method of construction prevents either concrete or other foreign material from entering and/or reducing the gap. Local

stress areas are thereby prevented, and the primary containment system is capable of accommodating both normal operating as well as postulate accident conditions.

The steel primary containment vessel, including all penetrations and welded attachments, is designed to act as a structural component within the reactor building as described in Section 3.8.2.4. The general configuration, and elevations are shown in Figures 3.8-1 and 3.8-2. The primary containment vessel is provided with two concentric circular skirts on the bottom ellipsoidal head integral with the vessel. The skirts are anchor bolted to the concrete foundation mat. The bottom ellipsoidal head and the upper portion of the skirts connected to the head are considered part of containment pressure boundary in accordance with the ASME Code Section III. The lower portion of the skirts follow the requirements of the American Institute of Steel Construction (AISC) Code. The skirts are backed up by concrete fill. The concrete fill and the concrete foundation mat discussed in Section 3.8.5 are not part of the containment vessel.

3.8.2.1.1 Description of Penetrations

Penetrations through the primary containment vessel are as follows.

3.8.2.1.1.1 Pipe Penetrations. Two general types of pipe penetrations are provided. The two types differ depending on whether the penetration is subject to a hot or cold operational environment. The cold penetrations pass through the steel primary containment vessel and are welded directly to it. The piping is normally welded directly to the penetration nozzles. The piping design includes the effects of thermal motion of the containment shell at the penetration connections.

The hot penetrations and multiple piping penetrations do not come in direct contact with the steel shell of the primary containment vessel. These penetrations pass through vessel shell nozzles which are welded to the steel shell of the primary containment vessel and function as thermal sleeves. Containment closure is accomplished by means of closure plates or flued head fittings, welded to the penetration nozzle and the piping at a suitable distance outside the containment shell. Detailed descriptions of pipe penetrations are given in Section 3.8.6.

3.8.2.1.1.2 Electrical Penetrations. Containment electrical penetrations are designed to safely accommodate all of the electrical requirements within the containment boundary. These are functionally grouped into low voltage power and control cable penetrations assemblies, medium voltage power cable penetration assemblies, signal cable penetration assemblies and thermocouple cable penetration assemblies. The medium voltage power cable electrical penetrations are canister type assemblies sized to be inserted into the containment vessel penetration nozzles. All other electrical penetrations are a unitized header plate assembly attached to the outboard end of the containment vessel penetration nozzles. Detailed descriptions of electrical penetrations are given in Section 3.8.6.

3.8.2.1.1.3 Traversing In-Core Probe Penetrations. Five traversing in-core probe (TIP) guide tubes pass from the reactor building to the drywell through the primary containment vessel. The penetrations are a Type 2 (see Section 3.8.6.1.2) piping penetration modified with a welding neck flange attached outside the containment. This flange is itself modified with dual concentric O-ring grooves machined into the face, which retain elastomeric O-rings. To this is bolted a blank flange which has been drilled for both a between-O-ring test port and a central hole in which an instrument tubing "bulkhead union" fitting is retained. This single penetration point is sealed by seal welding between the bulkhead union and the blank flange. The TIP guide tubes are attached to both sides of their respective bulkhead unions by flare fittings. These penetrations are also discussed in Section 3.8.6.1.2.

3.8.2.1.1.4 Personnel Access Lock and the (Combined) Equipment Hatch and Control Rod Drive Removal Hatch. The drywell has one manually operated personnel air lock. This air lock consists of a cylindrical shell with two doors, one at each end of the shell. The cylindrical shell is approximately 8 ft 10.5 in. in diameter which is sufficient to provide 6 ft 8 in. high by 3 ft 4 in. wide door openings above the floor. The minimum clear horizontal distance not impaired by the door swing is 5 ft. Each door has double compressible seals with an air space between them so that each door may be individually tested. Each door is hinged and swings toward the drywell.

The air lock doors are designed so as to permit either door to be operated from inside the drywell, inside the air lock, or outside the drywell. In addition, the air lock doors are interlocked to ensure that at least one door is locked when primary containment integrity is required. Signals and controls indicating the status of the doors are provided locally. The locking mechanisms are designed so that tight seals are maintained when the doors are subject to either the design internal or external pressure. A mechanical override is provided to permit temporary bypassing of the door interlock system to permit opening both doors under proper authorization. Quick acting equalizer valves are provided to equalize the pressure in the air lock when personnel enter or leave the primary containment vessel.

The drywell has one equipment removal hatch. The equipment hatch cover is dished and has steel stiffeners. The hatch cover is bolted to a flanged steel sleeve welded to the primary containment vessel shell such that the hatch cover can be removed and reinstalled from outside the drywell. The equipment hatch and cover is entirely supported by the steel containment vessel. Double compressible seals with an air space between them are used to permit leak testing at any time. The inside diameter of the equipment hatch is approximately 12 ft 6 in. which provides a minimum clearance above the floor at the hatch of 10 ft 1.5 in. high by 7 ft 0 in. wide.

Included within the equipment hatch cover is a control rod drive (CRD) removal hatch with its hinged cover. This hatch is provided with leak-testable, double-gasketed seals. The inside

diameter of this hatch is approximately 1 ft 11 in. in diameter which provides a minimum clearance of 11 in. high by 1 ft 4 in. wide at the hatch.

The personnel access lock and the equipment hatch extend radially outward across the annular gap of compressible isolation material and through the biological shield wall and are supported by the primary containment vessel only.

Both the personnel air lock and the equipment removal hatch are designed to withstand the normal environmental conditions which may prevail during normal plant operation and to maintain their functional integrity during a postulated LOCA. The design meets the requirements of the ASME Code Section III, Subsection NE, Class MC Components.

3.8.2.1.1.5 Pressure Suppression Chamber Access Hatch. Access to the pressure suppression chamber is provided at one location in the cylindrical well of the chamber approximately 7 ft 6 in. above the suppression pool operating water level. This access hatch is approximately 3 ft 5 in. in diameter, extends radially outward, is supported by the vessel and has a leak-testable, double-gasketed, bolted cover which is normally closed and is opened only when primary containment is not required. The minimum clearance at the hatch is 2 ft 2.75 in. wide by 2 ft 5.5 in. high. The design meets the requirements of the ASME Code Section III, Subsection NE, Class MC Components.

3.8.2.1.1.6 Access for Refueling Operations. The drywell containment head is removed during refueling operations. This head is held in place by bolts and is sealed with a double seal. It is bolted closed when primary containment is required and is opened only when primary containment is not required. The gasket seal is capable of limiting the leakage to below the design rate and is capable of being independently tested.

3.8.2.1.2 Description of Crane Girder (Not Applicable to CGS)

CGS does not have a crane girder inside containment.

3.8.2.1.3 Description of Vacuum Relief System

See Figure 9.4-8 for an illustration of the vacuum relief system described below.

Three 24-in. reactor building-to-wetwell vacuum relief lines, each containing a 24-in. vacuum breaker valve and an automatic air-operated butterfly valve, are provided between the reactor building and the suppression chamber. These valves prevent excessive vacuum from developing in the primary containment vessel from such causes as inadvertent containment spray actuation.

Each butterfly valve is equipped with a spring-to-open, air-to-close operator which, during normal plant operation is maintained in a closed position by means of a control air supply

through a three-way solenoid pilot valve. The plant control air supply to the valve is backed up by a Quality Class I air supply system, consisting of an accumulator pressurized by the plant control air system and an independent 10 nitrogen bottle manifold located in the vehicle air lock (railroad bay) which will automatically maintain accumulator pressure on loss of air supply from the control air system (see Figure 3.8-7). On venting of air from the air-operators, the spring- actuated butterfly valves open. Venting is accomplished through remote manual deenergization of the solenoid pilot valve or by a signal from the differential pressure switch which deenergizes the solenoid pilot valve when the secondary containment atmospheric pressure is more than 0.5 psi higher (analytical limit) than the suppression chamber atmospheric pressure.

Two limit switches, wired to indicator lights in the control room, are provided with each butterfly valve for position indication. One switch actuates when the valve is fully open and provides an alarm and "open" visual indication in the control room. The other switch actuates when the valve is fully closed and provides a "closed" visual indication in the control room (see Figure 3.8-8).

In series with each butterfly valve is a single disk check valve. The disk is maintained in the closed position during normal operation by means of a spring-actuated lever arm and magnets embedded in the periphery of the disk. The magnetic and spring forces are overcome, and the disk opens when the pressure differential across the valve is within the range of 0.10 to 0.35 psi. The disk is fully open when the pressure difference is 0.5 psi. In addition, pneumatic actuators are provided for remote operation of the disk. Compressed air is supplied by the plant control air system through the pneumatic operator Quality Class I accumulator and backup nitrogen supply as shown in Figure 3.8-7. Each disk pneumatic operator consists of two air cylinders, one to open and one to close the disk. Each air cylinder is actuated through energization of a three-way solenoid pilot valve. The two solenoid pilot valves associated with each disk are operated by a remote manual switch in the control room. During normal operation the remote manual switches are in the neutral position and the solenoids are deenergized. Each valve disk is provided with contact probe sensors for position indication. These sensors are wired to indicator lights in the control room to provide open and closed position indication. An additional sensor is also wired to a light in the control room that indicates when the disk is fully open.

Each reactor building to wetwell vacuum breaker is visually inspected at least once every 30 months.

Nine 24-in. wetwell-to-drywell vacuum relief valves attached to the downcomers in the suppression chamber are provided to return noncondensables from the wetwell to the drywell to prevent too large an upward pressure differential across the diaphragm floor after a LOCA.

Each wetwell-to-drywell vacuum relief valve assembly consists of two discs and seats which operate independently. The operation, controls, and position indication for each disc is as

described above for the single disc check valves. During normal plant operation the control air supply line to these valve assemblies is isolated. Also, residual pressure is vented from the line following vacuum relief valve testing to prevent inadvertent valve opening.

The vacuum breaker valves are sized to ensure that following design conditions are not exceeded.

a. The drywell internal design pressure of 2.0 psi below reactor building pressure,

b. The suppression chamber internal design pressure of 2.0 psi below reactor building pressure, and

c. The upward design pressure difference across the diaphragm floor of 6.4 psi.

The design evaluation for the vacuum relief is discussed in Section 6.2.1.1.4.

The vacuum relief valves are constructed to ASME Section III, Subsection NC for Class 2 components, 1974 Edition through the Summer 1975 Addenda.

Electrical systems associated with the control and position indication for the reactor building to wetwell vacuum breaker valves and the drywell-to-wetwell vacuum breaker valves are not Class 1E since electrical failure or malfunction will not prevent operation or cause inadvertent actuation under postulated accident conditions.

3.8.2.1.4 Containment Pressure Boundaries

The primary containment pressure boundaries for the steel primary containment vessel consist of those defined in Subarticle NE-1130 of the ASME Code Section III and the additional boundaries listed in the following:

a. The steel primary containment vessel shell including the top and bottom heads, and, as defined in the ASME Code Section III, Paragraphs NE-3364 and NE-4431, an upper portion of the skirts supporting the vessel, and

b. The attachment welds fastening pipe whip protection support rings, beam supports, and pads to the vessel for purposes of supporting piping support members, walkways, platforms, monorails, brackets, or other members to the containment vessel.

3.8.2.1.5 Primary Containment Environmental Conditions

The primary containment is designed to operate during all environmental conditions found in Section 3.11. See Table 3.8-2 for specific design conditions.

3.8.2.2 Applicable Codes, Standards, Specifications, and Regulatory Guides

3.8.2.2.1 Codes, Standards, and Specifications

The following describe the applicable codes, standards, specifications, code classification, and code compliance for the steel primary containment vessel.

a. The following sections of the ASME B&PV Code, 1971 Edition, including all addenda through Summer 1972 apply for the steel primary containment vessel.

   Section II, Material Specifications
   Part A - Ferrous
   Part B - Welding Filler Metals

   Section III, Nuclear Power Plant Components
   Subsection NE, Class MC Components
   Section V, Nondestructive Examination
   Section IX, Welding Qualifications;

b. The AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, February 12, 1969, applies for the steel construction beyond the boundaries established for the steel containment vessel;

c. Steel Structures Painting Manual, Volume 2, Systems and Specifications, 1964 Edition with the 1968 Supplement and the January 1971 Editorial changes.

   Specification SSPC-SP-6, SP-8 and SP-10;

d. Applicable ASTM and AWS Material Standard Specifications permitted by Article NE-2000 of III;

e. Applicable ASTM Standard Specifications for nondestructive methods of examination referenced in Article X-3000 of Section III of the ASME Code;

f. Plant Design Specification 2808-213; and

g. National Electrical Code.

3.8.2.2.2 Code Classification

The steel primary containment vessel is classified Class MC in accordance with Subarticle NA-2130, Section III of the ASME Code, 1971 Edition through the Summer 1972 Addenda.

3.8.2.2.3 Code Compliance

3.8.2.2.3.1 Containment Vessel. The steel shell and the top and bottom heads of the steel primary containment vessel, including all penetrations and attachments within the boundaries defined in Section 3.8.2.1.4, are designed and constructed in accordance with Subsection NE, Class MC Components, including the requirements for quality assurance of the Article NA-4000, and inspection requirements of Article NA-5000 of Section III of the ASME Code.

3.8.2.2.3.2 Code Stamp. The steel primary containment vessel is ASME Code stamped in accordance with requirements of the Code applicable to Class MC containment vessels.

3.8.2.2.3.3 Exceptions. No exceptions are taken to the requirements of Section III of the ASME Code for Class MC containment vessels.

3.8.2.2.3.4 Attachments. Structural steel attachments beyond the boundaries established for the steel primary containment vessel are designed and constructed according to the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Building, February 12, 1969, where applicable. Nonpressure vessel elements such as catwalks and interior beam connections with hatch floors are designed in accordance with the AISC Code. Stress intensity limits are in accordance with the allowable permitted by the AISC Code with the exception that, for loading combinations including OBE, no increase in allowable stress will be permitted.

The inner and outer skirts for the support of the primary containment vessel are designed in accordance with Subarticle NE-3100 of Section III of the ASME Code.

Supports for pipe whip guide rings are designed as described in Section 3.6.

3.8.2.2.4 Regulatory Guides

The following regulatory guides related to the primary containment vessel are applicable to CGS. Their implementation is discussed in Section 1.8.

a. Regulatory Guide 1.7, Rev. 1 - Control of Combustible Gas Concentrations in Containment Following a LOCA,

b. Regulatory Guide 1.11, Rev. 0 - Instrument Lines Penetrating Primary Reactor Containment,

c. Regulatory Guide 1.28, Rev. 0 - Quality Assurance Requirements (Design and Construction),

d. Regulatory Guide 1.29, Rev. 3 - Seismic Design Classification,

e. Regulatory Guide 1.46, Rev. 0 - Protection Against Pipe Whip Inside Containment,

f. Regulatory Guide 1.57, Rev. 0 - Design Limits and Loading Combinations for Metal Primary Containment System Components,

g. Regulatory Guide 1.63, Rev. 0 - Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants,

h. Regulatory Guide 1.84 - Code Case Acceptability - ASME Section III Design and Fabrication, and

i. Regulatory Guide 1.85 - Code Case Acceptability - ASME Section III Materials.

3.8.2.3 Loads and Loading Combinations

The primary containment vessel is designed to withstand forces due to

a. Dead load, including permanent equipment loads and hydrostatic loads,

b. Live loads, including movable equipment loads and other loads varying in intensity and occurrence,

c. Thermal effects and loads during startup, normal operating, and shutdown conditions, based on the most critical transient or steady-state conditions,

d. Earthquakes,

e. Wind during construction, and

f. Pressure and temperature effects, jet impingement, and missile impact all due to pipe break accident.

The vessel design includes analyses in the vicinity of the primary containment vessel penetrations for the penetration load combinations given in Section 3.8.6.3.

3.8.2.3.1 Design Pressures and Temperatures

a. Pressure suppression chamber

1.	Internal design pressure (LOCA pressure)	45 psig

	2.	External design pressure (due to negative internal pressure)	2.0 psig
	3.	Design temperature	275°F
b.	Drywell		
	1.	Internal design pressure (LOCA pressure)	45 psig
	2.	External design pressure (due to negative internal pressure)	2.0 psig
	3.	Design temperature	340°F
c.	Pressure suppression chamber and drywell		
	1.	Pneumatic over pressure test (115% of 45 psig)	51.8 psig at ambient temperature
	2.	Initial leak rate test leakage rate of 0.5% of air per any 24 hour period	Maximum permitted total weight of contained at 38 psig test pressure.
d.	Differential design pressure (downward on drywell floor following a LOCA)		25 psid
e.	Differential design pressure (upward on drywell floor following a LOCA)		6.4 psid
f.	Lowest service metal temperature		30°F

3.8.2.3.2 Operating Pressure and Temperature

See Table 3.8-1 for normal operating pressures and temperatures.

3.8.2.3.3 Dead Loads

The following dead loads are transmitted directly through the concrete fill inside the bottom head of the primary containment vessel, through the continuous concrete fill directly under the bottom head and then into the reactor building foundation mat:

a. Concrete fill inside the bottom head of the primary containment vessel,

b. Concrete columns in the suppression chamber and contributory portion of the drywell floor supported by the columns,

c. Contained air in the suppression chamber,

d. Water in the suppression chamber, and

e. Reactor vessel pedestal, reactor pressure vessel (RPV), SSW, and contributory portion of the concrete drywell floor supported by the pedestal.

The following are typical dead loads used in the design of the primary containment vessel:

a. Vessel and appurtenances,

b. Water in the suppression chamber with coincident hydrostatic pressure,

c. Attached equipment and supports, catwalks, platforms and attached piping, air ducts, electrical ducts, conduit and trays,

d. Header loads,

e. Contained air, under test conditions,

f. Weight of the gap filler material applied to the outside face of the shell (used 5 psf.),

g. Design load on welding ring pads of 1500 lb per lineal foot acting in any direction. Welding ring pads are in the drywell and are used for attaching pipe and duct hangers, hoist supports, etc. These pads are welded parallel to the containment vessel surface,

h. Stabilizer truss,

i. Construction dead loads, e.g., scaffolds, and

j. Water in the drywell with coincident hydrostatic pressure, under the flooded condition.

3.8.2.3.4 Live Loads

The following are typical live loads used in the design of the primary containment vessel:

a. Live loads on the drywell floor, platforms and catwalks discussed in Sections 3.8.3.1.3 and 3.8.3.1.4,

b. Monorail live and impact loads,

c. Live loads on floor section of personnel air lock (300 psf), equipment access hatch (1000 psf), and suppression chamber access (150 psf),

d. Live load on temporary construction scaffolds,

e. Operating weight of fluid in attached normally empty piping, headers and penetrations,

f. Head of water, 23 ft 6 in. high, on the refueling bellows seal with the containment vessel head removed and coincident hydrostatic pressure (under the refueling condition), and

g. Same as Section 3.8.2.3.4(f) above except without the containment vessel head removed (under the flooded condition).

3.8.2.3.5 Mechanical Piping Loads

Mechanical piping loads consist of

a. Piping reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition,

b. Pipe reactions under thermal conditions generated by a postulated break and including (a) above,

c. Equivalent static load generated by the reaction of a broken high-energy pipe during a postulated break (and including an appropriate dynamic load factor to account for the dynamic nature of the load),

d. Jet impingement equivalent static load generated by the design-basis accident (DBA) postulated break (and including an appropriate dynamic load factor to account for the dynamic nature of the load), and

e. Pipe reactions and thermal conditions during an event causing external pressure.

A description of certain loads included among those listed above follows:

3.8.2.3.5.1 Jet Forces in Drywell. The drywell shell, personnel air lock, equipment hatch, jet deflectors, and the removable top closure head are designed and constructed to withstand, in combination with other loads, jet forces consisting of either steam and/or water at 340°F and applied as follows:

	Jet Force (kips)	Area Subjected (in^2)
From the closure head flange to the top of the head	33	26
From the closure head flange down to the drywell floor	534	429

A jet force is considered to occur in any direction but is not considered to occur simultaneously with another jet force; however, a jet force is considered to occur coincident with the drywell internal design pressure of 45 psig and design temperature of 340°F. Local yielding may take place on the drywell shell from the jet force, but the shell will not rupture. On the top closure head and other areas, where the shell is not backed up by concrete, the primary stresses resulting from this combination of loads do not exceed the values specified in the ASME Code Section III, Paragraph NE-3131(c) at a temperature of 340°F.

3.8.2.3.5.2 Vent Pipe (Downcomer) Thrusts. The vent pipes (downcomers) and their connections to the drywell floor are designed for the following loads:

a. Jet blowdown thrust

A jet force of 20,000 lb acting upward on each of the downcomers is considered to occur simultaneously with an internal design pressure of 20 psig and a design temperature of 275°F;

b. Initial and final test conditions

A force equal to 1.25 times the design pressure multiplied by the flow area of the vent pipe; and

c. Accident conditions

Forces obtained from Section 3.8.2.3.5.2 (a) except that the temperature in the drywell is taken as 340°F and the temperature in the suppression chamber is taken as 275°F. The drywell floor concrete temperature is taken as 95°F.

3.8.2.3.5.3 Pipe Whip. Pipe whip protection support rings, which are fully circumferential rings, are attached to the primary containment vessel at el. 516 ft 6 in. and 542 ft 7.25 in. The basic function of these rings is to support pipe whip protection framework and to adequately distribute pipe whip loading into the vessel.

The pipe rupture loading is applied to the vessel through the support rings during normal operating conditions at normal operating temperature and at atmospheric pressure, as well as during an incident condition at maximum temperature of 340°F and at design pressure. The primary containment vessel analysis include the effects of a pipe rupture at any single location. For further discussion on function and design of load transmitting members see Section 3.6.

3.8.2.3.6 Thermal Loads

The thermal loads in the primary containment vessel steel are produced by the presence of temperature gradients within the containment and its appurtenances. Thermal effects and loads during normal operating conditions are based on the most critical transient or steady-state condition. Thermal loads are also considered under thermal conditions generated by a postulated pipe break.

3.8.2.3.7 Construction Loads

a. Wind load in the projected area of the steel primary containment vessel before the completion of the reactor building in accordance with Reference 3.8-1, with a basic wind of 100 mph as discussed in Section 3.3, and

b. Snow loads before the completion of the reactor building.

3.8.2.3.8 Missile Loads

There are no external missile loads considered since the primary containment vessel is protected by the biological shield wall. Potential internal missiles and protection provisions are discussed in Section 3.5.

3.8.2.3.9 Loss-of-Coolant Accident Loads

The LOCA imposes pressure and thermal loads plus jet forces associated with coolant flow from any ruptured pipe within the containment. This LOCA loading condition is determined

by analysis of the transient pressure and temperature effects which occur during a LOCA. The governing design condition for the LOCA is discussed in Section 6.2.

3.8.2.3.10 Accident Recovery Loads

Among the postulated LOCAs there may be an accident within the drywell that requires a contingency flooding of the pressure suppression chamber and the drywell to an elevation above the top of the active fuel zone in the reactor vessel as indicated in Section 3.8.2.3.12h; and, with the primary containment vessel head not removed, the reactor vessel cavity outside the primary containment vessel and above the refueling bellows seal flooded to a level above the refueling bellows seal noted in Section 3.8.2.3.12.

The structural design criteria for the primary containment vessel are consistent with the provisions of Regulatory Guide 1.57, Revision 0 (issued June 1973) except with respect to the stress limits specified in Section C-1-b (2) of the guide for the load combinations of accident recovery flooding plus OBE.

For the flooded condition loading combination described in Section 3.8.2.3.12h, the various stress categories shown in ASME Code Section III, Figure NB 3222-1 are satisfied, using allowable primary general membrane stress intensities specified in the ASME Code Section III paragraph NE 3131(c)(1) and NE 3131(c)(2).

The stress limits for the primary containment vessel for the accident recovery flooding plus OBE load condition are shown in Section 3.8.2.5.3. Justification for these stress limits, which are higher than those outlined in Regulatory Guide 1.57 is based on the extremely low probability that flooding would ever be required.

3.8.2.3.11 Seismic Loads

Equivalent static loads are developed through a dynamic analysis of Seismic Category I structures subjected to the OBE and the SSE.

3.8.2.3.12 Loading Combinations

The loading combinations considered in the design of the steel primary containment vessel apply to both the drywell and suppression chamber, unless otherwise noted, and include the following typical loads:

a. Initial proof load test condition (at ambient temperature)

   This is a normal condition.

   1. Dead load

(a) Vessel and appurtenances,

(b) Water in suppression chamber with coincident hydrostatic pressure,

(c) Concrete drywell floor,

(d) Catwalks and platforms,

(e) Contained air, and

(f) Header loads, empty;

2. Live Loads

(a) On drywell floor, and
(b) On catwalks and platforms;

3. Design pressure, internal, in suppression chamber;

4. Test pressure, internal in the drywell;

5. Lateral wind or OBE, horizontal and vertical, whichever is more severe; and

6. Jet blowdown thrusts in the suppression chamber taken at columns, reactor pedestal and penetrations;

b. Final proof load test condition (at ambient temperature)

This is a normal condition.

1. Dead load

Same items as in item a.1, plus:

(a) Drywell equipment, supports, platforms, and attached empty pipe and headers,

(b) Gap fill material, and

(c) Piping, ducts, hoist support loads, etc., on welding ring pads in drywell described in Section 3.8.2.3.3.h.

2. Live load

Same items as in item a.2;

3. Design pressure, internal (drywell and suppression chamber);

4. External pressure of 2 psi (consists of 2 psig restraint due to filler material);

5. Seismic OBE, horizontal, and vertical;

6. Jet blowdown thrusts in suppression chamber (same as item a.6); and

7. Drywell refueling bellows seal loads, namely, the effects of the spring action of refueling bellows seal onto the containment vessel;

c. Normal operating condition (at atmospheric pressure and at operating temperature ranging from 50°F to values specified in Table 3.8-1).

This is a normal condition with OBE and is in conformity with ASME Code, Section III, Paragraph NE-3131(c)(1) or (c)(2) with SSE.

1. Dead load

(a) Vessel and appurtenances,

(b) Water in suppression chamber and coincident hydrostatic pressure,

(c) Concrete drywell floor,

(d) Catwalks in suppression chamber,

(e) Drywell equipment, supports, platforms and attached pipes, headers, air ducts, electrical ducts, conduits, and trays,

(f) Gap filler material, and

(g) Loads on welding ring pads (same as in item b.1);

2. Live load

   (a) On drywell floor,

   (b) On catwalks and platforms,

   (c) Personnel and equipment access openings in the drywell and on personnel access openings in suppression chamber, and

   (d) Operating weight of fluid in normally empty piping and penetrations;

3. External pressure of 2 psi (same as in item b.4);

4. Seismic OBE or SSE, horizontal and vertical, including seismic loads from shear lugs between the drywell floor and the containment vessel and seismic loads from the reactor vessel stabilizers;

5. Jet blowdown thrusts in suppression chamber (same as in item a.6);

6. Drywell refueling bellows seal loads;

7. Drywell floor seal loads; and

8. Thermal loads described in Section 3.8.2.3.6

d. Refueling condition (with containment vessel head removed, at atmospheric pressure and at operating temperature ranging from 50°F to values specified in Table 3.8-1).

   This is a normal condition.

   1. Dead load

      Same as in item c.1;

   2. Live load

      Same items as in item c.2 plus water load due to 23 ft 6 in. head of water on the refueling bellows seal, at the top flange of the drywell, with coincident hydrostatic pressure;

   3. External pressure of 2 psi (same as in item b.4);

4. Seismic OBE, horizontal and vertical;

5. Drywell refueling bellows seal loads;

6. Drywell floor seal loads, namely, the effects of spring action on floor seal on to the containment vessel; and

7. Thermal loads described in Section 3.8.2.3.6;

e. Accident condition (at drywell maximum temperature of 340°F and suppression chamber maximum temperature of 275°F). This is a normal condition with SSE in conformance with ASME Code Section III, Paragraph NE 3131(c), or with OBE.

1. Dead load

   Same items as in item c.1;

2. Live load

   (a) On drywell floor,

   (b) On catwalks and platforms, and

   (c) Operating fluid weight in normally empty piping and penetrations;

3. Internal design pressure in drywell of 45 psig decaying to a negative pressure of 2 psig at 135°F;

4. Internal design pressure in suppression chamber of 45 psig decaying to a negative pressure of 2 psig at 275°F;

5. External pressure of 2 psi (same as in item b.4);

6. Seismic OBE or SSE, horizontal and vertical, including seismic loads from shear lugs between the drywell floor and the containment vessel, and seismic loads from the reactor vessel stabilizer;

7. Jet blowdown thrusts in suppression chamber (same as in item a.6);

8. Jet forces on drywell shell, personnel air lock, equipment hatch, drywell floor, jet deflectors on the drywell floor over the downcomers and at the floor periphery, and the head of the containment vessel. See Section 3.8.2.3.5.1;

9. Drywell refueling bellows seal loads;

10. Drywell floor seal loads;

11. Effects of header jet nozzle loads; and

12. Thermal loads described in Section 3.8.2.3.6;

f. Accident condition with pipe whip (at drywell maximum temperature of 340°F and suppression chamber temperature of 275°F).

This condition includes pipe whip support load effects onto the containment vessel shell.

Loads are the same as those in item e, with the addition of loads on pipe whip support rings due to main steam or reactor feedwater pipe ruptures;

g. Normal condition with pipe whip (at atmospheric pressure and at operating temperature ranging from 50°F to values specified in Table 3.8-1).

Earthquakes used are OBE or SSE. Stresses are in conformance with ASME Code, Section III, Paragraph NE-3131(c)(1) or (c)(2).

Loads are the same as those in item c, with the addition of loads on pipe whip support rings due to main steam or reactor feedwater pipe ruptures;

h. Flooded Condition (with drywell flooded to el. 552 ft 0 in., which is approximately 1 ft above the top of the active fuel zone in the reactor vessel, suppression chamber and downcomer vent system flooded; and, with the containment vessel head not removed, reactor vessel cavity outside the containment vessel and above the refueling bellows seal flooded to a level 23 ft 6 in. above the refueling bellows seal).

1. Dead load

(a) Vessel and appurtenances,

(b) Concrete drywell floor,

(c) Catwalks in suppression chamber,

(d) Drywell equipment, supports, platforms and attached piping, headers, air ducts, electrical ducts, conduits, and trays,

(e) Gap filler material,

(f) Loads on welding ring pads same as in item b.1,

(g) Suppression chamber filled with water at a maximum temperature of 212°F and with coincident hydrostatic pressure, and

(h) Drywell filled with water to el. 552 ft 0 in. at a maximum temperature of 212°F and with coincident hydrostatic pressure;

2. Live load

(a) Operating weight of fluid in normally empty piping and penetrations, and

(b) Water load on the refueling bellows seal at the top flange of the containment vessel, and coincident hydrostatic pressure, due to the reactor vessel cavity outside the containment vessel above the refueling bellows seal flooded to a level 23 ft 6 in. above the seal, as described in item h;

3. External pressure of 2 psi (same as in item b.4);

4. Seismic OBE, horizontal and vertical, including loads from shear lugs between the drywell floor and the containment vessel, and loads from the reactor vessel stabilizer; and

5. Drywell floor seal loads.

3.8.2.4 Design and Analysis Procedure

The steel primary containment vessel, which consists of a vertical free-standing bell-jar shell, bottom and top ellipsoidal heads and numerous penetrations and attachments, is considered to act as an independent structural component within the reactor building. The bottom head, which is supported by the concrete mat foundation, is designed as a pressure tight and leaktight membrane.

The steel primary containment vessel is designed to the ASME Code Section III, Subsection NE for Class MC components. Those areas of the steel primary containment vessel that are free from structural discontinuities and experience no discontinuity stresses due to thermal and/or mechanical loads are designed in accordance with Subarticle NE-3300 of ASME Code Section III. For the configurations and loadings located throughout the steel primary containment vessel including anchorage, embedment, and all appurtenances which are not explicitly treated in Subarticle NE-3130, the design is in accordance with the applicable references designated in paragraphs (b) and (c) of Paragraph NE-3131 of ASME Code Section III.

The design of nonpressure resisting components within the steel containment vessel that are within the jurisdiction of ASME Code Section III, Subsection NE, is performed in accordance with ASME Code Section III, NE-3131 (e).

The design of nonpressure resisting components within the steel containment vessel which are outside the jurisdiction of ASME Code Section III, Subsection NE, is performed in accordance with the general practices of the AISC Specification for the Design Fabrication and Erection of Structural Steel for Buildings, February 12, 1969.

A comparison of the design pressures to the maximum calculated pressures is shown in Table 3.8-3.

3.8.2.4.1 Description

The following describes the design and analysis procedures required to verify the structural integrity of critical areas present within the steel primary containment vessel.

3.8.2.4.1.1 Bottom Ellipsoidal Head and Bell-Jar Shaped Shell. The 2:1 ellipsoidal head is designed in accordance with Subarticle NE-3300, Vessel Design, of ASME Code Section III.

The compressive stresses within the knuckle region caused by internal pressure are limited to the allowable buckling stress in accordance with Paragraph NE-3133 of ASME Code Section III.

The design of the top and bottom ellipsoidal heads for both external pressure and compressive loadings is in accordance with Paragraph NE-3133 of ASME Code Section III.

The bell-jar shaped shell of the steel primary containment vessel is designed in accordance with Subarticle NE-3300 for internal pressure. For external pressure and mechanical loads which include compressive stresses in the bell-jar shaped shell the design is in accordance with Paragraph NE-3133 of ASME Code Section III.

The bell-jar shaped shell contains the following major structural discontinuities:

a. Steel containment vessel embedment, discussed in Section 3.8.2.4.1.2,

b. Vertical stiffeners on the inside periphery of the suppression pool portion of the steel containment vessel, to increase the longitudinal compression strength of the vessel,

c. Horizontal stiffening rings on the inside periphery of the steel containment vessel to provide capability to resist external pressure,

d. Ring girders attached to the inside periphery of the steel containment vessel at el. 516 ft 6 in. and el. 542 ft 7.25 in., as discussed in Section 3.8.2.3.5.3, and

e. Steel containment vessel closure flange and refueling bellows attachment at el. 583 ft 1.25 in.

3.8.2.4.1.2 Steel Containment Vessel Embedment Region. The anchorage of the steel containment vessel to the concrete foundation mat is facilitated by means of an embedded lower skirt. Analysis for this portion utilizes the computer as described in Section 3.8.2.4.2.

The anchorage itself is accomplished by the use of anchor bolts and embedded plates located at the bottom outer steel skirt of the primary containment vessel. The anchor bolts extend into the concrete mat foundation a distance sufficient to develop the required embedment length. The anchor bolts and embedded plates are designed to resist the moments and vertical shears developed by the axial tensile forces of the steel primary containment vessel shell.

3.8.2.4.1.3 Penetrations. The containment penetrations are designed to withstand the normal environmental conditions which may prevail during plant operation and to retain their integrity during all postulated accidents. Containment penetrations are fully described in Section 3.8.6.

An equipment hatch in the drywell is fabricated from welded steel and furnished with double-gasketed flanged and bolted doors. Provision is made to test pressurize the space between the double gaskets of the door flanges.

The personnel access lock is provided for drywell access. The personnel lock is a double door, latched, welded steel assembly. Quick-acting equalizing valves connect the personnel lock with the interior of the containment vessel for the purpose of equalizing pressure in the two systems when entering or leaving the containment.

The two doors in the personnel access lock are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. An emergency lighting and communication system operating from an emergency supply is provided in the lock interior.

The suppression chamber access hatch is fabricated from welded steel and furnished, on the reactor building side, with a double bolt, horizontal hinged, dished steel closure. The closure is flanged and double gasketed with provisions to test pressure between the double gaskets of the closure flange.

Electrical penetrations are provided with double seals and are separately testable at 45 psig. The test taps and seal are located so that tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or suppression chamber. Gaskets are separately testable at the containment maximum internal pressure of 45 psig to verify leaktightness. The covers on flanged closures, such as the equipment hatch cover, the drywell head, suppression chamber access hatch, and personnel access lock doors are provided with double seals so that these can be tested for leaktightness without pressurizing the entire containment system.

3.8.2.4.1.4 Personnel Access Lock and the (Combined) Equipment Hatch and Control Rod Drive Removal Hatch. The personnel access air lock and the equipment hatch/CRD removal hatch, are supported entirely by the steel containment vessel shell. The lock barrel is welded directly to a thick insert plate which in turn is welded at its periphery to the shell of the steel containment vessel. The barrel in the vicinity of its attachment to the insert plate is also thickened as required. The additional thickness in both the barrel and insert plate is provided to satisfy the area reinforcement requirements as well as to resist the external moments and shears due to the cantilevered construction. The discontinuity stresses induced by the combination of external, dead, and live loads including the effects of earthquake loadings are evaluated. The required analyses and limits for the resulting stress intensities are in accordance with Subarticles NE-3100 and NE-3200 of ASME Code Section III.

The doors for the personnel access air lock are dished. The analyses are in accordance with Paragraphs NE-3325 and NE-3326 of ASME Code Section III.

Additional stiffeners are provided as required around the opening of the equipment hatch.

Analytical methods developed by P. P. Bijlaard for the Welding Research Council and the summarization presented in Reference 3.8-4 are utilized to determine the stresses in a cylindrical shell with openings. With a thick insert plate, a flanged neck around the periphery and gusset plates, the stresses are within the ASME Code allowables.

The required analysis and the stress intensity limits are in accordance with Subarticles NE-3100 and NE-3200 of ASME Code Section III. The hatch cover with the bolted flange is designed in accordance with Paragraph NE-3326 of ASME Code Section III.

The following explains the method of analysis and the strength criteria used for the seismic design of the connections between the personnel access lock and equipment removal hatch and

the drywall shell; the provisions made to take care of shell, and the influence of complete local encasement of the drywell in concrete on the seismic design:

a. The entire containment vessel shell, including the personnel air lock and the equipment hatch, is designed to withstand seismic loading derived by seismic analysis as described in Section 3.7. The method of shell analysis is in accordance with ASME Code Section III, Class MC, utilizing allowable stress intensities required by this code;

b. Further, the containment vessel design also includes the seismic affects due to the inertia of the mass of the personnel access lock and the equipment hatch vibrating as an independent system; and

c. Since the personnel access lock and the equipment hatch are completely separated from the surrounding concrete wall by means of a gap, there is no interaction between these appurtenances and the concrete wall. The gap provided is of sufficient dimension to accommodate all vessel movements.

3.8.2.4.2 Computer Programs Utilized in Design and Analysis

The design and analysis of the steel containment vessel utilized the following three proprietary computer programs of the Pittsburgh-Des Moines Steel Company, Pittsburgh, Pennsylvania:

a. AX1, Analysis of Axisymmetric Solids,
b. AX2, Axisymmetric Shell Program, and
c. AX3, Analysis of Thin-Shell Solids of Revolution.

The italicized information is historical and was provided to support the application for an operating license. The AX1 computer program is a special purpose finite element program for the analysis of axisymmetric solids. Meridional-stiffening cannot be modeled in this program. Circumferential stiffening can be modeled with the axisymmetric finite elements. This program was used exclusively to evaluate jet load effects on the CRD Removal Hatch. All the members analyzed are unstiffened axisymmetric solids. It is capable of determining deformations and stresses within axisymmetric structures of arbitrary shape.

The AX2 computer program is a general purpose program for the analysis of composite shells of revolution loaded axisymmetrically. The shell theory used is for thin, isotropic, elastic shells. Meridional stiffening cannot be modeled. Circumferential stiffening members can be modeled as shell elements or as concentrated stiffnesses at node points. This program was used extensively in the analysis of the primary containment vessel. Meridional stiffening members were neglected in these AX2 analyses. Circumferential stiffening members were modeled as shell elements. It is capable of determining stresses and displacements in shells of revolution that are loaded axisymmetrically.

The AX3 computer program is a general purpose program for the analysis of composite axisymmetric shells. Loads can be applied axisymmetrically or non-axisymmetrically using fourier series representations. Meridional stiffening can be modeled by adding shell layers with zero modulus of elasticity in the circumferential direction. By adding a layer or layers of appropriate thickness' the axial and bending stiffness of the stiffening can be represented. Circumferential stiffening members can be modeled using shell elements or by using the layer method described above. It is capable of determining reactions and deformations for thin shell solids of revolution, such as discs, for axisymmetric loads.

The AX3 program was used to analyze two areas:

1. *The containment shell in the vicinity of the seismic stabilizers (el. 567 ft 5.5 in.) under a non-axisymmetric displacement loading. No meridional or circumferential stiffening exists in the region modeled.*

2. *The Equipment Hatch head/flange intersection under Jet loading. Meridional stiffening on the Equipment Hatch head and barrel was modeled using additional shell layers with zero modulus in the circumferential direction. No circumferential stiffening members exist in the area modeled.*

Computer programs AX1, AX2, and AX3 are further discussed in Section 3.12.

3.8.2.4.3 Seismic Analysis

The evaluation of the structural integrity of the steel primary containment vessel, when excited by seismic motion, is based on a dynamic analysis.

The steel primary containment vessel is designed to interact as a structural component with the reactor building (secondary containment structure) to which it is attached. The primary containment vessel is attached to the reactor building at the stabilizer truss level through the primary containment vessel/biological shield wall shear lug interface, and at the reactor building foundation mat level.

The structural components within the steel primary containment vessel, such as the reactor pedestal, reactor vessel and SSW are designed to interact with the reactor building and the primary containment vessel because of their connections at the reactor building foundation mat level, drywell floor level, and the top of the SSW level, through the reactor vessel shear lug/shear lug stabilizer/SSW stabilizer truss interfaces.

With the formulation of an overall mathematical model which provides for the realistic response of the containment system, response spectra and/or time histories are generated at the component interfaces, and at other desired points. These component response spectra and/or

time histories are used to perform detailed dynamic analyses of the individual components as previously mentioned.

Effects due to the presence of water in the suppression pool under earthquake excitations are established following the procedures described in Reference 3.8-2. The additional loads due to sloshing effect are included as part of the design loads of the primary containment vessel. The analytical results and methods of analysis utilized to determine the seismic sloshing effects in the suppression chamber are discussed in Section 3.8.2.4.3.2.

The model used for the seismic analysis and the method of seismic analysis to obtain the seismic moments, shears, displacements, and floor response spectra are discussed in Section 3.7. To obtain a detailed stress/strain analysis in local areas, the following additional methods are used.

In the dynamic analysis of the steel primary containment vessel component, a dynamic mathematical model is formulated which incorporates the general structural geometry and all significant boundary conditions present. The design of the numerous penetrations is such that any restraining forces on the steel primary containment vessel which could be developed can be considered as negligible. The effects of rotational inertia and shear deformations are also considered as negligible in the response of the steel containment vessel. In the determination of the seismic response of the steel containment vessel, damping effects are considered. The incorporation of damping into the dynamic analysis is facilitated by the use of viscous (velocity proportional) damping. The various damping values for both the OBE and SSE excitations for the steel containment vessel are discussed in Section 3.7.

The resulting equations of motion for the steel containment vessel were solved by the use of a computer program called DACSR. The solution algorithm used depends on the analytical method incorporated to evaluate the equations of motion for the system. A discussion of the solution technique is provided in Section 3.7.

The results of the dynamic seismic analysis contain values for maximum translation and rotational displacements and accelerations, moments and shears, as well as response spectra and/or time histories at desired points throughout the steel containment vessel.

These resultant forces are then combined with the various loading conditions is described in Section 3.8.2.3.12 and in accordance with Paragraph NE-3131 of Section III of the ASME Code. These combined forces are used in the structural analysis of the various critical areas present within the steel primary containment vessel. By using a response spectra and/or time history the cantilevered personnel locks, as well as any other appurtenance, are dynamically analyzed as previously discussed.

The resulting stress intensities due to the addition of seismic loads to the various loading conditions for the steel primary containment vessel and its appurtenances are in accordance

with the stress intensity limits as specified in Paragraph NE-3131 of the ASME Code, Section III.

3.8.2.4.3.1 Computer Program Utilized in the Seismic Dynamic Analysis. The seismic dynamic analysis utilized DACSR, a large capacity computer program discussed in Section 3.12. The program was capable of generating the required mass and stiffness matrices which were required to represent the mass and stiffness of the actual structure.

3.8.2.4.3.2 Seismic Dynamic Analysis of Water in Suppression Chamber (Sloshing Effects). Tests were not performed to arrive at the seismically induced sloshing loads in the suppression chamber. All of the loads were derived by calculations. The calculations provide the basis for the acceptance of these loads.

The method of analysis utilized to determine the seismic sloshing loads in the suppression chamber is taken from Reference 3.8-2 (Chapter 6 and Appendix F).

Two separate analyses were performed, using the formulations given in Reference 3.8-2 as follows:

a. In the first analyses, the entire suppression chamber is taken as a cylindrical rigid tank in plan having a flat bottom as modeled in the above-referenced document, in lieu of the actual 2:1 bottom ellipsoidal head, and supported on the foundation mat. In this analysis the RPV pedestal is excluded from the model, and the tank is considered as containing only the water to the full depth shown in Figure 3.8-1; and

b. In the second analysis, the RPV pedestal is included in the model. To include the pedestal, the suppression chamber is modeled to consist of theoretical rectangular tanks in plan, of the minimum quantity and the maximum size that can be fitted or inscribed adjacent to each other within the annulus formed by the cylindrical wall of the suppression chamber and the concentric cylindrical RPV pedestal. The tanks are each assumed as independent rigid bodies supported on the foundation mat, flat-bottomed and containing water to the full depth shown in Figure 3.8-1.

In both analyses, the structures are subjected to the maximum floor accelerations due to the SSE. The acceleration values are obtained from the time-history analysis performed for the reactor building given in Section 3.7.

Both analyses yield water displacements, velocities, and impulsive and convective water pressures on the walls of the suppression chamber and the reactor building foundation mat.

The first analysis, which considers the RPV pedestal excluded from the suppression chamber, yields the maximum impulsive pressures. The second analysis, which considers the RPV included in the suppression chamber, yields the maximum convective pressures. To obtain conservative values for the forces, bending moments and overturning moments on the suppression chamber and foundation mat, the maximum impulsive forces from the first analysis and the maximum convective forces from the second analysis are assumed to occur together.

The following tabulation gives the analytical results obtained for the additional horizontal wall pressures due to SSE. The additional wall pressures are found to be negligible.

Distance below water surface el. 466 ft 4.75 in. (ft)	Horizontal wall pressure due to SSE induced water sloshing (psi)
0	0.30
5	1.56
10	2.57
15	3.30
20	3.74
Below 20	5.84

The maximum vertical displacement (slosh height) and velocity of the oscillating water surface above the undisturbed equilibrium water surface el. 466 ft 4.75 in. are 9.5 in., and 17.2 in./sec respectively, at the suppression chamber face. This occurs in the second analysis which considers rectangular tanks with the RPV pedestal included.

The period of water oscillation (time required for the water to oscillate one complete cycle) is 6 sec in the first analysis (circular tank) and 3.5 sec in the second analysis (rectangular tanks).

The analytical results used for horizontal pressures in the suppression chamber due to the OBE are one-half of the values obtained for SSE.

3.8.2.4.4 Protective Coatings

Protective coatings are described in Section 6.1.2.

3.8.2.5 <u>Structural Acceptance Criteria</u>

3.8.2.5.1 Primary Stresses

The structural acceptance criteria for the steel primary containment vessel, namely, the basis for establishing allowable stress values, the deformation limits, and the factors of safety, are established by and in accordance with ASME Code Section III.

In addition to the structural acceptance criteria, the steel primary containment vessel is designed to meet minimum leakage rate requirements. The leakage rate requirements are discussed in Section 6.2.

Loading combinations are discussed in Section 3.8.2.4 and buckling criteria is discussed in Section 3.8.2.5.4.

3.8.2.5.2 Primary and Secondary Stresses

For loading combinations described in Sections 3.8.2.3.12 (except for f and h), the stress limits specified in ASME Code Section III, NE-3131(c) are utilized.

3.8.2.5.3 Peak Stresses

For loading combinations described in Sections 3.8.2.3.12f and 3.8.2.3.12h, the stress limits specified in ASME Code Section III, NE-3131(c)(1) and NE-3131(c)(2) are utilized.

3.8.2.5.4 Buckling Criteria for the Primary Containment Vessel

To ensure safety against buckling, the rules set forth in ASME Code Section III, NE-3133, are utilized. The buckling analysis of the containment vessel was performed as follows:

External pressure: The allowable working pressure, P_a, calculated in NE-3133.3 was compared with the specified maximum external pressure, -4 psi. Conical shell elements were analyzed as equivalent cylinders in accordance with NE-3133.7.

Longitudinal compression on unstiffened shell: The maximum allowable compressive stress, B, determined in NE-3133.6, was compared to the maximum longitudinal compressive stress produced under all the loading conditions specified, including the compressive stress due to the SSE overturning moment.

Longitudinal compression meridionally stiffened shell: Two independent checks were made on buckling of stiffened shell lengths:

a. NE-3133.6 was applied as above using an equivalent thickness in bending, $t_e = (12 \times I_s/b)^{1/3}$

   b = meridional stiffener spacing

   I_s = moment of inertia of the composite section comprised of the stiffener and a width of shell, b.

b. Additionally, shell lengths were analyzed by treating the composite stiffener-shell described in item a as a column pinned at the shell ring stiffeners. These columns were evaluated for buckling using the AISC criteria.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

The materials essential to containment integrity (items a through f below) comply with the requirements of Article NE-2000 of the ASME Code Section III. Material for the containment vessel (Section 3.8.2) and vessel penetrations (Section 3.8.6) conforms to but is not limited to the following:

a. Plate

ASME SA-516, Grade 70 (drywell head, flanges, containment vessel, and electrical penetration weld ring collars)	Specification for carbon steel plates for pressure vessels for moderate and lower temperature service
ASME SA-537, Class 2 (containment vessel plates and penetration insert plates)	Specification for carbon-manganese silicon steel plates heat treated for pressure vessels
ASME SA-240 Type 304 (electrical penetration header plates for austenitic pipes or fittings)	Specification for chromium and chromium-nickel stainless steel plate, sheet, and strip for fusion-welded unfired pressure vessels
ASME SA-299 (jet deflectors)	Specification for carbon-manganese-silicon steel plates for pressure vessels;

b. Forgings

ASME SA-350 Grade LF1 or LF2 (flued head fittings or flanges)	Specification for forgings carbon and low alloy steel requiring notch toughness testing for piping components (for low temperature service)

ASME SA-182 F304 (flued head fittings for austenitic stainless steel process piping)	Specification for forged or rolled alloy-steel pipe flanges, forged fittings, and valves and parts for high-temperature service
ASME SA-105, Grade II, (flued heads)	Specification for forgings, carbon steel, for piping components (for ambient and high temperature service);

c. Pipe

ASME SA-333 Grade 1 or 6 (seamless)	Specification for seamless and welded steel pipe for low temperature service
ASME SA-312, Grade TP 304	Specification for seamless and welded austenitic stainless steel pipe
ASME SA-376, Grade TP 304	Specification for seamless austenitic steel pipe for high-temperature central-station service;

d. Bolting

ASME SA-320 Grade 17 (ferritic steels)	Specification for alloy steel bolting materials for low-temperature service
ASME SA-320 Grade B8 (austenitic steels)	Specification alloy steel bolting materials for low-temperature service
ASME SA-193 Grade B7 (ferritic steels)	Specification for alloy-steel and stainless steel bolting materials for high-temperature services
ASME SA-193 Grade B8 (austenitic steel)	Specification for alloy steel bolting materials for high-temperature service
ASME SA-194 Grade 7 (ferritic steels)	Specification for carbon and alloy steel nuts for bolts for high-pressure and high-temperature service
ASME SA-194 Grade 8 (austenitic steel)	Specification for carbon and alloy steel nuts for bolts for high-pressure and high-temperature;

e. Castings

ASME SA-216 Grade WCB	Specification for carbon-steel castings suitable for fusion welding for high-temperature service
ASME SA-352 Grade LCB	Specification for ferritic steel castings for pressure containing parts suitable for low-temperature service
ASME SA-351 Grade CF8	Specification for austenitic steel castings for high-temperature service;

f. Weld fittings, elbows, tees, and reducers

ASME SA-420 Grade WPL 6	Specification for piping fittings of wrought carbon and alloy steel for low-temperature service;

g. Gib material

ASTM A-514 (bearing plate under radial beam)	Standard specification for high-yield strength, quenched, and tempered alloy steel plate, suitable for welding
ASTM B-22, Copper Alloy No. 863 or Lubrite Alloy No. 424 (Material for bearing plates under radial beams, except in suppression pool. Material not required in suppression chamber.)	Standard specification for bronze castings for bridges and turntables

Self-lubricating bearing plates are Lubrite, a manganese bronze material, produced by an established manufacturer of such material. They are provided with trepanned recesses which are filled with a lubricating compound capable of withstanding the design temperature and load and consisting of graphite and metallic substances with a lubricating binder. The compound is pressed into the recesses by hydraulic presses so as to form dense, nonplastic inserts. The lubricating area comprises not less than 25% of the total area. The coefficient of friction does not exceed one-tenth.

This material is not used in the suppression pool where it is classified a detrimental material and therefore prohibited. Such a material, however, is not required in the suppression chamber.

h. Material that does not fall within the scope of the ASME Code conforms to the following ASTM Specification:

ASTM A-36	Standard specification for structural steel;

i. Material for anchor bolts of bottom support skirts and temporary hoist structures conforms to the following ASTM specification:

ASTM A-307	Standard specification for carbon steel externally and internally threaded standard fasteners;

j. Gasketing material - Gaskets are solid and are fabricated in continuous rings of silicone, neoprene, or natural rubber or other material suitable for its intended service. Gaskets have a guaranteed life of not less than 12 months. The sealing pressure on each gasket is constant for the life of the gasket. Results of tests performed by an approved testing laboratory demonstrate satisfactory performance of the gaskets. Gaskets are used at hatches and flanges. Typical applications are described in Section 3.8.2.4.1.3;

k. Concrete fill between the top of the foundation mat and the underside of the primary containment vessel, within the area bounded by the bottom support skirts, is an expansive cement concrete. The minimum compressive strength of the concrete fill is 4000 psi at 28 days; and

l. Grout under bottom support skirts is the non-shrink type. Preparation, mixing and placing of grout is in accordance with the manufacturer's instructions. The design strength of grout is 4000 psi at 28 days.

3.8.2.6.2 Quality Control

Quality control measures used during the construction and installation of the steel primary containment vessel and the penetration assemblies (discussed in Section 3.8.6.6) are

a. The vessel vendor submitted shop and field quality compliance and quality assurance organization and procedures. These procedures include, as applicable, the methods of documentation of materials, material control, welder identification, and welding electrode handling and distribution; and the vendor's

methods of qualification of nondestructive testing and welding personnel, procedures, and equipment.

b. *The records pertaining to the steel primary containment vessel contain distinct categories; these are material certifications, welding data, test data, vendor drawings, and vendor-certified stress reports. All records are turned over to the Owner on completion of the work.*

1. *Material certification*

    (a) *Mill Test Reports - mechanical and chemical properties of material used, as detailed within the project specifications.*

    (b) *Nondestructive testing reports.*

    (c) *Plate forming procedures.*

2. *Welding data*

    (a) *Shop weld data - description of the procedures used, the welding process or processes, the welder or welders performing the welding operations, the type of weld, and the results of the following inspections: visual, radiography, ultrasonic magnetic particles or liquid penetrant, as well as the name of the inspector and date inspection was performed.*

    (b) *Field weld data - the same type of information reported under Shop Weld Data.*

    (c) *As-built drawings - drawings designating structural plate, member or part locations and field welds by number.*

    (d) *Inspection and quality assurance reports.*

3. *Test data*

    (a) *Cleaning records pertaining to cleanliness inspections.*

    (b) *Pneumatic tests - a record of pneumatic tests performed, including test pressure, hold time, and the results of the test.*

    (c) *Initial leak rate test - a record of leak test, including test pressure, hold time, and an error analysis of the test data.*

4. *Vendor drawings include outline drawings, assembly drawings, and shop detail drawings.*

5. *Vendor-certified stress reports contain the structural analysis and design calculations and associated design drawings of the various structural elements and components of the primary containment vessel.*

c. *All welding procedure qualifications and welder performance qualifications are in accordance with ASME Code Section IX. The welding design, fabrication, inspection, and acceptance, as a minimum, conform to the requirements of Subsection NE of Section III of the ASME Code. For magnetic particle or liquid penetrant inspection, the acceptance criteria conforms to ASME Code Section III, Subsection NE.*

d. *All procedural requirements for nondestructive testing as a minimum conform to the requirements of Appendix X of ASME Code Section III.*

e. *Erection tolerances - The steel primary containment vessel erection tolerances meet the requirements of NE-4221 and NE-4222 of ASME Code Section III for fabrication and erection. The tolerance for fabrication of remaining plates are as stated in the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, Code of Standard Practice for Steel Buildings and Bridges.*

3.8.2.6.3 Special Construction Techniques

No construction techniques unusual to vessel erection methods were required for the steel primary containment vessel.

3.8.2.7 <u>Testing and Inservice Inspection Requirements</u>

3.8.2.7.1 Inspection of Material and Parts for Fabrication

All materials used in constructing pressure retaining parts of the primary containment vessel were examined before fabrication to detect defects affecting functional integrity of the vessel. Materials were inspected to detect defects and cracks introduced during fabrication and to ensure that the work was properly executed and completed. In addition, pressure retaining parts were inspected to ensure that they conformed to the required shape and size after forming.

3.8.2.7.2 Testing of Primary Containment Vessel During Field Erection

During erection of the bottom ellipsoidal head of the primary containment vessel, vacuum box soap bubble tests of longitudinal and circumferential seams were made by placing a vacuum box over the weld areas after application of a soap solution to the test areas and adjacent base material. No leaks were detected in the area by this test and therefore no repairs and retesting were necessary.

On satisfactory completion of soap bubble testing, a halide testing program over the welded seams followed, inasmuch as internal concrete was placed prior to completion of vessel erection. After successful completion of the halide test program, concrete was placed in the interior region of the bottom ellipsoidal head.

3.8.2.7.3 Testing of the Erected Primary Containment Vessel

On completion of field erection of the primary containment vessel and prior to the installation of penetration internals, a test plan was established to outline the tests and inspections to be performed on the primary containment vessel and appurtenances. The tests were performed in accordance with the applicable requirements of the ASME Code to demonstrate the structural integrity and leaktightness of the completed vessel. The tests consisted of an initial and a final reference volume test, an initial and a final soap bubble test, a structural integrity overpressure test for structural acceptability, and an initial leak rate test. The initial soap bubble test was performed at 5 psig, and the final soap bubble test was performed at 45 psig between the 51.8 psig structural integrity test and the 45 psig initial leak rate test.

The initial and final reference volume tests, the first and last of the tests performed in the order of sequence, evaluated and leak tested the reference volumes in accordance with ANSI N45.4-1972.

The following tests and inspections outlined in the test plan were performed in the following order, in accordance with the plan.

a. Test prerequisites

   1. *Prior to pressurization of the primary containment vessel, preparatory steps were taken such as installation, calibration, and checking out the operability of all systems and instrumentation equipment including that required for measuring containment pressure, temperature, and humidity.*

   2. *Before the tests began, the suppression chamber was filled with water to el. 466 ft 2.75 in.*

b. Initial reference volume test

Before the primary containment vessel was pressurized the reference volume system was evaluated and leak tested in accordance with ANSI N45.4-1972. This test included the helium leak test and the vacuum retention test. The initial reference volume test was successfully completed.

c. Initial soap bubble test

The initial soap bubble test was performed at 5 psig, the first increment of pressure of the structural integrity pressurization test. Soap suds were applied to all weld seams and seals, including seams and seals of all piping and electrical penetrations which were not backed up by water, the equipment hatch, drywell personnel access lock, and suppression chamber access hatch. Where the weld seams of the primary containment vessel penetrations were below the water level of the suppression chamber, visual leak detection was performed. Only the inner door seals of the personnel access lock were soap bubble tested at this time, since the personnel access lock weld seams were tested in the shop. The outer door of the personnel access lock was in the open position and remained open during this test. Weld maps with signoffs for each weld to be soap bubble tested were used for each soap bubble test. This test was successfully completed.

d. Structural integrity pressurization test

This test is also referred to as the structural acceptance overpressure test. The test was performed by pressurizing the primary containment vessel drywell and suppression chamber in increments, holding at pressures of 5, 10, 15, 20, 25, 30, 35, 40, 45, 50, and 51.8 psig. The initial soap bubble test was performed at 5 psig. The outer door of the personnel access lock was left open as it was in the initial soap bubble test. The final test pressure level of 51.8 psig was maintained for at least 1 hr simultaneously in the vessel and on the inner door of the personnel access lock. The increments of holding levels were reached by allowing the pressure to rise 1 psig above the desired level and then reducing the pressure to the desired level. The final holding level of 51.8 psig did not include the 1 psig increase. The final holding test pressure of 51.8 psig is 15% more than the 45 psig design pressure and is in accordance with Subarticle NE-6300 of the Summer 1972 Addenda of ASME Code Section III, Nuclear Power Plant Components, Subsection NE, Class MC components. The structural integrity test was completed successfully.

e. Final soap bubble test

On completion of the structural integrity pressurization test, the primary containment vessel was depressurized from 51.8 psig to 45 psig. A 10 minute hold period at 45 psig was observed and then the inner door seals of the personnel access lock were soap bubble tested. The outer door of the personnel access lock was then closed and the air lock was pressurized to 45 psig. At this point the outer door seals of the personnel access lock were soap bubble tested. When soap bubble testing of the outer door seals was completed, the entire primary containment vessel was soap bubble tested as described in Section 3.8.2.7.3. The final soap bubble test was successfully completed.

f. Initial leak rate test

After the final soap bubble test, an initial leak rate test was performed on the primary containment vessel. During the test, the suppression chamber remained filled with water to el 466 ft 2.75 in. placed therein during the test preparation period described in Section 3.8.2.7.3. During the initial leak rate test, the inner door of the personnel access lock was open and the outer door closed. The primary containment vessel was completely sealed off. Since both the drywell and suppression chamber above the water level at el. 466 ft 2.75 in. are designed for the same pressure, the entire primary containment vessel was tested at the same time without the necessity of providing closures from the drywell.

The test pressure of 45 psig was used. The test pressure was arrived at by depressurizing the vessel from 51.8 psig to 45 psig upon completion of the structural integrity test.

The necessary instrumentation was provided and installed and furnished the data required to calculate and verify the leakage rate. The equipment used was capable of measuring with an accuracy consistent with the measurements made. Continuous hourly measurements were taken until it was shown that the integrated leakage rate from the primary containment vessel did not exceed the specified maximum leak rate of 0.5% of the total weight of contained air in any 24-hr period at test temperature and pressure. Measurements taken were in accordance with procedures outlined in ANSI N45.4-1972.

The leak rate was calculated by the reference volume method and verified by the absolute volume method. The reference volume method followed ANSI N45.4-1972. The leak rate arrived at by both the reference volume and the absolute volume methods was determined by a straight line least-squares fit of the results for the entire period during which data was taken.

g. Final reference volume test

On completion of the initial leak rate test, the primary containment vessel was depressurized to 0 psig and the outer door of the personnel access lock was opened. At this point a vacuum retention test was performed on the reference volume system. The vacuum retention test was successfully completed and the water in the suppression chamber was discharged.

All tests were successfully completed.

3.8.2.7.4 Tests on Electrical and Mechanical Penetrations

All electrical penetration nozzles were pressure tested by two methods. The first test consisted of a shop hydrostatic test of 150 psig prior to shipment of the nozzle to the field. The second pressure test of the electrical penetration nozzles was performed in conjunction with the overall overpressure test of the containment vessel. The pressure for this test was 1.15 times the design pressure of 45 psig as prescribed by ASME Code Section III, Class MC.

All primary containment vessel pipe penetrations were pressure tested by two methods. All internal piping, together with the flued head fitting (if applicable), was initially hydrostatically tested in the shop. Test pressures used were based on the design pressure for the particular system for which the penetration is designed to serve. On completion of the overall containment vessels the penetration was field tested in conjunction with the overall overpressure test of the containment vessel. The pressure of this test was 1.15 times the design pressure of 45 psig as prescribed by ASME Code Section III, Class MC, the same as for the electrical penetrations noted above.

3.8.2.7.5 Tests on Personnel Access Lock

On completion of erection of the primary containment vessel, the air lock was given an operational test consisting of repeated operation of each door and mechanism to determine that all parts operate smoothly without binding or other defects. All defects encountered were corrected and retested. The process of testing, correcting defects, and retesting was continued until no defects were detectable.

Pressure testing of the air lock was performed in conjunction with the containment vessel testing, as described in Section 3.8.2.7.3. During this operation, all double gasketed door seals were tested by pressurizing the space between the gaskets and checking the sealing area for leaks using the test outlined in Section 6.2.6.2.

Design features of the air lock which permit testing are described below:

a. Figures 3.8-10, 3.8-11, and 3.8-12 provide longitudinal and transverse sectional elevations of the personnel access air lock. The figures also identify mechanical and electrical penetrations on the inner and outer faces of both the containment bulkhead and the atmosphere bulkhead;

b. Figure 3.8-13 illustrates the door vacuum relief tube and test connection. This is a design provision which permits between-seal tests on the air lock door seals;

c. Figures 3.8-14 and 3.8-15 illustrate typical mechanical penetrations in the atmosphere and containment bulkheads. The penetration seals and test connections are a design provision that permits between-seal tests on bulkhead penetrations. Each such bulkhead penetration is sealed with a cartridge type seal unit having double dynamic seals, and each bulkhead penetration is provided with the test connection to test these seals;

d. Figure 3.8-16 illustrates a typical electrical penetration in the atmosphere and containment bulkheads. The penetration is accomplished by means of electrical fittings installed into the bulkheads. The fittings pressure seal the electrical leads passing through the bulkhead; and

e. The personnel air lock door seals and the bulkhead penetration seals are tested on a regular post installation periodic basis, subsequent to the tests after erection described in Section 3.8.2.7.3. These tests are of two types: air lock pressure test and individual seal tests.

The air lock design permits pressure testing of the entire air lock at a pressure of P_a. Special air lock features associated with this test include 12 removable test clamp mechanisms on the inner (air lock to containment) door. These clamps prevent the inner door from becoming unseated during leak testing and are located on and are accessible from the inside of the air lock. These clamps are installed for the duration of the internal air lock pressure test.

The air lock is pressurized through the emergency air penetrations. This test confirms sealing capability at a pressure of Pa for all air lock penetrations.

The individual mechanical and door seals are testable as detailed above. The door seals are tested individually on a schedule based on air lock use and as allowed by the containment leakage rate testing program plan as referenced by the Technical Specifications. The door seal pressure test is at 10 psig and is designed to confirm seal integrity.

3.8.2.7.6 Tests on Penetration Field Welds

Leaktightness tests of field welds connecting the penetrations to the primary containment vessel, including welds connecting the penetration sleeves to the vessel shell, were made.

Penetrations welded directly to the containment vessel shell and field welds on the containment vessel shell were tested with the completed containment vessel during the soap bubble, overpressure, and leak rate testing described in Section 3.8.2.7.3.

3.8.2.7.7 Preoperational Leakage Rate Test and Periodic Leakage Rate Tests

On completion of construction of the primary containment system, including installation of all portions of mechanical fluid, electrical and instrumentation systems penetrating the primary containment vessel, and prior to any reactor operating period, preoperational and periodic leakage rate tests are performed. The various leakage rate tests and associated acceptance criteria are in accordance with Option B of Appendix J to 10 CFR Part 50. The steel primary containment system overall leakage rate is tested by the implementation of type A tests. The detection of local leaks and the measuring of leaks across each pressure-containing or leakage limiting boundary is facilitated by the use of type B tests. The steel containment isolation valves and systems are tested by using type C tests. For each of the three types of tests the associated acceptance criteria is as discussed in Appendix J to 10 CFR Part 50. This compliance includes periodic testing, performance of appropriate tests after major modifications, and the initial preoperational tests. A visual inspection of accessible interior and exterior containment surfaces will be performed in conjunction with the leak rate testing. In addition, periodic visual inspections are performed as described in Sections 3.8.3.7 and 3.8.2.7.8. In addition, leakage rate tests will be reported as required in Appendix J. The tests will verify the leaktight integrity of the primary containment vessel and penetrations below the leak rate of 0.5% of the total contained weight of air per day. Type A, B, and C leak rate tests and satisfaction of commitments to 10 CFR 50, Appendix J, are discussed in Section 6.2.6.

Electrical penetrations are provided with double seals and are separately tested at the containment maximum calculated internal pressure (38 psig). The test taps and seals are so located that tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or suppression chamber.

Containment closures which are fitted with resilient seals or gaskets are tested at the containment maximum calculated internal pressure (38 psig) to verify leaktightness. The covers on flanged closures, such as the equipment hatch cover, the drywell head, and personnel access lock doors are provided with double seals so that they can be tested for leaktightness without pressurizing the entire containment system.

In addition, provision is made so that the space between the air lock doors can be pressurized to the full drywell test pressure. Since both doors on the lock swing in toward the drywell

vessel, structural members are provided to brace the inner door during this test. The resilient seals on the personnel access lock are tested at a reduced pressure.

3.8.2.7.8 Examination of Coatings on Immersed Surfaces

Periodic visual examination of concrete structures, structural steel, and the coatings on immersed surfaces inside primary containment will normally be conducted during refueling outages. A continuing examination program has been developed based on the combined observations from the first and successive visual examinations. The frequency of the visual examinations will be based on the findings and apparent degradation rates, and may not necessarily be repeated each refueling outage. If the visual examination of the containment coating indicates that significant deterioration has occurred, the affected areas will be checked to determine the extent of material loss and repairs will be made if necessary.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL CONTAINMENT

3.8.3.1 Description of Internal Structures

The following provides descriptive information of the various internal structures to define the primary structural aspects and elements relied on to perform their safety-related functions. Figure 3.8-1 gives an overall view of the internal structures of the steel containment.

The following internal structures are discussed:

a. Reactor pedestal,

b. SSW,

c. Drywell floor structural system and support elements,

d. Radial beam framing systems,

e. Stabilizer truss,

f. Refueling bellows seals,

g. Reactor steam supply system hangers and supports, and

h. Reinforced-concrete lining inside the bottom head of the steel primary containment vessel.

The safety-related functions of the internal structures include the following:

a. To provide support during normal operation and seismic disturbances and thereby prevent the occurrence of a LOCA, and

b. If a LOCA does occur, to act to mitigate its consequences by protecting the containment and other engineered safety features from the effects induced by the accident such as jet forces and whipping pipes.

The capability of structures internal to the steel containment to withstand the hydrodynamic effects resulting from actuation of SRV and specific loads associated with postulated LOCAs are discussed in Appendix 3A.

3.8.3.1.1 Reactor Pedestal

The general arrangement and principal features of the reactor pedestal are shown in Figure 3.8-17 and 3.8-18.

The reactor pedestal is a vertical cylindrical shell-type reinforced-concrete foundation. This foundation supports the RPV and the SSW. The reactions transmitted by the SSW and the RPV to the pedestal are due mainly to seismically induced loads, pipe break pressures, and pipe rupture restraints attached directly to the SSW and the loads transmitted to the SSW by the radial beam systems. Other pipe rupture restraints are attached directly to the reactor pedestal.

The reactor pedestal is also an important component in the structural system supporting the RPV against seismic disturbances and postulated pipe breaks. This system includes the SSW, reactor pedestal, and the stabilizer truss all in conjunction with the primary containment vessel, the biological shield wall, and the reactor building foundation mat.

In plan, the reactor pedestal is located on the centerline of the RPV and, therefore, on the centerline of the primary containment vessel. In elevation, the reactor pedestal is located directly under the RPV and SSW.

The bottom of the RPV skirt and the SSW are anchor-bolt-connected directly to the top of the reactor pedestal. The bottom of the reactor pedestal is keyed into the reinforced-concrete liner inside the bottom head of the primary containment vessel. Load transmitted from the reactor pedestal to the reactor building foundation mat is by means of the concrete liner inside the containment vessel bottom head, the continuous concrete fill under the containment vessel bottom head and by means of the inner circular skirt attached to the containment vessel bottom head, and anchor-bolted to the reactor building foundation mat. The skirt provides a direct link from the pedestal to the mat. Such load transfer is accomplished without imposing direct load onto the containment vessel bottom head as discussed in Section 3.8.3.4.1.

The features of the reactor pedestal concrete are shown in Figures 3.8-17 and 3.8-18 and include haunches for the support of radial beams and openings for CRDs. Information including plans and sections describing features at the connections of the RPV and SSW to the top of the reactor pedestal are given in References 3.8-5, 3.8-6, and 3.8-7. Information including plans and sections showing the reactor pedestal as part of the drywell floor pressure barrier between the drywell and suppression chamber, and plans and sections showing the monolithic construction at the joint between reactor pedestal and drywell floor, is presented in References 3.8-8 and 3.8-9.

Except at haunches and at any other special features, the reactor pedestal has an inside diameter of 20 ft 3 in., an outside diameter of 30 ft 4 in. and, therefore, a shell thickness of 5 ft 0.5 in. Its height is approximately 84 ft 0 in. The shell is reinforced on both faces by horizontal hoop and vertical meridional reinforcing steel and is designed and constructed integrally with the SSW and RPV, the drywell floor slab, and the reactor building foundation mat.

The inside and outside surfaces of the reactor pedestal are coated with a special decontaminable coating. This coating protects the pedestal surfaces against attack by either aggressive demineralized water or radioactive contamination and facilitates washdown.

3.8.3.1.2 Sacrificial Shield Wall

See References 3.8-5, 3.8-6, 3.8-7, 3.8-23, and 3.8-24 for descriptive information, primary structural functions, structural arrangement, and principal features of the SSW.

3.8.3.1.3 Drywell Floor Structural System and Support Elements

The general arrangements and principal features of the drywell floor structural system and support elements are given in Figures 3.8-17, 3.8-3, 3.8-19, and 3.8-20.

The drywell floor is part of the BWR containment system, utilizing the pressure suppression concept in a Mark II over-under containment configuration. The drywell floor is a leaktight pressure barrier dividing the containment vessel into a drywell portion above the floor and a suppression chamber (wetwell) below the floor and directly under the drywell. The drywell portion, including the drywell floor slab, serve to contain the effects (i.e., mass and radiation) of a LOCA and to direct the steam released from a reactor primary system pipe break into the suppression chamber. The suppression chamber provides a pool of water which serves as a heat sink capable of transforming the energy, in terms of pressure and temperature, that is released from a LOCA following a postulated rupture of the primary coolant piping. The energy transformation is achieved by directing the steam mixture through the drywell floor downcomer vent pipes into the suppression pool where the mixing effect condenses the steam and results in lower pressure and temperature. Noncondensible gases are carried over and

collected in the suppression free space and vented back to the drywell by way of vacuum breaker valves.

See Figures 3.8-1, 3.8-3, 3.8-17, and 3.8-19 and References 3.8-8 and 3.8-9 for information and figures relative to the function of the floor structural system and support elements in providing a leaktight drywell-wetwell pressure barrier.

The drywell floor structural system consists of

a. An outer annulus made of a 2-ft 0-in.-thick reinforced-concrete slab supported by structural steel beams in composite action, by reinforced-concrete columns and by the reinforced-concrete pedestal, and

b. An inner circular reinforced-concrete slab, inside the reactor pedestal, 5 ft 0 in. thick, lower in elevation than the outer slab by approximately 6 ft 10 in., and constructed monolithically with, and supported by, the reactor pedestal.

Additional elements supporting the drywell floor are

a. A continuous circular closure girder embedded in the drywell floor along its outer periphery as shown in Figure 3.8-3;

b. A drywell floor peripheral seal assembly. The seal assembly is discussed in Section 3.8.3.4.3.3; and

c. Shear lugs intermittently located along the outer periphery of the drywell floor and consisting of male lugs welded to the circular flanged girder and female lugs welded to the primary containment vessel.

There are 83 24-in. and 16 28-in. diameter downcomer vent pipes which provide the flow paths for uncondensed drywell steam into the suppression chamber pool. The upper part of each downcomer is embedded in and supported by the reinforced-concrete slab of the drywell floor. A horizontal steel plate ring, welded to each downcomer and embedded in the slab, serves as a downcomer support and as a seal in preventing leakage through the drywell floor. A jet deflector is provided at each downcomer to prevent the direct impingement of a fluid jet onto the downcomer from any pipe break.

A special decontaminable epoxy coating is applied to the drywell floor to reduce the permeability of the concrete slab and to provide additional leaktightness between the drywell and the suppression chamber.

3.8.3.1.4 Radial Beam Framing Systems

Structural steel beams span radially from support points on the SSW and reactor pedestal to the primary containment vessel to form radial beam systems at various levels. The beam seats provided on the primary containment vessel to support the radial beams are designed to support the beams vertically and tangentially and to allow the beams to move freely in the radial direction. The beam seats are also designed to account for differential motions of the primary containment vessel at one end of the beam and the SSW and reactor pedestal at the other end. The radial beam framing systems are erected to serve several purposes. They provide:

a. Supports for mechanical and electrical equipment, such as the recirculation pumps, monorails, valves, air handling units, cable tray runs, and piping systems; during normal operation and seismic disturbances,

b. Platform areas for access to equipment and materials and areas for performing inservice inspection and maintenance, and

c. Supports for pipe whip restraints designed to withstand postulated high energy pipe breaks (see Section 3.6.2).

Various radial beam support systems which are typical are discussed below.

At el. 512 ft 9.5 in. (see Figure 3.8-20) radial beams support the recirculation pump and motor; furnish pipe restraint supports primarily for the residual heat removal (RHR) shutdown cooling supply and return loops; provide support for piping, electrical trays and ventilating duct hung loads; and provide grating floor areas where required for access to material and equipment.

Figure 3.8-21 shows the type of steel framework erected to withstand high-energy pipe rupture loads in the unlikely event of a postulated main steam or feedwater pipe break provides the pipe restraint required to prevent damage to the adjacent redundant main steam and feedwater piping and adjacent structural elements. This framework interfaces with the circumferential pipe whip protection support ring attached to the primary containment vessel at el. 516 ft 6 in. The interface arrangement is one which is comprised of shear lugs which permit the transmission of tangential shear loads and axial loads from the framework and allow for differential vertical motions between the primary containment vessel and the structural framework (see Section 3.6.2).

Figure 3.8-20 shows a radial beam framing system at el. 524 ft 3-7/8 in. provides the platform area required for access to the feedwater valves.

The radial beams at el. 531 ft 0.25 in., shown in Figure 3.8-20, furnish the catwalk needed for inservice inspection of the RPV inlet and outlet nozzles.

The radial beam framing system at el. 541 ft 3.75 in., shown in Figure 3.8-22, is a platform that primarily supports the main steam and reactor feedwater piping at the main steam relief valve location. It also serves as a pipe restraint framework for the main steam and feedwater pipe lines. The steel framework is supported directly by a continuous circular ring girder attached directly to the primary containment vessel at el. 542 ft 7.25 in.

The radial beam platform at el. 557 ft 6.25 in. supports a monorail system used for servicing the main steam relief valves, (see Figure 3.8-23), while another platform at el. 548 ft 6.25 in. provides access to the same valves.

3.8.3.1.5 Stabilizer Truss

The reactor building stabilizer truss in Figure 3.8-24 and the RPV stabilizer in Figure 3.8-25 are important components in the structural system supporting the reactor vessel. This support system includes the reactor pedestal, reactor vessel, and SSW inside the primary containment vessel. The system is designed to interact with the reactor building and the primary containment vessel via their connections at the reactor building foundation mat level, drywell floor level, and the top of the SSW level through the reactor vessel shear lugs/shear lug stabilizers/SSW/stabilizer truss interfaces.

The stabilizer truss is a circular truss with 16 horizontal members hinged at eight panel points at the top of the SSW and at eight panel points at the containment vessel by means of horizontal pin plate/gusset plate connections. The gusset plates are rigidly field-welded to the top of the SSW at one end of each member and to the containment vessel at the other end. The hinges allow vertical translation of the members resulting from differential thermal growth between the SSW and the containment vessel. The reactor vessel, SSW, stabilizer truss, and primary containment vessel are interconnected and as a unit are restrained tangentially but free to move vertically and radially with respect to the biological shield wall. This tangential restraint is accomplished by means of shear lug assemblies at the common panel points on the primary containment vessel and biological shield wall (see Figure 3.8-24). The tangential forces due primarily to earthquake and pipe rupture are in turn transmitted directly to the biological shield wall. Truss restraint to differential radial expansion of the SSW and the containment vessel imposes some load radially at the eight panel points on the containment vessel and is accounted for in the design (see Figure 3.8-24).

The reactor vessel is supported laterally near the top by means of its stabilizer lugs and the stabilizers shown in Figure 3.8-25. The lugs are integral parts of the RPV wall, and the stabilizers are rigidly attached to the top plate of the SSW. Only shear tangential to the vessel is transmitted by this lug and stabilizer arrangement, allowing the vessel to “grow” freely both radially and vertically relative to the SSW.

3.8.3.1.6 Refueling Bellows Seals

The drywell of the primary containment vessel is isolated by means of the drywell floor and its peripheral seal (discussed in Section 3.8.3.1.3) and by means of an inner refueling bellows seal, of flexible stainless steel, at the top head flange of the reactor vessel (see Figure 3.8-26). The inner refueling bellows seal, which is welded to both the reactor vessel and the bulkhead plate, serves to seal the gap between the reactor vessel and the primary containment vessel.

The polyurethane-filled gap between the primary containment vessel and the biological shield wall is kept dry by means of an outer refueling bellows seal at the top head flange of the primary containment vessel. The outer refueling bellows seal is welded between the primary containment vessel and the biological shield wall which seals the space between the primary containment vessel and the biological shield wall.

3.8.3.1.7 Reactor Steam Supply System Hangers and Supports

The steam supply system piping and pumps are supported by various types of hangers which in turn are supported by the structural steel radial beam framing systems; and by pipe supports, the SSW, the drywell floor, and the drywell portion of the reactor pedestal. A description of these supports is found in Section 5.4.

3.8.3.1.8 Reinforced-Concrete Lining Inside the Bottom Head of the Primary Containment Vessel

The reinforced-concrete lining inside the bottom head of the primary containment vessel (Figure 3.8-1 and 3.8-17) facilitates the design and construction of concrete structures in the suppression chamber by providing the means, through a like material for attaching their bases, particularly for attaching the base of the reactor pedestal with a continuous connection to the reactor building foundation mat that transfers all the load directly to the mat, with no residual concentrated load transferred to the bottom ellipsoidal head. In addition, the liner inside the vessel bottom head and the concrete fill on the outside of the bottom head sandwich the bottom head in such manner as to enhance the distribution of uniform type loads. The concrete liner is anchored to the vessel bottom head by means of headed stud shear connectors welded to the vessel bottom head.

3.8.3.2 <u>Applicable Codes, Standards, Specifications, and Regulatory Guides</u>

This section lists the codes, standards, specifications, regulatory guides, and other accepted industry guidelines that were adopted, to the extent applicable, in the original design and construction of the structures inside the primary containment vessel. Modification to these structures may use the latest editions to eliminate repetitious listing for each structure, the codes, standards, specifications, and regulatory guides are listed in Table 3.8-4 and given a

reference number. For each structure inside the primary containment vessel the applicable reference numbers in Table 3.8-4 are given in the following.

3.8.3.2.1 Reactor Pedestal

a. 1A, 2A, 2B, 3, 4, 5A, SB,
b. 6 through 9,
c. 11 through 18,
d. 20 through 25,
e. 30 through 36,
f. 38, 40, 42, 43, 45, and
g. 50 through 54.

3.8.3.2.2 Sacrificial Shield Wall

See References 3.8-5 and 3.8-6 for applicable codes, standards, specifications, and Regulatory Guides. References 3.8-23 and 3.8-24 provide information relative to as-built conditions and compliance to applicable welding codes.

3.8.3.2.3 Drywell Floor Structural System and Support Elements

The structural system and support elements refer to the

a. Reinforced-concrete slab,

b. Reinforced-concrete columns,

c. Reinforced-concrete pedestal,

d. Circular closure girder embedded in the slab along the outer periphery of the floor,

e. Drywell floor peripheral seal and jet deflectors,

f. Shear lugs, and

g. Downcomers and jet deflectors.

3.8.3.2.3.1 Reinforced-Concrete Slab.

a. 1A, 2A, 2B, 3, 4, 5A, 5B,
b. 6 through 9,
c. 11 through 18,

d. 21, 25,
e. 32, 33, 34,
f. 35, 36, 38, 40, 42, 45, and
g. 50 through 54.

3.8.3.2.3.2 Reinforced-Concrete Columns. Same as in Section 3.8.3.2.3.1, except that reference 18 and 21 are not applied.

3.8.3.2.3.3 Reinforced-Concrete Reactor Pedestal. See Section 3.8.3.2.1.

3.8.3.2.3.4 Circular Closure Girder.

a. 1A,
b. 18, 20, 21, 25,
c. 33, 34,
d. 40, 41, 42, and
e. 50 through 54.

3.8.3.2.3.5 Drywell Floor Peripheral Seal.

a. 18, 20, 22, 23, 25,
b. 33, 34,
c. 40, 41, 42, 43, 46, and
d. 50 through 54.

3.8.3.2.3.6 Peripheral Seal Jet Deflectors. Same as Section 3.8.3.2.3.5, except that reference 43 is not applied.

3.8.3.2.3.7 Shear Lugs. Same as Section 3.8.3.2.3.5, except that reference 43 is not applied.

3.8.3.2.3.8 Downcomers

a. 1A,
b. 18, 20, 21, 22A,* 22B,* 23, 25,
c. 33, 34,
d. 40, 41, 42, 46, and
e. 50 through 54.

3.8.3.2.3.9 Downcomer Jet Deflectors. Same as Section 3.8.3.2.3.5, except that reference 43 is not applied.

* See Section 3.8.3.4.9.

3.8.3.2.4 Radial Beam Framing Systems

a. 1A,
b. 18, 20, 25,
c. 33, 34,
d. 40, 41, 42, and
e. 50 through 54.

3.8.3.2.5 Stabilizer Truss

Same as Section 3.8.3.2.4, except that reference 1A is not applied.

3.8.3.2.6 Refueling Bellows Seals

a. 22, 23, 25,
b. 33, 34,
c. 40 through 49, and
d. 50 through 54.

3.8.3.2.7 Reactor Steam Supply System Hangers and Supports

See Section 5.4.14 for the codes and standards applicable to the steam supply system hangers and supports.

3.8.3.2.8 Reinforced-Concrete Lining Inside Bottom Head of Primary Containment Vessel

a. 1A, 2A, 2B, 3, 4, 5A, SB,
b. 6 through 9,
c. 11 through 17,
d. 21, 25,
e. 32, 33, 34,
f. 38, 40, 42, 45, and
g. 50 through 54.

3.8.3.3 Loads and Load Combinations

The following, pertaining to load conditions, load categories, definition of load terms, loads and load combinations, apply to all the major loads encountered and/or postulated for concrete and structural steel internal structures described in Section 3.8.3. All the loads referred to are not necessarily applicable to all structures and their elements. Applicable loads and applicable load combinations for which each structure is designed depends on the conditions to which the particular structure could be subjected.

Hydrodynamic loads resulting from actuation of SRVs and those associated with specific postulated LOCA loads, which were addressed in the DAR are included in Appendix 3A.

Sections 3.8.3.3.1 through 3.8.3.3.3 may be used in conjunction with the appropriate tables on loads and load combinations (Tables 3.8-5 and 3.8-6).

The criteria used for the concrete and steel internal structures of the steel containment complies with the codes, standards, specifications, and regulatory guides listed in Section 3.8.3.2.

3.8.3.3.1 Load Conditions

The load conditions are the following:

a. Service load conditions

   Under service load conditions, the loads are those encountered during construction, testing, normal operation, shutdown, and severe environmental such as the OBE and the design basis wind, and

b. Factored load conditions

   Under factored load conditions, the loads are abnormal loads due to postulated accidents such as LOCAs due to pipe rupture, shutdown, earthquake, and tornado.

3.8.3.3.2 Load Categories

The load categories for loads under service load conditions and factored load conditions are the following:

a. Service load conditions

   1. Construction

      All events and loads during structural construction, excluding those during testing;

   2. Testing

      All events and loads applied during structural integrity tests and preoperational tests such as hydrostatic tests and pressure tests. Each testing event is considered to be mutually exclusive of other testing events;

3. Normal operation and shutdown

   All events and loads that could reasonably be expected during the operation, shutdown, and normal maintenance of the power plant. The magnitude of these events and loads are based on the probability of occurrence of at least once in the design life of the plant;

4. Severe environmental

   All site-related environmental events and loads that could infrequently be encountered during the plant life such as the OBE, design basis wind, and the flood of record;

b. Factored load conditions

1. Extreme environmental

   All loads due to site-related environmental events which are credible but highly improbable. These events include the SSE, design basis tornado, and probable maximum flood;

2. Abnormal

   All loads due to postulated accident events. They include pressure, temperature, pipe whip, jet impingement, and pipe reactions due to each rupture postulated for the design basis pipe accidents. This loading condition also includes plant-related nonenvironmental missiles. The loads from each postulated accident event are considered to be mutually exclusive of other postulated accidents;

3. Abnormal/severe environmental

   Loads due to the highly improbable simultaneous occurrence of abnormal and severe environmental loading categories. Only the specified combination of these categories, as determined by the credible cause-and-effect events, are considered;

4. Abnormal/extreme environmental

   Loads due to the extremely improbable simultaneous occurrence of the abnormal and extreme environmental loading conditions. Only the

specified combinations of these conditions as determined by the credible cause-and-effect events, are considered.

3.8.3.3.3 Load Definitions

The following definitions of load terms apply to the major loads encountered and/or postulated for CGS, under the service load condition and/or the factored load condition, for concrete and structural steel structures:

a. Normal loads

D = Dead loads or their related internal moments and forces, including any permanent equipment loads, hydrostatic loads, and soil loads. For equipment supports, it also includes static and dynamic head and fluid flow effects.

L = Live loads or their related internal moments and forces, including any movable equipment loads such as crane loads, and other loads which vary in intensity and occurrence. For equipment supports, it also includes loads due to vibration and any support movement effects. Appropriate impact factors are included for such moving loads as from trolleys and cranes.

T_o = Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

See Table 3.8-1 for drywell and suppression chamber temperatures.

R_o = Pipe, cable trays, and duct reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition. The reactions include

1. Self-weight including contents,

2. Most severe transient or steady-state thermal condition at normal operating or shutdown conditions, and

3. Effects of unbalanced pressure and thrust.

P_o = Normal operating external pressure differential pressure loads resulting from pressure variation either inside or outside the containment;

b. Construction Loads

The definitions for D, L, T_o in Section 3.8.3.3.3 (a) and W in Section 3.8.3.3.3 (d) are applicable, except that the construction load values are used;

c. Testing loads

The definitions for D and L given in Section 3.8.3.3.3 (a) are applicable, except that the test load values are used. In addition, the following loads are also considered:

P_t = Pressure during the structural integrity and leak rate tests. The primary containment vessel test pressure is 51.8 psig. For the drywell floor test pressure see Sections 3.8.3.3.4 and 3.8.3.7.

T_t = Thermal effects and loads during the tests;

d. Severe Environmental Loads

E = Loads generated by the OBE. The seismic effects include loads from structure, equipment, piping, cable trays dynamic soil pressures, hydrodynamic pressures, and all other items that could be considered as inertial forces for seismic analysis. The seismic excitations are discussed in Section 3.7.

W = Loads generated by the design basis wind as discussed in Section 3.3;

e. Extreme environmental loads

E′ = Loads generated by the SSE.

Seismic excitation from the SSE is discussed in Section 3.7. The seismic effects include loads from structures, equipment, piping, cable trays, dynamic soil pressures, hydrodynamic pressures, and all other items that could be considered as inertial forces for seismic analysis;

f. Abnormal loads

Abnormal loads are loads generated by the design basis accident under consideration.

P_a = Maximum differential pressure equivalent static load within or across a compartment generated by the postulated pipe break, and including an appropriate dynamic load factor to account for the dynamic nature of the load and an appropriate margin to account for uncertainty in the calculations. A small break case is also investigated.

1.	Internal P_a = 45 psig,
2.	External P_a = 2 psig.

T_a = Effects of thermal environment on the structure generated by a postulated pipe break. This includes T_o for all other areas not affected by the pipe break.

1.	Peak drywell = 340°F,
2.	Peak suppression chamber = 275°F.

R_a = Effects of thermal environment on the pipe reactions on the structure and equipment reactions on the structure generated by a postulated pipe break. This includes R_o for all other areas not affected by the pipe break.

R_r = Local effects in the structure (e.g., pipe support and whip restraints) generated by a postulated pipe break including appropriate dynamic load factors to account for the dynamic nature of the loads. These loads include

1. Reactions from pipe supports and whip restraint, Y_r,
2. Jet impingement, Y_j,
3. Missile impact due to a postulated ruptured pipe, Y_m.

Y_r = Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_j = Jet impingement equivalent static load on a structure generated by the postulated break, including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m = Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

3.8.3.3.4 Internal Structures in the Suppression Chamber

The structures are grouped for convenient referencing as either drywell structures or suppression chamber structures. The various internal structures in the steel containment are discussed in Section 3.8.3.1. Of these structures, structures referred to as drywell structures are listed in Section 3.8.3.3.5 and structures referred to as suppression chamber structures are as follows:

a. Reinforced-concrete structures

    1. Drywell floor slab,

    2. Drywell floor support columns,

    3. Reactor pedestal, and

    4. Reinforced-concrete lining inside bottom head of primary containment vessel;

b. Structural steel structures

    1. Drywell floor support steel immediately on underside of drywell floor slab,

    2. Drywell floor continuous circular closure girder,

    3. Drywell floor peripheral seal assembly,

    4. Drywell floor peripheral seal jet deflectors,

    5. Drywell floor peripheral shear lugs,

    6. Downcomer vent pipes,

    7. Downcomer jet deflectors,

    8. Radial beam framing system supporting downcomers against horizontal seismic loads, and

9. Catwalks.

The reinforced-concrete structures are designed using the loads, load combinations and load factors listed in Table 3.8-5.

The structural steel structures are designed using the loads, load combinations and load factors listed in Table 3.8-6. Appropriate dynamic load factors are applied to the calculated dynamic loads in the design of each structural steel element.

See Appendix 3A for a discussion of hydrodynamic loads associated with activation of SRVs and LOCAs.

Structures in the suppression chamber, inclusive of the drywell floor, are designed to resist in combination with other possible concurrent normal and/or accident loads the effects of hydrostatic and hydrodynamic forces associated with water set in motion by seismic disturbances.

Internal structures are designed for the reactions of all other structures or equipment that they may support.

The effects of concrete volume changes are minimized by designing the concrete mix for minimal volume changes and by prescribing construction techniques to minimize differential strains.

The drywell floor is designed for the following differential pressures which may develop across the floor during the containment response following a postulated pipe break accident:

a. During the short term response, a differential pressure equal to +25 psig due to overpressurization of the drywell relative to the suppression chamber and applied to the drywell side in a downward direction, and

b. During the long term response, a differential pressure equal to -6.4 psig due to a partial vacuum in the drywell relative to the suppression chamber and applied to the suppression chamber side in an upward direction.

The drywell floor is designed for a test pressure of +25 psig overpressurization of the drywell relative to the suppression chamber applied to the drywell side.

The drywell floor is also designed for the following loads in addition to its own dead and live loads:

a. The reactor pedestal support reactions (vertical, base shear, and overturning moment),

b. Pipe rupture loads, including jet impingement loads and reactions, transmitted by pipe restraints connected directly or indirectly through radial beam framing systems serving as pipe rupture support truss systems,

c. Thermal and pressure loads imposed during normal operating conditions, and

d. Forces induced during an OBE or SSE.

The reactor support pedestal is designed to resist the following loads, in addition to its own dead load and live leads:

a. Dead and live loads from the reactor vessel, SSW and radial beam framing systems,

b. Thermal and pressure loads under normal operating and accident conditions,

c. Pipe rupture loads transmitted by pipe to the reactor pedestal by pipe restraints connected directly or indirectly through the radial beam framing systems serving as pipe rupture support truss systems,

d. Forces induced during an OBE or SSE, and

e. Thermal, pressure, earthquake, and pipe rupture loads that act on the RPV and SSW and are transmitted to the reactor pedestal via the support reactions.

3.8.3.3.5 Internal Structures in the Drywell

The structures are grouped for convenient referencing as either drywell structures or suppression chamber structures. The various internal structures in the steel containment are discussed in Section 3.8.3.1. Of these structures, structures referred to as suppression chamber structures are listed in Section 3.8.3.3.4; and structures referred to as drywell structures are the following, all of which are treated as structural steel internal structures:

a. SSW,
b. Radial beam framing systems,
c. Stabilizer truss,
d. Refueling bellows seals, and
e. Reactor steam supply system hangers and supports.

The loads, load combinations, and load factors used in the design of the SSW are presented in References 3.8-6 and 3.8-7. Further information relative to the as-built wall is contained in References 3.8-23 and 3.8-24. The loads, load combinations, and load factors used in the design of the structures in the drywell are presented in Table 3.8-6. Appropriate dynamic load factors have been applied to the calculated dynamic loads in the analysis of each structural steel element for load application time.

The radial beam framing systems serving as floor levels and/or pipe restraint supports are designed for the following loads in addition to their own dead loads:

a. A uniform platform live load,

b. Loads from pipe hangers, ventilation ducts, and electrical cable trays,

c. Forces induced during an OBE or SSE,

d. Pipe whip restraint forces, including the jet impingement load, due to a rupture of the supported pipes (see Section 3.6.2), and

e. Temperature and pressure effects during normal operating and accident conditions.

Internal structures are designed for the reactions of all other structures or equipment that they may support including the steam supply system hangers and supports.

The reactor vessel stabilizer truss is designed primarily for lateral seismic loads. However, all the loads associated with a support at the top of the SSW such as pressure and pipe whip loads on the SSW are included in the design of the stabilizer truss. The applicable load combinations are found in Table 3.8-6.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Reactor Pedestal

The general approach in the analysis and design of the reactor pedestal is to determine the values of the controlling stress resultants on the basis of elastic analysis under design loadings and to provide the required capacity of the pedestal in accordance with the strength method of the ACI 318-71 Code (Reference 3.8-10). Design loadings are in conformance with the loads and load combinations of Section 3.8.3.3. The report on design and analysis procedures for the upper portion of the pedestal, including transmission into the pedestal of reactions from the SSW and the RPV, is contained in Reference 3.8-6.

The principal loadings controlling the design of the pedestal are due to seismic action and pipe break effects which include annulus pressurization, pipe reactions, and pipe whip forces. Controlling overall stress resultants are pedestal bending moment, shear, and axial force. The values of these stress resultants due to seismic action are obtained from a dynamic analysis of a discrete mathematical idealization of the entire reactor building structure as described in Section 3.7. For other loadings, the pedestal is analyzed as a cylindrical beam fixed at its base and simply supported at the level of the drywell floor.

The distribution over the cross section of the axial stresses and shearing stresses due to the aforementioned stress resultants is that associated with single flexural theory. The axial force per unit length of arc due to the overall bending moment varies linearly with the distance from the neutral axis. The shearing force per unit length of arc is circumferential (tangential) in direction and varies sinusoidally in magnitude with maxima at the neutral axis. Meridional and hoop steel requirements are determined at the locations of maximum stress; this reinforcement is then provided uniformly around the pedestal.

Specific analyses are made at pedestal discontinuities such as openings and boundaries to determine the radial shears and radial moments. Additional reinforcement consisting of radial ties and meridional steel is provided as required.

At the base of the pedestal, provision is made for transmission of the pedestal reactions. Capacity for the transmission of shear is available due to axial compression, shear friction and the continuous key of the pedestal into its base. The axial tensite forces are transmitted by continuing the meridional reinforcement into the base where the connection to the cadweld ring assembly is made. This assembly, in turn, connects through weldments to the inner steel skirt which is anchored to the basemats.

3.8.3.4.2 Sacrificial Shield Wall

The design and analysis procedures for the SSW are described in Reference 3.8-7. Further information relative to the as-built wall is contained in References 3.8-23 and 3.8-24.

3.8.3.4.3 Drywell Floor Structural System and Support Elements, Including the Peripheral Seal Assembly, Peripheral Seal Jet Deflectors, and Peripheral Shear Lugs

3.8.3.4.3.1 <u>Drywell Floor Slab and Columns</u>. The drywell floor slab and columns are each analyzed elastically to determine the values of controlling stress resultants under the design load combinations for concrete structures as in Section 3.8.3.3. The required capacity for each of these structures is then provided in accordance with the strength method of the ACI 318-71 Code (Reference 3.8-10).

Under vertical loading, the floor slab is considered to act as a one-way slab in the radial direction, supported by tangential beams below, and extending from the support at the pedestal

to the face end at the primary containment vessel. The slab is analyzed as continuous over the supporting beams except for the spans between downcomers which are taken to be simple spans. Significant loads include dead load and differential pressure on the slab (P_a) but the principal load controlling slab design is the pipe break jet impingement force (R_r). Slab capacity is provided to resist the calculated design shears and moments.

The floor slab is also analyzed for the effect of other significant loads besides vertical loads. The effect of differential temperature between drywell and wetwell including slab bending is checked. Also, the connection between the pedestal and the floor slab is checked for capacity to transmit the pedestal horizontal seismic reaction.

The wetwell columns are subjected to a combination of axial (vertical) and lateral (horizontal) loading resulting from the superimposed loads from the drywell floor and seismic action respectively. The significant loads from the drywell floor, which contribute to the design axial load, are the floor dead load, live load, vertical seismic load (E'), differential pressure, and jet impingement. Column flexure due to horizontal seismic action is determined from analysis of the column as an elastic member subjected to lateral inertial forces corresponding to the column seismic acceleration. The capacity of the column to sustain the design stress resultants is determined by the strength method of ACI 318-71 Code. With concurrent axial load and bending moment, the magnified moment due to axial load is utilized in conformance with the Code.

3.8.3.4.3.2 Structural Steel Members. The steel beam structural system of the drywell floor consists of a grid of radial and tangential beams which support the drywell floor slab. The 17 radial beams, which divide the floor area into 17 similar sectors, are supported by the pedestal and the wetwell columns and extend as cantilevers beyond the columns to the vicinity of the primary containment vessel. In each sector, the tangential beams carry the drywell floor slab and span between the radial beams.

Each of the radial and tangential beams is analyzed by conventional elastic methods as an overhanging or simple span beam, as appropriate, to determine the design moments and shears. Loadings for the radial beams and tangential beams are as discussed above for the wetwell columns and the drywell floor slab. The beams are designed as composite beams in conjunction with the slab above. The elastic working stress design method of the 1969 AISC Design Specification (Reference 3.8-11) for composite construction is followed.

3.8.3.4.3.3 Drywell Floor Peripheral Seal Assembly. The drywell floor peripheral seal assembly is shown in Figure 3.8-3. The drywell floor peripheral seal is made of steel and is welded to the primary containment vessel and to the underside of the circular closure girder embedded in the drywell floor. It is a 270 degree segment of a stainless-steel pipe in cross section, circular in plan, and is drained to the floor drain system which is routed to a point outside of primary containment. Design and construction are compatible with primary containment requirements of Class MC components. Assembly of the seal and attachment

thereof to both the floor and the primary containment vessel is by means of welding in accordance with the ASME B&PV Code Section III, Class MC. The floor seal is designed to accommodate the maximum vertical and radial differential thermal movements which may occur during plant startup, normal operation, and shutdown. It is also designed to withstand, in an elastic manner, the effects associated with a LOCA, including temperature changes and pressure differentials ranging from +25 psig to -6.4 psig, and seismic loads. No other loads are applied to this seal. Jet deflectors are provided at the seal to prevent the direct impingement of a fluid jet force on the seal due to any pipe break. To prevent differential lateral and torsional movements, shear lugs are furnished along the outer periphery of the drywell floor to ensure that movements of the interfacing drywell floor, floor seal, and primary containment vessel are in unison during seismic events.

A continuous circular closure girder, which is of structural steel and embedded in the reinforced-concrete drywell (or diaphragm) floor along its periphery, is provided. Its basic function is to complete the drywell floor closure. It consists of a cylindrical vertical web plate extending from the bottom of the radial steel beams supporting the drywell floor slab to the top of the drywell floor slab at el. 499 ft 6 in. and with annular flanges as illustrated in Figure 3.8-3. In addition to its sealing function, the closure girder also provides the means for connecting the drywell floor peripheral seal to the drywell floor, for attaching the male components of the shear lug assembly, and for supporting the concrete floor. The closure girder withstands the design basis accident loads, drywell floor and slab loads, tangential seismic shear loads, and loads from the drywell peripheral seal.

The closure girder is designed according to the 1969 AISC Specification and for normal operating load combinations, extreme environmental, and the abnormal/extreme environmental loading combinations. Among the loads included in the combinations are design basis accident loads, drywell floor and slab loads, tangential seismic shear loads, and loads from the drywell floor seal. The loads are effectively resisted by the girder in flexure, shear, bearing on the concrete slab, and in tension by the way of shear stud connectors and embedded structural steel.

3.8.3.4.3.4 <u>Drywell Floor Peripheral Shear Lugs</u>. Thirty-six male shear lugs, equally spaced around the drywell floor periphery, transmit horizontal load between the drywell floor and the primary containment vessel. Each of these lugs consist of an assembly of steel plates joined by welding and anchored to the concrete floor slab by stud shear connectors. Transmission of load to the primary containment vessel is via female shear lugs welded to the vessel. The joint between male and female lugs affords restraint only in the circumferential direction as relative motion in the vertical and radial directions is permitted.

Analysis and design of the shear lug assembly is in accordance with the elastic working stress design method, Part 1 of the 1969 AISC Design Specification (Reference 3.8-11). The principal load controlling the design of the lug assembly is the horizontal seismic force transmitted by the drywell floor as determined by the method in Section 3.7. In line with

elastic theory, the distribution of the shear force per unit length of periphery is taken to vary sinusoidally with maxima along the diameter perpendicular to the direction of the overall shear force. The maximum value of the distributed shear force per unit length is used to design the shear lugs.

3.8.3.4.4 Radial Beam Framing Systems

The radial beam framing systems considered are those which do not support pipe whip restraints. Analysis and design of those beam systems which do support pipe whip restraints is discussed in Section 3.6.2.3.3.2. The analysis and design of the former radial beam systems is in accordance with the elastic working stress design method, Part 1 of the 1969 AISC Design Specification (Reference 3.8-11). Conventional elastic beam analysis is used. The significant loads in the load combinations are dead, live, reactions under operating conditions, and seismic loads.

3.8.3.4.5 Stabilizer Truss

The stabilizer truss is a pin-connected plane truss which transmits horizontal force between the top of the SSW and the primary containment vessel biological shield wall, as described in Section 3.8.3.1.5. This transmitted force represents reactions from the SSW and the RPV. The supports for the truss joints at the SSW are fixed at the wall so that two components of reaction (radial and tangential) may occur. At the primary containment vessel the truss joint support is constrained only in the circumferential direction so that the only reaction is tangential force.

Analysis and design of the stabilizer truss is in accordance with the elastic working stress design method, Part 1 of the AISC Design Specification (Reference 3.8-11). The principal loads controlling the design of the truss result from seismic action and pipe break effects including pipe whip, pipe jet, and annulus pressurization. The forces transmitted by the stabilizer truss under these loadings are determined by analysis of the overall structural system from the pedestal to the primary containment vessel including the SSW and the RPV. In this regard, the SSW is modeled as a space frame as described in Reference 3.8-7 and the RPV as a beam to give the loads transmitted by the stabilizer truss. Analysis of the stabilizer truss as a pin-connected plane truss with supports as described above is accomplished using the proprietary computer program “McDonnell-ECI, ICES, STRUDL” which is based on MIT’s STRUDL II, Version 2, Update 2 as augmented by McDonnell Douglas Automation Co., St. Louis, Missouri. The computer analysis is described in Reference 3.8-7.

3.8.3.4.6 Refueling Bellows Seals

Design and analysis procedures for the inner and outer refueling bellows seals are based on applicable ASME Code Sections II, VIII, and IX; the Standards of the Expansion Joint Manufacturers Association, and Interpretation Case Number 1177-7 (see Figure 3.8-26).

3.8.3.4.7 Reactor Steam Supply System Hangers and Supports

Design and analysis procedures for the steam supply system hangers and supports are found in Section 5.4. Design and analysis procedures for the General Electric stabilizers are based on applicable ASME pressure vessel codes.

3.8.3.4.8 Reinforced-Concrete Lining Inside Bottom Head of Primary Containment Vessel

The design accounts for strains caused by creep, shrinkage, and elastic shortening. The methods and data used for the analysis are based on the applicable codes, standards, and specifications in Table 3.8-4 and the results of past experience. (See Figures 3.8-1 and 3.8-17).

The concrete lining is analyzed using elastic methods and designed in accordance with ACI 318-71 by the strength method.

The headed stud shear connectors anchoring the concrete liner to the bottom head of the primary containment vessel are capable of transferring horizontal shear from the concrete internal structures to the bottom head of the containment vessel and of resisting relative movements between the concrete liner and the bottom head of the containment vessel. See the discussion of the reactor pedestal in Sections 3.8.3.1.8 and 3.8.3.4.1.

The analysis and design account for any postulated loading conditions that would cause net uplift at the base of certain reinforced-concrete columns.

Uplift on any portion of the pedestal base is transmitted directly into the reactor building foundation mat as discussed in Section 3.8.3.4.1.

3.8.3.4.9 Downcomer Vent Pipes

The downcomer vent pipes are designed to contain and direct uncondensed drywell steam into the suppression pool following a pipe break accident. See Section 3.8.3.1.3 and Reference 3.8-8, and Appendix 3A for further description and the design and analysis procedures used.

Stainless-steel extension pieces were added to the ends of the downcomers to prevent coating damage from plugs which are installed for the preoperational bypass leakage rate tests. Downcomers were originally provided with exit flanges for these tests. These flanges were removed because of concern about the applicability of test data taken on prototype downcomers without flanges. The downcomers are designed and constructed in accordance with ASME Section III Class 2 requirements above 1 in. above the circumferential weld joining the stainless-steel extension pieces to the bottom of the downcomers. Below this point

the downcomers are designed and constructed to ASME Section III Class 3 requirements. The only effect of this code break is to eliminate radiography requirements for the circumferential weld.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

The maximum permissible stresses and strains used are given in Table 3.8-12. These permissible stresses and strains are used to keep the structures below the range of general yield, both under service load conditions and factored load conditions.

For each of the loading combinations listed in Table 3.8-5, the required sectional strength of concrete (U) is calculated using the strength design method of ACI 318-71 with the applicable capacity reduction factor.

The symbol U denotes the section strength required to resist design loads or their related internal moments and forces based on the strength design methods described in ACI 318-71.

The reinforced-concrete internal structures of the steel containment include the drywell floor, the drywell floor support columns, the reactor pedestal, and the reinforced-concrete lining inside the bottom head of the primary containment vessel. (See Figures 3.8-17 and 3.8-18).

3.8.3.5.2 Structural Steel

See Table 3.8-7 for the criteria used for

a. Required limits of section strength, S and Y, and
b. Section moduli

The symbol S denotes the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," February 12, 1969 (Reference 3.8-11). The symbol Y denotes the section strength required to resist design loads based on plastic design methods described in Part 2 of the AISC Specifications (Reference 3.8-11). For steel internal structures of steel containments, the elastic working stress design method of Part 1 of the AISC specification (see Table 3.8-4) is used. All the loads considered in the loading combinations are factored loads. The plastic design method of Part 2 of the AISC Specification (see Table 3.8-4) is used as may be required for such structures as pipe restraint supports.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Structures internal to the containment, except for the SSW, are not in a region of high-energy neutron flux.

It has been determined that in the 40-year life expectancy of the station the outside face of the SSW will experience a neutron fluence of less than 2 x 10^{16} nvt.

The construction materials and material quality control procedures for reinforced-concrete structures internal to the containment conform to the standards set forth in Section 3.8.4.6. Structural steel standards are also found in Section 3.8.4.6.

Materials and quality control programs comply fully with the "Building Code Requirements for Reinforced Concrete," ACI 318-71 (Reference 3.8-10), for concrete and with the AISC Specification (Reference 3.8-11), for steel as applicable.

See Table 6.1-1 for materials for specific components of internal structures in the containment.

Quality control meets the requirements of ANSI N45.2.5 (Reference 3.8-12), and Regulatory Guide 1.55, Revision 0. Reinforcing bars are generally not welded in any structure, except where mechanical cadweld fasteners are not feasible.

No construction techniques unusual to methods used during CGS construction were employed or required for the concrete and steel internal structures of the steel containment.

3.8.3.7 Testing and Inservice Surveillance Programs

All of the structural components in Section 3.8.3 are visually inspected during outages as defined by the inspection program. The frequency of periodic inspections is based on the results from the findings from previous inspections, as defined in the inspection program and summary finding reports. The inspections are made to determine if degradation to structural integrity has occurred. Inspections of concrete structures are conducted to check for possible deterioration, excessive cracking, or spalling of concrete. Similar inspections are made of structural steel members to check for deterioration of surface coatings and abnormal deformations or warpage.

Rigorous inspection was carried out during construction and in conjunction with the quality control assurance procedures for structural materials outlined in Section 3.8.4.6.

The drywell floor metal peripheral seal is designed, procured, fabricated, installed, and inspected in accordance with the 1971 Edition of the ASME Code Section III, Subsection NE, Class MC Components, Winter 1973 Addenda, except that hydrostatic or pneumatic testing of the seal is not performed, and the seal is not N-stamped. Leaktightness of the seal is tested,

however, as described below and in Reference 3.8-14. Butt welds joining the peripheral seal segments are radiographed, and welds joining the peripheral seal to the containment and to the diaphragm slab closure girder are liquid penetrant examined.

Periodic drywell-wetwell leakage tests are performed. See Section 6.2 for details of the periodic testing methods that are included in the containment leakage rate testing program.

The initial bypass leak rate test included tests at 25 psid and 15 psid in addition to the low differential pressure required in the periodic tests. Successful completion of the 25 psid bypass leak rate test served to prove the structural integrity of the drywell floor.

3.8.5 FOUNDATIONS

3.8.5.1 Descriptions of Foundations

All foundations described below are supported by Quality Class I compacted structural backfill as described in Section 2.5.4.5. As a means of providing a level working surface for construction of the mat type foundations, a 4-in. thick unreinforced concrete leveling slab is installed on the compacted subgrade.

The groundwater level at the site is sufficiently lower than the deepest foundation in the complex. For discussion of groundwater levels at the site, see Sections 2.5.4.6 and 3.4.

3.8.5.1.1 Reactor Building

The primary containment vessel and the reactor building enclosing the containment vessel are both supported on a common, reinforced-concrete mat foundation having a thickness of 16 ft. The mat is reinforced with top and bottom layers of reinforcing steel. Reinforcement is placed in an orthogonal grid pattern. The plan of the mat and the corresponding sections through the mat are shown in Figure 3.8-43.

Induced horizontal shears in the mat due to seismic, wind, and tornado disturbances are resisted by frictional resistance between the mat and the supporting media. Shear resulting from shear wall action is transferred to the mat through shear reinforcement. Lateral shears on vertical load-carrying elements are transferred to the mat through keys in the mat.

A gap is provided between the reactor building foundation mat and adjacent building foundations. The gap is of sufficient horizontal dimension to preclude interaction of Seismic Category I foundations with Seismic Category II foundations, during the SSE and the OBE.

The capability of the foundation mat to withstand loads associated with steam relief valve actuation and with vent clearing, and to withstand specified loads associated with postulated LOCAs which act in the drywell and wetwell of the steel containment vessel, are discussed in Appendix 3A.

For additional description of the reactor building foundation mat, see Section 3.8.2.1.

3.8.5.1.2 Radwaste and Control Building

The radwaste areas and the control room area are contained in one building unit supported on a reinforced-concrete mat foundation having a nominal thickness of 8 ft. The control room area occupies approximately 30% of the radwaste and control building plan at the north end of the building. For description of the radwaste and control building, see Section 3.8.4.1.2. The mat has the same structural characteristics as the reactor building mat described in Section 3.8.5.1.1.

The control room portion of the building unit shares common structural elements with the radwaste portion, which is Seismic Category I. Because of this interconnection and the resulting seismic interaction between the building portions, the entire building complex is modeled as a unit for Seismic Category I analysis (see Section 3.7.2).

3.8.5.1.3 Diesel Generator Building

The foundations for the diesel generator building consist of continuous reinforced-concrete wall footings under all building perimeter and interior load bearing walls and reinforced-concrete spread footings under each interior building column. The diesel generator units are each

supported on an individual reinforced-concrete foundation, isolated from the above mentioned building foundations. Induced horizontal shears in the foundations due to seismic, wind and tornado disturbances are transferred to the supporting media through the frictional resistance between the foundations and the supporting media. For a description of the diesel generator building, see Section 3.8.4.1.4.

The diesel oil storage tanks are buried and supported directly on Quality Class I structural backfill as referenced in Section 3.8.5.1. Since the groundwater level is lower than the invert of the tanks (see Section 3.8.5.1), an empty storage tank will not have buoyancy forces acting upon it.

3.8.5.1.4 Standby Service Water Pump House and Spray Pond

The standby service water pump house and spray pond are two structures joined together along one face of the pump house.

The spray pond is an inground water retention structure. The structure consists of reinforced-concrete cantilevered retaining walls, with an independent reinforced-concrete bottom slab on which is supported the spray piping. Spray pipe supports are secured to the slab with bolts screwed into expansion anchors drilled into the concrete slab. The pond is provided with a depressed sump in the bottom slab at one corner, at the pump house pump inlet bay.
A membrane vapor barrier is placed between the structural slab and the leveling slab.

The standby service water pump house foundation consists of three types of foundations. The pump inlet bay portion of the pump house is depressed to the same elevation as the pond sump whose bottom is a reinforced-concrete mat foundation having a thickness of 3.5 ft. The concrete walls which form the pump inlet bay and support portions of the building are also supported on this mat. The face of the pump house which is adjacent to the spray pond is supported by a reinforced-concrete retaining wall common to both the pump house and the spray pond. The balance of the pump house is supported by reinforced-concrete columns and spread footings. For description of the standby service water pump house and spray ponds, see Section 3.8.4.1.5.

Horizontal shears in retaining walls and piers are transmitted to their respective foundations through shear keys. Shears in the foundations are transferred to the supporting media through the frictional resistance between the foundation and the supporting media.

3.8.5.1.5 Condensate Storage Tank Retaining Area

The condensate storage tank retaining area consists of a reinforced-concrete mat foundation, at grade, having a thickness of 2 ft 4 in. The mat provides support to the two condensate storage tanks and the perimeter reinforced-concrete dike walls which are designed to contain the contents of the tanks in the event of their failure. The mat edges are thickened to provide the

strength required to sustain all of the seismic loads acting on the perimeter dike walls. The perimeter dike walls are keyed to the mat foundation to resist the seismically induced lateral horizontal shears in the walls. The mat rests directly on compacted backfill without any membrane waterproofing. Horizontal shears in the mat are transferred to the supporting media through the frictional resistance between the mat and the supporting media.

3.8.5.1.6 Turbine Generator Building

The turbine generator building foundation mat, like the superstructure, is designed to withstand the effects of an SSE and maintain its structural integrity thus providing adequate protection for the main steam lines designed as Seismic Category I, as described in Section 3.8.4.1.3. The turbine generator building is supported on a reinforced-concrete mat foundation having varying thicknesses ranging from 9 ft to 12 ft. For a description of the turbine generator building, see Section 3.8.4.1.3. The mat has the same structural characteristics as the reactor building mat described in Section 3.8.5.1.1.

3.8.5.1.7 Non-Seismic Category I Safety-Related Foundations

The makeup water pump house is a non-Seismic Category I structure but is a safety-related installation designed to withstand the design basis tornado and tornado-generated missiles. For additional description of the pump house, see Section 3.8.4.1.6. The makeup water pump house is supported on two types of foundations. The eastern end of the pump house is supported by a deep, square pump pit which in turn is supported on a reinforced-concrete mat foundation having a thickness of 3 ft. The remainder of the building is supported on continuous reinforced-concrete wall footings.

3.8.5.2 Applicable Codes, Standards, and Specifications

This section lists the codes, specifications, standards of practice, regulatory guides, and other accepted industry guidelines which are adopted to the extent applicable in the design and construction of the foundations for Seismic Category I structures. Modifications to the foundations may use the latest editions. To eliminate repetition, these codes, standards, and specifications are described and discussed in Table 3.8-4 and given a specification reference number. Listed below are the reference numbers for the foundations.

a. 1A through 9,
b. 11 through 17,
c. 25 and 28,
d. 32 through 36, and
e. 38, 41, 42, 43, 45, 49, 52, 53 and 54.

In addition, the "Supplementary Soils Investigation" prepared for Columbia Generating Station by Shannon & Wilson, Inc., soil consultants, is applicable to the design of the foundations (Reference 3.8-20).

3.8.5.3 Loads and Loading Combinations

The loads and loading combinations listed, defined, and discussed in Section 3.8.4.3 are applicable to the design of the foundations. Hydrostatic loads from the flood of record are not applicable to this installation as discussed in Section 3.4.

3.8.5.4 Design and Analysis Procedures

3.8.5.4.1 General

The analysis of Seismic Category I and non-Seismic Category I safety-related foundations is done using conventional elastic techniques. Seismic response coefficients used in the determination of loads applied to the foundations have been determined by mathematical models of the structures. All loads, interior and exterior to the structures, are transferred to their respective foundations through the elastic deformation of slabs, bearing and shear walls and columns. For additional discussion of the design and analysis procedures of the buildings, see Section 3.8.4.4.

The foundations are supported on compacted structural backfill. For discussion of site geology and soils characteristics and criteria, see Section 2.5. All foundation bearing pressures are within the allowables set forth in Appendix 2.5E. Maximum settlements based on allowable bearing pressures on the soil are tabulated in Table 5 contained in Appendix 2.5E.

Structural design of all foundations is in accordance with ACI 318-71.

3.8.5.4.2 Reactor Building Foundation Mat

The reactor building foundation mat is analyzed and designed by the finite element method utilizing NASTRAN, which is an accepted finite element computer program for static and dynamic structural analysis and is discussed in Section 3.12. The exterior building walls and the biological shield wall transmit the lateral loads and forces and overturning moments to the mat foundation, and are included in the model to account for their stiffening effect on the mat.

3.8.5.4.3 Radwaste and Control Building

The radwaste and control building mat foundation is analyzed and designed as a beam on an elastic foundation based on Abbett's "American Civil Engineering Practice" (Reference 3.8-18). The mat is divided into three strips, in each direction, with each strip considered as a beam. The lateral loads and forces acting on the structure are transmitted to

the mat foundation through interior shear walls, and the exterior, perimeter foundation walls. The mat beam strips are computer analyzed by FLXMAT, a proprietary computer program developed by Burns and Roe, Inc., for the analysis of flexible mat foundations on elastic foundations, which is discussed in Section 3.12.

3.8.5.4.4 Diesel Generator Building

Building wall footings and column spread footings are conventionally analyzed utilizing static and seismic loads and forces determined in the mathematical model seismic analysis of the superstructure in combination with other loads. The diesel generator foundations are analyzed utilizing the operating static and dynamic loads determined by the respective diesel generator manufacturers. The diesel generator foundations are seismically analyzed utilizing seismic response coefficients determined by mathematical model analysis.

3.8.5.4.5 Standby Service Water Pump Houses, Spray Ponds, and Condensate Storage Tank Retaining Area

The standby service water pump house building foundations are conventionally analyzed, utilizing seismic loads and forces determined in the mathematical model seismic analysis of the superstructure, in combination with other loads.

The spray pond bottom slab (nominally 7-in. thick) is considered to be a sufficiently flexible structure relative to the underlying supporting soil to essentially follow the displacements and deformations of the soil surface during a seismic event. The horizontal seismically induced shears at the points of pipe supports are resisted by horizontal plate action of the bottom slab which is horizontally constrained by the perimeter retaining wall footings. The spray pond walls are conventionally analyzed as rigid retaining walls subjected to static and dynamic lateral earth pressures.

The bottom slab of the condensate storage tank retaining area is a conventionally analyzed and designed mat foundation subjected to static and seismically induced loads acting on the storage tanks and the perimeter dike walls. The perimeter dike walls are analyzed and designed as vertical cantilever walls keyed and doweled to the thickened edge of the bottom slab mat foundation. The horizontal loads on the walls consist of hydrodynamic forces induced by the retained water within the diked area during a seismic event causing a failure of the condensate tanks, static water pressure of the contained water and the seismic forces acting on the wall itself.

3.8.5.4.6 Turbine Generator Building

The turbine generator building mat foundation is analyzed and designed as a beam on an elastic foundation based on Abbett's "American Civil Engineering Practice" (Reference 3.8-18). The mat is divided into three longitudinal beam strips covering the entire width of the building and

three representative transverse beam strips. The vertical and lateral loads and forces are transmitted to the mat foundation by perimeter foundation walls, interior shear walls, and columns through shear keys and shear friction reinforcement. The mat beam strips are computer analyzed by FLXMAT, a proprietary computer program developed by Burns and Roe, Inc., for the analysis of flexible mat foundations on elastic foundations and is discussed in Section 3.12.

3.8.5.4.7 Makeup Water Pump House

The mat foundation at the bottom of the pump pit and the continuous wall footings are conventionally analyzed. The mat foundation is analyzed as a two-way slab supported on four sides by the pump pit walls.

3.8.5.5 Structural Acceptance Criteria

Foundations are designed in compliance with ACI 318-71 and satisfy the strength and the serviceability requirements specified therein. The structural acceptance criteria for reinforced concrete described in Section 3.8.4.5.1 is applicable to the foundation designs.

The factors of safety against overturning and sliding for safety-related structures are as follows:

Safety-Related Structure	Safety Factors Overturning	Sliding
Reactor building	1.50	1.80
Radwaste and control building	3.13	2.29
Turbine generator building	6.80	2.70
Standby service water pump houses	1.24	2.30

Uplifting of foundation mats occurs for the Seismic Category I structures listed below. Maximum uplift occurs as a result of the combined effects of horizontal and upward SSEs. The maximum uplift conditions are described below.

a. Reactor building - Under maximum uplift, 48% of the mat maintains bearing contact with the soil. Maximum upward deflection is 1.1 in.;

b. Turbine generator building - Under maximum uplift, 89% of the mat maintains bearing contact with the soil. Maximum upward deflection is 0.10 in.; and

c. Radwaste and control building - Under maximum uplift, 84% of the mat maintains bearing contact with the soil. Maximum upward deflection is 0.14 in.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The materials, quality control, and special construction techniques for foundations conform to those set forth for Seismic Category I structures and are discussed in Section 3.8.4.6.

3.8.5.7 Testing and Inservice Surveillance Techniques

Tests to evaluate compaction of the Quality Class I structural backfill were performed by determining the relative densities of the compacted soil. Laboratory tests were made in accordance with ASTM D2049 which specifies the use of a vibrating table operating at a stipulated vibrating amplitude. In tests executed for this installation, it was found that the vibrating amplitudes of the tables used were lower than specified in ASTM D2049. Shannon & Wilson, Inc., soil consultants, investigated the relative densities of the compacted soil utilizing various vibrating tables and concluded that the relative density values are valid. The Shannon & Wilson report (Reference 3.8-19) covers their investigation of the foregoing and was submitted to the NRC by Energy Northwest.

In 1984 ASTM replaced ASTM D2049 with ASTM D4253 and D4254. These procedures are equivalent. All three standards are considered acceptable for use.

Routine observations were made of the foundations to determine the extent of cracking and/or imperfections.

Inspections were made during construction in conjunction with the quality control procedures for the structural materials as stated in Section 3.8.4.6. Structural integrity and/or performance tests, other than those described above, were not conducted.

3.8.6 PIPING AND ELECTRICAL PENETRATIONS

To maintain containment integrity, penetration assemblies which penetrate the steel primary containment vessel have the following characteristics:

a. They are capable of withstanding the peak pressures and temperatures encountered during all operating and testing modes,

b. They are capable of accommodating the thermal and mechanical loads which may be encountered during all modes of operation without failure,

c. They are capable of withstanding the forces caused by impingement of fluid from the rupture of the largest local pipe or connection without failure,

d. They are capable of withstanding the maximum piping reactions due to dead weight, seismic excitation, constraint to thermal expansion and other mechanical flow induced effects that may be exerted by the piping, to which they are attached, and

e. To account for the effects of postulated pipe rupture, the penetration assemblies are capable of withstanding the maximum reactions that the pipes to which they are attached are capable of exerting.

The quantities, types, service identifications, sizes, and pertinent data of the primary containment vessel penetrations are given in Figures 3.8-51 and 3.8-52.

The capability of the primary steel containment vessel penetrations to withstand the hydrodynamic effects resulting from the actuation of SRVs and specified loads associated with postulated LOCAs is discussed in Appendix 3A.

3.8.6.1 Description

3.8.6.1.1 Piping Penetrations - Type 1

Process lines traverse the boundary between the inside of the steel primary containment vessel and the outside of the biological shield wall by means of piping penetration assemblies made up of several elements. Two general types of piping penetration assemblies are provided: type 1 (also referred to as "hot" type piping penetration assemblies) and type 2 (also referred to as "cold" type piping penetration assemblies).

Figure 3.8-54 shows a type 1 piping penetration assembly. The type 2 penetration assembly is described in Section 3.8.6.1.2.

All piping is generally attached directly to the penetration nozzle. However, hot piping and multiple piping penetrations pass through the nozzle as type 1 and type 5 penetrations, respectively. The type 5 penetration is described in Section 3.8.6.1.5.

Type 1 penetration assemblies which penetrate the primary containment vessel consist of:

a. Penetration nozzle,
b. Flued head fitting,
c. Process pipe, and
d. Guard pipe, when required.

In all type 1 penetrations, containment closure is accomplished by means of the flued head fitting. The flued head fitting is located at the outer end of the nozzle, guard pipe and process

pipe, at a suitable distance external to the primary containment vessel and biological shield wall. At this location, the flued head fitting is welded to the nozzle. With the exception of the main steam penetrations, the portion of the process pipe, which is within the penetration assembly, is an integral part of the forged flued head fitting thus eliminating pipe welds which would be inaccessible for inservice inspection (See Figure 3.8-54). The main steam fluid heads are constructed from a flued head forging welded to a length of process pipe such that the welded assembly becomes a type 1 penetration (See Figure 3.8-51, Note 7). The flued head to process pipe weld is accessible for inservice inspection.

The inner end of the nozzle is welded to the reinforcing penetration insert plate, which is part of the primary containment vessel shell. The penetrations are designed for long-term integrity.

The guard pipe is provided between the nozzle and the process pipe to form an additional annulus between the nozzle and the process pipe. The additional annulus minimizes thermal stresses at the primary containment vessel shell during normal operation. The guard pipe also prevents direct impingement of hot fluid on to the primary containment vessel shell in the unlikely event of pipe rupture within the penetration, thereby minimizing thermal shock loading. The guard pipe is guided radially, at its inner end, near the primary containment vessel shell, by means of an enveloping sleeve and shims in the annulus between it and the nozzle, to limit seismic movement. Guard pipes are generally furnished with the type 1 penetration assemblies only when the operating process pipe temperature exceeds 300°F.

Pipe insulation is provided when required in the annulus between the process pipe and the guard pipe to reduce thermal stresses and heat losses.

The piping design includes the effects of seismic and thermal motion of the primary containment vessel shell at the penetration connections. Bellows type seals are not used for CGS.

Penetration analysis includes verification of the containment adequacy of the penetrations and of the primary containment vessel in the vicinity of the penetrations. The rupture loads are applied at the outboard interface of the flued head fitting and the process pipe. Analyses are performed to demonstrate that if local contact between the nozzle and the biological shield wall sleeve is possible, the nozzle is capable of maintaining containment.

No equipment or piping is anchored to the biological shield wall. However, lateral restraint assemblies are provided outside the primary containment vessel shell at certain of the type 1 penetrations to provide, where required by a rupture analysis of the penetration nozzle on the vessel side, containment integrity for all postulated maximum nozzle loads due to a self-break of the piping (a guillotine break or side break) as well as an adjacent pipe break (impingement).

The restraint assemblies are in the form of coaxial extensions of and attached to the steel pipe sleeves in the biological shield wall. The steel pipe sleeves are fully anchored into the biological shield wall.

The design loadings and the various load combinations, as well as the allowable stress intensities, are discussed in Sections 3.8.6.3 and 3.8.6.5, respectively.

3.8.6.1.2 Piping Penetrations - Type 2

Process lines for low energy (or cold) fluids utilize penetration assemblies without flued head fittings, to penetrate the primary containment vessel shell. Type 2 penetration assemblies are illustrated in Figure 3.8-55.

The type 2 penetration assemblies consist of a process pipe which also acts as the penetration nozzle in the vicinity of the primary containment vessel wall. The inner end of the nozzle is welded to the reinforcing penetration insert plate, which is a part of the primary containment vessel shell. The piping configuration and supports on both sides of the penetration are designed to preclude overstressing of the steel primary containment vessel nozzle under any loading condition, including postulated accidents.

Five TIP guide tubes pass from the reactor building to the drywell through the primary containment vessel. The penetrations are a type 2 piping penetration modified with a welding neck flange attached outside the containment. This flange is itself modified with dual concentric O-ring grooves machined into the face, which retain elastomeric O-rings. To this is bolted a blank flange which has been drilled for both a between-O-ring test port and a central hole in which an instrument tubing "bulkhead union" fitting is retained. This single penetration point is sealed by seal welding between the bulkhead union and the blank flange. The TIP guide tubes are attached to both sides of their respective bulkhead unions by flare fittings. These penetrations are also discussed in Section 3.8.2.1.1.3.

The design loadings and various load combinations, as well as the allowable stress intensity, are discussed in Sections 3.8.6.3 and 3.8.6.5, respectively.

3.8.6.1.3 Piping Penetrations - Type 3

The primary containment vessel is furnished with a number of spare penetrations of various sizes. These penetrations have a specific possible future use, and may be used as a type 1 or type 2 process pipe penetration or, as an electrical or instrumentation penetration. The type 3 penetration assembly is illustrated in Figure 3.8-55.

Type 3 penetration assemblies consist of a nozzle which penetrates the primary containment vessel and extends outboard beyond the biological shield wall. At the latter location, the

nozzle is furnished with a permanent closure. The inner end of the nozzle is welded to the reinforcing penetration insert plate which is part of the primary containment vessel shell.

Although the type 3 spare penetrations have a specifically intended future use, they are designed so that they can be utilized as any type penetration having more severe loads. Thus, a type 3 penetration which is intended to be a future type 2 penetration is designed for loadings which may be incurred by a type 2 penetration and for loadings which may be incurred by a type 1 penetration. A stipulated pipe rupture load is included in the analysis and is as shown in Figures 3.8-51 and 3.8-52.

The design loadings and various load combinations that may be incurred by either a future type 2 or a future type 3 penetration, as well as the allowable stress intensity, are discussed in Sections 3.8.6.3 and 3.8.6.5.

3.8.6.1.4 Electrical Penetrations - Type 4

The electrical penetration assemblies provide a means for the continuity of power, control and signal circuits through the primary containment vessel while maintaining the leaktight integrity of the vessel. Each penetration assembly is designed to function in the environmental conditions specified in Table 3.8-13. Electrical penetrations are categorized into three classes. Each of the three classes of penetration assemblies is designed to meet the requirements of a particular group of electrical cables (see Reference 3.8-15). The three general classes are

a. Signal penetration assemblies for shielded signal cables for circuits that require high signal integrity (neutron monitoring penetrations), penetrations X-100A, B, C and D;

b. Low voltage penetration assemblies for power, control or instrumentation leads. Penetrations series X-101, X-102, X-105, X-104, and X-107; and

c. Medium voltage penetration for power cables. Penetrations X-103A, B, C, and D.

The basic configuration of each of these three classes of penetration assemblies is a completely enclosed, sealed unit. Each penetration assembly is designed and fabricated so as to permit leak testing of the pressure barrier from a single point outside the primary containment vessel. (See Figures 3.8-56 and 3.8-57.) The penetration assemblies are designed considering the size and physical parameters of the containment vessel penetration nozzles.

The two basic types of electrical penetration assemblies are as follows:

a. The canister type as shown in Figure 3.8-56, and
b. The unitized header plate or noncanister type as shown in Figure 3.8-57.

The canister type electrical penetration (assembly series X-103) is used for medium voltage power cables. The penetration assembly extends the length of the containment vessel penetration nozzle. A plate at each end of the penetration canister provides double barriers between which can be pressurized and monitored for leak testing purposes.

The plates are constructed of stainless steel to eliminate eddy current heating. The containment vessel pressure boundary is established by welding the penetration assembly to the outboard end of the containment vessel penetration nozzle. A weld ring is attached to the inboard end of the containment vessel penetration nozzle to provide lateral support to the penetration assembly. Bolt-on terminal boxes are attached to the inboard weld ring and to the outboard end of the penetration assembly.

The remaining electrical penetration assemblies, series X-100, X-101, X-102, X-104, X-105, and X-107 are the unitized header assembly type. The unitized header assembly consisting of the bulkhead extension and monitoring plate is welded to the outboard end of the containment vessel penetration nozzle. Electrical penetration of the pressure boundary is achieved by the insertion of sealed modules into the monitoring plate. The modules are designed with two sets of dual O-rings between which sets can be pressurized and monitored for leak testing purposes. Modules may be substituted or added to the unitized header assembly as necessary in the future.

Bolt-on terminal boxes are attached to the unitized header assembly on the outboard side of the containment vessel and to a slip-on flange welded to the inboard side of the containment vessel penetration nozzle.

A cable support tray is provided inside the containment vessel penetration nozzle.

3.8.6.1.4.1 Configuration.

a. Neutron monitoring penetration

   Shielded signal cables are provided to interconnect low noise circuits between the reactor and the associated control room. Two types of signal circuits are required: Type A service consists of reactor neutron monitoring circuits with a dc analog signal (power range neutron monitoring); type B service consists of reactor neutron monitoring circuits with a pulse output signal (startup neutron monitoring);

   For type A service, the wire within the penetration module consists of, at a minimum, No. 18 AWG wires that are connected to the external wiring with insulated crimp splices;

For type B service, the cable within the penetration matches those to which it is connected. The source range monitor (SRM)/intermediate range monitor (IRM) cables consist of two shields which are connected together at the connectors to form a common shield for each cable. Inline connectors mounted on the side of the terminal boxes are used for cable terminations. The connectors are insulated from the penetration.

The concentric geometry of shielded cables is maintained through the penetration assembly without interruption to either shield or conductor.

Type B connectors are insulated from ground by at least 10^8 ohms at room temperature. The insulation resistance, between the center conductor and the shield, is greater than 10^{12} ohms at room temperature;

b. Low voltage control and indication penetration

The low voltage control and indication penetration assemblies (series X-101, X-102, and X-105) are suitable for 600-V ac (or below), whose circuits are designed to supply control power for the plant auxiliary systems.

Cable connections (see Reference 3.8-15 are as follows: (1) multipin connectors mounted on the header plate or box, (2) pigtail with inline pin connector, (3) pigtail and crimp type connector to be routed to a terminal box, or (4) pigtail to be routed to a splice.

Where connectors are furnished, they are the environment-resistant type in accordance with MIL-C-5051D. The insulation resistance between conductors is not less than 10^8 ohms at room temperature. All receptacles and pin contacts for No. 16 AWG and smaller wires are insertion pin types either soldered or gold plated crimp-on;

c. Low voltage power penetration

The low voltage power penetration assembly (X-104 series) is suitable for low voltage, 600-V ac and 250-V dc. The cables used within the penetration have their copper conductors and insulations sized in accordance with the conditions specified in Section 3.8.6.5.4.1 (b).

Wire is switchboard type SIS, stranded, tinned copper with 1000 V, 90°C insulation.

Insulation resistance between conductors is not less than 10^8 ohms at room temperature. The header plates are constructed of a suitable material to minimize eddy current heating. Power penetrations are provided with two copper-constantan thermocouples for temperature monitoring;

d. Medium voltage power penetration

The medium voltage power penetration assembly, (X-103 series) is suitable for three, 8-kV high resistance grounded cable.

The cables used inside the penetration assembly have their copper conductors and insulators sized in accordance with the conditions specified in Section 3.8.6.5.4.1 (c).

Cable terminations are either a mechanical splice (crimp-on lug type) or bushings mounted on the headerplates. Basic impulse level (BIL) is 95 kV. The insulation resistance between conductors is 10^{10} ohms at room temperature. These penetrations are provided with copper-constantan thermocouples for penetration temperature monitoring.

Specific provision is made in the design of the 8-kV penetration assembly to avoid electrical stress on the cable insulation. No part of the shielded cable insulation comes in contact with the ground plane either at the header or in the assembly. Holes in the header have well-rounded edges and are large enough to provide adequate supplementary insulation or bushing for the cable.

The ends of the canister are factory welded to nonmagnetic stainless steel header assemblies containing pressure-tight high-alumina ceramic seals;

e. Low voltage power, control and indication penetration

The low voltage power, control, and indication penetration assembly (X-107 Series), is suitable for 600-V ac and 250-V dc power cables, and 600-V control cable. The cables are used for 120-V ac (or below) circuits that are designed to supply control power for the plant auxiliary systems inside of the suppression chamber.

The cables used within the penetration have their copper conductors and insulations sized in accordance with the conditions specified in Section 3.8.6.5.4.1.

The continuity of shielded cables is to be maintained through the penetration assembly using single, conductor without interruption to either conductor or shield.

Cable connections (for Reference 3.8-15) are as follows: (1) multi-pin connectors mounted on the header plate or box, (2) pigtail with inline pin connector, (3) pigtail and crimp type connector to be routed to a terminal box, or (4) pigtail to be routed to a splice (for drywell wetwell vacuum breakers).

Where connectors are furnished, they are the environment-resistant type in accordance with MIL C-5015D. The insulation resistance between conductors is not less than 10^8 ohms at room temperature. All receptacles and pin contacts for No. 16 AWG and smaller wires are insertion pin types either soldered or gold plated crimp-on.

The header plates are constructed of nonmagnetic austenitic stainless steel to minimize eddy current heating.

3.8.6.1.4.2 Ampacity. The cables through the electrical penetration assembly are sized to ensure the following:

a. Adequate cable current carrying capability (ampacity) of no less than the values indicated in Reference 3.8-15, and

b. After each penetration assembly is installed and operating continuously during normal environmental conditions, the temperature of the concrete adjacent to the containment vessel penetration nozzle does not exceed a maximum of 150°F along the length of the penetration when operating at 100% rated current load on all conductors.

3.8.6.1.4.3 Auxiliary Hardware.

a. Temperature monitoring: Each power penetration (series X-104 and X-103) assembly has two internal temperature monitoring, copper-constantan thermocouples to monitor the temperature within the assembly.

b. Terminal boxes: Terminal boxes and their mounting terminal blocks, connectors and terminal lug connectors are provided at both ends of each penetration. Terminal boxes for unitized header type penetrations are designed to permit installation of the penetration with only the removal of the outer (or front) enclosure cover. The terminal boxes are type National Electrical Manufacturers Association (NEMA) 4 boxes with an environment resistant coating. Boxes are arranged to avoid interference with adjacent penetrations.

All terminal boxes located inside of containment have holes punched in the box to allow equalization of pressure outside to inside the box during a LOCA so that the box will not implode. Plugs are installed in the holes of the boxes to help keep moisture out. These plugs are designed to push into the box during a LOCA. In addition, holes are punched in the bottom of the boxes, as required for moisture drainage.

c. Cables: Cables within penetration assemblies and/or nozzles are restrained by supporting structures to prevent cables from excessive motion due to high electrodynamic or mechanical forces. Cable supports also minimize compressive loading of cable insulation.

d. Radiation shielding: Provision is made for fastening and supporting up to 150 lb of radiation shielding on the inside end of the penetration assembly.

3.8.6.1.4.4 Shop Painting of Electrical Penetrations. All ferrous metal surfaces, interior and exterior, are furnished with a coating to withstand the environmental conditions listed in Table 3.8-13.

3.8.6.1.5 Instrumentation Piping Penetrations - Type 5, 6, and 7

Figure 3.8-58 illustrates the instrumentation penetration assembly. The penetration assembly consists of a penetration nozzle and lengths of instrumentation pipes. The instrumentation pipes pass through the penetration nozzle from the reactor building side, through the containment vessel shell, to the interior of the containment vessel. The inner end (containment vessel end) of the nozzle is welded to the reinforcing penetration insert plate the penetration are designed to preclude over-stressing the steel primary containment vessel nozzle under any loading condition, including postulated accidents.

The penetration of the instrumentation pipes through the primary containment nozzle end plates is sealed by welding which meets the requirements of the ASME Code Section III, Subsection NE, Class MC components.

3.8.6.2 Applicable Codes, Standards, Specifications, and Regulatory Guides

3.8.6.2.1 Codes, Standards, and Specifications

Piping and electrical penetration assemblies are designed and constructed to meet the requirements of the following applicable codes, standards and specifications.

a. The following sections of the ASME B&PV Code, 1971 Edition, including Summer 1972 Addenda and all previous addenda (ASME Code):

Section II, Material Specifications, Part A - Ferrous (ASME Code Section II)

Section III, Nuclear Power Plant Components, Subsection NE, Class MC Components (ASME Code Section III)

Section V, Nondestructive Examination (ASME Code Section V)

Section IX, Welding Qualifications (ASME Code Section IX);

b. Steel Structures Painting Manual, Volume 2, Systems and Specifications, 1964 Edition with the 1968 Supplement and the January 1971 Editorial changes.

Specifications SSPC-SP-6, SP-8 and SP 10;

c. American Society for Testing and Materials

A370 - Standard Methods and Definitions for Mechanical Testing of Steel products

A380 - Recommended Practice for Descaling and Cleaning Stainless Steel Surfaces;

d. Institute of Electrical and Electronics Engineers

IEEE 317-1972 IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations

[NOTE: For the replacement of the electrical penetration modules, use the version of IEEE 317 that is in effect at the time of purchase and documented in the design specification.]

IEEE 336-1971 - Installation, Inspection and Testing Requirements for Instrumentations and Electrical Equipment During the Construction of Nuclear Power Generating Stations

IEEE 344 - (Trial Use) - Guide for Seismic Qualifications of Class I Electric Equipment for Nuclear Power Generating Stations;

e. American Society for Testing and Materials

D635 - Test for Flammability of Self-Supporting Plastics (Fire Resistant);

f. Insulated Power Cable Engineers Association (IPCEA) and NEMA:

S-61-402 - (IPCEA) Standard for Thermoplastic - Insulated Wire WC-5-1968 - (NEMA) - Cable for the Transmission and Distribution of Electrical Energy

S-19-81 - (IPCEA) - Rubber Insulated Wire & Cable for Transmission and Distribution of Electrical Energy

S-66-524 - (IPCEA) - Standard for Cross-Linked Thermosetting

WC-7-1971 - (NEMA) - Polyethylene-Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy

S-68-561I - #1 - Cables Rated 0-35000 Volts Having Ozone-Resistant Ethylene-Propylene-Rubber Insulation

P-32-382 - (IPCEA) - Short Circuit Characteristics of Insulated Cable

S-68-561I - #2 - Cables Rated 5000 Volts and Less and Having Ozone-Resistant Ethylene-Propylene-Rubber Integral Insulation and Jacket

For electrical penetration assembly, use the applicable IPCEA and NEMA Standards that are in effect at the time of purchase;

g. American National Standard for Temperature Measurement Thermocouples C96.1-1964;

h. Military Specification Electric Connector AN Type

MIL-C-5051D - Electric Connector, AN Type;

i. Underwriters Laboratories Standards (UL); and

j. National Electric Code, 1975.

The nozzle, which is part of the containment boundary, is classified as Class MC and designed in accordance with Subsection NE, Section III of the ASME Code. The Type I penetration guard pipe, which is not part of the containment boundary, is designed, tested, and of materials selected in accordance with Subsection NE, Section III of the ASME Code. The part of the

process pipe which is inside the guard pipe and the flued head fitting is of the same code class as the remainder of the process pipe.

3.8.6.2.2 Conformance with Regulatory Guides

The applicable regulatory guides are discussed in Section 3.8.2.2.4.

3.8.6.3 Loads and Loading Combinations

3.8.6.3.1 Loads

The analysis and design of piping and electrical penetrations consider the following loads:

a. Load (a) - Pressure and temperature due to LOCA,

b. Load (b) - Live and dead loads, including the primary loads applied to the containment vessel directly as well as dead weight, seismic inertia loads, and other mechanical and flow induced loads from attached pipes and equipment,

c. Load (c) - OBE,

d. Load (d) - Thermal expansion load, including loads caused by constraint due to thermal expansion of the primary containment vessel, as well as reactions from attached piping due to constraint from thermal expansion of the piping system,

e. Load (e) - SSE, and

f. Load (f) - Jet force or pressure on structures and equipment resulting from a pipe break.

Load (f) includes guillotine type break or longitudinal split of the piping in question, or pressure impingement from adjacent piping. Load (f) for penetration assemblies is tabulated in Figure 3.8-51 under the heading "Flued Head Fitting - Pipe Rupture Loads."

Load (b) includes jet forces resulting from a normal operating effluent discharge of an open-ended pipe.

3.8.6.3.2 Loading Combinations

The loading combinations (of the loads defined in Section 3.8.6.3.1) used in the design of penetrations are as follows:

a. Normal accident: loads (a) + (b) + (c) + (d) (normal condition),

b. Maximum seismic: loads (a) + (b) + (d) + (e),
c. Pipe rupture: loads (a) + (b) + (c) + (d) + (f), and
d. Pipe rupture plus maximum seismic: loads (a) + (b)+ (d) + (e) + (f).

Allowable stresses are discussed in Section 3.8.6.5.4.

All process pipe penetrations, including spare penetrations intended for future process piping, are analyzed for the load combinations defined above.

Electrical penetration assemblies are designed to withstand environmental conditions present during a postulated LOCA including the jet impingement force from a ruptured pipe to the junction boxes. The assemblies are designed to maintain containment integrity for extended periods of time in a postaccident environment.

Electrical penetrations, including penetration nozzles intended for future electrical penetration assemblies, are analyzed for a pipe rupture impingement load [load (f)].

3.8.6.4 Design and Analysis Procedures

3.8.6.4.1 Piping Penetrations

Loads on the piping penetrations due to thermal expansions of the pipes, thermal and pressure movements of the primary containment vessel, and the piping system weight are determined by a flexibility analysis of the piping system and are discussed in Section 3.9.

Seismic loads on piping penetrations are determined by the method described in Sections 3.7 and 3.9. Jet loads on nozzles are analyzed in accordance with Section 3.6.

3.8.6.4.2 Flued Head Fitting Design

The design of the flued head fittings is in accordance with both the ASME Code Section III for Class MC as well as for the same ASME Code Class shown in Figure 3.8-51 for process pipe. The flued head fitting to penetration nozzle weld is excluded from the limit of process pipe classification, and is a Class MC Weld. The flued heads are designed to withstand the loads and combinations thereof noted in Figure 3.8-51, including the containment design pressures and temperatures as noted in Table 3.8-14. The configuration of the flued head is such as to minimize stress concentrations by using gradual transitions from one thickness to another and by shaping the fitting so as to avoid any sudden changes in contour.

The flued head fitting contouring at the process pipe interfaces and the penetration nozzle interface is in accordance with the ASME Code, Section III, Figure NB-4233-1, except that the initial maximum slope of the outside surface adjoining the process pipe is 1-on-4 in lieu of the

1-on-3 slope shown in Figure NB-4233-1. The exterior shape of the fittings is as shown in Figure 3.8-54.

3.8.6.4.3 Thermal Stress Analysis for Flued Head Fittings

Thermal stress due to steady-state and/or transient condition is accounted for.

3.8.6.4.4 Primary Containment Vessel Design at Penetration Nozzle Interface

Influence coefficients are provided for all primary containment vessel penetration nozzles resulting from unit forces and moments applied at the intersection of the centerline of the penetration and the neat inside face of the containment vessel (such as W.P. 1 in Figure 3.8-54). These influence coefficients represent local shell membrane stresses and shell bending stresses at the juncture of the nozzle to shell reinforcing insert plate and the juncture of the reinforcing insert plate to vessel shell plate for unit forces and moments. The primary containment vessel stresses due to applied forces and moments are derived by use of the influence coefficients and proper combination at specified locations. The vessel is designed and analyzed for all loading combinations and satisfies the following design requirements:

a. Penetration assemblies are designed in accordance with the ASME Code Section III, Subsection NE for MC Components. Process piping is designed in accordance with the ASME Code Section III, Subsection NB or NC as required for Class 1 or Class 2 components. The ASME Code Class for each process pipe is indicated in Figures 3.8-51 and 3.8-52;

b. The type, size, and thickness of the penetration nozzle conforms to those indicated in Figures 3.8-51 and 3.8-52. Penetration reinforcement is generally a single plate insert;

c. Loads and loading combinations as described in Section 3.8.6.3;

d. The flued head fittings meet Class MC requirements, as well as the requirements of the ASME Code class specified for the attached process pipes. The flued head fittings are integral with the process pipe at the inboard side of the fitting (W.P. 3 in Figure 3.8-54).

   The flued head fittings are capable of transmitting all applicable loads from the process pipe to the penetration nozzle. Pipe rupture load (f) in Section 3.8.6.3.1 is considered as primary load and is applied at the inner and outer process pipe flued head fitting interface (W.P. 3 and W.P. 2 in Figure 3.8-54) separately.

The flued head fittings are designed in accordance with the allowable stress criteria in Section 3.8.6.5 and meet the allowable stress criteria of the applicable ASME Code Class. When ASME Code Class I is applicable, emergency condition loading criteria is applied for load combination (b) and (c) in Section 3.8.6.3.2 at all locations where the structure can be considered integral and continuous; and upset condition loading criteria is applied at locations where the structure cannot be considered integral and continuous. When emergency condition loading criteria is applicable, the design is based on elastic analysis. Limit analysis is not used.

The analytical procedures utilized for flued head fitting design was performed by means of "ISOFINITE" (see Section 3.12) using the three-dimensional finite element method. The analysis was performed for the normal and emergency conditions described previously. "ISOFINITE" calculates six stress components at the centroid of each element. These stress components are then converted into principal stress and stress intensities. The stress intensities are then compared to 1.5 S_m. The sum of the principal stresses are likewise compared to 1.5 S_m. For maximum primary plus secondary stress intensity, the allowable stress criteria is 3 S_m.

All the flued head fittings satisfy the design criteria of the ASME Code Section III; and

e. The penetration design analysis for pipe rupture and maximum seismic loadings includes verification of containment adequacy of the penetration assemblies and the primary containment vessel in the vicinity of penetrations. For this analysis, the pipe rupture loads utilized are those shown in Figure 3.8-51 for the flued head fitting. The rupture loads are applied at the outboard interface of the flued head fitting and the process pipe (Refer to W.P. 2 in Figure 3.8-54).

Steel pipe sleeves within the biological shield wall provide an annular gap around the penetration nozzles. The analysis includes verification that, if local contact between nozzles and sleeves is possible, the nozzles have sufficient strength to maintain containment. As shown in Figure 3.8-54, lateral restraint assemblies with an annular gap are provided around the flued head fittings, when required.

The steel pipe sleeves are not within the jurisdiction of primary containment and are therefore designed to AISC criteria for Class I structures.

3.8.6.4.5 Electrical Penetrations

All penetration assemblies are a part of the containment system and are capable of meeting the requirements listed in Table 3.8-13. The design, fabrication materials, inspection, and testing of the pressure retaining parts of the penetration assembly are in accordance with the ASME Code Section III, Subsection NE, Class MC components. All assemblies are code stamped in accordance with applicable requirements of the ASME Code Section III.

The power, control and instrumentation wiring complies with the standards listed in Sections 3.8.6.2.1(f) and 3.8.6.6.1 (cable and cable insulation) as applicable to a particular cable construction.

The penetration assemblies are designed to meet the requirements of IEEE Standard 317.

Electric penetration assemblies are designed by analysis, test, and combinations thereof. Design criteria for the penetration seals are as follows:

a. For normal operation, electrical loadings causing heating or electromagnetic forces do not violate primary containment integrity or disrupt the integrity of the electrical circuit,

b. Electrical rated current loadings do not exceed the requirements of the National Electrical Code with regard given to the application of derating factors to account for the effects of ambient temperature and grouping,

c. For seismic loadings, during normal operation, primary containment and electrical circuit integrity are not violated,

d. Jet force impingement may be allowed to disrupt electrical circuits but does not violate primary containment integrity,

e. For LOCA conditions, short circuit faults may disrupt electrical circuits but do not violate primary containment integrity,

f. Motor startup currents produce no detrimental effect on circuit integrity,

g. Gamma radiation exposure produces no detrimental effect on primary containment or circuit integrity,

h. Thermal cycling due to plant startup and shutdown produces no detrimental effect on primary containment nor circuit integrity,

i. The penetration assembly has leak rate less than IEEE 317 leak rate requirements (see Section 3.8.6.7), and

j. Low voltage control, instrumentation, and power operate electrically during LOCA, as specified by the plant specification.

3.8.6.4.6 Protective Coatings

Protective coatings are applied to all exposed steel surfaces of the penetration assemblies, as discussed in Section 3.8.2.4.4.

3.8.6.5 Structural Acceptance Criteria

3.8.6.5.1 Type I Piping Penetrations

The requirements of Article NE-3000 of the ASME Code Section III, as modified by Regulatory Guide 1.57, Revision 0, and discussed in Section 3.8.2.2.4 are met for each of the load combinations and for each component of type 1 penetration assemblies. In addition, the process pipe portion of the assembly is designed to the appropriate piping code class; that is, ASME Code Section III, Code Class 1 or 2.

3.8.6.5.2 Types 2 and 3 Piping Penetrations

The requirements of Article NE-3000 of Section III of the ASME Code are met for each of the load combinations and for each component of types 2 and 3 Piping Penetrations.

3.8.6.5.3 Type 4 Electrical Penetrations and Type 5, 6, and 7 Instrumentation Penetrations

Structural acceptance criteria for the containment penetration assemblies is in accordance with the rules of the ASME Code Section III, Class MC and IEEE 317-1972.

[NOTE: If replacement of the electrical penetration modules is required, the version of IEEE-317 that is in effect at the time of purchase and documented in the design specification is used.]

3.8.6.5.4 Allowable Stresses for Piping, Electrical, and Instrumentation Penetrations

Maximum allowable stress values are in accordance with the ASME Code Section III, Paragraph NE-3131.

Doubler plates (or pads), usually continuous and attached to the pressure boundary by means of fillet welds, and junctures of nozzles to containment vessel, are considered to be structures which are not integral and continuous. Therefore, the rules of the ASME Code Section III,

Paragraph NE-3131(c) (1) are applied. An example of doubler plates or pads is the welding ring discussed in Section 3.8.2.3.3.

Locally thickened vessel shell plates having a properly tapered transition as shown in the ASME Code Section III, Figure NE 3361-1, and full penetration butt weld connections, are considered to be structures which are integral and continuous, and the rules of the ASME Code Section III, Paragraph NE-3131(c) (2) are applied.

3.8.6.5.4.1 Electromagnetic Conditions. Each penetration assembly is designed to meet continuous, short time overload, and fault current conditions (see Reference 3.8-15). The penetration assemblies meet the requirements of IPCEA Standard P32-382.

a. Low voltage control and indication penetration (Nos. X-101 Series, X-102 Series, and X-105 Series): the low voltage control and indication penetration assemblies are designed for the following electromagnetic conditions:

    1. Maximum momentary short-circuit current: 60 x rated amp RMS asym for 0.25 sec, and

    2. Cable and continuous ampacity as tabulated in Reference 3.8-15.

b. Low voltage power penetration (No. X-104 Series in Reference 3.8-15) and low voltage power control and indication penetration (No. X-107 Series in Reference 3.8-15): These two types of penetration assemblies are designed for the following electromagnetic conditions without damaging conductor and insulation:

    1. Maximum short-circuit current: 60 x rated RMS asym for 0.20 sec,
    2. Maximum starting current: 6.5 x rated amp for 15 sec, and
    3. Continuous ampacity as tabulated in Reference 3.8-15.

c. Medium voltage power penetration (No. X-103 Series in Reference 3.8-15): The medium voltage power penetration assemblies are designed for the following electromagnetic conditions without damaging conductor and insulation:

    1. Maximum continuous current: 700 amps,

    2. Maximum inrush current: 4000 amps/phase for 20 sec,

    3. Maximum short-circuits: 66,000 amps asym momentary and 44,000 amp sym. for 75 cycles.

3.8.6.6 Materials, Quality Control, and Special Construction Techniques

3.8.6.6.1 Materials

Materials used comply with the requirements of NE-2000 of Subsection NE of the ASME Code Section III.

Materials used in piping, electrical and instrumentation penetration assemblies are included in Section 3.8.2.6.1. In addition to materials in Section 3.8.2.6.1, the following materials are used in electrical penetration:

a. Conductor

Material for conductor is as specified in Reference 3.8-15 and in Section 3.8.6.1.4.1.

b. Cable insulation

Unless otherwise specified in Reference 3.8-15 and in Section 3.8.6.1.4.1, the cable insulation is of nonmetallic, organic materials as qualified by IEEE-317.

c. The inner and outer bulkheads of each penetration assembly portions of the assembly encircling individual single conductor power cables are made of nonferromagnetic material to eliminate hysteresis heating. Soft metals such as copper, brass, and aluminum are not exposed to the containment environment.

3.8.6.6.2 Quality Control

Quality control measures discussed in Section 3.8.2.6.2 apply to penetration assemblies. The measures include the vessel vendor's submitted shop and field quality compliance and quality assurance organization and procedures, material certifications, weld data, test data, and welding and testing procedures.

The quality control procedures are in accordance with the ASME Code Section III, Appendix X; IEEE-317; and Regulatory Guide 1.63, Revision 0. In summary, steel quality control begins with the selection of basic shapes and with ranges of properties and characteristics defined by industry standards. Quality control extends from testing of specimens sampled from basic shapes to fabrication, installation and joining procedures.

Nondestructive testing for penetration assemblies is in accordance with the ASME Code Section III for the applicable code class, and in accordance with supplemental requirements specified in the plant specifications.

Nondestructive testing requirements include the following:

a. 100% radiography of all welds;

b. Any weld not radiographed because of restricting geometry receive a 100% magnetic particle or liquid penetrant inspection for both root and final surface. If the weld falls into the ASME Code Section III, Category A or B, ultrasonic testing is performed; and

c. Flued head forgings are examined in the finished condition on all accessible surfaces by either the liquid penetrant or the magnetic particle methods.

3.8.6.6.3 Special Construction Techniques

No construction techniques unusual to erection methods used during the original construction were required for the penetration assemblies.

3.8.6.7 Testing and Inservice Inspection Requirements

3.8.6.7.1 Inspection of Material and Parts for Fabrication

The discussion in Section 3.8.2.7.1 applies to penetration assemblies.

3.8.6.7.2 Shop Hydrostatic Testing

Shop hydrostatic tests discussed in Section 3.8.2.7.4 apply to penetration assemblies.

3.8.6.7.3 Shop Tests on Electrical Penetration Assemblies

The following shop tests were performed on the original electrical penetrations in accordance with applicable standards:

[NOTE: If replacement of the electrical modules is required, the version of IEEE 317 that is in effect at the time of purchase and documented in the design specification is used.]

a. Flame resistant tests

All wires successfully pass the "Flame Resistant Test" specified in IPCEA publication No. S-61-402, whether or not other applicable IPCEA Standards require this test.

b. Fire resistance

The electrical penetration assembly meets the requirements of ASTM D635, Test for Flammability of Self-Supporting Plastics, as well as Underwriters Standards (UL).

c. Prototype tests

Qualification tests performed on at least one prototype electrical penetration assembly of each type, namely, low voltage penetration Nos. X-101 Series, X-102 Series, X-105 Series, X-104 Series, and X-107 Series; and medium voltage penetration No. X-100 Series, as specified in IEEE Standard 317, Paragraph 5, demonstrate the suitability of the penetrations, assemblies, design and all materials selected for use with the assemblies.

d. Production test

Prior to shipment, production tests are performed on each penetration assembly as specified in IEEE Standard 317, Paragraph 6.

e. Containment environmental tests

1. *Testing to qualify for normal operation*

   Temperature and moisture resistance: Evidence of qualification for normal operation (see Table 3.8-13) is provided by certified data which demonstrates that cable has been manufactured, tested, and has successfully passed the requirements contained in the applicable IPCEA standards, with regard to the application of derating factors for the ambient temperatures specified.

   Thermal and radiation aging: The following test sequence demonstrates that the cable is operational after exposure to the combined effect of thermal and radiation aging (Table 3.8-8):

   Test Step #1: Conditioning

   The cable under test is conditioned in a circulating air oven for 7 days at 150°C.

Test Step #2

Heat aged samples are exposed to gamma radiation from a nuclear source such as Cobalt 60 to a dosage of 1 x 10^8 rads.

Test Step #3

The samples are subjected to a combined mechanical/electrical proof test consisting of bending around a mandrel maintained at room temperature in accordance with the procedure designated in paragraph 6.19.3 of IPCEA S-19-81, and the cable is subsequently subjected to an AC voltage withstand test equal to 80% of the final factory test voltage called out in the applicable IPCEA standard.

2. *Testing to qualify for operation during the postulated design basis LOCA.*

Test Step #1

A new set of samples is subjected to heat aging as specified in Section 3.8.6.7.3(e) under thermal and radiation aging.

Test Step #2

Test samples are subjected to a radiation dosage of 1 x 10^8 rads.

Test Step #3

Test samples are subjected to environmental conditions as encountered in a LOCA.

The conditioned samples are placed in a pressure vessel so constructed that cables can be operated under rated voltage and load while exposed to the pressure, temperature, and humidity specified in Table 3.8-13.

After conditioned samples are installed inside the pressure vessel, they are energized at rated voltage and loaded with current to the level specified herein. Then the samples are exposed to the environmental extremes specified in Table 3.8-13 and function properly throughout this exposure to environmental extremes.

3. Sampling

The samples tested contain the conductor, insulation, fillers, jacket, binder tape, overall jacket, shielding, which are representative of the cable category being qualified. Table 3.8-8 lists sizes which are considered representative of the categories of cable supplied for electrical penetration assemblies. The sample lengths are sufficient to permit reliable test readings and evaluation consistent with accepted testing practice.

3.8.6.7.4 Field Tests on Electrical Penetration Assemblies

a. Field tests

Field tests made on electrical penetrations in accordance with IEEE 336 demonstrate that the material and equipment meets the specified performance.

b. After installation

All electrical penetration assemblies are installed as an integral part of the primary containment system and meet the requirements of IEEE 317, Paragraph 7.

3.8.6.7.5 Testing of Penetrations After Erected

On completion of field erection of the primary containment vessel and prior to installation of penetration internals, tests of penetrations were performed according to an established test plan discussed in Sections 3.8.2.7.3 and 3.8.2.7.4.

3.8.6.7.6 Tests on Penetration Field Welds

Leaktightness tests of field welds connecting the penetrations to the primary containment vessel, including the welds connecting the penetration nozzle to the vessel shell, are conducted as discussed in Section 3.8.2.7.6.

3.8.6.7.7 Preoperational Leakage Rate Tests and Periodic Leakage Rate Tests

A discussion of the preoperational and periodic leakage rate tests of the primary containment vessel penetrations is provided in Sections 3.8.2.7.7 and 6.2.6.

3.8.7 REFERENCES

3.8-1 ASCE Task Committee Report, Wind Forces on Structures, Paper No. 3269, Vol. 126, 1961.

3.8-2 Lockheed Aircraft Corp. and Holmes and Narver, Inc., Nuclear Reactors and Earthquakes, Div. of Reactor Development of Atomic Energy Commission, ID-70424, Washington, D.C., August 1963, Chapter 6 and Appendix F.

3.8-3 Harris, Suer, Skene and Benjamin, "The Stability of Thin-Walled, Unstiffened, Circular Cylinders Under Axial Compression Including the Effects of Internal Pressure," Journal of the Aeronautical Sciences, August, 1957.

3.8-4 Wichman, K. R., Hopper, A. G., and Mershon, J. L., "Local Stresses In Spherical and Cylindrical Shells Due to External Loadings. Welding Research Council," Bulletin No. 107, New York, NY, August 1965.

3.8-5 "Sacrificial Shield Wall," Burns and Roe, Inc., Hempstead, New York, Report No. WPPSS-74-2-R2, March 1974.

3.8-6 "Sacrificial Shield Wall Design Supplemental Information," Burns and Roe, Inc., Hempstead, New York, Report No. WPPSS-74-2-R2-A, February 1975.

3.8-7 "Sacrificial Shield Wall Design Supplemental Information," Burns and Roe, Inc., Hempstead, New York, Report No. WPPSS-74-2-R2-B, August 1975.

3.8-8 "Drywell to Wetwell Leakage Study," Burns and Roe, Inc., Hempstead, New York, Report No. WPPSS-74-2-R5, July 1974.

3.8-9 "Drywell to Wetwell Leakage Study Additional Information," Burns and Roe, Inc., Hempstead, New York, Report No. WPPSS-74-2-R5-A, February 1975.

3.8-10 ACI 318-1971, "Building Code Requirements for Reinforced Concrete," American Concrete Institute, 1971.

3.8-11 AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," American Institute of Steel Constriction, 1969.

3.8-12 ANSI N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," Draft 3, Revision 1, American National Standards Institute, January 1974.

3.8-13 Letter GI2-75-10, from W. R. Butler to J. J. Stein, Transmitting Request for Additional Information, dated January 14, 1975, on Drywell to Wetwell Leakage Study, Docket 50-397.

3.8-14 Letter GO2-76-156, from D. L. Renberger to W. R. Butler, entitled WPPSS Nuclear 2 Project No. 2, Drywell/Wetwell Leakage Study, transmitting response to request for additional information in Reference 3.8-13, dated April 23, 1976.

3.8-15 Supply System Analysis Report, "Overcurrent Protection of Primary Containment Electrical Penetrations," E/I-02-93-04 and Drawing E539.

3.8-16 Savin, G. N., Stress Distribution Around Holes, Translation of "Raspredeleniye Napryazheniy Okolo Otverstiy," Naukova Dumka Press, 1968, National Aeronautics and Space Administration, NASA TT F-607, Washington, D.C., November 1970.

3.8-17 Roark, R. J., Formulas for Stress and Strain, Fourth Edition, McGraw-Hill Book Company, Inc., New York, 1965.

3.8-18 Abbett, R. W., American Civil Engineering Practice, John Wiley and Sons, Inc., New York, 1956.

3.8-19 Shannon and Wilson, Inc., Soil Compaction Evaluation of Quality Class I Backfill, Washington Public Power Supply System, WPPSS Nuclear Project No. 2 (WNP-2).

3.8-20 Shannon and Wilson, Inc., Supplementary Soils Investigation, Washington Public Power Supply System, Hanford No. 2 Nuclear Power Plant, Central Plant Facilities, Benton County, Washington, July 28, 1972.

3.8-21 "Primary Containment Vessel for Washington Public Power Supply System, Hanford No. 2, Jet Impingement Analysis," FIRL Technical Report F-C14121, May 21, 1975.

3.8-22 "HYBOS," FIRL Users Manual, July 1973.

3.8-23 Engineering Evaluation of the Sacrificial Shield Wall, submitted to the NRC with WPPSS letter G02-80-172, August 8, 1980.

3.8-24 Engineering Evaluation of the Sacrificial Shield Wall, Supplement No. 1, submitted to NRC with WPPSS letter GO2-80-182, August 19, 1980.

Table 3.8-1

Primary Containment
Drywell and Pressure Suppression Chamber
Principal Design

Parameters	Characteristics
Pressure suppression chamber	
Internal design pressure	45 psig
External design pressure (due to negative internal pressure)	2.0 psig
Drywell	
Internal design pressure	45 psig
External design pressure (due to negative internal pressure)	2.0 psig
Drywell free volume, including downcomer vent pipes	200,540 ft^3 (maximum)
Pressure suppression chamber free volume	144,184 ft^3 (maximum)
Pressure suppression pool water volume	112,197 ft^3 (minimum)[a]
Submergence of downcomer vent pipe below pressure suppression pool surface	11.67 ft (minimum) 12.0 ft (maximum)
Design temperature of drywell	340°F
Design temperature of pressure suppression chamber	275°F
Downcomer vent pipe pressure loss factor	2.77
Total downcomer vent pipe area	309 ft^2
Break area/total downcomer vent pipe area	0.0105
<u>Calculated maximum pressure after blowdown (no prepurge):</u>	
Drywell	37.4[b] psig
Pressure suppression chamber	30.5[b] psig
Number of downcomer vent pipes	99
Minimum spacing of downcomer vent pipes	4 ft 3 in.

Table 3.8-1

Primary Containment
Drywell and Pressure Suppression Chamber
Principal Design (Continued)

Parameters	Characteristics
Normal operating temperature - suppression chamber pool	90°F (maximum)[c]
Normal operating temperature - suppression chamber air space	95°F[c] (150°F maximum)[d]
Normal operating temperature - drywell	135°F (150°F locally)
Normal operating pressure - drywell and suppression chamber	0 psig to 2 psig

[a] The value for the pool water volume does not include the water within the reactor pedestal (10,065 ft^3) and the 12 ft of water below the downcomer vent pipe exits (15,000 ft^3).

[b] Based on an initial containment pressure of 2.0 psig. The value of P_a to be used for 10 CFR 50 Appendix J testing was conservatively chosen to be 38 psig.

[c] Average or bulk temperature.

[d] Average of two thermocouples located near ceiling.

Table 3.8-2

Containment
Environmental Design Conditions

Parameter	Inside Primary Containment Drywell	Outside Primary Containment
Normal operating environment (capable of continuous operation)		
Temperature	135°F to 150°F	40°F to 104°F
Pressure	2 psig	-0.012 psig to +0.25 psig
Relative humidity	40% to 100%	20% to 90%
Maximum emergency environment (equipment is capable of maintaining containment integrity for not less than 2 hr)		
Temperature	340°F	
Pressure	45 psig	
Relative humidity	100%	

Table 3.8-3

Primary Containment
Pressure Suppression System
Maximum Accident Pressure Comparison

	Pressure (psig)	
	Design	Calculated
Drywell	45	37.4[a]
Pressure suppression chamber	45	30.5[a]
Differential pressure on drywell floor, downward	25	21.5

[a] The value of P_a to be used for 10 CFR 50 Appendix J testing was conservatively chosen to be 38 psig.

Table 3.8-4

List of Applicable Codes, Standards, Specifications, and Regulatory Guides

Reference Number	Designation	Title	Edition
1A	ACI 318-71	Building Code Requirements for Reinforced Concrete	February 9, 1971
1B	ACI 318-63	Building Code Requirements for Reinforced Concrete	June 1983
2A	ACI 301-72	Specifications for Structural Concrete for Buildings	May 1972
2B	ACI 301-66	Specifications for Structural Concrete for Buildings	1966
3	ACI 347-68	Recommended Practice for Concrete Formwork	March 1968
4	ACI 605-72	Recommended Practice for Hot Weather Concreting	1972
5A	ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete	1970
5B	ACI 211.1-74	Recommended Practice for Selecting Proportions for Normal Weight Concrete	1974
6	ACI 614-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete	1971
7	ACI 315-74	Manual of Standard Practice for Detailing Reinforced Concrete Structures	1971
8	ACI 306-66	Recommended Practice for Cold Weather Concreting	1966

Table 3.8-4

List of Applicable Codes, Standards, Specifications, and Regulatory Guides (Continued)

Reference Number	Designation	Title	Edition
9	ACI 609-72	Recommended Practice for Consolidation of Concrete	March 1972
10	ACI 322-72	Building Code Requirements for Structural Plain Concrete	1972
11	ACI 308-71	Recommended Practice for Curing Concrete	1971, Title 69-1
12	ACI 212	Guide for Use of Admixtures in Concrete	ACI Journal, September 1971, Title 68-56
13	ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete	1965
14	ACI 311-64	Recommended Practice for Concrete Inspection	1964
15	ACI SP-2	Manual of Concrete Inspection	1968 (5th Edition)
16	Report by ACI Committee 304	Placing Concrete by Pumping Methods	ACI Journal, May 1971, Title 68-33
17	Report by ACI Committee 437, Subcommittee 1	Strength Evaluation of Existing Concrete Structures	ACI Journal, November 1967, Title 64-61
18	AISC-69	Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings	February 12, 1969
19	AISC-68	Specification for the Design of Light Gauge Cold-Formed Steel Structural Members	1968

Table 3.8-4

List of Applicable Codes, Standards, Specifications, and Regulatory Guides (Continued)

Reference Number	Designation	Title	Edition
20	AWS D1.1-72	Structural Welding Code	1972
21	AWS D12.1-61	Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connection in Reinforced Concrete Construction	1961
25	ASTM	Annual Books of ASTM Standards	1972
26	ANSI B31.1.0	Standard Code for Pressure Piping, Power Piping	Latest edition
27	API Spec. No. 620	Specification for Welded Steel Storage Tanks	February 1970
28	UBC	Uniform Building Code	1970
29	NEC	National Electric Code	Latest edition
30	ASTM C 1107	Standard Specification for Packaged Dry, Hydraulic-Cement Grout (Nonshrink)	Latest edition
31	ASTM C 1090	Standard Test Method for Measuring Changes in Height of Cylindrical Specimens of Hydraulic-Cement Grout	Latest edition
32	CRSI	Manual of Standard Practice	1972
33	ANSI 45.2.5-74	Supplementary Quality Assurance Requirements for Installations, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1974

Table 3.8-4

List of Applicable Codes, Standards, Specifications, and Regulatory Guides (Continued)

Reference Number	Designation	Title	Edition
34	---------	Steiger Occupational Safety and Health Act	Latest edition
35	Regulatory Guide 1.10	Mechanical Cadweld Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1)	January 2, 1973
36	Regulatory Guide 1.12	Instrumentation for Earthquakes (Revision 2)	March 1997
37	Regulatory Guide 1.13	Fuel Storage Facility Design Basis	March 10, 1971
38	Regulatory Guide 1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1)	December 28, 1972
40	Regulatory Guide 1.26	Quality Group Classification and Standards (Revision 3)	September 1974
41	Regulatory Guide 1.27	Ultimate Heat Sink for Nuclear Power Plants (Revision 1)	March 1974
42	Regulatory Guide 1.29	Seismic Design Classification (Revision 3)	September 1978
43	Regulatory Guide 1.31	Control of Stainless Steel Welding (Revision 1)	June 1973
44	Not Used		
45	Regulatory Guide 1.55	Concrete Placement in Category I Structures (Revision 0)	June 1973
46	ASME	1971 Boiler and Pressure Vessel Code, Section XI	Summer of 1972 Addenda

Table 3.8-4

List of Applicable Codes, Standards, Specifications, and Regulatory Guides (Continued)

Reference Number	Designation	Title	Edition
47	ASME	1971 ASME Boiler and Pressure Vessel Code, Section VIII	Summer of 1972 Addenda
48	ASME	1971 Code Interpretations Case Number 1177-7	
49	EJMA	Standards of the EJMA	Latest editions
50	SSPC	Painting Specifications	Latest editions
51	ANSI N 101.4	Protective Coating Applied to Nuclear Facilities, Quality Assurance	Latest edition
52	10 CFR Part 50, Appendix A	General Design Criterion 2, "Design Bases for Protection Against Material Phenomena"	July 15, 1971
53	10 CFR Part 50, Appendix A	General Design Criterion 4, "Environmental and Missile Design Basis"	July 15, 1971
54	Regulatory Guide 1.94	Quality Assurance Requirements for Installation; Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Revision 1)	April 1976
55	Regulatory Guide 1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions	March 1997
56	Regulatory Guide 1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	March 1997

Table 3.8-4

List of Applicable Codes, Standards, Specifications,
and Regulatory Guides (Continued)

Legend:

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ANSI	American National Standards Institute
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
CRSI	Concrete Reinforcing Steel Institute
NEC	National Electric Code
UBC	Uniform Building Code
EJMA	Expansion Joint Manufacturers Association
SSPC	Steel Structure Painting Council

NOTE: All of the previously referenced codes and standards may not appear as direct references in the construction specifications. However, they were used by the architect-engineer either in preparing the specification or in design.

Table 3.8-5

Load Combinations and Load Factors
Concrete Internal Structures of Steel Containment

Category ACI 318-71 Strength Design Method	Load	Normal					Severe Environment	Abnormal				Extreme Environment
		D	L	R_o	T_o	P_o	E	P_a	T_a	R_a	R_r	E′
Service load conditions												
Normal	1	1.4	1.7			1.7						
	1b	1.4	1.7	1.4	1.4	1.7						
Severe environmental	2	1.4	1.7			1.7	1.9					
	2b	1.4	1.4	1.4	1.4	1.4	1.4					
Factored load conditions[a,b]												
Extreme environmental	3	1.0	1.0	1.0	1.0	1.0						1.0
Abnormal[c]	4	--	--			--		--	--	--		
Abnormal/severe environmental[d]	5	--	--			--	--	--	--	--	--	
Abnormal/extreme environmental	6	1.0	1.0		1.0	1.0		1.0	1.0	1.0	1.0	1.0

[a] In combination 6, the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, are used unless a time-history analysis is performed to justify otherwise (R_r includes Y_j, Y_r, and Y_m.)

[b] When considering Y_j, Y_r, and Y_m, local section strength capacities may be exceeded under these concentrated loads provided there is no loss of function of any safety-related system.

Table 3.8-5

Load Combinations and Load Factors
Concrete Internal Structures of Steel Containment (Continued)

[c] In abnormal load category, DBA is not considered alone. DBA is considered with E′ as in load combination 6.

[d] In abnormal/severe environmental load category, DBA with E is not considered. DBA is considered with E′ as in load combination 6.

<u>NOTE</u>:

All of the loads listed are not necessarily applicable to all concrete structures in containment. Loads not applicable to a particular structure are deleted. If for any combination, the effect of any load other than D reduces the stress it is deleted from the combination. Combination numbers correspond to those in NUREG-0800, Standard Review Plan for Section 3.8.3. Dashed lines indicate that the load combination is not used. For load definitions, see Section 3.8.3.3.

Table 3.8-6

Load Combinations and Load Factors
Steel Internal Structures of Steel Containment

Load Category	Load	Normal					Severe Environment	Abnormal				Extreme Environment
		D	L	R_o	T_o	P_o	E	P_a	T_a	R_a	R_r	E′
Elastic Working Stress Design Method												
Service load conditions												
Normal	1	1.0	1.0			1.0						
	1a	1.0	1.0	1.0	1.0	1.0						
Severe environmental	2	1.0	--			1.0	1.0					
	2a	1.0	--	1.0	1.0	1.0	1.0					
Factored load conditions[a-c]												
Extreme environmental[d]	3	--	--	--	--	--						--
Abnormal[e]	4	--	--			--		--	--	--		
Abnormal/severe environmental[f]	5	--	--			--	--	--	--	--	--	
Abnormal/extreme environmental	6	1.0	--			1.0		1.0	1.0	1.0	1.0	1.0
Plastic Design Method												
Service load conditions												
Normal	1	1.0	1.0			1.0						
	1b	1.0	1.0	1.0	1.0	1.0						
Severe environmental	2	1.0	--			1.0	1.0					
	2b	1.0	--	1.0	1.0	1.0	1.0					
Factored load conditions[a-c]												
Extreme environmental[d]	3	--	--	--	--	--						--
Abnormal[e]	4	--	--			--		--	--	--		
Abnormal/severe environmental[f]	5	--	--			--		--	--	--	--	
Abnormal/extreme environmental	6	1.0	--			1.0		1.0	1.0	1.0	1.0	1.0

Table 3.8-6

Load Combinations and Load Factors
Steel Internal Structures of Steel Containment (Continued)

[a] In combination 6, the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, are used unless a time-history analysis performed to justify otherwise (R_r includes Y_j, Y_r, and Y_m).

[b] When considering Y_j, Y_r, and Y_m, local section strength capacities may be exceeded under these concentrated loads provided there will be no loss of function of any safety-related system.

[c] Thermal loads for factored load conditions are neglected when it can be shown that they are secondary and self-limiting in nature.

[d] Extreme environmental E′ is considered with DBA as in load combination 6.

[e] Abnormal: DBA not considered alone. DBA is considered with E′ as in load combination 6.

[f] Abnormal/severe environmental: DBA with E not considered. DBA is considered with E′ as in load combination 6.

NOTE:

The drywell floor support steel is considered a steel internal structure in the suppression chamber. All the loads listed are not necessarily applicable to all steel structures. Loads not applicable to a particular structure are deleted. If, for any load combination, the effect of any load other than D reduces the stress, it is deleted from the combination. Combination numbers correspond to those in NUREG-0800, Standard Review Plan, for Section 3.8.3. Dashed lines indicate that the load or load combination is not used. For load definitions, see Section 3.8.3.3.

Table 3.8-7

Section Strength Limits and Section Modulus
for Structural Steel Internal Structures
of Steel Containment

Load Category	Load	Strength Limit[a,b]	Section Modulus of Steel Shapes
Elastic Working Stress Design Method			
Service load conditions			
Normal	1	S	Elastic
	1a	1.5S	Elastic
Severe environmental	2	S	Elastic
	2a	1.5S	Elastic
Factored load conditions			
Extreme environmental	3	1.6S	Elastic
Abnormal	4	1.6S	Elastic
Abnormal/severe environmental	5	1.6S	Plastic[c]
Abnormal/extreme environmental	6	1.7S	Plastic[c]
Plastic Design Method			
Service load conditions			
Normal	1	Y	Plastic
	1b	Y	Plastic
Severe environmental	2	Y	Plastic
	2b	Y	Plastic
Factored load conditions			
Extreme environmental	3	0.9Y	Plastic
Abnormal	4	0.9Y	Plastic
Abnormal/severe environmental	5	0.9Y	Plastic
Abnormal/extreme environmental	6	0.9Y	Plastic

[a] S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, February 12, 1969.
[b] Y is the section strength required to resist design loads based on plastic design methods in Part 2 of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, February 12, 1969.
[c] Plastic section modulus may be used.

Table 3.8-8

Radwaste and Control Building
Quality Class and Design Bases Criteria

Location	Materials of Construction[a]	Quality Class	Environmental Disturbances			
			Seismic Category	Design Basis Wind	Design Basis Tornado	Tornado-Generated Missiles
Those portions of the radwaste and control building that house systems or components necessary for safe shutdown of the reactor	Reinforced-concrete	I	I	Yes	Yes	Yes
Those portions of the radwaste building that house equipment containing significant quantities of radioactive material	Reinforced-concrete	I	I	Yes	Yes	Yes
Other portions	Structural steel	II	II	Yes	No[b]	No[b]

[a] See Figure 3.8-42.

[b] “Other portions” denote those portions of the radwaste and control building which are not required to withstand the effects of the design basis tornado and tornado-generated missiles.

Table 3.8-9

Load Combinations and Load Factors
Seismic Category I and Nonseismic Category I Safety-Related
Concrete Structures Outside Primary Metal Containment

		Normal									Severe Environmental					Abnormal					Extreme Environmental		
Load Category ACI 318-71 Strength Design Method	Load	D	L	Ro	To	Po	Pt	Tt	F	Q	E	W	H	F*	Q*	Pa	Ta	Ra	Rr	P′	E′	W′	H′
Service load conditions																							
Construction	U1	1.4																					
	U2	0.9	1.3	0.9	1.3							1.3											
Normal	1	1.4	1.7			1.7																	
	1b	1.4	1.7	1.4	1.4	1.7																	
	U3	1.4	1.7			1.7			1.4	1.7													
	U4	1.4	1.7		1.4	1.7			1.4	1.7													
	U5	0.9							1.4	1.7													
	U6	0.9			1.4				1.4	1.7													
Severe	2	1.4	1.7			1.7					1.9												
	2b	1.4	1.4	1.4	1.4	1.4					1.4												
	2b′	0.9									1.4												
	3	1.1	1.3			1.3						1.3											
	3b	1.1	1.3	1.1	1.1	1.3						1.3											
	3b′	0.9										1.3											
	U7	1.1	1.3			1.3			1.1	1.3		1.3											
	U8	1.1	1.3		1.1	1.3			1.1	1.3		1.3											
	U9	1.4	1.7			1.7					1.9			1.4	1.7								
	U10	1.4	1.4		1.4	1.4					1.4			1.4	1.4								
Factored load conditions[a, b]																							
Extreme environmental	4	1.0	1.0	1.0	1.0																1.0		
	5	--	--	--	--																	--	
Abnormal	6	1.0	1.0													1.5	1.0	1.0					
Abnormal/severe environmental	7	1.0	1.0								1.2 5					1.25	1.0	1.0	1.0				
Abnormal/extreme environmental	8	1.0	1.0													1.0	1.0	1.0	1.0		1.0		
	U11	1.0	1.0	1.0	1.0															1.0	1.0		
		1.0	1.0	1.0	1.0															1.0		1.0	

Table 3.8-9

Load Combinations And Load Factors
Seismic Category I and Nonseismic Category I Safety-Related
Concrete Structures Outside Primary Metal Containment (Continued)

[a] In combinations 6, 7, and 8, the maximum values of P_a, T_a, R_a, Y_r, and Y_j, including an appropriate dynamic load factor, are used. The value of Y_m is arrived at by an energy balance method of structural action (Section 3.6.1.6.3.2), to account for the dynamic nature of the load.

In combinations 7 and 8, local stresses due to concentrated load Y_r., Y_j, and Y_m may be permitted to exceed the allowable stresses, provided there is no loss of function of any safety-related system as a result thereof.

[b] In considering the concentrated tornado missile load in combination U11, local section strength capacities may be exceeded under these concentrated loads provided there is no loss of function of any safety-related system as a result thereof.

NOTES:

All the loads listed are not necessarily applicable to all concrete structures. Loads not applicable to a particular structure are deleted. If, for any combination, the effect of any load other than dead loads reduces the stress it is deleted from the combination. Combinations 1 through 8, 1b, 2b, 2b′, 3b, and 3b′ correspond to those in NUREG-0800, Standard Review Plan for Section 3.8.4. Combinations U1 through U11 are not in the review plan for Section 3.8.4 and are used in addition to those of the review plan combinations. Dashed lines indicate that the load or load combination is not used. For load definitions, see Section 3.8.4.3. Combinations 6, 7, and 8 are used only when abnormal loads generated by a postulated pipe break are included.

Table 3.8-10

Load Combinations and Load Factors
Seismic Category I and Nonseismic Category I Safety-Related
Steel Structures Outside Primary Metal Containment

		Normal					Severe Environmental		Abnormal					Extreme Environmental	
Load Category	Load	D	L	R_o	T_o	P_o	E	W	P_a	T_a	R_a	R_r	P′	E′	W′
Elastic Working Stress Design Method															
Service Load Conditions															
Normal	1	1.0	1.0			1.0									
	1a	---	---	---	---										
Severe environmental	2	1.0	---			1.0	1.0								
	2a	---	---	---	---		---								
	3	1.0	---			1.0		1.0							
	3a	---	---	---	---			---							
Factored Load Conditions															
Extreme environmental	4	---	---	---	---									---	
	5	---	---	---	---										---
Abnormal	6	---	---						---	---	---				
	U12	1.0	1.0										1.0		
Abnormal/severe environmental	7	---	---				---		---	---	---	---			
Abnormal/extreme environmental	8	---	---						---	---	---	---		---	
	U13	1.0											1.0	1.0	
	U14	1.0											1.0		1.0

Table 3.8-10

Load Combinations and Load Factors
Seismic Category I and Nonseismic Category I Safety-Related
Steel Structures Outside Primary Metal Containment (Continued)

		Normal					Severe Environmental		Abnormal					Extreme Environmental	
Load Category	Load	D	L	R_o	T_o	P_o	E	W	P_a	T_a	R_a	R_r	P′	E′	W′
<u>Plastic Design Method</u>															
Service load conditions															
Normal	1	1.0	1.0			1.0									
	1b	---	---	---	---										
Severe environmental	2	1.0	---			1.0	1.0								
	2b	---	---	---	---		---								
	3	1.0	---			1.0		1.0							
	3b	---	---	---	---			---							
Factored load conditions															
Extreme environmental	4	---	---	---	---									---	
	5	---	---	---	---										---
Abnormal	6	1.0	1.0						1.5	1.0	1.0				
	U12	1.0	1.0										1.0		
Abnormal/severe environmental	7	1.0	1.0				1.25		1.25	1.0	1.0	1.0			
Abnormal/extreme environmental	8	1.0	1.0						1.0	1.0	1.0	1.0		1.0	
	U13	1.0											1.0	1.0	
	U14	1.0											1.0		1.0

Table 3.8-10

Load Combinations and Load Factors
Seismic Category I and Nonseismic Category I Safety-Related
Steel Structures Outside Primary Metal Containment (Continued)

Notes:

1. In combinations 6, 7, and 8, (factored load conditions, Plastic Design Method), the maximum values of P_a, T_a, R_a, Y_j, and Y_r, including an appropriate dynamic load factor, are used; and the value of Y_m is arrived at by an energy balance method of structural action (Section 3.6.1.6.3.2), to account for the dynamic nature of the load.

   In combinations 7 and 8, (factored load conditions, Plastic Design Method), local stresses due to concentrated loads Y_r, Y_j, and Y_m may be permitted to exceed the allowable stresses, provided there is no loss of function of any safety-related system as a result thereof.

   In considering the concentrated tornado missile load in combination U14, local section strength capacities may be exceeded under these concentrated loads provided there is no loss of function of any safety-related system as a result thereof.

2. Thermal loads for factored load conditions are neglected when it can be shown that they are secondary and self-limiting in nature.

3. All the loads listed are not necessarily applicable to all concrete structures. Loads not applicable to a particular structure are deleted.

4. If, for any load combination, the effect of any load other than D reduces the stress, it is deleted from the combination.

5. Combinations 1 through 8, 1a, 2a, 3a, 1b, 2b, and 3b correspond to those in the NRC Standard Review Plan for Section 3.8.4. Combinations U12, U13, and U14 are not in the review plan for Section 3.8.4, and are used in addition to those of the review plan combinations.

6. Dashed lines indicate that the load or load combination is not used.

7. For load definitions, see Section 3.8.4.3.

8. This table applies to Section 3.8.4.

9. Combinations 6, 7, and 8 in the Plastic Design Method are used only when abnormal loads generated by a postulated pipe break are included.

Table 3.8-11

Seismic Category I and Nonseismic
Safety-Related Steel Structures
Outside Primary Metal Containment

Load Category	Load	Strength Limit[a, b]	Section Modulus of Steel Shapes
Elastic Working Stress Design Method			
Service load conditions			
Normal	1	S	Elastic
	1a	1.5S	Elastic
Severe environmental	2	S	Elastic
	2a	1.5S	Elastic
	3	S	Elastic
	3a	1.5S	Elastic
Factored load conditions			
Extreme environmental	4	1.6S	Elastic
	5	1.6S	Elastic
Abnormal	6	1.6S	Elastic
	U12	1.6S	Elastic
Abnormal/severe environmental	7	1.6S	Plastic[c]
Abnormal/extreme environmental	8	1.7S	Plastic[c]
	U13	1.7S	Elastic
	U14	1.7S	Elastic
Plastic Design Method			
Service load conditions			
Normal	1	Y	Plastic
	1b	Y	Plastic
Severe environmental	2	Y	Plastic
	2b	Y	Plastic
	3	Y	Plastic
	3b	Y	Plastic
Factored load conditions			
Extreme environmental	4	0.9Y	Plastic
	5	0.9Y	Plastic
Abnormal	6	0.9Y	Plastic
	U12	0.9Y	Plastic

Table 3.8-11

Seismic Category I and Nonseismic
Safety-Related Steel Structures
Outside Primary Metal Containment (Continued)

Load Category	Load	Strength Limit[a, b]	Section Modulus of Steel Shapes
Abnormal/severe environmental	7	0.9Y	Plastic
Abnormal/extreme environmental	8	0.9Y	Plastic
	U13	0.9Y	Plastic
	U14	0.9Y	Plastic

[a] S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, February 12, 1969.

[b] Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, February 12, 1969.

[c] Plastic section modulus may be used.

Table 3.8-12

Maximum Permissible Stresses
Seismic Category I and Nonseismic Category I
Safety-Related Reinforced-Concrete Structures

	Concrete			Reinforcing Steel		
		Shear				
Loading Conditions	Flexural and Axial Compression	Flexural	Punching	Flexural Tension	Shear Tension	Axial Compression
Service load and factored load conditions	ϕ (0.85 $f'c$) Varies from: $0.70(0.85f'c) = 0.60f'c$ to $0.90(0.85f'c) = 0.76f'c$	$2\sqrt{f'c}$	$4\sqrt{f'c}$	$0.9f_y$	$0.85f_y$	Varies from $0.7f_y$ to $0.75f_y$

Notes:

1. Concrete tensile strength is not relied upon.
2. $f'c = 4000$ psi; $\eta = 8$
3. $f_y = 40{,}000$ psi for reinforcing steel up to and including #5 bar size.
4. $f_y = 60{,}000$ psi for reinforcing steel over #5 bar size.
5. The maximum permissible compression and flexural stress of $0.85f'c$ corresponds to a limiting strain of 0.003 in./in.
6. The symbol ϕ represents the capacity reduction factor.
7. Normal permissible bearing stresses are as specified in ACI 318-71.
8. This table applies to the Sections 3.8.3, 3.8.4, and 3.8.5.

Table 3.8-13

Primary Containment Vessel Electrical Penetrations
Principal Design Parameters

Environmental Condition[a]	Inside Primary Containment Vessel	Outside Primary Containment Vessel
Normal operating pressure	-0.5 psig to 2 psig	-0.10 in. to -1.0 in. water gauge
Design pressure	-2.0 psig to 45 psig	7 in. water gauge
Test pressure	52 psig	N/A
Normal operating temperature	135°F average 150°F maximum	70°F average 104°F maximum
Design temperature (accident - 6 hr)	340°F	212°F
Design temperature (accident up to 6 months)	250°F	150°F
Relative humidity (normal)	40% to 55% 90% maximum	40%
Relative humidity (LOCA)	100%	100%
Gamma radiation (normal operating)	50.0 rad/hr	N/A
Neutron radiation (normal operating)	1.4 x 10^5 neutrons/cm^2 sec	N/A
Lowest service metal temperature	30°F	N/A
Integrated dose - gamma (normal condition)	1.8 x 10^7 rad	N/A
Integrated dose - neutron (normal condition)	1.8 x 10^{14} neutrons/cm^2	N/A
Integrated dose - gamma (accident conditions)	2.6 x 10^7 rad	N/A
LOCA dose rate - gamma[b]	1.3 x 10^6 rad	N/A

[a] Under normal operating and accident conditions, the design parameters are integrated over 40 years.

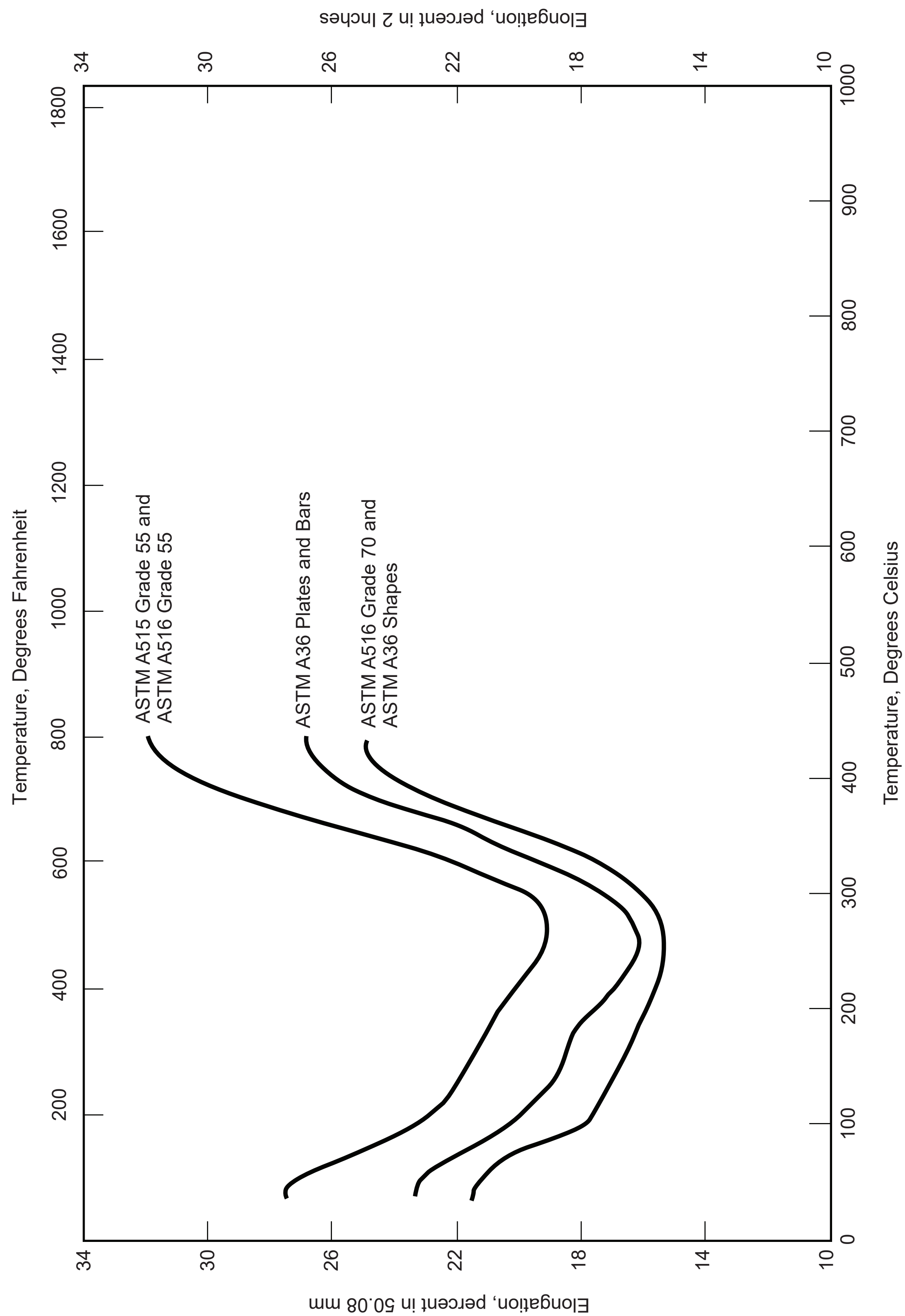
[b] LOCA analysis is based on the assumption that 100% of the noble gases, 50% of the halogens, and 1% of the solid fission products are released from the core.

Table 3.8-14

Primary Containment Vessel
Piping Penetrations
Principal Design

Parameters	Characteristics
Pressure suppression chamber	
Internal design pressure	45 psig
External design pressure (due to negative internal pressure)	2.0 psig
Drywell	
Internal design pressure	45 psig
External design pressure (due to negative internal pressure)	2.0 psig
Design temperature of drywell	340°F
Design temperature of pressure suppression chamber	275°F
Normal operating temperature - suppression chamber air space	95°F[a] 150°F (maximum)[b]
Normal operating temperature - drywell	135°F[a]
Normal operating pressure - drywell and suppression chamber	0 psig to 2 psig

[a] Average or bulk temperature.
[b] Average of two thermocouples located near ceiling.



Columbia Generating Station Final Safety Analysis Report	Minimum Total Elongation Structural Materials, Carbon Steels, Medium Carbon Steels		
	Draw. No. 990306.71	Rev.	Figure 3.8-5

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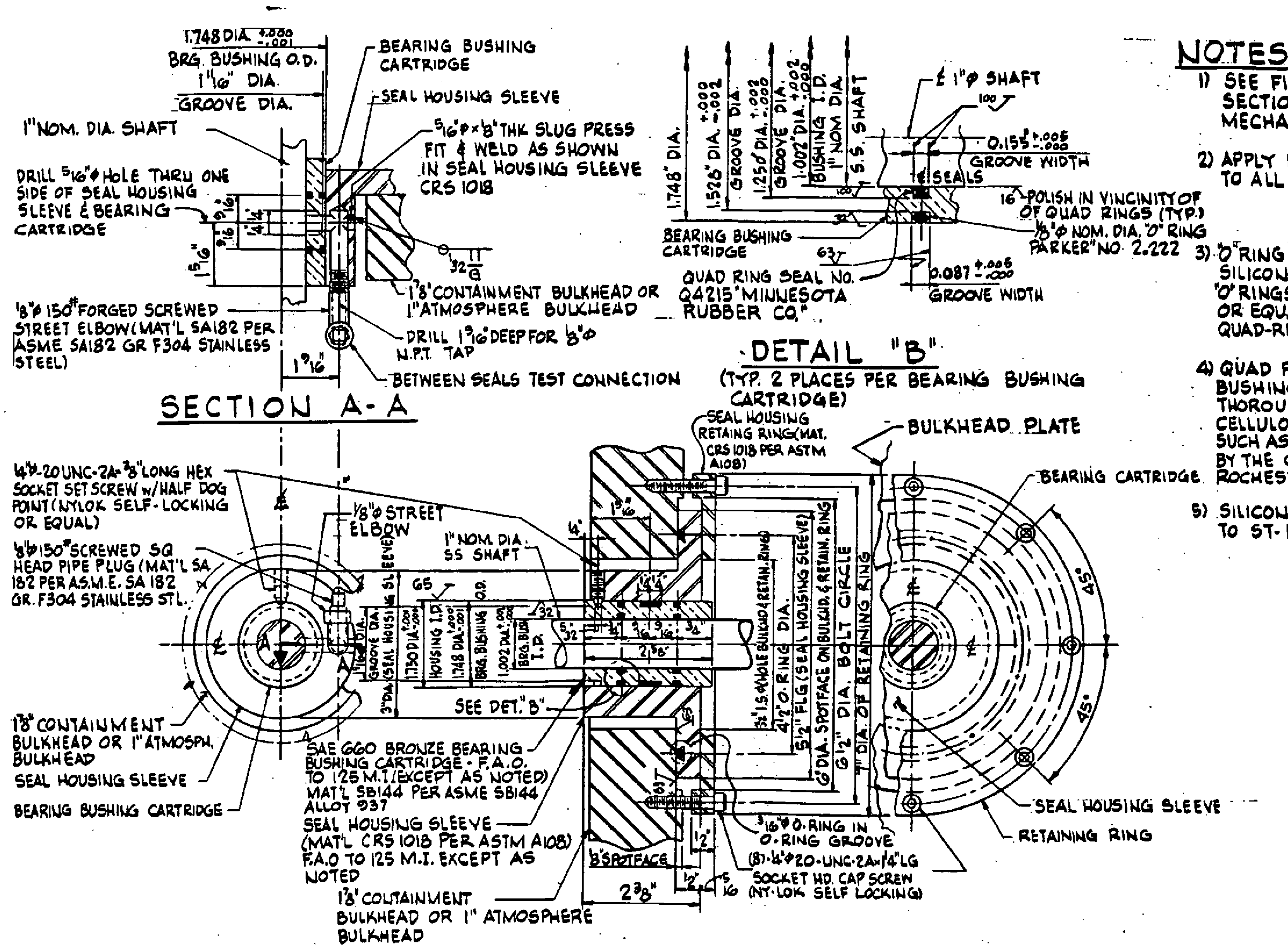
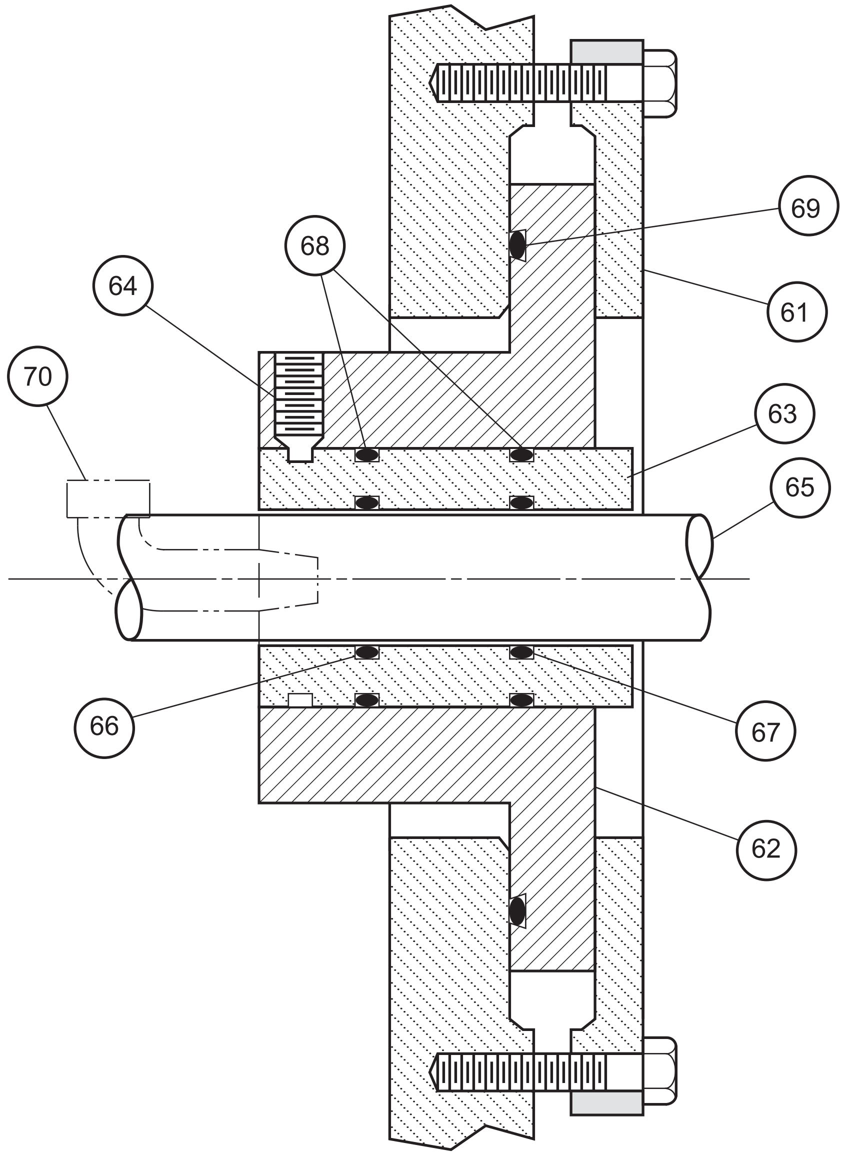


NOTES:

1) SEE FIG. 3.8-15 FOR AN ENLARGED SECTION THRU A BULKHEAD MECHANICAL PENETRATION SEAL.
2) APPLY LOCTITE PIPE SEALANT CAT. NO. 71 TO ALL PIPE THREADS.
3) "O" RING & QUAD RING SEAL MATERIAL: SILICONE TO 70 DUROMETER "O" RINGS: "PARKER" COMPOUND NO. S604-7 OR EQUAL QUAD-RING: M.R. COMPOUND NO. 71417
4) QUAD RING SEALS, SHAFT & BEARING BUSHING CARTRIDGE MUST BE THOROUGHLY LUBRICATED WITH A CELLULOSE ESTER & CASTER OIL COMPOUND SUCH AS "CELVACENE" MANUFACTURED BY THE CONSOLIDATED VACUUM CO., ROCHESTER N.Y. (OR EQUAL)
5) SILICONE RUBBER MATERIAL TO BE TEST TO ST-1

TYPICAL SECTION THRU 1"φ SHAFT PENETRATION IN ATMOSPHERE & CONTAINMENT BULKHEAD

Columbia Generating Station Final Safety Analysis Report	Personnel Access Airlock - Typical Bulkhead Mechanical Penetration		
	Draw. No. 020552.10	Rev.	Figure 3.8-14



Section through Bulkhead Penetration Seal

Key	Description
61	Seal Housing Retaining Ring
62	Seal Housing Sleeve
63	Bronze Bushing Cartridge
64	Locking Set Screw
65	Shaft
66	Outboard Quad Seal
67	Inboard Quad Seal
68	Cartridge O-Ring Seals
69	Housing O-Ring Seal
70	Seal Test Connection

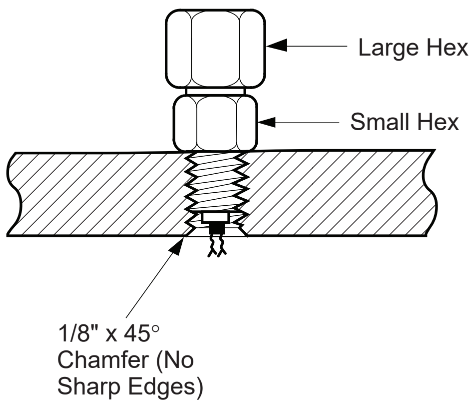
Notes:

1. Each shaft penetrating the containment bulkhead or the atmosphere bulkhead is sealed with a cartridge type seal unit having double dynamic seals. A test connection is provided for pressure testing between the seals.

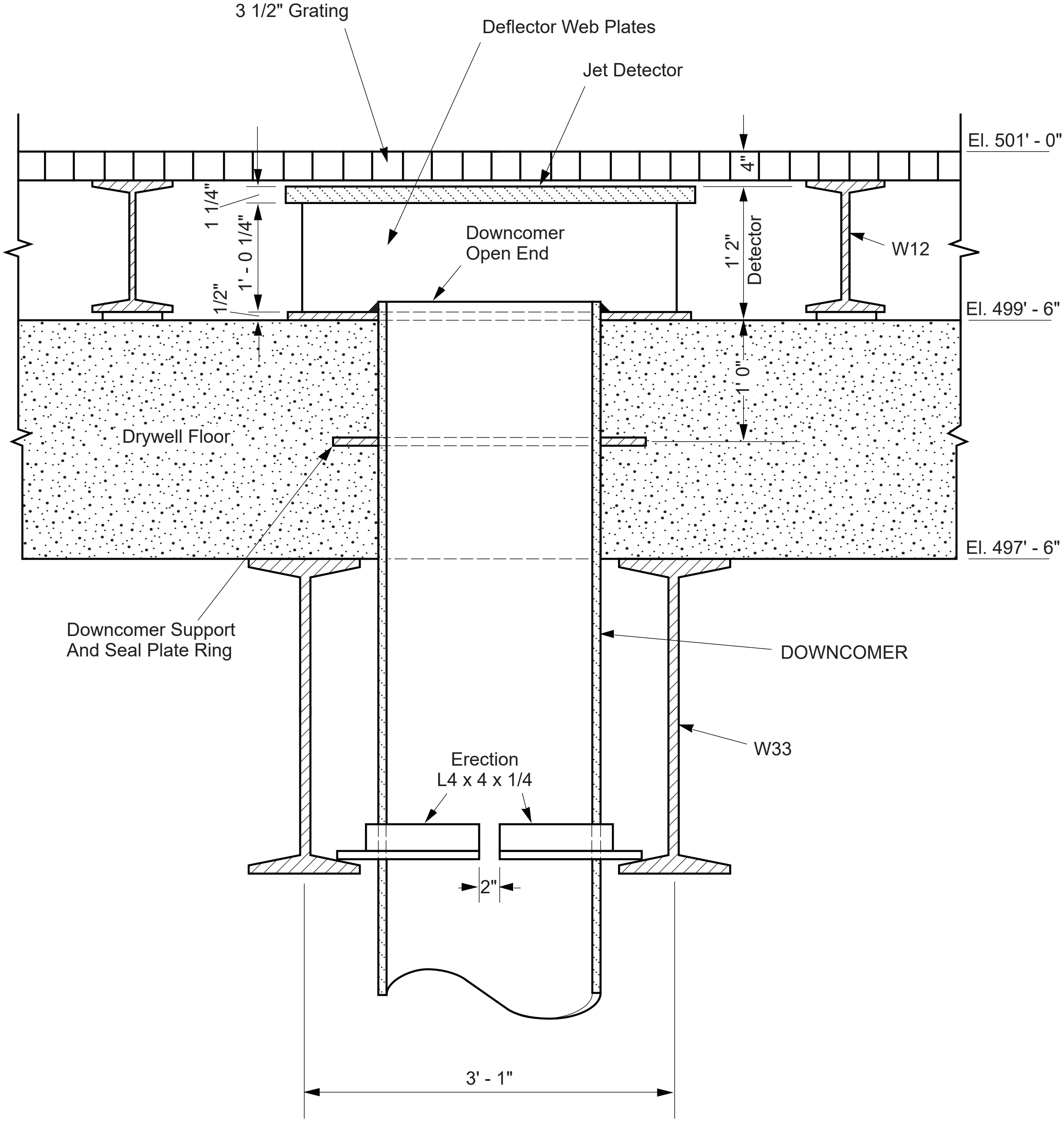
2. See Figure 3.8-14 for additional views and details of this seal in a typical bulkhead mechanical penetration.

Instructions Followed for Installing
Conax Fittings (P1 thru P4)

1) Install the Conax fitting into the bulkhead and pressure seal the leads before electrical boxes are fastened in place.
2) Use only the small hex for installing the Conax fitting into the bulkhead.
3) Use the larger hex for pressure sealing the leads.
4) Caution: Inside parts of the Conax fittings are easily lost. Do not take fitting apart.
5) Conax fittings to be installed from the pressure side only which means Conax fittings to be installed on the side of the bulkhead which the door hinges on.



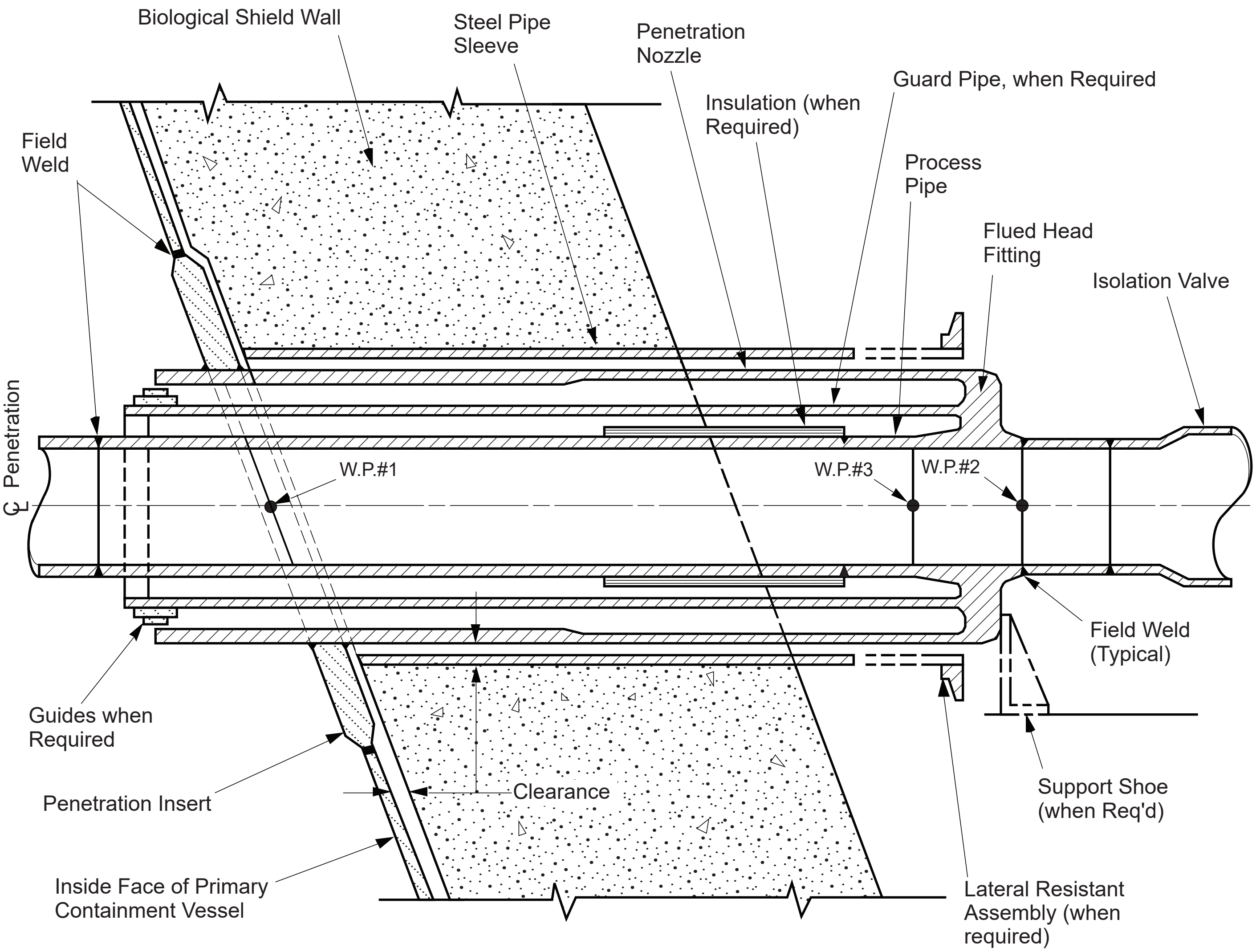
Note:
The locations of the Conax fittings are indicated on Figures 3.8-11 & 3.8-12



Columbia Generating Station
Final Safety Analysis Report

Primary Containment Vessel Drywell Floor Downcomer Vent Pipes

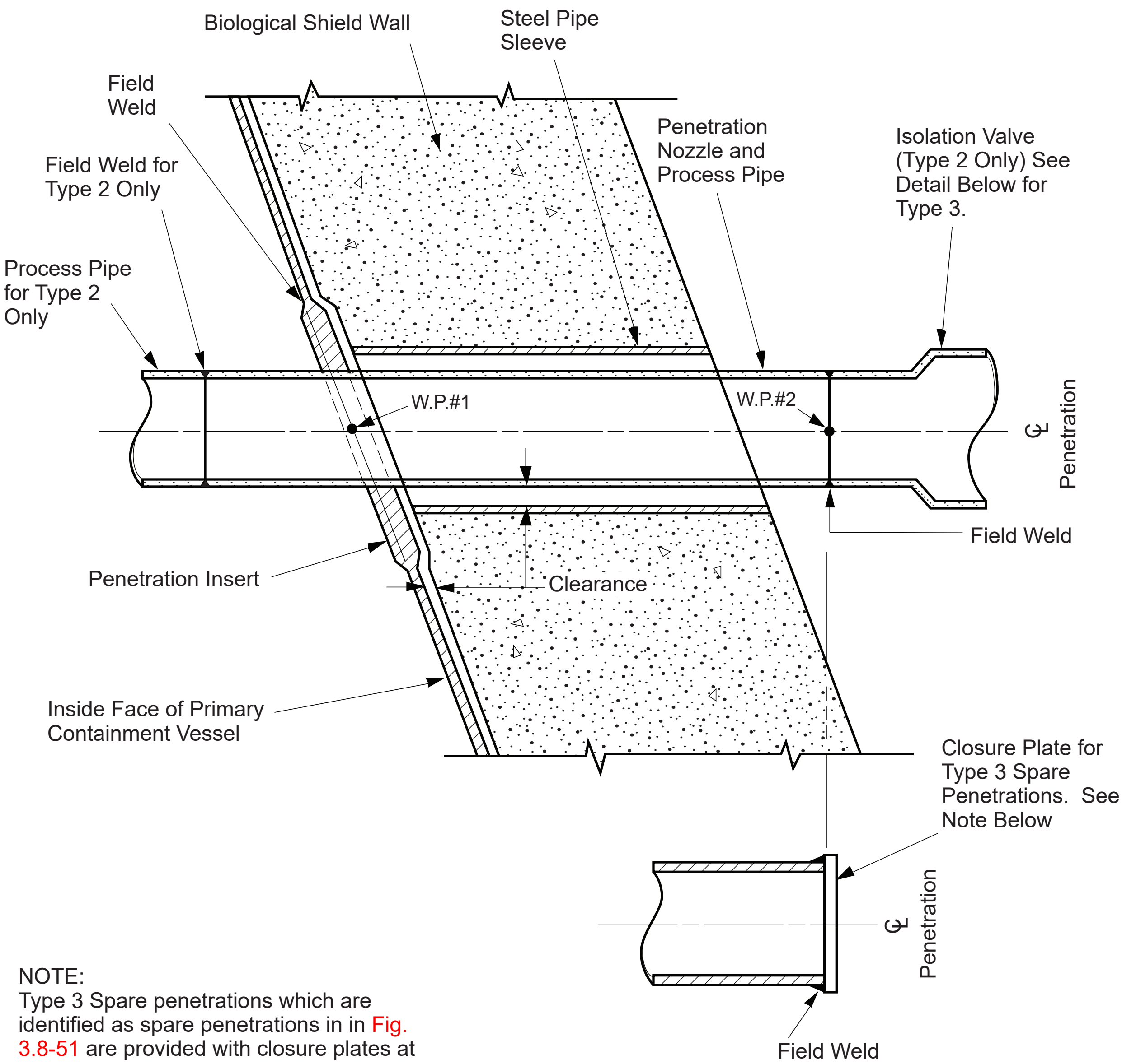
Draw. No. 990306.73 | Rev. | Figure 3.8-19



Columbia Generating Station
Final Safety Analysis Report

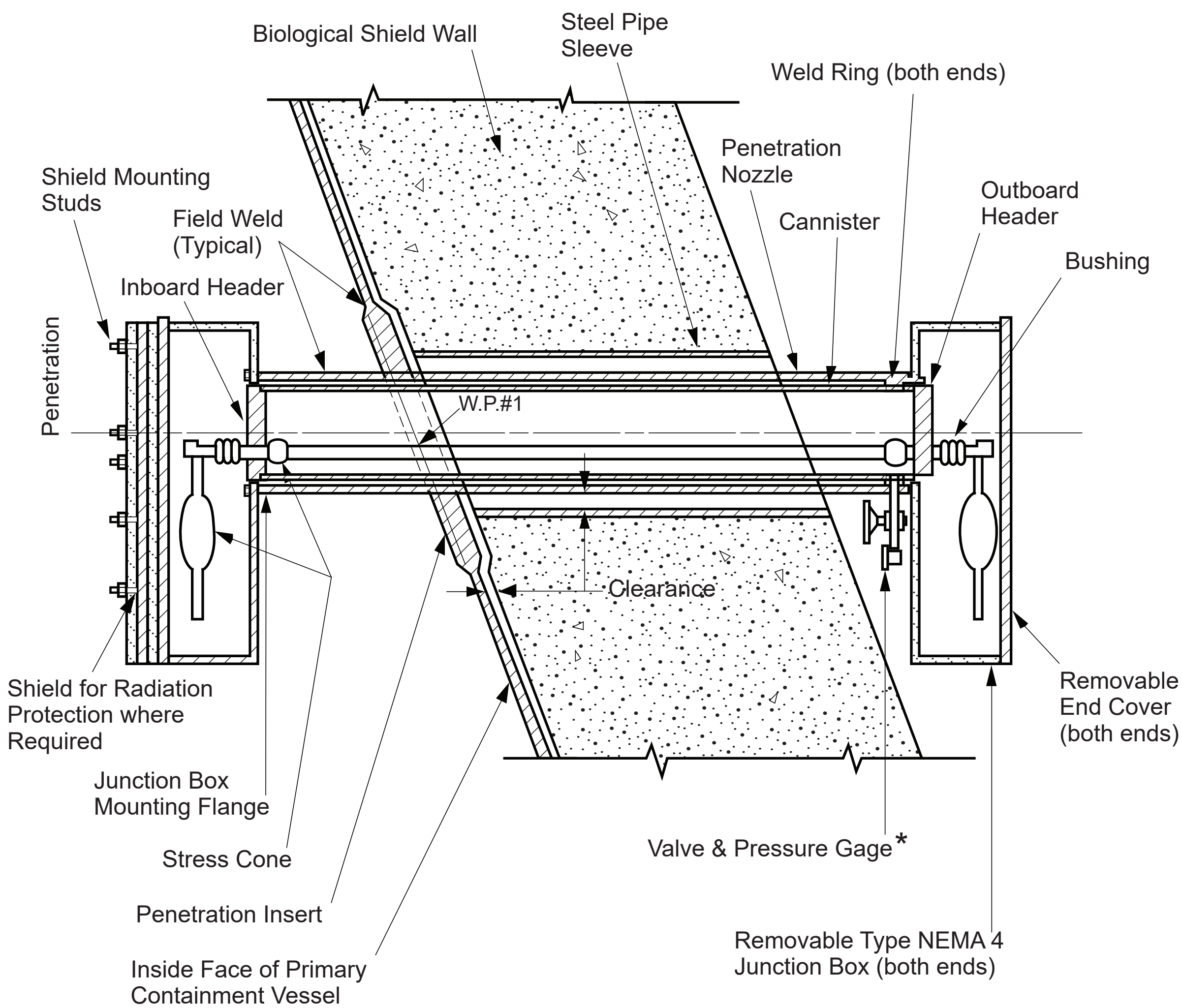
Primary Containment Vessel Penetration Assembly - Type 1 Pipe

Draw. No. 990306.74 | Rev. | Figure 3.8-54



NOTE:
Type 3 Spare penetrations which are identified as spare penetrations in in Fig. 3.8-51 are provided with closure plates at W.P.#2 location. Type 3 penetrations which are not spare penetrations as indicated in Fig. 3.8-51, are not provided with permanent closure plates.

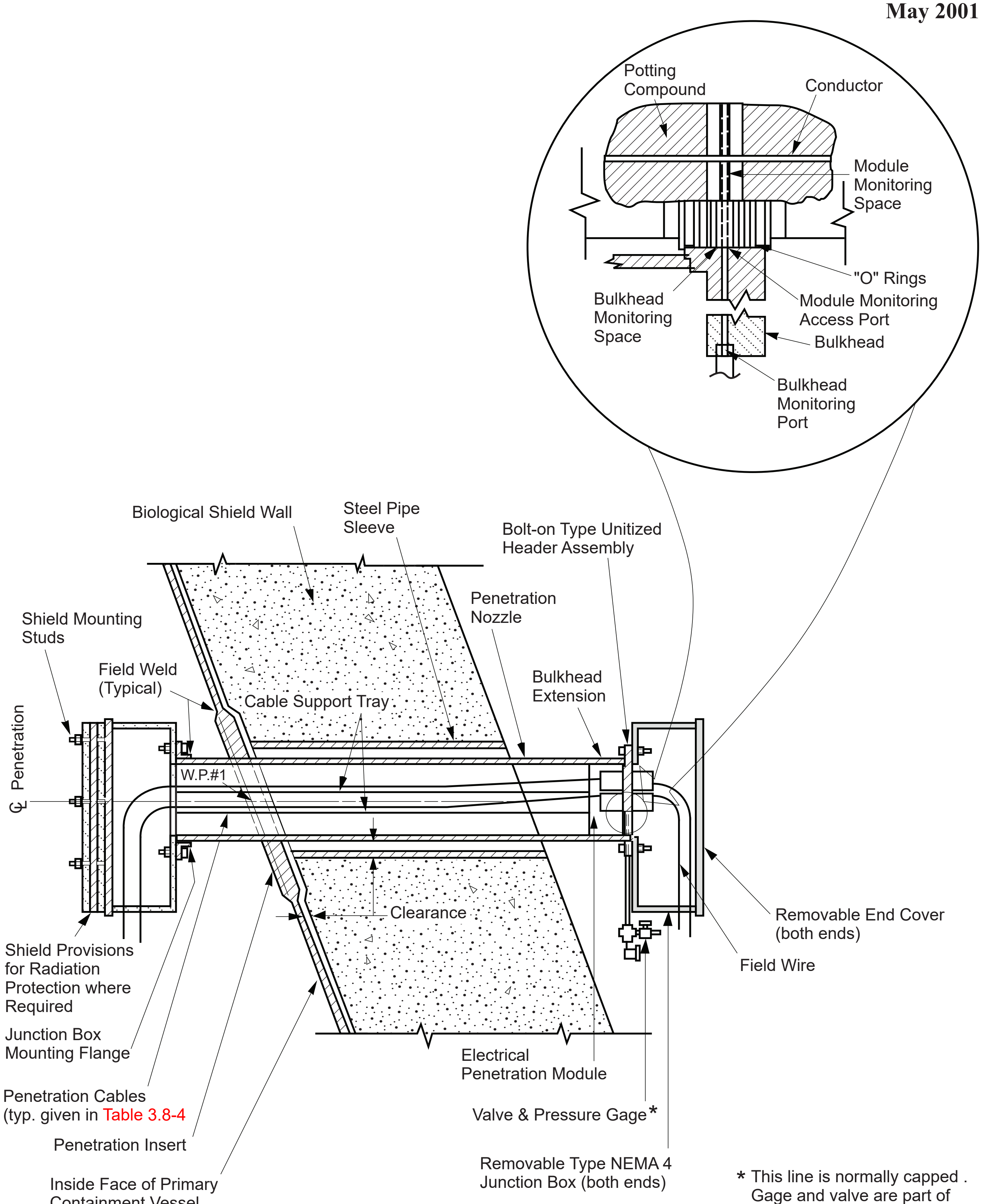
Columbia Generating Station Final Safety Analysis Report	Primary Containment Vessel Penetration Assembly - Types 2 and 3 Pipe		
	Draw. No. 990306.75	Rev.	Figure 3.8-55



* This line is normally capped. Gage and valve are part of the local leak rate testing.

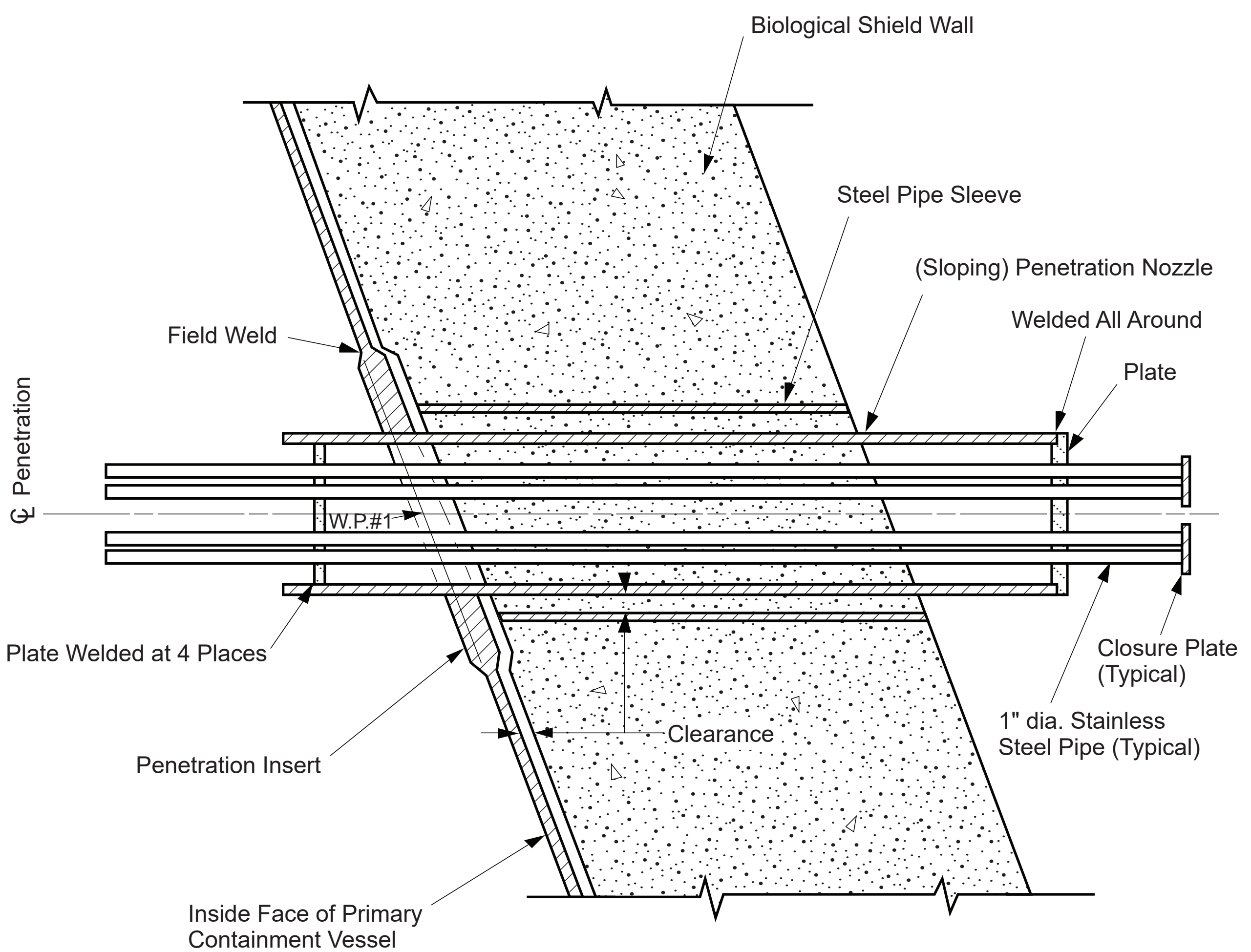
Primary Containment Vessel Penetration Assembly - Type 4 Electrical Canister Type

Draw. No. 990306.76 | Rev. | Figure 3.8-56



* This line is normally capped . Gage and valve are part of the local leak rate testing.

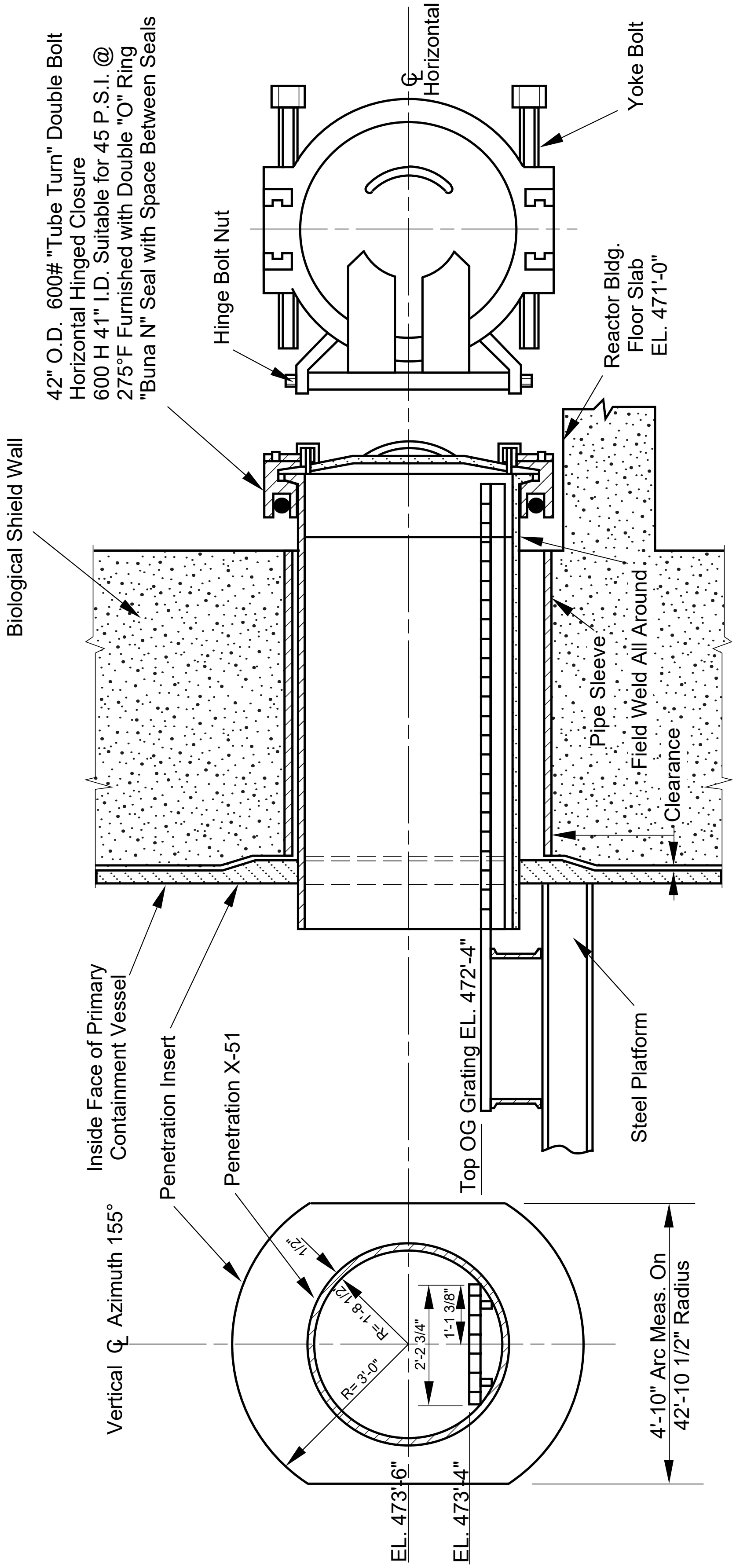
Columbia Generating Station Final Safety Analysis Report	**Primary Containment Vessel Penetration Assembly - Type 4 Electrical Non-Canister Type**
	Draw. No. 990306.77 | Rev. | Figure 3.8-57



Primary Containment Vessel Penetration Assembly - Types 5, 6, And 7 Instrumentation

Draw. No. 990306.78	Rev.	Figure 3.8-58

Amendment 55
May 2001

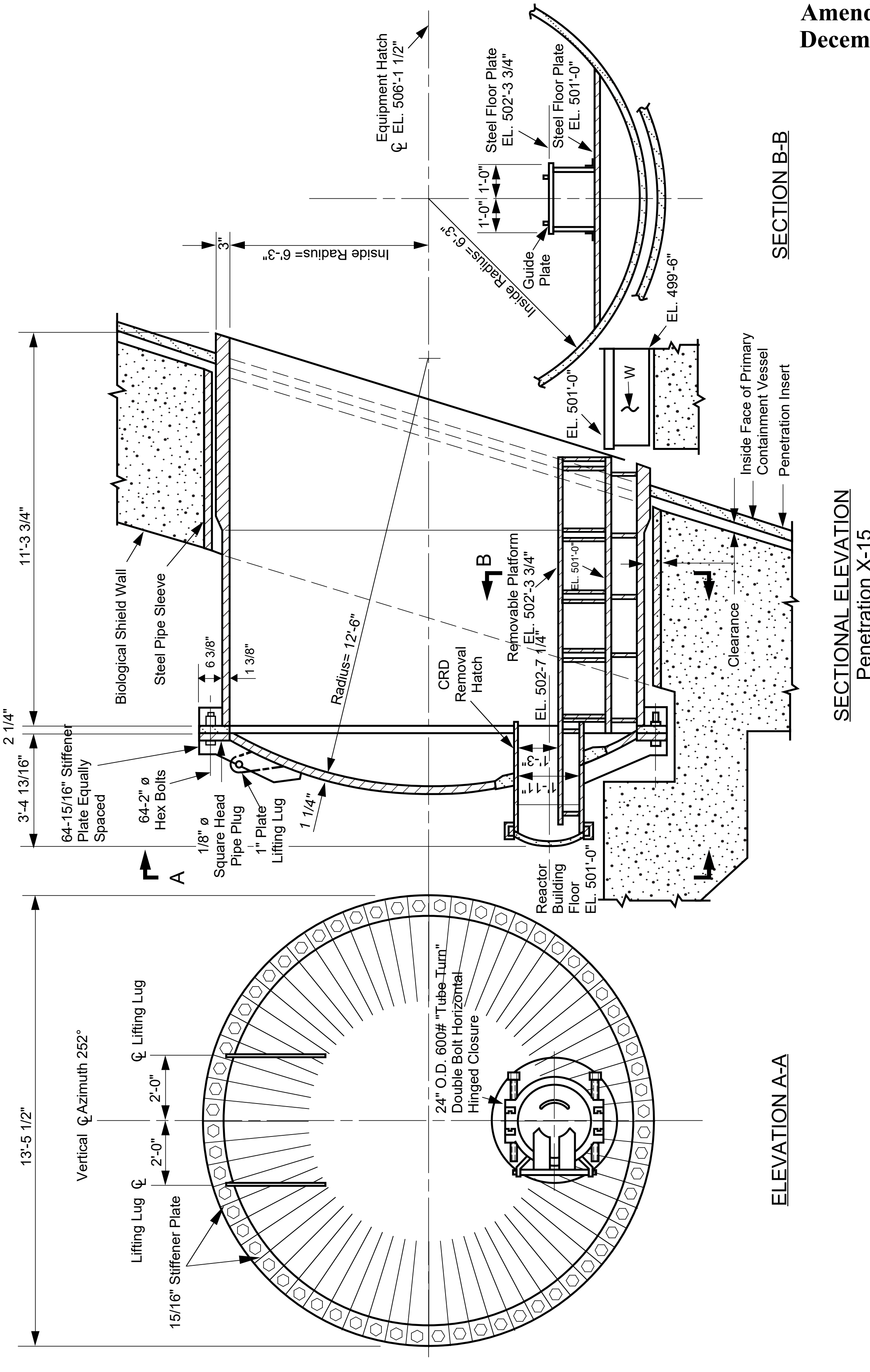


Columbia Generating Station
Final Safety Analysis Report

Primary Containment Vessel Suppression Chamber Access Hatch

Draw. No. 010126.16 | Rev. | Figure 3.8-59

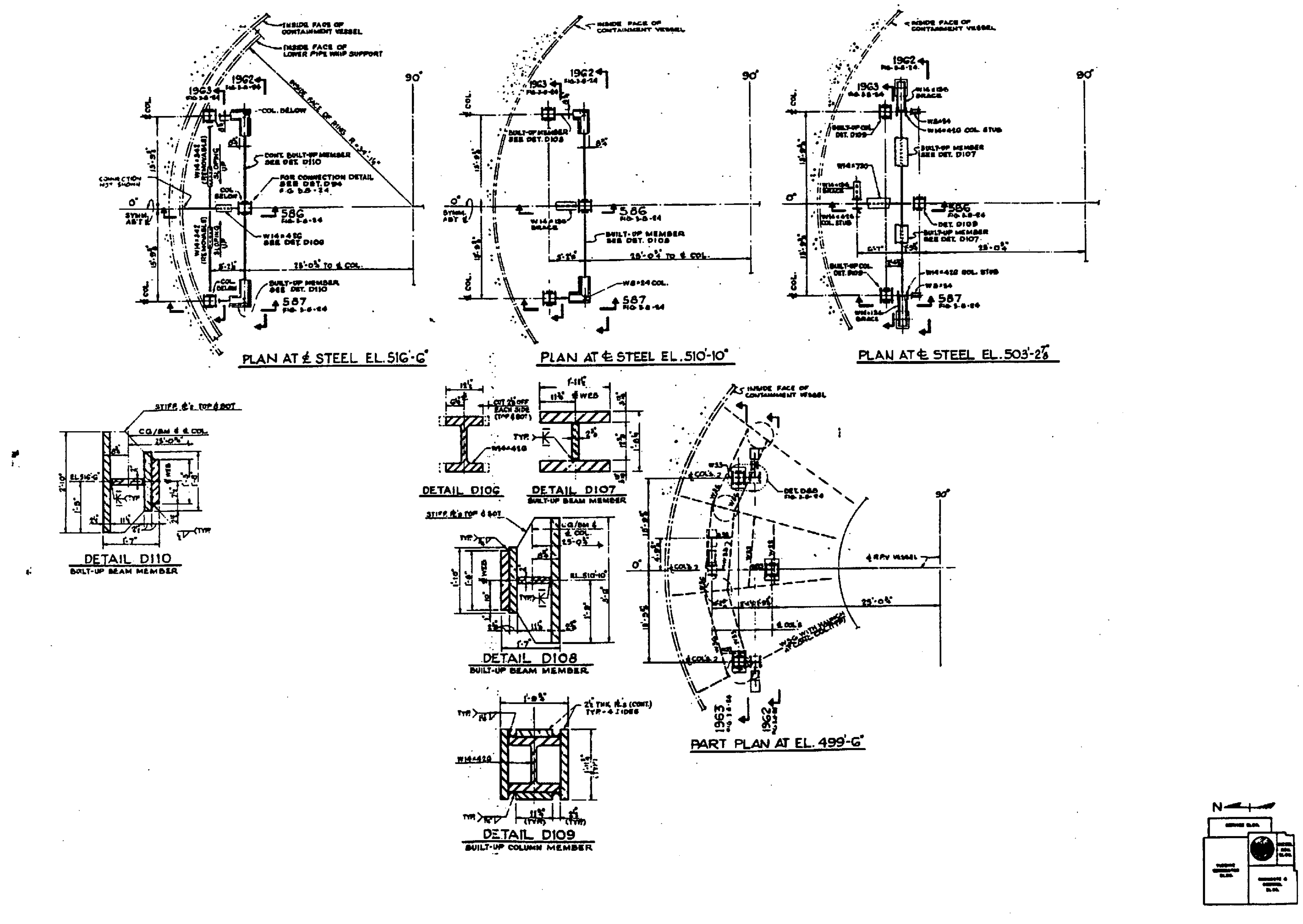
Form No. 960690



Columbia Generating Station
Final Safety Analysis Report

Primary Containment Vessel Equipment Hatch and CRD Removal Hatch

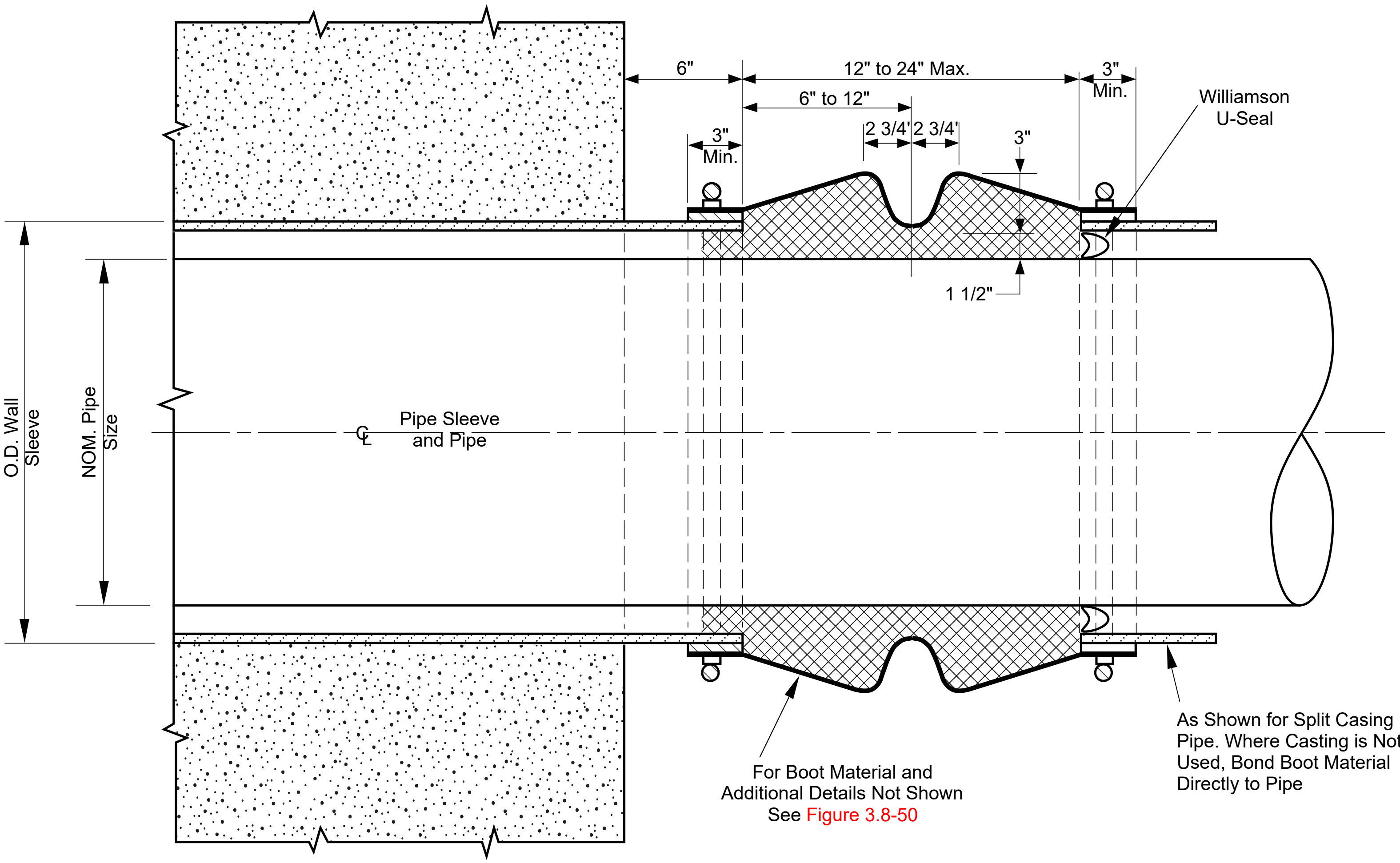
Draw. No. 010126.19	Rev.	Figure 3.8-60



Columbia Generating Station
Final Safety Analysis Report

Primary Containment Vessel Radial Beam Framing Systems

Draw. No. 020552.12	Rev.	Figure 3.8-61



Amendment 55
May 2001

Columbia Generating Station
Final Safety Analysis Report

Underground Pipe Penetration Flexible Watertight Closure Boot

Draw. No. 010126.20

Rev.

Figure 3.8-62

Form No. 960690

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section shows the transients that are used in the design of the ASME Boiler and Pressure Vessel (B&PV) Code Section III, Class 1, control rod drive (CRD) components, reactor assembly including core supports and reactor internals, main steam, and recirculation systems. The number of cycles or events for each transient is included. The design transients shown in this section are included in the design specifications for the 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 Code if applicable. (The first four operating condition categories correspond to Service Levels A, B, C, and D, respectively, which are used in Section III after the Winter 1976 Addenda.)

3.9.1.1.1 Control Rod Drive Transients

The normal and test service load cycles used for design purposes for the 40-year life of the CRDs are as follows:

	Transient	Category	Cycles
a.	Reactor startup/shutdown	Normal/upset	120
b.	Vessel pressure tests	Normal/upset	130
c.	Vessel overpressure	Normal/upset	10
d.	Scram test plus startup scrams	Normal/upset	300
e.	Operational scrams	Normal/upset	300
f.	Jog cycles	Normal/upset	30,000
g.	Shim/drive cycles	Normal/upset	1000

In addition to the above cycles, the following have been considered in the design of the CRD.

h.	Scram with inoperative buffer	Normal/upset	10

i.	Scram with stuck control blade faulted	Normal/upset	1
j.	Operating Basis Earthquake (OBE)*	Normal/upset	10
k.	Safe Shutdown Earthquake (SSE)**	Faulted	1

All ASME Class 1 components of the CRD have been analyzed according to ASME Code Section III.

The capability of the CRDs to withstand emergency and faulted conditions is verified by test rather than analysis.

3.9.1.1.2 Control Rod Drive Housing and In-Core Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and in-core housing are as follows:

	Transient	Category	Cycles
a.	Normal startup and shutdown	Normal/upset	120
b.	Vessel pressure tests	Normal/upset	130
c.	Vessel overpressure tests	Normal/upset	10
d.	Interruption of feedwater flow	Normal/upset	80
e.	Scram	Normal/upset	200
f.	OBE	Normal/upset	10
g.	SSE	Faulted	1

* The frequency of occurrence of this transient would indicate emergency category. However, for conservatism, the OBE condition was analyzed as an upset condition. Ten peak OBE cycles are postulated.

** SSE is a faulted condition; however, in the stress analysis, it was treated as emergency with lower stress limits.

CRD Housing Only

h.	Stuck rod scram	Normal/upset	1
i.	Scram no buffer	Normal/upset	10

3.9.1.1.3 Hydraulic Control Unit Transients

The normal and test service load cycles used in the original design and fatigue analysis for the 40-year life of the hydraulic control unit (HCU) are as follows:

	Transient	Category	Cycles
a.	Normal startup and shutdown	Normal/upset	120
b.	Vessel pressure tests	Normal/upset	130
c.	Vessel overpressure tests	Normal/upset	10
d.	Scram tests (cold)	Normal/upset	300
e.	Operational scrams (hot)	Normal/upset	300
f.	Jog cycles	Normal/upset	30,000
g.	Drive cycles	Normal/upset	1000
h.	Scram with stuck scram discharge valve	Normal/upset	1
i.	OBE	Normal/upset	10
j.	SSE	faulted	1

3.9.1.1.4 Core Support and Reactor Internals Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40-year life of the core support and reactor internals are shown in Table 3.9-1.

3.9.1.1.5 Nuclear Steam Supply System Scope Main Steam System Transients

The following transients are considered in the stress analysis of the main steam piping between the reactor pressure vessel (RPV) and the outer containment isolation valve:

	Transient	Category	Cycles
a.	Startup	Normal	120
b.	Loss of feedwater pump isolation valves closed	Upset	10
c.	Scrams	Upset	180
d.	Shutdown	Normal	111
e.	Reactor overpressure delayed scram	Emergency	1
f.	Single safety/relief valve (SRV) blowdown	Upset	8
g.	Automatic blowdown	Emergency	1
h.	Design pressure leak test	Test	130
i.	OBE	Upset	50
j.	1.25 P hydrotest	Test	3
k.	SSE	Faulted	1
l.	Pipe rupture	Faulted	1

3.9.1.1.6 Recirculation System Transients

The following transients are considered in the stress analysis of the recirculation piping:

	Transient	Category	Cycles
a.	Startup	Normal	120
b.	Turbine roll and increase to power	Normal	120
c.	Loss of feedwater heater	Upset	10

d.	Partial feedwater heater bypass	Upset	70
e.	Scrams	Upset	180
f.	Shutdown	Normal	111
g.	Loss of feedwater pump isolation valves closed	Upset	10
h.	Reactor overpressure with delayed scram	Emergency	1
i.	Single SRV blowdown	Upset	8
j.	Automatic blowdown	Emergency	1
k.	Design pressure leak test	Test	130
l.	OBE	Upset	50
m.	1.25 P hydrotest	Test	3
n.	SSE	Faulted	1
o.	Pipe rupture	Faulted	1

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes the RPV, support skirt, and shroud support. The cycles listed in Table 3.9-1 were specified in the reactor assembly design and fatigue analysis.

3.9.1.1.8 Main Steam Isolation Valve Transients

The main steam isolation valves (MSIV) are designed for the following service conditions and thermal cycles:

	Transient	Category	Cycles
a.	Preoperational at 100°F	Normal/upset	150
b.	Startup (heating 100°F)	Normal/upset	120

c. Shutdown

	Transient	Category	Cycles
1.	Cooling cycles at 100°F/hr 540°F to 375°F	Normal/upset	120
2.	Cooling cycles at 270°F/hr 375°F to 330°F	Normal/upset	120
3.	Cooling cycles at 100°F/hr 330°F to 100°F	Normal/upset	120

d.	Scram cooling cycles at 100°F/hr	Normal/upset	180

e. Emergency and faulted transients

1.	546°F to 281°F in 15 sec	Emergency/faulted	1
2.	546°F to 375°F in 3.3 minutes	Emergency/faulted	1
	375°F to 281°F at 300°F/hr	Emergency/faulted	1
3.	546°F to 375°F in 10 minutes	Emergency/faulted	8
4.	375°F to 281°F at 100°F/hr	Emergency/faulted	8
5.	546°F to 583°F in 2 sec	Emergency/faulted	1
	583°F to 538°F in 30 sec	Emergency/faulted	1
	538°F to 400°F at 100°F/hr	Emergency/faulted	1

	400°F to 546°F at 100°F/hr	Emergency/faulted	1
6.	561°F to 500°F in 7 minutes	Emergency/faulted	10
7	500°F to 400°F at 100°F/hr	Emergency/faulted	10
8.	400°F to 546°F at 100°F/hr	Emergency/faulted	10

3.9.1.1.9 Main Steam Safety/Relief Valve Transients

The transients used in the analysis of the SRV are as follows:

	Transient	Category	Cycles
a.	Preoperational and inservice testing (100°F/hr)	Normal/upset	150
b.	Startup (100°F/hr) and pressure increase (0 psig to 1000 psig)	Normal/upset	120
c.	Shutdown (100°F/hr, pressure decrease to 0 psig)	Normal/upset	120
d.	Scram	Normal/upset	180
e.	System pressure and temperature decay from 1000 psig 546°F to 35 psig and 281°F within 15 sec	Emergency/faulted	1
f.	System temperature change from 546°F to 375°F within 3.3 minutes and from 375°F to 281°F at a rate of 300°F/hr. Pressure change from 1000 psig to 35 psig.	Emergency/faulted	1

g.	System temperature change from 546°F to 375°F within 10 minutes and from 375°F to 281°F at a rate of 100°F/hr. Pressure change from 1000 psig to 35 psig.	Normal/upset	8
h.	System temperature change from 546°F to 583°F within 2 sec, from 583°F to 538°F within 30 sec, and from 538°F to 400°F and return to 546°F at a rate of 100°F/hr. Pressure change from 1000 psig to 1350 psig thence to 240 psig and return to 1000 psig.	Emergency/faulted	1
i.	System temperature changes, greater than 30°F, from 561°F to 500°F within 7 minutes and from 500°F to 400°F and return to normal operating temperature of 546°F at a rate of 100°F/hr. Pressure change from 1000 psig to 1180 psig to 240 psig and return to normal operating of 1000 psig.	Emergency/faulted	10

Paragraph NB-3552 of ASME Code Section III excludes various transients and provides means for combining those which are not excluded. Review and approval of the equipment supplier's certified calculation provides assurance of proper accounting of the specified transients.

3.9.1.1.10 Recirculation Flow Control Valve Transients

The following pressure and temperature transients were considered in the design of the recirculation system flow control valve (this valve has been blocked open):

	Transient	Category	Cycles
a.	Startup (100°F/hr) heating rate 70°F to design temperature	Normal/upset	300
b.	Small temperature changes (29°F)	Normal/upset	600

c.	50°F step changes	Normal/upset	200
d.	Safety/relief valve blowdowns (single valve) (546°F to 375°F in 10 minutes)	Normal/upset	30
e.	Safety valve transient (110% of design pressure)	Normal/upset	1
f.	Installed hydrotests		
	1. 1300 psig	Testing	130
	2. 1670 psig	Testing	3
g.	Automatic blowdown (546°F to 281°F in 15 sec)	Emergency	2
h.	Improper start of pump in cold loop (130°F step to 546°F for 15 sec)	Emergency	1

3.9.1.1.11 Recirculation Pump Transients

The following transients are listed in the design specification as a requirement for design considerations. However, a submitted certified analysis considering thermal stresses was not required. The vendor was required to submit a certification of compliance. The submitted certified design calculations only considered pressure transient. Nozzle piping loads were considered in accordance with the following ASME paragraph:

> "The pump case shall be designed to withstand secondary stresses due to piping reactions in accordance with ASME Code, Section III, 1971."

	Transient	Category	Cycles
a.	Heatup and cooldown at 100°F/hr	Normal/upset	300
b.	±29°F temperature changes	Normal/upset	600
c.	±50°F temperature changes	Normal/upset	200

d.	RPV pressure transients to 110% design pressure	Normal/upset	1
e.	SRV blowdowns	Emergency	8
f.	Improper pump startup, 100°F to 552°F in 15 sec	Emergency	1
g.	Cooling transient 552°F to 281°F in 15 sec	Faulted	1
h.	Hydrotest to 1300 psig	Testing	130
i.	Hydrotest to 1670 psig	Testing	3

3.9.1.1.12 Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves:

	Transient	Cycles
a.	50°F to 575°F at 100°F/hr	300
b.	$\pm$29°F between limits of 50°F and 575°F, instantaneous	600
c.	$\pm$50°F between limits of 50°F and 546°F, instantaneous	200
d.	546°F to 375°F, instantaneous	30
e.	546°F to 281°F, instantaneous	2
f.	130°F to 546°F, instantaneous	1
g.	110% design pressure at 575°F	1
h.	1300 psi at 100°F installed hydrostatic test	130
i.	1670 psi at 100°F installed hydrostatic test	3

3.9.1.1.13 Balance-of-Plant Transients

The transients used in design and fatigue analysis of the balance-of-plant (BOP) components are listed in Table 3.9-1 with the exception that 50 maximum stress cycles due to an OBE are used in fatigue evaluations. Table 3.9-1 also shows the thermal cycles which are tracked to provide an indication of reactor cumulative fatigue usage.

3.9.1.2 Nuclear Steam Supply System Computer Programs Used in Original Analysis

The following italicized text is historical information provided in the FSAR to support the application for an operating license. As such, it is not subject to change and therefore has not been verified for accuracy or updated during the FSAR upgrade process per 10 CFR 50.71(e).

The following sections discuss computer programs used in the analysis of the major safety-related components. (Computer programs were not used in all components; hence not all components are listed.) The nuclear steam supply system (NSSS) programs can be divided into two categories.

Vendor Programs

The verification of the following two groups of vendor programs is ensured by contractual requirements between GE and the vendor. The quality assurance procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10 CFR 50, Appendix B.

CB&I Programs

711 GENOZZ
948 NAPALM
1027
846
781 KALNINS
979 ASFAST
766 TEMAPR
767 PRINCESS
928 TGRV
962E962A
984
992 GASP
1037 DUNHAM'S
1335
1606 and 1657 HAP

1635
953
955 MESHPLOT
1028
1038

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10 CFR 50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

MASS.
SNAP (MULTISHELL)
HEATER
PISYS
ANSI7
SAP4G
FTFLG01
ANSYS
BSTIF01
RTRMEC
POSUM
BILRD
RVFOR
TSFOR
LUGST
PDA
DYSEA
SPECA
SEISM

3.9.1.2.1 Reactor Pressure Vessel and Internals

3.9.1.2.1.1 Reactor Pressure Vessel. CB&I Programs are used to analyze the RPV. Detailed descriptions are provided in Sections 3.9.1.2.1.1.1 through 3.9.1.2.1.1.20.

3.9.1.2.1.1.1 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 will not comply with the specifications, the program will modify the design and redesign it to yield an acceptable result.

3.9.1.2.1.1.2 <u>CB&I Program 9-48 - "NAPALM"</u>. 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 0 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.

3.9.1.2.1.1.3 <u>CB&I Program 1027</u>. This program is a computerized version of the analysis method contained in the "Welding Research Council Bulletin F107, December 1965."

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 (two normal stresses and one shear stress). This program provides the same information as the manual calculation and the input data is essentially the geometry of the vessel and attachment.

3.9.1.2.1.1.4 <u>CB&I Program 846</u>. 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.

3.9.1.2.1.1.5 <u>CB&I Program 781 - "KALNINS"</u>. 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 axisymmetric loading:

a. Preload condition,
b. Internal pressure, and
c. Thermal load.

3.9.1.2.1.1.6 CB&I Program 979 - "ASFAST". ASFAST Program (Program 979) performs the stress analysis of axisymmetric, bolted closure flanges between head and cylindrical shell.

3.9.1.2.1.1.7 CB&I Program 766 - "TEMAPR". This program will reduce any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient will have the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plant 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.

3.9.1.2.1.1.8 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 Pressure Vessel Code.

3.9.1.2.1.1.9 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 can be represented by a three-dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates, TGRV will calculate and give as output the steady state or transient temperature distributions in the structure as a function of time.

3.9.1.2.1.1.10 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.

3.9.1.2.1.1.11 <u>CB&I Program 984</u>. 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.

3.9.1.2.1.1.12 <u>CB&I program 992 - "GASP"</u>. The GASP computer program, originated by Professor E. L. Wilson of the University of California at Berkeley, uses the finite element method to determine the stresses and displacements of plane or axisymmetric structures of arbitrary geometry and is written in FORTRAN IV. For a detailed account, see Reference 3.9-1.

As mentioned above, the program determines the stresses and displacements of plane or axisymmetric 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.

3.9.1.2.1.1.13 <u>CB&I Program 1037 - "DUNHAM'S"</u>. DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacements of axisymmetric structures of arbitrary geometry subjected to either axisymmetric loads or non-axisymmetric loads represented by a Fourier series.

This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle non-axisymmetric 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.

3.9.1.2.1.1.14 <u>CB&I Program 1335</u>. To obtain stresses in the shroud support, the baffle plate must be made 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.

3.9.1.2.1.1.15 <u>CB&I Programs 1606 and 1657 - "HAP"</u>. The HAP program is an axisymmetric 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.

3.9.1.2.1.1.16 <u>CB&I Program 1635</u>. Program 1635 offers three features to aid the stress analyst in preparing a stress report.

a. Generates punched card input for program 767 (PRINCESS) from the stress output of program 781 (KALNINS),

b. Writes a stress table in a format such that it can be incorporated into a final stress report, and

c. Has the option to remove through-wall thermal bending stress and report these results in a stress table similar to the one mentioned above.

3.9.1.2.1.1.17 <u>CB&I Program 953</u>. The program is a general purpose program, which does the following:

a. Prepares input cards for the thermal model,

b. Prepares the node and element cards for the finite element model, and

c. 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.

3.9.1.2.1.1.18 <u>CB&I Program 955 - "MESHPLOT"</u>. 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.

3.9.1.2.1.1.19 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.

3.9.1.2.1.1.20 CB&I Program 1038. This program calculates the loads required to satisfy the compatibility between the shroud baffle plate and the jet pump adapters in the RPV.

3.9.1.2.1.2 Reactor Internals. The following computer programs are used in the analysis of the core support structures and other safety-related reactor internals: MASS, SNAP (MULTISHELL), GASP, NOHEAT, FINITE, DYSEA, SHELLS, HEATER, FAP-71, and CREEP-PLAST. Detailed descriptions of these programs are provided in 4.1.

3.9.1.2.2 Nuclear Steam Supply System Piping

3.9.1.2.2.1 Piping Analysis Program/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 3.9-2.

3.9.1.2.2.2 Component Analysis/ANSI7. The ANSI7 computer program determines stress and accumulative usage factors in accordance with NB-3600 of the ASME Code, Section III. The program was written to perform stress analysis in accordance with the ASME Code sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.2.3 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.

3.9.1.2.2.4 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.

3.9.1.2.2.5 Integral Attachment/LUGST. The computer program "LUGST" evaluates the stresses in the pipe wall that are produced by loads applied to the integral attachments.

3.9.1.2.2.6 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 also 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.

3.9.1.2.3 Recirculation Pump

No computer programs were used in the design of the recirculation pumps.

3.9.1.2.4 Emergency Core Cooling System Pumps and Motors

3.9.1.2.4.1 Rotor Assembly Analysis 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 shaft (rotating parts only). The shaft deflection due to 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.

3.9.1.2.4.2 Structural Analysis Program/SAP4G. SAP4G is used to analyze the structural and functional integrity of the emergency core cooling system (ECCS) pump/motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress, and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.4.3 Effects of Flange Joint Connections/FTFLG01. The flange joints connecting the pump bowl castings are analyzed using FTFLG01. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix XI, Article XI-3000, and Appendix L of Section III of the ASME Boiler and Pressure Vessel Code.

3.9.1.2.4.4 Structural Analysis of Discharge Head/ANSYS. ANSYS is used to analyze the pump discharge head flange and bolting taking into account the prying action developed by the flat face contact surface. The program is described in detail in Section 3.12.

3.9.1.2.4.5 Beam Element Data Processing/POSUM. POSUM is a computer code designed to process SAP generated beam element data for pump or heat exchanger models. The purpose is to determine the load combination that would produce the maximum stress in a selected beam element. It is intended for use on residual heat removal (RHR) heat exchangers with four nozzles or ECCS pumps with two nozzles.

3.9.1.2.5 Residual Heat Removal Heat Exchangers

3.9.1.2.5.1 Structural Analysis Program/SAP4G. SAP4G is used to analyze the structural and functional integrity of the RHR heat exchangers. The description of this program is provided in Section 3.9.1.2.4.2.

3.9.1.2.5.2 Local Stiffness Calculations/BSTIF01. BSTIF01 is used to estimate the local stiffness of the heat exchanger shell at the attachment point of the supports. The method used in this program is based on the shell stiffness calculations by P. P. Bijlaard as groundwork for Welding Research Council Bulletin 107. The results of BSTIF01 are used to determine equivalent beam properties of the lower and upper heat exchanger support bracket to shell attachments included in the finite element model of the heat exchanger.

3.9.1.2.5.3 Calculation of Shell Attachment Parameters and Coefficients/BILRD.

BILRD is used to calculate the shell attachment parameters and coefficients used in the stress analysis of the support to shell junction. The method, per Welding Research Council Bulletin No. 107, is implemented in BILRD to calculate local membrane stress due to the support reaction loads on the heat exchanger shell.

3.9.1.2.5.4 Beam Element Data Processing/POSUM. POSUM is used to process SAP generated beam element data. The description of this program is provided in Section 3.9.1.2.4.5.

3.9.1.2.6 Dynamic Loads Analysis

3.9.1.2.6.1 Dynamic Analysis Program/DYSEA. DYSEA simulates a beam model in the annulus pressurization dynamic analysis. A detailed description of DYSEA is provided in Section 4.1. DYSEA employs a preprocessor program named GEAPL. GEAPL corrects pressure, time histories into time varying loads, and forcing functions for DYSEA. The overall resultant forces and moments time histories at specified points of resolution can also be obtained from GEAPL.

3.9.1.2.6.2 Acceleration Response Spectrum Program/SPECA. SPECA generates acceleration response spectrum for an arbitrary input time history of piece-wise linear accelerations, i.e., to compute maximum acceleration responses for a series of single-degree-of-freedom systems subjected to the same input. It can accept acceleration time histories from a random file. It also has the capability of generating the broadened/enveloped spectra when the spectral points are generated equally spaced on a logarithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analysis.

3.9.1.2.6.3 Fuel Support Loads Program/SEISM. SEISM02 computes the vertical fuel support loads using the component element methods in dynamics. The methodology is based on the Reference 3.9-3.

3.9.1.2.7 Balance-of-Plant Computer Programs

A list of the principal computer programs used in dynamic and static analyses in the BOP scope is given in Section 3.12. With the exception of the Burns and Roe developed program, these programs are recognized and widely used in the industry with a history of successful applications. The Burns and Roe developed program listed in Section 3.12 is documented, verified, and maintained by Burns and Roe as described in SRP 3.9.1, II2.b.

3.9.1.2.7.1 S/RVDAM4. S/RVDAM4 (Safety/Relief Valve Discharge Analyses Model 4) is a computer model which simulates the transient flow of steam, air, and water in a SRV discharge

line (SRVDL) for a time period of approximately 0.5 sec after SRV opening. The model calculates transient fluid properties, forces, and thermal distributions in the SRVDL.

The piping system is initially filled with air and a water slug at the exit submerged in the suppression pool. Upon SRV actuation, steam enters the line and compresses the air which expels the water slug. The piping system is represented by two models: (a) a gas (steam and air), and (b) a water slug, which are coupled by common pressure and velocity at the air-water interface. The gas flow equations are expressed in finite difference form solved with the method of characteristics. Provision for axial variation in flow area is included. Motion of the water slug is solved with a one-dimensional ordinary differential form of the momentum equation is integrated axially to determine flowrate and displacement.

S/RVDAM4 is based on the analytical model described in the GE Report NEDE-23749-P (Reference 3.9-4) and GE computer code RVFOR04 described in NEDE-24695 (Reference 3.9-5).

Program Version and Computer

Currently S/RVDAM version 4 is being used by Burns and Roe, Inc. in conjunction with a CDC Computer. The system used was CDC 175.

Extent of Application

S/RVDAM4 is a transient piping fluid analysis program which began development in 1975 and is supported by Burns and Roe. It has been used on several in-house projects.

Test Problems

S/RVDAM4 has been benchmarked against problems provided in Reference 3.9-4 and 3.9-5 which have been compared with in-plant test data from Quad Cities, Monticello, and CAORSO boiling water reactor (BWR) plants.

3.9.1.3 Experimental Stress Analysis

When experimental stress analysis is used in lieu of analytical methods for Seismic Category I ASME Code items, the requirements for experimental testing enumerated in the ASME Code which are applicable for the specific components under test shall be applied. When testing is required for Seismic Category I non-ASME Code parts account shall be taken of size effects and dimensional tolerances which exist between the actual part and the test part or parts as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts, to ensure that the loads obtained from the test are a realistic or conservative representation of the load carrying capability of the actual structure under the postulated loading.

3.9.1.3.1 Experimental Stress Analysis of Piping Components

The following are the only NSSS components on which experimental stress analysis is used. These components have been tested to verify their design adequacy:

a. Pipe whip restraints, and
b. Snubbers.

Descriptions of the whip restraint and snubber tests are discussed in Sections 3.6 and 3.9.3.4, respectively.

3.9.1.3.2 Orificed Fuel Support, Vertical, and Horizontal Load Tests

The orificed fuel support experimental stress analysis is discussed in Section 3.9.1.4.2.4.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

All Seismic Category I equipment is evaluated for the faulted loading conditions. However, emergency stress limits rather than faulted stress limits are used in many cases. In all cases, with the exception of the refueling platform, the calculated stresses are within allowable limits. The slight overstress calculated in one member of the refueling platform under the emergency condition (which is greater than the faulted condition in this instance). (See Table 3.9-2s.) This calculated overstress has been judged to be acceptable because of the conservatism in the calculations and because functional use of the equipment is not impaired with the deformation of only one member of the platform. The following paragraphs show examples of the treatment of faulted conditions for the major components on a component by component basis. Additional discussion of faulted analysis can be found in Sections 3.9.3 and 3.9.5 and Table 3.9-2.

Sections 3.9.2.2 and 3.7 discuss the treatment of dynamic loads resulting from the postulated seismic and hydrodynamic events. Section 3.9.2.5 discusses the dynamic analysis of the reactor internals under faulted conditions including additional blowdown forces. Deformations under faulted conditions have been evaluated in critical areas. In all cases the identified design limits, such as clearance limits, are not violated.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Control Rod Drives. The ASME III Code components of the CRD have been analyzed for faulted conditions. The CRD component which is analyzed for the faulted condition is the indicator tube. The method of analysis and the maximum stresses for this component for various plant operating conditions are given in Table 3.9-2u.

The design adequacy of noncode components of the CRD has been verified by analysis and extensive testing programs on components parts, specially instrumented prototype drives, and production drives. The testing included postulated abnormal events as well as the service life cycle listed in Section 3.9.1.1.1.

3.9.1.4.1.2 Hydraulic Control Unit. The HCU has been qualified by test for upset and faulted conditions. The test response spectra (TRS) for the HCU enveloped the CGS required response spectra (RRS) for all frequencies ranging from 5 Hz to 100 Hz. The CGS unique HCU structural support configuration was considered in this evaluation.

3.9.1.4.1.3 Control Rod Drive Housing. A stress analysis of the CRD housing demonstrated that the calculated stresses were below the allowable stresses for all loading cases. The emergency condition calculated stresses were less severe than the normal and upset conditions. The calculated and allowable stresses at various loading conditions are shown in Table 3.9-2v.

3.9.1.4.2 Standard Reactor Internal Components

3.9.1.4.2.1 Control Rod Guide Tube. The maximum calculated stress on the control rod guide tube occurs at the flange during the faulted conditions. The faulted limit is 2.4 S_m where S_m is 16,000 psi at the design temperature of 575°F per ASME Code Section III, Table I-1.2 and F 1322-1. Table 3.9-2aa shows the calculated stresses are within the allowable limits.

3.9.1.4.2.2 Jet Pump. The elastic analysis for the jet pump faulted conditions shows that the maximum stress occurs at the riser brace and is 54,450 psi. The maximum allowable for this condition per ASME Code Section III, Subsection NG, is 3.6 S_m or 60,480 psi. Table 3.9-2w shows the loads summary.

3.9.1.4.2.3 Low-Pressure Coolant Injection Coupling. The maximum stress during the faulted condition on the low-pressure coolant injection (LPCI) coupling is bounded by the allowable limit of 3.6 S_m. Table 3.9-2y shows that the calculated stresses are within the allowable limits.

3.9.1.4.2.4 Orificed Fuel Support. Due to its complex configuration, a series of vertical and horizontal load tests were performed on the orificed fuel support (OFS) to verify the design. Results from these tests indicate that the component seismic and hydrodynamic loading of the OFS are well below the stress limit allowables with a safety margin of 1.26 for normal and upset and 1.5 for faulted conditions. (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 upset and 1.5 x 0.7 S_u for faulted.)

3.9.1.4.3 Reactor Pressure Vessel Assembly

The RPV assembly includes the RPV, support skirt, and shroud support. For the faulted conditions, RPV assembly is evaluated using elastic-analysis methods. For the support skirt and shroud support, buckling is evaluated for the compressive load. Table 3.9-2a shows that the calculated stresses are within the allowable limits.

3.9.1.4.4 Core Structure

The dynamic evaluations for faulted conditions of the core structure are discussed in Section 3.9.5. The calculated and allowable stresses are summarized in Table 3.9-2b.

3.9.1.4.5 Main Steam Isolation, Recirculation Gate, and Safety/Relief Valves

Standard design rules, as defined in ASME Code Section III, are utilized in the analysis of pressure boundary components of Seismic Category I valves. Conventional, elastic stress analysis was used to evaluate components not covered by the ASME Code. The Code allowable stresses were applied to determine acceptability of structure under applicable loading conditions including the faulted condition. Maximum stresses and highest calculated loads are summarized in Tables 3.9-2g, 3.9-2h, and 3.9-2j; and Tables 3.9-2d and 3.9-2e, respectively, for the SRVs, MSIV, and recirculation gate valves.

3.9.1.4.6 Recirculation System Flow Control Valve

The recirculation system flow control valve was analyzed for faulted conditions using the elastic analysis criteria from the ASME Code Section III. The analysis and results for various plant operating conditions are summarized in Tables 3.9-2e and 3.9-2f.

3.9.1.4.7 Main Steam and Recirculation Piping

For main steam and recirculation system piping, elastic analysis methods are used for evaluating faulted loading conditions. The 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 3.9-2d and 3.9-2e, respectively.

3.9.1.4.8 Nuclear Steam Supply System Pumps, Heat Exchangers, and Turbine

The recirculation, emergency core cooling system (ECCS), reactor core isolation cooling (RCIC), and standby liquid control (SLC) pumps, residual heat removal (RHR) heat exchangers, and RCIC turbine have been analyzed for the faulted loading conditions identified in Section 3.9.1.1. In all cases, stresses were within the elastic limits. The analytical

methods, stress limits, and allowable stresses are shown in Table 3.9-2 under the respective equipment table.

3.9.1.4.9 Control Rod Drive Housing Supports

The stress criteria, loadings, calculated stresses, and stress limits for faulted condition for the CRD housing supports are shown in Table 3.9-2ac.

3.9.1.4.10 Fuel Storage Racks

The stress criteria, loadings, calculated stresses, and stress limits for the faulted conditions for the new fuel storage racks are shown in Table 3.9-2s.

3.9.1.4.11 Fuel Assembly Including Channels

The BWR fuel assembly design bases and analytical methods including those applicable to the faulted conditions are contained in References 3.9-6, 3.9-8, 3.9-19 and 3.9-21.

3.9.1.4.12 Refueling Equipment

Refueling and servicing equipment (Table 3.9-2s) which is important to safety is classified per the requirements of 10 CFR 50, Appendix A. This equipment, the failure of which is prevented from degrading a safety related component is listed in Table 3.2-1 and is classified to the appropriate Seismic Category. This equipment was 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 were generated and combined for normal, upset, and faulted conditions. Stresses were compared, depending on the specific safety class of the equipment, to allowables specified in ASME, ANSI, AISC, or other industrial codes and standards. The loading conditions, acceptance criteria, calculated, and allowable stresses are shown in Table 3.9-2s.

3.9.1.4.13 Balance-of-Plant Equipment

With the exception of pipe whip restraint design, the faulted condition was evaluated in accordance with ASME Section III by elastic systems and components analysis. Inelastic stress analysis methods were not utilized for design of any of these components. Pipe whip restraint design is described in Section 3.6.2.

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

The test program was divided into three phases: preoperational vibration, startup vibration, and operational transients. See Section 14.2.12.3.17 for a discussion of the piping thermal expansion test program.

The following italicized text is historical information provided in the FSAR to support the application for an operating license. As such, it is not subject to change and therefore has not been verified for accuracy or updated during the FSAR upgrade process per 10 CFR 50.71(e).

3.9.2.1.1 Preoperational Vibration Testing

During the preoperational test phase it is verified that operating vibrations in all safety-related piping systems included in the preoperational test program are within acceptable limits. This phase of the test uses visual observation. If, during the initial system operation, visual observation indicates that piping vibration is significant, measurements are made with a hand-held vibrograph. The results of those measurements will be reviewed by the appropriate engineering group to determine the acceptability of the measured vibration values. If the measured vibration values are not acceptable, appropriate design modifications will be made and the system retested. Visual observations are made during initial operation of all piping systems. During the preoperational test program described in Section 14.2, all safety-related systems with the exception of the main steam, recirculation, RCIC, feedwater, and SRV discharge piping are operated up to rated system flow condition. These remaining piping systems are monitored and/or visually inspected during the startup program. Refer to Sections 3.9.2.1.3 and 14.2.12.3.33.

3.9.2.1.2 Small Attached Piping

During visual observation special attention is given to small attached piping and instrument connections to ensure that they are not in resonance with their associated main process piping. If the operating vibration acceptance criteria are not met, appropriate corrective action will be taken and retesting performed.

3.9.2.1.3 Startup Vibration

The purpose of this phase of the program is to verify that the main steam, recirculation, reactor water cleanup (RWCU), feedwater, RHR, SRV discharge, and RCIC steam piping vibration are within acceptable limits. Because of limited access during power operation caused by high radiation levels, remote monitoring is required for drywell piping systems during this phase of the test. The piping vibration startup test is described in 14.2.12.3.17 and 14.2.12.3.33.

3.9.2.1.4 Operating Transient Loads

The purpose of the operating transient test phase is to verify that pipe stresses are within Code limits. Compliance with the acceptance criteria is the method used to accept the data collected during the transient. Remote vibration and deflection measurements are taken during the following transients:

a. Recirculation pump starts,

b. Recirculation pump trips at 100% of rated flow,

c. Turbine stop valve closure at 75% and 100% power,

d. Manual discharge of each SRV at 1,000 psig and at planned transient tests that result in SRV discharge, e.g., MSIV full isolation,

e. RCIC operation at maximum steam flow,

f. MSIV full isolation,

g. RHR pump starts and trips,

h. Generator load reject at 25% power;

i. Reactor feed pump trip at 100% power, and

j. Recirculation pump transfer to 15 Hz during simulated RPT.

3.9.2.1.5 Test Evaluation and Acceptance Criteria

The piping response to test conditions are considered acceptable if the organization responsible for the stress report reviews the test results and determines that the tests 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. 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 Level 1 and 2, are described in the following paragraphs.

For steady-state vibration, the piping peak stress due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criterion and 5000 psi for Level 2 criterion. These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of ASME Code for 10^6 cycles. The definitions of Level 1 and Level 2 criteria are clarified in the text revision attached.

3.9.2.1.5.1 Level 1 Criterion. Level 1 establishes the 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 will be placed in a satisfactory hold conditions and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1.5.2 Level 2 Criterion. Level 2 specifies the level of pipe motion which, if exceeded, requires that the responsible piping design engineer be advised. If the Level 2 limit is not satisfied, 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. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

3.9.2.1.6 Corrective Actions

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 held or terminated as soon as Level 1 criteria is violated. As soon as possible after the test hold or termination, the following corrective actions will be taken:

a. Installation Inspection. A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. Snubbers shall be about the midpoint of the total travel range at operating temperature. Hangers shall be in their operating range between hot and cold settings. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

b. Instrumentation Inspection. The instrumentation installation and calibration are checked and any discrepancies corrected. Additional instrumentation is added, if necessary.

c. Repeat Test. If actions (a) and (b) identify discrepancies that could account for failure to meet Level 1 criteria, the test is repeated.

d. Resolution of Findings. If the Level 1 criteria is violated on the repeat test or no relevant discrepancies are identified in (a) and (b), the organization responsible for the stress report shall review the test results and criteria to determine if 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:

a. *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 must be identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.*

b. *Instrumentation Inspection. The instrumentation installation and calibration are checked and any discrepancies corrected.*

c. *Repeat Test. If actions (a) and (b) 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.*

d. *Documentation of Discrepancies. If the test is not repeated, the discrepancies found under actions (a) and (b) above are documented in the test evaluation report and correlated with the test condition. The test is not considered complete until the test results are reconciled with the acceptance criteria.*

3.9.2.1.7 Measurement Locations

Remote shock and vibration measurements are made in the three orthogonal directions at appropriate locations on the main steam, recirculation, feedwater, RCIC, and SRV discharge piping. The exact locations are finalized and are documented in the Startup Vibration Test Procedure described in Section 14.2.12.3.33. During preoperational testing prior to fuel load, visual inspection of all safety-related piping is made, and any visible vibration measured with a handheld instrument.

For each of the selected remote measurement locations, Level 1 and 2 deflection and vibration limits are prescribed in the startup test specification. Level 2 limits are based on the results of the stress report adjusted for operating mode and instrument accuracy; Level 1 limits are based on maximum allowable Code stress limits.

3.9.2.2 Seismic and Hydrodynamic Loads Qualification of Safety-Related Mechanical Equipment

This section describes the criteria for dynamic load qualification of mechanical safety-related equipment and also describes the qualification testing and/or analysis applicable to this plant for all the major components on a component by component basis. In some cases, a module or assembly consisting of mechanical and electrical equipment was qualified as a unit; for

example, motor-powered pumps. These modules are generally discussed in this paragraph rather than providing discussion of the separate electrical parts in Section 3.10. Dynamic load qualification testing for pumps is discussed in Section 3.9.3.2. Electrical supporting equipment such as control consoles, cabinets, panels, and valve motor operators is discussed in Section 3.10.

3.9.2.2.1 Test and Analysis Criteria and Methods

The ability of equipment to perform its safety-related function during and after the application of dynamic loads is demonstrated by tests and/or analysis. Selection of testing, analysis, or a combination of the two was determined by the type, size, shape, and complexity of the equipment being considered. When practical, equipment operability is demonstrated by testing; otherwise the operability is demonstrated by mathematical analysis or by a combination of in situ testing and mathematical analysis.

Equipment which is large and/or simple, is usually qualified by analysis or test to show that the loads, stresses, and deflections are less than the allowable maximum. Analysis and/or test is also used to show there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for hydrodynamic loads (see Section 3.7.3.4). If a natural frequency lower than these is discovered, dynamic tests may be conducted and in conjunction with mathematical analysis used to verify operability and structural integrity at the required dynamic input conditions.

When analysis was chosen as the method of qualification, in most cases, the horizontal loads were combined by square-root-of-the-sum-of-the-squares (SRSS) to find the worst-case horizontal load. See References 3.9-10 through 3.9-12 for basis of using SRSS. This load was then applied normal to the weakest axis of the equipment being qualified and the resultant stresses, strains, and deflections were combined with those resulting from the vertical load using absolute sum. The qualifying analysis, in all cases, complied with the intent of Regulatory Guide 1.92.

The response spectra or time history of the attachment point determined by building or piping analysis is used as the input motion in the equipment test or analysis. Equipment is tested in its operational mode and verified during and after the test. The tested equipment is either an exact duplicate of the supplied equipment or is representative of a family of equipment of the same design and structure. See Section 3.10 for details of test input load development.

Valve operability was demonstrated by dynamic tests, application of a static load/stroke test, analysis, or a combination of these methods. The load/stroke test is defined as a test during which a static load, equivalent to the worst-case faulted load in the worst direction, determined by analysis, is applied, in situ, to the extended structure. The valve is stroked before, during, and after the test. The stroke time required in the valve specification must be met on each stroke.

a. Deflection analysis was used to demonstrate operability for valves whose mechanism for becoming inoperable is known to be metal-to-metal contact between moving parts. In such cases the analytically determined clearance between the critical moving parts must be greater than the manufacturer's design clearance with margin when the worst-case dynamic plus operational load is applied. Valves of the same type, which could not meet these analytic criteria, successfully passed the in situ load/stroke test;

b. Valves whose body and operator are supported rigidly to the same structure did not require operability under the load to be demonstrated;

c. Operability under worst-case dynamic plus operational loads was demonstrated for the following three valves by demonstrating similarity to successfully tested valves;

1. HPCS-V-23 is qualified by similarity to tested valve HPCS-V-11, both of which are Anchor-Darling 900-lb globe valves with identical materials. The yokes that are the critical structural elements have the same cross section. HPCS-V-11 is a 10-in. valve while HPCS-V-23 is a 12-in. valve. The tested valve is equipped with a Limitorque model SMB-3 (150) operator (weight 1150 lb) while the valve being qualified is equipped with a SMB-4 (150) operator (weight 1765 lb). The larger SMB-4 is less likely to become stalled during a dynamic event. The test load applied was the equivalent of a 2.5g load to the larger operator. The required load to demonstrate operability during a faulted event is 2.5g.

2. Valve HPCS-V-1 is qualified based on its similarity to HPCS-V-15. Both are Anchor-Darling 150-lb gate valves of the same type, made of identical materials. HPCS-V-15 is an 18-in. valve equipped with a Limitorque SB-2-60 operator, while HPCS-V-1 is a 14-in. valve equipped with an SMB-00-25 operator. A deflection analysis was performed showing that the tested valve (HPCS-V-15) had less clearance between moving parts during its test than the valve being qualified (HPCS-V-1) would have during a postulated faulted event.

3. Valve CRD-V-10 is qualified based on its similarity to CRD-V-11. Both valves are manufactured by I.T.T. Hammel-Dahl of the same design. Both are 600-lb fail-closed-gate valves with pneumatic operators of the same type. The faulted load on each is approximately 1g. Valve CRD-V-11 was statically tested at 1.6g. Structural analysis of the two valve assemblies shows that no yielding occurs for loads up to 6g in any direction and that the tested valve (CRD-V-11) is stressed to a higher

level and deflects more than the valve being qualified (CRD-V-10) by similarity for any applied acceleration.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude, a hammer blow Fourier transform in situ test or analysis.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

3.9.2.2.1.1 Random Vibration Input. See Section 3.10.

3.9.2.2.1.2 Application of Input Motion. See Section 3.10 for test input motion.

3.9.2.2.1.3 Fixture Design. The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

3.9.2.2.1.4 Prototype Testing. Some equipment testing is conducted on prototypes of the equipment installed in this plant.

3.9.2.2.2 Seismic and Hydrodynamic Load Qualification of Specific Nuclear Steam Supply System Mechanical Components

The following sections discuss the testing or analytical qualification of NSSS equipment. Dynamic qualification is also described in Sections 3.9.1.4, 3.9.3.1, and 3.9.3.2.

3.9.2.2.2.1 Jet Pumps. A dynamic analysis of the jet pumps was performed and the resulting stresses were below the design allowable.

3.9.2.2.2.2 Control Rod Drive and Control Rod Drive Housing. The dynamic qualification of the CRD housing (with enclosed CRD) was done analytically and the stress results of these analysis established the structural integrity of these components. Dynamic tests have been conducted to verify the operability of the CRD during seismic and hydrodynamic events. A simulated test imposing a static bow in the fuel channels was performed with the CRD functioning satisfactorily.

3.9.2.2.2.3 Core Support (Orificed Fuel Support and Control Rod Guide Tube). A detailed analysis imposing dynamic effects due to seismic and hydrodynamic events has shown that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2.2.4 Hydraulic Control Unit. The HCU was evaluated by comparing the floor response spectra with the maximum HCU capability as determined by test and analysis.

3.9.2.2.2.5 Fuel Assembly Including Channels. General Electric initial core BWR fuel channel design bases, analytical methods, and evaluation results, including seismic and loss-of-coolant accident (LOCA) consideration, are contained in References 3.9-6 and 3.9-7. The Global Nuclear Fuel reload fuel design bases are contained in Reference 3.9-22.

3.9.2.2.2.6 Recirculation Pump and Motor Assembly. Calculations were made to ensure that the recirculation pump and motor assembly is designed to withstand the specific static equivalent seismic and hydrodynamic loads. The flooded assembly was analyzed from the brackets on the motor-mounting member with snubbers attached to brackets located on the pump case and the top of the motor frame.

Primary stresses due to horizontal and vertical seismic and hydrodynamic forces were considered to act simultaneously and are conservatively added directly. Horizontal and vertical seismic and hydrodynamic forces were applied to mass centers and equilibrium reactions determined for motor and pump brackets.

3.9.2.2.2.7 Emergency Core Cooling System Pumps and Motors Assembly. A three-dimensional finite element model of each ECCS pump/motor assembly and its supports was developed and dynamically analyzed using the response spectrum method to verify that the pump/motor assemblies could withstand seismic and hydrodynamic loadings. The same model was statically analyzed to evaluate the effect of the external piping loads and dead weight to ensure that nozzle load criteria and stress limits were met. Critical location stresses were evaluated and compared with the allowable stress criteria. The results of the analysis demonstrated that the stresses at all investigated locations were less than their corresponding allowable values.

3.9.2.2.2.8 Reactor Core Isolation Cooling Pump Assembly. The RCIC pump assembly is safety-related mechanical equipment and dynamic qualification for an active safety function has been provided.

3.9.2.2.2.9 Reactor Core Isolation Cooling Turbine Assembly. The RCIC turbine assembly is safety related. The turbine and subcomponents have been dynamically qualified by testing and analysis.

3.9.2.2.2.10 Standby Liquid Control Pump and Motor Assembly. The SLC pump and motor are not considered safety related (see Section 7.4). The equipment has been analyzed for seismic loads.

3.9.2.2.2.11 Residual Heat Removal Heat Exchangers. A three-dimensional finite element model of the RHR heat exchanger and its support was developed and analyzed using the

response spectrum method to verify that the heat exchanger can withstand seismic and hydrodynamic loads. The same model was statically analyzed to evaluate the effect of the external piping loads and dead weight to ensure that the nozzle load criteria and stress limits were met. Critical location stresses were evaluated and found to be lower than the corresponding allowable values.

3.9.2.2.2.12 Standby Liquid Control Tank. The SLC tank is not considered safety related (see Section 7.4). The tank has been analyzed for seismic loads.

3.9.2.2.2.13 Main Steam Isolation Valves. The MSIV structures were analyzed and representative models statically tested to demonstrate operability at the specified faulted conditions. Static testing consisted of mechanically loading the extended mass of the valve actuator to equivalent seismic loading while valve closure was performed. Operation of the valve was demonstrated by this test.

3.9.2.2.2.14 Main Steam Safety/Relief Valves. Due to the complexity of this structure and the performance requirements of the valve, the total assembly of the SRV (including electrical, pneumatic devices) was dynamically tested at dynamic accelerations equal to or greater than the combined SSE and hydrodynamic loading determined for this plant. Satisfactory operation of the valves was demonstrated during and after the test.

3.9.2.2.2.15 Fuel Pool Cooling and Cleanup System. The cooling portion of the fuel pool cooling and cleanup (FPC) system is Seismic Category I (see Section 9.1.3). In addition, an interconnection with the standby service water (SW) system will ensure the availability of safety-grade makeup and cooling water in the event that the normal source of makeup and cooling water from the RCC system is unavailable.

The cleanup portion of the system is of Seismic Category II design. It will be isolated from the cooling portion of the system by Seismic Category I valves in the event of its unavailability (see Section 9.1.3.1).

3.9.2.2.3 Balance-of-Plant Safety-Related Mechanical Equipment

Balance-of-plant Seismic Category I equipment, components, and accessories were designed based on results determined analytically (see Section 3.9.2.2) or through dynamic testing. The dynamic program is performed to confirm the ability of the equipment to function as needed during and after an earthquake of magnitude up to and including the SSE. These test programs implement the criteria stated in Sections 3.9.2.2.1 through 3.9.2.2.1.4. The dynamic tests met the seismic loading requirements as defined by the applicable floor response spectrum curves for the appropriate damping coefficients.

3.9.2.2.4 Suppression Pool Hydrodynamic Loads Qualification of Safety-Related Equipment

Suppression pool hydrodynamic loads due to postulated intermediate break accident (IBA), design basis accident (DBA), SRV, SRV(X), SRV(ADS), and SRV(ALL) events were developed for Columbia Generating Station, and are discussed in Appendix 3A. The SRV building responses are appropriately combined with OBE, SSE, IBA, and DBA building responses to provide the basis for evaluating acceptability of Class 1E electrical and safety-related mechanical equipment originally qualified to seismic only dynamic loading. Detailed reevaluation of each component of Seismic Category I equipment was not performed wherever direct comparison of original qualification RRS with new seismic plus hydrodynamic RRS demonstrates satisfactory qualification of the equipment. When such comparisons could not be made, other means of evaluating the original qualification against the new dynamic load combinations were used. In regard to the load combination SRV (1) + SSE + DBA, plant design adequacy assessments for Class 1E electrical and safety-related mechanical equipment were performed using the generic basis established by the BWR Mark II Owner's Group for this load combination.

Each of these analyses complied with the intent of Regulatory Guide 1.92.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The major reactor internal components within the vessel were subjected to extensive testing coupled with dynamic system analyses to properly describe the resulting flow-induced vibration phenomena incurred from normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analyses 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 design.

This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

a. Dynamic analysis of major components and subassemblies is performed to identify natural vibration modes and frequencies. The analytical models used for Seismic Category I structures are similar to those outlined in Section 3.7.2;

b. 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;

c. 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;

d. 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; and

e. 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 (a) above.

The dynamic modal analyses also form the basis for interpretation of the prototype plant preoperational and initial startup test results (see Section 3.9.2.4). 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.

3.9.2.4 Confirmatory Flow-Induced Vibration Testing of Reactor Internals

Reactor internals for CGS are substantially the same as the internals design configurations which have been tested in prototype BWR/4 plants. The only exception is the jet pumps which are of the BWR/5 design. A vibration measurement and inspection program was conducted in the Tokai-2 plant to verify the design of the jet pumps with respect to vibration.

A comprehensive vibration assessment of BWR/4 and BWR/5 internals is presented in a licensing topical report (Reference 3.9-9). This report also contains additional information on the jet pump vibration measurement and inspection programs performed in the Tokai-2 plant.

The CGS reactor internals were tested in accordance with provisions of Regulatory Guide 1.20, Revision 2, for nonprototype Category IV plants using Tokai-2 as the limited valid prototype. The test procedure involves taking vibration measurements to determine the vibration characteristics of reactor vessel internals during the initial approach to full power operation.

Vibratory responses are recorded at various power levels and recirculation flow rates using accelerometers on the shroud head assembly and strain gauges on two selected jet pump riser pipe braces.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

To ensure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted on by the applied forces. These periods are determined from a 12-node vertical dynamic model of the RPV and internals. Only motion in the vertical direction is considered here; hence, each structural member (between two mass points) can only have an axial load. Besides the real masses of the RPV and core support structures, account is made for the water inside the RPV.

The time varying pressures were 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 3.7.2.1. The pressure dynamic loads were combined with other dynamic loads (including seismic and hydrodynamic) by the SRSS method. The resultant force was then combined with other steady-state and static loads on an absolute sum basis to determine the design load.

The results of the dynamic analysis of the reactor internals are summarized in Table 3.9-2b.

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

Prior to initiation of the instrumented vibration test program for the prototype plant (Tokai-2), extensive dynamic analyses of the reactor and internals were performed. The results of these analyses were used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test are always 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 insight into the dynamic behavior to the reactor internals. The additional knowledge gained is utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are the same as those used for the vibration analysis of the prototype plant (Tokai-2).

The flow vibration test data are supplemented by data from forced oscillation tests of reactor internal components to provide the analysts with additional information concerning the dynamic behavior of the reactor internals.

3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and hydrodynamic events for the design of safety-related ASME Code components (except containment components which are discussed in Section 3.8).

When two or more dynamic loads (seismic, LOCA, SRV discharge, etc.) are included in the load combination, responses to individual dynamic loads may be combined by the SRSS method. The basis for this is provided in References 3.9-10 through 3.9-12.

This section also lists the major ASME Class 1, 2, and 3 equipment and associated pressure retaining parts on a component-by-component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Design transients for ASME Class 1 equipment are discussed in Section 3.9.1.1. Dynamic related loads are discussed in Sections 3.7 and 3.9.2.2.

Table 3.9-2 is the major part of this section. It presents the loading combination, analytical methods (by reference or example) and the calculated stress, or other design values, for the most critical areas in the ASME design of each component applicable to all ASME Code Class 1, 2, and 3 components, component supports, and core support structures.

3.9.3.1.1 Plant Conditions

Events that the plant might credibly experience during the plant life 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. The current ASME code designations for these design conditions are given in parentheses.

3.9.3.1.1.1 Normal Condition (Level A). 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.

3.9.3.1.1.2 Upset Condition (Level B). 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 that 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 or an OBE. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.1.3 Emergency Condition (Level C). Those deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the reactor coolant pressure boundary (RCPB). The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a 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; and vibration motions of an OBE in combination with associated system transients.

3.9.3.1.1.4 Faulted Condition (Level D). 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: a control rod drop accident, a fuel handling accident, a main steam line break, a recirculation loop break, the combination of small break accident/large break accident dynamic motion associated with an SSE and hydrodynamic loads plus a loss of offsite power or the SSE.

3.9.3.1.1.5 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 Encountered 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}

3.9.3.1.1.6 Safety Class Functional Criteria. For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could deteriorate 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 shall be 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.

3.9.3.1.1.7 Compliance with Regulatory Guide 1.48. 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 Seismic Category I fluid system components (i.e., water and steam containing components). This guide is applicable to the ASME Code Section III, Class 1, 2, and 3 components, such as vessels, piping, pumps, and valves designed to Seismic Category I conditions with particular emphasis on active pumps and valves to ensure operability. A comparison between the design limits and load combinations required by Regulatory Guide 1.48 and those utilized by CGS for vessels, piping, and active and nonactive pumps and valves shows that compliance with the regulatory guide is satisfied for both NSSS and BOP components.

The state of compliance with the requirements of Regulatory Guide 1.48 for ASME Code Section III, Class 1, components is summarized as follows:

a. Piping design acceptance is based on satisfying the load combinations and stress intensity limits for normal and upset, emergency, and faulted conditions of the ASME Code and is thus in accordance with the requirements of Regulatory Guide 1.48. The loadings include various combinations of pressure, temperature, seismic, and hydrodynamic plant unique inertial and displacement loads developed for CGS. The hydrodynamic events include chugging, pool swell, annulus pressurization, and SRV actuations. For each plant condition the worst-case loading combination is applied and the piping evaluated for satisfaction of the applicable ASME Code allowables. In addition, essential piping is evaluated for functional capability per the rules of NUREG/CR-0261 or for NSSS piping (GE scope of supply) NEDO-21985;

b. Similar to piping, ASME Code Class 1 vessels and nonactive pumps and valves are designed and accepted per the applicable stress allowables of Section III of the ASME Code. The load combinations include pressure, temperature, seismic, and hydrodynamic events. The Regulatory Guide 1.48 cited ASME Code requirements are satisfied for all loading conditions;

c. Compliance with Regulatory Guide 1.48 for active Class 1 pumps is not required since no such pumps are included in the design of CGS;

d. The primary basis for demonstration of operability and satisfaction of Regulatory Guide 1.48 for active Class 1 valves at CGS is provided by means of analysis and/or test. Dynamic analyses of the piping system under building seismic and hydrodynamic loads are performed to obtain the enveloped (or maximum) valve acceleration response. Flexibility of the valve and its operator is also examined to establish the absolute maximum response. These acceleration responses are then compared to safe acceleration limits established by vibratory testing of the subject valve or a representative class of valves. In those cases where operability of valves is demonstrated by analysis, deflections, or deformation of critical valve components under the maximum calculated accelerations (forces) are examined and shown not to impair valve function. Where test data and/or analyses fail to conclusively show valve operability, in situ testing of the valve is completed. These tests include frequency response measurements and/or operational proof tests of the valve under static loading of the valve operator such that maximum dynamic loads are simulated. All CGS pressure retaining safety-related valves are designed for normal as well as plant accident condition loads using either the standard or the alternate design rules of ASME Code Section III;

CGS MSIVs and SRVs have undergone extensive analysis and testing to demonstrate operability and Regulatory Guide 1.48 compliance. The MSIVs are modeled into the piping analysis where thermal, pressure, seismic, steam hammer, and hydrodynamic loads are imposed on the valves in the assessment of ASME Code compliance. In addition, the MSIVs operability following a downstream line break was demonstrated by the “Static Line Test” as defined in the GE report APED-5750 (March 1969). Later tests involved application of hydrodynamic loads in the valves’ operability qualification.

The SRV valves are qualified by test for operability during a combined seismic and hydrodynamic load event. Structural integrity of the SRVs during a dynamic event is demonstrated by both ASME Code compliance and test. A dynamic qualification test (shake table) which applied moment and shear loads greater than the required design limit loads was completed to demonstrate SRV operability.

The state of compliance with the requirements of Regulatory Guide 1.48 for ASME Code Section III, Class 2 and 3 piping, vessels, pumps, and valves is demonstrated in a program analogous to the Class 1 component evaluations. The applied stress limits comply with the ASME Code Class 2 and 3 allowables and meet the requirements of Regulatory Guide 1.48 for

all plant conditions. Piping functional capability assessment of essential systems are completed according to the rules for Class 2 piping defined by NUREG/CR-0261.

All active ECCS pumps are qualified for operability by first being subject to rigid tests before and after installation in the plant. These tests include hydrostatic pressure, seal leakage, flow, thermal response, and vibration monitoring of these pumps under full load. Similar ECCS pump motors were dynamically tested operating at full load under the combined seismic and abnormal environmental conditions existing during and after a LOCA. Detailed analyses of the pump structure under piping reaction and building inertial (seismic/hydrodynamic) loads have been completed to further assess operability and ASME Code compliance.

Engineered safety feature (ESF) pumps (other than ECCS) are designed to the appropriate section of the ASME Code using conservatively derived loads for both normal and accident condition events. The ESF pump designs for operability are based on analytical results or appropriate dynamic testing to meet the seismic loading requirements as defined by the applicable floor response spectrum.

Except for stress limits and application of hydrodynamic loads (which may not be applicable outside the first piping anchor external to primary containment), Regulatory Guide 1.48 compliance for active and nonactive Class 2 and 3 valves was completed using the programmatic procedures instituted for Class 1 valves.

The design limits for BOP Class 1 components and for all Class 2 and 3 components are based on the stress criteria outlined in the ASME Code. The ASME criteria established allowable stresses for these components under all design load combinations. Table 3.9-2 defines the loading combinations and stress limits for NSSS and BOP components. Tables 3.9-3 and 3.9-4 show the load and stress criteria for ASME Code Class 1, 2, and 3 components.

3.9.3.1.2 Reactor Pressure Vessel Assembly

The reactor vessel assembly consists of the RPV, vessel support skirt, and shroud support. The shroud support consists of the shroud support plate, the shroud support cylinder, and its legs. The RPV is an ASME Class 1 component constructed to the requirements of the Summer 1971 Section III Code. The Summer 1971 Code did not include requirements for supports or for vessel internals. A complete stress report on the RPV has been prepared in accordance with ASME requirements. Table 3.9-2a provides a summary of the stress criteria, load combinations, calculated stresses, and allowable stresses, including the effects of hydrodynamic loads. The stress analysis performed for the reactor vessel assembly (including the faulted condition) was completed using elastic methods. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are discussed in Section 3.9.5.

3.9.3.1.3 Main Steam Piping

The main steam piping discussed in this paragraph includes that piping extending from the RPV to the outboard MSIV. This piping is designed in accordance with the ASME Code Section III, Subarticle NB-3600. The loading conditions, stress criteria, calculated stresses, and allowable stresses are summarized in Table 3.9-2d.

The rules contained in Appendix F of ASME Code Section III were 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 paragraph F-1360.

3.9.3.1.4 Recirculation Loop Piping

This section discusses the recirculation system piping which is bounded by the RPV nozzles. This piping is designed in accordance with the ASME Code Section III, Subarticle NB-3600. The loading conditions, stress criteria, calculated stresses, and allowable stresses are summarized in Table 3.9-2e.

The rules contained in Appendix F of ASME Code Section III were 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 paragraph F-1360.

3.9.3.1.5 Recirculation System Valves

The recirculation system flow control (kept in a mechanically blocked full open position) and suction and discharge gate valves are designed in accordance with the ASME Code Section III, Class 1, Subarticle NB-3500. The discharge gate valve is required to close for LPCI flow injection. Loading combinations and other stress analysis information are presented in Tables 3.9-2f and 3.9-2j.

3.9.3.1.6 Recirculation Pump

All pump parts that are subject to reactor system water pressure comply with ASME Code Section III, 1971 Edition for Class 1 components.

In the design of the recirculation pumps, the ASME Code Section VIII, Division 1, was used as a guide in calculations made for determining the thickness of pressure retaining parts and in sizing the pressure retaining bolting.

The pump vendor made calculations for the design of the pressure containing components to include the determination of minimum wall thickness, allowable stress, and pressures. The loading conditions and other stress analysis information are presented in Table 3.9-2i.

Load, shear, and moment diagrams were constructed to scale using live loads, dead loads, and calculated snubber reactions. Combined bending, tension, and shear stresses were determined for each major component of the assembly, including the pump driver mount, motor flange bolting, and pump case.

The maximum combined tensile stress in the cover bolting was calculated using tensile stress from design pressure.

Combined primary stresses did not exceed the code allowable stress shown in ASME Code Section VIII, 1971 Edition. For the faulted condition (SSE), the maximum stress was limited to 1.5 S_m. This limit is conservative since 1.5 $S_m \cong 0.5\ S_u$ (S_u is the ultimate strength). The code allows 0.7 S_u for the faulted condition. These methods and calculations demonstrate that the pump will maintain pressure integrity at all times.

3.9.3.1.7 Standby Liquid Control Tank

The SLC tank is designed in accordance with ASME Code Section III. A summary of the design calculations and stress criteria used is shown in Table 3.9-2m.

3.9.3.1.8 Residual Heat Removal Heat Exchangers

The RHR heat exchanger is designed in accordance with the ASME Code Section III. The loading combinations considered and stress analysis for the RHR heat exchangers are presented in Table 3.9-2o.

3.9.3.1.9 Reactor Core Isolation Cooling Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine has been designed and fabricated following the basic guidelines of ASME Code Section III for Class 2 components.

3.9.3.1.10 Reactor Core Isolation Cooling Pump

The RCIC pump has been designed and fabricated to the requirements of the 1971 Edition, Winter 1971 Addenda of the ASME Code Section III as a Class 2 component.

Table 3.9-2r contains a summary of the RCIC pump loading conditions, stress criteria, calculated stresses, and the allowable stresses.

3.9.3.1.11 Emergency Core Cooling System Pumps

The RHR, low-pressure core spray (LPCS), and high-pressure core spray (HPCS) pumps are designed in accordance with ASME Code Section III. The stress analysis methods, calculated stresses, and allowable limits for the ECCS pumps are provided in Table 3.9-2n.

3.9.3.1.12 Standby Liquid Control Pump

The SLC pump has been designed and fabricated following the requirements for an ASME Code Section III, Class 2 component. Table 3.9-2L contains a summary of the SLC pump loading conditions, stress criteria, calculated stresses, and the allowable stress limits.

3.9.3.1.13 Safety/Relief Valves and Main Steam Isolation Valves

The SRVs and MSIVs are designed in accordance with the requirements of ASME Code Section III, Subarticle NB-3500, Class 1 components.

Loading combinations, analytical methods, calculated stresses, and allowable limits are shown for the SRVs and MSIVs in Tables 3.9-2g and 3.9-2h, respectively.

3.9.3.1.14 Safety/Relief Valve Discharge Piping

3.9.3.1.14.1 Main Steam Safety/Relief Valve Piping. This piping is designed in accordance with the ASME Code Section III, Subsection ND for Class 3 piping within the drywell and Subsection NC for Class 2 piping within the suppression chamber. The load combinations and allowables are shown in Table 3.9-4. The main steam SRVs relieve to closed discharge systems; therefore, Regulatory Guide 1.57 is not applicable.

3.9.3.1.14.2 Residual Heat Removal Suction Shutdown Thermal Relief Valve Piping. The discharge piping for the thermal relief valve on the RHR system relieves into the containment suppression pool. It is designed in accordance with ASME Code Section III for Class 2 piping and is Seismic Category I supported. However, due to the very small discharged quantities of fluid required to relieve pressure the intent of Regulatory Guide 1.67 is not considered applicable. See Section 1.8.2.

3.9.3.1.15 Reactor Water Cleanup System Pump and Heat Exchangers

The RWCU pump and regenerative and nonregenerative heat exchangers are not part of a safety system and are not designed to Seismic Category I requirements.

The requirements of ASME Code Section III, Class 3, components were used in evaluating the RWCU system pump and heat exchanger components. The loading conditions, stress criteria,

calculated stress, and allowable stresses for the RWCU pumps are summarized in Table 3.9-2p.

3.9.3.1.16 Fuel Pool Cooling and Cleanup System Heat Exchangers and Pumps

The cooling portion of the FPC system has been analyzed to Seismic Category I requirements.

The FPC heat exchanger design has been analyzed by performing a response spectrum dynamic analysis for seismic loads. The model used to represent the heat exchangers utilized finite element and ANSYS programs to perform the detailed calculations. For certain subcomponents such as the tube bundle, hand calculations were performed to determine the natural frequency. The corresponding acceleration coefficients were then used to calculate the stress in these subcomponents.

The piping in the cooling portion of the system has been analyzed to Seismic Category I requirements utilizing the ADLPIPE computer program. Existing manually operated valves have been upgraded by the manufacturer to comply with Seismic Category I requirements. Additional and replacement valves have been purchased to Seismic Category I, Quality Class 1 requirements.

The cleanup portion of the system is not safety related and is designed to Seismic Category II requirements as defined in Section 3.2.1. Seismic Category I isolation valves automatically isolate the portion of the system on low fuel pool water level. Following a seismic occurrence it can be manually isolated if necessary.

3.9.3.1.17 Control Rod Drive Piping

The safety-related portion of the CRD piping is designed in accordance with the ASME Code Section III, Subsection NB for Class 1 piping and NC for Class 2 piping. The criteria and load combinations are shown in Table 3.9-3 for Class 1 piping and Table 3.9-4 for Class 2 piping.

The remainder of the CRD piping, which is not safety related, is designed in accordance with ANSI B31.1.

3.9.3.1.18 Balance-of-Plant Piping

Safety associated piping is classified as Seismic Category I. The code class of such piping is ASME Code Class 1, 2, or 3.

3.9.3.1.18.1 Criteria and Results. For ASME Code Class 1, the loading combinations, appropriate design criteria, and applicable allowable stresses are presented in Tables 3.9-2 and 3.9-3, respectively. Procedures used to evaluate stresses and stress indices in ASME Class 1

piping at integral attachments shall reference the applicable ASME Code Case. Calculated stresses and fatigue usage factors for these systems are documented in the applicable stress reports. These values are within the allowable limits shown in Table 3.9-3. The design of 1 in. and under ASME Class 1 piping is performed to Subsection NC rules in accordance with NB-3630.

For ASME Code Classes 2 and 3, the loading combinations, appropriate criteria, and applicable allowable stresses are presented in Tables 3.9-2 and 3.9-4. Actual calculated stresses for these systems are within the allowables given in Table 3.9-4 and are documented in piping stress calculations.

3.9.3.1.18.2 Method of Analysis. The pipe stress analyses and fatigue analyses are performed using computer programs listed in Section 3.12. Stresses due to seismic loading are evaluated by use of multimass, multispring analytical models, in conjunction with seismic shock response spectra. Other dynamic effects when applicable are also considered in the calculation by performance of dynamic analysis, using the appropriate shock response spectra or time-history loading.

3.9.3.1.18.3 Seismic Loading. For Seismic Category I piping, the procedure of combining the effect of the three components of earthquake motion is discussed in Section 3.7.2.6.

Internal moments and forces derived from the seismic responses of the piping system are combined with loads from deadweight, pressure, thermal, and other mechanical loads to complete the stress analysis of all Seismic Category I and some Seismic Category II piping. For ASME Class 1 piping, stress indices and cumulative usage factors of the piping system are computed based on the formulation specified in ASME Code Section III, Subarticle NB-3600; and for ASME Code Class 2 and Class 3 piping, the formulations in Subarticles NC-3600 and ND-3600 are used.

In the simplified dynamic analysis described in Section 3.7.2.1.8.2 for BOP-supplied Seismic Category I piping, a constant load factor is used as the vertical and horizontal amplified floor response loadings.

ASME Code Class 2 and Class 3 piping systems specified as Seismic Category I, and 2-in. nominal diameter or smaller (i.e., equipment drain and instrument lines) are generally subject to simplified dynamic analyses as described above.

Where it is not feasible or practical to isolate the Seismic Category I piping system from the nonseismic Category I piping system, the adjacent nonseismic piping is then seismically designed according to the same criteria applicable to the Seismic Category I piping system. The attached nonseismic piping is also designed in such a manner that during an SSE, it does not cause a failure of the Seismic Category I piping.

3.9.3.1.18.4 Other Dynamic Loadings. Dynamic loadings resulting from sudden closure of an isolation valve or a turbine throttle valve on the piping system (for example, transient loading on steam line due to turbine trip) are included as occasional mechanical loads in piping analysis. Shock suppresser constraints are used as required to control excessive displacements or moments due to these transient loadings.

3.9.3.1.18.5 Analytical Models for Piping Systems. Piping systems are designed and analyzed as complete systems from anchor to anchor. The relative rigidity of major equipment terminal points is generally considered sufficient to effectively decouple the piping at that location. All major piping branches and all inline equipment (such as inline valves, etc.) are included in the analytical model used to determine relative flexibility and resulting stresses and deflections. Relatively flexible branches, having significantly smaller pipe diameter than the remainder of the piping system, are decoupled from the main run and are analyzed separately. For further discussion on modeling procedures see Section 3.7.2.3.

3.9.3.2 Pump and Valve Operability Assurance

For all active ASME Class 1, 2, and 3 pumps and valves see the Inservice Testing Program Plan.

These valves are designed according to ASME Code Section III rules to ensure that the calculated primary stresses are within the elastic range for all code-specified loading conditions. See Section 3.9.3.2.4 for additional information relative to active valve operability assurance.

The inactive valves and pumps within the RCPB are ASME Class 1. All inactive valves and pumps within the RCPB meet the stress and pressure limits of ASME Code Section III, NB-3500.

Active mechanical equipment classified as Seismic Category I is designed to perform its function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include active pumps and valves in fluid systems such as the ECCS. Active equipment must perform a mechanical motion during the course of accomplishing a safety function.

Operability is ensured by satisfying the requirements of the following programs. Safety-related valves are qualified by testing and analysis and by satisfying stress and information criteria at all critical locations. The content of these programs is described in the following sections.

3.9.3.2.1 Emergency Core Cooling System Pumps

All active pumps are qualified for operability by first being subject to rigid tests before and after installation in the plant. The in-shop tests include (a) hydrostatic tests of pressure-retaining parts as required by the applicable Edition and Addenda of the ASME Code, (b) seal leakage tests, and (c) performance tests, while the pump is operated with flow to determine total developed head, minimum and maximum head, NPSH requirements, and other pump and/or motor parameters. 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 inservice inspection and operation. 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 analyzed for operability during faulted conditions by ensuring that (a) the pump will not be damaged during the seismic and hydrodynamic event, and (b) the pump will continue operating despite the faulted loads.

3.9.3.2.1.1 Analysis of Loading, Stress, and Acceleration Conditions. To avoid damage during the faulted plant condition the stresses caused by the combination of normal operating loads, SSE, and dynamic loads are limited to the material elastic limit, as indicated in Section 3.9.3.1 and Table 3.9-2. A three-dimensional finite element model of the pump-motor and its supports was developed and dynamically analyzed using the response spectrum analysis method. The same model was analyzed for static nozzle loads, pump thrust loads, and dead weight. Critical location stresses were evaluated and compared with the allowable stress criteria. Critical location deflections and accelerations were evaluated to ensure operability. The average membrane stress (P_m) for the faulted condition loads is maintained at 1.2S, or approximately 0.75 S_y (S_y = yield stress) and the maximum stress in local fibers is limited to 1.8S, or approximately 1.1 S_y. The maximum allowable nozzle loads are also considered in the analysis of the pump supports to ensure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9-2 as allowables ensures critical parts of the pump are not damaged during the faulted condition and, therefore, the reliability of the pump for postfaulted condition operation is not impaired by the seismic and hydrodynamic events.

A dynamic analysis was made to determine the seismic load from the applicable floor response spectra. Analysis was made to check that faulted condition nozzle loads and dynamic accelerations did not impair the operability of the pumps during or following the faulted condition. Components of the pump, when having a natural frequency above 33 Hz, are considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas.

3.9.3.2.1.2 Pump Operation During and Following Faulted Condition Loading. Active pump/motor rotor combinations are desired to rotate at a constant speed under all considerations. 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 seismic and hydrodynamic event, will prevent the rotor from becoming seized. In actuality, the dynamic loadings will 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 will not shut down during the faulted event and will operate at the design speed despite the faulted loads.

The functional ability of the active pumps after a faulted condition is ensured 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 and hydrodynamic loads on the equipment itself. The faulted event is infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the postfaulted condition operating loads will be 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 postfaulted conditions are limited by the magnitudes of the normal condition allowable nozzle loads. The postfaulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.2 Standby Liquid Control Pump and Motor Assembly and Reactor Core Isolation Cooling Pump Assembly

These equipment assemblies are compact, rigid assemblies. Each equipment assembly has been dynamically qualified by way of static analysis only. This static qualification verifies operability under seismic conditions and ensures structural loading stresses within Code limitations.

3.9.3.2.3 Emergency Core Cooling System Pump Motors

Qualification of the Class 1E motors used for the ECCS pumps is in compliance with NUREG-0588 Cat II. The qualification of CGS motors is based on a type test of a similar motor. All manufacturing, inspection, and routine tests by motor manufacturer on production units were performed on the test motor.

The type test has been performed on a 1250-hp vertical motor in accordance with IEEE 323-1971, first simulating normal operation during the design life, then the motor being subject to a number of seismic events, and then to the abnormal environment condition possible during and after a LOCA. The test plan for the type test was as follows:

a. Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on IEEE 275-1966 for the insulation type used on the ECCS motors. The amount of aging equaled the total estimated operation days at maximum insulation surface temperature;

b. Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma irradiation during normal and abnormal conditions;

c. The normal induced vibration effect on the insulation system has been simulated by horizontal vibration of 1.5g at 60 Hz for 1 hr duration;

d. Motor bearings are selected based on bearing manufacturer's test and operating data using the loads calculated to act on the bearings. Operating life is determined by condition monitoring;

e. The dynamic load deflection analysis on the rotor shaft, performed to ensure adequate rotation clearance, was verified by static loading and deflection of the rotor for the type test motor;

f. Dynamic load aging and testing was performed on a biaxial test table in accordance with IEEE 344-1975. During this type test, the shake table was activated simulating the maximum design limit of the SSE with motor starts and operation combinations as may possibly occur during a plant life; and

g. An environmental test simulating a LOCA condition with 100 days duration time has been performed with the test motor fully loaded, simulating pump operation. The test consisted of startup and 6 hr operation at 212°F ambient temperature and 100% quality steam. Another startup and operation of the test motor after 1-hr 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 for the insulation type used on the ECCS motors.

3.9.3.2.4 ASME Code Class 1 Active Valves

The Class 1 active valves are the MSIVs, SRVs, and SLC valves. Each of these valves is designed to perform its mechanical motion in conjunction with a DBA including hydrodynamic loads. Qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.4.1 <u>Main Steam Isolation Valve</u>. The MSIVs are evaluated for operability during seismic and hydrodynamic loads events by both analysis and test.

a. First, the valve body is designed in accordance with the ASME Code Section III, Class 1 which limits deformation in the operating area of the valve body to be within the elastic limit of the material by limiting pressure and pipe reaction input loads (including seismic), thereby ensuring no interference with valve operability. (See Table 3.9-2h);

b. A dynamic (including hydrodynamic loads) test was conducted on the MSIV to ensure operability at design seismic and hydrodynamic loading requirements. A sine sweep was used to determine resonance frequencies of the actuator assembly. A sine beat was used to excite the actuator assembly at all frequencies up to 33 Hz with special emphasis at the resonance frequency. Operability was demonstrated at each test point. No significant change in valve closing rate resulted from the test. It was also demonstrated that the valve configuration had sufficient integrity to withstand, without compromise of structure or electrical function, the required simulated seismic and hydrodynamic loads event.

To ensure design limits are not exceeded for both piping input loads and actuator dynamic loads the MSIV is mathematically modeled in the main steam line system analysis. 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 steam line analysis. Pipe anchors and restraints are applied as required to limit pipe system resonance frequencies and amplified acceleration to within acceptable limits for the MSIVs; and

c. The MSIVs operability following a downstream line break was demonstrated by the "Static Line Test" as defined in the report APED-5750 (March 1969). The test specimen was a 20-in. valve of a design representative of the MSIVs.

Environmental qualification of sensitive electrical/pneumatic equipment to meet performance requirements defined in Tables 3.11-1 and 3.11-2 have been successfully completed by product design and test evaluation methods.

3.9.3.2.4.2 Main Steam Safety/Relief Valves. The SRVs are qualified by test for operability during seismic and hydrodynamic load events. Structural integrity of the configuration during a dynamic event is demonstrated by both code analysis and test.

a. Valve is designed for maximum moments which may be imposed when installed in service. These moments are resultants due to dead weight plus seismic and hydrodynamic loadings on both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge;

b. The operability of a production SRV was demonstrated by a dynamic qualification (shake table) test which applied moment and ‘g’ loads greater than the required design limit loads and conditions.

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; and

c. The SRV is generically qualified via testing for seismic and hydrodynamic loading. The input shock spectrum contained waveforms with frequencies up to 100 Hz.

The sensitive electrical/pneumatic equipment is qualified to performance requirements during and after emergency environmental conditions defined in Section 3.11.

3.9.3.2.4.3 Standby Liquid Control Valve (Explosive Valve). The SLC explosive valve has been generically qualified to IEEE 344-1975. The generic qualification test demonstrated the absence of natural frequencies below 33 Hz and the ability to remain operable after the application of horizontal seismic loading equivalent to 6.5g and a vertical seismic loading equivalent to 4.5g at 33 Hz.

3.9.3.2.4.4 High-Pressure Core Spray Valve. This valve is a motor-operated gate valve. The body design analysis and testing is in accordance with the ASME Code Section III, Class 1 requirements.

3.9.3.2.5 Class 2 and 3 Active Valves

General Electric has six HPCS valves which are “Class 2 Active” in their scope of supply. There are no “Class 3 Active” valves in GE’s scope of supply. All gate/globe valves are motor operated and check valves are air operated.

The gate/globe valves are generically qualified by testing valves that are typical of the valves supplied by GE. Operability is ensured by testing under both the static design basis load and at the maximum capability static load. The tests ensure operability during and after the design basis load. The actuators are qualified to IEEE 382-1972, to levels that exceed the design loadings.

3.9.3.2.6 Engineered Safety Features Pumps

The ESF pumps are designed to the appropriate section of the ASME Code using conservatively derived loads including the effects of SSE. The ESF pump design is based on analytical results or appropriate dynamic testing to meet the seismic loading requirements as

defined by the applicable floor response spectrum for the appropriate damping coefficients. For the small, compact pumps comprising rigid assemblies with natural frequencies well above 33 Hz, the assemblies may have been qualified by static analytical methods only. The ECCS pumps discussed in Section 3.9.3.2.1 are included in the ESF categorization.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

The design criteria for all safety and relief piping are in accordance with the rules in paragraphs NB-3677 and NC-3677 of ASME Section III, and the rules of Code Case 1569, applicable to the piping component under investigation. For relief systems the design criteria and the analyses used to calculate maximum stresses and stress intensities are in accordance with subarticles NB-3600 and NC-3600 of ASME Code Section III. The maximum stresses are calculated based on the full discharge loads, including the effects of the system dynamic response, and the system design internal pressure. Stresses are determined for all significant points in the piping system including the safety valve inlet pipe nozzle and the nozzle to shell juncture.

Detailed evaluations are performed only for valves which produce transient effects; small relief valves [for example, those relieving temperature induced water (expansion)], where pressure relief is accomplished without transient effects, are not evaluated.

3.9.3.3.1 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 SRV until a steady discharge flow from the RPV 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 SRV cause the SRV discharge piping to vibrate. This in turn produces forces that act on the main steam piping and the discharge piping.

The analysis of the relief valve discharge transient consists of a stepwise time history solution of the fluid flow equation, to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated 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. 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. Figure 3.9-1 shows a pipe section load transient typical of that produced by relief valve discharge.

The methods of analysis applied to determine piping system response to relief valve operation are either time-history integration method or the normal mode superposition technique. The

forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the SRV, the main steam line, and the discharge piping are combined with loads due to other effects as specified in Section 3.9.3.1. The code stress limits, corresponding to load combinations classification as normal, upset, emergency, and faulted are applied to the steam and discharge pipe.

3.9.3.3.2 Open Relief Systems

There are no open discharge pressure relief valves mounted on Class 1, 2, or 3 systems in the NSSS and BOP system.

3.9.3.3.3 Closed Relief System

For relief valves discharging into a closed system, an analytical model of one-dimensional transient flow characteristics following the blowoff of the upstream SRV into the discharging piping system is established. The time-dependent pressure, temperature, density, velocity, and hence the momentum of the downstream pipe flow are then computed from this conservative hydrodynamic/thermodynamic flow model. The phenomena such as flow restrictions, frictional resistance, and flow discontinuities (shock waves) are considered. This model also considers the influence of valve opening time and the effect of loop seal water contained in the upstream valve seat.

The unbalanced transient hydraulic forcing function acting on the piping system computed from the flow model is then used to determine the transient dynamic responses of the piping structural model. Adapting the lumped-parameter method incorporated with the modal analysis of piping system, the time-history modal responses are computed. Computations of maximum stress intensities for ASME Code Class 1 piping, or maximum stress levels for ASME Code Class 2 and 3 piping, are based on the dynamic analysis of the system.

3.9.3.4 Component Supports

Component supports are discussed in Section 5.4.14. The discussion of design loading combinations, design procedures, and acceptability criteria is presented below.

All safety-related component supports in the BOP are designed in accordance with ASME Code Section III, Subsection NF, 1971 Edition, Winter 1973 Addenda. The bases for allowable buckling loads and allowable buckling stresses are the allowable load equations and stress equations from ASME Code Section III, 1974 Edition, as referenced below:

a. Subsection NA, “General Requirements,” Appendix XVII; “Design of Linear Type Supports by Analysis,” subarticle XVII-2213; “Stress In Compression,” and subarticle XVII-2215, “Combined Stresses;”

b. Subsection NA, Appendix XVII, subarticle XVII-2220, "Stability and Slenderness and Width Thickness Ratios;" and

c. Subsection NA, Appendix F, "Rules for Evaluation of Faulted Conditions."

Critical buckling loads for plant faulted conditions are determined by methods described in Subsection NA, Appendix F, and allowable loads are limited to two-thirds of critical buckling loads.

3.9.3.4.1 Pipe Supports

Pipe supports for ASME jurisdictional systems are designed in accordance with Subsection NF of ASME Code Section III, 1971 Edition, Winter 1973 Addenda. Component standard supports for ASME piping are designed in accordance with ASME Code Section III, Subsection NF, 1971 Edition, Winter 1973 Addenda, or later code editions. Supports are either designed by load rating or to the stress limits for the applicable code. In general, the load combinations for the various operating conditions correspond to those used to design the supported pipe. The design and evaluation of welded attachments to Class 2 and 3 piping shall be made in accordance with ASME Code Cases N-318, N-224, N-224-1, and N-392. No pipe support fatigue evaluation is necessary since the code stipulated minimum number of cyclic events for high cycle fatigue evaluation are not met (see Subsection NA, Appendix XVII-3000).

The design criteria and dynamic testing requirements for supports are as follows:

a. Standard supports

   Standard pipe support hardware encompasses items such as struts, pipe clamps, U-bolts, saddles, lubrite plates, rods, turnbuckles, eyenuts, and other attachment and bracket fixtures. These items are selected based on certified load ratings, which in each specific application are selected to bound the applied pipe loading. All standard pipe support hardware is designed, fabricated, and installed so that it cannot become disengaged by the movement of the supported pipe or equipment after it has been placed in service.

b. Spring supports

   The design load on spring supports is the load caused by dead weight. The supports are calibrated to ensure that they support the design load at both their hot and cold settings. Spring supports provide a specified downtravel and uptravel in excess of the specified thermal movement. Spring supports are selected such that bottoming will not occur during design bases dynamic loading events.

c. Snubbers

The design load on snubbers includes those loads caused by seismic forces (OBE and SSE) and containment hydrodynamic loading for those snubbers within the containment anchor boundary. System movements and reaction forces caused by relief valve discharge, turbine stop valve closure, and similar transients may also be included when applicable.

The snubbers are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

Two snubbers of each size and each model were tested under upset and faulted loads in the manner described below:

1. Snubbers were tested dynamically to ensure that they could perform as required under upset loading condition in the following manner:

   (a) The snubbers were subjected to a force that varied approximately as the sine wave,

   (b) The frequency (Hz) of the input force was in increments of 5 Hz within the range of 3 Hz to 33 Hz,

   (c) The test was conducted with the snubber at room temperature and at 200°F,

   (d) The peak load in both tension and compression was equal to or higher than the rated load of the snubbers, and

   (e) The duration of the test at each frequency was 10 sec or more.

2. Snubbers were tested dynamically to ensure that they could perform as required under emergency and faulted loading conditions in the following manner:

   (a) The snubbers were subjected to a force that varied approximately as the sine wave,

   (b) The test was conducted with the snubbers at room temperature,

(c) The peak load in both tension and compression was equal to 1.5 times the rated load of the snubbers,

(d) The duration of the test was 10 sec,

On completion of the above abnormal environmental transient test, the snubber was tested dynamically at a frequency within a specified frequency range. The snubber did operate normally during the dynamic test, and

(e) Bolting and welding of pipe support structures (i.e., nonpipe class attachments) meet the requirements of Subsection NF and Subsection NE for primary containment attachments (1971 ASME Code Edition, Winter 1973 Addenda). Bolting and welding of pipe support structures outside of the NF jurisdictional boundary complies with the requirements of the AISC Code, 1970, 7th Edition.

(f) Operability assurance of snubbers

Snubber operability was verified during the plant Power Ascension Test Program in the system expansion test described in Section 14.2.12.3.17.

Supply system design and plant administrative procedures are established which require that applicable documentation (SAR, Technical Specifications, Design Drawings, etc.) is revised to reflect the plant configuration as altered by design changes. These design change control procedures require documented acceptance testing for all additional snubber installations on safety-related systems. One step of this design verification requires evaluation of the impact of the design change on the operation of the facility. Snubbers added to safety-related systems or requiring surveillance testing will be added to the Snubber Program.

The Snubber Program includes a compilation of safety-related snubbers and specifies snubber inspection and testing requirements. Snubber accessibility, maintenance and repair/replacement are addressed in the Snubber Program.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The RPV support skirt is designed as an ASME Code Class 1 plate and shell type component support per the requirements of ASME Code Section III, Subsection NB, 1971 Edition, Summer 1971 Addenda. The loading conditions, stress criteria, calculated stresses, and the allowable stresses in the critical support areas for various plant operating conditions are summarized in Table 3.9-2a.

In design of the reactor vessel support skirt as a plate and shell-type component support, the allowable compressive load was limited to 90% of the load, which produces a stress equivalent to yield stress in the material, divided by the safety factor for the plant condition being evaluated. The safety factor for the faulted condition was 1.125. The effects of fabrication and operational eccentricity were included in stress calculations.

A buckling analysis of the RPV support skirt for faulted conditions shows the support skirt meets ASME Code Section III, paragraph F-1370(c) faulted condition limits of 0.67 times the critical buckling strength of the support at temperature. The design basis faulted condition for this analysis included compressive loads due to the design basis maximum earthquake, overturning moments and shears due to the jet reaction load from a postulated severed pipe, and compressive effects on the support skirt from thermal and pressure expansion of the reactor vessel. The expected maximum loads for the CGS vessel support skirt are less than 50% of the maximum design basis loads used in the buckling analysis; therefore, the reactor vessel support skirt is adequately designed to prevent buckling.

3.9.3.4.3 Nuclear Steam Supply System Floor-Mounted Equipment (Pumps and Heat Exchangers)

The RHR, HPCS, LPCS, RCIC, and SLC pumps and heat exchangers are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the stresses in the critical support areas are within ASME Code allowables. The loading conditions, stress criteria, and allowable stresses in the critical support areas are summarized in Tables 3.9-2L, 3.9-2n, 3.9-2o, and 3.9-2r.

3.9.3.4.4 Supports for ASME Code Class 1, 2, and 3 Active Components

The ASME Code Class 1, 2, and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Seismic Category I active pump supports are qualified for seismic and hydrodynamic loads by testing when the pump supports along with the pumps are fulfilling the following conditions:

a. Simulate actual mounting conditions;

b. Simulate all static and dynamic loadings on the pump;

c. Monitor pump operability during testing;

d. The normal operation of the pump during and after the test indicates that the supports are adequate. Any deflection or deformation of the pump supports which precludes the operability of the pump is not accepted; and

e. Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Seismic and hydrodynamic qualification of component supports by analysis is generally accomplished as follows:

a. Stresses at all support elements and parts such as pumps holddown and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to be within the allowable limits as specified in ASME Subsection NF;

b. For normal and upset plant conditions, the deflections and deformations of the supports are ensured to be within the elastic limits and not exceed the values permitted by the design based on design verification tests to ensure the operability of the pumps; and

c. For emergency and faulted plant conditions, the deformations must not exceed the values permitted by design to ensure that operability of the pumps.

3.9.3.5 Pipe Support Analysis

The complex structural design of many pipe supports necessitates the use of computer programs to confirm their integrity under various loading conditions (including seismic loading). Computer program use extends from commercially available packages such as ANSYS and STRUDL (Structural Analysis Programs) to those which were developed specifically for CGS. Pipe support design analyses were performed (when required) using computer programs as described in Section 3.12.

3.9.4 CONTROL ROD DRIVE SYSTEM

CGS is equipped with a hydraulic CRD system which includes the CRD mechanism, the HCU, the condensate supply system and the scram discharge volume, and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information Regarding Control Rod Drive Systems

Descriptive information on the CRDs as well as the entire control and drive system is contained in Section 4.6.

3.9.4.2 Applicable Control Rod Drive Systems Design Specification

The CRD system is designed to meet the functional design criteria as outlined in Section 4.6 and consists of the following:

a. Locking piston CRD,
b. Hydraulic control unit,
c. Hydraulic power supply (pumps),
d. Interconnecting piping,
e. Flow and pressure and isolation valves, and
f. Instrumentation and electrical controls.

Those components of the CRD system forming part of the primary pressure boundary are designed according to ASME Code Section III.

The quality group classification of the components of 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 quality groups.

Pertinent aspects of the design and qualification of the CRD system components are discussed in the following locations: transients in Section 3.9.1.1, faulted conditions in Section 3.9.1.4, and seismic testing in Section 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRD system have been evaluated analytically and the design loading conditions plus the effects of hydrodynamic loads where applicable, stress criteria, calculated stresses, and allowable stresses are summarized in Tables 3.9-2u and 3.9-2v. For the noncode components, experimental testing was used to determine the CRD performance under all possible conditions as described in Section 3.9.4.4.

Deformation is not a limiting factor in the analysis based on the results of the numerous tests performed on the drive.

The CRD housing support system functions are described in Section 4.6.

The American Institute of Steel Construction (AISC) Manual of Steel Construction (Reference 3.9-13), was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding the allowable tension and bending stresses used were 90% of yield and the shear stress used was 60% of yield. These design stresses are 1.5 times the AISC allowable stresses (60% and 40% of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with the reactor operating pressure of 1086 psig (at the bottom of the vessel) acting on the area of the separated housing. The weight of the separated housing, CRD, and blade, plus the pressure of 1086 psig acting on the area of the separated housing, gives a force of approximately 32,000 lb. This force is multiplied by a factor of three to calculate the impact force, conservatively assuming that the housing travels through a 1-in. gap before it contacts the supports. The impact force (109,000 lb) is then treated as a static load in design. The CRD housing supports are designed as Seismic Category I equipment. Loading conditions and examples of stress analysis results and limits are shown in Table 3.9-2ac.

3.9.4.4 Control Rod Drive Performance Assurance Program

The CRD system test program consists of the following tests:

a. Development tests,
b. Factory quality control tests,
c. 5-year maintenance life tests,
d. 1.5 x design life tests,
e. Operational tests,
f. Acceptance tests, and
g. Surveillance tests.

All of the above tests except c and d are discussed in Sections 4.6.3 through 4.6.3.1.1.5. Tests c and d are discussed as follows:

<u>“5-Year Maintenance Life” Tests</u>

Four CRDs are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and one-eighth of the cycles specified in Section 3.9.1.1.1.

On completion of the test program, the CRDs must meet or surpass the minimum specified requirements.

1.5 x 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 Section 3.9.1.1.1.

Two CRDs underwent such testing in 1976. On completion of the test program, the CRDs met or surpassed the minimum specified performance requirements.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

This subsection identifies and discusses the structural and functional integrity of the major RPV internals.

3.9.5.1 Design Arrangements

The core support structure and reactor vessel internals (exclusive of fuel, control rods, CRDs, and in-core nuclear instrumentation) are identified below:

a. Core support structures

  1. Shroud,
  2. Shroud support,
  3. Core support plate, holddown bolts, and core plate wedges,
  4. Top guide (including bolts and keepers),
  5. Fuel supports, and
  6. Control rod guide tubes.

b. Reactor internals

  1. Jet pump assemblies and instrumentation,
  2. Feedwater spargers,*
  3. Vessel head spray nozzle,
  4. Differential pressure line,
  5. In-core flux monitor guide tubes,*
  6. Initial startup neutron sources,*
  7. Surveillance sample holders,*
  8. Core spray lines and spargers and SLC injection,
  9. In-core instrument housings (Dry Tubes),
  10. LPCI coupling,
  11. Steam dryer,*

* Non-safety-class components.

12. Shroud head and steam separator assembly,*
13. Guide rods,* and
14. CRD thermal sleeves.

A general assembly drawing of the important reactor components is shown in Figure 5.3-5.

The floodable inner volume of the RPV can be seen in Figure 3.9-2. 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 free to expand.

The LPCI couplings incorporate sleeves to allow free thermal expansion.

3.9.5.1.1 Core Support Structures and Vessel Internals

The core support structures and vessel internals consist of those items listed in Section 3.9.5.1. 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. Figure 3.9-2 shows the reactor vessel internal flow paths.

3.9.5.1.1.1 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 and the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the bottom by the core support. The lower portion, surrounding part of the lower plenum, is welded to the RPV shroud support.

3.9.5.1.1.2 Shroud Head and Steam Separator Assembly. The shroud head and steam separator assembly is bolted to 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.

* Non-safety-class components.

3.9.5.1.2 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, incore 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.

3.9.5.1.3 Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings and fastened to a peripheral rim. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, one or two fuel assemblies. Sockets are provided in the bottom of the beam intersections to anchor the incore flux monitors and startup neutron sources. The rim of the top guide rests on a ledge between the upper and central portions of the shroud. The top guide has alignment pins that engage and bear against slots in the shroud which are used to correctly position the assembly before it is secured. Lateral restraint is provided by wedge blocks between the top guide and the shroud wall.

3.9.5.1.4 Fuel Support

The fuel supports shown in Figure 3.9-3 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 assure 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 assure 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 support. 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 4.2.2).

3.9.5.1.5 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the CRD housings up through holes in the core support 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 CRD 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 CRD housing from below and is rotated to lock the control rod guide tube in place. A key is

inserted into a locking slot in the bottom of the CRD housing to hold the thermal sleeve in position.

3.9.5.1.6 Jet Pump Assemblies

The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the jet pump is covered in detail in References 3.9-14 and 3.9-15. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, slip joint clamp, and diffuser (see Figure 3.9-4). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High pressure water from the recirculation pumps is supplied through a single riser pipe welded to the recirculation inlet nozzle thermal sleeve of each pair of jet pumps. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is restrained laterally by a bracket attached to the riser. Some restrainer brackets may have one or more auxiliary or restrainer wedges installed to replace the function of restrainer bracket set screws found having an excessive throat-to-set screw gap. There is a slip-fit joint between the throat and diffuser. A removable clamp is installed at the slip-fit joint to suppress abnormal flow-induced vibration caused by slip joint leakage. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

3.9.5.1.7 Steam Dryers

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 would occur only under accident conditions, is restricted by steam dryer hold-down brackets attached to the reactor vessel top head.

3.9.5.1.8 Feedwater Spargers

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle and is shaped to

conform to the curve of the vessel wall. Sparger end brackets are 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.

3.9.5.1.9 Core Spray Lines and Standby Liquid Control Injection

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). The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the upper shroud. 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 shroud 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 shroud. 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).

The SLC system also injects to the RPV via the HPCS header. The SLC injection can be accomplished with HPCS flow either on or off.

3.9.5.1.10 Vessel Head Spray Nozzle

When reactor coolant is returned to the reactor vessel part of the flow can be diverted to a spray nozzle in the reactor head. This spray maintains saturated conditions in the reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. The spray also decreases thermal stratification in the reactor vessel coolant. This ensures that the water level in the reactor vessel can rise. The higher water level provides conduction cooling to more of the mass of metal of the reactor vessel and, therefore, help to maintain the cooldown rate.

The vessel head spray nozzle is mounted to a short length of pipe and a flange, which is bolted to a mating flange on the reactor vessel head (see Section 5.4.7).

3.9.5.1.11 Differential Pressure Line

The differential pressure line is used to sense the differential pressure across the core support plate (described in Section 5.4). This line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support plate. It is used to sense the pressure below the core support plate during normal operation. The inner pipe was also designed to reduce thermal shock to the vessel nozzle should the SLC system be actuated. However, the SLC injection piping has been relocated to the HPCS injection line and it no longer uses this nozzle. The outer pipe terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.12 In-Core Flux Monitor Guide Tubes

The in-core flux monitor guide tubes 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) 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 (SRM) and intermediate range monitoring (IRM) detectors are inserted through the guide tubes. A lattice-work 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.

3.9.5.1.13 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (see Section 5.4). 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.

3.9.5.1.14 Low-Pressure Coolant Injection Lines

Three LPCI lines penetrate the core shroud through separate LPCI nozzles. Coolant is discharged inside the core shroud.

The following italicized text is historical information provided in the FSAR to support the application for an operating license. As such, it is not subject to change and therefore has not been verified for accuracy or updated during the FSAR upgrade process per 10 CFR 50.71(e).

3.9.5.1.15 Startup Neutron Sources

The startup neutron sources are held in place by spring pressure between the top of the core support and the bottom of the top guide. Each source consists of two irradiated antimony rods within a single beryllium cylinder. Both the antimony and the beryllium are encased in stainless steel tubes. The design provides for a sufficient source of neutrons present in the core to assure that the core neutron flux is continuously detectable by installed neutron monitors and to assure that significant changes in core reactivity are readily detectable by installed neutron flux instrumentation.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of condition for which the safety design bases must be satisfied reveals five significant faulted events:

a. A recirculation line break between the reactor vessel and the recirculation pump suction,

b. Annulus pressurization load resulting from an asymmetric pressure build up due to a break in the recirculation inlet or feedwater lines in the annulus region between the shield wall and the RPV,

c. A steam line break accident of one main steam line between the reactor vessel and the flow restrictor. This accident results in significant pressure differentials across some of the structures within the reactor,

d. An earthquake which subjects the core support structures and reactor internals to significant forces as a result of ground motion, and

e. A SRV discharge in combination with an SSE.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other ESF reactor internals are less severe than these five postulated events.

The faulted conditions for the RPV internals are discussed in Section 3.9.1.4.2. The analysis and loading combinations for the RPV internals are discussed in Section 3.9.3.1 and Tables 3.9-1 and 3.9-2.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

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. Figure 3.9-5 shows the nodes. The computer code used is the GE Short-Term Thermal-Hydraulic Model described in Reference 3.9-16. This model has been approved for use in ECCS conformance evaluation under 10 CFR Part 50, 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 3.9-16

a. 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 steam line;

b. 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 steam line 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; and

c. The enthalpies in the guide tubes and the bypass are calculated separately since the fuel assembly P is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Recirculation Line and Steam Line Break

3.9.5.2.3.1 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 DBA for the ESF reactor internals. The recirculation line break is the same as the design basis LOCA described in Section 6.3. 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 DBA for internal pressure differentials.

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow. The maximum internal pressure loads can be considered to be composed of steady-state and transient pressure differentials. For a given plant the core flow and power are the two major factors that 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. On the other hand, 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 RPV 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 lower power.

To ensure that the calculated pressure differences bound those which could be expected if a steam line break should occur, an analysis was conducted at a low power-high recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (that is, the drive flow necessary to achieve rated core flow at rated power).

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; and second, because high core flow is neither required nor desirable at such a reduced power condition.

3.9.5.2.4 Seismic and Hydrodynamic Events

The seismic and hydrodynamic loads acting on the structures within the reactor vessel are based on a dynamic analysis as described in Section 3.7.

3.9.5.3 Design Loading Categories

Loading combinations for the core support structures are shown in Table 3.9-2. The basis for determining faulted loads on the reactor internals is shown for seismic and hydrodynamic loads in Section 3.7 and for pipe rupture loads in Sections 3.9.5.2.3 and 3.9.5.4.3.

Table 3.9-2b includes loading combinations, analytical methods, and allowable and calculated stress values for typical core support structures and reactor internal components.

Stress intensity and other design limits for reactor internals are discussed in Section 3.9.5.4.4. The core support structures which are fabricated as part of the RPV assembly are discussed in Section 3.9.5.4.5.

The design requirements for equipment classified as "other," e.g., steam dryers and shroud heads, were specified by the designer giving appropriate consideration to the intended service of the equipment and expected plant and environmental conditions under which it will 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.

3.9.5.4 Design Bases

3.9.5.4.1 Safety Design Bases

The reactor core support structures and internals meet the following safety design bases:

a. Arrangement provides 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,

b. Deformation is limited to ensure that the control rods and ECCS can perform their safety functions, and

c. Mechanical design of applicable structures ensures that safety design bases items a and b are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.4.2 Power Generation Design Bases

The reactor core support structures and internals are designed to the following power generation design bases:

a. They provide the proper coolant distribution during all anticipated normal operating conditions up to full power operation of the core without fuel damage,

b. They are arranged to facilitate refueling operations, and

c. They are designed to facilitate inspection.

3.9.5.4.3 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 greatest with operation at the minimum power associated with the maximum core flow. 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 less.

3.9.5.4.4 Stress, Deformation, and Fatigue Limits for Reactor Internals (Except Core Support Structure)

These limits are summarized in Table 3.9-2b.

Design Condition	Minimum Safety Factor
Normal	2.25
Upset	2.25
Emergency	1.5
Faulted	1.125

Elastic displacement is considered in the design of reactor internal components in which deflection can affect control rod insertability. No plastic deformation occurs in any permanent core support structure components or the reactor vessel. In the case of the core support plate, a detailed elastic-plastic analysis was performed to verify the adequacy of the core plate buckling capability. The analysis used an incremental loading technique and showed the core plate adequacy under the worst combination of seismic plus hydrodynamic loads. Radiation induced deformation can occur in the fuel channel over the core life. These effects are considered in control rod insertability tests. No fatigue analysis is required under the faulted conditions due to the low encounter frequency of faulted events and the low number of cycles. The forcing functions applicable to the reactor internals are discussed in Section 3.9.2.5.

3.9.5.4.5 Stress, Deformation, and Fatigue Limits for Core Support Structures

These limits are summarized in Tables 3.9-2a, 3.9-2b, and 3.9-2aa.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

This section addresses the program for inservice testing for operational readiness of ASME Code Section III, Class 1, 2, and 3 pumps and valves. As required by 10 CFR 50.55a(f), inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month inspection interval, will comply with the requirements in the latest Edition and Addenda of the Code incorporated twelve months prior to the date of the operating license. Where compliance with certain Code requirements specified in 10 CFR 50.55a(f) are found to be impractical for CGS, relief requests are included in the IST Program Plan submitted to the NRC. For subsequent 120-month inspection intervals, the IST Program Plan shall be updated to comply with the requirements in the latest Edition and Addenda of the Code incorporated by 10 CFR 50.55a(b)(3) twelve months prior to end of previous inspection interval. The IST Program Plan lists all ASME Code Section III, Class 1, 2, and 3 pumps and valves required to perform a specific function in shutting down the reactor to the cold shutdown condition, in maintaining the cold shutdown condition, or in mitigating the consequences of an accident. The IST Program Plan also lists testing requirements for all listed pumps and valves and applicable relief requests.

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Table 3.9-1

Plant Events
(For Nuclear Steam Supply System and Balance-of-Plant)

Conditions	Number of Cycles
Normal, upset, and testing	
Bolt up/unbolt[a,b]	123
Design pressure hydrostatic test[b]	130
Startup (100°F/hr heatup rate)[b,c]	117
Daily reduction to 75% power[a]	10,000
Weekly reduction to 50% power[a]	2000
Control rod pattern change[a]	400
Loss of feedwater heaters (80 cycles total)[b]	80
Operating basis earthquake event at rated operating conditions	10/50[d]
Scrams	
Turbine generator trip, feedwater on, isolation valves stay open[b]	40
Other scrams[b]	140
Loss of feedwater pumps, isolation valves closed[b]	10
Single safety or relief valve blowdown[b]	8
Reduction to 0% power, hot standby, shutdown (100°F/hr cooldown rate)[b,c]	111
High-pressure core spray operation (10), standby liquid control operation (10), low-pressure core spray operation (10), and low-pressure coolant injection operation (10)[b]	40
Emergency	
Scrams	
Reactor overpressure with delayed scram feedwater stays on, isolation valves stay open	1[e]

Table 3.9-1

Plant Events
(For Nuclear Steam Supply System and Balance-of-Plant) (Continued)

Conditions	Number of Cycles
Automatic blowdown	1[e]
Improper start of cold recirculation loop	1[e]
Sudden start of pump in cold recirculation loop	1[e]
Improper startup with reactor drain shutoff followed by turbine roll and increase to rated power	1[e]
<u>Faulted</u>	
Pipe rupture	1[e]
Safe shutdown earthquake at rated operating conditions	1[e]
<u>ASME hydrostatic test</u>	
1.25 x design pressure hydrostatic test ASME Section III, NB-6222 and NB-3114, allows up to 10 of these tests without stress calculation	No additional

[a] Applies to reactor pressure vessel only.

[b] Thermal cycles are tracked for indication of reactor cumulative fatigue usage.

[c] Bulk average vessel coolant temperature change in any 1-hr period.

[d] Includes 50 peak OBE cycles for NSSS piping and 10 peak OBE cycles for other NSSS equipment and components. Fifty peak OBE cycles are postulated for all BOP piping and components.

[e] The annual encounter probability of the one-cycle events is 10^{-2} for emergency and 10^{-4} for faulted events.

Table 3.9-2

Loading Combination and Acceptance Criteria
For ASME Code Class 1, 2, and 3
Piping and Equipment

INTRODUCTION

This table lists the major safety-related and selected important-to-safety mechanical components in the plant. Various parts of the table are referenced in Section 3.9. The format in certain sections of the table is changed since analytical methods and depth of detail necessary to demonstrate the safety aspects of various components are different. The ASME Code allowable stresses, loads or limits are shown in all cases. As a result of reanalysis and computer accuracy, the maximum listed stresses and loads may vary slightly from current calculations. The tabulated values are provided as a demonstration that the Code allowables have been met. Any reanalysis will assure that the allowable stresses, loads or limits are still met.

INDEX TO INDIVIDUAL COMPONENTS

Table 3.9-2

Loading Combination and Acceptance Criteria
For ASME Code Class 1, 2, and 3
Piping and Equipment (Continued)

Table 3.9-2

Loading Combination and Acceptance Criteria
For ASME Code Class 1, 2, and 3
Piping and Equipment (Continued)

GE SCOPE OF SUPPLY

Load Combination SRSS[a]	Design Basis	Evaluation Basis
$N + SRV_{(ALL)}$	Upset	Upset (B)
$N + OBE$	Upset	Upset (B)
$N + [OBE^2 + SRV^2_{(ALL)}]^{1/2}$	Emergency	Upset (B)
$N + [SSE^2 + SRV^2_{(ALL)}]^{1/2}$	Faulted	Faulted[b] (D)
$N + [SBA^2 + SRV^2]^{1/2}$	Emergency	Emergency[b] (C)
$N + [IBA^2 + SRV^2]^{1/2}$	Faulted	Faulted[b] (D)
$N + [SBA^2 + SRV^2_{(ADS)}]^{1/2}$	Emergency	Emergency[b] (C)
$N + [SBA^2 + OBE^2 + SRV^2_{(ADS)}]^{1/2}$	Faulted	Faulted[b] (D)
$N + [IBA^2 + OBE^2 + SRV^2_{(ADS)}]^{1/2}$	Faulted	Faulted[b] (D)
$N + [SBA^2/IBA^2 + SSE^2 + SRV^2_{(ADS)}]^{1/2}$	Faulted	Faulted[b] (D)
$N + [{}^{c}LOCA^2 + SSE^2]^{1/2}$	Faulted	Faulted[b] (D)

Table 3.9-2

Loading Combination and Acceptance Criteria
For ASME Code Class 1, 2, and 3
Piping and Equipment (Continued)

BALANCE-OF-PLANT[d]

Load Cases	SRSS Load Combinations[e,f,g]	Design Assessment Acceptance Criteria
1	P + DW	Normal (A)
2	$N + [OBE^2 + SRV^2_{ONE}]^{1/2}$	Upset (B)
3	$N + [OBE^2 + SRV^2_{TWO}]^{1/2}$	Upset (B)
4	$N + [OBE^2 + SRV^2_{ALL}]^{1/2}$	Upset (B)
5	$N + [OBE^2 + SRV^2_{ADS} + SBA^2]^{1/2}$	Emergency[b] (C)
6	$N + [OBE^2 + SRV^2_{TWO} + SBA^2]^{1/2}$	Emergency[b] (C)
7	$N + [SSE^2 + SRV^2_{ADS} + SBA^2/IBA^2]^{1/2}$	Faulted[b] (D)
8	$N + [SSE^2 + SRV^2_{TWO} + SBA^2/IBA^2]^{1/2}$	Faulted[b] (D)
9	$N + [SSE^2 + SRV^2_{ONE}]^{1/2}$	Faulted[b] (D)
10	$N + [SSE^2 + SRV^2_{TWO}]^{1/2}$	Faulted[b] (D)
11	$N + [SSE^2 + SRV^2_{ALL}]^{1/2}$	Faulted[b] (D)
12	$N + [SSE^2 + DBA^2]^{1/2}$	Faulted[b] (D)

LEGEND

N Normal (and abnormal loads depending on acceptance criteria) include internal pressure (P) and dead weight (DW).

OBE Operating basis earthquake loads.

SSE Safe shutdown earthquake loads.

Table 3.9-2

Loading Combination and Acceptance Criteria
For ASME Code Class 1, 2, and 3
Piping and Equipment (Continued)

SRV	Safety/relief valve discharge induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling).
SRV_{TWO}	Safety/relief valve discharge induced loads from two adjacent valves. (This load is conservatively enveloped by the SRV_{ONE} case.)
SRV_{ALL}	The loads induced by actuation of all safety/relief valves which activate within msec of each other (e.g., turbine trip operational transient). (This load is the largest of the axisymmetric and nearly symmetric all valve loading conditions.)
SRV_{ADS}	The loads induced by actuation of safety/relief valves associated with the automatic depressurization system which actuate within msec of each other during the postulated small or intermediate size pipe rupture. (This load is conservatively taken as the SVR_{ALL} case.)
SRV_{ONE}	The loads induced by the actuation of one safety/relief valve.
LOCA[c]	The loss-of-coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping).
$LOCA_1$	Pool swell drag/fallout loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
$LOCA_2$	Pool swell impact loads on piping and components located above the suppression pool water upper surface.
$LOCA_3$	Oscillating pressure induced loads on submerged piping and components during condensation oscillations.
$LOCA_4$	Building motion induced loads from chugging.
$LOCA_5$	Building motion induced loads from main vent air clearing.
$LOCA_6$	Vertical and horizontal loads on main vent piping.
$LOCA_7$	Annulus pressurization loads.
SBA	The abnormal transients associated with a small break accident.

Table 3.9-2

Loading Combination and Acceptance Criteria
For ASME Code Class 1, 2, and 3
Piping and Equipment (Continued)

IBA The abnormal transients associated with an intermediate break accident.

DBA The abnormal transients associated with a design basis break accident.

[a] Square root of the sum of the squares (SRSS) combination of dynamic loads.

[b] All ASME Code Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet the requirements of NRC's memorandum, "Evaluation of Topical Report - Piping Functional Capability Criteria," dated July 17, 1980.

[c] The most limiting case of load combination among $LOCA_1$ through $LOCA_7$

[d] Equipment includes pumps, valves, supports, and vessels.

[e] All dynamic loads are combined using the SRSS method and the results are added to the static loads. As an analysis option the simpler but more conservative absolute sum method was used in some load evaluations.

[f] As required by the appropriate subsection, i.e., NB, NC, or ND of ASME Code Section III, Division 1; other loads, such as thermal transient, thermal gradients, and anchor points displacement portion of the OBE or SRV may require consideration in addition to those primary stress-producing loads listed.

[g] SBA, IBA, and DBA include all event-induced loads, as applicable, such as chugging, pool swell, drag loads, annulus pressurization, etc.

Table 3.9-2a

Reactor Pressure Vessel and Shroud Support Assembly
(i) Vessel Support Skirt

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
Material: SA 533 GR. B CL #1				
A. Normal and upset condition:				
$P_m \leq Sm$	1. Normal loads	Primary membrane	26,700	19,911
$Sm = 26,700$ @ 575°F	2. Upset pressure			
$P_L + P_b \leq 1.5\ Sm$	3. Operating basis earthquake	Primary membrane plus bending	40,050	28,369
$1.5\ Sm = 40,050$ @ 575°F	4. Safety/relief valve			
B. Emergency condition:				
$P_m \leq Sy$	1. Normal loads	Primary membrane	42,300	39,245[a]
$Sy = 42,300$ @ 575°F	2. Upset pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Operating basis earthquake	Primary membrane plus bending	63,450	56,485[a]
$1.5\ Sy = 63,450$ @ 575°F	4. Safety/relief valve			
C. Faulted condition:				
$P_m \leq Sy$	1. Normal loads	Primary membrane	42,300[b]	39,245
$Sy = 42,300$ @ 575°F	2. Faulted pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Chugging	Primary membrane plus bending	63,450[b]	56,485
$1.5\ Sy = 63,450$ @ 575°F	4. Safety/relief valve (ADS)			
	5. Safe shutdown earthquake			

D. Maximum cumulative usage factor: 0.064 at skirt-base junction

[a] Calculated stresses under faulted loading (greater than emergency loading).

[b] Allowable stresses under emergency loading (less than faulted loading).

Table 3.9-2a

Reactor Pressure Vessel and Shroud Support Assembly (Continued)
(ii) Shroud Support

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
Material: Inconel				
A. Normal and upset condition:				
$P_m \leq Sm$	1. Normal loads	Primary membrane	23,300	16,890
$Sm = 23{,}300$ @ 575°F	2. Upset pressure			
$P_L + P_b \leq Sy$	3. Operating basis earthquake	Primary membrane plus bending	28,100	25,540
$Sy = 28{,}100$ @ 575°F	4. Safety/relief valve			
B. Emergency condition:				
$P_m \leq Sy$	1. Normal loads	Primary membrane	28,100	23,000
$Sy = 28{,}100$ @ 575°F	2. Upset pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Chugging	Primary membrane plus bending	42,150	33,880
$1.5\ Sy = 42{,}150$ @ 575°F	4. Safety/relief valve (ADS)			
C. Faulted condition:				
$P_m \leq Sy$	1. Normal loads	Primary membrane	28,100	23,000
$Sy = 28{,}100$ @ 575°F	2. Faulted pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Chugging	Primary membrane plus bending	42,150	33,880
$1.5\ Sy = 42{,}150$ @ 575°F	4. Safety/relief valve (ADS)			
	5. Safe shutdown earthquake			

D. Maximum cumulative usage factor: 0.399 at shroud support plate

Table 3.9-2a

Reactor Pressure Vessel and Shroud Support Assembly (Continued)
(iii) Reactor Pressure Vessel, Feedwater Nozzle

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
Material: Inconel				
A. Normal and upset condition:				
$P_m \leq Sm$	1. Normal loads	Primary membrane	23,300	13,848
$Sm = 23{,}300$ @ 575°F	2. Upset pressure			
$P_L + P_b \leq Sy$	3. Operating basis earthquake	Primary membrane plus bending	35,000	15,120
$1.5\ Sm = 35{,}900$ @ 575°F	4. Safety/relief valve			
B. Emergency condition:				
$P_m \leq 1.2\ Sm$	1. Normal loads	Primary membrane	27,960	17,719
$1.2\ Sm = 27{,}960$ @ 575°F	2. Upset pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Chugging	Primary membrane plus bending	42,000	20,060
$1.5\ Sy = 42{,}000$ @ 575°F	4. Safety/relief valve (ADS)			
C. Faulted condition:				
$P_m \leq 1.2\ Sm$	1. Normal loads	Primary membrane	27,960	17,719
$1.2\ Sm = 27{,}960$ @ 575°F	2. Faulted pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Jet reaction	Primary membrane plus bending	42,000	33,610
$1.5\ Sy = 42{,}000$ @ 575°F	4. Annulus pressurization			
	5. Safe shutdown earthquake			

D. Maximum cumulative usage factor: 0.696 at thermal sleeve to safe end

Table 3.9-2a

Reactor Pressure Vessel and Shroud Support Assembly (Continued)
(iv) Control Rod Drive Penetration

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
Material: Inconel SB157 (stub tube)				
A. Normal and upset condition:				
$P_m \leq Sm$	1. Normal loads	Primary membrane	16,650	2,099[a]
$Sm = 16,650$ @ 550°F	2. Upset pressure			
$P_L + P_b \leq Sy$	3. Operating basis earthquake	Primary membrane plus bending	24,100	16,893[a]
$Sy = 24,100$ @ 550°F	4. Safety/relief valve			
B. Emergency condition:				
$P_m \leq 1.2\ Sm$	1. Normal loads	Primary membrane	19,980	2,099
$1.2\ Sm = 19,980$ @ 550°F	2. Upset pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Operating basis earthquake	Primary membrane plus bending	36,150	16,893
$1.5\ Sy = 35,150$ @ 550°F	4. Safety/relief valve			
C. Faulted condition:				
$P_m \leq 1.2\ Sm$	1. Normal loads	Primary membrane	19,980[b]	7,623
$1.2\ Sm = 19,980$ @ 550°F	2. Faulted pressure			
$P_L + P_b \leq 1.5\ Sy$	3. Jet reaction	Primary membrane plus bending	36,150[b]	31,996
$1.5\ Sy = 35,150$ @ 550°F	4. Scram			
	5. Safe shutdown earthquake			
D: Maximum cumulative usage factor: 0.196 at CRD housing				

[a] Calculated stresses under faulted loading (greater than normal and upset loading).
[b] Allowable stresses under emergency loading (less than faulted loading).

Table 3.9-2b

Reactor Internals and Associated Equipment
(i) Top Guide - Highest Stressed Beam

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
Material: 304 Stainless Steel				
A. Normal and upset condition:				
$P_m \leq Sm$	1. Normal loads	Primary membrane	16,900	819
Sm = 16,900 @ 550°F	2. Upset pressure			
$P_L + P_b \leq 1.5$ Sm	3. Operating basis earthquake	Primary membrane plus bending	31,690[a]	28,548
1.5 Sm = 25,350 @ 550°F	4. Safety/relief valve			
B. Emergency condition:				
$P_m \leq 1.2$ Sm	1. Normal loads	Primary membrane	20,280	318
1.2 Sm = 20,280 @ 550°F	2. Upset pressure			
$P_L + P_b \leq 1.8$ Sm	3. Chugging	Primary membrane plus bending	30,420	26,638
1.8 Sm = 30,420 @ 550°F	4. Safety/relief valve			
C. Faulted condition:				
$P_m \leq 2.4$ Sm	1. Normal loads	Primary membrane	40,560	900
2.4 Sm = 40,560 @ 550°F	2. Upset pressure			
$P_L + P_b \leq 3.0$ Sm	3. Safety/relief valve (ADS)	Primary membrane plus bending	50,700	35,919
3.0 Sm = 50,700 @ 550°F	4. Safe shutdown earthquake			

D. Maximum cumulative usage factor: 0.1625 at longest beam slot

[a] Includes a factor of 1.25 as a result of certified yield strength test data.

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Table 3.9-2b

Reactor Internals & Associated Equipment (Continued)
(ii) Core Plate (Ligament In Top Plate)

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
Material: <u>304 Stainless Steel</u>				
A. <u>Normal and upset condition:</u>				
$P_m \leq Sm$	1. Dead weight	Primary membrane	16,900	4,830
$Sm = 16{,}900$ @ 550°F	2. Pressure			
$P_L + P_b \leq 1.5\ Sm$	3. Operating basis earthquake	Primary membrane plus bending	25,350	13,980
$1.5\ Sm = 25{,}350$ @ 550°F	4. Safety/relief valve			
B. <u>Emergency condition:</u>				
$P_m \leq 1.2\ Sm$	1. Normal loads	Primary membrane	20,280	1,630
$1.2\ Sm = 20{,}280$ @ 550°F	2. Pressure			
$P_L + P_b \leq 1.8\ Sm$	3. SBA (chugging)	Primary membrane plus bending	30,420	12,080
$1.8\ Sm = 30{,}420$ @ 550°F	4. Safety/relief valve			
C. <u>Faulted condition:</u>				
$P_m \leq 2.4\ Sm$	1. Dead weight	Primary membrane	40,560	5,330
$2.4\ Sm = 40{,}560$ @ 550°F	2. Pressure			
$P_L + P_b \leq 3.0\ Sm$	3. Loss-of-coolant accident	Primary membrane plus bending	50,700	20,600
$3.0\ Sm = 50{,}700$ @ 550°F	4. Safety/relief valve			
	5. Safe shutdown earthquake			

D. <u>Maximum cumulative usage factor:</u> <u>0.005</u> at <u>stiffener-rim junction</u>

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Table 3.9-2b

Reactor Internals & Associated Equipment (Continued)
(iii) Differential Pressure & Liquid Control Lines

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
Material: 304 stainless steel				
A. Normal and upset condition: $P_m \leq Sm$ $Sm = 16{,}400$ @ 550°F $P_L + P_b + Q_m + Q_b \leq 3\ Sm$ $3.0\ Sm = 49{,}200$ @ 550°F	1. Dead weight 2. Upset pressure 3. Safety/relief valve	Primary membrane plus bending and secondary membrane plus bending	49,200	42,725
B. Emergency condition: $P_m \leq 1.2\ Sm$ $1.2\ Sm = 19{,}680$ @ 550°F $P_L + P_b \leq 1.8\ Sm$ $1.8\ Sm = 29{,}520$ @ 550°F	1. Normal loads 2. Upset pressure 3. Operating basis earthquake 4. Safety/relief valve	Primary membrane plus bending	29,520	17,015
C. Faulted condition: $P_m \leq 2.4\ Sm$ $2.4\ Sm = 39{,}360$ @ 550°F $P_L + P_b \leq 3.0\ Sm$ $3.0\ Sm = 49{,}200$ @ 550°F	1. Normal loads 2. Faulted pressure 3. Jet reaction 4. Annulus pressurization 5. Safe shutdown earthquake	Primary membrane plus bending	49,200	21,664

Table 3.9-2b

Reactor Internals & Associated Equipment (Continued)
(iv) Vessel Head Spray Nozzle

ASME B&PV Code Section III Primary Stress Limit Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
A. Normal and upset condition: S_m @ 575°F = 17.7 k/psi S_u @ 575°F = 60.0 Ksi S_{limit} = 3.0 S_m	1. Normal loads 2. Normal pressure 3. Operating basis earthquake	Primary membrane plus bending and secondary membrane plus bending	53,100	31,357
B. Emergency condition: S_m @ 575°F = 17.7 k/psi S_u @ 575°F = 60.0 Ksi S_{limit} = 1.5 (0.7 S_u)	1. Normal loads 2. Upset pressure 3. Operating basis earthquake 4. Safety/relief valve	Primary membrane plus bending	31,900	17,713
C: Faulted condition:[a] S_m @ 575°F = 17.7 k/psi S_u @ 575°F = 60.0 Ksi S_{limit} = 1.5 (0.7 S_u)	1. Normal loads 2. Faulted pressure 3. Chugging 4. Safety/relief valve (ADS) 5. Safe shutdown earthquake OR 1. Normal loads 2. Faulted pressure 3. Jet reaction 4. Annulus pressurization 5. Safe shutdown earthquake	Primary membrane plus bending	63,000	27,840

[a] Calculated faulted stress value is the maximum stress resulting from either of the two faulted loading conditions.

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Table 3.9-2b

Reactor Internals & Associated Equipment (Continued)
(v) GE Fuel Assembly (Including Channel)

Acceptance Criteria	Loading	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis Acceleration
Acceleration envelope	Horizontal direction: 1. Peak pressure 2. Safe shutdown earthquake 3. Annulus pressurization	Horizontal acceleration profile	1.5g	(1)
	Vertical direction: 1. Peak pressure 2. Safe shutdown earthquake 3. Safety/relief valve 4. Chugging	Vertical accelerations	5.1g (4)	(1)

Notes:

1. Evaluation basis accelerations and evaluations are contained in Reference 3.9-7.
2. The calculated maximum fuel assembly gap opening for the most limiting load combination is 0.25[(4)] in.
3. The fatigue analysis indicates that the fuel assembly has adequate fatigue capability to capability to withstand the loadings resulting from multiple SRV actuations and the OBE+SRV event.
4. These values are determined using the methodology contained in Reference 3.9-7.

Table 3.9-2c

Deleted

Table 3.9-2d

ASME Code Class 1 Main Steam Piping and Pipe Mounted Equipment - Highest Stress Summary

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress
ASME B&PV Code Section III, NB-3650						
Design Condition: Eq. 9 ≤ 1.5 S_m	Primary	22,881 psi	26,550 psi	0.86	1. Pressure 2. Weight 3. Operating basis earthquake	At MSIV (steam line C)
Service levels A & B (normal and upset) condition: Eq. 12 ≤ 3.0 S_m	Secondary	45,951 psi	53,100 psi	0.87	1. Pressure 2. Weight 3. Operating transient 4. Operating basis earthquake	At sweepolet (steam line B)
Service levels A & B (normal and upset) condition: Eq. 13 ≤ 3.0 S_m	Primary plus secondary (except thermal expansion)	52,789 psi	53,100 psi	0.99	1. Pressure 2. Weight 3. Operating basis earthquake 4. Operating transient	At sweepolet (steam line D)
Service levels A & B (normal and upset) condition: Cumulative usage factor:	N/A	0.72	1.0		1. Pressure 2. Weight 3. Operating basis earthquake 4. Operating transient	At sweepolet (steam line B)

Table 3.9-2d

ASME Code Class 1 Main Steam Piping and Pipe Mounted Equipment - Highest Stress Summary (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress (psi)	Allowable Stress Limits (psi)	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress
Service level B (upset) condition: Eq. 9 $\leq$ 1.8 S_m & 1.5 S_y	Primary	5,741	31,860	0.81	1. Pressure 2. Weight 3. Operating basis earthquake 4. Operating transient	At sweepolet (steam line D)
Service level C (emergency) condition: Eq. 9 $<$ 2.25 S_m & 1.8 S_y	Primary	23,429	39,825	0.60	1. Pressure 2. Weight 3. Operating transient 4. Small break accident	At MSIV (steam line C)
Service level D (faulted) condition: Eq. 9 $<$ 3.0 S_m	Primary	32,673	53,100	0.62	1. Pressure 2. Weight 3. Safe shutdown earthquake 4. Loss-of-coolant accident	At sweepolet (steam line B)

Table 3.9-2d

ASME Code Class 1 Main Steam Piping and Pipe Mounted Equipment - Highest Stress Summary (Continued)

Component/Load Type	Highest Calculated Load (lb)	Allowable Load (lb)	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads
Snubber/level B	35,352	120,000	0.30	1. Pressure 2. Weight 3. Operating basis earthquake 4. Operating transients	MS-SC-1 Line C
Snubber/level C	49,559	160,000	0.31	1. Pressure 2. Weight 3. Small break accident 4. Associated operating transients	MS-SC-2 Line C
Snubber/level D	51,296	190,000	0.27	1. Pressure 2. Weight 3. Intermediate break accident 4. Safe shutdown earthquake 5. Associated operating transients	MS-SC-2 Line C

Table 3.9-2d

ASME Code Class 1 Main Steam Piping and Pipe Mounted Equipment - Highest Stress Summary (Continued)

Component/Load Type	Highest Calculated Load (lb)	Allowable Load (lb)	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads
Safety/relief valves (SRV)					
Horizontal acceleration	3.02g	5.2g	0.58	1. Pressure 2. Weight 3. Safe shutdown earthquake 4. Loss-of-coolant accident	SRV 4B line B
Vertical acceleration	1.55g	4.4g	0.35	1. Pressure 2. Weight 3. Safe shutdown earthquake 4. Loss-of-coolant accident	SRC 5C line C
Main steam isolation valve (MSIV)					
Level B - moment	1960 in.-Kips	28,750 in.-Kips	0.07	1. Pressure 2. Weight 3. OBA 4. Operating transients	Nozzle line D
Level C - moment	2287 in.-Kips	28,750 in.-Kips	0.08	1. Pressure 2. Weight 3. Operating transients 4. Small break accident	Nozzle line D
Level D - moment	2395 in.-Kips	28,750 in.-Kips	0.08	1. Pressure 2. Weight 3. Operating transients 4. Safe shutdown earthquake 5. Loss-of-coolant accident	Nozzle line D

Table 3.9-2e

ASME Code Class 1 Recirculation Piping and Pipe Mounted Equipment - Highest Stress Summary

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points
ASME B&PV Code Section III, NB-3650						
Design condition: Eq. 9≤ 1.5 S_m	Primary	15,333 psi	25,013 psi	0.61	1. Pressure 2. Weight 3. OBE	RHR supply Loop A Node 327
Service levels A & B (normal and upset) condition: Eq. 12 ≤ 3.0 S_m	Secondary	33,468 psi	50,025 psi	0.67	1. Pressure 2. Weight 3. OBE 4. Operating transients	Header sweepolet Loop A Node 082
Service levels A & B (normal and upset) condition: Eq. 13 ≤ 3.0 S_m	Primary plus secondary (except thermal expansion)	36,760 psi	50,025 psi	0.74	1. Pressure 2. Weight 3. OBE 4. Operating transients	RHR supply Loop A Node 330
Service levels A & B (normal and upset) condition: Cumulative usage factor	N/A	0.85	1.0	0.85		Header sweepolet Loop A Node 082

Table 3.9-2e

ASME Code Class 1 Recirculation Piping and Pipe Mounted Equipment - Highest Stress Summary (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points
Service level B (upset) condition:					1. Pressure 2. Weight 3. Operating basis earthquake 4. Operating transients	RHR TEE Loop A Node 322
Eq. 9 $\leq 1.8\ S_m$ & $1.5\ S_y$	Primary	25,217 psi	28,200	0.89		
Service level C (emergency) condition:					1. Pressure 2. Weight 3. Operating transients 4. Small break accident	RHR TEE Loop A Node 322
Eq. 9 $\leq 2.25\ S_m$ & $1.8\ S_y$	Primary	22,172 psi	33,840	0.66		
Service level D (faulted) condition:					1. Pressure 2. Weight 3. Loss-of-coolant accident 4. Safe shutdown earthquake	RHR TEE Loop A Node 322
Eq. 9 $\leq 3.0\ S_m$ & $2.0\ S_y$	Primary	27,965 psi	37,600	0.74		

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Table 3.9-2e

ASME Code Class 1 Recirculation Piping and Pipe Mounted Equipment - Highest Stress Summary (Continued)

Component/ Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated Allowable	Identification of Equipment with Highest Loads
<u>Suction gate valve</u>				
- Moment (in.-lb)	773,261	3,146,380	0.25	Body C.G. Loop A
<u>Discharge gate valve</u>				
- Moment (in.-lb)	326,189	1,000,000	0.33	Body C.G. Loop B
<u>Recirculation pump</u>				
- Horizontal acceleration	1.72g	4.5g	0.38	Motor C.G. Loop B
- Vertical acceleration	1.8g	3.5g	0.51	Motor C.G. Loop B
<u>Flow control valve</u> (mechanically blocked open)				
- Horizontal acceleration	3.0g	9.0g	0.33	Operator Loop A
- Vertical acceleration	2.3g	6.0g	0.38	Operator Loop A

Table 3.9-2e

ASME Code Class 1 Recirculation Piping and Pipe Mounted Equipment - Highest Stress Summary (Continued)

Component/Load Type	Highest Calculated Load (lb)	Allowable Load (lb)	Ratio Calculated Allowable	Loading	Identification of Equipment with Highest Loads
Snubber/level B	53,009	120,000	0.44	1. Pressure 2. Weight 3. Operating basis earthquake 4. Operating transients	SB6 Loop B
Snubber/level C	30,490	159,600	0.19	1. Pressure 2. Weight 3. Small break accident 4. Associated operating transients	SA6 Loop A
Snubber/level D	57,001	180,000	0.32	1. Pressure 2. Weight 3. Loss-of-coolant accident 4. Safe shutdown earthquake 5. Associated operating transients	SB6 Loop B

Table 3.9-2f

Recirculation Flow Control Valve
(24 in. - Neles/Jamesburg, Formerly Hammel-Dahl)
(Kept in Mechanically Blocked Full Open Position)

	Criteria	Method of Analysis	Allowable Value	Analytically Determined Value
1.	Body primary stress	Per Subarticle NB-3500 of the ASME Code, Section III	19,600 psi	17,286 psi
2.	Body to top housing flange joint at maximum stress point	Per Subarticle NB-3500 of the ASME Code, Section III	28,650 psi	24,910 psi[a]
3.	Housing to bonnet flange joint at maximum stress point	Per Subarticle, NB-3500 of the ASME Code, Section III	28,650 psi	2,934 psi[a]
4.	Body to bottom cover joint at maximum stress point	Per Subarticle NB-3500 of the ASME Code, Section III	N/A	N/A
5.	Top housing to top cover joint at maximum stress point	Per Subarticle NB-3500 of the ASME Code, Section III	28,650 psi	22,230 psi[a]
6.	Body minimum wall	Per Subarticle NB-3541 of the ASME Code, Section III	2.417 in.	2.437 in.

[a] Flow control valve accelerations on Table 3.9-2e have been reduced, therefore these stresses will be reduced as well.

Table 3.9-2g

Safety/Relief Valves (Main Steam) Spring-Loaded, Direct Acting Type
ASME Code, Section III, July 1971

Topic	Method of Analysis	Crosby 6-R-10 Analysis	Allowable Value	Calculated
Body inlet and outlet flange stresses	$S_H = \frac{fMo}{Lg_2^1 B} + \frac{Pb}{4g_o} < 1.5\ Sm$	$\frac{fMo}{Lg_1^2 B} + \frac{P_bB}{4g_1} P_b(\text{Crosby}) + P\ (\text{Codes})$	1.5 Sm = 27,300 psi (inlet)	<u>Inlet</u>: S_H = 1.2 Sm = 0.8 (allowable)
	$S_R = \frac{(4te / 3 + 1)Mo}{Lt^2B} < 1.5\ Sm$	$\frac{(4te / 3+1)Mo}{Lt^2J;} g_1(\text{Crosby}) = g_o, g_1(\text{Codes})$	and = 29,000 psi (outlet)	S_R = 0.3 Sm = 0.2 (allowable)
	$S_T = \frac{YMo}{t^2b} - ZS_R < 1.5\ Sm$	$\frac{YMo}{t^2b} - ZS_R\ \ J(\text{Crosby}) = B(\text{Codes})$		S_T = 1.4 Sm = 0.92 (allowable)
	where: S_H = Longitudinal "hub" wall stress, psi S_R = Radial "flange" (body, base, inlet) S_T = Tangential "flange" stress, psi	Material: A-105, Grade II Inlet: Sm @ 575°F = 18,200 psi Outlet: Sm @ 500°F = 19,400 psi		<u>Outlet</u> S_H = 0.82 Sm = 0.55 (allowable) S_R = 0.99 Sm = 0.66 (allowable) S_T = 0.27 Sm = 0.18 (allowable)
Inlet and outlet stud area requirement	Total cross-sectional area shall exceed the greater of: $Am_1 = \frac{Wm_1}{Sb}$,or $Am_2 = \frac{Wm_2}{Sa}$ where: Am_1 = total required bolt (stud) area for operation condition. Am_2 = total required bolt (stud) area for gasket seating.	$Am_1 = \frac{Wm_1}{Sb}$ $AM_2 = \frac{Wm_2}{Sa}$	<u>Inlet</u>: Am_1 (Am_2) <u>Outlet</u>: Am_1 (Am_2)	<u>Inlet</u>: Am (actual)/(required minimum) = 1.61 <u>Outlet</u>: Am (actual)/(required minimum) = 2.0

Table 3.9-2g

Safety/Relief Valves (Main Steam) Spring-Loaded, Direct Acting Type (Continued)
ASME Code, Section III, July 1971

Topic	Method of Analysis	Crosby 6-R-10 Analysis	Allowable Value	Calculated
Nozzle wall thickness	1. Valve Wall thickness Criterion: $t_{minimum} < t_A$ where: $t_{minimum}$ = minimum calculated thickness requirement, including corrosion allowance. t_A = Actual nozzle wall thickness. (<u>NOTE</u>: This $t_{minimum}$ is t_m per notation of the codes.)	Thin section near valve seat: $t_{m_{1-1}} < t_{a_{1-1}}$ Section at about middle of nozzle: $t_{m_{2-2}} < t_{a_{2-2}}$	$t_{m_{1-1}} =$ $t_{m_{2-2}} =$ 0.468 in. Actual thickness greater than t_m at the section under consideration.	$t_{a_{1-1}} = 1.2(t_{m_{1-1}})$ $t_{a_{2-2}} = 1.3(t_{m_{2-2}})$
	2. Cyclic Rating: <u>Thermal</u> $It = \sum \frac{Nri}{Ni}$	$It = \sum \frac{Nri}{Ni} (i = 1, 2, \& \ 3)$	It (max.) < 1	It = 0.032 (= 0.032 (It max.))
	<u>Fatigue</u> Na > 2000 cycles, as based on Sa, where Sa is defined as the larger of $SP_1 = (2/3)Q_P + \frac{P_{eb}}{2} + Q_{T_2} + 1.30Q_{T_1}$ (Uses same notation as codes) $SP_2 = 0.4\,Q_P + \frac{K}{2} + (P_{eb} + 2Q_{T_2})$	Na ≥ 2000 cycles, as based on SP, where SP (Crosby) = Sa (Codes)	Na ≥ 2000 cycles	Na (based on SP = SP_1) > 10^6 cycles: satisfies criterion

Table 3.9-2g

Safety/Relief Valves (Main Steam) Spring-Loaded, Direct Acting Type (Continued)
ASME Code, Section III, July 1971

Topic	Method of Analysis	Crosby 6-R-10 Analysis	Allowable Value	Calculated
Nozzle wall thickness (continued)	Fatigue (continued) where: SP_1 = Fatigue stress intensity at inside surface of crotch, psi. SP_1 = Fatigue stress intensity at outside surface of crotch, psi.			
Bonnet flange strength	$S_H = \frac{fMo}{Lg_1^2 B} + \frac{PB}{4g_o} < 1.5Sm$ $S_R = \frac{(4te/3+1)Mo}{Lt^2 B} < 1.5Sm$ $S_T = \frac{YMo}{t^2 B} - ZS_R < 1.5Sm$	$\frac{fMo}{Lg_1^2 B} + \frac{P_B B}{4g_1}$ $\frac{(4te/3+1)Mo}{LT^2 B}$ $\frac{YMo}{t^2 B} - ZS_R$	1.5 Sm (for maximum S_H' S_R' S_T) P_B (Crosby) = PC(Codes) = 29,100 psi g_1(Crosby) = $g_0 g_1$,(Codes)	S_H = 1.47 Sm = 0.98 (allowable) S_R = 0.45 Sm = 0.3 (allowable) S_T = 0.46 Sm = 0.31 (allowable)
	where: S_H = Longitudinal "hub" wall stress, psi. S_R = Radial "flange" stress, psi. ST = Tangential "flange" stress, psi.	Material: A-105, Grade II Sm at 500 °F = 19,400 psi		

LDCN-99-026

Table 3.9-2g

Safety/Relief Valves (Main Steam) Spring-Loaded, Direct Acting Type (Continued)
ASME Code, Section III, July 1971

Topic	Method of Analysis	Crosby 6-R-10 Analysis	Allowable Value	Calculated
Bonnet stud area requirements	Total cross-sectional area shall exceed the greater of: $Am_1 = \frac{Wm_1}{Sb}$, or $Am_2 = \frac{Wm_2}{Sa}$ where Am_1 = Total required bolt (stud) area for operating condition. Am_2 = Total required bolt (stud) area for gasket seating.	$Am_1 = \frac{WM_1}{S_b}$ $Am_2 = \frac{Wm_2}{Sa}$	Am_1 ($>Am_2$)	Am (actual) = 1.4 (required minimum)
Disc insert	Part No. N97499 (Valve Dwg. DS-A-63790-3) Method of Analysis: Algor SUPERSAP (finite element) Computer Program, SSAP0. Due to the complex geometrical shape of the 6R10 HB-BP-DF disc insert, a finite element computer model of the disc was constructed as follows: A half cross section two dimensional axisymmetric model was selected to represent the disc insert, since it is axisymmetric in shape, loading, and restraint about the vertical centerline. The most severe loading condition, "the valve is unpressurized with the spring compressed to establish valve set load" was compared with the lowest allowable stress intensity (at design temperature 575°F).			

LDCN-99-026

Table 3.9-2g

Safety/Relief Valves (Main Steam) Spring-Loaded, Direct Acting Type (Continued)
ASME Code, Section III, July 1971

Topic	Method of Analysis	Crosby 6-R-10 Analysis	Allowable Value	Calculated
Disc insert (continued)	Part No. N97499 (Valve Dwg. DS-A-63790-3) Where: P_{SET} = Set Pressure, 1250 psig F_{SET} = Set Load, 24,950 lb a = ½ disc insert undercut diameter, 2.782 in. t = Design minimum thickness, 0.616 in. S_i = Maximum bending stress intensities, psi	w = System pressure, 0 psig D_s = Disc inside diameter, 5.041 in. b = Hub radius 0.591 in. m = Reciprocal of Poisson's ratio, 3.333		
	Material: SA-637 TP 718 (SA/SB-637 Gr.718) S_m = 45,225 psi (575°F), Allowable bending stress intensity: 1.5 S_m = 67,838 psi (@ 575°F)			
	SSAP0	{ Algor SUPERSAP (finite element) Computer Program }		
		(At lower surface) (At upper surface)	1.5 S_m = 67,838 psi 1.5 S_m = 67,838 psi	S_i = 23,100 psi S_i = 19,600 psi
Spring washer stress requirements	$S_S = \frac{F_T}{A_S}$ where: F_T = Total spring load at full lift A_S = Shear area, $(in.)^2$	(Same notation) Material: A-105, Grade II Sm (at 400°F) = 26,000 psi	$S_S < 0.6$ Sm	S_S = 0.15 Sm = 0.25 (allowable)

Table 3.9-2h

Main Steam Isolation Valve

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness	Calculated Stress or Actual Thickness
Design of pressure retaining parts	All references are made to ASME Boiler & Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1971 Edition as Addended through Winter 1971. Reference the same code for explanation of the symbols used.		
Body minimum wall thickness	Reference paragraph NB-3543, Nonstandard Pressure-Rated Valve, Table NB-1542-1. For design condition of 1250 psig and 575°F. The primary service rating = 495 based on a core diameter of 23.9 in. t_m = 1.58 in. (Including a corrosion allowance of 0.12 in.).	1.58 in.	2.12 in.
Body shape rule	Reference paragraph NB-3544, Body Shape Rules		
Radius of crotch	Reference paragraph NB-3544.1(a), Radius of Crotch criterion $r_2 \geq 0.3\ t_m$ as r_2 = 1.00 in., t_m = 1.58 + 1.00 0.3 x 1.58 = 0.47 criterion satisfied		
Corner radii on internal surfaces	Reference paragraph NB-3544.1(b), Corner Radii on Internal Surfaces criterion $r_4 < r_2$ as r_4 = 0.69 in., r_2 = 1.00 in. + 0.69 < 1.00 criterion satisfied		
Out of roundness	Reference paragraph NB-3544.5, Out of Roundness, Figure NB-3545.1-2 $\frac{b}{t_b} + \frac{3}{4}\frac{3b^2 - 2ab - a^2}{t_b^2} + 1 \leq 1.5\frac{S_m}{P_s}$ a = 10.20 in., b = 15.75 in., c = 4.13 in. Sm = 19,400 psi 18.83 ≤21.56 Criterion satisfied		

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness	Calculated Stress or Actual Thickness
Longitudinal curvature	Reference paragraph NB-3544.6 Longitudinal Curvature		
	Criterion $\gamma \frac{1}{\text{Long.}} + \gamma \frac{1}{\text{Lat.}} - \frac{4}{3d_m}$ as $r_{\text{Long.}} = 35.31$ in.,		
	$r_{\text{Lat.}}$ = 15.75 in., d_m = 23.90 in., + 0.09 ≤ 0.06		
	criterion satisfied		
Flat wall limitation	Reference paragraph NB-3544.7, Flat Wall Limitation		
	$\frac{d}{t} - \frac{3dm}{2tm}$ dm = 23.90 in. tm = 1.58 in. d = 35.76 in. t = 4.06 in.		
Minimum wall at weld end	Reference paragraph NB-3544.8, Minimum Wall at Weld End		
	Actual thickness at 1 x t_m (i.e., 1.58 in. measured along the run direction) is 3.80 in.	1.58 in.	3.80 in.
Primary crotch Stress due to internal pressure	Reference paragraph NB-3545.1		
	criterion $P_m = (\frac{A_f}{A_m} + 0.5) P_S \quad S_m$		
	where A_f = 591.8 in.^{2}, A_m = 128.4 in.^{2}, P_S = 1,350 psig,	19,400 psi	6,897 psi
	P_m = 6,897 psi, S_m = 19,400 psi, since S_m P_m		
	criterion satisfied		
Valve body secondary stress	Reference paragraph NB-3545.2		

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress (psi)	Calculated Stress (psi)
Primary plus secondary stress due to internal pressure	Reference paragraphs NB-3545.2(a)(1), NB-3545.2(a)(2) $Q_P = C_P(\frac{r_i}{t_e} + 0.5) P_S Ca$ where C_P = 3, r_i = 11.80 in., P_S = 1350 psi. t_e = 3.80, Q_P = 14,601 psi for wye-type valve C_a = 1.33 + Q_P = 19,420		
Secondary stress due to pipe reaction	Reference paragraph NB-3545.2(b), Figures NB-3545.2-3. NB-3545.2-5, and NB-3545.2-6		
Direct or axial load effect	$P_{ed} = \frac{F_d S}{G_d}$ where $S = 41{,}000\ F_{d.} = 34.5\ \text{in.}^2 G_d = 314.8\ \text{in.}^2$ $\rightarrow P_{ed} = 4493$ psi	29,100	4,493
Bending load effect	$P_{eb} = C_b \frac{F_b S}{G_b}$ where $S = 41{,}000, F_b = 380\ \text{in.}^3$ $\text{i.d.} = 23.65\ \text{in.}, r_i = 11.80, t_e\ 3.80_{\bar{r}} = 13.70\ \text{in.}$ $\text{as}\ \frac{t_e}{r} = 0.257 > 0.19 + C_b = 1$ $G_b = \frac{I}{r_i + t_e}$ where $I = 30{,}453\ \text{in.}^4, r_i = 11.83\ \text{in.},$ t_e = 3.80 G_b 1948 in. ^{3} $\rightarrow P_{eb}$ = 7998 psi	29,100	7,998

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress (psi)	Calculated Stress (psi)
Torsion load effect	Reference paragraphs 3545.29(b)(1). 3545.2(b)(6)(c) $P_{et} = 2\frac{F_b S}{G_t}$ where $F_b = 380\text{ in.}^3$, $S = 41{,}000$ psi $G_t = C_t t\overline{A}$ $t = 3.43\text{ in.}, \overline{A} = 596\text{ in.}^2, C_t = 1.78, G_t = 3639\text{ in.}^3$ $P_{et} = 8563$ psi	29,100	8,563
Thermal secondary stress at crotch region	Reference paragraph NB-3545.2(c), Figures NB-3545.2(c)-2, NB-3545.2(c)-3, and NB-3545.2(c)-4 $Q_T = Q_{T_1} + Q_{T_2}$ where $Te_1 = 5.20\text{ in.}, Q_{T_1} = 3200$ $C_{T_2} = C_6 C_2\ \Delta T_2$ where $C_2 = 0.48\ C_6 = 210$ and $\Delta T_2 = 1.6^\circ F$ $Q_{T_2} = 161\text{ psi}, Q_T = 3361\text{ psi}$ criterion $S_N = Q_P + P_{ed} + 2Q_{T_2} \leq 3\,S_m$ where $Q_P = 19{,}420\ P_{ed} = 4493\ Q_{T_2} = 161$ as 24,235 ≤ 58,200 criterion satisfied	58,200	24,235
Normal duty valve fatigue requirements	Reference paragraphs 3545.3, NB-3545.3(a), NB-3545.3a, and Figure 1-9-1 criterion $N_a \geq 2{,}000$ cycles		

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress (psi)	Calculated Stress (psi)
Normal duty valve fatigue requirements (continued)	$S_{P_1} = \frac{2}{3} Q'P + \frac{P_{eb}}{2} + Q_{T_3} + 1.3 Q_{T_1}$ $S_{P_2} = 0.4Q \;\; P + \frac{K}{2}(P_{eb} + 2Q_{T_3})$		
	where $Q'_P = 19{,}420$ $P_{eb} = 7998$ Kr2, $Q_{T_1} = 3200$		
	$Q_{T_3} = 175\,\text{psi} + S_{P_1} = 21{,}267$ $S_{P_2} = 16{,}116$ S_a equal to the larger of S_{P_1} and		
	$S_{P_2} \rightarrow S_a = 21{,}267 + N_a = 75{,}000 \geq 2{,}000$ criterion satisfied		
Cyclic loading requirements at valve crotch	Reference paragraph NB-3550 For the largest temperature change range criterion Q' + $P_{ed} + C_6C_2C_4 \, \Delta T_f \max \leq 3\, S_m$		
	where $Q'_P = 19{,}420$ psi, $P_{ed} = 4493$ $C_6 = 210$ at ΔT_f max of 342°F, $C_2 = 0.48$, $C_4 = 0.15$ $S_m = 19{,}400$		
	$\rightarrow 29{,}084 \leq 58{,}200$ criterion satisfied	58,200	29,084
	Thermal Transients Not Excluded by Code Criterion $\sum \frac{N_{ri}}{N_i} < 1$		
	Calculate the fatigue usage factor (I_t) as follows:		
	$S_n \text{ Max} = Q'_P + P_{eb} + C_6C_3C_4 \Delta T_f \max + S_n \max = 33{,}343$ psi		
	Since $S_n \max < 3\, S_m$ (= 58,200) the following equation is used:		

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness	Calculated Stress or Actual Thickness
Cyclic loading requirements at valve crotch (continued)	$S_i = \frac{4}{3} Q'\ P + P_{eb} + C_6 (C_3 C_4 + C_5) \Delta T_{f_i}$		
	for ΔT_{fi} + 122 N_{ri} = 10, S_i = 64,956 psi, N_i = 18,000		
	N_{ri}/N_i = .0005		
	ΔT_{fi} = 90 N_{ri} = 120 S_i = 56,808 psi, N_i = 22,000		
	N_{ri}/N_i = .0050		
	ΔT_{fi} = 342 N_{ri} = 8, S_i = 120,973 psi, N_i = 2100		
	N_{ri}/N_i = .0038		
	$\text{as } I_t = \sum \frac{N_{ri}}{N_i} = .0093 < 1 \text{ criterion satisfied}$		
Disk design calculation	Reference paragraph NB-3546.3, Table I-1.1 Roark, 4th Edition Pages 220, 222		
	Disk design conditions, P_s = 1350 psi at 500°F, S_m = 20,800 psi @ 500°F		
	$\text{Case No. 13 } S = \frac{3W}{4mt^2(a^2 - b^2)} (a^4(3m+1) + b^4(m-1) - 4_m a^2 b^2 - 4(m+1)a^2 b^2 (1n(a / b)1$		
	$\text{Where } W = 1350 \text{ psi}, m = \frac{10}{3'}\ t + 5.875 \text{ in.},\ a = 10.75 \text{ in.}, b = 2.28 \text{ in.},\ S_t = 424{,}828 \text{ psi}$		

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness	Calculated Stress or Actual Thickness
Disk design calculation (continued)	Case No. 14 $S = \frac{3W}{2mt^2} \frac{(2a^2(m+1)}{a^2 - b^2} \ln (\frac{a}{b}) + (m-1)$ where $W = 61{,}072\ lb_f{}'\ t = 1.0$ in., $m = \frac{10}{3'}$ $a = 11.93$ in., $b = 2.28$ in., $S_t = 150{,}633$ psi	5.50 in.	6.60 in.
	Case No. 21 $S_r = \frac{3W}{4t^2} \frac{(4a^4(m+1)\ln\left(\frac{a}{b}\right) - a^4(m+3) + b^4(m-1) + 4a^2b^2)}{a^2(m+1) + b^2(m-1)}$ where $W = 1350$, $m = 10/3$, $t = 2.67$ in, $a = 11.93$ in., $b = 8.88$ in. $\rightarrow S_r = 6{,}090$ psi Case No. 22 $S_r = \frac{3W}{2t^2} \frac{(2a^2(m+1)\ln\left(\frac{a}{b}\right) - a^2(m-1) - b^2(m-1)}{a^2(m+1) + b^2(m-1)}$ where $W = 448{,}998$ $m = 10/3$, $t = 2.67$ in., $a = 11.93$ in., $b = 9.88$ in $\rightarrow S_r = 11{,}996$ psi Total stress $= S_{r_{21}} + S_{r_{22}} = 14{,},638$ psi, allowable stress 17,800 psi	20,800 psi	14,638 psi
	S_{shear} at inner edge disk $S_{shear} = \frac{F}{\lambda}$ where $F = 61{,}072$ lb., $\lambda = 98.08$ in.^{2} $\rightarrow S_{shear} = 623$ psi	12,480 psi	623 psi
	$S_{shear} = \frac{F}{\lambda}$ where $F = 620{,}458$ lb. $\lambda = 134$ in.^{2} $\rightarrow S_{shear} = 4{,}630$ psi Allowable shear stress = 0.6 x allowable stress = 0.6 x S_m = 12,480 psi	12,480 psi	4,630 psi

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness	Calculated Stress or Actual Thickness
Tensile stress at thread	$S_A = \frac{F}{A_t}$ where F = 440,893 lb, A_t = 193.7 in.^{2} , S = 2276 lb S_m = 20,800 psi		
Stem disk design calculation	Ref. Roark 4th Ed. P. 216 Design Conditions: P_s = 1350 psi @ 500°F, S_{all} = 20,800 psi		
Tensile and shear	Case No.1 $S_r = S_t = \frac{3W_P}{8\ mt^2}(3m+1)$ W_P = PA = 26,085 lb., m - 10/3, t = 1 (unit thickness) = > S_t' = S_r = 10,285 psi Case No.3 $S_r = S_t = \frac{3W}{2\eta mt^2}\left(1/2(m-1)+(m+1)\ln(\frac{a}{r_o})-(m-1)\frac{r_o^2}{2a^2}\right)$ W = 35,210 lb., t = 1 in. (unit thickness), a = 2.48 in. r_o = .94 in., m = 10/3 = > S_{t3} = S_r = 26,190 psi => $t_r = \frac{S_{t1}+S_{t3}}{S_m}$ = 1.32 in. $t_{required}$ = t_r + 2(12) = 1.56 in.	1.56 in.	1.85 in.
Shear stress above seat	$S_s = \frac{Fs}{A}$, Fs = 65,535 lbs., A = 22.2 in.^{2} , S = 2920 psi, $S_{all} = 0.6 S_m$ = 12,480 psi	12,480 psi	2,920 psi

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness	Calculated Stress or Actual Thickness
Bonnet design calculations including seismic accelerations for SSE	Reference paragraph NB-3647.1(a), paragraph UG-34 (k)(2) of Section VIII Div. 1, 1971 Edition		
Minimum thickness	$P_{fd} = P + P_{eq}, P_{eq} = \frac{16M}{G^3} + \frac{4F}{G^2}$ where M = 834,415 in. -lb, F = 39,400 lb, G = 24.72 in. P_{eq} = 381 psi, P_{fd} = 1756 psi $t = d\sqrt{\frac{CP}{S} + \frac{1.78\,W\,hg}{S_d{}^3}}$ where C = 0.3, P = 1714 psi, S = 19,400 psi, hg = 3.05 in. W = 1,077,640 lb, d = 24.72 in. t = 5.329 in., t = 5.329 + 0.120 = 5.45 in. (corrosion allowance is 0.120 in.)	5.45 in.	8.88 in.
Reinforcement	Reference paragraph UG-39 (d) (2) Section VIII Div. I, 1971 Edition. To account for the opening for stem in the bonnet. $t = 2\left(d\frac{CP}{S} + \frac{1.78\,W\,hg}{S_d{}^3}\right)$ t = 7.54 in., t = 7.54 + 0.12 = 7.66 in.		
Bonnet studs design calculation	Reference paragraph NB-3232.1 and Article E-1000 Bolt used 24 pieces of 1 5/8 - 8 UNC Bolts Total bolt area = 42.72.^{2}		

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress (psi)	Calculated Stress (psi)
Normal operation	1. Pressure stress at Operating Condition $S_1 = \frac{W_{ml}}{A_b} = 25{,}226\ \text{lb / in .2}$ where $W_{ml} = 1{,}077{,}640\ \text{lb}$, $A_b = 42.72\ \text{in.}^2$ 2. Gasket Load at ambient condition with no internal pressure $S_2 = \frac{W_{m2}}{A_b} = 2{,}616\ \text{lb / in}^2$ where $W_{m2} = 111{,}774\ \text{lb}_f$ $A_b = 42.72\ \text{in.}^2$ Maximum tensile stress = 25,226 lb/in.^{2} Thermal stress is assumed negligible because the coefficients of thermal expansion of bonnet plate and stud are the same. Standard preload 45,000 psi S_{all} = 69,000 psi	69,400	45,000
Body flange design calculations	Reference paragraph NB-3647.1 and Section VIII Div. I, 1971 Edition. Total flange moment under operating conditions $M_O = M_D + M_G + M_T$ $M_D = H_D h_D$' $H_D = 0.785\ B^2P$, $h_D = R + 0.5g$, where B = 24.14 in., P = 1,714 psi $\rightarrow$ H_D 784,070 lb_f, h_D = 1.94 in. M_D = 1,521,096 in.-lb $M_G = H_G h_g, H_G = W - H, h_G = \frac{C - G}{2}$ where W is the higher of W_{m1} and W_{m2} $W_{ml} = 0.785\ G^2P + (2b \times 3.14\ G\ m\ P)$ $W_{m2} = 3.14\ G_{by}$		

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress (psi)	Calculated Stress (psi)
Body flange design calculations (continued)	where G = 24.72 in., b = .32 in., m = 3, y = 4,500 → W_{ml} = 1,077,640 lb, W_{m2} = 111,774 lb → H_G = 255,440 lb, h_G = 3.02 in. →M_G = 771,429 in.-lb $M_T = H_T h_T, H_T = H - H_D, h_T = \frac{R + g_1 + h_G}{2}$ where H = 822,200, H_D = 784,070, R = .575 in., q_1 = 2.73 in.-lb. h_G = 3.02 → H_T = 38,130 lb, h_T = 3.16 in., M_T = 120,586 in.-lb. M_O = 2,413,111 in.-lb, where M_D = 1,521,096 in.-lb M_G = 771,429 in.-lb, M_T = 120,586 Total flange moment under gasket seating condition $M_O = W\left(\frac{C - G}{2}\right),\ W = \left(\frac{A_m + A_b}{2}\right) s_a$ where C = 30.75 in., A_b = 42.72 in.^{2}, G = 24.72 in., A_m = 31.06 in.^{2}, s_a = 40,000 psi at 100°F → W = 1,475,600 lb, M_O = 4,448,934 lb-in.		
Longitudinal hub stress	Reference Paragraph NB-3647.1(d) $S_H = \frac{fM_O}{Lg_1^2 B} + \frac{PB}{4g_O}, 1.5\,S_m = 29{,}100\ \text{lb/in.}^2$ $(S_H)_{oper}$ = 14,484 psi, $(S_H)_{atmos}$ = 23,507 psi	29,100	14,484

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress (psi)	Calculated Stress (psi)
Radial stress	Reference UA-51(1), Equation (7) of Section VIII of ASME B&PV Code, 1971 Edition	34,950	23,507
	$S_R = \frac{(1.33t_e + 1)M_O}{Lt^2B}; 1.5S_m = 29{,}100$ psi	29,100	5,144
	$(S_R)_{open} = 5144$ psi, $(S_R)_{atmos} = 9483$ psi	34,950	9,483
Tangential stress	$S_T = \frac{(YM_O)}{t^2B} - ZS_R; 1.5\, S_m = 29{,}100$		
	where Y = 57, t = 5.56 in., Z = 3.50, B = 24.14 in.	29,100	3,015
	$(S_T)_{open}$ = 3,015 psi; $(S_T)_{atmos}$ = 5533 psi	34,950	5,533
Flange stress criteria	$\frac{S_H + S_R}{2} S_m, \frac{S_H + S_T}{2} S_m$		
	Open / Atmos.		
	$\frac{S_M + S_R}{2} = 9814$ psi / 16,495 psi	19,400	9,814 (open)
	$\frac{S_H + S_T}{2} = 8750$ psi / 14,295 psi	19,400 23,300	8,750 (atmos.) 16,495 (open)
		23,300	14,295 (atmos.)

Table 3.9-2h

Main Steam Isolation Valve (Continued)

Criteria	Method of Analysis	Allowable Stress (psi)	Calculated Stress (psi)
Stem calculation			
Back seated stress	$S = \frac{F}{A}$		
	where F = 9916 lb net upward force		
	A = 2.268 in.^{2}, the smallest cross-sectional area on the stem		
	S = 4372 psi < 26,700 psi	26,700	4,372
Valve close stem stress	$S = \frac{F}{A}$		
	where F = 39,450 lb net down force		
	A = 2.268 in.^{2}, the smallest cross-sectional area on the stem		
	S = 17,394 psi < 26,700 psi	26,700	17,394
Stem thread strength	Reference Federal Thread Standard Stem thread Mating With Disk		
	Thread 1.875 in. - 12 UN - 2 Thread		
	A_{S1} = 5.23 in.^{2}/inch engagement		
	$r = \frac{F}{A_{S1}}$		
	where F = 39,450 lb_f, A_{S1} = 5.23 in.^{2}, t_{sd} = 7543 psi	16,020	7,543

Table 3.9-2i

Recirculation Pump[a]

	Criteria	Method of Analysis	Analytical Results	Allow, Stress, or Actual Thickness
1.	<u>Casing minimum wall thickness</u>	$t = \frac{PR}{SE - 0.6P} + C$	t = 2.855 in.	S_{allow} = 15,075 psi $t_{act.}$ = 2.858 in.
	A. <u>Loads</u>: <u>Normal and upset condition</u> Design pressure & temperature	where:		
	B. <u>Primary membrane stress limit</u>: Allowable working stress per ASME Section III, Class C	t = minimum required thickness, in. P = design pressure, psig R = maximum internal radius, in. S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in.		
2.	<u>Casing cover minimum thickness</u>			
	A. <u>Loads</u>: <u>Normal and upset condition</u> Design pressure & temperature	$S_r = \frac{3W}{4t^2}\left(a^2 - 2b^2 + \frac{b^4(m-1) - 4b^4(m+1)\ln(a/b) + a^2b^2(m+1)}{a^2(m-1) + b^2(m+1)}\right)$	s_r = 8,015 psi	S_A = 15,075 psi 1.5 S_m = 22,607 psi
	B. <u>Primary bending stress limit</u>:	$+\frac{3W}{2t^2}\left(1 - \frac{2mb^2 - 2b^2(m+1)\ln(a/b)}{a^2(m-1) + b^2(m+1)}\right)$		
	1.5 S_m per ASME Code for Pumps & Valves for Nuclear Power Class I.	$S_t = -\frac{3W(m^2-1)}{4mt^2}\left(\frac{a^4 - b^4 - 4a^2b^2\ln(a/b)}{a^2(m-1) + b^2(m+1)}\right) +$	S_t = 3,984 psi	S_A = 15,075 psi
		$\frac{3W}{2mt^2}\left(1 + \frac{ma^2(m-1) - mb^2(m+1) - 2(m^2-1)a^2\ln(a/b)}{a^2(m-1) + b^2(m+1)}\right)$		

Table 3.9-2i

Recirculation Pump[a] (Continued)

Criteria	Method of Analysis	Analytical Results	Allow, Stress, or Actual Thickness
	where: S_r = radial stress at outer edge, psi S_t = tangential stress at inner edge, psi w = pressure load, psi W = Uniform load along inner edge, lb t = disc thickness, in. m = reciprocal of Poisson's ratio a = radius of disc, in. b = radius of disc hole, in.		
3. Pump discharge nozzle stress (pressure, bending, axial and torsional)	Pressure $P_p = \frac{SET}{R + 0.6t}$	P_P = 1,644 psi	1.5 S_m = 29,400 psi
A. Loads: Normal and upset condition Design pressure & temperature piping reactions during normal operation.	Bending $P_{eb} = \frac{C_b F_b S}{G_b}$	P_{eb} = 7,232 psi	
B. Combined stress limit: 1.5 S_m per ASME Code for Pumps and Valves for Nuclear Power Class 1.	Axial $P_{ed} = \frac{F_d S}{G_d}$	P_{ed} = 3,605 psi	
	Torsional $P_{et} = \frac{2F_b S}{G_t}$	P_{et} = 7,233 psi	
	Combined $P_c = P_P + P_{eb} + P_{ed} + P_{et}$	P_c = 19,714 psi	

Table 3.9-2i

Recirculation Pump[a] (Continued)

	Criteria	Method of Analysis	Analytical Results	Allow, Stress, or Actual Thickness
4.	Cover and seal flange bolt areas	Bolting loads, areas and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Appendix II.	Cover Flange Bolts $S_{act,}$ = 16,772 psi A_m = 118.8 in.^{2}	$S_{All,}$ = 25,325 psi $A_{min.}$ = 78.6 in.^{2}
	A. Loads: Normal and upset condition Design pressure and temperature Design gasket load			
	B. Bolting stress limits Allowable working stress per ASME Section III			
5.	Cover clamp flange thickness	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Appendix II.	Flange Thickness and Stress	
	A. Loads: Normal and upset condition Design pressure and temperature. Design gasket load Design bolting load		t = 7.64 in. $S_{Act.}$ = 12,497 psi	$t_{Act.}$ = 8-3/8 in. S_{Allow} = 15,000 psi
	B. Tangential flange stress limit Allowable working stress per ASME Sect. III			
6.	Seal compartment wall thickness	$t = \frac{PR}{SE - 0.6P} + C$	t = 1.071 in.	S_m = 15,075 psi $t_{Act.}$ = 1.854 in.
	A. Loads: Normal and upset condition Design pressure and temperature	where: t = minimum required. thickness, in. P = design pressure, psig R = max. internal radius, in. S = allowable working stress, psi		
	B. Primary membrane stress limit Allowable working stress per ASME Section III.	E = joint efficiency C = corrosion allowance, in.		

Table 3.9-2i

Recirculation Pump[a] (Continued)

	Criteria	Method of Analysis	Analytical Results	Allow, Stress, or Actual Thickness
7.	<u>Seal gland retainer</u>	$S_s = \frac{W}{W_{dt}}$	S_s = 5,486 psi	S_s = 9,480 psi
	A. <u>Loads</u>: <u>Normal and upset condition</u> Design pressure and temperature	W = load imposed d = diameter at shear resistance t = thickness at shear resistance		
	B. Allowable working stress per ASME Code Section VIII.			
8.	<u>Shock suppressor lug combined stress</u>	Loads shall be applied in the normal direction simultaneously to determine tensile, shear and bending stresses in the brackets. Tensile, shear, and bending stresses shall be combined to determine maximum combined stresses.	Combined stress (bending plus tensile)	S_m = 19,435 psi S_Y = 21,600 psi
	A. <u>Loads</u>: SSE + hydrodynamic force = 1.72 g horizontal 1.8 g vertical		Lug #1 S_C = 21,430 psi Lug #2 S_C = 12,070 psi Lug #3 S_C = 15,540 psi	S_C = 64,800 psi
	B. <u>Combined stress limits</u> Yield stress per ASME Section III			
9.	<u>Hanger Bracket Combined Stress</u>	Bracket vertical loads shall be determined by summing the equipment and fluid weights and vertical seismic forces.	S_C = 5,520 psi	S_m = 12,600 psi
	A. <u>Loads</u>: Flooded weight of equipment SSE hydrodynamic force = 1.4 g	Load = (W_B + W_C + W_D).33 <u>Note</u>: The multiplier (.33) is added as a safety factor specified on the Purchase Part Drawing.		
	B. <u>Combined stress limit</u> Yield stress per ASME Section VIII	W_B = weight of motor W_C = weight of motor mount W_D = weight of pump case		

Table 3.9-2i

Recirculation Pump[a] (Continued)

Criteria	Method of Analysis	Analytical Results	Allow, Stress, or Actual Thickness
10. Stresses due to seismic loads	The flooded pump-motor assembly shall be analyzed as a free body supported by constant support hangers from the pump brackets. Horizontal and vertical seismic forces shall be applied at mass center of assembly and equilibrium reactions shall be determined for the	Motor bolt tensile stress:	
A. Combined stress limit Limit stress per ASME Section VIII	motor and pump brackets. Load sheer and moment diagrams shall be constructed using live loads and calculated snubber reactions. Combined bending, tension, and shear stresses shall be determined for each major component of the assembly including motor support barrel, bolting, and pump casing. The maximum combined tensile stress in the cover bolting shall be calculated using tensile stresses	$S_{Act.}$ = 19,703 psi Pump cover bolt tensile stress: $S_{Act.}$ = 18,542 psi	$S_{All.}$ = 45,000 psi $S_{All.}$ = 38,062 psi
B. Loads: Operating pressure and temperature SSE + Hydrodynamic = 1.72g Horizontal 1.8g vertical	determined from loading diagram plus tensile stress from operating pressure.	Motor support barrel lugs combined stress: $S_{Act.}$ = 8,327 psi	 $S_{Act.}$ = 12,600 psi

[a] This pump constructed to the requirements of the 1971 Edition of Section III, however, the 1971 Edition stated that requirements for pumps were in course of preparation and that until they were issued any method which had been shown to be satisfactory could be used. Consequently, several different references have been used.

Table 3.9-2j

Reactor Recirculation System Gate Valves, 24 in. Discharge Structural and Mechanical Loading Criteria[a]

	Discharge Valve	Design Procedure	Required Design Value	Actual Design Value
1.0	<u>Body and Bonnet</u>			
1.1	Loads: Design pressure, design	Vendor's design calculation	1,675 psi 575°F	1,675 psi 575°F
1.2	Pressure rating, psi	Used NB-3543, Table NB-3531-5, and NB-3531-6 of Section III	$P_r = 908$ psi	$P_r = 908$ psi
1.3	Minimum wall thickness, in.	Used NB-3543 and Table NB-3542-1	$t_m \geq 2.33$ in.	$t_m = 2.375$ in.
1.4	Primary membrane stress, psi	Used NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	$P_m = 11,850$ psi
1.5	Secondary stress due to pipe reaction	Used NB-3545.2(b)	1.5 S_m = 29,400 psi	$P_{ed} = 6,700$ psi $P_{eb} = 13,600$ psi
1.6	Primary plus secondary stress due to internal pressure	Used NB-3545-2(a)	See 1.8 below	Qp = 23,930 psi
1.7	Thermal secondary stress	Used NB-3545-2(c)	See 1.8 below	$Q_{T1} = 3,000$ psi $Q_{T2} = 1,220$ psi $Q_{T3} = 1,405$ psi
1.8	Sum of primary plus secondary stress	Used NB-3545.2	$S_n = Q_p + P_{ed} + 2Q_{T2}$ $S_n = \leq 35_m (500^\circ F) = 58,800$ psi	$S_n = 33,070$ psi
1.9	Fatigue requirements	Used NB-3545.3	$N_a \geq 2,000$ cycles	Na > 10^5 cycles
1.10	Cyclic rating	Used NB-3550	$I_t \leq 1.0$	$I_t = .0036$

Table 3.9-2j

Reactor Recirculation System Gate Valves, 24 in. Discharge Structural and Mechanical Loading Criteria[a] (Continued)

	Discharge Valve	Design Procedure	Required Design Value	Actual Design Value
2.0	<u>Body to bonnet bolting</u>			
2.1	Loads: design pressure and temperature, gasket loads, stem operational load, seismic load (design basis earthquake)	Used NB-3546.1, NB-3647.1 and Section VIII		
2.2	Bolt area	Used NB-3546.1, NB-3647.1, and Section VIII	$A_b \geq 42.79$ in.^{2} $S_b \leq 27{,}975$ psi	$A_b = 53.04$ in.^{3} $S_b = 22{,}800$ psi
2.3	Body bonnet flange stresses	Used NB-3546.1, NB-3647.1, and Section VIII		
2.3.1	Operating condition	Used NB-3546.1, NB-3647.1, and Section VIII	$S_H \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_R \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_T \leq 1.5\ S_m$ (575°F) = 28,837 psi	S_H = 16,777 psi S_R = 4,447 psi S_T = 5,306 psi
2.3.2	Gasket seating condition	Used NB-3546.1, NB-3657.1, and Section VIII	$S_H \leq 1.5\ S_m$ (150°F) = 30,000 psi $S_R \leq 1.5\ S_m$ (150°F) = 30,000 psi $S_T \leq 1.5\ S_m$ (150°F) = 30,000 psi	S_H = 22,531 psi S_R = 6,550 psi S_T = 7,811 psi
3.0	<u>Stresses in stem</u>			
3.1	Loads: operator thrust and torque			
3.2	Buckling of stem	Calculate slenderness ratio. If greater than 30, calculate allowable loads from Rankine's Formula using safety factor of 4.	Maximum allowable load = 102,050 lb	Slenderness ratio = 56 Actual thrust on stem = 51,000 lb No buckling

Table 3.9-2j

Reactor Recirculation System Gate Valves, 24 in. Discharge
Structural and Mechanical Loading Criteria[a] (Continued)

	Discharge Valve	Design Procedure	Required Design Value	Actual Design Value
3.3	Stem thrust stress	Calculate stress due to operator thrust in critical cross-section.	$S_{T,C} \leq S_m$ = 43,950 psi	$S_{T,C}$ = 15,045 psi
3.4	Stem torque	Calculate shear stress due to operator torque in critical cross-section.	$S_C \leq .6S_m$ = 16,785 psi	S_C = 10,748 psi
4.0	<u>Disc analysis</u>			
4.1	Loads: maximum differential pressure			
4.2	Maximum stress	Calculate maximum stress according to R.J. Roark, "Formulas for Stress and Strain"	$S_{max} \leq 1.5\ S_m$ (575°F) = 23,700 psi	S_{max} = 15,294 psi
5.0	<u>Yoke and yoke connections</u>			
5.1	Loads: stem operational load	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods.		
5.2	Tensile stress in yoke legs bolts		$S_{max} \leq S_m$ = 25,000 psi	Max. stress = 15,064 psi
5.3	Bending stress of yoke legs		$S_{max} \leq S_m$ = 17,500 psi	S_b = 13,701 psi

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Table 3.9-2j

Reactor Recirculation System Gate Valves, 24 in. Discharge
Structural and Mechanical Loading Criteria[a] (Continued)

	Suction Valve	Design Procedure	Required Design Value	Actual Design Value
1.0	Body and bonnet			
1.1	Loads: design pressure, design temperature pipe reaction thermal effects	Kinder's Design calculations	1,250 psig 575°F	1,250 psig 575°F
1.2	Pressure rating, psi	Used NB-3543, Table NB-3531-4, and NB-3531-5 of Section III	P_r = 678 psi	P_r = 678 psi
1.3	Minimum wall thickness	Used NB-3543 and Table NB-3542-1	$t_m \geq$ 1.747 in.	t_m = 1.75 in.
1.4	Primary membrane stress	Used NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	P_m = 11,075 psi
1.5	Secondary stress due to pipe reaction	Used ASME Section III Paragraph NB-3545.2(b) (S = 30,000 psi)	$P_{ed} \leq 1.5\ S_m$ = 29,400 psi $P_{eb} \leq 1.5\ S_m$ = 29,400 psi $P_{et} \leq 1.5\ S_m$ = 29,400 psi	P_{ed} = 6,400 psi P_{eb} = 13,150 psi P_{et} = 13,100 psi
1.6	Primary plus secondary stress due to internal pressure	Used ASME Section III Paragraph NB-3545.2(a)	See 1.8 below	Q_P = 21,730 psi
1.7	Thermal secondary stress	Used ASME Section III Paragraph NB-3545.2	See 1.8 below	Q_{T1} = 2,000 psi Q_{T2} = 830 psi Q_{T3} = 960 psi
1.8	Sum of primary plus secondary stress	Used ASME Section III Paragraph NB-3545.2	$S_n = Q_P + P_{ed} + 2Q_{T2}$ $S_n \leq 3S_m$ (500°F) = 58,800 psi	S_n = 29,790
1.9	Fatigue requirements	Used ASME Section III Paragraph NB-3545.3	$N_a \geq$ 2,000 cycles	$N_a = 10^6$ cycles
1.10	Cyclic rating	Used ASME Section III Paragraph NB-3550	I ≤ 1.0	I = .003

Table 3.9-2j

Reactor Recirculation System Gate Valves, 24 in. Discharge
Structural and Mechanical Loading Criteria[a] (Continued)

	Suction Valve	Design Procedure	Required Design Value	Actual Design Value
2.0	<u>Body to bonnet bolting</u>			
2.1	Loads: design pressure & temperature gasket loads stem operational load (design basis earthquake)	Used NB-3546.1 and NB-3657.1		
2.2	Bolt area	Used NB-3546.1 and NB-3657.1	$A_b \geq 31.53$ in.^{2} $S_b \leq 27,975$ psi	$A_b = 47.73$ in.^{2} $S_b = 18,480$ psi
2.3	Body flange stresses	Used NB-3546.1 and NB-3657.1		
2.3.1	Operating condition	Used NB-3546.1 and NB-3657.1	$S_H \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_R \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_T \leq 1.5\ S_m$ (575°F) = 28,837 psi	S_H = 17,170 psi S_R = 5,735 psi S_T = 6,120 psi
2.3.3	Gasket seating condition	Used NB-3546.1 and NB-3657.1	$S_H \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30,000 psi	S_H = 25,245 psi S_R = 9,425 psi S_T = 10,140 psi
3.0	<u>Stress in stem</u>			
3.1	Load operator thrust and torque			
3.2	Buckling of stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 4.	Maximum allowable load = 100,000 lb	Slenderness ratio = 42.5. Actual load on stem = 18,100 lb. No buckling.

Table 3.9-2j

Reactor Recirculation System Gate Valves, 24 in. Discharge
Structural and Mechanical Loading Criteria[a] (Continued)

	Suction Valve	Design Procedure	Required Design Value	Actual Design Value
3.3	Stem thrust stress	Calculate stress due to operator thrust in critical cross-section	$S_{T,C} \leq S_m$ = 30,800 psi	$S_{T,C}$ = 5,340 psi
3.4	Stem torque stress	Calculate shear stress due to operator torque in critical cross-section	$S_C \leq .6\ S_m$ 18,480 psi	S_C = 4,155 psi
4.0	<u>Disc analysis</u>			
4.1	Loads: maximum differential pressure 50 psi			
4.2	Maximum stress in the disc	Calculate maximum stress according to Table 10 of Roark's "Formula for Stress and Strain"	$S_{max} \leq 1.5\ S_m$ (575°F) = 23,700 psi	S_{max} = 15,529 psi
5.0	<u>Yoke and yoke connections</u>			
5.1	Loads: stem operational load	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods		
5.2	Tensile stress in yoke legs bolts		$S_{max} \leq S_m$ = 25,000 psi	Max Stress = 11,060 psi
5.3	Bending stress at yoke legs		$S_{max} \leq S_m$ = 17,500 psi	S_b = 14,740 psi

[a] This table is for design criteria only. The calculated loads are less than allowable loads. See Table 3.9-2e.

Table 3.9-2k

Not Used

Table 3.9-2L

Standby Liquid Control Pump

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Based on ASME B&PV Code, Section III				
Pressure Boundary parts:				
1) Fluid Cylinder - SA 182-F304 S_Y = 30,000 psi				
2) Discharge valve stop, cylinder head extension, and stuffing box SA 479-304, S_Y = 30,000 psi				
3) Discharge valve cover, cylinder head and stuffing box flange plate, SA-240-304, S_Y = 30,000 psi				
4) Stuffing box gland, SA-564-630, S_Y = 90,000 psi				
5) Studs, SA 193-B7, S_Y = 105,000 psi				
6) Dowel pins[a] alignment, SAE 4140, S_A = 117,000 psi				
7) Studs, cylinder tie, SA 193-B7, S_A = 25,000 psi				
8) Pump holddown bolts, SAE GR.8, T_A = 15,000 psi, Q_A = 12,000 psi				
9) Power Frame, foot area, cast iron, S_A = 7,500 psi				
10) Motor holddown bolts, SAE GR.1 T_A = 12,000 psi, Q_A = 15,000 psi				
11) Motor frame foot area, cast iron, S_A = 7,500 psi				
<u>Normal and upset condition loads</u>:	1. Fluid Cylinder	General membrane	17,800	
1. Design pressure	2. Discharge valve stop	General membrane	17,800	(b)
2. Design temperature	3. Cylinder head extension	General membrane	17,800	
3. Operating basis earthquake	4. Discharge valve cover	General membrane	17,800	
4. Nozzle loads[c]	5. Cylinder head	General membrane	17,800	
5. Safety/relief valve discharge	6. Stuffing box flange plate	General membrane	17,800	
6. Dead weight	7. Stuffing box gland	General membrane	35,000	
7. Thermal expansion	8. Cylinder head studs	Tensile	25,000	
	9. Stuffing box studs	Tensile	25,000	

Table 3.9-2L

Standby Liquid Control Pump (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Emergency or faulted conditions	1. Fluid cylinder	General membrane	17,295 psi	3,640 psi
1. Design pressure	2. Discharge valve stop	General membrane	21,360 psi	13,600 psi
2. Design temperature	3. Cylinder head extension	General membrane	21,360 psi	13,600 psi
3. Weight of structure	4. Discharge valve cover	General membrane	21,360 psi	8,150 psi
4. Thermal expansion	5. Cylinder head	General membrane	21,360 psi	8,150 psi
5. Safe shutdown earthquake	6. Stuffing box flange plate	General membrane	30,000 psi	10,690 psi
6. Safety/relief valve discharge load	7. Stuffing box gland	General membrane	42,000 psi	11,420 psi
7. Loss-of-coolant accident	8. Cylinder head studs	Tensile	25,000 psi	18,620 psi
8. Nozzle loads	9. Dowel pins[a]	Shear only[a]	23,400 psi	19,400 psi
	10. Studs, cylinder tie	Tensile[a]	25,000 psi	24,750 psi
	11. Pump holddown bolts	Shear	12,000 psi	9,050 psi
	12. Pump holddown bolts	Tensile	15,000 psi	14,026 psi
	13. Power frame-foot area	Shear	15,000 psi	1850 psi
	14. Power frame-foot area	Tensile	15,000 psi	11,390 psi
	15. Motor holddown bolts	Shear	12,000 psi	3,480 psi
	16. Motor holddown bolts	Tensile	15,000 psi	6,315 psi
	17. Motor frame-foot	Shear	7,500 psi	2,550 psi
Faulted condition	SLC Pump Assembly	Acceleration	1.75g vertical	0.41g
Dynamic loads			1.75g horizontal	0.73g
1. Safe shutdown earthquake				
2. Safety/relief valve				
3. Loss-of-coolant accident				

Table 3.9-2L

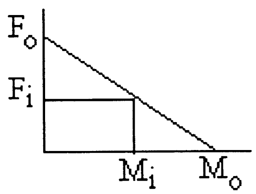
Standby Liquid Control Pump (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Loads (lb, ft-lb)	Calculated Loads (lb, ft-lb)

<u>Nozzle load definition</u>

Units: Forces - lb
Moments - ft-lb

Allowable combination of forces and moments are as follows:



$$\frac{F_i}{F_o} + \frac{M_i}{M_o} \leq 1$$

where:

F_i = The largest absolute value of the three actual external orthogonal forces (F_x, F_y, F_z) that may be imposed by the interface pipe, and,

M_i = The largest absolute value of the three actual internal orthogonal moments ((M_x, M_y, M_z) permitted from the pipe when they are combined simultaneously for a specific condition.

Criteria/Loading	Component	Limiting Stress Type	Allowable Loads (lb, ft-lb)	Calculated Loads (lb, ft-lb)
<u>Normal and upset condition loads</u>: 1. Design pressure 2. Design temperature 3. Dead weight 4. Thermal expansion 5. Operating basis earthquake		F_o = Allowable value of F_i when all moments are zero. M_o = Allowable of M_i when all forces are zero.	Suction: F_o = 770 M_o = 490	Less than or equal to allowable
			Discharge: F_o = 370 M_o = 110	Acceptable per GE review

Table 3.9-2L

Standby Liquid Control Pump (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Loads (lb, ft-lb)	Calculated Loads (lb, ft-lb)
Emergency or faulted condition loads:				
			Suction:	Less than or equal to allowable
1. Design pressure			$F_o = 920$	
2. Design temperature			$M_o = 590$	
3. Dead weight				
4. Thermal expansion				
5. Safe shutdown earthquake			Discharge:	Acceptable per GE review
			$F_o = 440$	
			$M_o = 130$	

[a] Dowel pins take all shear.

[b] Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition loads, therefore, the normal and upset condition is not evaluated.

[c] Nozzle loads produce shear loads only.

Note: Operability: The sum of the plunger and rod assembly, pounds mass times 1.75, acceleration is much less than the thrust loads encountered during normal operating conditions. Therefore, the loads during the faulted condition have no significant effect on pump operability.

Table 3.9-2m

Standby Liquid Control Tank

Criteria	Method of Analysis	Allowable Stress (psi) or Acceleration (g) or Minimum Thickness Required (in.)	Calculated Stress (psi) or Acceleration (g) or Actual Thickness (in.)
1. Shell thickness	Minimum thickness	0.010 in.	0.25 in.
Loads: normal and upset	Cylindrical shell		
Design pressure and temperature			
Stress limit			
Allowable working stress per ASME Section III			
2. Shell stress	Loads will not produce excessive tensile or compressive		
Loads: emergency	(buckling) stresses.		
Design basis earthquake (SSE) nozzle load	Brownell & Young "Process Equipment Design" $t = \frac{PR}{SE - .6P}$		
Stress limit		18,000 psi	3,314 psi
ASME Section III	t = Minimum required thickness, in.		
1/3 yield	P = Design pressure, psi		
	R = Shell inside radius, in.	Tensile (bolts)	
	S = Allowable stress, psi		
	E = Joint efficiency	10,000 psi	8104 psi
3. Dynamic loads	Equivalent static	1.75g horizontal	0.73g
		1.75g vertical	0.73g
Standby liquid control tank			
Safe shutdown earthquake			
Safety/relief valve			
Loss-of-coolant accident			

Table 3.9-2n

Emergency Core Cooling System Pumps
Residual Heat Removal Pump

Location	Loading Condition	Criteria	Calculated Stress (psi)	Allowable Stress (psi)
Suction barrel shell	<u>Faulted condition</u> Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	6,399	20,400
Stuffing box pipe	Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	13,630	18,000
Nozzle shell inter section	<u>Faulted condition</u> Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	19,029	34,650
Discharge elbow or suction pipe (max. moment location)	<u>Faulted condition</u> Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	10,643	21,600
Motor stand	<u>Faulted condition</u> Static loads Dynamic loads	Buckling Loads and Stresses per ASME Section III	2,996	15,200
Motor Mounting Flange Bolting	<u>Faulted condition</u> Static Loads Dynamic Loads	Bolting Loads and Stresses per ASME, Section III Subsection NF	6,081	17,500

Table 3.9-2n

Emergency Core Cooling System Pumps (Continued)
Low-Pressure Core Spray Pump

Location	Loading Condition	Criteria	Calculated Stress (psi) or Calculated Stress Ratio	Allowable Stress (psi) or Allowable Stress Ratio
Suction barrel shell	<u>Faulted condition</u> Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	9,037 psi	21,000 psi
Stuffing box pipe	Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	11,355 psi	15,000 psi
Nozzle shell inter section	<u>Faulted condition</u> Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	12,170 psi	34,650 psi
Discharge elbow or suction pipe (max.)	<u>Faulted condition</u> Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	8,758 psi	17,500 psi
Motor stand	<u>Faulted condition</u> Static loads Dynamic loads	Buckling Loads and Stresses per ASME Section III	9,530 psi	15,200 psi
Motor bolting	<u>Faulted condition</u> Static loads Dynamic loads Nozzle loads	Bolting Loads and Stresses per ASME, Section III Subsection NF	4,824 psi	17,500 psi

Table 3.9-2n

Emergency Core Cooling System Pumps (Continued)
High-Pressure Core Spray Pump

Location	Loading Condition	Criteria	Calculated Stress (psi) or Actual Thickness (in.)	Allowable Stress (psi) or Minimum Thickness (in.)
Suction barrel shell	Faulted condition Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	5,115 psi	21,000 psi
Stuffing box pipe	Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	12,851 psi	18,000 psi
Nozzle shell intersection	Faulted condition Design pressure Static loads Dynamic loads Nozzle loads	ASME Boiler and Pressure Vessel Code, Section III	13,533 psi	34,650 psi
Discharge elbow or suction pipe (max.)	Faulted condition Design pressure Static loads Dynamic loads	ASME Boiler and Pressure Vessel Code, Section III	12,499 psi	21,000 psi
Motor stand	Faulted condition Static loads Dynamic loads Nozzle loads	Buckling Loads and Stresses per ASME Section III	6,840 psi	15,200 psi
Motor bolting	Faulted condition Static loads Dynamic loads Nozzle loads	Bolting Loads and Stresses per ASME, Section III Subsection NF	3,821 psi	21,000 psi

Table 3.9-2o

Residual Heat Removal Heat Exchanger

	Loading/Component	Criteria/Location	Allowable Stress or Minimum Thickness Required	Actual Stress or Actual Thickness
1.	Closure bolting	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections", ASME Section III, App. XI		
	Loads: normal and upset			
	Design pressure and temperature			
	Design gasket load			
	Bolting stress limit	a. Shell to tube sheet bolts	25,000 psi	24,950 psi
	Allowable working stress per ASME Section III	b. Channel cover bolts	25,000 psi	24,390 psi
2.	Wall thickness	Shell side ASME Section III Class 2 and TEMA Class C		
	Loads: normal and upset			
	Design pressure and temperature	Tube side ASME Section III Class 3 and TEMA Class C		
	Stress limit			
	ASME Section III	a Shell	0.896 in.	1.0 in.
		b. Shell cover	0.885 in.	0.885 in.
		c. Channel	0.924 in.	1.0 in.
		d. Tubes	0.0515 in.	0.054 in.
		e. Channel cover	8.11 in.	8.12 in.
		f. Tube sheet	7.08 in.	7.12 in.

Table 3.9-2o

Residual Heat Removal Heat Exchanger (Continued)

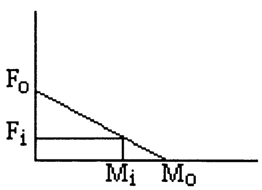
Loading/Component	Criteria/Location	Allowable Stress (psi)	Actual Stress (psi)
3. Nozzle	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits.	(a) (b)	
Loads: faulted			
Design pressure and temperature			
Dead weight, thermal expansion,	Primary Stress Smaller of 0.75 S_u or 2.4 S_m		
Safe shutdown earthquake SRV, LOCA	ASME Section III allowable.		
4. Support brackets and attachment welds	Stress allowables as per ASME Section III Subsection NF (upset condition).		
Loads: faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake, safety/relief valve	Lower bracket welds - Principal stress	14,438	7,400
5. Anchor bolts	Stress allowable as per ASME III, Appendix XVII		
Loads: faulted			
Design pressure and temperature, dead weight, nozzle loads, SSE, S/RV	Lower support bolting - Tension - Shear	 29,000 11,990	 14,421 2,592

Table 3.9-2o

Residual Heat Removal Heat Exchanger (Continued)

Loading/Component	Criteria/Location	Allowable Stress (psi)	Actual Stress (psi)
6. Shell adjacent to support brackets	Shell stress allowables as per ASME Section III Subsection NC (Upset Conditions).		
Loads: faulted			
Design pressure and temperature, dead weight, nozzle loads, SSE, SRV	a. Maximum principal stress adjacent to upper support	28,875	24,825
	b. Maximum principal stress adjacent to lower support	28,875	27,053
7. Shell	Stress allowable as per ASME Section III Subsection NC (upset condition)		
Loads: faulted			
Design pressure and temperature, dead weight, nozzle loads, SSE, SRV	Principal	19,250	11,908

[a] Maximum allowable piping load combinations for faulted conditions (including SSE) do not exceed the following relationship for each nozzle:



$$\left|\frac{F_i}{F_o}\right| + \left|\frac{M_i}{M_o}\right| \le 1$$

where:

F_i = The largest of the three actual external orthogonal forces (F_x, F_y, and F_z).

M_i = The largest of the three actual external orthogonal moments (M_x, M_y, and M_z) for the same reference coordinates.

F_o = The allowable value of F_i when all moments are zero.

M_o = The allowable value of M_i when all forces are zero.

Table 3.9-2o

Residual Heat Removal Heat Exchanger (Continued)

One coordinate axis must be the nozzle centerline. Another coordinate axis must be parallel to the heat exchanger centerline except where the heat exchanger centerline is parallel to the nozzle centerline. In this case, the coordinate axis must be orthogonal to the nozzle centerline and at 0°-180° or 90°-270° azimuths.

[b] Allowable limits (design basis)

		N1	N2	N3	N4
F_x	=	15,500 lb	15,500 lb	15,500 lb	15,500 lb
F_y	=	15,500 lb	15,500 lb	15,500 lb	15,500 lb
F_z	=	15,500 lb	15,500 lb	15,500 lb	15,500 lb
M_x	=	60,000 ft-lb	60,000 ft-lb	60,000 ft-lb	60,000 ft-lb
M_y	=	60,000 ft-lb	60,000 ft-lb	60,000 ft-lb	60,000 ft-lb
M_z	=	60,000 ft-lb	60,000 ft-lb	60,000 ft-lb	60,000 ft-lb

Table 3.9-2p

Reactor Water Cleanup Pump

Following is a summary of the design calculations on the RWCU Pump:

Part(ASME Code Calculation)	Calculated Stress (psi)	Allowable Stress (psi)
Pump part		
Casing wall	10,476	12,814
Suction wall	5,112	12,814
Discharge wall	3,337	12,814
Cover bolting	23,032	30,750
Seal gland bolting	26,532	30,750
Pedestal bolt (shear)	18,015	44,000
Motor part		
Motor foot bolts (shear)	3787	44,000

Table 3.9-2q

Not Used

Table 3.9-2r

Reactor Core Isolation Cooling Pump

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Pressure boundary stress limits of the various components for the RCIC pump assembly are based on the ASME B&PV Code Section III, for pressure boundary parts @ 140°F.				
1. Forged barrel, SA105 GR.II $S_Y = 36,000$ psi				
2. End cover plates, SA105 GR.II $S_Y = 36,000$ psi				
3. Nozzle connections, SA105 GR.II $S_Y = 36,000$ psi				
4. Aligning pins, SA105 GR.II $S_Y = 36,000$ psi				
5. Closure bolting, SA193-B7 $S_Y = 107,000$ psi				
6. Pump holddown bolting, SA325 $S_Y = 77,000$ psi				
7. Taper pins, SA108 GR. B1112, $S_Y = 75,000$ psi				
<u>Normal and upset condition loads</u>:				
1. Design pressure	1. Forged barrel	General membrane		
2. Design temperature	2. End cover	General membrane		
3. Operating basis earthquake	(Suction)	General membrane		
4. Suction nozzle loads	3. End cover	General membrane	(a)	
5. Discharge nozzle loads	(Discharge)	Shear		
	4. Nozzle reinforcement	Tensile shear		
	5. Alignment pin	Tensile		
	6. Closure bolting			
	7. Taper pins			
	8. Pump holddown bolts			

Table 3.9-2r

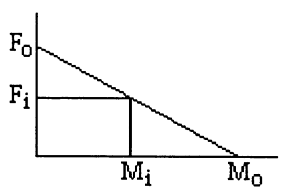
Reactor Core Isolation Cooling Pump (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Emergency or faulted condition loads:[b]				
1. Design pressure	1. Pump holddown bolts	Tension	12,800	12,646
2. Design temperature	2. Taper pins (bearing housing)	Shear	15,000	2,230
3. Safe shutdown earthquake	3. Alignment pin	Shear	17,500	2,680
4. Suction nozzle loads	4. Pump outer case	General membrane	17,500	7,052
5. Discharge nozzle loads	5. Discharge nozzle	General membrane	26,250	7,855

Nozzle load definitions:

Units: Forces - lb
Moments - ft-lb

The allowable combinations of forces and moments are as follows:



$$\frac{F_i}{F_o} + \frac{M_i}{M_o} \leq 1$$

where:

F_i = Largest absolute value of the three actual external orthogonal forces (F_x, F_y, F_z) that maybe imposed by the interface pipe and,

M_i = Largest absolute value of the three actual external orthogonal forces (M_x, M_y, M_z) permitted from the interface pipe when they are combined simultaneously for a specific condition.

Table 3.9-2r

Reactor Core Isolation Cooling Pump (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Loads (lb. ft-lb)	Calculated Loads (lb. ft-lb)
Normal and upset condition loads: 1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Operating basis earthquake		F_o = Allowable value of F_i when all moments are zero M_o = Allowable value of M_i when all forces are zero	Suction: F_o = 1940 M_o = 2460 Discharge: F_o = 3715 M_o = 4330	Less than or equal to allowable Less than or equal to allowable
Emergency or faulted condition loads: 1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Safe shutdown earthquake			Suction: F_o = 2325 M_o = 2950 Discharge: F_o = 4450 M_o = 5200	Less than or equal to allowable Less than or equal to allowable
where: F_i = The largest absolute value of the three actual external orthogonal forces (F_x, F_y, F_z) that may be imposed by the interface pipe and	Emergency or faulted condition loads: Design pressure and temperature Dead weight & thermal expansion Safe shutdown earthquake			

Table 3.9-2r

Reactor Core Isolation Cooling Pump (Continued)

Criteria/Loading		Component	Limiting Stress Type	Allowable Loads (lb. ft-lb)	Calculated Loads (lb. ft-lb)
M_i =	The largest absolute value of the three actual external orthogonal moments (M_x, M_y, M_z) permitted from the interface pipe when they are combined simultaneously for a specific condition.				

[a] Calculated stresses for emergency or faulted condition are less than the allowable for normal plus upset condition, therefore the normal and upset condition is not evaluated.

[b] Per Regulatory Guide 1.48, the allowable stresses under faulted condition are 1.2 times those under the upset conditions.

Note: Operability static analysis for emergency or faulted condition show that the maximum shaft deflection is 0.0038 in. with 0.0055 in. allowable, shaft stresses are 5975 psi with 17,200 psi allowable, and bearing loads of, drive end 376 lb with 7670 lb allowable and thrust end 1323 lb with 17,200 lb allowable.

Table 3.9-2s

Reactor Refueling and Servicing Equipment[a]
New Fuel Storage Racks

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The allowable stress is based on Part 1 of AISC Manual for type ASTM B221, 6061-T6 Alum Alloy $F_u = 38{,}000$ psi $F_Y = 35{,}000$ psi				
For normal condition: $S_{Limit} = 0.66\ F_Y$	Normal operating loads	Axial load plus bending	23,100	15,230[a]
For emergency condition[bc]: $S_{Limit} = 0.88\ F_Y$	Normal operating loads Operating basis earthquake Safety/relief valve Loss-of-coolant accident	Axial load plus bending	30,800	<30,800[a]
For faulted condition[bc]: $S_{Limit} = 0.88\ F_Y$	Normal operating loads Operating basis earthquake Safety/relief valve Loss-of-coolant accident	Axial load plus bending	30,800	<30.800[a]

Table 3.9-2s

Reactor Refueling and Servicing Equipment[a] (Continued)
Refueling Platform

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The allowable axial load stress is based on AISC Part 5, Section 1.5 for type ASTM A36 Structural Steel F_u = 58,000 psi F_y = 36,000 psi				
For normal condition: $S_{Limit} = 0.66\ F_Y$	Static	Axial load plus bending	23,760	3,597
For emergency condition: $S_{Limit} = 0.88\ F_Y$	Normal operating loads Operating basis earthquake Safety/relief valve	Axial load plus bending	31,680	32,040[d]
For faulted condition: $S_{Limit} = 0.7\ F_u$	Normal operating loads Safe shutdown earthquake Safety/relief valve Loss-of-coolant accident	Axial load plus bending	40,600	26,064

Table 3.9-2s

Reactor Refueling and Servicing Equipment[a] (Continued)
Fuel Preparation Machine

Acceptance Criteria	Loading	Primary Stress Type	Allowable Acceleration[e]	Calculated Acceleration[e]
The allowable axial load stresses are based on AISC Code; ASME Code Section III for 320 S.S. Side Plates 17-4 PH and 17-7 PH Rollers. F_Y = 30,000 psi F_u = 75,000 psi $S_{m\,200}$ = 17,800 psi				
For normal condition:	Static	Axial load	1.56	1.00
For upset/emergency condition:	Normal operating loads Operating basis earthquake Safety/relief valve	Axial load	2.11	1.24
For faulted condition:	Normal operating loads Safe shutdown earthquake Safety/relief valve Loss-of-coolant accident	Axial load	4.61	1.40

[a] Calculated stresses are recorded in GE document 386HA625, “Load Combinations and Acceptance Criteria for Reactor Refueling and Servicing Equipment.”

[b] A one-third margin is added to the normal limit to obtain the upset limit per AISC, 7th Edition, Part 1, Section 1.5.6.

[c] The upset allowable is used to evaluate emergency and faulted conditions for conservatism.

[d] Due to structural response and damping effects, the OBE produces greater stresses at this location than does the SSE. One member yields in the plastic material range but does not impair the functional use of the equipment. Operability assurance is demonstrated by analysis.

[e] Equivalent g-load. Operability assurance is demonstrated by analysis.

Table 3.9-2t

Not Used

Table 3.9-2u

Control Rod Drive (Indicator Tube)

Acceptance Criteria	Loading	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code Section III.				
For normal and upset condition: S_{allow} = 1.5 x S_m	1. Normal loads[a] 2. Scram with OBE and SRV	($P_M + P_B + Q$)[b] ($P_M + P_B$)[b]	51,700 25,860	47,100 24,728
For emergency condition: S_{allow} = 1.8 x S_m	1. Normal loads[a] 2. Chugging 3. Safety/relief valve 4. Scram	($P_M + P_B$)[b]	31,000	24,728[c]
For faulted condition: S_{allow} = 3.6 x S_m @ weld joint	1. Normal loads[a] 2. Scram 3. Safe shutdown earthquake 4. Safety/relief valve 5. Chugging	($P_M + P_B$)[b]	40,000	37,600

[a] Normal loads (include pressure, temperature, weight, and mechanical loads.

[b] P_M = Primary membrane stress, P_B = Primary bending stress,
Q = Secondary membrane and secondary bending stresses.

[c] Less severe than the upset condition $P_M + P_B$ calculated stress.

Table 3.9-2v

Control Rod Drive Housing

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary stress limit - The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Section III, Class I, for Type 304 stainless steel.				
For normal and upset condition: $S_{limit} = 1.0\ S_m$ $= 16{,}660$ psi @ 575°F	1. Design pressure 2. Stuck rod scram loads 3. Operational basis earthquake, with housing lateral support installed 4. Safety/relief valve	Maximum membrane stress intensity occurs at the tube weld near the center of the housing for normal, upset, emergency, and faulted conditions.	24,900	15,450
For faulted conditions: $S_{limit} = 29{,}880$	1. Design pressure 2. Stuck rod scram loads 3. Safe shutdown earthquake, with housing lateral support installed 4. Safety/relief valve 5. Loss-of-coolant accident	Membrane plus bending	29,880	18,180

Note: Emergency condition results are not shown because they are less than normal/upset dynamic loads. This occurs because the emergency condition includes primarily vertical accelerations which have less effect on this long vertical tube than the horizontal accelerations of the OBE loading in normal and upset.

Table 3.9-2w

Jet Pumps

Acceptance Criteria	Loading/Combinations	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary membrane plus bending stress based on ASME B&PV Code Section III Subsection NG				
For service levels A and B (normal and upset) condition: For Type 304 S.S. @ 550°F S_m = 16,800 psi S_{Limit} = 1.5 S_m psi	Normal loads[a] Operating basis earthquake Safety/relief valve	Primary membrane plus bending	50,700	17,640[b]
For service levels C (emergency) condition: For Type 304 S.S. @ 550°F S_m = 16,800 psi S_{Limit} = 2.25 S_m psi	Normal loads[a] Chugging Safety/relief valve	Primary membrane plus bending	37,800	15,505[b]
For service levels D (faulted) condition: For Type 304 S.S. @ 550°F S_m = 16,800 psi S_{Limit} = 3.6 S_m psi	Normal loads[c] Annulus pressurization Jet reaction Safe shutdown earthquake	Primary membrane plus bending	60,480	54,450[b]

[a] Design internal pressure, hydraulic and pressure reaction loads.

[b] Riser brace only. Stresses on other components are much lower.

[c] Design external pressure, hydraulic and pressure reaction loads.

Table 3.9-2x

Not Used

Table 3.9-2y

Low-Pressure Coolant Injection Coupling

Acceptance Criteria	Loading	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary membrane plus bending stress based on ASME B&PV Code Section III for Type 316L Stainless Steel.				
For service levels A and B (normal and upset) condition: $S_{limit} = 3\ S_m = 41{,}850$ psi	Normal loads Operating basis earthquake SRV_{ALL}	Primary and secondary membrane plus bending	41,850	13,455
For service levels C (emergency) condition: $S_{limit} = 2.25\ S_m = 31{,}400$ psi	Normal Loss-of-coolant accident (chugging) SRV_{ADS}	Primary membrane plus bending	31,400	22,938
For service levels D (faulted) condition: $S_{limit} = 3.6\ S_m = 50{,}220$ psi	Normal Safe shutdown earthquake Loss-of-coolant accident (annulus pressurization)	Primary membrane plus bending	50,220	41,329

Table 3.9-2z

Not Used

Table 3.9-2aa

Control Rod Guide Tube

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)[a]
Primary stress limit The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section III for type 304 stainless steel tubing..				
For normal and upset conditions: $S_{limit} = 1.5\ S_m = 1.5 \times 16{,}000$ $S_{limit} = 24{,}000$ psi	Applied loads: 1. Delta pressure force 2. Metal and water weight 3. Operating basis earthquake 4. Safety/relief valve	Applying vertical seismic plus dead weight, the maximum stress under normal and upset conditions occurs at the guide tube base. Primary membrane plus bending	24,000	8,189
For faulted conditions $S_{limit} = 2.4\ S_m = 2.4 \times 16{,}000$ $= 38{,}400$ psi	Applied loads: 1. Delta pressure force 2. Metal and water weight 3. Safe shutdown earthquake 4. Safety/relief valve 5. Local (chugging)	Applying vertical seismic plus dead weight, the maximum stress under faulted conditions occurs at the guide tube base. Primary membrane plus bending	38,400	13,169

[a] Because of different loading conditions, calculated stresses for emergency conditions are less severe than the normal and upset condition loads.

Table 3.9-2ab

Incore Housing

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary stress limit The allowable Primary Membrane Stress plus Bending Stress is based on the ASME Boiler and Pressure Vessel Code, Section III for Class 1 vessels for type 304, SA 213 stainless steel tubing.		Maximum membrane stress intensity occurs at the outer surface of the vessel penetration (including bending stresses)		
For normal and upset conditions: S_m = 16,660 psi at 575°F	1. Design pressure 2. Operating basis earthquake 3. Safety/relief valve		16,660	15,548
For faulted condition $S_{limit} = 2.4\ S_m = 2.4 \times 16,660$ $= 39,984$	1. Design pressure 2. Safe shutdown earthquake 3. LOCA (annulus pressure) 4. Jet reaction	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration (including bending stresses)	39,984	25,796

Table 3.9-2ac

Reactor Vessel Support Equipment
(i) Control Rod Drive Housing Support

Acceptance Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
Primary stress limit AISC specification for the design, fabrication and erection of structural steel for buildings	Faulted condition loads 1. Dead weight	Beams (top chord)	33,000 33,000	f_a = 12,200 f_a = 16,500
	2. Impact force from failure of a CRD housing	Beams (bottom chord)	33,000 33,000	f_a = 10,300 f_b = 11,700
For normal and upset condition $f_a = 0.60\ f_Y$ (tension) $f_b = 0.60\ f_Y$ (bending) $f_v = 0.40\ f_Y$ (shear) For faulted conditions: f_a limit = 1.5 f_a (tension) f_b limit = 1.5 f_b (bending) f_v limit = 1.5 f_v (shear) f_Y = Material yield strength	(Dead weight and earthquake loads are very small as compared to jet force)	Grid structure	41,500 27,500	f_b = 40,700 f_v = 11,100

Table 3.9-2ac

Reactor Vessel Support Equipment (Continued)
(ii) Reactor Pressure Vessel Stabilizer

Acceptance Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
Primary stress limit				
AISC specifications for the construction, fabrication, and erection of structural steel for buildings				
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads	Upset condition	Rod	84,000[a]	f_t = 54,000
	1. Spring preload	Bracket	22,000	f_b = 22,000
	2. Operating basis earthquake	Bracket	14,000	f_v = 4,600
For emergency conditions 1.5 x AISC allowable stresses	Emergency condition	Bracket	33,000	F_b = 24,400
	1. Spring preload	Bracket	21,000	f_b = 22,000
	2. Design basis earthquake	Rod	126,000[b]	f_v = 108,000
For faulted conditions Material yield strength	Faulted condition			
	1. Spring preload	Bracket	36,000	f_b = 26,000
	2. Design basis earthquake	Bracket	21,500	f_v = 11,330
	3. Jet reaction load	Rod	140,000	f_t = 132,000

[a] 0.6 x yield based on the AISC criterion for tension.

[b] 1.5 x normal and upset limit.

Table 3.9-2ac

Reactor Vessel Support Equipment (Continued)
(iii) Reactor Pressure Vessel Support (Bearing Plate)

Acceptance Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
Primary stress limit AISC specification for the design, fabrication and erection of structural steel for buildings				
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads	Normal and upset condition 1. Dead loads 2. Operating basis earthquake 3. Loads due to scram	Bearing plate	22,000[a]	$f_b = 8,000$
For emergency conditions	Emergency condition 1. Dead loads 2. Design basis earthquake 3. Loads due to scram	Bearing plate	33,000[b]	$f_b = 16,000$
For faulted conditions	Faulted condition 1. Dead loads 2. Design basis earthquake 3. Jet reaction load	Bearing plate	36,000[c]	$f_b = 16,800$

[a] Two-thirds of yield strength for bending gives 24,000 psi, but 22,000 psi is used for conservatism.

[b] A 1.5 factor is applied to the normal and upset limit since the emergency condition is not critical for an inactive equipment.

[c] For A-36 material, the yield strength is 36,000 psi.

Table 3.9-2ac

Reactor Vessel Support Equipment (Continued)
(iv) Stabilizer Bracket-Adjacent Shell

Acceptance Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
ASME B and PVC Section III Primary Local Membrane Plus Primary Bending Limit for SA 533 Grade B, Class 1:				
For design mechanical load condition: 1.5 x 26,700 = 40,050	Design mechanical load 1. Design earthquake (operating basis earthquake) 2. Design pressure	Local membrane plus bending	40,050	37,635
For faulted and emergency condition: 1.5 Sy = 63,450	Faulted and emergency condition load: 1. Maximum credible earthquake (design basis earthquake) 2. Jet reaction forces 3. Design pressure	Local membrane plus bending	63,450	57,745

Note: Faulted category loads were evaluated with emergency allowable stresses.

Table 3.9-3

Load and Stress Criteria For
ASME Code Section III Class 1 Piping

Criteria	Load Combination
The required minimum wall thickness, t_m, for piping under internal design pressure is calculated by using the indicated formula, Equation (1). The actual minimum wall thickness, t_a, must be equal to or greater than the required minimum wall thickness, t_m. Thus $t_m = \frac{PD_o}{2(S_m + yP)} + A$ (See note 1)	Design pressure (P)
Primary stress intensity is calculated by Equation (9) for any normal and upset condition loading combination. The maximum calculated value is 1.5 S_m	Design pressure Weight Other sustained mechanical loads Inertia effects due to occasional mechanical loads as specified in Table 3.9-2
Primary plus secondary stress intensity range for every pair of load sets is calculated by Equation (10) $\leq$ 3 S_m (see note 2)	Operating pressure Mechanical loads other than weight Inertia effect due to normal and upset occasional mechanical loads as specified in Table 3.9-2 Anchor movements due to any cause Thermal expansion Linear temperature gradient at pipe wall and thermal effects of gross discontinuity

Table 3.9-3

Load and Stress Criteria for
ASME Code Section III Class 1 Piping (Continued)

Criteria	Load Combination
If Equation (10 cannot be satisfied for all load sets, then for those pairs of load sets which do not satisfy Equation 10):	
The nominal value of expansion stress is calculated by Equation (12) $\leq 3S_m$	Thermal anchor movements Thermal expansion
The range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending and thermal expansion stresses, are calculated by Equation (13) $\leq 3S_m$	Operating pressure Weight plus other sustained mechanical loads Inertia effect due to normal and upset occasional mechanical loads as specified in Table 3.9-2 Thermal effect of gross discontinuity

For every pair of load sets the peak stress intensity range (S_p) is calculated by Equation (11) due to operating pressure, mechanical loads other than weight, inertia effect of normal upset occasional mechanical loads, anchor movement due to any cause, thermal expansion, temperature gradient at pipe wall, and thermal effects of gross discontinuity.

The alternating stress intensity (S_{alt}) is calculated as follows:

1/2 S_p, for every pair of load sets where Equation (10) is satisfied.

[(K_e) (1/2) (S_p)], for every pair of loads sets, where Equation (10) cannot be satisfied, but Equations (12) and (13) are satisfied. (K_e) is as defined in NB-3653.6.

The appropriate design fatigue curve is used to determine the number of allowable cycles (N_i) corresponding to the alternating stress intensity, for every pair of load sets. The usage factor (U_i), is calculated by taking the expected number of load set cycles (N_i) and dividing this value by (N_i).

The cumulative usage factor (U), which is the sum of the individual usage factors (U_i), is calculated

Table 3.9-3

Load and Stress Criteria for
ASME Code Section III Class 1 Piping (Continued)

Criteria	Load Combination
Cumulative Usage Factor (U) ≤ 1.0	
The permissible pressure under any emergency condition is shown not to be greater than 1.5 times the design pressure (P)	N/A
The permissible pressure under any faulted condition is shown not to be greater than 2 times the design pressure (P) (see note 3)	N/A
The primary stress intensity is calculated by Equation (9) for any emergency condition loading combination. The maximum calculated value is ≤ 2.25 S_m.	Peak pressure Weight Other sustained mechanical loads Inertia effects due to occasional mechanical loads as specified in Table 3.9-2
The primary stress intensity is calculated by Equation (9) for any faulted condition loading combination (see note 3). The maximum calculated value is ≤ 3.0 S_m.	Pressure Weight Other sustained mechanical loads Inertia effects of occasional mechanical loads as specified in Table 3.9-2

NOTES:

1. All equations and symbols are as defined in ASME Code Section III, Subarticle NB-3600.

2. This need not be satisfied if the rules of ASME Code Section III, Paragraph NB-3653.6 are complied with.

3. This need not be satisfied if other rules, as set forth in ASME Code Section III Appendix F are complied with.

Table 3.9-4

Load and Stress Criteria for ASME Code
Class 2 and 3 Piping System

Criteria	Load Combination
The required minimum wall thickness, t_m, for piping under internal design pressure is calculated by using the indicated formula, Equation (3). The actual minimum wall thickness, t_a, must be equal to or greater than the required minimum wall thickness, t_m. Thus $t_m = \frac{PD_o}{2(S_m + yP)} + A$ (See notes 1 and 2)	Design pressure (P)
The sum of the longitudinal stresses due to pressure, weight, and sustained loads are calculated by Equation (8) $\leq$ 1.0 S_h.	Design pressure Weight Other sustained mechanical loads
The sum of the longitudinal stresses due to pressure, weight, sustained loads, and inertia effects of normal and upset occasional loads are calculated by Equation (9) $\leq$ 1.2 S_h.	Peak pressure (see note 3). Weight Sustained mechanical loads Inertia effects of occasional mechanical loads as specified in Table 3.9-2
The thermal expansion stress range, including effects of anchor displacement due to normal and upset occasional mechanical loads are calculated by Equation (10) $\leq$ S_a (see note 4).	Thermal expansion Anchor movements due to thermal expansion
The sum of the longitudinal stresses due to pressure, weight, and other sustained loads, plus the thermal expansion stress range, including effects of anchor displacement due to thermal expansion and normal and upset occasional	Design pressure Weight Other sustained mechanical loads Thermal expansion Anchor displacements due to

Table 3.9-4

Load and Stress Criteria for ASME Code
Class 2 and 3 Piping System (Continued)

Criteria	Load Combination
mechanical loads are calculated by Equation (11) ≤ (S_h + S_a) (see note 4).	thermal expansion and normal and upset mechanical loads as specified in Table 3.9-2
The sum of the longitudinal stresses due to pressure, weight, sustained loads, inertia effects of Emergency condition mechanical loads are calculated by Equation (9) ≤ 1.8 S_h.	Design pressure Weight Other sustained mechanical loads Inertia effect of occasional mechanical loads as specified in Table 3.9-2
The sum of the longitudinal stresses due to pressure, weight, sustained loads, inertia effects of faulted condition mechanical loads are calculated by Equation (9) ≤ 2.4 S_h.	Peak pressure (see note 3) Weight Other sustained mechanical loads Inertia effect of occasional mechanical loads as specified in Table 3.9-2

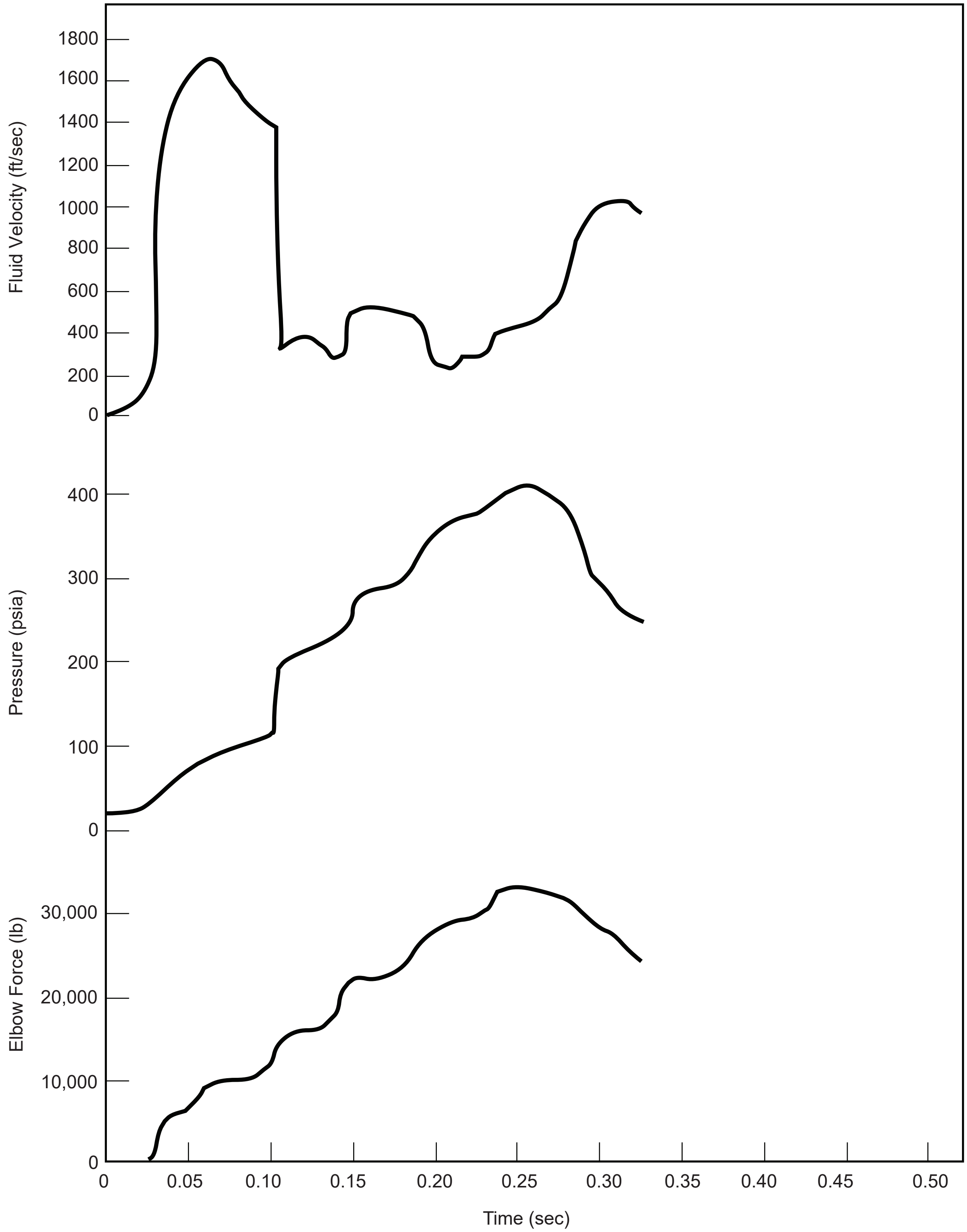
NOTES:

1. All equations and symbols are defined in ASME Code Section III, Subarticle NC-3600.

2. Joint efficiency (E) is as defined in ASME Code Section III, Paragraph ND-3641.1. For ASME Code Class 2 piping E equal to 1.0 is used.

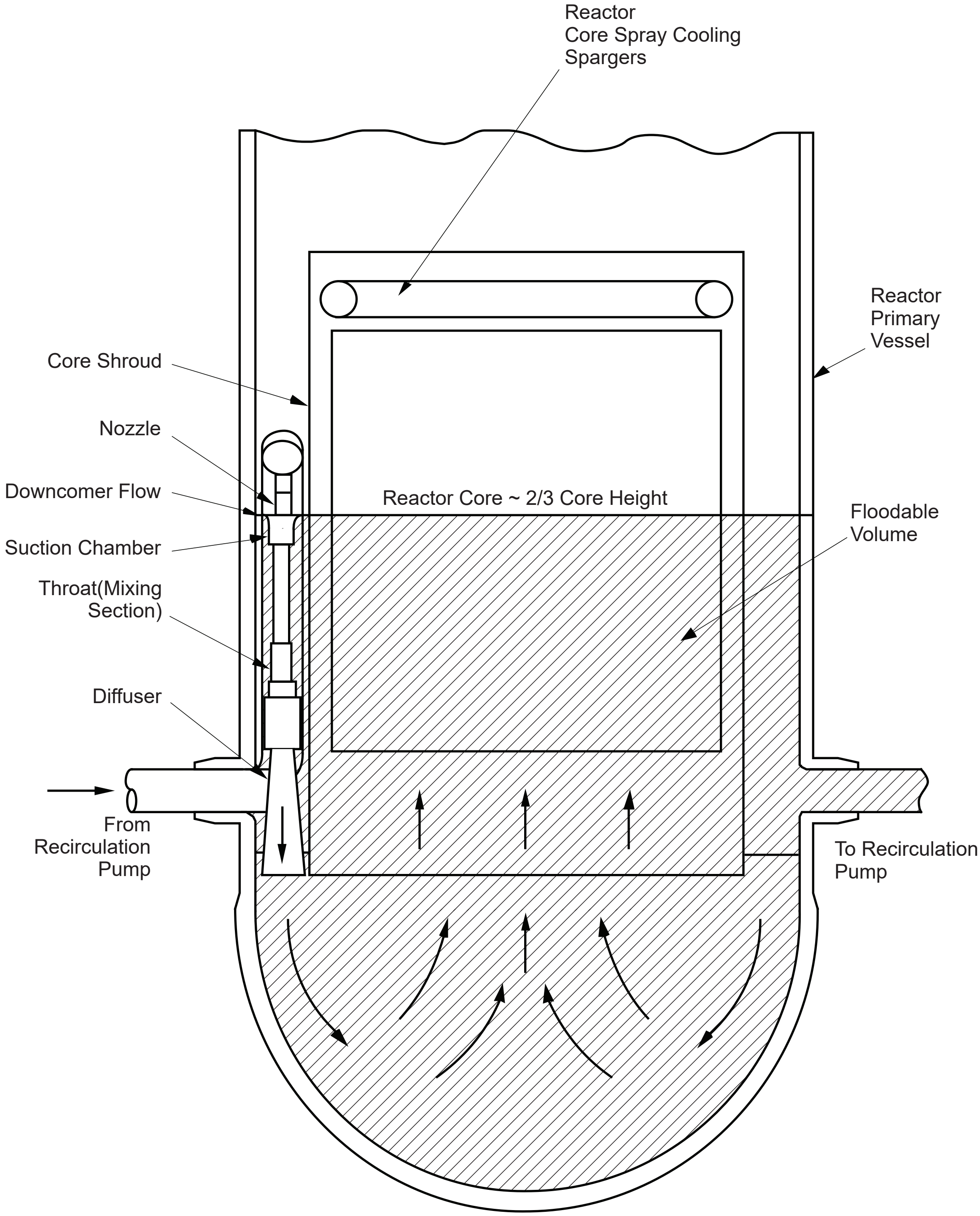
3. Design pressure may be used if the Design Specification states that peak pressure and earthquake need not be taken as acting concurrently.

4. The requirements of either Equation (10) or Equation (11) must be met.

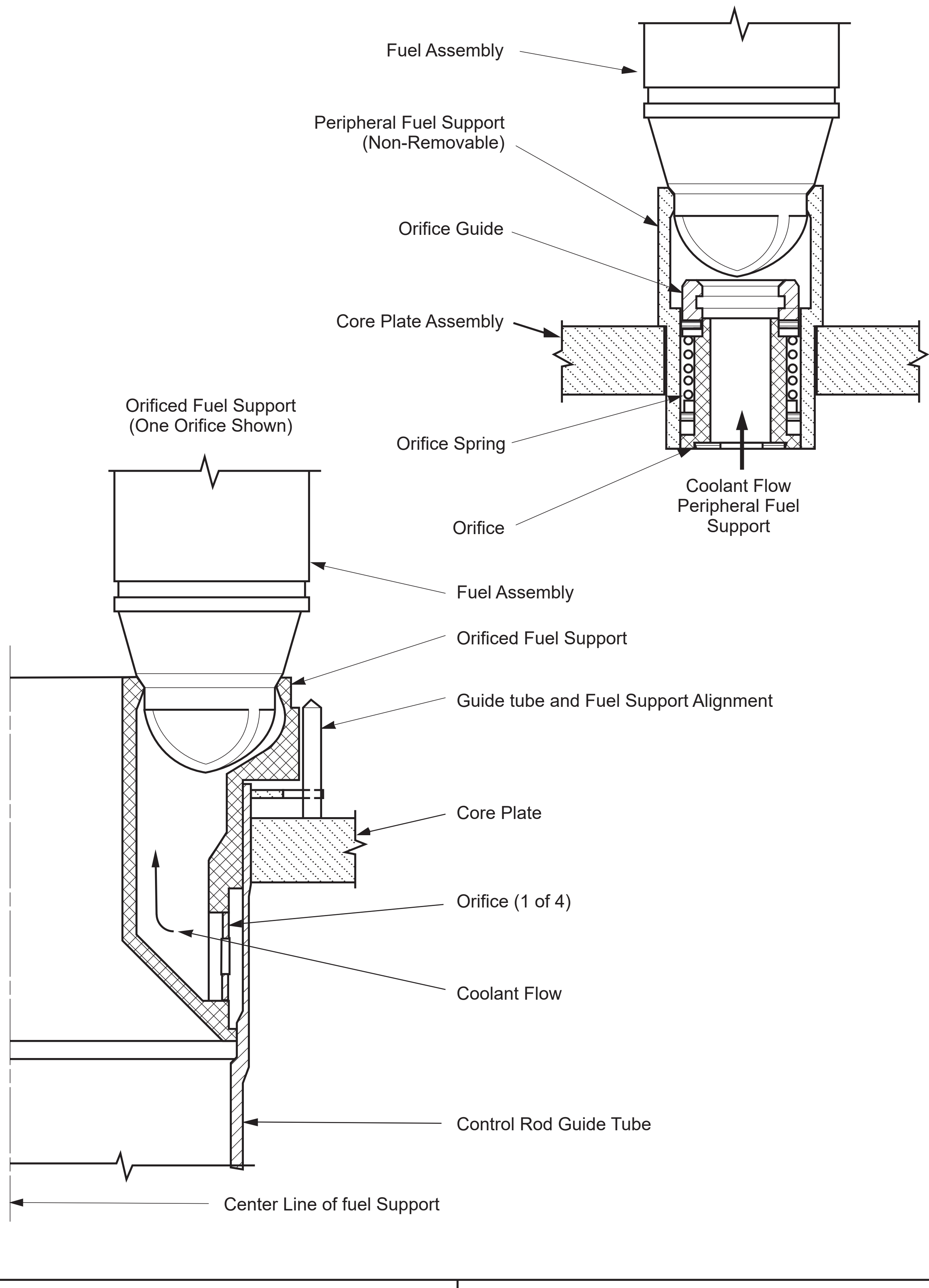


Typical Relief Valve Transient

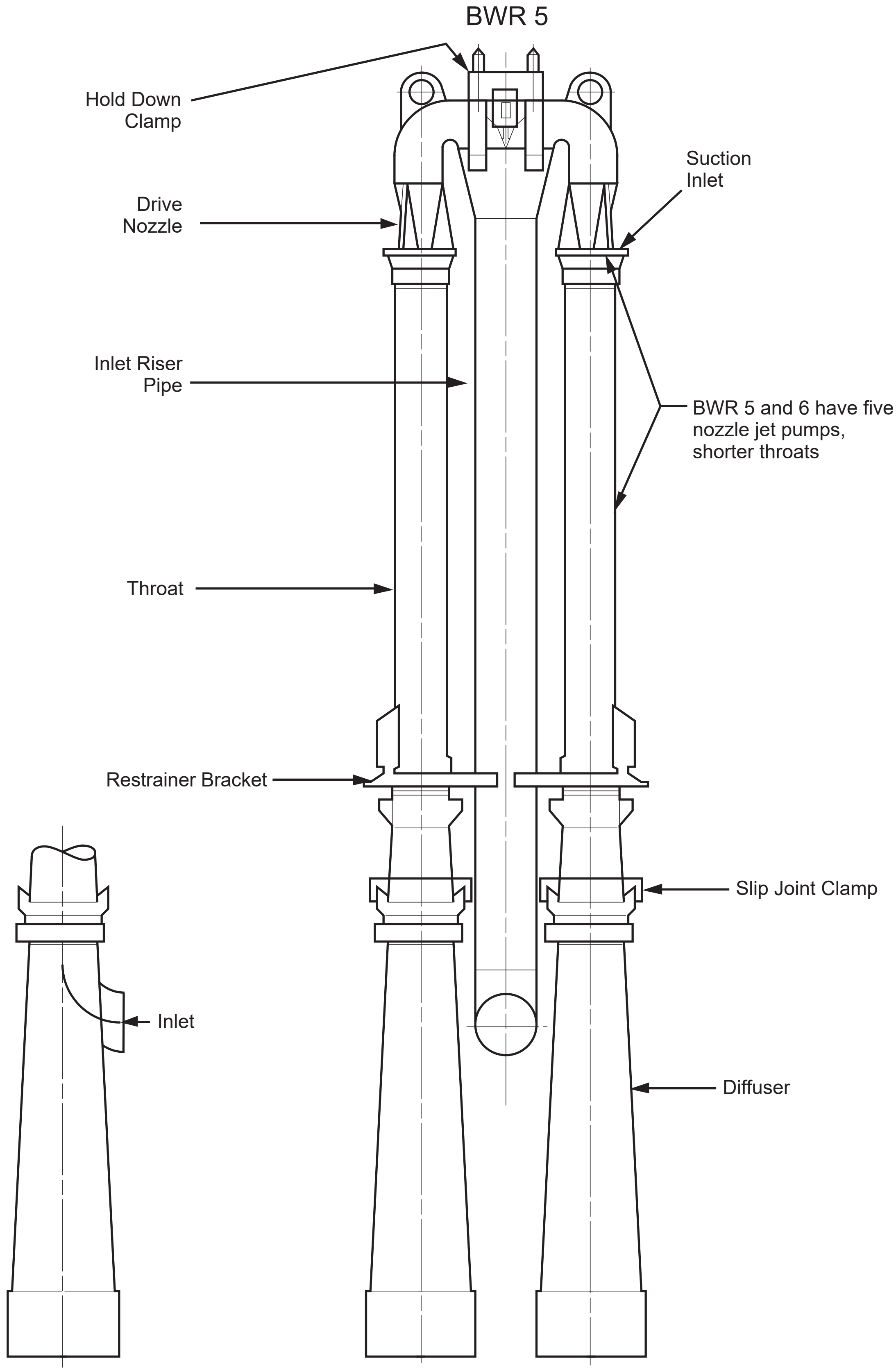
Draw. No. 990306.79 | Rev. | Figure 3.9-1



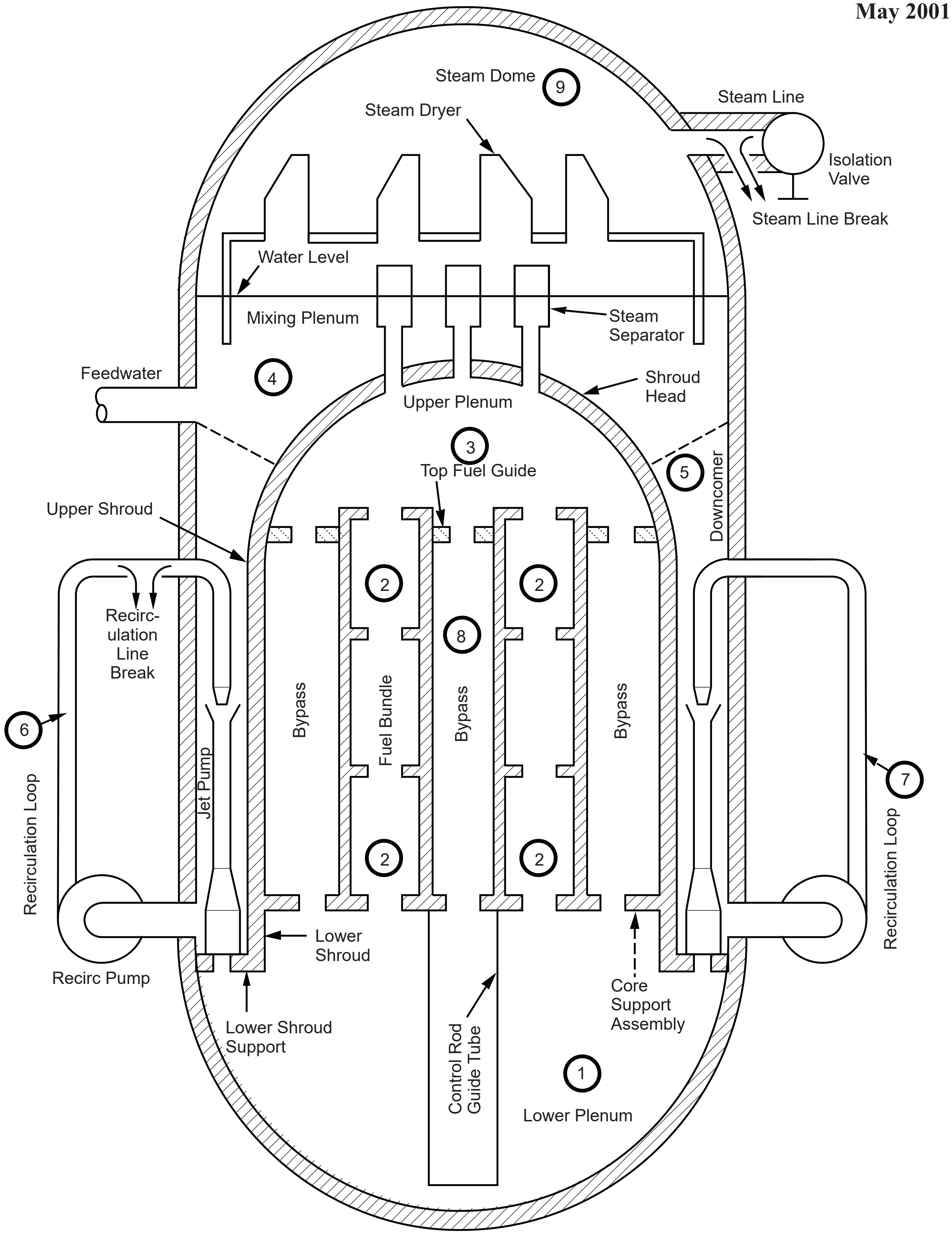
Columbia Generating Station Final Safety Analysis Report	Reactor Internals Flow Paths		
	Draw. No. 990306.80	Rev.	Figure 3.9-2



Columbia Generating Station Final Safety Analysis Report	Fuel Support Pieces		
	Draw. No. 990306.81	Rev.	Figure 3.9-3



Columbia Generating Station Final Safety Analysis Report	Jet Pump		
	Draw. No. 990306.82	Rev.	Figure 3.9-4



Columbia Generating Station
Final Safety Analysis Report

Pressure Nodes Used For Depressurization Analysis

Draw. No. 990306.83	Rev.	Figure 3.9-5

3.10 SEISMIC AND DYNAMIC QUALIFICATION OF SAFETY-RELATED INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC AND DYNAMIC QUALIFICATION CRITERIA

3.10.1.1 Safety-Related Equipment Identification

The Master Equipment List (MEL) is a computerized database of CGS equipment identification numbers and related information. The Safety Related Mechanical (SRM) list is a subset of MEL which contains all of the mechanical (nonelectrical) equipment which is in engineered safety features and reactor protection systems. The seismic and dynamic qualification of SRM equipment is discussed in Section 3.9.2.2. The equipment on the C1E list, also a subset of MEL, and the equipment necessary for the operators to follow the course of an accident (Regulatory Guide 1.97) are addressed in Section 3.10. The C1E list also includes equipment supporting structures (cabinets, racks, etc.). Class 1E is defined per IEEE 323-1974 as the safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment.

All parameters required to perform the qualification evaluation have been determined, including normal and accident operational requirements, operating data, and manufacturer's data. The location of the equipment has been verified by plant walk down or by plant records, to ensure the appropriateness of the required response spectra (RRS).

The C1E list in the MEL includes the Qualification Information and Documentation (QID) file number.

The QID file contains the following information:

a. The name of the company or organization which prepared the qualification report,

b. The report identification number and date,

c. The complete report (if the reports were proprietary and not released to Energy Northwest, audits were conducted, and summary reports prepared from the audit),

d. Required input loads or applicable required and test response spectra for the equipment,

e. Normal and accident operational requirements of the equipment,

f. The identification of whether the equipment is subject to fatigue due to hydrodynamic loading, and

g. The building location for each piece of equipment.

3.10.1.2 Criteria for Acceptability

The original equipment seismic qualification requirement for CGS was described in the Preliminary Safety Analysis Report (PSAR). These requirements specified that the nuclear steam supply system (NSSS) and balance-of-plant (BOP) equipment be designed and tested to good industry practices. IEEE 344-1971 represented the established industry practices at that time and equipment purchases were made to those requirements.

In March 1979 Energy Northwest was notified that the NRC would review CGS equipment seismic qualification to upgraded criteria. This criteria was defined as IEEE 344-1975 as supplemented by Regulatory Guides 1.100 and 1.92 and NUREG-0800, Standard Review Plan (SRP) Sections 3.9.2 and 3.10. A complete review (reevaluation) of the seismic/ hydrodynamic load basis, along with reevaluation of past equipment qualification documentation, was performed. Energy Northwest undertook an equipment requalification program to ensure all Class 1E equipment would perform their safety functions during the seismic/hydrodynamic loading conditions postulated to occur at CGS. This program included

a. Identification of Class 1E equipment,
b. Definition of seismic and hydrodynamic loads,
c. Collection of seismic qualification documentation to current criteria,
d. Reevaluation of the seismic qualification documentation to current criteria,
e. Identification of document deficiencies, and
f. Correction of identified deficiencies.

The NRC staff assembled a Seismic Qualification Review Team (SQRT) and conducted a site audit of the qualification program and equipment installation.

All C1E instrumentation and electrical equipment was designed to withstand the effects of the safe shutdown earthquake (SSE) described in Section 3.7.1.

The safety-related (Class 1E) instrumentation and electrical equipment were reevaluated to ensure performance of their safety function during and after operating basis earthquakes (OBEs), SSE, and/or the hydrodynamic loads which result from a loss-of-coolant accident (LOCA) or other design-basis event.

Suppression pool hydrodynamic loads were developed for the CGS plant and are discussed in the "Plant Design Assessment for SRV and LOCA Loads," Revision 3 (see Appendix 3A). The safety/relief valve (SRV) building responses are appropriately combined with OBE, SSE, intermediate break accident (IBA), and design-basis accident (DBA) building responses to provide the basis for evaluating acceptability of Class 1E electrical and safety-related mechanical equipment originally qualified to seismic only dynamic loading. Detailed reevaluation of each component of Seismic Category I equipment was performed by comparison of original qualification RRS with revised seismic plus hydrodynamic RRS to demonstrate satisfactory qualification of the equipment. When such comparisons were not sufficient, other means of evaluating the original qualification against the new dynamic load combinations are used.

Hydrodynamic loads were limited to equipment located within containment or pipe-mounted equipment located between containment and the first equivalent six-way anchor outside containment. For that equipment, the hydrodynamic response spectra was added by square-root-of-the-sum-of-the-squares (SRSS) to the response due to the SSE computed using the finite element soil-structure interaction analysis to determine adequacy. The use of SRSS methodology for combining response to dynamic loads was justified in Reference 3.10-1. In regard to the load combination SRV1 + SSE + DBA, CGS plant design adequacy assessments for Class 1E electrical equipment were performed on the generic basis established by the BWR Mark II Owner's Group for this load combination.

The equipment affected by hydrodynamic loads were identified and an evaluation documented in the QID files. A list of the equipment was included in the CGS Dynamic Qualification Report (Reference 3.10-2).

The equipment located in other buildings was reevaluated for the motion caused by the SSE. That motion is defined by the lumped-mass model analysis described in Section 3.7.2.

The reevaluation was based on IEEE 344-1975, as supplemented by Regulatory Guide 1.100 and SRP Section 3.10. There are four exceptions to the use of these criteria:

a. Interim criteria was established to reevaluate equipment mounted on piping systems whose analyses was not completed. The interim criteria was to use the peak of the applicable 0.5% damping floor response spectrum about 8 Hz as input acceleration for analysis or for sine dwell testing. The piping systems were designed, in turn, not to respond to frequencies less than 8 Hz. The as-built piping analyses were subsequently completed using the damping values specified in Regulatory Guide 1.61 or the criteria in ASME Code Case N411. The computed accelerations of the equipment were compared to the interim acceleration criteria to verify that the interim criteria was conservative;

b. Equipment which was qualified by testing using single frequency motion was reevaluated using the following criteria to establish its adequacy;

1. If the equipment was rigid [no resonant frequency below the zero period acceleration (ZPA) of the applicable response spectra] the test input acceleration must be greater than the acceleration corresponding to the ZPA of the response spectrum of the mounting point of the equipment;

2. If the equipment had only one natural frequency, the response acceleration to the test motion must be calculated at the appropriate damping ratio. To account for cross coupling, the required response acceleration was calculated by multiplying the acceleration corresponding to the equipment's natural frequency found on the applicable response spectrum by the square root of 2 (1.41). If test response acceleration exceeded the required response acceleration, the test motion was considered adequate for requalification of the equipment;

3. If the equipment had multiple resonant frequencies, it was tested at each of them. The response to each test was calculated at each resonant frequency. That is, the response to a test at one frequency was calculated at that frequency and at all other resonant frequencies. The responses were then combined using the SRSS method. The test motion was considered adequate if the SRSS of the response accelerations to every test was greater than 1.4 times the SRSS of the accelerations found at the resonant frequencies on the applicable response spectrum;

4. If the equipment had closely spaced modes, the criteria of (3), above, was used except the responses to the closely spaced modes were combined by the absolute sum rather than SRSS;

c. For equipment which was panel, rack, or duct mounted, the maximum transmissibility of the mounting system was found by a combination of testing and analysis. The ZPA of the applicable response spectrum was then multiplied by this transmissibility to find the required acceleration for the equipment. Test accelerations of the equipment were then compared with the calculated required acceleration to establish qualification of the equipment; and

d. IEEE 344-1975 references IEEE 323-1974. Section 6.3.5 of IEEE 323-1974 recommends thermal and radiation aging before vibration testing. It has not been shown that normal service condition environmental aging reduces equipment's ability to withstand a seismic event.

The Electric Power Institute (EPRI) conducted testing to confirm that aging has insignificant effects on the ability of electrical and electronic equipment to survive a seismic event. That work is documented in EPRI NP-3326 and NP-5024.

Documentation demonstrating qualification for each item is assembled in QID files. These files provide the details of the qualification method utilized in demonstrating the equipment's adequacy to function when exposed to seismic/hydrodynamic vibrational input. See Section 3.10.2 for a discussion on methods.

3.10.1.2.1 Cable Tray and Conduit Supports

Regardless of cable tray or conduit function, all supports located in Seismic Category I structures are designed to Seismic Category I requirements, with the exception of supports for field routed trays and conduits containing cabling for the communication system and all ac lighting outside of the main control room. All supports are qualified by dynamic analysis using appropriate seismic response spectra. The design considers both dead loads (loads which do not change magnitude and/or position), live loads (loads which do change magnitude and/or position), and SSE acceleration loads. Tray and conduit cable loadings (lb/ft) are accounted for in support design, based on maximum permissible raceway loading for the types of cable utilized.

Routing of trays and conduits containing cabling for the communication system and all ac lighting outside of the main control room were reviewed or inspected prior to February 1993 to ensure that failure of the support system cannot result in these trays and conduits having an impact on Class 1E equipment.

In February 1993, the installation procedure was changed to ensure future installations of conduit are aligned with engineering standards which define Seismic Category I areas of the plant. Installation requirements are accomplished by using Seismic Category II over I supports and appropriate span lengths.

3.10.1.2.2 Decision Criteria (Original Construction Permit Basis)

3.10.1.2.2.1 Structurally Simple Equipment. See Section 3.7.2.1.9.

3.10.1.2.2.2 Structurally Complex Equipment. Class 1E equipment determined to be structurally complex such that it cannot be described by a simple model may also be qualified by analysis. The equivalent SSE horizontal loads (plus hydrodynamic loads where applicable) are determined using a dynamic model analysis of the equipment represented as a multidegree of freedom system. Horizontal floor response spectra for the particular equipment location and appropriate damping coefficients are used as input to determine the horizontal equipment response. A similar procedure is used to determine the vertical equipment response.

3.10.1.2.2.3 Optional Dynamic Testing. In lieu of calculations and analyses, vibration testing of Class 1E equipment is an acceptable method of demonstrating the capability of equipment furnished to meet seismic loading requirements given by the applicable floor response spectra. Test data furnished are either data acquired by testing equipment specifically for CGS or data acquired from previously tested comparable equipment. Dynamic tests were performed in accordance with IEEE 344-1971. (See Section 3.10.1.2.3 for reevaluation to IEEE 344-1975.)

3.10.1.2.2.4 Mandatory Dynamic Testing. When potential failure of Class 1E equipment cannot be evaluated structurally (e.g., opening or closing of electrical circuits), then vibration tests are required to demonstrate seismic adequacy. No analytical procedures are considered acceptable in these instances.

3.10.1.2.2.5 Combined Test-Analysis. Where Class 1E equipment cannot be practically qualified by either analysis or testing alone (due to equipment size, complexity, etc.), a combination test and analysis is required to demonstrate seismic capability. The combination test and analysis were performed in accordance with IEEE 344-1971. (See Section 3.10.1.2.3 for reevaluation to IEEE 344-1975.)

3.10.1.2.3 Reevaluation of Original Seismic Qualification

Section 3.10.1.2.2 describes the decision criteria employed in the selection of the qualification method used in the initial equipment specification. As discussed in Section 3.10.1.2, a reevaluation of the original qualification basis and methods employed was required by the NRC. This section discusses the reevaluation decision criteria.

3.10.1.2.3.1 Reevaluation Decision Criteria. The basis and method of qualification for each equipment item was reviewed to determine its adequacy to meet IEEE 344-1975. In addition, requalification to hydrodynamic load conditions was assessed.

Original qualification of the equipment generally is in the following categories:

a. Existing documentation fully satisfied existing new loads and criteria. Requalification consists of the preparation of appropriate comparative and certification documents;

b. New dynamic loads or criteria impact the previous qualified status of a component. Requalification can be completed by providing additional analysis for that component to supplement the original testing or analysis;

c. Previous qualification method is not applicable to load criteria now prescribed. For example, a static analysis may have been performed where a dynamic analysis would now be appropriate. Requalification would require proper analysis, test or a combination thereof; and

d. Qualifying documentation consisted of certificates of conformance. In some cases certificates of conformance did not exist because of equipment relocations or modified safety class determinations. Often, qualifying documentation could be purchased from the manufacturer. If not, requalification would consist of performing proper analysis, test, or a combination thereof.

Qualification procedure methods vary with the nature and load criteria of the equipment and consist of static techniques, dynamic analysis, or test procedures. The following describes these methods.

3.10.1.2.3.2 Static Analysis. Static analysis was allowed for rigid equipment; that is, equipment whose natural frequency is above the ZPA of applicable response spectra. Rigid equipment items are analyzed by static methods which determine forces and moments resulting from center of gravity loading of a lumped mass acted on by the resultant acceleration from multidirectional earthquake motions. Conventional analysis determined stresses and/or deflections at all critical sectional areas, mounting attachment points, and anchor bolts. All stress level findings were additive to operational loads. Structural integrity was established by comparison of stress levels with prescribed codes or manufacturers' acceptance criteria. Selection of acceleration coefficients was based on response spectra at the equipment item mounting location.

Static analysis methods were particularly suited to equipment for which structural integrity is the primary criterion for qualification. Application of static analysis required adequate demonstration that the equipment could be realistically represented by the simple model and that the method produced conservative findings.

3.10.1.2.3.3 Dynamic Analysis. Dynamic analysis methods employed a mathematical model accurately representing the structural mass and stiffness distribution with sufficient degrees of freedom to determine dynamic response to cyclic loadings. These methods were employed when equipment could not be characterized as relatively simple or when interactive effects had to be included in the demonstration of adequacy. Dynamic analysis was also used to qualify equipment for which static analysis methods were too conservative.

Detailed dynamic analyses are accomplished with the use of sophisticated computer programs, such as STRUDL, ANSYS, and STARDYNE. The programs require development of a mathematical model that describes the mass and stiffness properties of the equipment. This involves preparation of model geometry, material constants, section properties, boundary conditions, and applied loads for input into the selected computer program. Standard Review

Plan Section 3.9.2 modeling guidance is applied. As with static analysis, the results are combined with all other loads acting on or within the equipment item. The use of twice the peak values of the SSE/OBE seismic response spectra to compensate for the inability to realistically model structurally complex equipment was not used in the reevaluation.

3.10.1.2.3.4 Demonstration of Operability By Analysis. Where the function of a component could be demonstrated by analysis alone, analysis was often chosen as the cost-effective method for qualifying mechanical equipment and electric motors.

Where structural failure was the only known related failure mode, the allowables and rules of Section III of the ASME Boiler and Pressure Vessel (B&PV) Code were used for pressure retaining ASME materials. For nonpressure boundary materials either the rules and allowables of the AISC Code were used or the ASME Code allowables were extended to nonpressure retaining parts. For ANSI B31.1 components, the allowables and rules of ANSI B31.1 were used. For components which must produce mechanical motion after the faulted condition, the allowables were limited to emergency values for faulted loads. Where the listed codes do not apply (e.g., electric motors), good engineering practice was used to ensure stresses were within the working stress of the material with suitable safety factors.

For some components, operability was established by ensuring parts that have relative motion do not come into contact with each other. The manufacturer's drawings were obtained for those components and the minimum clearances were determined. In order to qualify these components, a deflection analysis was performed at peak load which showed that the parts did not come into contact. Also, it was shown that stress limits were not exceeded.

The damping values used in the analyses were those specified in Regulatory Guide 1.61 unless another was justified and documented.

In the analyses performed, horizontal and vertical loads are assumed to occur simultaneously in the most unfavorable combinations. Normal operating loads are also combined with the accident loads to produce the most severe stress combination. The "no loss of function" stresses are limited to 90% of the materials minimum yield strength with an SSE added to hydrodynamic loads and the normal operating loads.

3.10.1.2.3.5 Consideration of Fatigue. A fatigue reevaluation was performed for all components for which hydrodynamic loads are significant. To be qualified by analysis the stress levels and number of cycles at that stress are calculated. Then using the methods of ASME Section NB-3222.4, a cumulative damage fraction is calculated. That damage fraction cannot equal or exceed 1.0 for the component to be considered qualified.

3.10.1.2.3.6 Testing Methods. Qualification by testing was required when complex or active equipment could not be efficiently modeled to correctly predict response. The methods employed were laboratory tests conducted to simulated service conditions and in-situ tests conducted in the installed configuration.

Dynamic tests are performed to a test response spectrum (TRS) which envelops and closely resembles RRS over the critical frequency range. Equipment was tested to simulated inservice load conditions whenever practical and are appropriately justified when not applied. Operability was verified during and/or after the testing as applicable to the equipment being evaluated. The test specimens were checked for spurious operation during testing. If there were spurious operations, it was determined that they had no detrimental effects on the safety function of the equipment. Spurious operation of relays (e.g., contact chatter) was limited to 2 msec maximum, per IEEE 501-1978 unless it had been demonstrated that the spurious action found would not affect the safety function of the component as applied in its safety system.

Multiple frequency, multiple axis testing was used to qualify most of the Class 1E electrical equipment. This testing was performed according to IEEE 344-1975. The equipment was operated before, during, or after the test as required by the system safety function. No functional or structural failure was allowed.

Single frequency input motion, such as sine beats, was allowed when (a) the characteristics of the seismic input motion indicated that the motion was dominated by one frequency, (b) the anticipated response of the equipment was adequately represented by one mode, or (c) the test input motion had sufficient intensity and duration to excite all modes to the required amplitudes, such that the testing response spectra envelop the corresponding response spectra of the individual modes.

Some equipment was previously qualified to IEEE 344-1971 using single frequency testing. The NRC required that justification be supplied for use of single frequency single axis qualification where the component could respond to multiple modes. These criteria were seldom used to qualify CGS equipment. Justification for those particular cases was made available to the NRC SQRT reviewers during an audit in January 1983 and resolved.

Several in-situ test approaches were used to complete qualification. These may be grouped into tests that verify accuracy of the analytical model, verify rigidity, justify reducing stress analysis conservatism, and demonstrate operability under simulated load conditions. This latter test, more appropriately called an in-situ static load test, simulates seismic deflections by means of an appropriately directed static force application. Operability is exhibited before, during, and after application of the deflecting load. Another type of in-situ test was used to qualify components attached to the high-pressure core spray (HPCS) system diesel generator. Large reciprocating engines produce starting and running vibrations to attached components much greater than that received due to seismic acceleration. If it was determined that the acceleration to the component under actual running operation exceeded the calculated input due to seismic effects by a factor of 3, the component was accepted as having sufficient seismic adequacy.

3.10.1.2.3.7 Combination of Testing and Analysis. Most qualification of equipment employs a combination of testing and analysis. Sections 3.10.2 and 3.10.3 provide examples of typical equipment qualifications that used this methodology.

3.10.2 METHODS AND PROCEDURES FOR QUALIFYING INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.2.1 Methods For Qualifying Nuclear Steam Supply System Equipment (Excluding Motors and Valve-Mounted Equipment)

a. Procedures - GE-supplied Seismic Category I equipment meets the requirements that the seismic qualification should demonstrate the capability to perform the required function during and after the SSE and other postulated events. Both analysis and testing were used, but most equipment was tested. Analysis was primarily used to determine the adequacy of mechanical strength (mounting bolts, etc.) after operating capability was established by testing;

   1. Analysis - GE-supplied Safety 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 impractical. Analytical methods described in Section 3.10.1.2.3 and sanctioned by IEEE 344-1971 were utilized in such cases;

   2. Testing - GE-supplied Safety Class 1E equipment, having primarily an active electrical safety function, was tested in compliance with IEEE 344-1971, and reevaluated per Section 3.10.1.2.3. Supplemental test data meeting IEEE 344-1975 was obtained when single frequency testing could not be justified; and

b. Documentation - Documentation used to seismically qualify GE-supplied Safety Class 1E equipment is in accordance with the requirements of IEEE 344-1975. The documentation is maintained in a permanent file by Energy Northwest or GE and is available for audit.

3.10.2.2 Methods For Qualifying Balance-of-Plant Equipment

Suppliers of Seismic Category I equipment such as batteries and racks, instrument racks, control consoles, electrical distribution equipment, etc., are required to demonstrate that their components or systems do not suffer loss of function during or after SSE or OBE seismic loading. Tests or analysis are performed in accordance with the criteria described in Section 3.10.1.2.3 and IEEE Standard 344-1975. The magnitude of the SSE and OBE loadings which each component experiences is determined by its location within the plant. The response spectra for the various plant locations are included in each equipment's QID file

and a comparison of the equipment's qualification and capability is made to the RRS to document its adequacy.

3.10.2.3 General Electric-Supplied Equipment Seismic Analyses and/or Testing Procedures

3.10.2.3.1 Testing Procedures for Qualifying Equipment (Excluding Motors and Valve-Mounted Equipment)

The testing procedure for qualifying electrical equipment and instrumentation (excluding motors and valve-mounted equipment) required that the devices be mounted on the vibration machine table the same way it was to be installed in the plant. The device was tested in the operating states that it would experience in performing its safety functions and these states were monitored before, during, and after the test to ensure proper function and absence of spurious reactions. In the case of a relay, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay were used in those configurations in its safety functions.

The initial seismic qualification of GE-supplied electrical equipment was based on single frequency "continuous" testing in which the applied vibration was a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. Each frequency and acceleration combination was maintained for about 30 sec except when a resonance search was made (see IEEE 344-1971). The vibratory excitation was applied in three orthogonal axes individually with the axes chosen as those coincident with the most probable mounting configuration.

The first step was to search for resonances in each device. This was done since resonances cause amplification of the input vibration and are the most likely cause of malfunction or spurious operation. The resonance search was usually run at low acceleration levels (0.2g) to avoid destroying the test sample in case a severe resonance was encountered. The search was made from 0.25 Hz to 33 Hz in accordance with IEEE 344-1971 for a test period of no less than 7 minutes; 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. Usually, the devices were either too small for an accelerometer, had their critical parts in an inaccessible location, or had critical parts that would be adversely affected by the mounting of an accelerometer. In these cases, the resonances were detected visually (strobe light), by audible observation, or by performance.

Following the frequency scan and resonance determination, the devices were tested to determine their malfunction limit. The 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 (33 Hz,) since that allowed the maximum acceleration to be obtained from deflection-limited machines.

Under the reevaluation program described in Section 3.10.1.2.3, the adequacy of the single frequency, single axis testing was reviewed. Supplemental test data, both on full cabinet assemblies and components, were obtained that utilized multifrequency biaxial input motion in accordance with IEEE 344-1975. A complete review of the control room panels and local instrument panels was performed and qualification upgrade to IEEE 344-1975 was achieved (see Section 3.10.3.1).

3.10.2.3.2 Qualification Procedures for Motors

Seismic qualification of the emergency core cooling system (ECCS) motors is discussed in Section 3.9.2.2.2.7 in conjunction with the ECCS pump and motor assembly.

3.10.2.3.3 Qualification Procedures for Valve-Mounted Equipment

The piping analyses established the response spectra, power spectral density function or time history characteristics, and developed a horizontal and a vertical acceleration for the pipe-mounted equipment. Pipe-mounted valves are normally furnished with a natural frequency greater than or equal to 33 Hz. Amplification of the seismic/hydrodynamic motion through the valve yoke structure was considered in the valve operator qualification. Class 1E motor-operated valve actuators were qualified per IEEE 382-1972, which invokes IEEE 344-1971 for dynamic qualification.

Three methods were used to demonstrate operability of the full valve assembly: (a) testing per IEEE 344-1975 criteria of representative samples that included the valve, yoke structure, valve operator, and associated limit switches and solenoids (if air operated), (b) testing of the complete valve assembly, using the in-situ static load test method, and (c) analysis of the valve assembly where the input load was small enough such that only minor deflections of the valve assembly occur and it could be demonstrated that critical parts would not bind up under maximum postulated loads. Section 3.9.2.2.1 provides additional detail on these last two methods.

The main steam SRVs, including the electrical components mounted on the valve, were subject to a dynamic seismic test. This testing is described in Sections 3.9.2.2.2.14 and 3.9.3.2.4.2.

3.10.2.4 Balance-of-Plant Equipment Analyses and/or Testing Procedures

Descriptions of the various acceptable analyses and/or test procedures, as well as criteria for determining which procedures are applicable to the particular units of equipment or systems, are described in Section 3.10.1.2 and in the reevaluation and upgrade program described in Section 3.10.1.2.3.

Seismic qualification by analysis followed the procedures described in Section 3.10.1.2.2. Most electrical equipment was qualified by test.

Seismic qualification by testing followed the procedures described in Sections 3.10.1.2.2.3 and 3.10.1.2.2.4 as follows: the specimen is fastened to a test table in a configuration simulating inservice mounting. Generally, a low level resonant search is made, followed by the OBE and SSE tests. Input excitations such as continuous single frequency sinusoidal motions, sine beat motions, are acceptable if justification as discussed in Section 3.10.1.2.3.6 is established. Random motions are used primarily to qualify the equipment. Each horizontal axis is excited separately, then simultaneously with the vertical axis, unless otherwise justified. Where applicable, voltage and current outputs are monitored. Seismic qualification by a combination of test and analysis (where applicable) was performed under the procedures described in Section 3.10.1.2.2.5. Combined test/analysis procedures were often used for large assemblies such as panels, racks, large reciprocating equipment, and large air handling units. Under this procedure, the supporting structure may be analyzed for adequacy and amplification factors and the components tested to the amplified values. Extension of a test on a prototype unit often required additional analysis to extend its results to a family of equipment of similar designs from that manufacturer.

3.10.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS FOR INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.3.1 Nuclear Steam Supply System Equipment (Other Than Motors and Valve-Mounted Equipment)

The Class 1E equipment supplied by GE is used in many systems on many different plants under widely varying seismic requirements. The seismic qualification tests were performed at all frequencies from 5 Hz to 33 Hz (the required qualification range was 0.25 Hz to 33 Hz but since test facility capability usually limited the lower frequency test to 5 Hz, a combination of test and analysis was used to ensure there were no untested lower resonances). A sample analysis is located in Appendix 3.10C.

Some GE-supplied support structures that contain Class 1E devices were qualified by analysis. A procedure of such analysis is discussed in Appendix 3.10A. Analysis was used for passive mechanical devices and was also 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 seismic frequency range (see IEEE 344-1975). If the equipment was determined to be free of natural frequencies, then it was assumed to be rigid and a static analysis was performed. A sample static analysis is discussed in Appendix 3.10B (see IEEE 344-1975). 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) containing devices which have had seismic malfunction limits established were tested by mounting the assembly on the table of a vibration machine in the manner it was to be mounted when in use. The vibration testing was performed by running a low level resonance search. The assemblies were tested in the three major orthogonal axes. The resonance search was run in the same manner as described for devices (see Section 3.10.2.3.1).

If resonances were present, a transmissibility between the input and the location of each Seismic Category I device was determined by measuring the accelerations throughout the structure and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response at any Seismic Category I device location for any given input. It was assumed that the transmissibilities were linear as a function of acceleration even though they actually decrease as acceleration is increased. Therefore, the assumption is conservative.

Since control panels and racks constitute the majority of Class 1E electric assemblies supplied by GE, seismic qualification testing of these will be discussed in more detail. The four generic panel types considered include vertical boards, instrument racks, local racks, and National Electrical Manufacturers Association (NEMA) type 12 enclosures. One or more of each type was tested using the above procedures. Figures 3.10-1 through 3.10-4 illustrate the four basic panel types referenced above and show typical accelerometer locations.

For each of the control room panels and local instrument racks a transmissibility map was established by testing, analysis, or similarity with tested panel bench boards. Required acceleration levels were established for zones within the panels.

Class 1E electrical device locations were determined through design review of panel drawings and plant inspections. Required acceleration levels were then established for each device within the panel. Then this required level was compared to the level the device received during full level assembly tests or individual device tests to ensure devices received sufficient testing to demonstrate their adequacy in their installed locations.

3.10.3.2 Balance-of-Plant Supports

Battery racks, instrument racks, cabinets, panels, and cable trays were originally specified to be qualified in accordance with IEEE 344-1971. Response spectra for all locations where the equipment is to be mounted were specified to the vendor. Subsequently, all original qualification programs were reviewed and reevaluated using IEEE 344-1975.

Cable trays, raceways, and support systems were furnished and installed by the electrical contractor in accordance with specified criteria and qualification procedures. A complete set of floor response spectra was provided to the contractor who was required to perform an analysis of the raceways and supports. This included trays and conduits, horizontal shelf members, vertical support members, internal bracing members, lateral or longitudinal

supports, and all connections. The contractor determined seismic forces based on specified levels of acceleration and frequency for each building and elevation. Maximum tray and conduit loading, minimum structural properties, as well as specified maximum spacing of supports and stress limits, are specified to the vendor. Limitations are also placed on stresses in supports to steel and to concrete. The contractor was required to submit all calculations of the seismic analysis and design for review and approval to demonstrate qualification.

3.10.4 OPERATING LICENSE REVIEW

This section discusses Energy Northwest's Seismic/Hydrodynamic Equipment Qualifications Program Results. This program has established seismic/hydrodynamic vibratory service conditions and detailed documentation of the qualification of safety-related electrical equipment, all contained in the QID files. The results of the reevaluation program and the NRC's audit of the adequacy of the qualification methods and results of the reevaluation program are also discussed.

3.10.4.1 Establishment of Service Conditions

The original service condition basis, which was seismic and operational loads has been augmented with vibrational loads due to event-induced loads, including SRV discharge, small break accident (SBA), IBA, and DBA as applicable. These loads include chugging, pool swell, drag loads, annulus pressurization, etc. These hydrodynamic loads have been established and factored into the load definition for safety-related equipment.

Energy Northwest has submitted analyses justifying the combination methodology and the plant areas affected by the loads to the NRC. The load basis for dynamic qualification of electrical equipment was reviewed and accepted by the NRC. Sections 3.7 and 3.9 provide the details of the methodology used and the load results. NUREG-0892 Supplement 1 and Supplement 4 discuss the NRC acceptance of the CGS hydrodynamic loads. Service conditions which include load definition and levels, and normal and accident operating conditions, are established for each safety-related electrical equipment and are contained in QID files.

3.10.4.2 Establishment of Qualification Information and Documentation Files

The QID files provide the details of the qualification method and results for each equipment item on the safety-related electrical equipment list. These files are the centralized collection and evaluation location for the documentation that demonstrates the equipment's qualification. Each item on the safety-related equipment list references this central file and qualification information is easily retrievable.

Included in these files is a comparison of the dynamic service conditions and the qualification levels. Test reports, analyses, qualification summary checklists, and data to verify installation similarity to qualified conditions are also included in the QID files.

The QID files are generally arranged by manufacturer and model type. Special files have also been established for complex assemblies such as control panels and local instrument racks.

3.10.4.3 Seismic Qualification Review Team Results of the Operating License Review by NRC

The NRC's evaluation of Energy Northwest's program for qualification of safety-related electrical and mechanical equipment for seismic and dynamic loads consisted of (a) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (b) an audit of selected equipment items to develop the basis for the NRC's judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program. The SQRT consisted of NRC staff members and personnel from the Idaho National Engineering Laboratory (INEL, EG&G). The SQRT reviewed the equipment dynamic qualification information contained in FSAR Sections 3.9.2 and 3.10 to determine the extent to which the qualification of equipment as installed meets the current licensing criteria as described in IEEE 344-1975, Regulatory Guides 1.92 and 1.100, and SRP Section 3.10. Conformance with these criteria was required to satisfy the applicable portions of General Design Criteria (GDC) 1, 2, 4, 14, and 30, as well as Appendix B to 10 CFR 50 and Appendix A to 10 CFR 100. A representative sample of safety-related electric and mechanical equipment as well as instrumentation, included in both NSSS and BOP scopes, was selected for the audit. The plant site visit consisted of field observations of the actual, final equipment configuration and its installation. This was immediately followed by a review of the corresponding test and/or analysis documents maintained in Energy Northwest's central files (QID). Observing the field installation of the equipment was required to verify and validate equipment modeling used in the qualification program.

On the basis of this audit, both generic and plant-specific concerns relating to the seismic and dynamic qualification of equipment were identified. In subsequent submittals, Energy Northwest developed acceptable approaches to address and resolve the audit generic findings.

Adverse effects (loss of unit function) on Class 1E equipment caused by seismic-induced spurious actuation of non-Class 1E instrumentation and electrical equipment were considered. The scope included both direct causes (i.e., equipment damage, automatic shutdown) and indirect causes (manual shutdown) and it was determined that Class 1E equipment would not be adversely affected. The loss of offsite power is not a factor since outside standby power sources are not affected.

All of the issues were resolved before the plant exceeded 5% of rated power. Energy Northwest also provided additional information relative to the specific findings and clarified the details of qualification for the pieces of equipment in question, including the air operators for the purge and vent valves.

3.10.4.4 Pump and Valve Operability Review Team Audit Results of the Operating License Review by NRC

To ensure that Energy Northwest provided an adequate program for qualifying safety-related pumps and valves to operate under normal and accident conditions, the NRC staff performed a two-step review. First, the NRC reviewed FSAR Section 3.9.3.2, the description of Energy Northwest's pump and valve operability assurance program, and compared it to SRP Section 3.10. Second, the Pump and Valve Operability Review Team (PVORT) conducted an onsite audit of a small representative sample of safety-related pumps and valves supporting documentation.

The onsite audit included:

a. A plant inspection to observe the as-built configuration and installation of the equipment,

b. A discussion of the system in which the pump and valve are located,

c. Examination of the normal and accident conditions under which the component must operate, and

d. A review of the qualification documentation (stress reports, test reports, etc).

The two-step review was performed to determine the extent to which the qualification of equipment, as installed, meets the current licensing criteria as described in SRP Section 3.10 and conformance with GDC 1, 2, 4, 14, and 30 and Appendix B to 10 CFR 50.

3.10.5 ESTABLISHMENT OF OPERATIONAL PHASE SEISMIC QUALIFICATION PROGRAM

The operational phase program for maintaining qualification of the equipment and the ongoing process of ensuring that qualified equipment is selected for plant design changes is described in the following.

Energy Northwest has established an operational phase seismic qualification program. This program is integrated with design changes to Columbia Generating Station to ensure compliance with the criteria as described in IEEE 344-1975, Regulatory Guide 1.100, 1.92, and SRP Section 3.10. Conformance to these criteria as clarified in Sections 1.8.2 and 1.8.3 ensures continuing adequacy to meet the applicable portions of GDC 1, 2, 4, 14, and 30, Appendix B to 10 CFR 50, and Appendix A to 10 CFR 100.

The operational phase qualification program is applicable for all design changes that add, modify, or delete safety-related equipment.

Design control procedures establish a special qualification review as part of the design change package preparation. Adequacy of the equipment selected to be added to the plant in the design change is assessed and documented to an "as designed" qualification status. Special procurement requirements are detailed for the purchase order. During and after installation, inspections are conducted by Quality Control to ensure that critical attributes of the final installation conforms to the design package. Documentation of qualification is established in the QID files.

For design changes that modify or reclassify existing plant equipment or support structures for safety-related equipment, a special review is also required similar to the above preprocurement review. Documentation of qualification is established in QID files.

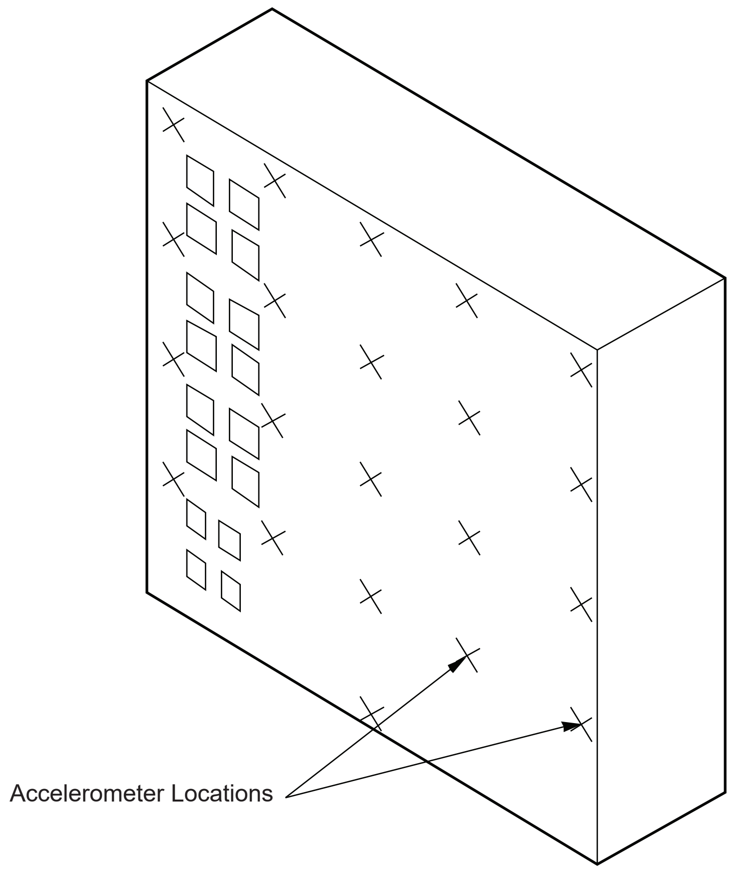
For design changes that delete existing plant equipment or downgrade its safety-related status, a special review is also conducted. Modification of the QID file is considered optional under the program. Component model or part changes that do not affect the function or capability of a component are not considered design changes. Substitutions that have a potential to affect dynamic qualification are evaluated and if necessary qualification is documented in the QID files prior to approval of the substitution.

Design control procedures also require the safety-related equipment list in the MEL to be kept current with actual plant configuration as part of the operational phase program. Thus, the link with the plant equipment configuration and qualification documentation (QID files) is maintained throughout plant life.

3.10.6 REFERENCES

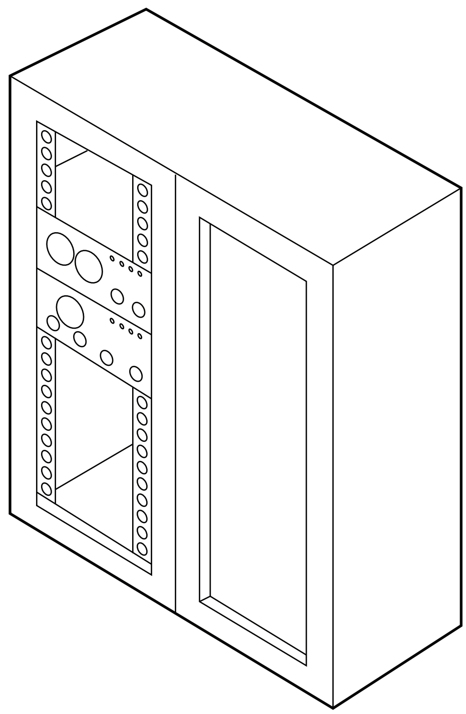
3.10-1 Letter from Supply System to NRC, Subject: "Applicability of the Use of the Square Root of the Sum of the Squares Method for Combining Peak Dynamic Responses to Dynamic Loads for WNP-2" (GO2-83-090).

3.10-2 J.L. Sullivan and D.A. Armstrong, "WNP-2 Dynamic Qualification Report for Safety-Related Equipment," Engineering Report, Vols. I and II, October 5, 1982 (historical).

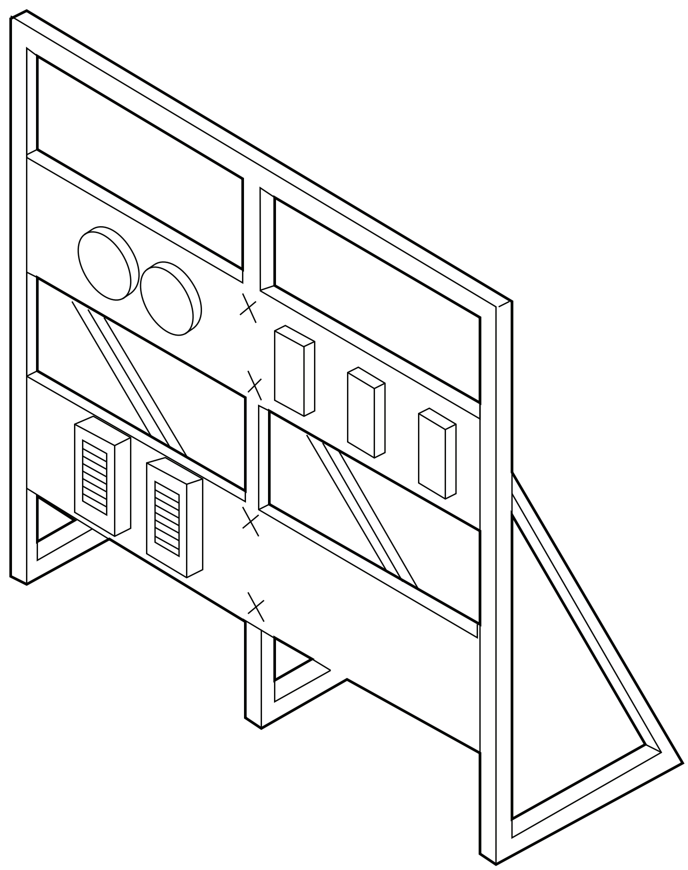


NOTE: Benchboard would be the same with a bench section protruding out approximately half-way down

Typical Vertical Board



Note: Cabinet would contain gages or other special instruments instead of simply drawer type instruments.

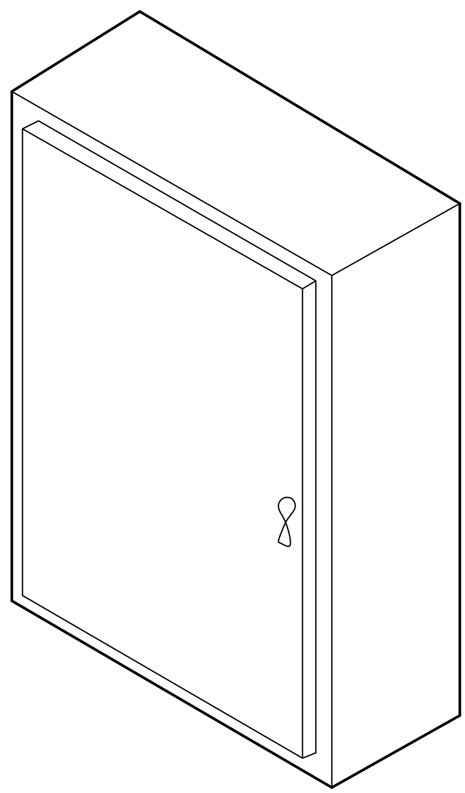


Note: Piping and other external connections not shown

Columbia Generating Station
Final Safety Analysis Report

Typical Local Rack

Draw. No. 990306.86 | Rev. | Figure 3.10-3



Note: Instruments mounted inside on internal membrane mounted on standoffs attached to back.

Columbia Generating Station
Final Safety Analysis Report

Typical NEMA Type 12 Enclosure

Draw. No. 990306.87	Rev.	Figure 3.10-4

APPENDIX 3.10A

DYNAMIC ANALYSIS BY RESPONSE SPECTRUM METHOD FOR NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT

The italicized information is historical and was provided to support the application for an operating license.

Appendix 3.10A

DYNAMIC ANALYSIS BY RESPONSE SPECTRUM METHOD FOR NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT

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Appendix 3.10A

3.10A DYNAMIC ANALYSIS BY RESPONSE SPECTRUM METHOD FOR NSSS EQUIPMENT

3.10A.1 UNCOUPLING THE EQUATION OF MOTION

3.10A.1.1 Method

The system stiffness and mass matrices are generated using standard techniques. A seismic analysis is performed using the following equations of motion and procedure to uncouple these equations:

The equations of motion in matrix form are as follows:

$$M(\ddot{X} + \ddot{Y}) + C\,\dot{X} + K\,X = 0 \qquad (1)$$

where

M = mass matrix, n x n (this includes the hydrodynamic mass)

X = column vector of displacement relative to ground (n x 1)*

C = damping matrix (n x n)

K = stiffness matrix (n x n)

Y = column vector of ground accelerations (n x 1)

$\dot{}$ = *first derivative with respect to time*

$\ddot{}$ = *second derivative with respect to time*

It should be noted that for equipment containing fluid, a hydrodynamic mass coupling exists between real structural masses. This hydrodynamic mass appears as diagonal and off-diagonal terms in the mass matrix. The overall system stiffness matrix K is determined by either the matrix force method or the matrix displacement method. The resulting stiffness matrix is similar.

Removing the driving-point acceleration vector to the right side of equation (1), the equation reduces to the classical form:

$$M\,\ddot{X} + C\,\ddot{X} + K\,X = -M\ddot{Y} \qquad (2)$$

In order to uncouple equation (2), we set:

$$X = \phi q \qquad (3)$$

Equation (2) then becomes

$$M\phi q + C\phi q + K\phi q = -M\ddot{Y} \qquad (4)$$

Pre-multiplying (4) by ϕ^T, the transpose of ϕ, and performing the coordinate transformation described in (4) such that f is defined by the following orthogonality conditions:

$$\phi^T M\phi = I \qquad (5)$$

$$\phi^T K\phi = \omega^2 \qquad (6)$$

where I is an identify matrix (Nxn) and ω^2 is a diagonal matrix of the eigenvalues. Then (4) becomes

$$\phi^T M\phi\ddot{q} + \phi^T C\phi\dot{q} + \phi^T K\phi q = -\phi^T \ddot{M}Y \qquad (7)$$

$$\ddot{q} + \phi^T C\phi\dot{q} = \omega^2 \quad q = -\phi^T M\ddot{Y} \qquad (8)$$

3.10A.1.2 Assumptions

The above procedure for uncoupling the equation of motion by using the modal matrix of the undamped system assumes that damping in the system is small. It will further be assumed that the damping matrix C is such that $\phi + TC\phi$ is a diagonal matrix. The elements of this diagonal-matrix are the modal damping values.

3.10A.2 MAXIMUM PHYSICAL DISPLACEMENT AND MAXIMUM LOAD RESPONSE

With the assumptions of 3.10A.1.2, equation (8) may be written in the following uncoupled form:

$$\ddot{q} \dot{+} 2\ \beta_i\ \omega_i \dot{q}_i + \omega_i{}^2 q_i = S_i \quad \ddot{U}_g$$

$$i = 1,\ 2,\ \text{---}\ n \qquad (9)$$

where

$$x_i = \begin{bmatrix} x_{1i} \\ x_{2i} \\ . \\ . \\ . \\ . \\ x_{ni} \end{bmatrix} \qquad \phi_i = \begin{bmatrix} \phi_{1i} \\ \phi_{2i} \\ . \\ . \\ . \\ . \\ \phi_{ni} \end{bmatrix}$$

The maximum physical displacement for each mass is then taken to the square root of the sums of the squares of each of the maximum displacement responses for each mode, i.e.,

$$(X)\,maximum = \left[\sum_{j=1}^{n} x_{ij^2} \right]^{1/2}, \quad i = 1, \quad 2, \ldots, m$$

where (X) maximum is the column vector of maximum displacements. Similarly, the maximum load response for the i+th mode is found from

$$L_{ji} = \beta_j X_i$$

$$L_{ji} = \begin{bmatrix} L_{1i} \\ L_{2i} \\ . \\ . \\ . \\ L_{mi} \end{bmatrix}$$

where

β_j *is the stress matrix for element j, j=1, … m*

m = total number of elements.

where

β_i *= damping ratio for the i+th mode expressed as percent of critical damping*

ω_i *=* i^{th} *natural angular frequency of the system*

S_i *= modal participation factor the* i^{th} *mode =* $-\phi_i^t\, MD$

U_g = ground or floor acceleration time history

ϕ_i^t = transpose of the i^{th} mode shape

D = earthquake direction vector

The response is calculated using the response spectra specified for the location of the input to the analytical model. The analytical procedure is described briefly in the following paragraphs.

The system of one-degree-of-freedom equations represented by equation (8) or (9) can be solved by the response spectrum method. With this method, the maximum modal response for each natural frequency of interest is found from the applicable response spectra. Response spectrum curves are essentially plots of the maximum responses of single-degrees-of freedom systems described by equation (9) with $S_i = 1.0$ as a function of their natural frequencies.

Having found the maximum modal displacements q, i = 1...m, the maximum physical displacement for the i+th mode is given by:

$$X_i = \phi_i S_i q_i$$

The maximum load response is taken to be the square root of the sums of the squares of each of the maximum responses for each mode, i.e.,

$$(L_j)\,maximum = \left[\sum_{i=1}^{n} L^2_{ji}\right]^{1/2}, j = 1, 2, ..., m$$

where (L) maximum is the column vector of maximum loads.

3.10A.3 MAXIMUM ACCELERATIONS

The accelerations for each mode are determined by multiplying the displacements vector for that mode (X_i) by the natural frequency (ω_i^2) of that mode.

$$Ai = X_i\, \omega_i^2$$

The maximum accelerations are then determined by

$$(A)\,maximum = \left[\sum_{i=1}^{n} A_{i^2}\right]^{1/2}$$

Appendix 3.10B

SAMPLE SEISMIC STATIC ANALYSIS FOR NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT

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SAMPLE SEISMIC STATIC ANALYSIS FOR NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT

3.10B.1 STATIC SEISMIC ANALYSIS OF STANDARD ENCLOSURES

Presented herein is a set of curves from which static seismic analysis of standard enclosures can be quickly performed. A standard enclosure is any enclosure listed in the Enclosure Standards Manual. The enclosures are assumed to be floor mounted, using all mounting holes with 5/8 in. steel bolts or studs each having an effective area of 0.2256 in^2. Using an elastic limit of one half the ultimate strength, the bolts are assumed to have a maximum safe tension stress and maximum safe shear stress of 28,000 psi and 21,000 psi, respectively. The curves are based on a design basis earthquake having a horizontal acceleration of 1.5g and a vertical acceleration of 0.5g. It is assumed that each enclosure is mounted alone and not coupled directly to any other enclosure.

The static analysis consists of determining the maximum allowable safe weight of the enclosure and its components for which the mounting bolt stresses are not exceeded. The curves of Figure 3.10B-1 have been derived for this purpose. To use the curves given in Figure 3.10B-1, first determine from Table 3.10B-1 the curve designation of the enclosure being considered. Next, using the corresponding curve in Figure 3.10B-1, determine the maximum safe weight per bolt for a given height of the center of gravity. The maximum safe enclosure weight is then determined by multiplying the weight per bolt by the total number of enclosure mounting bolts. Comparison with the actual weight of the enclosure and its components then indicates whether or not the mounting bolt stresses are exceeded. If the comparison shows that the maximum safe weight per bolt is exceeded, steps should be taken to increase the effective bolt area by welding the enclosure to its mounting, increasing the number of mounting bolts, adding top braces to a wall, or using another appropriate method to ensure safe operation during seismic disturbance.

3.10B.2 ASSUMPTIONS AND EQUATIONS FOR THE CALCULATION OF MAXIMUM NORMAL AND SHEAR STRESSES IN ENCLOSURE MOUNTING BOLTS

For the calculation of the maximum normal and shear stresses in the mounting bolts of any enclosure under seismic disturbance, the following necessary assumptions and conventions are made:

a. *The enclosure under consideration is assumed to be a rigid body in equilibrium with respect to its mounting.*

b. *The forces on the enclosure due to seismic accelerations are assumed to act through the enclosure's center of gravity.*

c. *The enclosure is assumed to have a known weight W as well as a known center of gravity located at X, Y, Z with respect to a right-handed coordinate system.*

d. *The right-handed coordinate system is arbitrarily assumed to be located at the front left-hand lower corner of the enclosure with the positive X-axis to the right along the front edge, the positive Y-axis toward the back of the enclosure, and the positive Z-axis toward the top of the enclosure.*

e. *The stresses on the enclosure mounting bolts are assumed to be greatest when the horizontal component of the floor acceleration is perpendicular to a side of the enclosure and the vertical component of the acceleration is downward.*

f. *It is assumed that the enclosure tends to rotate about an axis parallel to either the X-axis or the Y-axis, dependent upon the direction of the horizontal acceleration. The location of the axis of rotation is dependent upon the mounting configuration of the enclosure.*

g. *There is assumed to be no friction between the enclosure and its mounting.*

h. *The horizontal shear force due to the horizontal component of the acceleration is assumed to be distributed equally among the mounting bolts.*

i. *All mounting bolts are assumed to be identical.*

The following procedure outlines the equations involved in determining the mounting bolt stresses.

From the geometric configuration of the mounting bolts it is found that the tension forces in the bolts are related by

$$F_i = \frac{d_i}{d_j} F_j \qquad (1)$$

where F_i and F_j are the tension forces acting on the i-th and the j-th bolts, respectively, and d_i and d_j are the perpendicular distances of the i-th and the j-th bolts, respectively, from the axis about which the enclosure tends to rotate. When the enclosure is mounted directly to the floor, the axis of rotation will be an edge of the enclosure. For other mounting configurations, care must be exercised in determining this axis.

Summing moments about the enclosure's axis of rotation, the equation relating the unknown bolt tension forces to known quantities is found to be

$$F_1 d_1 + F_2 d_2 + \ . \ . \ . \ + F_N d_N = W[A1 \ . \ Z + (A2\text{-}1)L] \qquad (2)$$

where N is the number of mounting bolts, A1 and A2 are the relative magnitudes of the horizontal and vertical components of the floor acceleration, respectively, and L is the perpendicular distance between the line of action of the vertical acceleration through the center of gravity and the axis about which the enclosure tends to rotate.

Substituting (1) into (2), the j-th tension force is

$$F_j = \frac{d_j \cdot W\left[A1 \cdot Z + (A2 - 1)L\right]}{d_1^2 + d_2^2 + \cdots + d_N^2} \qquad (3)$$

The other tension forces are determined using Equation (1).

The tension stress T_i is related to the tension force by

$$T_i = \frac{F_i}{A} \qquad (4)$$

Where A is the effective cross-sectional area of a mounting bolt.

Summing forces in the direction of the horizontal force acting upon the enclosure and making use of assumptions 7 and 8, the shear stress on the i-th bolt is

$$S_i = \frac{W \cdot A1}{N \cdot A} \qquad (5)$$

Due to the combined tension and shear stresses, the maximum tension stress, $(T_i)_{max}$; and the maximum shear stress, $(S_i)_{max}$, present in the i-th bolt are (1):

$$(T_i)_{max} = \frac{T_i}{2} + \sqrt{\left(\frac{T_1}{2}\right)^2 + (S_i)^2} \qquad (6)$$

and

$$(S_i)_{max} = \sqrt{\left(\frac{T_i}{2}\right)^2 + (S_i)^2} \qquad (7)$$

To apply the above equations to determine the maximum tension and shear stresses, the following is required:

Total Weight	*W (pounds)*

Center of Gravity	*X, Y, Z (in.)*
Horizontal Seismic Acceleration	*A1(G)*
Vertical Seismic Acceleration	*A2(G)*
Distance to CG (see eq. (2))	*L (in.)*
Number of Bolts	*N*
Area Each Bolt	*A* ($in.^2$)
Bolt distance from Axis of Rotation	$d_1, d_2 \ldots d_N$*(in.) **

The procedure to be used as follows:

a. *Determine the axis about which the cabinet tends to rotate for a given floor motion.*

b. *Determine, using Equation (3), the tension force acting on the j-th mounting bolts (arbitrarily choose one).*

c. *Determine the tension forces acting on the remaining mounting bolts from application of Equation (1).*

d. *Calculate the tension stress acting on each bolt using Equation (4) and the results of Step 3.*

e. *Calculate the horizontal shear stress from Equation (5).*

f. *Determine the maximum tension stresses using Equation (6) and the results of Steps 4 and 5.*

g. *Determine the maximum shear stresses using Equation (7) and the results of Steps 4 and 5.*

h. *Compare these maximum stresses and allowable stresses of one half the ultimate strength (in PSI) for the bolt material.*

3.10B.3 VERIFICATION OF MOUNTING BOLT SEISMIC WITHSTAND CAPABILITY

3.10B.3.1 Purpose

The purpose of the sample analysis is to document a particular static seismic analysis which was performed to verify that the mounting bolts of the standard cabinets are capable of withstanding seismic environment.

3.10B.3.2 Scope

The scope of this sample analysis is limited to the static analysis of the mounting bolt stresses of five (5) standard cabinets. The standard cabinets are:

a. *Area Radiation Monitor, 236x400 (911)*
b. *TIP Control, 236x401 (913)*
c. *Start-up Neutron Monitor, 236x402 (936)*
d. *Power Range Monitor, 236x403 (937)*
e. *Rod Position Information System, 236x404 (927)*

3.10B.3.3 Discussion

GE Seismic Design Guide, 225A4582, was used in conducting the static seismic analysis. Each cabinet was assumed to be floor mounted using 5/8 in. bolts in all mounting holes. The maximum safe tension stress and maximum safe shear stress were assumed to be 28,000 psi and 21,000 psi, respectively (equal to one-half the ultimate strength as given in Machinery's Handbook, Fourteenth Edition). The design basis earthquake was assumed to have a horizontal acceleration of 1.5g and a vertical acceleration of 0.5g. The weight of each cabinet was estimated using the weight of each major component listed in the parts lists for each cabinet. The height of the center of gravity of each cabinet was calculated using the weight and center of gravity of each of the major components.

The following includes the necessary information for determining the factor of safety for each cabinet:

Cabinet Name: Area Radiation Monitor 236x400

Applied Horizontal Acceleration	*1.5g*
Applied Vertical Acceleration	*0.5g*
Tension Stress (Maximum Safe)	*28,000 psi*
Shear Stress (Maximum Safe)	*21,000 psi*
Weight of Cabinet	*675 lb*

Number of Mounting Bolts	*4*
Height of Center of Gravity	*48 in.*
Maximum Allowable Weight Per Bolt (From Curve No. C1 on Page 8 of Seismic Design Guide 225A4582)	*830 lb/bolt*
*Maximum Allowable Cabinet Weight 830 lbs/bolt * 4 bolts =*	*3320 lb*

$$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} = 4.9$$

Cabinet Name: TIP Control, 236x401 (913)

Applied Horizontal Acceleration	*1.5g*
Applied vertical Acceleration	*0.5g*
Tension Stress (Maximum Safe)	*28,000 psi*
Shear Stress (Maximum Safe)	*21,000 psi*
Weight of Cabinet	*755 lb*
Number of Mounting Bolts	*8*
Height of Center of Gravity	*50 in.*
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	*1110 lb*
*Maximum Allowable Cabinet Weight 1110 lbs/bolt * 8 bolts =*	*8,880 lb*

$$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} = 11.7$$

Cabinet Name: Start-Up Neutron Monitor, 236x402 (936)

Applied Horizontal Acceleration	*1.5g*

Applied Vertical Acceleration	*0.5g*
Tension Stress (Maximum Safe)	*28,000 psi*
Shear Stress (Maximum Safe)	*21,000 psi*
Weight of Cabinet	*1910 lbs.*
Number of Mounting Bolts	*12*
Height of Center of Gravity	*50 in.*
Maximum Allowable Weight Per Bolt (From Curves No. C3 on Page 8 of Seismic Design Guide, 225A4582)	*1110 lb/bolt*
*Maximum Allowable Cabinet Weight 1110 lbs/bolt * 12 bolts =*	*13,320 lb*

$$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} = 11.9$$

Cabinet Name: Power Range Monitor, 236x403 (937)

Applied Horizontal Acceleration	*1.5g*
Applied Vertical Acceleration	*0.5g*
Tension Stress (Maximum Safe)	*28,000 psi*
Shear Stress (Maximum Safe)	*21,000 psi*
Weight of Cabinet	*4345 lb*
Number of Mounting Bolts	*40*
Height of Center of Gravity	*46 in.*
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	*1210 lb/bolt*
*Maximum Allowable Cabinet Weight 1,210 lbs/bolt * 40 bolts =*	*48,400 lb*

$$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} = 11.1$$

Cabinet Name: Rod Position Information System, 236x404 (927)

Applied Horizontal Acceleration	*1.5g*
Applied Vertical Acceleration	*0.5g*
Tension Stress (Maximum Safe)	*28,000 psi*
Shear Stress (Maximum Safe)	*21,000 psi*
Weight of Cabinet	*2500 lb*
Number of Mounting Bolts	*20*
Height of Center of Gravity	*45 in.*
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	*1225 lb/bolt*
*Maximum Allowable Cabinet Weight 1225 lb/bolt * 20 bolts =*	*24,500 lb*

$$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} = 9.8$$

3.10B.3.4 Conclusion

Review of the Factor of Safety of each standard cabinet indicates that the mounting bolts of each cabinet are capable of withstanding seismic disturbance as specified in the GE Seismic Design Guide.

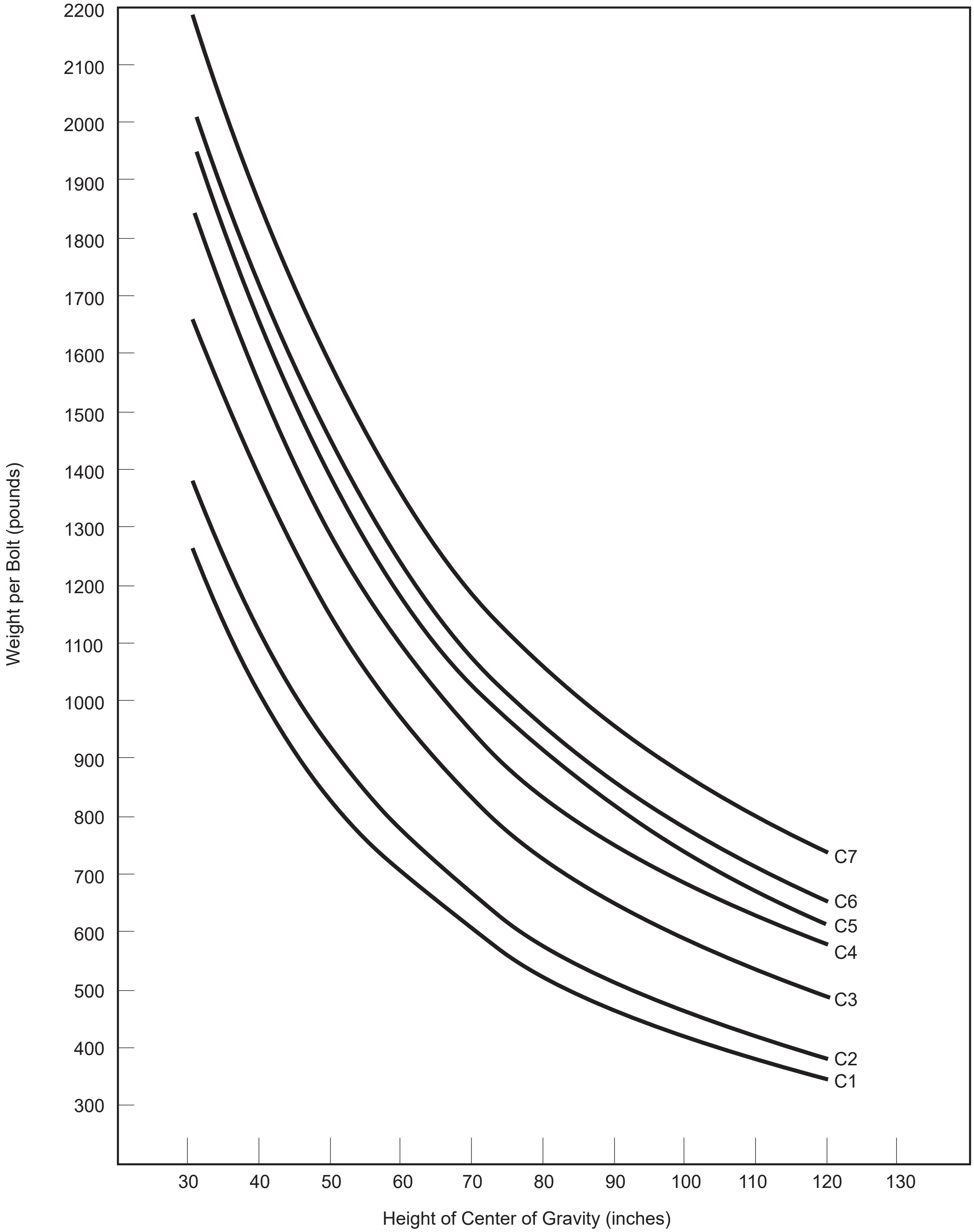
3.10B.4 REFERENCES

3.10B.4-1 Singer, Ferdinand L., Strength of Materials, Chapter 9, Section 6.

Table 3.10B-1

Standard Enclosures

Curve	*Enclosure*	*Width*	*Depth*	*Mode of Failure*
C1	*Instrument Rack*	*24 in.*	*24 in.*	*S-S*
C1	*Instrument Rack*	*24 in.*	*30 in.*	*S-S*
C1	*Vertical Board*	*24 in.*	*24 in.*	*S-S*
C1	*Vertical Board*	*24 in.*	*30 in.*	*S-S*
C1	*Benchboard*	*24 in.*	*48 in.*	*S-S*
C1	*Benchboard*	*24 in.*	*54 in.*	*S-S*
C2	*Instrument Rack*	*30 in.*	*30 in.*	*F-B or B-F*
C2	*Instrument Rack*	*30 in.*	*24 in.*	*F-B or B-F*
C2	*Instrument Rack*	*48 in.*	*24 in.*	*F-B or B-F*
C2	*Instrument Rack*	*60 in.*	*24 in.*	*F-B or B-F*
C2	*Instrument Rack*	*72 in.*	*24 in.*	*F-B or B-F*
C2	*Instrument Rack*	*96 in.*	*24 in.*	*F-B or B-F*
C2	*Vertical Board*	*36 in.*	*24 in.*	*F-B or B-F*
C2	*Vertical Board*	*48 in.*	*24 in.*	*F-B or B-F*
C2	*Vertical Board*	*60 in.*	*24 in.*	*F-B or B-F*
C2	*Vertical Board*	*72 in.*	*24 in.*	*F-B or B-F*
C2	*Vertical Board*	*96 in.*	*24 in.*	*F-B or B-F*
C3	*Instrument Rack*	*48 in.*	*30 in.*	*F-B or B-F*
C3	*Instrument Rack*	*60 in.*	*30 in.*	*F-B or B-F*
C3	*Instrument Rack*	*72 in.*	*30 in.*	*F-B or B-F*
C3	*Instrument Rack*	*96 in.*	*30 in.*	*F-B or B-F*
C3	*Vertical Board*	*36 in.*	*30 in.*	*F-B or B-F*
C3	*Vertical Board*	*48 in.*	*30 in.*	*F-B or B-F*
C3	*Vertical Board*	*60 in.*	*30 in.*	*F-B or B-F*
C3	*Vertical Board*	*72 in.*	*30 in.*	*F-B or B-F*
C3	*Vertical Board*	*96 in.*	*30 in.*	*F-B or B-F*
C4	*Console*	*96 in.*	*42 in.*	*B-F*
C5	*Benchboard*	*48 in.*	*54 in.*	*S-S*
C5	*Benchboard*	*48 in.*	*48 in.*	*S-S*
C6	*Benchboard*	*72 in.*	*48 in.*	*F-B*
C6	*Benchboard*	*96 in.*	*48 in.*	*F-B*
C6	*Console*	*96 in.*	*48 in.*	*F-B*
C7	*Benchboard*	*72 in.*	*54 in.*	*B-F*
C7	*Benchboard*	*96 in.*	*54 in.*	*B-F*



Columbia Generating Station Final Safety Analysis Report	Maximum Safe Weight Per Bolt for Standard Enclosure as a Function of the Height of the Center of Gravity
	Draw. No. 990306.88 / Rev. / Figure 3.10B-1

Appendix 3.10C

SAMPLE PANEL FREQUENCY ANALYSIS FOR NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT

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SAMPLE PANEL FREQUENCY ANALYSIS FOR NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT

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SAMPLE PANEL FREQUENCY ANALYSIS FOR NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT

3.10C.1 METHOD OF ANALYSIS

The method of analysis used to determine the resonant frequency of the panel is as follows:

a. *Calculate the moment of inertia of the corner post structure.*

b. *First assume a simplified structure and calculate the frequency using the expression:*

$$f = 1 / 2\pi\sqrt{Kg / w} = \left(\sqrt{g / 2\pi}\right)\left(\sqrt{k / w}\right) = 3.13/\sqrt{w / k}$$

$$f = 3.13/\sqrt{\sigma}$$

where

f = *frequency, Hz*

g = *386 in./sec*^{2}

k = *spring rate lbs/in.*

w = *weight lbs*

σ = *deflection* = w/k = *in.*

weight distribution is assumed to be uniform.

c. *Additional structural components are added and the moment and frequency recalculated.*

The calculated resonant frequency of 7.4 Hz for the panel and 5.9 Hz for the benchboard was obtained using only the corner posts and the top. The addition of skin (3/8-in. steel) and 2-in. x 1/4-in. steel stiffeners will raise the frequency further. This proves that resonances cannot exist in the unstable region below 5 Hz.

3.10C.2 FIRST APPROXIMATION

For first approximation lump the four corner posts together and assume the panel is a cantilever beam fixed on one end and uniformly loaded (see Figure 3.10C-1).

The natural frequency is 2.6 Hz resulting in the use of more of the structure.

3.10C.3 SECOND APPROXIMATION

For a second approximation, consider two 0.18-in. x 30-in. barriers in addition to the corner posts. The plan view of the panel is shown in Figure 3.10C-2.

In the X direction just one barrier will raise the frequency to 30 Hz. Use 4 in. of the back panel for each of the two barriers (see Figure 3.10C-3) and the natural frequency in the Y direction becomes 4 Hz.

3.10C.4 DEFLECTION

The deflection equation used so far is very conservative. It assumes that the four corner posts are lumped together and that the structure can reflect like a simple cantilever beam. Actually the corners are separated by an angle frame which is stiffer than the corner posts. This will force the structure to deflect as shown in Figure 3.10C-4.

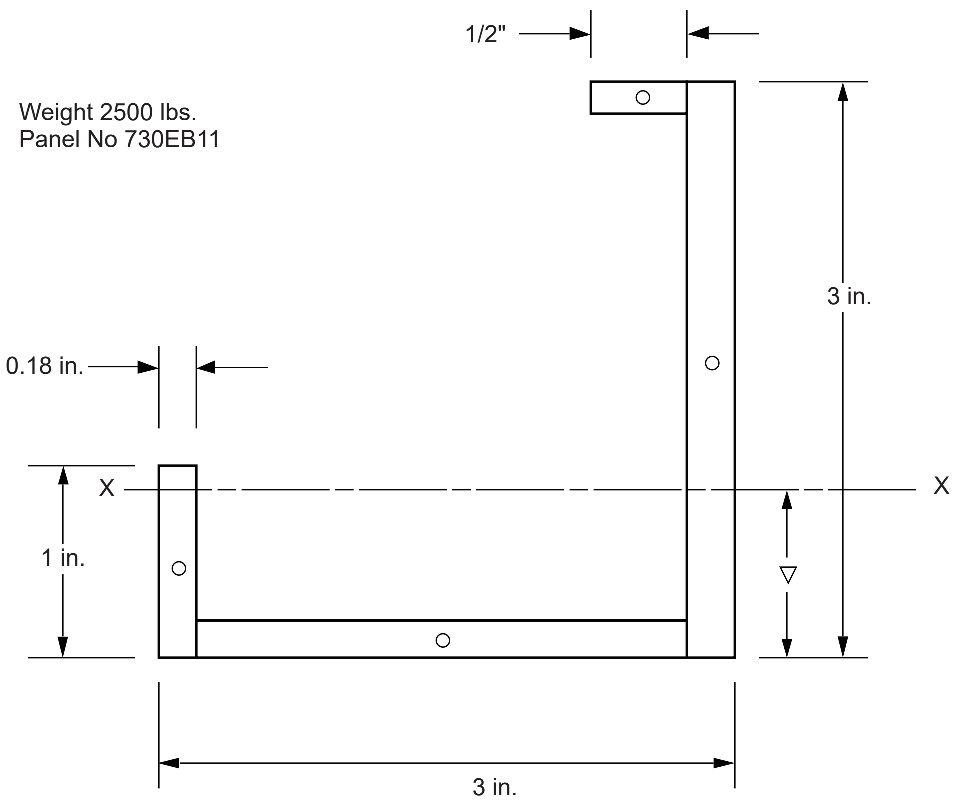
The stiff top frame will also deflect slightly as shown in Figure 3.10C-5. The calculated frequency using this conservative panel frequency analysis method is 7.4 Hz which is above the necessary 5 Hz.

For benchboard H13-P601 which weighs 4,000 lbs, the calculated natural frequency is 5.9 Hz which is still above the 5 Hz test frequency minimum.

NOTE: This neglects the barriers, the end and front panels, top plate, the stiffening of the lower part of the structure due to the bench board geometry, and all other members of the structure.

3.10C.5 ASSESSMENT OF CONSERVATIVENESS

In order to assess the conservativeness of the above method for determining benchboard and panels natural frequency, analysis using a finite element model of Benchboard H13-P601 was performed by the response spectra analysis method described in Appendix 3.10A. This was performed because of the addition of some heavy components to the benchboard in the field. Due to its increased weight, the original 5-9 Hz natural frequency needed to be evaluated as increased weight results in lowering of the natural frequency. The results of the subsequent analysis yielded a natural frequency of 12.9 Hz for the benchboard. This result conclusively verified the conservativeness the above approach.



Columbia Generating Station
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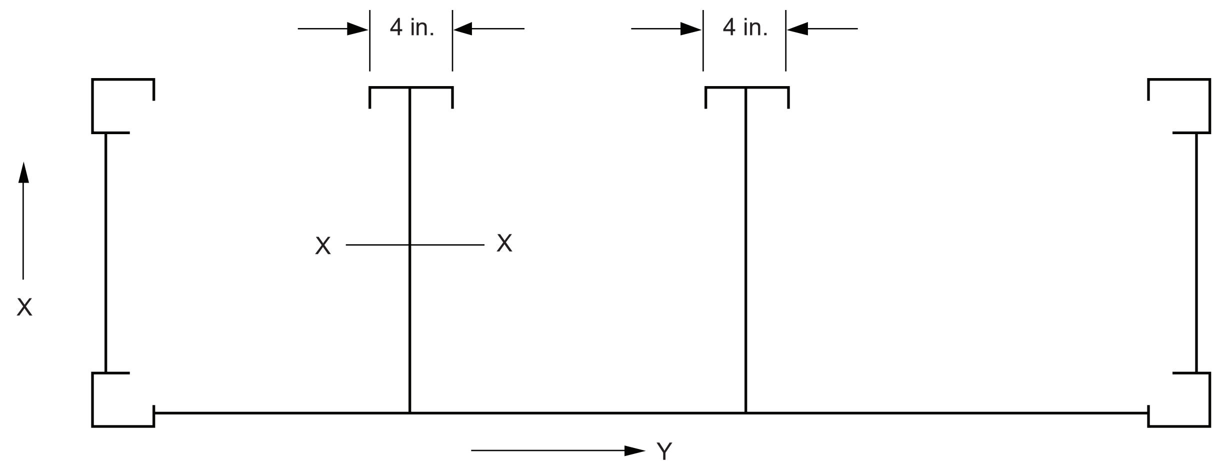
Corner Post

Draw. No. 990306.89	Rev.	Figure 3.10C-1



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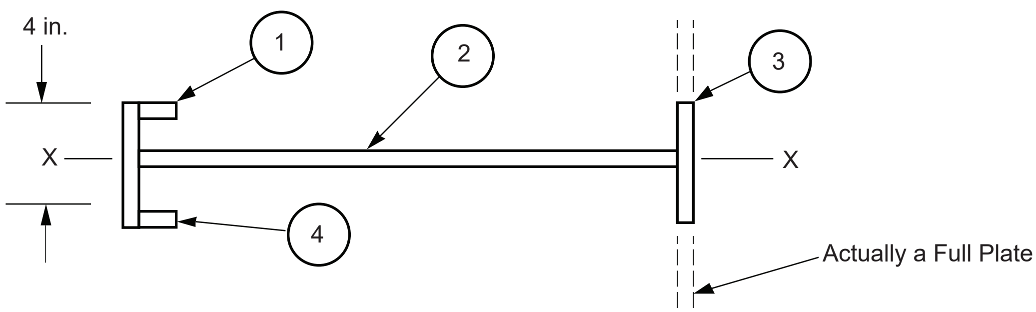
Plan View of Panel

Draw. No.	Rev.	Figure
990306.90		3.10C-2



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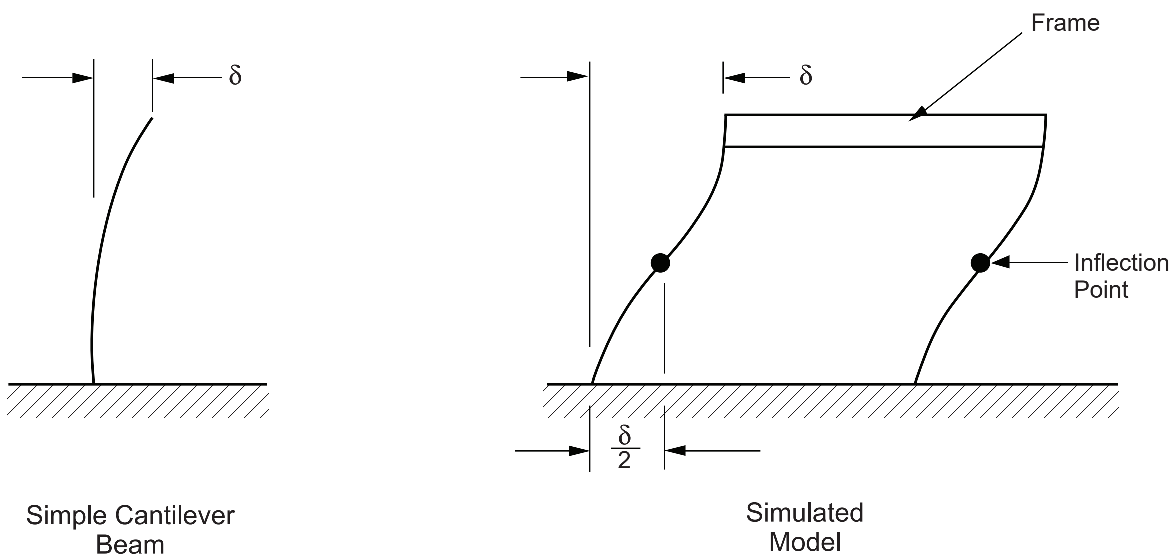
Barrier with Two End Plates

Draw. No. 990306.91 | Rev. | Figure 3.10C-3



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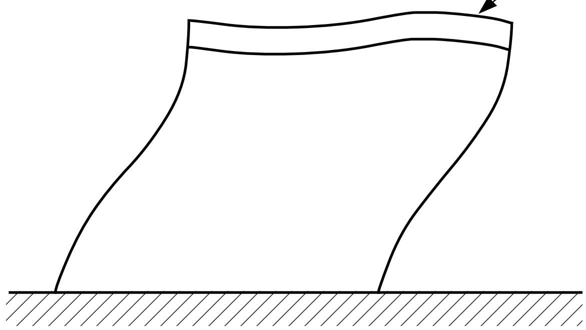
Panel Deflections

Draw. No. 990306.92 | Rev. | Figure 3.10C-4



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Top Frame Deflection

Draw. No.	990306.93	Rev.		Figure	3.10C-5

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section provides information on the environmental conditions and design bases for which the instrumentation and electrical portions of the engineered safety features (ESF) and reactor protection systems (RPS) have been designed and qualification documentation generated to ensure acceptable performance in all environments in which the equipment is or to which it may be potentially exposed.

The design bases for equipment qualification for CGS safety-related equipment is IEEE 323, 1971. This standard was selected as the design basis for qualification requirements contained in specifications and equipment purchase orders. Equipment suppliers were required to provide certification that supplied equipment meets the requirements of IEEE 323, 1971, for equipment intended for installation in potentially harsh environment areas of CGS.

Subsequent to NRC acceptance of this design basis at the construction permit stage, the NRC required evaluation of the equipment's environmental qualification to NUREG-0588 Category II. The NRC also required that equipment whose qualification documentation does not meet the requirements set forth in NUREG-0588 Category II be re-qualified or justification provided as to why the existing documentation is sufficient.

Subsequent to the issuance of NUREG-0588 the NRC amended 10 CFR Part 50.49. This expanded the scope of equipment to be considered for electrical equipment qualification and required that replacement of electrical equipment be qualified to the provisions of 10 CFR Part 50.49 unless sound reasons to the contrary are provided. The CFR also instituted a requirement to provide an analysis as to why the plant can be operated safely, pending completion of any requalification or equipment replacement action that could not be accomplished prior to operation of CGS.

The CGS Equipment Qualification Program includes two major subprograms. They are the CGS Dynamic Qualification Program and the CGS Environmental Qualification Program. The CGS Environmental Qualification Program, discussed within this section, is designed to ensure that electrical equipment important to safety is qualified to the provisions of NUREG-0588 Category II and applicable provisions of 10 CFR Part 50.49. Details of the initial CGS environmental qualification program are provided in Reference 3.11-1. The following summarizes the current CGS environmental qualification program.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

3.11.1.1 Identification of Electrical Equipment Important to Safety

The criteria for identification of electrical equipment important to safety are defined in Section 3.11.1.1.1 through 3.11.1.1.4. The electrical equipment defined in these sections is included in the Master Equipment List (MEL).

Electrical equipment in these sections may experience the conditions of design basis accidents due to their plant location. In addition, the cumulative gamma radiation dose to equipment in these areas is, in general, above 1 x 10^4 rad. This equipment is located in the primary and secondary containment areas of the CGS reactor building as well as certain areas in the radwaste building.

Safety-related equipment outside of these areas and in some reactor building electrical equipment rooms is not exposed to a significant change from the normal service environment or anticipated operating occurrences as a result of design basis accidents and therefore are not a part of the environmental qualification program. In addition, the cumulative gamma radiation dose to equipment in these areas is generally below 1 x 10^4 rad. Also excluded from the environmental qualification program is safety-related electrical equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of such accidents; or that has performed its safety function prior to the exposure to an accident environment; and whose failure (in any mode) is deemed not detrimental to plant safety or accident mitigation and will not mislead the operator. This equipment has been identified and included in the CGS dynamic qualification program described in Section 3.10.

Electrical equipment important to safety that only performs a mechanical safety function, such as the maintenance of mechanical pressure integrity, and whose electrical failure (in any mode) or change of state is of no safety significance, is considered to be similar to mechanical devices and is not addressed by the CGS environmental qualification program.

The following equipment specific data was determined for electrical equipment important to safety and is included in either the MEL or Passport databases:

a. Equipment piece number (i.e., unique plant tag number),

b. Manufacturer,

c. Model number or manufacturer's identification reference,

d. Active or passive classification,

e. Equipment use classification per the equipment categories defined in Appendix E of Regulatory Guide 1.89, Revision 1,

f. System/component level safety function(s),

g. Equipment plant location, and

h. Equipment period of operability during accident condition.

3.11.1.1.1 Engineered Safety Features and Reactor Protection System Equipment

An ESF is a safety-related system that provides a safety function to mitigate the consequences of a design basis accident that may cause major fuel damage. An ESF includes the primary auxiliary systems of the safety system. The identification, location, and accident environmental design bases for these safety-related systems and/or components are provided in the MEL, calculations, and supporting documents. This equipment includes safety-related electrical equipment (Class 1E) that is relied on to remain functional during and following accident exposure by loss-of-coolant accidents (LOCA), main steam line break (MSLB) accidents, high-energy line break (HELB) accidents, and rod drop accidents. The electrical equipment included in the program is that equipment necessary to ensure the

a. Integrity of the reactor coolant pressure boundary,

b. Capability to shut down the reactor and maintain it in a safe shutdown condition, or

c. Capability to prevent or mitigate the consequences of the above accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines.

3.11.1.1.2 Postaccident Monitoring Electrical Equipment

In addition to Class 1E equipment identified in Section 3.11.1.1.1, electrical and instrumentation equipment identified by Regulatory Guide 1.97, Revision 2 (Category I and II), and NUREG-0737 are included in the environmental qualification program. Instruments meet the requirements by category and type as described in Regulatory Guide 1.97, Revision 2, unless noted in text discussions as meeting Regulatory Guide 1.97, Revision 3, requirements (see Section 7.5.2.2.3).

3.11.1.1.3 Other Electrical Equipment Important to Safety

Included in the environmental qualification program is equipment that is not classified as safety-related (Class 1E). This equipment may experience the environmental conditions of

design basis accidents and the results of failure of this equipment due to the design basis accidents may prevent satisfactory accomplishment of the safety function of safety-related equipment. Qualification requirements for electrical equipment required to mitigate anticipated transients without scram (ATWS) or used for remote shutdown are provided in Section 1.5.1.1.2, 7.1.2.4, and 7.4.2.3.

3.11.1.1.4 Electrical Equipment Whose Failure Does Not Affect Safety Function Performance

In the development of the list of electrical equipment important to safety, electrical equipment was identified that may be exposed to environments created by design basis accidents whose failure during or following the accident exposure would have no affect on safety function performance. Failure modes and effect evaluations were performed to justify their removal from the environmental qualification program.

3.11.1.1.5 Safety-Related Mechanical Equipment

Safety-related mechanical equipment is not included in the environmental qualification program. This includes ATWS equipment with only a mechanical augmented quality function. The design verification requirements cited in Reference 3.11-2 are met outside the environmental qualification program.

3.11.1.1.6 Identification of Safety-Related Equipment In Mild Environmental Plant Areas

Safety-related electrical equipment located in areas of the plant that are not exposed to significant environmental effects resulting from design basis accidents are considered mild environments and are also identified within the CGS equipment qualification program.

Equipment in ESF and RPS located in mild environments will not experience a significant change in its environmental conditions and, therefore, need only be qualified to the service conditions resulting from seismic design basis events. This also includes equipment that may be exposed to a total integrated radiation dose (normal plant operations including anticipated operational occurrences + accident) in excess of 1×10^4 rad if the accident radiation does not significantly exceed the 40-year integrated radiation dose due to normal plant operations and anticipated operational occurrences. Safety-related equipment located in a mild environment where the total integrated radiation dose exceeds 1×10^4 rad, or exceeds 1×10^3 rad for equipment containing certain classes or categories of solid state electronics as defined within qualification program procedures and supporting documents, have been specifically excluded from the environmental qualification program. However, this equipment is either procured to the required radiation dose provided in the procurement specifications or is evaluated for acceptable operability due to radiation exposure effects only, in its service environment. The identification of this equipment is provided in the MEL. See Section 3.10 for additional details.

3.11.1.2 Normal and Accident Environmental Service Conditions

The normal and accident environmental service conditions have been defined for electrical equipment important to safety and are identified in this section. The environmental parameters include temperature, pressure, relative humidity, demineralized water spray, submergence, and radiation.

3.11.1.2.1 Normal Plant Operational Environmental Conditions (Except Radiation)

Plant operational environmental conditions are without significant abnormalities and include the following: planned startups, power range, shutdown, and normal hot standby. These environmental conditions, specified for the various buildings and/or areas, are listed in Table 3.11-1 and represent normal temperatures and maximum and minimum design bases for temperature, relative humidity, and pressure to which equipment may be exposed during routine plant operations and short-term abnormal conditions due to anticipated operating occurrences. See Figure 3.11-1 for primary containment zone locations.

3.11.1.2.2 Accident Environmental Service Conditions (Except Radiation)

The primary containment, most areas of the secondary containment within the reactor building, and certain areas in the radwaste building, potentially could be exposed to elevated temperature, pressure, and high humidity conditions resulting from LOCA, MSLB, or HELB type accidents.

Pressure and temperature profiles were defined for two accident types: LOCA/MSLB in primary containment and HELBs in the secondary containment of the reactor building and certain areas of the radwaste building. Break locations were determined (see Section 3.6), and equipment required to operate during or following exposure to the environmental conditions resulting from the breaks identified.

The CGS plant-specific pressure/temperature profiles of the primary containment and the equipment required to operate during or following the postulated breaks are provided in the MEL, LOCA and MSLB analysis calculations, and supporting documents. Table 3.11-2 provides generic environmental conditions obtained from a GE analysis of the response of a BWR Mark II containment to a full spectrum of possible LOCA and MSLB. These generic environmental conditions were used for equipment qualification evaluations of equipment in primary containment. Plant-specific profiles are used for the postaccident period as required by NUREG-0588. In addition to pressure/temperature and humidity conditions inside primary containment, demineralized spray and potential flooding was included in the definition of the environmental service conditions.

Plant-specific pressure/temperature and humidity profiles were also defined for equipment spaces within the secondary containment areas of the reactor building and certain areas of the radwaste building. General Electric generic conditions were not used. These profiles are provided in

HELB analysis calculations and supporting documents. In addition to pressure/temperature and humidity, the affect of flooding was evaluated. The results of the flooding analysis determined that CGS can be safely shut down with alternative safety-related equipment not affected by flooding. Based on this, flooding is not a required environmental service condition for equipment qualification in the secondary containment except for the high-pressure core spray (HPCS), low-pressure core spray (LPCS), reactor core isolation cooling (RCIC), and residual heat removal (RHR) reactor building pump rooms.

3.11.1.2.3 Radiation Service Conditions

The primary containment radiation conditions determined for environmental qualification include the normal (40-year dose) and the accident radiation dose (180 days). The criteria for determining the radiation dose were NUREG-0737 and NUREG-0588. The calculated radiation environment is based on normal service conditions, plus the most severe nonmechanistic design basis accident during or following which equipment must function. Required period of operability for the equipment was a factor used to determine equipment specific radiation dose. The accidents considered are the entire spectrum of Chapter 15 accidents that can lead to a degraded core condition. The source term assumptions for postulated accidents are consistent with those defined in NUREG-0588 and Regulatory Guides 1.3 and 1.7.

The secondary containment radiation conditions are defined according to Section II.B.2 of NUREG-0737 and NUREG-0588. The total integrated dose includes the sum of direct accident gamma dose, airborne gamma dose, 40-year normal gamma dose, and where equipment could be beta sensitive, a beta dose. This beta dose was determined through use of energy dependent geometry factors, a ratio of the internal equipment volume to an infinite cloud, and the dose to a target at the center point face for a hemispherical cloud of gases.

The dose and dose rate were determined for equipment areas within the secondary containment. The secondary containment was divided into radiation zones to define the equipment radiation service conditions. The worst target (component with the highest dose) in each zone was chosen. This radiation value was then used as a screening value for all equipment in the zone. In some zones, additional targets were chosen to more realistically map the radiation conditions throughout some zones.

The methodology and results of the radiation evaluations are provided in CGS radiation dose calculations. The results of evaluations for specific equipment to radiation conditions are contained in the CGS Qualification Information Documents (QIDs).

3.11.1.2.4 Accident Conditions - Harsh Environments

The primary containment, most areas of the reactor building, and certain areas of the radwaste building will be exposed to a harsh environment following a postulated LOCA/HELB.

A harsh environment is defined as:

> An area that would be exposed to a significant increase in the maximum temperature, pressure, and humidity as a direct result of design basis events and/or the total radiation dose (normal + accident) is above 1 x 10^4 rad, or above 1 x 10^3 rad for certain classes or categories of solid state electronics, with a significant increase in radiation dose during accident conditions compared to that during normal plant operations, including anticipated operational occurrences.

3.11.2 QUALIFICATION TESTS AND ANALYSES

In accordance with NUREG-0588 Category II requirements, descriptions of the qualification tests and analyses have been provided in the QIDs. These QIDs are prepared in accordance with Energy Northwest Engineering Standards. The information provided is consistent with Appendix E of NUREG-0588 and includes the methods employed to address temperature, pressure, humidity, spray, radiation, and aging.

Specific values (both required and demonstrated) are provided in equipment qualification summary sheets in the QIDs.

These summary reports indicate how the general requirements of General Design Criteria 1, 4, 23, and 50 of Appendix A to 10 CFR Part 50, and 10 CFR Part 50.49 are met. The CGS equipment qualification program meets Criterion III of Appendix B to 10 CFR Part 50 through implementation of Quality Class 1 procedures for all aspects of the evaluation, documentation, and corrective action resulting from the program.

Section 1.8 addresses how Regulatory Guides 1.30, 1.40, 1.63, 1.73, and 1.89 were applied to the CGS qualification program.

3.11.3 QUALIFICATION PROGRAM RESULTS

The results of qualification test and analysis for each piece of equipment in the CGS Environmental Qualification Program is provided in the QIDs. A typical Equipment Qualification Report Summary from a QID is shown in Figure 3.11-2.

3.11.3.1 NRC Review of the CGS Environmental Qualification Program

The NRC staff evaluation of the CGS Environmental Qualification Program included an examination of the installed electrical equipment and audits of both electrical and mechanical equipment qualification documentation. A review of Energy Northwest's submittals for completeness and acceptability of systems and components, qualification methods, and accident environments was also conducted. The criteria described in Standard Review Plan Section 3.11 (NUREG-0800) and NUREG-0588, Category II, formed the basis for the NRC's evaluation of the adequacy of the CGS program.

An audit was performed of the CGS qualification documentation and installed equipment in February 1983 and was documented by the NRC in NUREG-0892, Supplement 3.

A subsequent audit was made to assess implementation of the concerns and resolutions raised in the February 1983 audit. These files were found acceptable by the NRC as reported in Supplements 4 and 5 of NUREG-0892, Reference 3.11-2.

Mechanical equipment is no longer a part of the environmental qualification program and is addressed separately (see Section 3.11.1.1.5).

3.11.3.2 Establishment of Operational Phase Environmental Qualification Review

Energy Northwest has established an operational phase environmental qualification review. This review is integrated with design changes to CGS to ensure compliance with the criteria as described in Section 3.11.

The operational phase qualification review is required for all design changes that add, modify, or delete safety-related equipment for CGS because of plant betterment, regulatory, or license conditions requirements.

Design control procedures establish a special qualification review as part of the design change package preparation. Adequacy of the equipment selected to be added to the plant in the design change is assessed and documented to an "As Designed" qualification status. Special procurement requirements are detailed for the purchase order. During or after installation, quality control inspections are conducted to ensure final installation conforms to design documents and final qualification documentation is established in QID files. A walkdown is performed by equipment qualification personnel if needed to clarify the final installation configuration.

For design changes that modify or reclassify existing plant equipment, a special review is also required similar to the above. Final documentation is established in QID files on completion of the design change.

For design changes that delete existing plant equipment or change its status to non-safety-related, a special review is also conducted. In these instances, modification of the QID file is considered optional under the program.

Design control procedures also require the safety-related equipment list in the MEL is to be kept current with actual plant configuration. Thus, consonance of the plant equipment configuration and qualification documentation (QID files) is maintained throughout plant life.

Plant modifications to environmental control systems that alter environmental profiles to which equipment has previously been qualified are reviewed to ensure equipment is qualified to revised profiles.

The NRC Bulletins and Information Notices that affect qualification of plant equipment are reviewed and factored into the qualification review. Design changes that result from NRC Bulletin and Information Notice reviews also receive the special qualification review.

The qualification program is linked with the maintenance program to ensure that required maintenance or replacement due to qualified life considerations are performed. Refinement of the qualified life is included based on actual service conditions and actual equipment degradation evaluations through monitoring, inspection, and surveillance programs.

Replacement equipment is qualified in accordance with Regulatory Guide 1.89, Revision 1. Where necessary, engineering evaluations are performed using the criteria and guidance for establishing sound reasons to the contrary provided in Regulatory Guide 1.89, Revision 1, and Generic Letter 82-09, as applicable.

3.11.4 LOSS OF VENTILATION

3.11.4.1 Main Control Room Air Conditioning and Ventilation System

Controls and electrical equipment necessary for safe plant shutdown are located in the main control room. The control room is air conditioned and shielded against radiation to allow the operators safe and continued occupancy under optimum environmental conditions. Air conditioning equipment and associated components are designed to Seismic Category I requirements. Redundant equipment is provided and on loss of offsite power, emergency power from the onsite diesel generator sets is automatically supplied to the equipment. No single failure can result in loss of control room air conditioning. Hence, design of the system ensures that the operability of the safety-related control and electrical equipment located in the control room will not be impaired and will continue to function in an acceptable environment. Therefore, no special environmental design requirements for loss of ventilation or air conditioning need be incorporated in the design of safety-related electrical or instrumentation equipment located in the control room.

3.11.4.2 Reactor Building Emergency Cooling System

Some equipment located within the reactor building which must operate in the event of an accident could not survive direct exposure to the accident environment. This equipment is located in enclosed electrical equipment rooms. In the event of an accident, these rooms are automatically isolated from the normal reactor building ventilation system and subsequently are maintained at the required environmental conditions by the reactor building emergency cooling system (see Section 9.4.9).

The reactor building emergency cooling system consists of independent air recirculation systems each of which is fully enclosed within the room it serves and is designed to Seismic Category I standards. The air recirculation system for a room is powered from the same emergency diesel generator bus as the equipment in the room. Since each air recirculation system is independent and serves redundant emergency equipment systems, a failure of one air recirculation system does not affect the operation of other air recirculation systems or the safe shutdown of the reactor.

3.11.4.3 Miscellaneous Ventilation Systems

Each of the following areas are serviced by at least one ESF heating, ventilating, and air conditioning (HVAC) system:

b. Critical switchgear area, radwaste building (Section 9.4.1),

c. Emergency diesel generator building (Section 9.4.7),

d. Critical electrical cable runs, between the diesel generator building and the radwaste building (Section 9.4.8), and

e. Standby service water pump houses (Section 9.4.10).

Ventilation failures in these areas are considered and evaluated in Section 9.4. During some design basis events, short periods of elevated service water temperature may affect the normal temperatures of some mild environment rooms (see Section 9.4 and Table 3.11-1). No significant thermal aging will result from the temperature excursions. Design reviews were conducted to ensure that the resulting operating conditions will not cause safety-related equipment located in these areas to fail.

3.11.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

3.11.5.1 Suppression Pool, Residual Heat Removal System, and Emergency Core Cooling System Water Quality

The water in these systems is not chemically inhibited. The maximum limits for the suppression pool for normal operation have been established to be compatible with those of the primary coolant and are listed in Table 3.11-3 for comparison.

The fuel pool cooling and cleanup system has provisions for circulating the suppression pool water through demineralizers to maintain water quality limits.

During reactor shutdown cooling, the RHR system water mixes with reactor water. Therefore, to ensure reactor water quality, the shutdown cooling piping and equipment is flushed with water of the quality specified in Table 3.11-3 for maximum limit with suspended solids concentration of 5 ppm or less. During layup the RHR system will be filled with water which meets the limits specified in Table 3.11-3.

3.11.5.2 Qualification

Qualification of equipment for severe chemical exposure is considered unnecessary since no sources of detrimentally high or low pH have been identified for the postulated post-LOCA conditions. The post-LOCA water qualities would be expected to be similar to that shown in Table 3.11-3 (see Section 6.1.3). Radiation environment conditions and equipment qualifications are discussed in Section 3.11.1.2.3 and 3.11.2 respectively.

3.11.6 REFERENCES

3.11-1 WNP-2 Environmental Qualification Report for Safety-Related Equipment, dated September 1983 (historical document).

3.11-2 Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2, NUREG-0892, Supplement 3 (May 1983), Supplement 4 (December 1983), and Supplement 5 (April 1984).

Table 3.11-1
Normal Operating Conditions

	Area	Pressure (as noted)	Temperature (°F)	Relative Humidity (%)
I.	Primary containment (not otherwise noted)	0.25 to 0.75 psig (nominal)[m]	135 bulk average maximum[a] 70 minimum 150 maximum	40-55 normal NA - minimum 100 maximum
	Vicinity recirculation pump motors zone 4[b]	0.25 to 0.75 psig (nominal)[m]	70 minimum[c] 135 maximum	40-55 normal NA - minimum 100 maximum
	Area beneath RPV zone 3[b]	0.25 to 0.75 psig (nominal)[m]	70/100 minimum average[d] 165 maximum	40-55 normal NA - minimum 100 maximum
	Sacrifical shield wall lower/mid annulus	0.25 to 0.75 psig (nominal)[m]	70/100 minimum average[d] 185 maximum/150 average	40-55 normal NA - minimum 100 maximum
	Suppression pool air volume zone 6[b]	0.25 to 0.75 psig (nominal)[m]	50 minimum 150 maximum 117 (operating limit)	40-55 normal NA - minimum 100 maximum
	Suppression pool water volume		50 minimum Maximum - See Technical Specification 3.6.2.1	
II.	Reactor building (not otherwise noted)	Nominal -0.6 in. water gauge, static pressure (unless otherwise noted)	70-90 normal 40/50 minimum[l] 104 maximum[h]	40 normal 20 minimum 90 maximum
	RCIC equipment area		70-90 normal 60 minimum 104 maximum equip not running 150 maximum equip running	40 normal 20 minimum 90 maximum (e)

Table 3.11-1

Normal Operating Conditions (Continued)

Area	Pressure (as noted)	Temperature (°F)	Relative Humidity (%)
HPCS, LPCS and RHR equipment area		70-90 normal 40 minimum 104 maximum equip not running 150 maximum equip running	40 normal 20 minimum 90 maximum (e)
II. Reactor building (Continued)			
Steam tunnel		130 normal[i] 40 minimum 140 maximum	40-50 normal 20 minimum 98 maximum
RWCU piping and pump area (R406, R407, R409)		118 maximum	40-50 normal 20 minimum 98 maximum
Pipe space south (R405)		109 maximum	40-50 normal 20 minimum 98 maximum
Shielded pipe space south (R511)		111 maximum	40-50 normal 20 minimum 98 maximum
New fuel storage vault		116 maximum	40-50 normal 20 minimum 98 maximum
Refueling floor		40/64 minimum[j]	40-50 normal 20 minimum 98 maximum
III. Turbine building	0.0 in. to -0.25 in. water gauge, static pressure	70-90 normal 40 minimum (non-elec) 50 minimum (elec) 104 maximum (elec) 120 maximum (non-elec)	40 normal 20 minimum 90 maximum

Table 3.11-1
Normal Operating Conditions (Continued)

	Area	Pressure (as noted)	Temperature (°F)	Relative Humidity (%)
	Area above main steam lines		200 maximum	40 normal 20 minimum 90 maximum
IV.	Radwaste building	0.0 in. to -0.25 in. water gauge, static pressure	70 normal[f] 40/55/60 minimum[g] 104 maximum[h]	50 normal 90 maximum 20 minimum
	Radwaste building equipment cells	0.0 in. to -0.5 water gauge, static pressure	70 normal 40 minimum 120 maximum	40 normal 20 minimum 90 maximum
V.	Control room	0.10 in. to 1.0 in. water gauge, static pressure	72-78 normal 40 minimum 85/104 maximum[k]	40-50 normal 60 maximum 10 minimum
VI.	Critical switchgear rooms	0.0 in. to -0.25 in. water gauge, static pressure	70 normal 55 minimum 104 maximum (elec)[h]	40 normal 20 minimum 90 maximum
VII.	Diesel generator building	Ambient	60-95 normal[n]	60 normal 10 minimum 90 maximum
	Emergency diesel generator engine rooms		40 minimum 120 maximum	
	HPCS diesel generator engine room		40 minimum 112 maximum	
	Emergency diesel generator electrical switchgear rooms		50 minimum 104 maximum	

Table 3.11-1
Normal Operating Conditions (Continued)

	Area	Pressure (as noted)	Temperature (°F)	Relative Humidity (%)
	HPCS diesel generator electrical switchgear/ battery room		60 minimum 104 maximum	
	Cable Corridor		40 minimum 104 maximum	
VIII.	Service water pump building	0.0 in. to +0.25 in. water gauge, static pressure	50-104 normal 40 minimum 114 maximum (elec)[h]	60 normal 10 minimum 90 maximum

[a] Average or bulk temperature. See Technical Specification 3.6.1.4.

[b] See Figure 3.11-1 for zone locations.

[c] Minimum temperature for operational and maintenance for RRC pump loop piping general bolting materials specifications.

[d] The same minimum normal average operational temperature (100°F) shall apply at the inside base of the sacrificial shield wall. Air velocity over vessel insulation and exposed vessel parts shall be approximately 6 ft/sec. 70°F average applies when not in mode 1, or mode 2 or 3 with RPV piping >275 psig and 200°F.

[e] The maximum temperature and humidity will occur simultaneously in these spaces less than 1% of the time.

[f] Normal for cable spreading room is 80°F.

[g] Minimum for critical switchgear area is 55°F (60°F in battery rooms). LCS 1.7.1 lists minimum normal for battery rooms as ≥ 74°F to support Station Black Out (SBO) commitment.

[h] During design basis events this temperature may be exceeded for a short period of time (<30 days). These room temperatures are used to support the evaluated design capability of equipment during design basis events. Analysis has confirmed that equipment with a design safety function is capable of operating at the temperature listed below.

Table 3.11-1

Normal Operating Conditions (Continued)

Room/Area	Description	Operability Temp Limit	Room/Area	Description	Operability Temp Limit
C121	RadW/React Bld Corridor	N/A	D107	DG1 Engine Room	130°F
C206	Elec Switchgr Rm 2	120°F	D108	DG1 Day Tk Room	162°F
C207	Remote Shutdown	124°F*	D110	DG2 Engine Room	130°F
C208	Elec Switchgr Rm 1	120°F*	D111	DG2 Day Tk Room	162°F
C210	Div I Battery Room	110°F	D113	DG Bld HVAC Room	126°F
C211	RPS Rm 1	120°F*	D114	Div III Battery Area in DG	122°F*
C213	RPS Rm 2	120°F*	D114	HPCS DG3 Elec Equip Room	111°F*
C215	Div II Battery Room	110°F	D115	DG1 Elec Equip Room	122°F*
C216	Battery Charger Rm 1	122°F*	D116	DG2 Elec Equip Room	122°F*
C224	Battery Charger Rm 2	122°F*	D201	HPCS DG3 Air Handling Room	122°F
C230	Cable Chase	136°F	D203	DG1 Air Handling Room	130°F
C502	Equip Access Area	140°F	D204	Deleted	
C507	HVAC Equip Room 1	120°F*	D205	DG2 Air Handling Room	130°F
C508	HVAC Equip Room 2	120°F*	R105	441' Railroad Bay	137°F
C510	Instrument Shop	140°F	R212	DC MCC Rm	129°F
D100	HPCS DG3 Engine Room	122°F	R410	MCC Rm Div 2	129°F
D101	DG1 Storage Tk/Transfer Room	142°F	R411	MCC Rm Div 1	129°F
D102	DG2 Storage Tk/Transfer Room	142°F	R611	Hydrogen Recombiner Rm Div 1	129°F*
D103	HPCS DG3 Storage Tk/Transfer Room	142°F	R612	Hydrogen Recombiner Rm Div 2	129°F*
D104	React Bld/DG Bld Corridor	137°F	N/A	SW Pump House A	122°F*
D105	HPCS Day Tk Room	162°F	N/A	SW Pump House B	122°F
			R506	Fuel Pool HX Room	128°F

*Lowest value for certain equipment in this room/area.

[i] 130°F normal with one fan always on standby.

[j] 64°F minimum when Reactor Building Crane is making full rated lift.

[k] 85°F maximum for SBO capability and control room habitability; 104°F maximum environmental qualification temperature for equipment.

[l] 50°F minimum in electrical equipment areas, 40°F minimum elsewhere.

[m] Maximum pressure is the high drywell pressure trip set point; minimum pressure is vacuum breaker open set point.

[n] Per LCS 1.7.1 for HPCS battery area minimum normal is 65°F.

Table 3.11-2

Accident Environment Conditions (Primary Containment) for Essential Equipment

Condition	Component[a]	Temperature (°F)	Pressure (psig)	Relative Humidity	Duration[b]
1	Core spray injection check valves, LPCI-RHR injection check valves, reactor shutdown cooling suction valve, safety/relief valves,[c] vessel level indicators, structural components (e.g., loop restraints, vessel skirt, etc.)	340 320 250 200[d]	-2 to 45 -2 to 45 0 to 25 0 to 20	All steam All steam 100% 100%	3 hr 6 hr 1 day 180 days
	Control instrumentation air (including accumulators), reactor building closed cooling (RCC) valves,[e] drywell unit coolers[e]				
2	Feedwater check valves, RCIC steam line isolation valve, reactor water cleanup suction valve, reactor water sample line valve, 2 in. and smaller isolation valves, cables to intermediate range and power range monitors, reactor vessel head spray isolation valve	340[f] 320	-2 to 45 -2 to 45	All steam All steam	3 hr 6 hr
3	Main steam isolation valves, main steam drain isolation valves	340	-2 to 45	All steam	1 hr
4	Recirculation gate valves,[e] reactor protection system neutron monitoring system	340 320	-2 to 45 -2 to 45	All steam All steam	3 hr 4.5 hr
5	Feedwater check valves, RCIC steam line isolation valve, recirculation flow control valves, reactor vessel head spray isolation valve, reactor water cleanup suction, reactor water sample line valve, 2 in. and smaller isolation valves	250 200[d]	-2 to 25 -2 to 20	100% 100%	1 day 180 days

Table 3.11-2

Accident Environment Conditions (Primary Containment) for Essential Equipment (Continued)

Condition	Component[a]	Temperature (°F) Drywell Side	Temperature (°F) Wetwell Side	Pressure (psig)	Relative Humidity	Duration[b]
6	Wetwell-drywell vacuum relief valves	340	275	-2 to 45	All steam	3 hr
		320	275	-2 to 45	All steam	6 hr
		250	250	0 to 25	100%	1 day
		200[d]	200[d]	0 to 20	100%	180 days
7	Main steam isolation valves, main steam drain isolation valves, standby liquid control injection check valve	340		-2 to 45	All steam	3 hr
		320		-2 to 45	All steam	6 hr
		250		-2 to 25	100%	1 day
		200[d]		-2 to 20	100%	180 days

[a] Typical components are listed. Components listed may have required operating time different than shown here. Specific hours to operate are described in Reference 3.11-1 (historical) and currently in the Columbia Generating Station Master Equipment List or MEL. Also included are the valve operators and cabling (instrumentation and power) required for proper operation of the valves listed.

[b] Durations shown are termination times for conservative environmental conditions measured from the initiation of the postulated accident, i.e., Condition 1, the 6-hr duration, is the period from 3 hr through 6 hr, the 1-day duration is the period from 6 hr through 1 day (24 hr).

[c] The safety/relief valves are required to be operable for 2 days and to remain in whichever position the valve operator has been set to at the end of the 2 days for the remainder of the long-term transient. Safety/relief valves used for automatic depressurization must remain operable throughout the long-term transient.

[d] Represent peak starting temperature of extended long-term temperature in the containment following a postulated design basis accident.

[e] The RCC system, drywell air coolers, and the reactor recirculation system have no safety requirements during a LOCA. The specified conditions are desirable to enable a normal shutdown cooling procedure during a steam leak.

[f] The TIP system isolation valves are located outside of primary containment in the TIP room. Analysis has determined that maximum accident condition will be 130°F.

Table 3.11-2

Accident Environment Conditions (Primary Containment) for Essential Equipment (Continued)

Basic Accident Environmental Pressures and Temperatures

The following is a compilation of basic accident environmental pressures and temperatures and a description of the time durations expected. The full spectrum of simultaneous environmental possibilities is not presented in a series of curves, but rather as a description of the boundaries within which designated equipment will be qualified for discrete times during the cycles/modes of the reactor's operation for the first 24 hr. Plant-specific temperature/pressure profiles have been calculated to define the conservatism inherent in these generic environmental values. No margin is required when these values are met in equipment qualification test programs.

Temperatures

340°F Superheat temperature for a steam leak with the reactor vessel at 400-500 psi, containment at 45 psig.

320°F Superheat temperature during shutdown cooling line flush after reactor has been depressurized to 150 psia which corresponds to the pressure at which the shutdown cooling system is activated.

250°F Long-term temperature in the containment during the first day following a postulated design basis accident.

200°F Peak starting temperature of extended long-term temperature in the containment following a postulated design basis accident. Plant-specific calculations have been determined per NUREG-0588.

Pressures

-2 psig Assumed minimum pressure of the primary containment.

45 psig Maximum positive internal pressure of the primary containment.

25 psig Pressure up to 1 day following a postulated design basis accident.

20 psig Pressure at 1 day and longer following a postulated design basis accident.

Table 3.11-2

Accident Environment Conditions (Primary Containment)
for Essential Equipment (Continued)

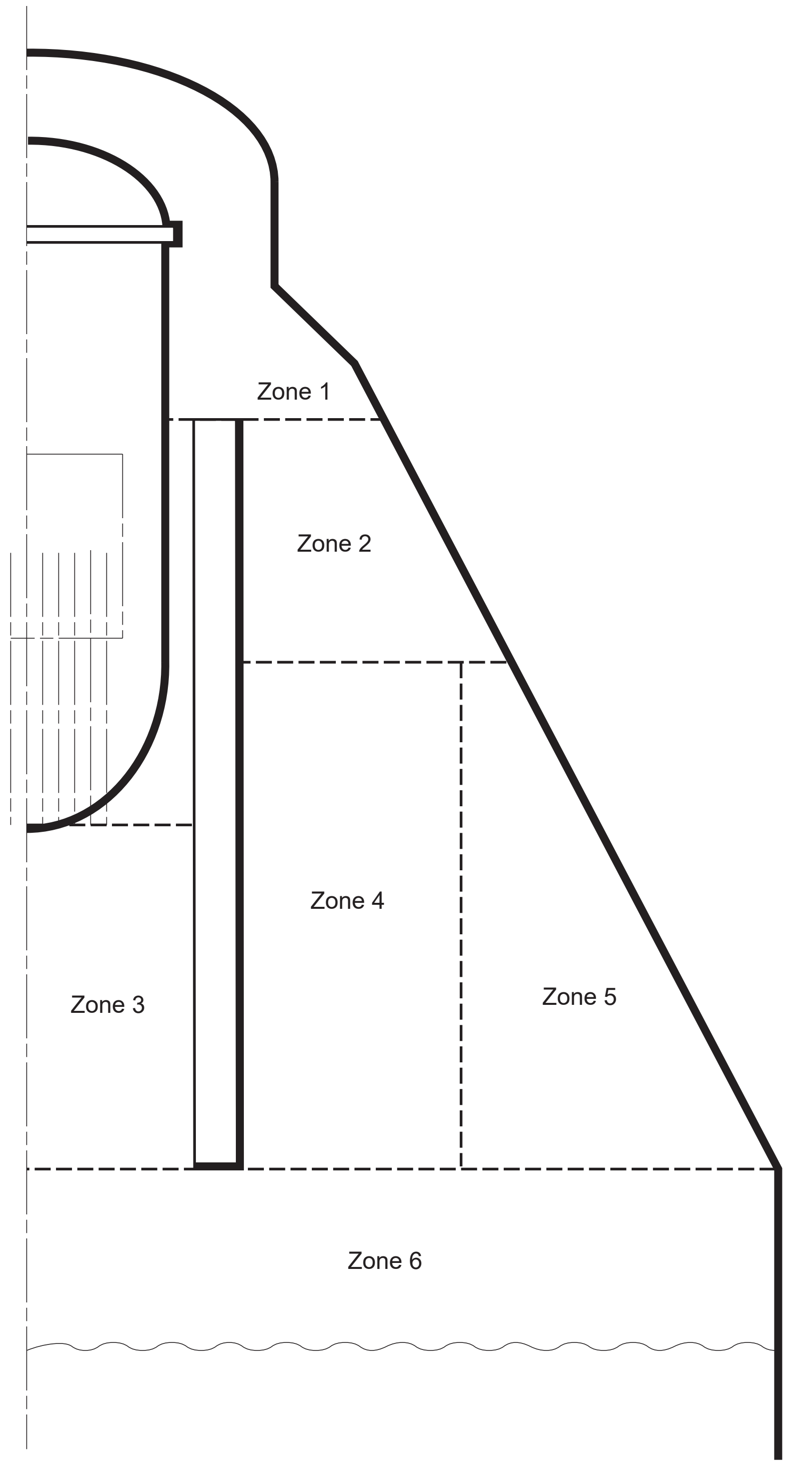
<u>Duration</u>

1 hr	Applies to valves that isolate automatically on low RPV pressure or level. This time interval represents a conservative duration during which the valves must be qualified. Required operating time is 0.17 hr.
3 hr	Time duration to depressurize the reactor pressure vessel at a rate not exceeding 100°F/hr, down to 150 psia.
4.5 hr	Time at which shutdown cooling system flush is complete. Normal shutdown cooling necessitates closure of recirculation line valves.
6 hr	Duration of time to complete vessel depressurization to near atmospheric pressure. This time includes RPV depressurization to 150 psia not exceeding a rate of 100°F/hr, flushing of system, depressurization to near atmospheric pressure. If shutdown cooling is not available, containment sprays should be activated before 6 hr to limit temperature to less than 250°F.

Table 3.11-3

Water Quality Limits

System Water	Parameter			
	Conductivity	Chlorides as Cl	pH	Total Suspended Solids and/or Insolubles
Reactor (shutdown condition, depressurized maximum)	< 10 µS/cm at 25°C	< 0.5 ppm	5.3 to 8.6 at 25°C	< 1 ppm
Suppression pool (quality expected) (see Section 6.1.1.2)	< 5 µS/cm at 25°C	< 20 ppb	5.3 to 8.6 at 25°C	< 5 ppm
Suppression pool (maximum)	< 10 µS/cm at 25°C	< 0.5 ppm	5.3 to 8.6 at 25°C	< 5 ppm
RHR (layup condition, maximum and/or refueling)	< 3 µS/cm at 25°C	< 0.5 ppm	5.3 to 7.5 at 25°C	< 1 ppm



Columbia Generating Station
Final Safety Analysis Report

Primary Containment Zones

Draw. No. 990306.94 | Rev. | Figure 3.11-1

EQUIPMENT QUALIFICATION REPORT

QID #213012

Owner: Energy Northwest
Facility: Columbia Generating Station
Spec.: 2808-67

MPL:
PPD:

Page No.
Revision 4
Date: July 1983

Equipment Description	Environment: Parameter	Environment: FSAR	Environment: Qualification	Document Ref.: FSAR	Document Ref.: Qual.	Qualification Method	Outstanding Items
System Reactor Building Return Air	Operating Time	6 months	Equivalent to > 6 months	1	4	Simultaneous Test and Engineering Analysis	None
Tag Number RRA-M-(See Note 1)	Temperature (°F)	90 Max Normal 104 Max Abnormal Accident Profiles 4, 8	410	2	4	Simultaneous Test and Engineering Analysis	None
Manufacturer Westinghouse	Pressure (PSIA)	14.7 Normal Accident profiles 8	Accident Profiles 8	2	4	Engineering Analysis	None
Model Number See Note 1	Relative Humidity (N)	40 Normal 90 Abnormal 100 Accident	100	2	4	Sequential Test	None
Component Motors	Chemical Spray	N/A	N/A	2	N/A	N/A	None
Function/Service See Note 1	Radiation (Rad)	3.1×10^6	1×10^6	3	4	Sequential Test	None
	Aging	40 Years	40 Years	2	4	Simultaneous Test and Engineering Analysis	None
Location Bldg R Elevation Column See Note 1	Accuracy	N/A	N/A	N/A	N/A	N/A	None

Flood Level Elev:
Above Flood Level?
Yes_X__ No____ (Ref. 5)

Prepared by:______________________________ Reviewed by: _________________________________

Documentation References

1. BRI CIE List, Rev. 8, 6/1/83
2. FSAR Paragraph 3.11
3. EDS Report 0740-004-44IJ
3. QID = 213017
5. BRI Calc. #5.51.055

Notes

Qualified

1.

Tag Number	Model	Function/Service	Elevation	Column
RRA-M-1	SBFC	Motor for RRA-FN-1	441	H.7/4.3
RRA-M-2	SBFC	Motor for RRA-FN-2	445	L.0/8.3
RRA-M-3	SBFC	Motor for RRA-FN-3	442	L.8/8.3

3.12 COMPUTER PROGRAMS FOR STRUCTURAL ANALYSIS AND DESIGN

The computer programs referenced in Sections 3.6, 3.7, 3.8, and 3.9 by their acronyms are listed below.

ADLPIPE
ANSYS
AX1
AX2
AX3
BPIPE-CS085 (Version 2)
CB&I PROGRAM 979 - ASFAST
CB&I PROGRAM 1027
CB&I PROGRAM 1037 - DUNHAM'S
CB&I PROGRAM 1335
CB&I PROGRAM 1606 AND 1657 - HAP
CB&I PROGRAM 1635
CB&I PROGRAM 711 - GENOZZ
CB&I PROGRAM 766 - TEMAPR
CB&I PROGRAM 767 - PRINCESS
CB&I PROGRAM 781 - KALNINS
CB&I PROGRAM 846
CB&I PROGRAM 928 - TGRV
CB&I PROGRAM 948 - NAPALM
CB&I PROGRAM 953
CB&I PROGRAM 962 - E0962A
CB&I PROGRAM 984
CB&I PROGRAM 992 - GASP
DACSR
FLXMAT
GOTHIC
ISOFINITE
MASS
MULTISHELL
NASTRAN
NUPIPE-IIM (Version 1.6.3)
P001 (Version 6.0)
P002 (Version 4.0)
P003 (Version 5.0)
PDA
PIPESUP-CS102 (Version 2)
RELAP3
RELAP4/MOD5
SAP4G07

S/RVDAM
SSAPO ALGOR SUPERSAP
SPLUG
STARS-2S
STRUDL II
T-MOVE (Version V0-00)
TPIPE (Version 4.2)

The italicized information is historical and was provided to support the application for an operating license.

The computer program summaries, including descriptions of assumptions, limitations, capabilities, and any appropriate parameters and features associated with the use of these computer programs, are provided in the following.

All programs are verified, within the stated assumptions and limitations, for correctness of theory used and for validity of results obtained for a variety of typical problems. Results are checked against known solutions, solutions obtained from other programs, or hand calculations. Examples of validation problems are included with the program descriptions. Wherever applicable, internal checks are included as an aid in checking the validity of the results obtained from the computer program for each problem analyzed.

3.12.1 DACSR

DACSR (Dynamic Analysis of Cantilever Structures) is a computer program used for the response spectrum analysis of structures subjected to horizontal excitations. The structural idealization consist of lumped mass cantilevers attached to a common rigid mat which is supported by soil springs. Other springs can be used to connect any two masses.

Each mass has two degrees of freedom: translational and rocking. The program has the capability to perform an analysis on a model containing up to three cantilevers. DACSR does not consider off-diagonal masses.

The program calculates the eigenvalues (natural frequencies) eigenvectors (natural mode shapes), modal participation factors, and modal damping values which are based on Whitman's average weighted damping calculation (Reference 3.12-1). The input ground motion (horizontal base excitation) is defined by a set of pseudo-accelerations response spectrum curves corresponding to different damping values which are expressed as a fraction of critical damping. The program then calculated modal accelerations, velocities, and displacements. The total response is found by the following methods:

a. Absolute sum of the individual modal responses

b. Square root of the sum of the squared modal responses

c. Combination of the above two methods, which accounts for the effect of closely spaced modes. (Closely spaced modes are defined as those whose frequencies are within 10%.)

The option of having member forces calculated is available.

Program input requires member properties and stiffness, structure stiffness, spring constants, damping ratings, and ground acceleration.

DACSR was developed by Burns and Roe, Inc., in 1970. The program can successfully operate on IBM 370 series hardware maintained by Call Data Systems, Inc., a subsidiary of Grumman Data Systems.

To demonstrate the validity of DACSR, results from a production problem were compared to those obtained from the public domain program STARDYNE. Results from the two programs are compared in Table 3.12-1.

As demonstrated in the table, results compare favorably.

This program is used for dynamic analyses discussed in 3.7.

3.12.2 AX1

AX1 (Analysis of Axisymmetric Solids) is a finite element program which determines deformations and stresses within axisymmetric structures of arbitrary shape. The effects of displacement or stress boundary conditions, concentrated loads, gravity forces and temperature changes are included. Nonlinear material properties are included by a successive approximation technique. Orthotropic material behavior is included. The program can be used for structures with up to 800 elements and 900 modes with a maximum bandwidth of 27 modes. The program has a mesh generating capability which reduces the amount of input required.

The program was written by E. L. Wilson of the University of California at Berkeley, Reference 3.12-2, and is in the public domain.

The program is referred to in Section 3.8.2.

3.12.3 AX2

AX2 (Axisymetric Shell Program) is a computer program used for the analysis of single layer, axisymmetric thin shells of revolution composed of spherical, toroidal, conical, cylindrical or

circular plate segments. Shells may be stiffened with discrete circumferential rings of arbitrary cross section. Structural geometry and loading is defined such that behavior is linear and deflections are small with respect to segment thicknesses.

Any elastic, isotropic material with constant elastic modulus, Poisson's ratio, and coefficient of thermal expansion may be modeled. Structural loading is static and axisymmetric. AX2 allows mechanical loading in the form of normal and tangential pressures and line loads and moments. Thermal loading in the form of meridional and cross thickness gradients is also available. Both rigid and elastic edge restraints may be used in global coordinates and in user-specified local coordinate systems inclined with respect to the global axes. Nodal displacements may also be specified.

Output from AX2 includes

- *a. Nodal displacements in global coordinates,*
- *b. Reaction loads at constraints in global coordinates,*
- *c. Body displacements in local coordinates,*
- *d. Stress resultants, and*
- *e. Stresses and ASME stress intensities.*

AX2 uses the stiffness method (References 3.12-6 and 3.12-8) to determine the deflections of the nodal circles or segment junctions. The load-deflection relationships required for the assembly of the stiffness matrix are derived from closed form solutions found in References 3.12-3, 3.12-4, 3.12-7 and 3.12-9. An exception to this procedure is the toroidal segment, which required development of a separate numerical integration procedure. The remaining output quantities are calculated from the displacement solution using formulations derived from the same references.

AX2 is a program proprietary to the Pittsburgh-Des Moines Steel Company, Pittsburgh, Pennsylvania. This is verified by comparing its results with those obtained from the computer code BOSOR 4. BOSOR 4 was written by David Bushnell of the Lockheed Missile and Space Company (Palo Alto, California) and is in the public domain.

A cone segment with an apex angle of 60° and a 30° spherical segment (R/t = 500) is connected rigidly, without a discontinuity, to form a shell. (See Figure 3.12-1.) Loading consists of 20 psi internal pressure (normal) and 200°F uniform temperature rise. Normal and tangential deflections at points 1 and 3 are constrained. Normal and tangential deflections are plotted against arc length for both AX2 and BOSOR 4, as shown in Figures 3.12-2 and 3.12-3. AX2 deviation from BOSOR 4 results for these deflections are calculated and also appear on these figures.

This program is referred to in Section 3.8.2.

3.12.4 AX3

AX3 (Analysis of Thin Shell Solids of Revolution) is a computer program used for the analysis of laminated, thin shell solids of revolution with orthotropic material properties. The program is predicated on the Kirchoff-Love hypothesis regarding deformation. It accommodates asymmetrically and axisymmetrically loaded shells composed of an arbitrary number of bonded layers, each with a different thickness and different orthotropic material properties. In this method of analysis, a series of short, truncated conical shell elements connected by nodal circles is substituted for the continuous shell. An approximate displacement function similar to the following is used:

$$W_r = b_1 + b_2 r + b_3 z$$

$$W_z = b_4 = b_5 r + b_6 z$$

where:

W_r, W_z = *linear displacement functions*

$b1, \cdots b6$ = *generalized coordinates*

r,z = *radial and axial coordinates*

From the above, the displacements, moments, and stresses for loadings (such as concentrated nodal circle forces and moments) and distributed surface normal and shearing pressures are obtained directly.

The AX3 program was written by S. K. Takahashi of the Naval Civil Engineering Laboratory, Port Hueneme, California and S. B. Dong, Assistant Professor of Engineering, University of California, Los Angeles, (Reference 3.12-10), and is in the public domain.

This program is referred to in Section 3.8.2.

3.12.5 STRUDL II

STRUDL II (Structural Design Language) is a static and dynamic analysis program offering solutions to a wide range of structural problems. Primarily, the program is used to analyze two-dimensional trusses, frames, plates and grids, as well as three-dimensional trusses and frames. Problems which contain both planar and spatial components may also be analyzed.

In addition to standard truss and framing members, the program accepts directly input curved members, piping sections and member dimensions (for concrete sections). Provisions exist to

idealize members of variable cross sections, to input stiffness or flexibility arrays, or to reference predefined tables of member properties. Anisotropic materials may be also used.

The program flexibility which exists for the specification of joint and member data also applies to loadings. It can accept input from concentrated, uniform or linearly varying member loads, member distortions, temperature loadings (including transverse gradients), joint loads, and joint displacements. Program output includes member forces, displacements, reactions, member distortions and equilibrium check at every point. The program is capable of identifying infinitely stiff end joints, member eccentricities and elastic supports.

STRUDL II has the capability to develop a large variety of finite element structural models, such as may be required for certain building types and its components. Analysis of three-dimensional elastic solids subjected to arbitrary loads, temperature changes, or specified displacements can be performed. Either earthquake accelerations or time history force may be used for dynamic analysis. Analysis results can be output in terms of node displacements and element stresses and strains or member forces and moments. Eigen-values, eigenvectors and time history response or nodal responses may be obtained for dynamic analysis.

In addition to analysis, the program is capable of performing structural steel design according to the AISC-69 Code (Reference 3.12-23) and reinforced-concrete design according to the ACI 318-71 Code (Reference 3.12-24).

STRUDL-II was developed as part of the Integrated Civil Engineering System at the Massachusetts Institute of Technology. See Reference 3.12-11.

STRUDL-II has been in the public domain since 1968. The program is currently maintained by McDonnell Douglas Automation Company and is operated on a IBM 360/168 machine.

This program is referred to in Sections 3.8.3.4.5 and 3.8.4.4.1.

3.12.6 STARS-2S

The STARS-2S Shell [Theory Automated for Rotational Structures - 2 (Statics)] program was developed to solve complex mathematics and numerical techniques required to analyze structural shell problems using an accurate shell theory. The program is based on the Love-Reissner first order shell theory.

This program can analyze orthotropic thin shells of revolution, subjected to unsymmetric distributed loading or concentrated line loads, as well as thermal strains (Reference 3.12-12). Furthermore, a shell with arbitrary boundary conditions, under loads which vary arbitrarily

with position and under a temperature variation through the thickness, is tractable with this program. The shell can consist of any combination of the following geometric shapes:

- *a. Ellipsoidal-spherical (offset from the axis of revolution allowed),*
- *b. Ogival-toroidal,*
- *c. Modified ellipse shape,*
- *d. Conical-circular plate,*
- *e. Cylindrical,*
- *f. General point input geometry,*
- *g. Dummy geometry slot to be filled in by user,*
- *h. Discrete ring, and*
- *i. Elastic support.*

The program has the capabilities to analyze a shell wall cross section that is composed of a sheet, sandwich, or reinforced sheet or sandwich. Reinforcement can consist of rings and/or stringers, a waffle construction rotated at any angle to the principal coordinates, or an isogrid construction. General stiffness input options are also available. The program is capable of accommodating reinforcement material properties that differ from the main shell, and a temperature variation that can cause different properties in the two face sheets of a sandwich wall. The present program is also capable of a nonlinear analysis of axisymmetrically loaded shells.

The approach to analysis is to cut the structure into several shell regions. The regions are further subdivided into several shell segments, each being free to have its own geometric shape. There is a restriction on the length of the shell segments. Physically, the restriction demands that boundary disturbances at one edge be distinctly felt at the other edge. One of the limiting factors is that the ratio of the radii of revolution at the initial and final points of a segment be greater than one hundredth and less than 100.

The required input data is subdivided into three main parts; namely, geometric, topological (or coupling orientation), and joint data (degree of freedom description for each joint component). Each segment requires its own geometric configuration and numerical integration control.

The output consists of stiffness coefficients with symmetry checks for each shell segment. Region stiffnesses and their symmetry checks are also provided. Final stresses, displacements and Huber-Von Mises-Hencky "effective stresses" are provided for each shell segment at specified intervals along the segment.

STARS-2S was developed by Grumman Aerospace Corp. and is in the public domain. Official documentation is "User's Manual for the STARS-2 System of Programs, No. 000-STMECH-039 Volume I," by V. Svalbonas (Reference 3.12-12).

STARS-2S is a new version of the STARS programs but has retained the basic features of these programs which have been in use since 1963. The program operates on an IBM 370 series computer maintained by Call Data Systems, Inc., a subsidiary of Grumman Data Systems.

This program is referred to in Section 3.8.4.4.1.1.

3.12.7 NASTRAN

The acronym NASTRAN is formed from NAsa STRuctural ANalysis. NASTRAN is a general purpose digital computer program for the analysis of large complex structures and has its origins in the research councils of NASA.

NASTRAN is a large digital computer program for static and dynamic structural analysis by the finite element approach. The program has been specifically designed to treat large problems with many degrees of freedom. The only limitations on problem size are those imposed by practical considerations of running time and by the ultimate capacity of auxiliary storage devices.

The program is currently capable of performing twelve rigid formats, each corresponding to a different problem type as indicated in the following list:

- *a. Basic static analysis,*
- *b. Static analysis with inertial relief,*
- *c. Normal modes analysis,*
- *d. Static analysis with differential stiffness,*
- *e. Buckling analysis,*
- *f. Piecewise linear analysis,*
- *g. Direct complex eigen value analysis,*
- *h. Direct frequency and random response analysis,*
- *i. Direct transient analysis,*
- *j. Modal complex eigen value analysis,*
- *k. Modal frequency and random response, and*
- *l. Modal transient analysis.*

NASTRAN embodies a lumped element approach, wherein the distributed physical properties of a structure are represented by a model consisting of a finite number of idealized substructures or elements that are interconnected at a finite number of grid points, to which loads are applied. All input and output data pertain to the idealized structural model.

Input for the reactor building foundation mat consists of sectional properties, loads at node points, surface (pressure) loads, and soil spring constants at node points. The program output includes displacements, moments and shears. The use of the NASTRAN program for this

particular application was checked by solving initially a small sample problem which gave satisfactory results.

NASTRAN is in the public domain and official documentation consists of the following four manuals:

a.	*NASTRAN Theoretical Manual*	*NASA-SP-221(01)*
b.	*NASTRAN User's Manual -* *NASA Level 15* *Level 15.5 Update*	 *NASA-SP-222(01)* *NASA-SP-222* *Rev. CS1*
c.	*NASTRAN Programmer's Manual*	*NASA-SP-223(01)*
d.	*NASTRAN Demonstration Problem Manual*	*NASA-SP-224*

The program is currently operational on the following computing systems:

a. CDC 6000 series under the SCOPE 3 operating system,
b. IBM System 360/370 under OS operating system, and
c. Univac 1108 under EXEC 8 operating system.

This program is referred to in Sections 3.8.4.4.1.1 and 3.8.5.4.2.

3.12.8 FLXMAT

FLXMAT (Flexible Mat on an Elastic Foundation) analyzes a beam on an elastic foundation for arbitrary load conditions. The analysis is based on the solution to the basic differential equation for a beam on an elastic foundation:

$$EI\frac{d^4y}{d\chi^4} = \rho = -ky$$

or

$$\frac{d^4y}{d\chi^4} = -4\beta y$$

in which ρ is the pressure, k is the foundation (soil) modulus, and β is defined by the equation

$$\beta = \sqrt[4]{\frac{k}{4EI}}$$

In general the bending of a beam on an elastic foundation falls into three divisions as follows:

a. *For short beams: $\beta L < 0.60$,*
b. *For beams of medium length: $0.6 < \beta L < 5.00$, and*
c. *For long beams: $\beta L < 5.00$.*

where L represents the length of the beam.

In analyzing this type of problem two basic assumptions are made:

a. *The foundation is elastic. This implies that the settlement at any point is proportional to the pressure at that point; and*

b. *The foundation modulus in tension is equal to that in compression. The foundation modulus is defined as the pressure which is required to produce a unit settlement.*

The program input requires sectional properties, applied loads, Young's modulus, and the soil modulus. The program output includes soil reactions, displacements, shears, and moments at specified intervals along the beam.

FLXMAT was developed by Burns and Roe, Inc., in 1972. The program can successfully operate on IBM 370/168 hardware maintained by Call Data Systems, Inc., a subsidiary of Grumman Data Systems.

To demonstrate the validity of FLXMAT, results from the program are compared with results of two sample problems presented in the text by Robert W. Abbett. (See Reference 3.12-13.) The results are as follows:

EXAMPLE PROBLEM		SOIL REACTIONS (kips/ft^2)	
		Text Results	FLXMAT Results
1.	Short beam	$\rho_1 = 0.613$	$\rho_1 = 0.6217$
		$\rho_2 = 2.487$	$\rho_2 = 2.4872$

2.	*Medium beam*	$\rho_1 = 0.714$	$\rho_1 = 0.7146$
		$\rho_2 = 1.233$	$\rho_2 = 1.2334$
		$\rho_3 = 1.173$	$\rho_3 = 1.1721$
		$\rho_4 = 0.573$	$\rho_4 = 0.5743$

Long beam validation is not considered for the reason that all foundation mats utilizing the FLXMAT program do not fall into the long beam category.

A comparison of results indicated above demonstrates the accuracy of the program.

This program is referred to in Sections 3.8.5.4.3 and 3.8.5.4.6.

3.12.9 ISOFINITE

ISOFINITE is used in the design of flued head fittings. ISOFINITE is an isoparametric, three dimensional finite element program that is utilized for stress analysis of three-dimensional elastic continua. The program uses an eight-noded box element of arbitrary shape to build up the stiffness and stress characteristics by Gaussian integration. Each box has 33 degrees of freedom, 24 corresponding to the three motions at each of the eight nodes, and 9 internal degrees of freedom used to minimize strain energy. The 9 internal degrees of freedom are highly effective in eliminating shear error, thereby permitting the use of far fewer elements than are conventionally required.

Program input consists of coordinates of nodal points, element nodal numbers, load conditions, and boundary conditions. In the case of flued head fitting design, because of circular symmetry, a quarter plane representation involving 75 elements and 180 nodes is used. Pressure loads at the pipe inner surface are input as radial nodal forces.

Program output includes all input data, nodal displacements and stresses. Stress values are given on the six surfaces of each element. Corresponding principal stresses are also calculated and printed out.

ISOFINITE was originally developed by S. Levy of the General Electric Company in 1969 (Reference 3.12-14). It was further expanded and modified by N.E. Rieger of the Rochester Institute of Technology in 1971 for use by the National Forge Company. The program uses the Fortran IV computer language.

ISOFINITE is verified using the computer program NASTRAN. The stresses for penetration X-2 are determined using a 3-D flued head model with radial pressure forces applied to the internal surface. This simulated the 2-D axisymmetric pressure stress analysis obtained with NASTRAN.

For the axisymmetric finite element program, NASTRAN, the effects of the process pipe internal pressure is determined through the application of equivalent radial pressure forces at the nodes along the pipe inner surface. Pressure force magnitudes are given by

$$F_i = 2\,p\,r_i\,L_i$$

where

F_i = *radial force at node*

p = *internal pressure, 1250 psi*

r_i = *internal radius, 2.881 in.*

L_i = *equivalent length of pipe associated with the node*

For the axisymmetric pressure calculation using ISOFINITE, internal pressure is included in the computer model as x-, y- forces applied at the inner surface of the process pipe. Values of these forces are obtained from the expression

$$F_i = p\,r_i\,\frac{\pi}{10}\,L_i$$

where

F_i = *radial force at node*

p = *internal pressure, 1250 psi*

$\frac{\pi}{10}$ = *arc length of element surface*

L_i = *axial length of pipe relating to node*

The x-, y- coordinate forces are, therefore,

$$F_{xi} = F_i \sin \theta$$

$$F_{yi} = F_i \cos \theta$$

where is the angle subtended between the vertical y-axis and the local element plane.

Results of both ISOFINITE and NASTRAN are given in Table 3.12-2. As can be seen, there is close correlation between the deflections, with NASTRAN giving larger values throughout the flued head than ISOFINITE. This is due to the lack of rotational freedom at nodes with NASTRAN over the more flexible shear elements in ISOFINITE. This leads to prediction of higher stresses using ISOFINITE (as can be seen by comparing pages 2 and 3 of Table 3.12-2). The computer program ISOFINITE is therefore a conservative method for determining stresses in flued head fittings.

This program is referred to in Section 3.8.6.4.4.

3.12.10 ANSYS

ANSYS is a general-purpose finite element computer program designed to solve a variety of problems in engineering analysis.

The ANSYS program features the following capabilities:

a. *Structural analysis including static elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis;*

b. *One-dimensional fluid flow analyses;*

c. *Transient heat transfer analysis including conduction, convection, and radiation with direct input to thermal-stress analyses;*

d. *An extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), direct interfaces (compression only), curved elbows, etc. Many of the elements contain complete plastic, creep, and swelling capabilities;*

e. *Plotting - Geometry plotting is available for all elements in the ANSYS library, including isometric and perspective views of three-dimensional structures; and*

f. *Restart Capability - The ANSYS program has restart capability for several analyses types. An option is also available for saving the stiffness matrix once it is calculated for the structure, and using it for other loading conditions.*

The program is maintained current by Swanson Analysis Systems, Inc., of Pittsburgh, Pennsylvania, and is supplied to General Electric for use on the Honeywell 6000.

The ANSYS program has been used for productive analyses since early 1970. Users now include the nuclear, pressure vessels and piping, mining, structures, bridge, chemicals, and automotive industries, as well as many consulting firms.

3.12.11 RELAP3

This program describes the behavior of water-cooled nuclear reactors during postulated accidents such as loss-of-coolant pump failure, or power transients. The behavior of the primary cooling system and the reactor is emphasized. The program calculates flaws, mass inventories, energy inventories pressures, temperatures, and qualities along with variables associated with reactor power, reactor heat transfer, or control systems.

RELAP3 is an NRC-accepted computer program and is in the public domain. For a complete discussion of this program see Reference 3.12-18.

This program is referred to in Sections 3.6.2.2.1b and 3.6.2.3.1.

3.12.11.1 RELAP4/MOD5

RELAP4 is a computer program written in FORTRAN IV for the digital computer analysis of nuclear reactors and related systems. It is primarily applied in the study of system transient response to postulated perturbations such as coolant loop rupture, circulation pump failure, power excursions, etc. The program was written to be used for water-cooled (PWR and BWR) reactors and can be used for scale models such as LOFT and SEMISCALE. Additional versatility extends its usefulness to related applications, such as ice condenser and containment subcompartment analysis. Specific options are available for reflood (FLOOD) analysis and for the NRC evaluation model.

3.12.11.2 S/RVDAM

This program is described in Section 3.9.

3.12.11.3 GOTHIC 7.0/8.1

GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is a general purpose thermal-hydraulics computer program for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. Applications of GOTHIC include evaluation of containment and containment sub-compartment response to the full spectrum of high energy line breaks within the design basis envelope as described in FSAR Chapter 6, Section 2. Applications may include pressure and temperature determination, equipment qualification profiles and inadvertent system initiation, and degradation or failure of engineered safety features.

GOTHIC was developed by Numerical Applications, Inc., and has been accepted by the NRC for numerous applications. For a complete description of this program, see References

3.12-27 and 3.12-28. Use of GOTHIC 7.0 is restricted to applications consistent with NRC approved River Bend Safety Evaluation (Reference 3.12-29). GOTHIC 8.1 is strictly approved only for HELB analysis of RWCU piping within the Radwaste Building.

3.12.12 CB&I PROGRAM 711 - "GENOZZ"

This program is described in Section 3.9.

3.12.13 CB&I PROGRAM 948 - "NAPALM"

This program is described in Section 3.9.

3.12.14 CB&I PROGRAM 1027

This program is described in Section 3.9.

3.12.15 CB&I PROGRAM 846

This program is described in Section 3.9.

3.12.16 CB&I PROGRAM 781 - "KALNINS"

This program is described in Section 3.9.

3.12.17 CB&I PROGRAM 979 - "ASFAST"

This program is described in Section 3.9.

3.12.18 CB&I PROGRAM 766 - "TEMAPR"

This program is described in Section 3.9.

3.12.19 CB&I PROGRAM 767 - "PRINCESS"

This program is described in Section 3.9.

3.12.20 CB&I PROGRAM 928 - "TGRV"

This program is described in Section 3.9.

3.12.21 CB&I PROGRAM 962 - "E0962A"

This program is described in Section 3.9.

3.12.22 CB&I PROGRAM 984

This program is described in Section 3.9.

3.12.23 CB&I PROGRAM 992 - "GASP"

This program is described in Section 3.9.

3.12.24 CB&I PROGRAM 1037 - "DUNHAM'S"

This program is described in Section 3.9.

3.12.25 CB&I PROGRAM 1335

This program is described in Section 3.9.

3.12.26 CB&I PROGRAM 1606 and 1657 - "HAP"

This program is described in Section 3.9.

3.12.27 CB&I PROGRAM 1635

This program is described in Section 3.9.

3.12.28 CB&I PROGRAM 953

This program is described in Section 3.9.

3.12.29 ALGOR SUPERSAP COMPUTER PROGRAM

The Algor SUPERSAP computer program, developed by Algor Interactive systems, Inc. is a widely used general purpose computer program employing finite element technology for the solution of several classes of engineering analysis problems. Structural analysis may be performed in two or three dimensions including axisymmetric and planar problems. The matrix displacement method of analysis is mathematically modeled as a system of node points interconnected by various "finite elements." The interconnecting finite elements are assigned stiffnesses equivalent to that of the actual structure. The static analysis portion of SUPERSAP is used to solve for the displacements, stresses, and strains in structures under the action of applied loads.

3.12.30 MASS

This program is referred to in Section 3.9.

3.12.31 MULTISHELL

This program is referred to in Section 3.9.

3.12.32 SAP4G07

Structural Analysis Program 4 G07 (SAP4G07) is a finite element program that conducts static and dynamic analysis of linear, three-dimensional structural or piping systems. The main applications of this program at General Electric Hitachi are mainly to perform the dynamic and vibration analyses of reactor, piping, heat exchanger, pump and containment structures.

The finite element model serves as an input to the SAP4 program. This model consists of definitions in terms of element data, joint data, mass matrix data and/or concentrated mass data. The static loads are defined in terms of material weight, temperature, pressure and concentrated loads. Likewise the dynamic loads are defined in terms of acceleration and force time histories, or displacement/acceleration response spectra.

The SAP4 program then computes the following outputs: vibration modes and frequencies, static deflections and stresses, maxima and time histories of nodal displacement, nodal accelerations, element forces and stresses.

3.12.33 PDA

This program is referred to in Section 3.6.2.2.2.2 and described in Section 3.9.

3.12.34 DELETED

3.12.35 DELETED

3.12.36 DELETED

3.12.37 ADLPIPE

"ADLPIPE" is a computer program used in the analysis of complex piping systems. This program is accessed on the Burns & Roe NAS Series 5000/6000 computer. The ADLPIPE Program has the capability of handling static and dynamic loading with stress reports meeting both ASME and ANSI Codes.

Static loads may be thermal, deadweight, static "g" seismic, externally applied loads, and movements, wind, and pressure. Dynamic loads are multiple seismic response spectra or time history forcing functions. Output may also be supplemented by geometry plots (orthographic, isometric and steroscopic drawings).

3.12.38 NUPIPE-IIM (Version 1.6.3)

"NUPIPE" is a general purpose piping analysis program with features and capabilities analogous to those described for the ADLPIPE computer code referenced in Section 3.12.37.

NUPIPE is a product of the Quadrex Corporation and is executed via Control Data Corporation's Cybernet computer system. The main application of NUPIPE at CGS is qualification of safety-related instrumentation lines (i.e. tubing), with limited application to small and large bore piping systems. The program, similar to ADLPIPE, is fully benchmarked for accuracy and correctness and may be applied to the full range of ASME and ANSI tubing/piping analyses.

3.12.39 TPIPE (Version 4.2)

Developed by PMB Systems Engineering Inc., San Francisco, and the Tennessee Valley Authority. TPIPE is a 3-dimensional structural analysis program used by Gilbert Commonwealth on the analysis of safety class 1, 2, and 3 piping systems. The program is capable of static and dynamic analysis for very large piping systems and has a postprocessor giving pipe member forces and moments to the requirements of the stress equations of ASME Code Section III.

The integrity of the program was accomplished by comparison of results (for many examples) with three independent computer programs: PIPESD, POSOL, AND SAPIV.

3.12.40 PIPESUP-CS102 (Version 2): Gilbert Commonwealth

Developed for use on the TRS-80 Model II, this Microcomputer program is used in the design of the three most commonly used smallbore pipe support types.

- *a. Cantilever type,*
- *b. Cantilever with one brace, and*
- *c. Cantilever with two braces.*

3.12.41 P001 (Version 6.0): Gilbert Commonwealth

Developed for use on the TRS-80 Model II Microcomputer, this program calculates support reactions, maximum deflections, movements, and bending stresses for single span piping. The program can handle gravity and seismic loadings and four restraint configurations are available: simply supported, cantilever, propped cantilever, and fixed. These conditions are available for 0.25 in. through 3 in. piping (schedules 10S, 40, 80, and 160). Any combination of up to nine uniform and eight point loads can be modeled; however, all loads and reactions must be normal to the pipe and in the same plane. Note that uniform loads may be applied to the entire piping system or any portion of it.

3.12.42 P002 (Version 4.0): Gilbert Commonwealth

Developed for use on the TRS-80 Model II Microcomputer, this program calculates additional pipe span tables in accordance with Energy Northwest contract CO-208. Modifications are

made to Tables 4.3.3.1 in accordance with Tables 4.3.3.1 through 4.3.3.9 and 5.9. These provide modifications for period of vibration, temperature, weight, concentrated mass, and change of direction. The complete modifications are printed with the total multiplication factor used to modify the original table also printed.

3.12.43 P003 (Version 5.0): Gilbert Commonwealth

Developed for the use on the TRS-80 Model II Microcomputer, this program calculates pipe weights for 0.25 in. through 2 in. pipe for schedules 10, 40, 80, and 160 with or without water and various insulations. (All weights are from Tables D, E, F of Energy Northwest Contract CO-208.)

3.12.44 T-MOVE (Version V0-00): Gilbert Commonwealth

Developed for the use on the TRS-80 Model II Microcomputer, this program is used to predict thermal movements in the containment building. Thermal displacements are calculated for normal plant operation and accident operation.

3.12.45 BPIPE-CS085 (Version 2)

Written in Fortan 4, this program is accessed by Gilbert Commonwealth on their "Librarian System." This program computes stresses at the elbow of a buried pipe due to thermal pressure and seismic loads. The following program features include

a. *Axial, bending, and shear effects in both legs of the pipe are coupled using a matrix formulation;*

b. *Friction and elastic soil shear deformation effects are included in the expression for pipe axial stiffness. The nonlinear nature of the friction problem is incorporated through an iterative procedure;*

c. *Far end fixity effects (boundary conditions) can be specified for each leg;*

d. *Member releases at the elbow can be specified for each leg to simulate slip couplings, etc.;*

e. *An arbitrary angle (not just 90) is specified for the bend at the elbow. A proper matrix transformation is employed to incorporate all coupling effects (shear-axial etc.);*

f. *A different elbow wall thickness than the straight pipe thickness can be specified;*

g. Stress intensification is included to compute bend or elbow bending stress. Stress combinations are computed and compared with code; and

h. The program computes the axial and shear-bend fixity lengths to alert the user as to the importance of far end fixity conditions.

3.12.46 SPLUG (Originators J. Kutzen, J. B. Mahoney, V. Ral-Bahade)

Accessed through the NAS Series 5000/6000 Main Frame computer at Burns & Roe, New Jersey, this program is an application of WRC Bulletin No. 107 from which local stresses in cylindrical and spherical shell attachments are calculated. The program calculates and tabulates local membrane and bending stresses, combined stress intensity and shear stresses.

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Table 3.12-1

DACSR Computer Program Verification DACSR Program Versus STARDYNE Program Frequency Comparison (Hz)

Mode	*DACSR*	*STARDYNE*	*% Difference*
1	*3.41899*	*3.4173*	*0.05*
2	*4.04607*	*4.0454*	*0.01*
3	*6.52622*	*6.5262*	*0*
4	*10.4538*	*10.4539*	*0*
5	*13.4950*	*13.4950*	*0*
6	*15.5987*	*15.5987*	*0*
7	*20.9143*	*20.9131*	*0*

Table 3.12-1

Modal Shape Comparison (Continued)

Mode	Node	Direction	DACSR	STARDYNE	% Difference
1	*1*	*TRANSLATION*	*1.0*	*1.0*	*0*
1	*2*	*TRANSLATION*	*0.094798*	*0.09443860*	*0.38*
1	*3*	*TRANSLATION*	*0.073115*	*0.074203256*	*1.49*
1	*4*	*TRANSLATION*	*0.056265*	*0.059767895*	*6.23*
1	*5*	*TRANSLATION*	*0.044278*	*0.044066432*	*0.48*
1	*6*	*TRANSLATION*	*0.056264*	*0.055998528*	*0.47*
1	*7*	*TRANSLATION*	*0.050659*	*0.050418222*	*0.48*
1	*1*	*ROTATION*	*0.000346255*	*0.000345357*	*0.26*
1	*2*	*ROTATION*	*0.000321196*	*0.000320296*	*0.28*
1	*3*	*ROTATION*	*0.000254565*	*0.000254659*	*0.04*
1	*4*	*ROTATION*	*0.000211389*	*0.000210694*	*0.33*
1	*5*	*ROTATION*	*0.000117812*	*0.000117394*	*0.36*
1	*6*	*ROTATION*	*0.000135774*	*0.000135256*	*0.38*
1	*7*	*ROTATION*	*0.000131294*	*0.000130802*	*0.38*

Table 3.12-1

Modal Shape Comparison (Continued)

Mode	Node	Direction	DACSR	STARDYNE	% Difference
3	*1*	*TRANSLATION*	*0.69002*	*0.688620885*	*0.20*
3	*3*	*TRANSLATION*	*0.49725*	*0.497217159*	*0.01*
3	*5*	*TRANSLATION*	*0.45323*	*0.453265733*	*0.01*
3	*7*	*TRANSLATION*	*0.21734*	*0.217404322*	*0.03*
3	*1*	*ROTATION*	*0.016681355*	*0.016682023*	*0*
3	*3*	*ROTATION*	*0.014636790*	*0.014636205*	*0*
3	*5*	*ROTATION*	*0.007712905*	*0.007712407*	*0.01*
3	*7*	*ROTATION*	*0.00843554*	*0.00843500*	*0.01*
5	*1*	*TRANSLATION*	*0.038966316*	*0.038921798*	*0.01*
5	*3*	*TRANSLATION*	*1.000000000*	*1.00000000*	*0*
5	*5*	*TRANSLATION*	*0.50942392*	*0.509626317*	*0.04*
5	*7*	*TRANSLATION*	*0.25701805*	*0.257044865*	*0.01*
5	*1*	*ROTATION*	*0.032264308*	*0.032274208*	*0.03*
5	*3*	*ROTATION*	*0.016006215*	*0.016005601*	*0*

Table 3.12-1

Modal Shape Comparison (Continued)

Mode	Node	Direction	DACSR	STARDYNE	% Difference
5	*5*	*ROTATION*	*0.005604970*	*0.00560528*	*0*
5	*7*	*ROTATION*	*0.010036175*	*0.010036397*	*0*
7	*1*	*TRANSLATION*	*0.080347174*	*0.0803057676*	*0.05*
7	*3*	*TRANSLATION*	*0.12327235*	*0.123366541*	*0.08*
7	*5*	*TRANSLATION*	*0.14277601*	*0.142648893*	*0.09*
7	*7*	*TRANSLATION*	*1.00000000*	*1.000000000*	*0*
7	*1*	*ROTATION*	*0.062406298*	*0.062542867*	*0.22*
7	*3*	*ROTATION*	*0.006965539*	*0.006097712*	*0.32*
7	*5*	*ROTATION*	*0.0023545011*	*0.002347585*	*0.29*
7	*7*	*ROTATION*	*0.024423497*	*0.02442371*	*0*

Table 3.12-1

Acceleration (g) Comparison Obtained By The Absolute Sum Method[a] (Continued)

Node	*DACSR X1*	*DYNRE 4 X1*	*% Difference*
1	*1.82661E+00*	*1.82324E+00*	*0.18*
2	*3.38043E-01*	*3.35377E-01*	*0.79*
3	*2.82879E-01*	*2.81895E-01*	*0.35*
4	*2.16947E-01*	*2.14633E-01*	*1.08*
5	*1.98629E-01*	*1.98131E-01*	*0.25*
6	*2.34196E-01*	*2.33873E-01*	*0.14*
7	*2.14037E-01*	*2.12973E-01*	*0.49*

Node	*DACSR X6*	*DYNRE 4 X6*	*% Difference*
1	*1.86357E-03*	*1.82812E-03*	*2.33*
2	*1.73161E-03*	*1.71051E-03*	*1.23*
3	*1.44085E-03*	*1.43783E-03*	*0.21*
4	*1.17287E-03*	*1.16839E-03*	*0.38*
5	*8.16169E-04*	*8.13861E-04*	*0.28*
6	*1.03815E-04*	*1.02027E-03*	*1.75*
7	*9.54772E-04*	*9.45200E-04*	*1.01*

[a] *X1 denotes horizontal translational acceleration due to horizontal base excitation. X2 denotes rotational acceleration due to horizontal base excitation.*

Table 3.12-1

Acceleration (g) Comparison Obtained By The Root Mean Square Method[a] (Continued)

Node	*DACSR X1*	*DYNRE 4 X1*	*% Difference*
1	*1.32587E+00*	*1.32352E-01*	*0.05*
2	*1.93621E-01*	*1.93606E-01*	*0.01*
3	*1.74340E-01*	*1.74406E-01*	*0.04*
4	*1.52166E-01*	*1.52236E-01*	*0.05*
5	*1.26814E-01*	*1.26899E-01*	*0.07*
6	*1.54137E-01*	*1.54238E-01*	*0.07*
7	*1.40968E-01*	*1.41061E-01*	*0.07*

Node	*DACSR X6*	*DYNRE 4 X6*	*% Difference*
1	*1.01148E-03*	*1.01043E-03*	*0.10*
2	*9.62895E-04*	*9.62230E-04*	*0.07*
3	*8.53000E-04*	*8.52735E-04*	*0.03*
4	*7.20125E-04*	*7.19931E-04*	*0.03*
5	*4.52437E-04*	*4.52359E-04*	*0.02*
6	*5.40420E-04*	*5.40097E-04*	*0.06*
7	*5.08601E-04*	*5.08468E-04*	*0.03*

[a] *X1 denotes horizontal translational acceleration due to horizontal base excitation. X6 denotes rotational acceleration due to horizontal base excitation.*

Table 3.12-1

Displacement Comparison Obtained by the Root Mean Square Method[a] (Continued)

Node	*DACSR X1*	*DYNRE 4 X1*	*% Difference*
1	*8.74855E-02*	*8.74131E-02*	*0.08*
2	*1.07928E-02*	*1.07850E-02*	*0.07*
3	*9.58983E-03*	*9.58743E-03*	*0.03*
4	*8.32096E-03*	*8.32080E-03*	*0.00*
5	*6.71647E-03*	*6.71782E-03*	*0.02*
6	*8.28021E-03*	*8.28137E-03*	*0.01*
7	*7.56965E-03*	*7.57101E-03*	*0.02*

Node	*DACSR X6*	*DYNRE 4 X6*	*% Difference*
1	*3.37770E-05*	*3.37147E-05*	*0.18*
2	*3.19740E-05*	*3.19160E-05*	*0.18*
3	*2.74793E-05*	*2.74401E-05*	*0.14*
4	*2.34891E-05*	*2.34484E-05*	*0.13*
5	*1.40948E-05*	*1.40810E-05*	*0.10*
6	*1.71246E-05*	*1.71122E-05*	*0.07*
7	*1.63059E-05*	*1.62933E-05*	*0.08*

[a] *X1 denotes horizontal translational acceleration due to horizontal base excitation. X2 denotes rotational acceleration due to horizontal base excitation.*

Table 3.12-2

ISOFINITE Computer Program Verification
Using Computer Program NASTRAN

Node (See Figure 3.12-4)	*ISOFINITE Deflection (10^{-3} in.)*	*NASTRAN Deflection (10^{-3} in.)*
01	*0.700*	*0.796*
03	*0.603*	*0.768*
05	*0.288*	*0.499*
07	*0.170*	*0.318*
11	*0.031*	*0.160*
15	*0.240*	*0.251*
23	*0.514*	*0.519*

Table 3.12-2

NASTRAN Pressure Loads (Continued)

Point[a]	T-Stress (psi)	R-Stress (psi)	Z-Stress (psi)	Maximum Stress Intensity (psi)
a	8795	-6683	5573	15,478
b	8806	-59	6571	8865
c	7365	-575	4587	7940
d	5430	-643	3316	6073
e	3735	-498	2303	4233
f	2846	-616	1905	3462
g	1560	-15	1190	1575
h	1306	-716	1032	2022
i	842	126	-10	852
j	603	33	161	570
k	318	-218	111	536
l	297	-43	389	432
m	80	10	396	386
n	19	-13	414	427
o	-42	-22	423	465
p	-60	-23	648	708
q	-107	11	479	586
r	-72	1	441	513

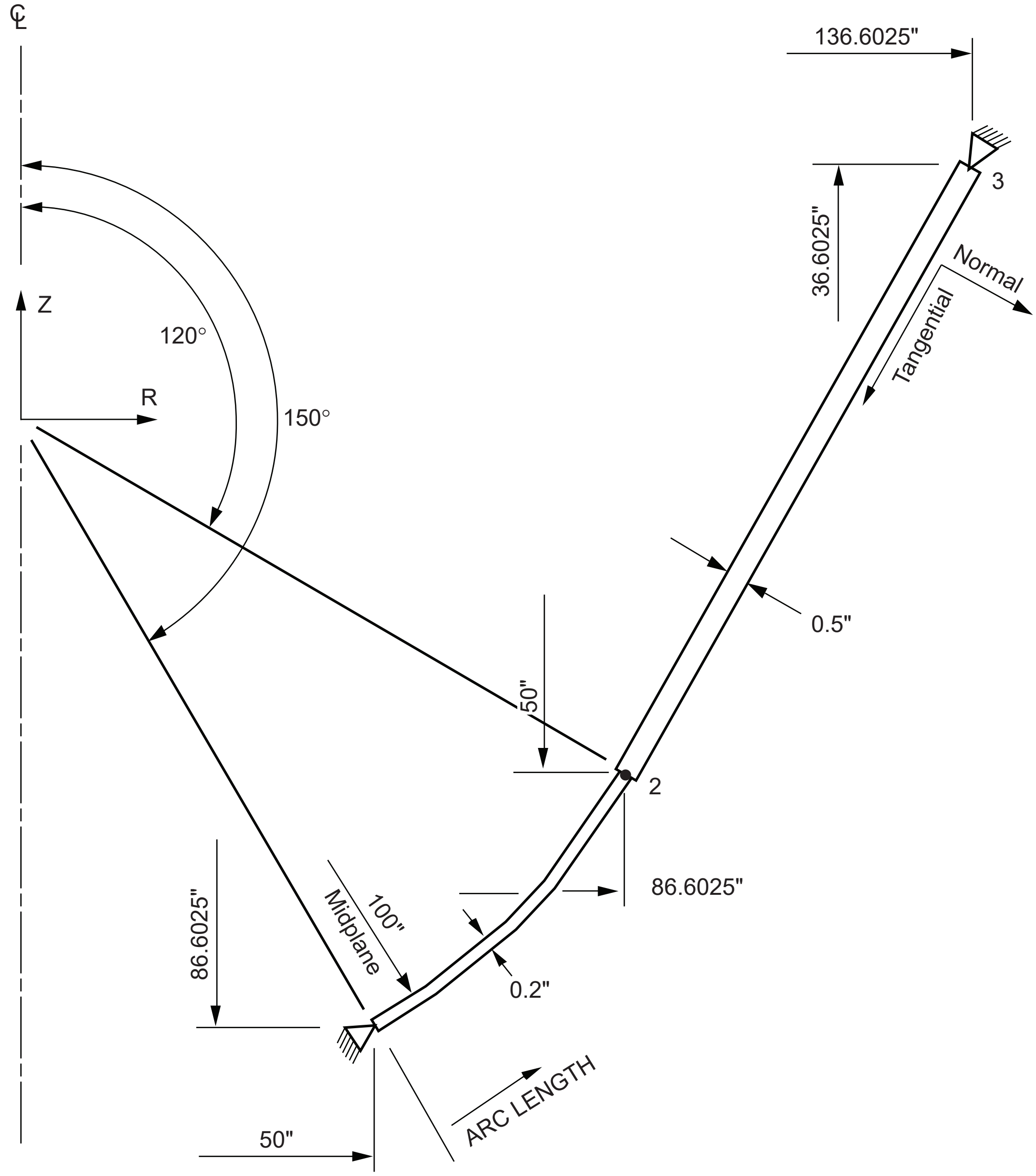
[a] *Refer to Figure 3.12-4*

Table 3.12-2

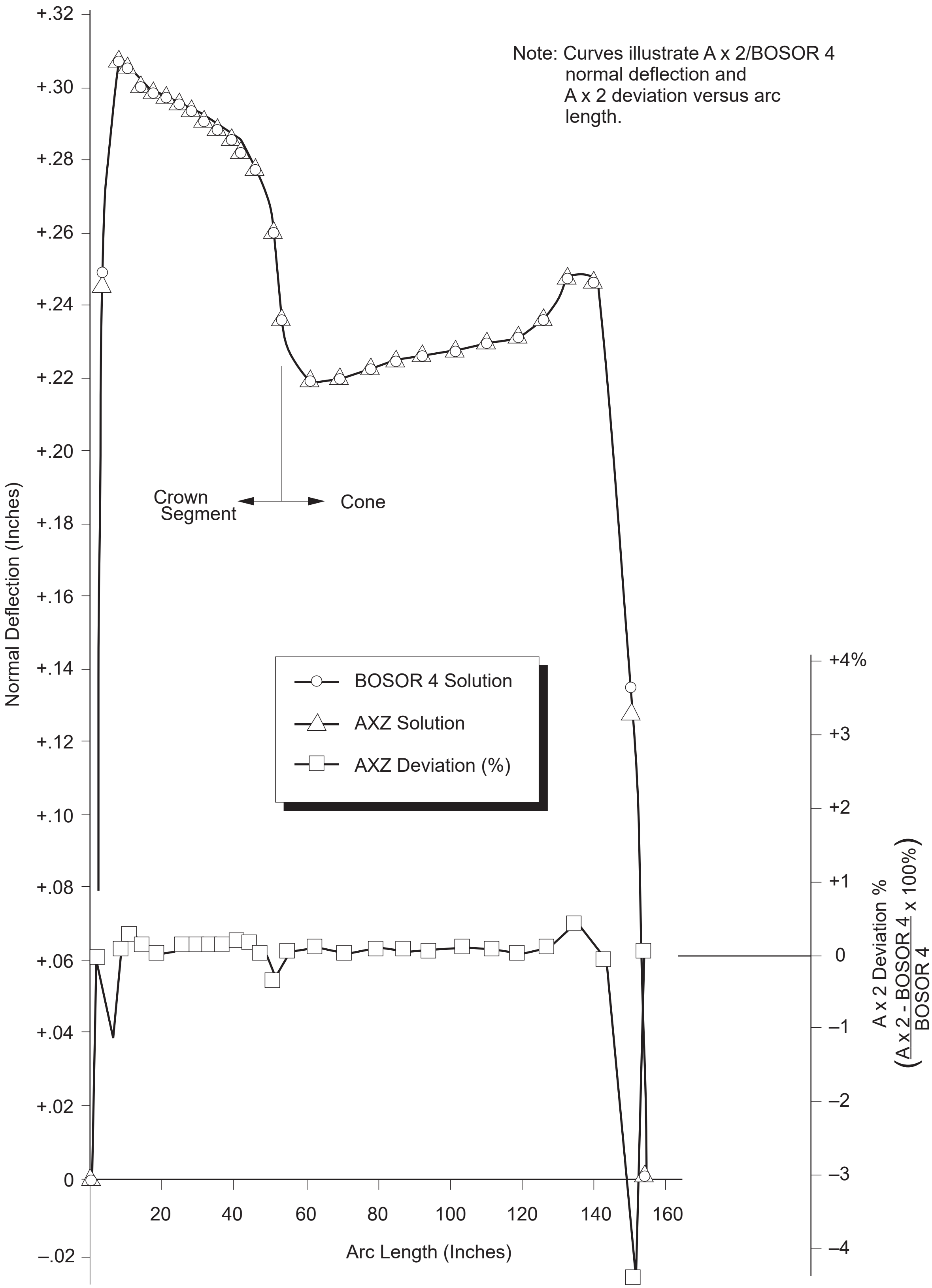
ISOFINITE Pressure Loads (Continued)

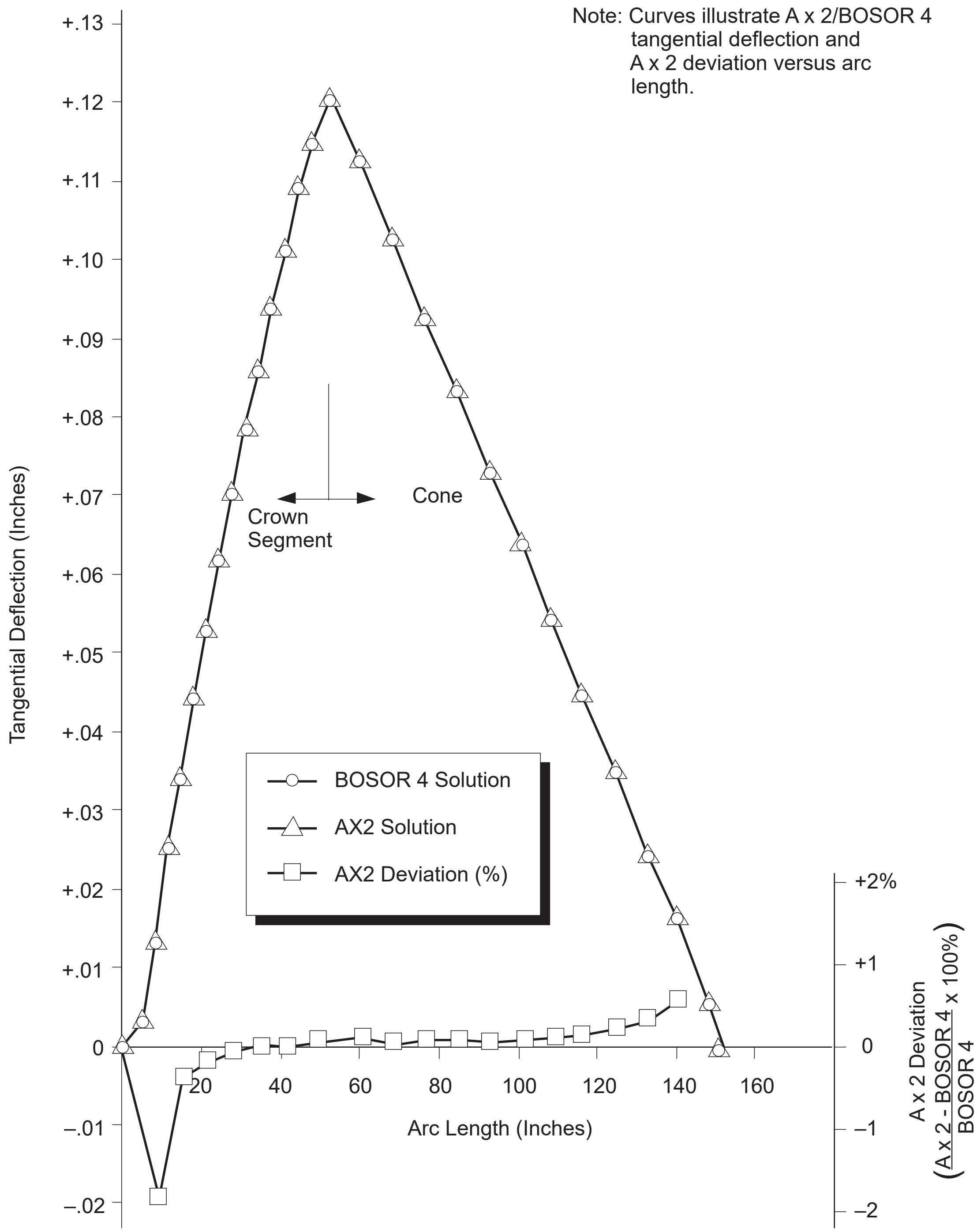
Point[a]	X-Stress (psi)	Y-Stress (psi)	Z-Stress (psi)	Maximum Stress Intensity (psi)
a	14,916	1702	8530	13,214
b	16,462	1656	15,401	14,806
c	9537	-1322	8619	10,859
d	5530	-114	4936	5644
e	4579	-169	3967	4748
f	3729	21	3148	3708
g	2527	-386	2047	2913
h	1725	-108	1716	1833
i	766	215	283	551
j	281	-10	-46	327
k	94	-170	331	501
l	44	50	384	340
m	-17	128	478	495
n	-94	41	509	603
o	-136	19	513	649
p	-100	-26	518	618
q	-64	-71	501	565
r	70	73	511	441

[a] Refer to Figure 3.12-4

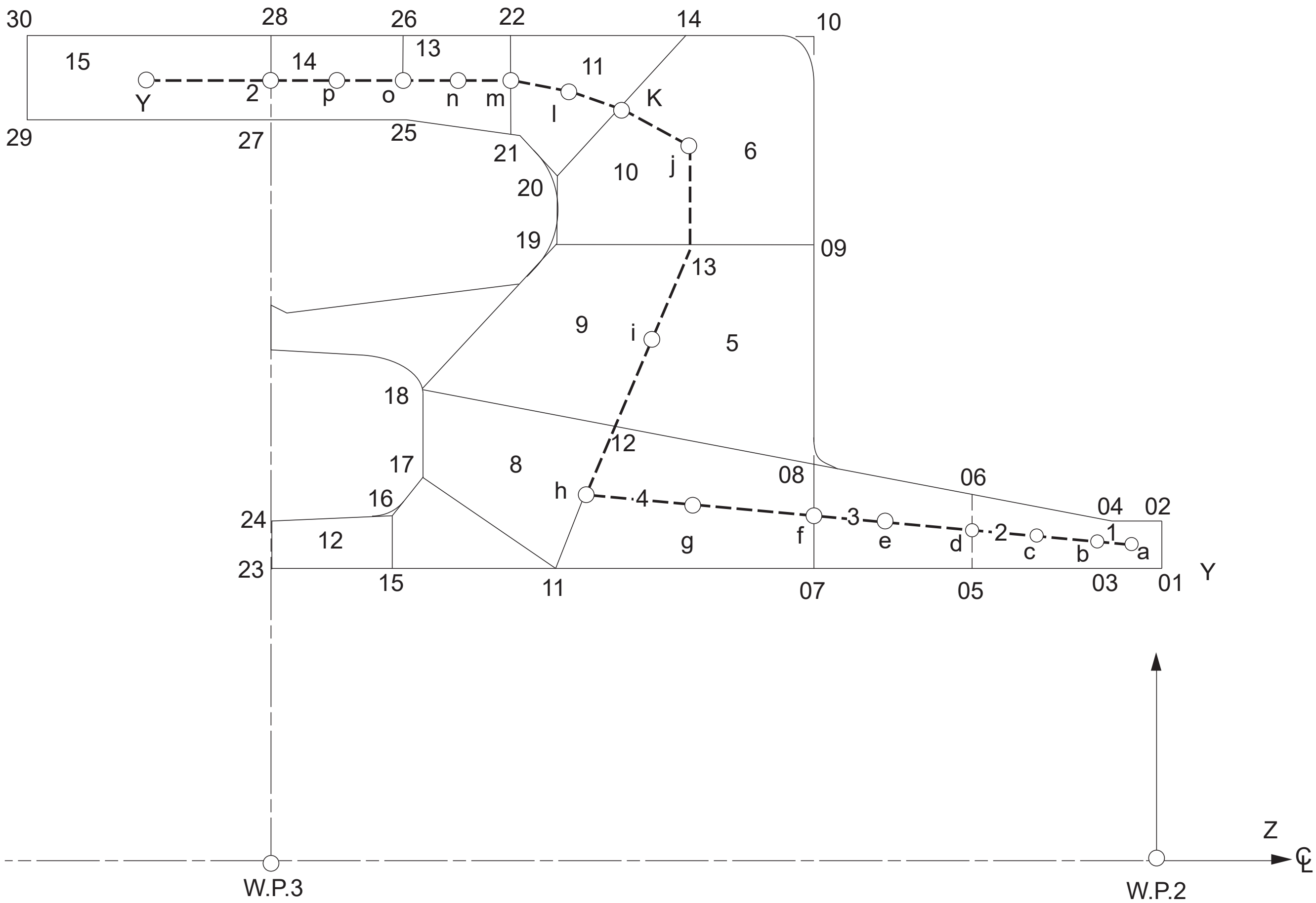


Loading: a) 20 psi (internal), Normal to Shell
b) 200°F rise above stress-free temperature


Note: Curves illustrate A x 2/BOSOR 4 normal deflection and A x 2 deviation versus arc length.
Normal Deflection (Inches)
+.32
+.30
+.28
+.26
+.24
+.22
+.20
+.18
+.16
+.14
+.12
+.10
+.08
+.06
+.04
+.02
0
–.02
Crown Segment
Cone
BOSOR 4 Solution
AXZ Solution
AXZ Deviation (%)
+4%
+3
+2
+1
0
–1
–2
–3
–4
A x 2 Deviation %
(A x 2 - BOSOR 4 / BOSOR 4 x 100%)
20
40
60
80
100
120
140
160
Arc Length (Inches)



Columbia Generating Station Final Safety Analysis Report	Tangential AX2/BOSOR 4 Verification Problem		
	Draw. No. 990306.98	Rev.	Figure 3.12-3



Columbia Generating Station
Final Safety Analysis Report

Isofinite Grid for Flued Head X2

Draw. No.	Rev.	Figure
990306.99		3.12-4

Appendix 3A

PLANT DESIGN ASSESSMENT REPORT FOR SAFETY/RELIEF VALVES AND LOSS-OF-COOLANT ACCIDENT LOADS

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LIST OF FIGURES (Continued)

Appendix 3A

PLANT DESIGN ASSESSMENT REPORT FOR SAFETY/RELIEF VALVES AND LOSS-OF-COOLANT ACCIDENT LOADS

LIST OF FIGURES (Continued)

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PLANT DESIGN ASSESSMENT REPORT FOR SAFETY/RELIEF VALVES AND LOSS-OF-COOLANT ACCIDENT LOADS

This third revision of the “Plant Design Assessment Report,” (DAR) together with the safety/relief valve (SRV), condensation oscillation and chugging reports, finalizes the Columbia Generating Station load definition, load application, load combination, and design margins for hydrodynamic loading conditions.

In July 1993 Energy Northwest requested an amendment to the operating license to allow an increase in the power level of the plant. The effects of power uprate on the containment system response are described in NEDC-32141-P. Specifically, Section 4.1 of NEDC-32141-P states that for short-term containment pressure response, the peak pressure values are below design values and remain virtually unaffected by power uprate and extended load line limit. The loss-of-coolant accident (LOCA) containment dynamic loads are not affected by power uprate, and SRV containment loads will remain below their design allowables. Appendix 3A has not been updated to reflect the minor changes to the LOCA and SRV load analyses described in NEDC-32141-P or the additional minor changes due to evaluations for Maximum Extended Load Line Limit Analysis (MELLLA).

3A.1.1 CONFORMANCE TO NRC ACCEPTANCE CRITERIA

The DAR specifies the Columbia Generating Station position for each of the pool dynamic loads. The table further provides a detailed description of each load, the NRC evaluation, and the Columbia Generating Station position on the acceptance for each load. The Columbia Generating Station positions (Attachment 3A.H) were discussed, reviewed, and approved by NRC at various meetings with Energy Northwest. The NRC acceptance was formalized in the Columbia Generating Station Safety Evaluation Report (SER).

3A.1.2 ROLE OF THE DESIGN ASSESSMENT REPORT

The Columbia Generating Station DAR serves the primary purpose of assessing the adequacy of structures and equipment affected during SRV actuation or a postulated LOCA. This report utilizes the load definition data from the SRV report, chugging report, DFFR (Reference 3A.3.2-2), and applicable NRC guidelines as outlined in Attachment 3A.H.

Specifically Revision 3 of the DAR serves the following purposes:

a. Summarize the loads and effects agreed upon with NRC which are most important to the design of the plant,

b. Identify the design modifications implemented to withstand these loads, and

c. Identify the design margins for hydrodynamic loading conditions.

3A.1.3 ASSESSMENT APPROACH

The information developed in the SRV and Chugging Reports (References 3A.3.1-1 and 3A.3.2-15 respectively) together with other information available as outlined in Attachment 3A.H, was used to assess all major structures, systems, and components in the wetwell region. The effects of hydrodynamic loads outside the wetwell region are discussed in the appropriate sections of the FSAR.

3A.1.4 SUMMARY OF DESIGN ASSESSMENT REPORT CONTENT

The DAR, Revision 3 is summarized as follows:

Chapter 1

a. Introduction to CGS Loads,
b. Review of the purpose of the plant specific loads, and
c. Discussion of the assessment of containment components since DAR Revision 2.

Chapter 2

a. General description of the CGS containment, and
b. Summary and conclusions.

Chapter 3

A discussion of the manner in which the plant specific loads for CGS are determined, based on information provided in the SRV and chugging reports, DFFR, and other associated and referenced documents.

Chapter 4

A review of the design assessment for the CGS containment system. Assessment and conclusions are included for suppression pool boundary structures (steel containment, vertical and horizontal tees, basemat, pedestal, diaphragm floor, and diaphragm floor seal) and for suppression pool major structures (downcomer bracing, columns, downcomers, SRV piping systems, quenchers, and platforms), and for suppression pool miscellaneous piping systems.

Chapter 5

Provides the building response due to SRV discharge, LOCA.

Attachments

The attachments to this report provide

a. Attachment not utilized,

b. Theoretical formulation for the calculation of three dimensional source flows in exact containment geometry,

c. A method to calculate drag forces on submerged structures caused by hydrodynamic flow fields,

d. Calculation models for short term LOCA phenomena,

e. Description of the suppression pool temperature monitoring system,

f. Description of computer programs utilized for CGS load definition and plant assessment,

g. Attachment not utilized,

h. Table of conformance of CGS design to NRC acceptance criteria, and

i. Summary of the methodologies used for defining SRV and LOCA loads on submerged structures.

3A.2 SUMMARY AND CONCLUSIONS

3A.2.1 GENERAL DESCRIPTION OF PLANT

The Energy Northwest Nuclear Project No. 2 (CGS) is a nuclear fueled electrical generating station which utilizes a General Electric Company BWR-5 (1969 product line) nuclear reactor.

The primary containment utilizes a Mark II over/under pressure-suppression configuration (see Figure 3A.2.1-1). The primary containment consists of a steel pressure vessel enclosed by a concrete shield wall both supported by a concrete basemat. The primary containment is enclosed by the reactor building, a reinforced-concrete structure functioning as a secondary containment.

The drywell is connected to the suppression chamber by 99 downcomer pipes. Originally 102 downcomer pipes were provided but three were capped, as discussed in Sections 3A.3.2.1 and 3A.3.2.2. Steam that could be released in the drywell during a postulated loss-of-coolant accident (LOCA) is channeled through these downcomer pipes into the suppression pool where it is condensed thus effecting pressure-suppression.

Eighteen safety/relief valves (SRVs) are mounted on the four main steam lines. When SRVs are actuated, steam from the reactor pressure vessel (RPV) flows through the SRV discharge lines into the suppression pool where the steam is condensed. The discharge lines from all 18 SRVs are routed inside selected downcomers into the suppression chamber *(Figures 3A.2.1-6 through 3A.2.1-8)*. Each discharge line terminates with a quencher device having four arms. Seven of the 18 SRVs are part of the automatic depressurization system (ADS) (Table 3A.3.1-1) which is designed to function, under certain conditions, following a postulated intermediate or small size line break.

3A.2.1.1 Structures, Piping, and Components Directly Affected by Pool Dynamic Loads

The structures in the suppression chamber are shown in *Figures 3A.2.1-2 through 3A.2.1-8.* The structures, piping and components directly affected by the hydrodynamic events associated with the LOCA pressure suppression and the SRV discharge are identified below. The applicable hydrodynamic loads are identified in Section 3A.4.

a. Boundary elements

The suppression chamber boundary elements are: the steel containment including the vertical and horizontal tee stiffeners (Figure 3A.4.1-4), the concrete basemat, the concrete pedestal, the diaphragm floor and the diaphragm floor seal;

b. Major structures and components

The major vertical structures are shown in *Figure 3A.2.1-2*. They are the 102 downcomers, the 18 SRV lines including quenchers and support towers, and the 17 concrete columns supporting the diaphragm floor. The major horizontal structures are the steel truss shown in *Figure 3A.2.1-3* which provides lateral support to the downcomers and the SRV lines, and the platform at el. 472 ft 4 in. shown in Figures *3A.2.1-3 and 3A.2.1-8*. The downcomer bracing truss is submerged and the platform is located in the pool swell zone; and

c. Miscellaneous piping systems

A developed elevation of the CGS containment showing the location of the containment penetrations is shown in Figure 3A.2.1-9. The piping systems of the suppression pool are classified as described below.

1. Fully submerged piping systems

Eleven piping systems are fully submerged in the suppression pool *(see Figure 3A.2.1-2)*. Seven systems enter the pool through containment penetrations at el. 452 ft 0 in. One pipe [4 in.-fuel pool cooling and cleanup (FPC)] enters the pool through the pedestal at el. 451 ft 8.25 in. Two short lengths of pipe (instrumentation stubs) enter the pool at el. 462 ft 0 in. A third (instrumentation line) enters at el. 455 ft 0 in. Pipes below the downcomer vent exit at el. 454 ft 4.75 in.

2. Partially submerged piping systems

Thirteen partially submerged piping systems enter the suppression chamber through containment penetrations at el. 467 ft 9 in. *(see Figure 3A.2.1-3)* and enter the pool vertically within 3 ft 0 in. distance from the containment as shown in *Figures 3A.2.1-6 through 3A.2.1-8*.

3. Piping systems in pool swell zone

The pool swell zone is identified in Section 3A.3.2.3 to be between the elevations of the initial pool surface (466 ft 4.75 in.) and the maximum pool rise during a LOCA (design basis accident) (484 ft 4.75 in.).

Piping systems in the pool swell zone include short projections into the chamber from the containment at one access hatch and 10 miscellaneous piping systems as shown in *Figure 3A.2.1-3 and Figures 3A.2.1-6 through 3A.2.1-8.*

4. Piping systems above the pool swell zone

   Piping systems above the pool swell zone include short lengths of pipe entering at el. 491 ft 0 in. and two penetrations for the wetwell spray header also at el. 491 ft 0 in. as shown in *Figure 3A.2.1-4 and Figures 3A.2.1-6 through 3A.2.1-8.*

3A.2.1.2 Structures, Piping, and Components Indirectly Affected by Pool Dynamic Loads

In the drywell region the containment structures, piping, and components are also affected by pool dynamic loads. This is a result of loading applied to the suppression chamber boundary (basemat, pedestal, and containment shell) which would result in vibratory motion transmitted through the reactor pedestal and the primary containment. This is commonly referred to as "building response."

3A.2.2 SUMMARY OF CHANGES AND CONCLUSIONS

3A.2.2.1 Summary of Changes to Preserve Design Margins

Structures, piping, and components which were affected by pool dynamic loads were divided into two general categories, i.e., those directly affected by pool dynamic loads (those in and bounded by the suppression chamber) and those affected only indirectly by pool dynamic loads (outside the suppression chamber). The Design Assessment Report (DAR) addressed the structures, piping, and components in and bounding the suppression chamber. For these structures several changes in design were implemented as a result of consideration of SRV discharge and LOCA hydrodynamic loads. Table 3A.2.2-1 provides a list of the structures and components that were impacted by the DAR and includes the design margins, controlling load combination, and the design changes that have been made. The steel containment structure has been reinforced by the addition of seven horizontal rows of tee stiffeners as shown in Figure 3A.4.1-1. The downcomer bracing system has been redesigned from a system of radial beams to a pipe truss system. This bracing system also is designed to provide lateral restraint for the SRV discharge pipes. Quenchers have been provided as exit devices for the SRV discharge pipes. Additions and modifications of pipe supports for miscellaneous piping systems have been provided. Other miscellaneous changes are noted in Table 3A.2.2-1.

3A.2.2.2 Conclusions

The DAR, Revision 3 concluded that the modified design of the wetwell for CGS is capable of withstanding the effects of the hydrodynamic loads resulting from SRV actuation and postulated LOCA events in conjunction with other applicable loads.

The effects due to hydrodynamic loads outside the wetwell region are discussed in the FSAR.

Table 3A.2.2-1

Suppression Pool Assessment Summary

Structure	Controlling Margin[a]	Controlling Load Combination[b]	Changes to Structures Due to SRV and LOCA Loads
Steel containment	1.26	3	Added horizontal tee stiffeners, revised platform location, and connection to containment.
Basemat	Bending - 1.14 Shear - 1.27	7	None
Pedestal diaphragm floor	1.11 Downward - 1.62 Uplift - 1.27	4 4a	None None
Diaphragm floor seal	See Section 3A.4.1.5.5	See Section 3A.4.1.5	None
Downcomer	1.68	5	Redesigned bracing system as a pipe truss system
Columns	1.19	1	None
Downcomers	1.08	See Section 3A.4.2.3	Added stainless-steel spool piece, provided local reinforcement where SRV pipe penetrates downcomer, raised the vacuum breaker valves out of the pool swell zone. Capped three downcomers.
SRV piping systems	1.05	See Section 3A.4.2.4	Provided lateral restraint at downcomer bracing system, rerouted SRV lines.
Quenchers	1.25	See Table 3A.4.2-3	Added quencher device and support to the end of SRV lines

Table 3A.2.2-1

Suppression Pool Assessment Summary (Continued)

Structure	Controlling Margin[a]	Controlling Load Combination[a]	Changes to Structures Due to SRV and LOCA Loads
Platforms and ladders	1.18	5a	Removed floor plate, replaced with grating, revised locations and connections to containment, added platform supports to service vacuum breaker valves; strengthened grating connections and supports.
Miscellaneous wetwell piping system	See Section 4.3	See Section 3A.4.3	Added and modified pipe supports. Stiffened penetrations. Relocated two piping systems.

[a] See Section 3A.4 for a complete discussion.

[b] See Section 3A.3.5 for a complete discussion.

3A.3 CONTAINMENT DYNAMIC FORCING FUNCTIONS

3A.3.1 LOADS ASSOCIATED WITH SAFETY/RELIEF VALVE ACTUATION

3A.3.1.1 Description of the Safety/Relief System

The safety/relief system is comprised of 18 safety/relief valves (SRVs) connected to the main steam lines in the drywell chamber. From each of the valves, a discharge line with two vacuum breakers is routed from the drywell into the wetwell where it terminates in a quencher in the suppression pool, as shown in *Figures 3A.2.1-3 to 3A.2.1-8*. To pass through the drywell floor, the discharge lines are routed through selected downcomers to about el. 490 ft. At this elevation, they exit the downcomers and are routed horizontally, as shown in *Figure 3A.2.1-4*, to points directly above their respective quenchers. They then are routed vertically down into the suppression pool, as shown in *Figures 3A.2.1-6 to 3A.2.1-8*. Under normal operating conditions, each quencher is filled with water and its discharge line is filled to the same level as the surface of the suppression pool. The remaining volume of the discharge line up to the SRV is filled with air. Table 3A.3.1-1 summarizes the characteristics of the safety/relief system.

3A.3.1.2 Description of the Phenomena and Resulting Loads

During plant operation, if one or more of the SRVs is actuated, three transient events occur consecutively for each:

a. The water in each quencher and discharge line is expelled into the suppression pool through the holes in the quencher arms,

b. The air which fills each discharge line is expelled into the suppression pool, and

c. The steam from the discharge line being vented is expelled into the suppression pool and condensed.

Each of these events creates disturbances in the suppression pool. The first creates water jets, the second creates air discharge related pressure oscillations and the third creates pressure fluctuations as the steam is condensed. These disturbances, in turn, produce hydrodynamic loads both on the structures which are submerged in the suppression pool and on the pool boundaries. Sections 3A.3.1.2.1, 3A.3.1.2.2, and 3A.3.1.2.3 briefly describe the characteristics of the load producing phenomena, while Section 3A.3.1.3 discusses the loads produced.

3A.3.1.2.1 Water Clearing Loads

When an SRV opens and permits steam to pass, the steam flow compresses the air above the water standing in the discharge line increasing the line pressure. The increased air pressure forces the water into the suppression pool through the holes in each side of the quencher arms. As the water flows from the adjacent sides of the quencher arms, it coalesces into four turbulent jets. These jets flow away from the quencher producing acceleration and standard drag loads on structures in their paths. Due to the turbulent nature of the jets, their momentum diffuses rapidly and their effective velocity decreases to zero in a distance comparable to a quencher arm length.

Based on the results of scaled experiments and Caorso test results (Reference 3A.3.1-3), the region throughout which the jets produce a significant load is small, at most existing to a distance comparable to a quencher arm length. There are no structures located in this region except a small intrusion of the concrete column which is designed for significantly higher air clearing loads. Detailed definition of water clearing loads therefore is unnecessary for the purposes of this report. It is noted that a clear demarcation between the water clearing loads and air clearing loads is not possible from the recorded Caorso test data. For the purpose of this report it is assumed that significant pressure peaks represent air clearing loads and that the slowly rising pressure before the first significant peak represents water clearing loads.

3A.3.1.2.2 Air Clearing Loads

Following the expulsion of water from the SRV discharge line, the air trapped in the line is forced through the holes in the sides of the quencher arms into the suppression pool. As a result of this disturbance of the pool, oscillations are produced in the pool which induce time varying pressure and velocity fields. These fields create acceleration drag loads on the submerged structures and time varying loads on the pool boundaries. The definition of these air clearing loads is provided in Section 3A.3.1.3.

3A.3.1.2.3 Steam Condensation Loads

After the water and air have been expelled from the SRV discharge line, high pressure, high temperature, high mass flux steam is discharged into the suppression pool. As the steam condenses and collapses, vibrations or small pressure fluctuations are produced in the water. Suppression pool temperature transient analyses were performed for CGS, for a stuck open relief valve, isolation scram, and small break accident (SBA), in accordance with the requirements of NUREG-0783. The peak pool temperature for each of these cases is maintained within the limits of NUREG-0783. As a result unstable steam condensation due to extended SRV discharge to the suppression pool will not occur. Details of this analysis are provided in Reference 3A.3.1-7. The suppression pool temperature monitoring system is discussed in FSAR Appendix 3A Attachment 3A.E.

3A.3.1.3 Safety/Relief Valve Air Clearing Loads

Testing and analytical efforts have been performed by the Mark II Owners' Group to define the loads resulting from discharge through a quencher device upon actuation of the SRV. The SRV testing carried out at the Caorso plant in Italy represents the most extensive test program to date with geometry and plant conditions similar to CGS. An analytical effort was undertaken by Burns and Roe to evaluate the data taken during the Caorso Phase I and II tests (References 3A.3.1-3 and 3A.3.1-4) which resulted in an improved SRV load definition and application methodology for Mark II containments (Reference 3A.3.1-1). These results and a detailed description of the analysis have been submitted, reviewed, and approved by the NRC (Reference 3A.3.1-6) as part of the SRV reports (Reference 3A.3.1-1).

Hydrodynamic loads due to an SRV discharge affect both pool boundaries and submerged structures. A summary of the improved load definition for the specific applications is provided in Sections 3A.3.1.3.1 and 3A.3.1.3.2.

3A.3.1.3.1 Boundary Loads

Expulsion of water and air in a discharge line during an SRV discharge creates disturbances in the suppression pool which induce dynamic pressure loads on the pool boundary. Resulting dynamic effects depend upon the definition of a rigid wall pressure incident on this boundary. Analytical interpretations and subsequent definitions of the spatial pressure distribution, pressure wave forms, and maximum pressure amplitudes recorded during the Caorso tests are the basis for defining the design boundary pressure load. Conversion of these Caorso results for application to CGS requires a correlation between the test conditions at Caorso and the design condition at CGS. Reference 3A.3.1-1 details this correlation along with the derivation of the design boundary pressure with regard to all the possible SRV discharge events that may occur during the life of the plant (see 3A.4.1.1.1.1). Sections 3A.3.1.3.1.1, 3A.3.1.3.1.2, and 3A.3.1.3.1.3 summarize the more important aspects of the derivation.

3A.3.1.3.1.1 Spatial Distribution of Boundary Pressures. The spatial distribution of boundary pressures during an SRV discharge contributes to a complete definition of the design boundary pressure load. Based on analytical studies of data recorded at Caorso, it was concluded that the spatial distribution of pressure is independent of the time variable. As stated in Reference 3A.3.1-1, this implies that the maximum pressure amplitude can be representatively used when studying the spatial distributions.

The circumferential pressure distribution (Figure 3A.3.1-1) as well as the vertical pressure distribution (Figure 3A.3.1-2) adopted for the SRV load specification is obtained through comparisons of various available distributions (Reference 3A.3.1-1). It should be noted that both distributions have been normalized for 1 psi peak pressure.

3A.3.1.3.1.2 Pressure Wave Forms. The SRV boundary pressure load specification depends on the evaluation of the pressure wave forms measured during the Caorso tests. Based on the experimental data recorded, two distinct characteristic wave forms prevail, the multiple frequency pressure (MFP) wave form and the single frequency pressure (SFP) wave form.

The design MFP wave form, shown in Figure 3A.3.1-3, reflects the characteristics of all such MFP wave forms measured at Caorso. Initially, there are several pressure spikes as the pressure wave reaches the pool boundary. They exhibit a frequency content in the range of 15.0 Hz to 40 Hz. Following the pressure spikes are damped oscillations exhibiting primarily a single frequency in the range of 6.0 Hz to 10.0 Hz. Maximum pressure amplitude occurs in the initial period and decays rapidly thereafter. Figure 3A.3.1-4 illustrates the frequency spectrum of the pressure amplitude trace indicating rich frequency content in the range of 15.0 Hz to 40.0 Hz and a distinct peak in the range of 6.0 Hz to 10.0 Hz. There is negligible frequency content beyond 40.0 Hz.

The design SFP wave form, shown in Figure 3A.3.1-5, reflects the characteristics of all such SFP wave forms measured at Caorso.

Unlike the MFP wave form, a single characteristic frequency of oscillation predominates for the entire time history. As shown in the corresponding frequency spectrum (Figure 3A.3.1-6), the dominant frequency is in the range of 6.0 Hz to 10.0 Hz. Again, there is negligible frequency content beyond 40.0 Hz.

Due to the randomness and variability in the characteristic/dominant frequencies of the pressure wave forms recorded, the time histories of the design pressure wave forms are compacted and expanded to obtain a characteristic frequency covering the range of 4.0 Hz to 12.0 Hz at intervals of 1.0 Hz. As a result, each type of wave form (MFP and SFP) is depicted by nine separate pressure time histories (see Reference 3A.3.1-1 for details).

3A.3.1.3.1.3 Design Maximum Pressure Amplitude. Conversion of the Caorso maximum pressure amplitude computed for application to CGS yields a design maximum pressure amplitude of 15.0 psi (References 3A.3.1-1 and 3A.3.1-5). This is the rigid wall pressure incident on the suppression pool boundary resulting from an SRV actuation. Application of this design boundary pressure for all SRV discharge cases that may possibly occur during the life of the CGS plant, as specified in the DFFR (Reference 3A.3.1-2), is discussed in Section 3A.4.1.1.1.1. As discussed in Reference 3A.3.1-2, the design pressure reflects a 90% confidence level and 90% probability of nonexceedence.

3A.3.1.3.2 Submerged Structure Loads

The methodology for calculating the loads on submerged structures during SRV discharge uses the predicted pressure time histories directly rather than the velocity and acceleration

transients. The pressure predictions are based on the pressures measured on submerged structures in Caorso tests and their correlation with the boundary pressure loads.

3A.3.1.3.2.1 Peak Safety/Relief Valve Dynamic Loads. The methodology used to define peak SRV dynamic loads on submerged structures is outlined in Attachment 3A.I (see also Reference 3A.3.1-8).

3A.3.1.3.2.2 Time Dependence of Safety/Relief Valve Loads and Dynamic Load Factors. The pressure time histories recorded on submerged structures at Caorso show waveform characteristics similar to those recorded at pool boundary. As indicated in Reference 3A.3.1-1, boundary pressure time histories consist of SFP time histories and MFP time histories.

Dynamic load factor (DLF) versus frequency curves are presented in Figure 3A.3.1-10. A typical curve, such as the curve labeled SFP, 1% damping, was calculated as follows:

a. Response spectra that correspond to all the SFP time histories described in Section 3A.3.1.3.1.2 are computed using 1% damping,

b. The response spectra obtained in (a) are enveloped, and

c. The DLF curve is obtained from the response spectrum envelope by dividing the responses at various frequencies by the zero period response.

3A.3.1.3.2.3 Safety/Relief Valve Loads on Structures. Loads on submerged structures are shown for the submerged structures listed below. Unless otherwise noted, calculation of an equivalent static load is completed via a DLF as obtained from Figure 3A.3.1-10.

a. Downcomers

Figure 3A.3.1-7 shows dynamic load on a downcomer;

b. Columns

Figure 3A.3.1-8 shows dynamic load on a concrete column. Note, for the subsequent actuation load case the maximum direct pressure load is shown (see Reference 3A.3.1-8). Reference 3A.3.1-9 computes the maximum dynamic reaction of the column, and these time-history analyses results are applied in Reference 3A.3.1-10 for final column structural (i.e., code) qualification. Table 3A.4.2-2 tabulates maximum column reaction load results;

c. Safety/relief valve line and quencher supports

Figure 3A.3.1-9 shows dynamic load on SRV line and the quencher support. Horizontal flow past an SRV line due to actuation of its quencher is negligible; and

d. Piping, supports, and bracing truss

Figure 3A.3.1-11 shows equivalent static loads on piping, supports, and bracing truss.

3A.3.1.4 References

3A.3.1-1 "SRV Loads - Improved Definition and Application Methodology for Mark II Containments," Technical Report, Burns and Roe, Inc., July 1980. Transmitted to NRC by letter GO2-80-172, of August 8, 1980.

3A.3.1-2 "Mark II Containment Dynamic Forcing Functions Information Report (DFFR)," General Electric Company, NEDO-21061, Revision 4, November 1981.

3A.3.1-3 "Mark II Containment Supporting Program Caorso SRV Discharge Tests, Phase I Test Report," General Electric Company, NEDE-25100-P, May 1979.

3A.3.1-4 "Mark II Containment Supporting Program Caorso SRV Discharge Tests, Phase II ATR Report," General Electric Company, NEDE-25118.

3A.3.1-5 Letter, GO2-82-35, 1/13/82, "Responses to CSB Open Items 44 through 48," G. D. Bouchey (WPPSS) to A. Schwencer (NRC).

3A.3.1-6 "Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2," NUREG-0892, Supplement No. 1, USNRC, August 1982.

3A.3.1-7 "Suppression Temperature Analysis," Stone and Webster Report 14057-4 (D-1). Transmitted to NRC by letter GO2-81-524 of December 15, 1981.

3A.3.1-8 Supply System Calculation No. ME-02-93-22, "SRV Air Clearing Loads on Diaphragm Slab Columns."

3A.3.1-9 Supply System Calculation No. CE -02-93-12, "Diaphragm Slab Column Evaluation for SRV Air Clearing Loads."

3A.3.1-10 Burns & Roe Calculation No. 6.19.36, Book SV-830, "Calculation for Reactor Building Diaphragm Floor Column and Plant DAR for Hydrodynamic Loads, Rev. 3."

Table 3A.3.1-1

Summary of Safety/Relief System Characteristics

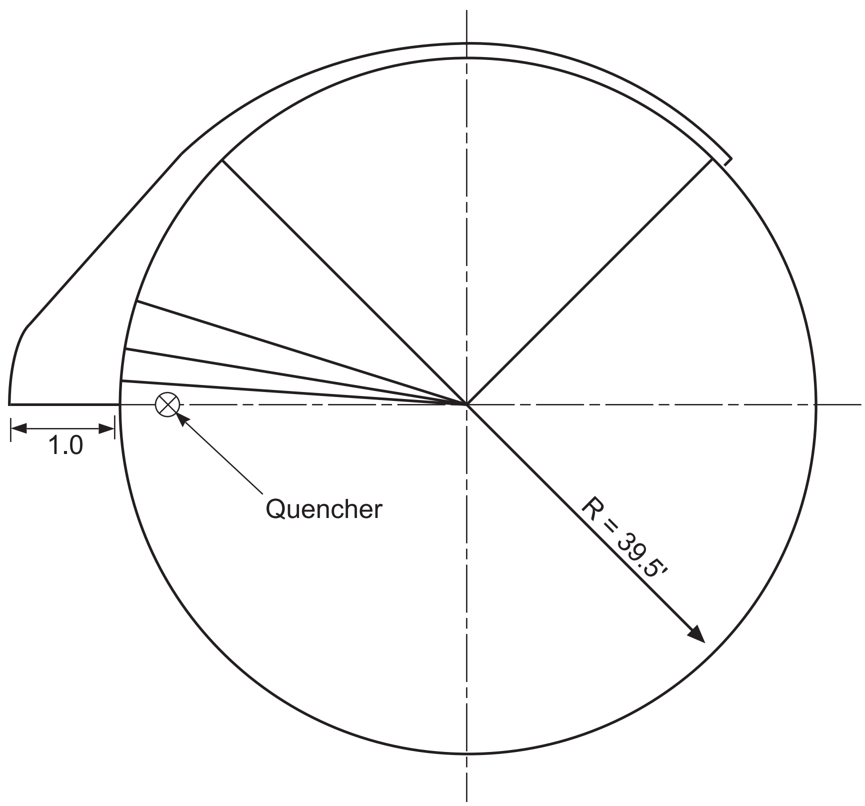
1.	Number of SRVs	18	
2.	SRV manufacturer	Crosby	P.S.P
3.	Designations of SRVs and pressure setpoints (Refer to *Figure 3A.2.1-2*)	<u>Valves</u>	<u>(psi)</u>
		1B[a], 1C[a]	1165
		1A[a], 2B[a], 2C[a], 1D[a]	1175
		2A, 3B, 3C, 2D	1185
		3A, 4B, 4C, 3D	1195
		4A, 5B, 5C, 4D	1205
4.	Number of automatic depressurization valves [automatic depressurization system (ADS)]	7	
5.	Designations of automatic depressurization valves	MS-RV-3D, 4A, 4B, 4C, 4D, 5B, 5C	
6.	Nominal range of valve opening times (ms)	20-150	
7.	Nominal range of nameplate steam rates (lbm/sec)	236.33-251.81	
8.	Number of vacuum breakers per discharge line	2 (in parallel)	
9.	Size of each vacuum breaker (in.)	10	
10.	Each discharge line pipe sizes (in./schedule)	10/80 expanded to 12/80 at approximate el. 493 ft	
11.	Discharge line rage of lengths to normal water level (ft)	107.99-161.99	
12.	Discharge line range of air volumes to normal water level (ft^3)	57.2-88.1	
13.	Depth of suppression pool at RPV pedestal to normal water level (ft)	31.0	
14.	Submergence of quenchers to high water level (ft)	17.4	

Table 3A.3.1-1

Summary of Safety/Relief System Characteristics (Continued)

15. Elevation of inner/outer ring quenchers above basemat (ft)	13.6-8.2
16. Quencher area defined by circumscribed circle (ft^2)	74.6

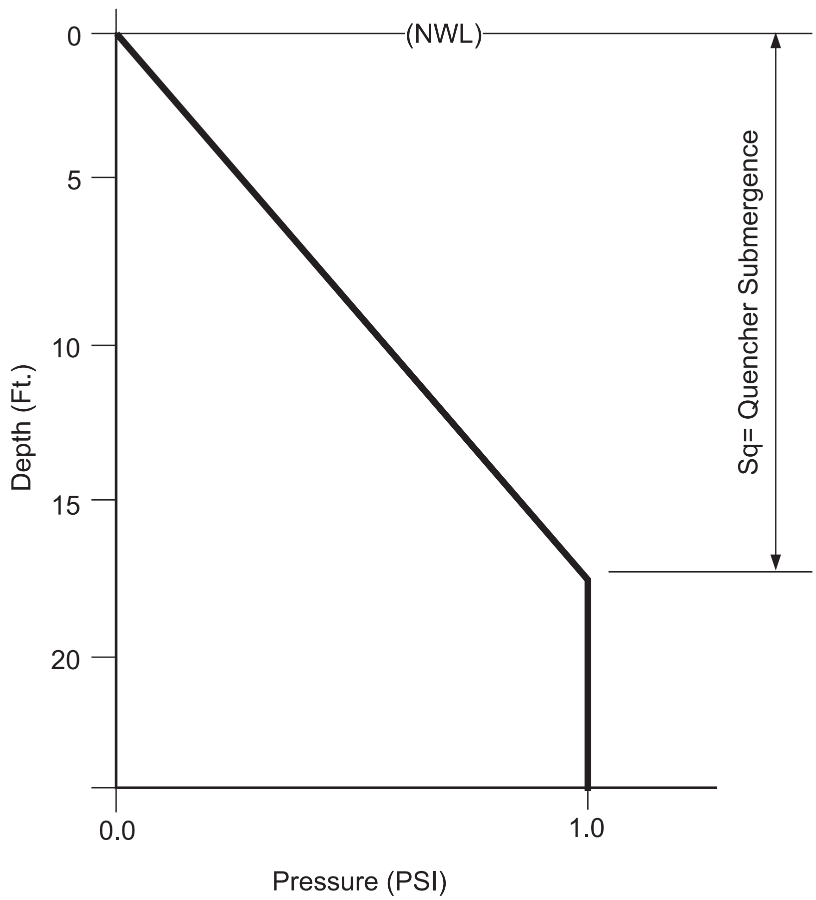
[a] These valves are the low setpoint valves which are prone to subsequent actuation. However, subsequent actuation may occur (though unlikely) in the higher set pressure SRV groups (see also discussion contained in Reference 3A.3.1-8).



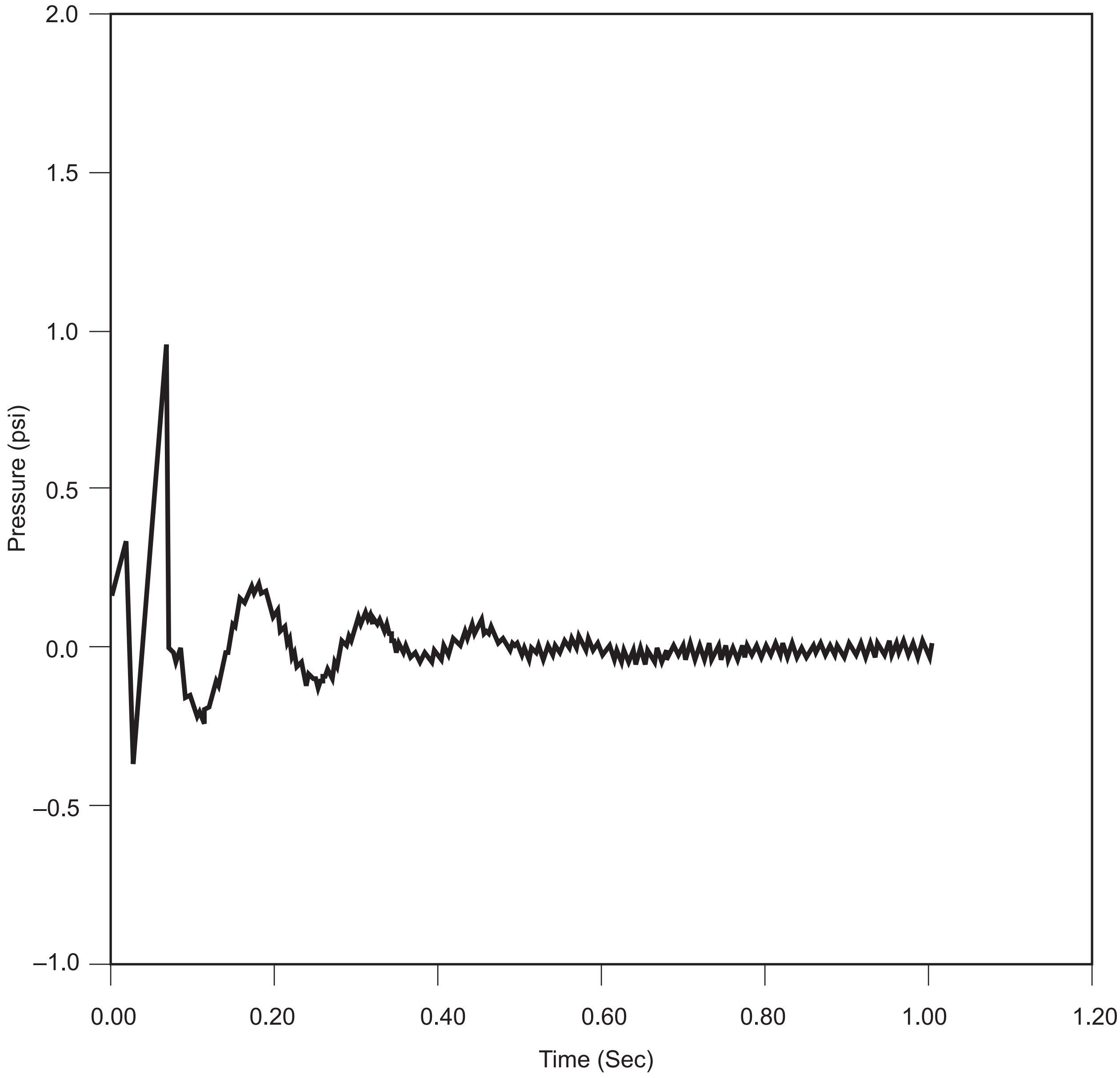
Columbia Generating Station
Final Safety Analysis Report

Normalized Design Circumferential Distribution of Pool Boundary Pressures at Containment

Draw. No. 950021.64	Rev.	Figure 3A.3.1-1

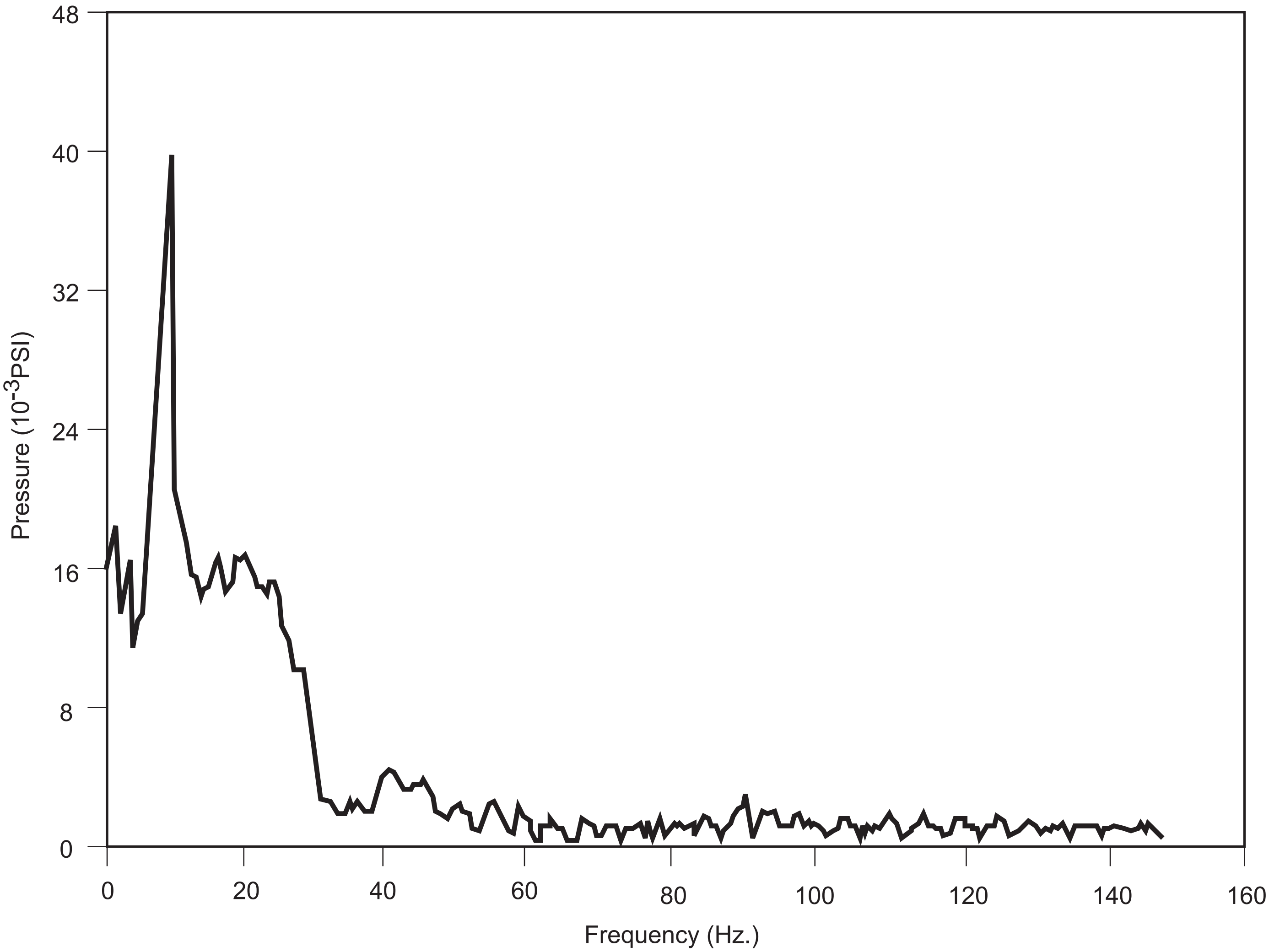


Normalized Design Vertical Distribution of Pool Boundary Pressures at Containment



MFP Design Wave Form (Normalized)
Time History

Draw. No. 950021.66 | Rev. | Figure 3A.3.1-3



Columbia Generating Station
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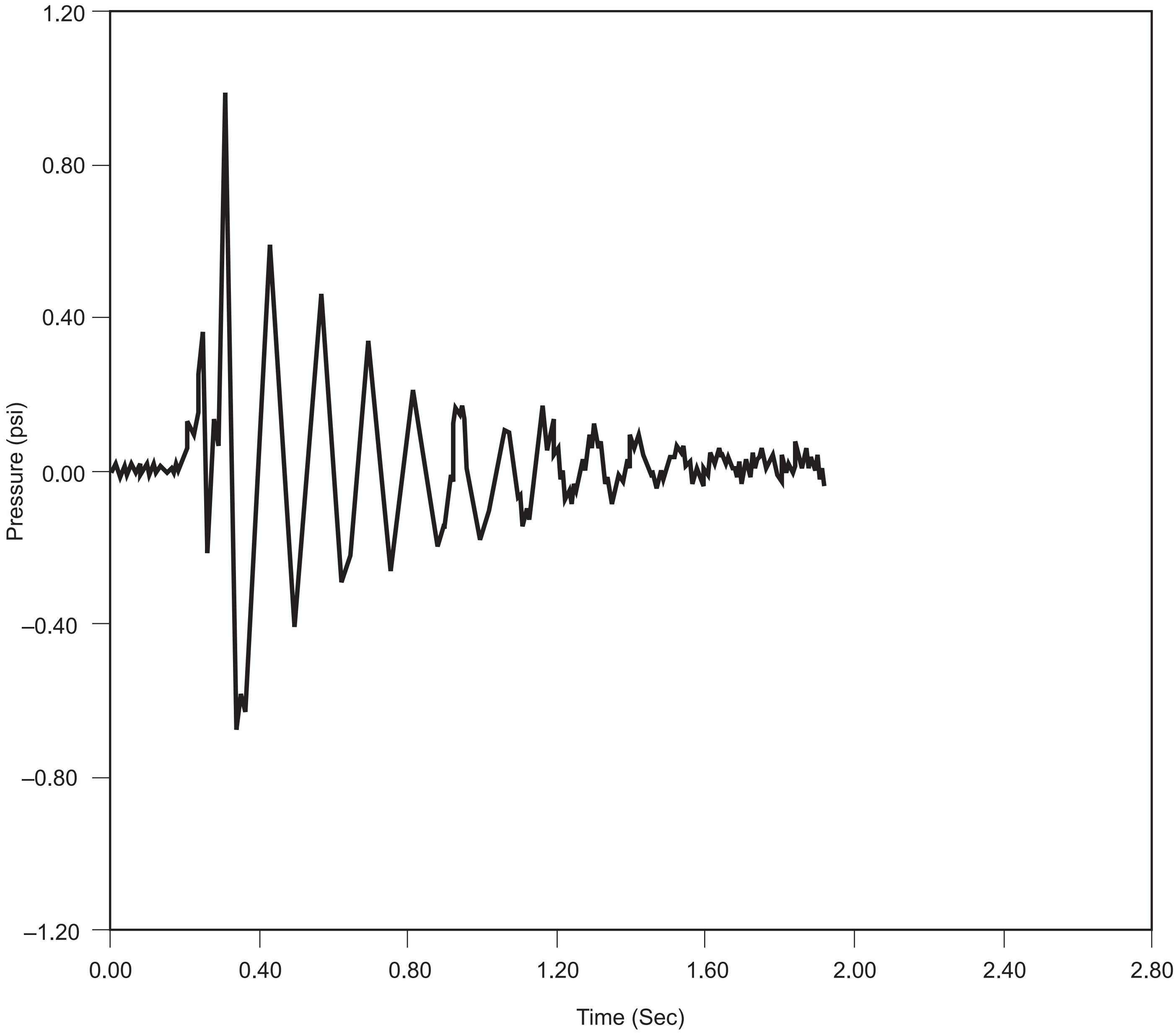
MFP Design Wave Form (Normalized) Amplitude of Frequency Spectrum

Draw. No. 950021.67 | Rev. | Figure 3A.3.1-4



Columbia Generating Station
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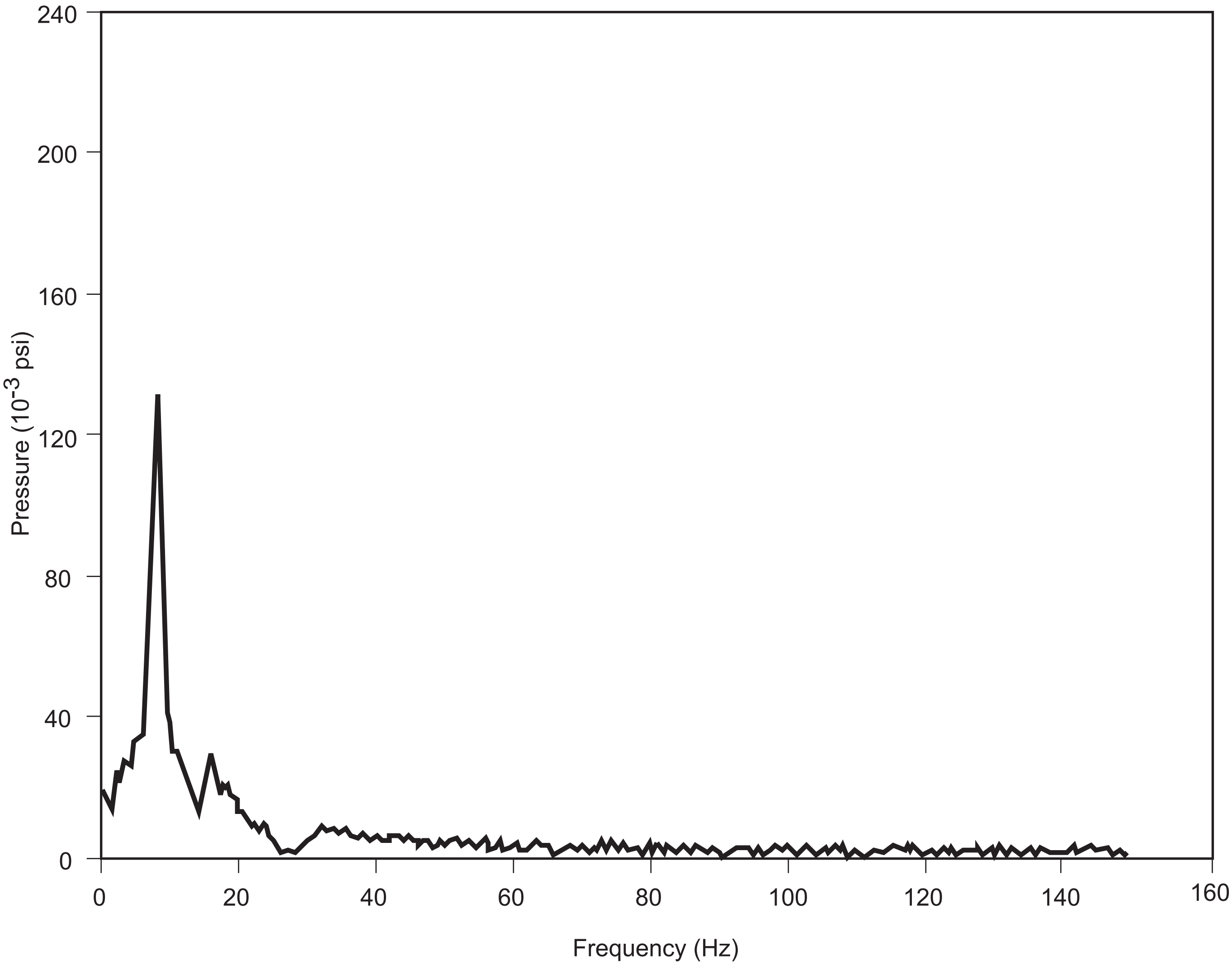
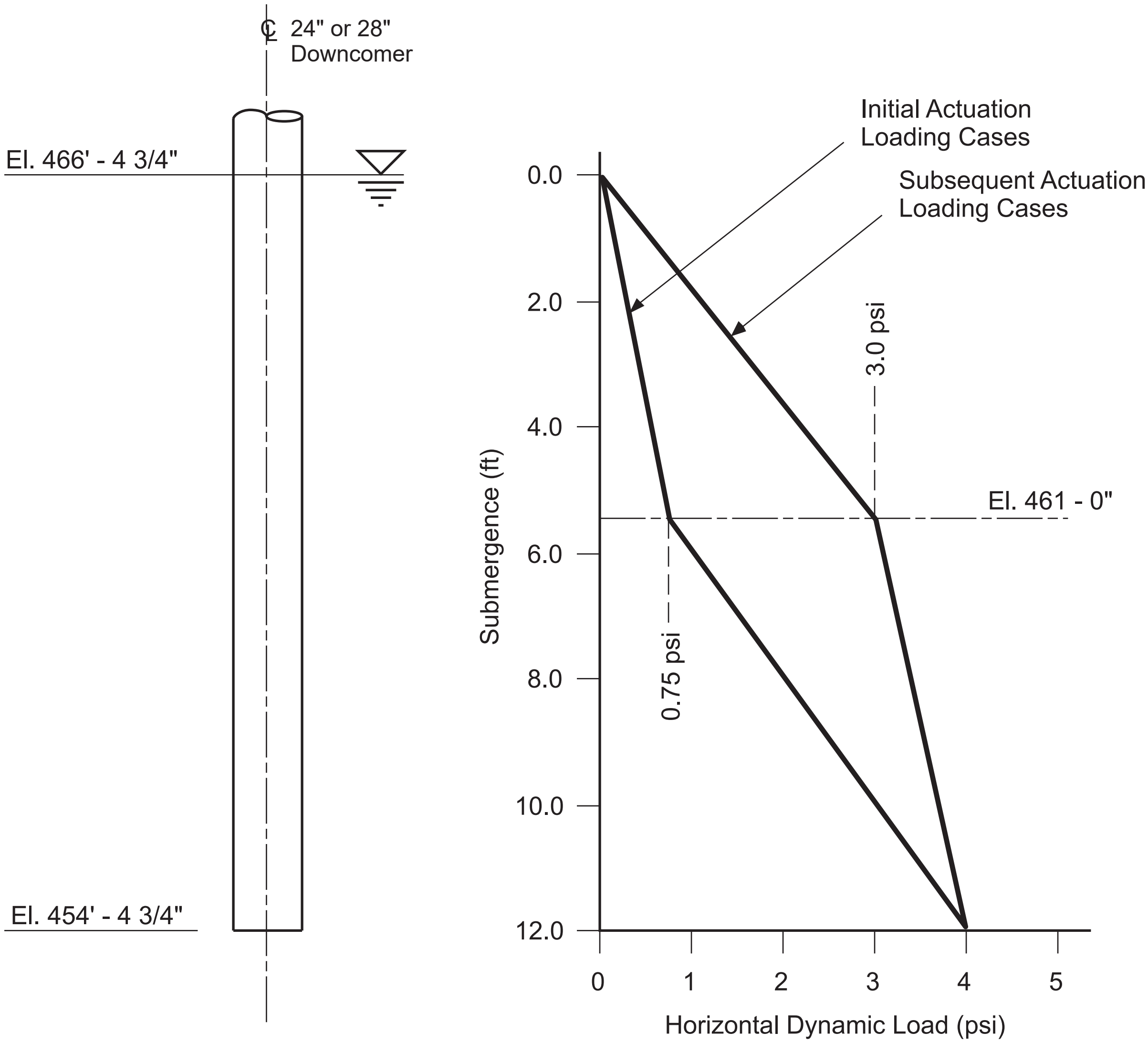
SFP Design Wave Form (Normalized)
Time History

Draw. No. 950021.68 | Rev. | Figure 3A.3.1-5



SFP Design Wave Form (Normalized)
Amplitude of Frequency Spectrum

Draw. No. 950021.69 | Rev. | Figure 3A.3.1-6

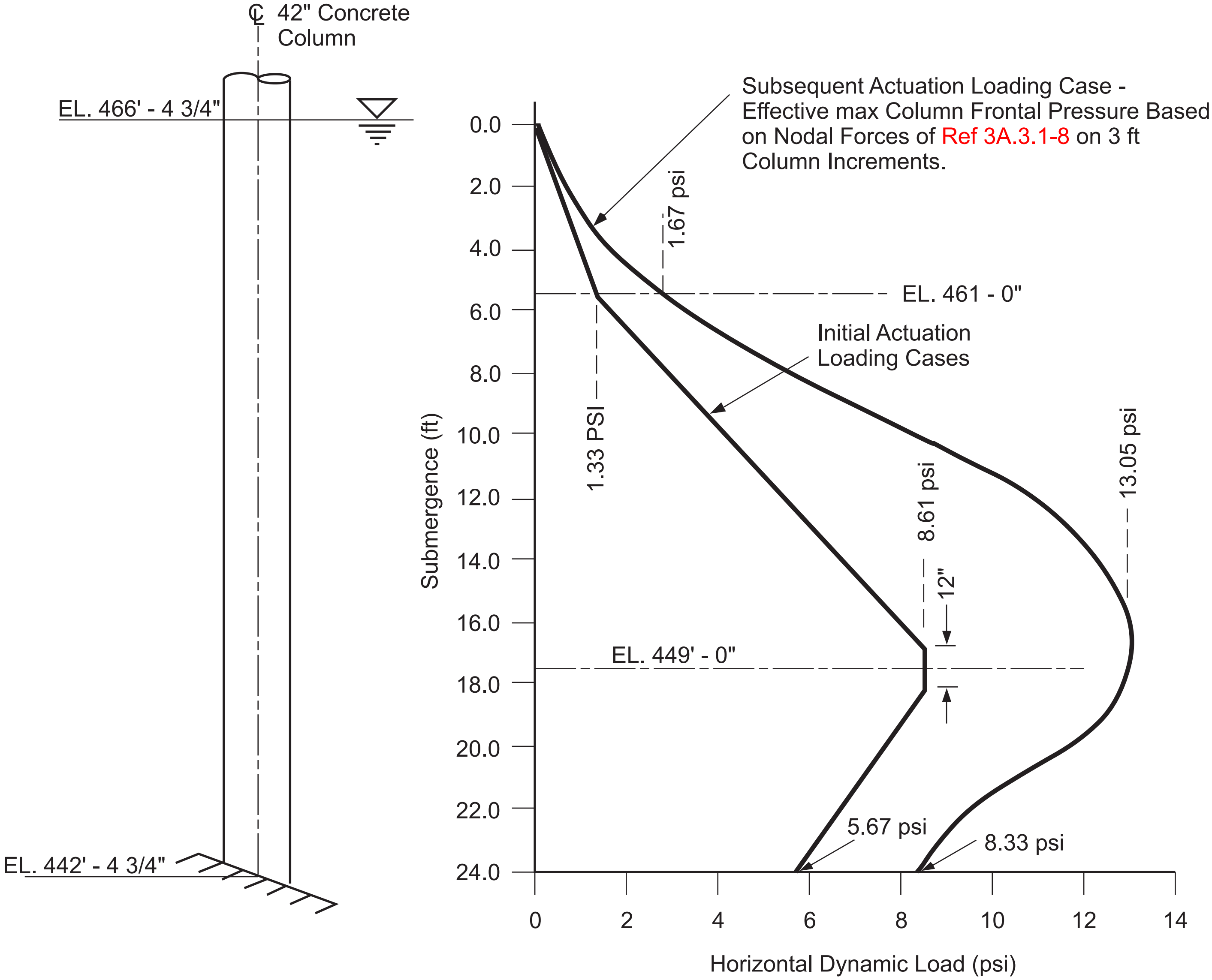


Notes:

1. Dynamic load factors are obtained from Figure 3A.3.1-10.

2. Horizontal loads on downcomers are applied in any horizontal direction producing maximum load effects.

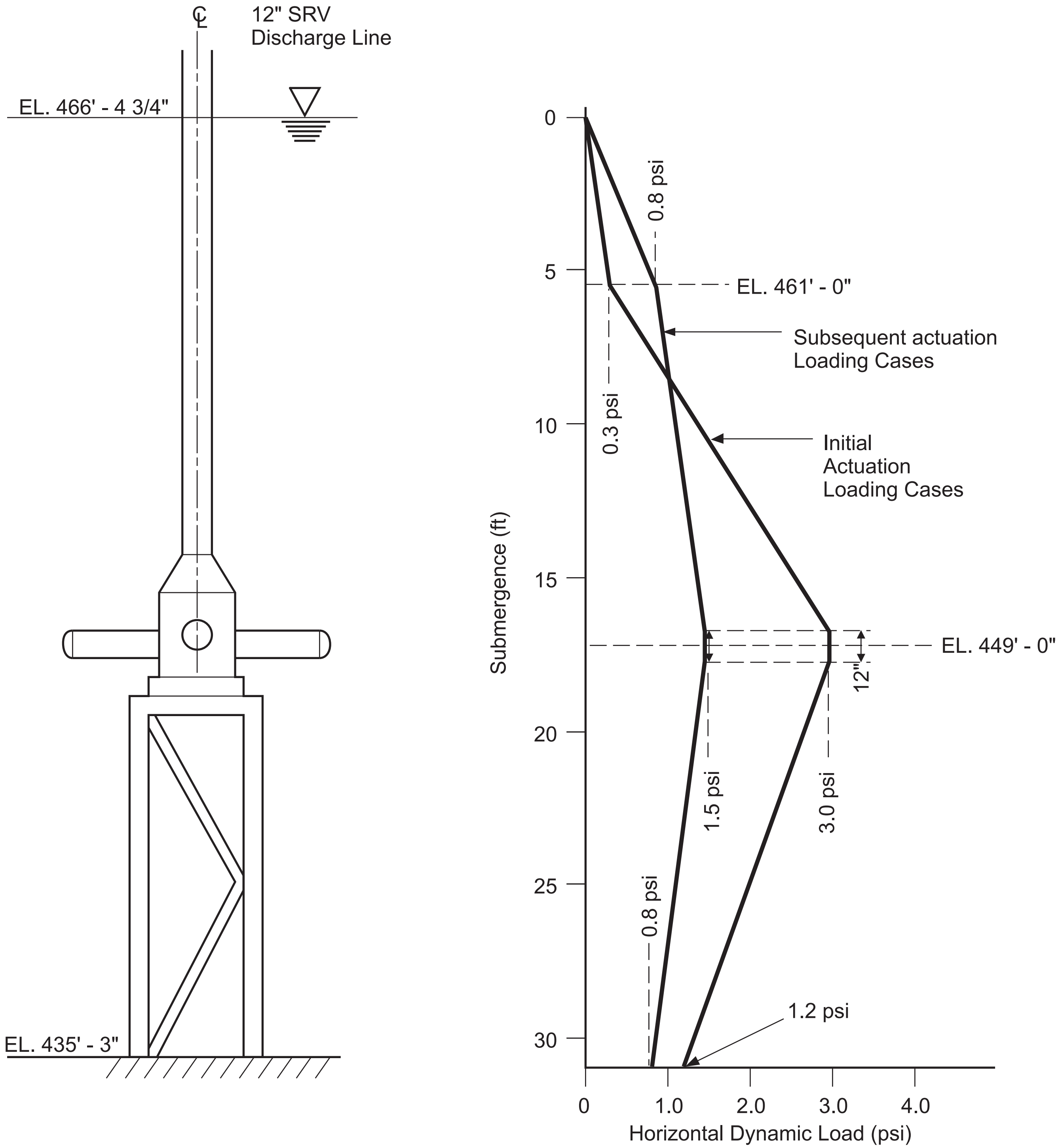
Columbia Generating Station Final Safety Analysis Report	**SRV Air Clearing Load Distribution on a Downcomer**		
	Draw. No. 950021.70	Rev.	Figure 3A.3.1-7



Notes

1. Dynamic load factors for initial actuation load cases are obtained from Figure 3A.3.1-10, dynamic load response reactions for the subsequent actuation case are tabulated in Table 3A.4.2-2 (see also References 3A.3.1-9 and 3A.3.1-10).

2. Horizontal loads on concrete column are applied in any horizontal direction producing maximum load effects.



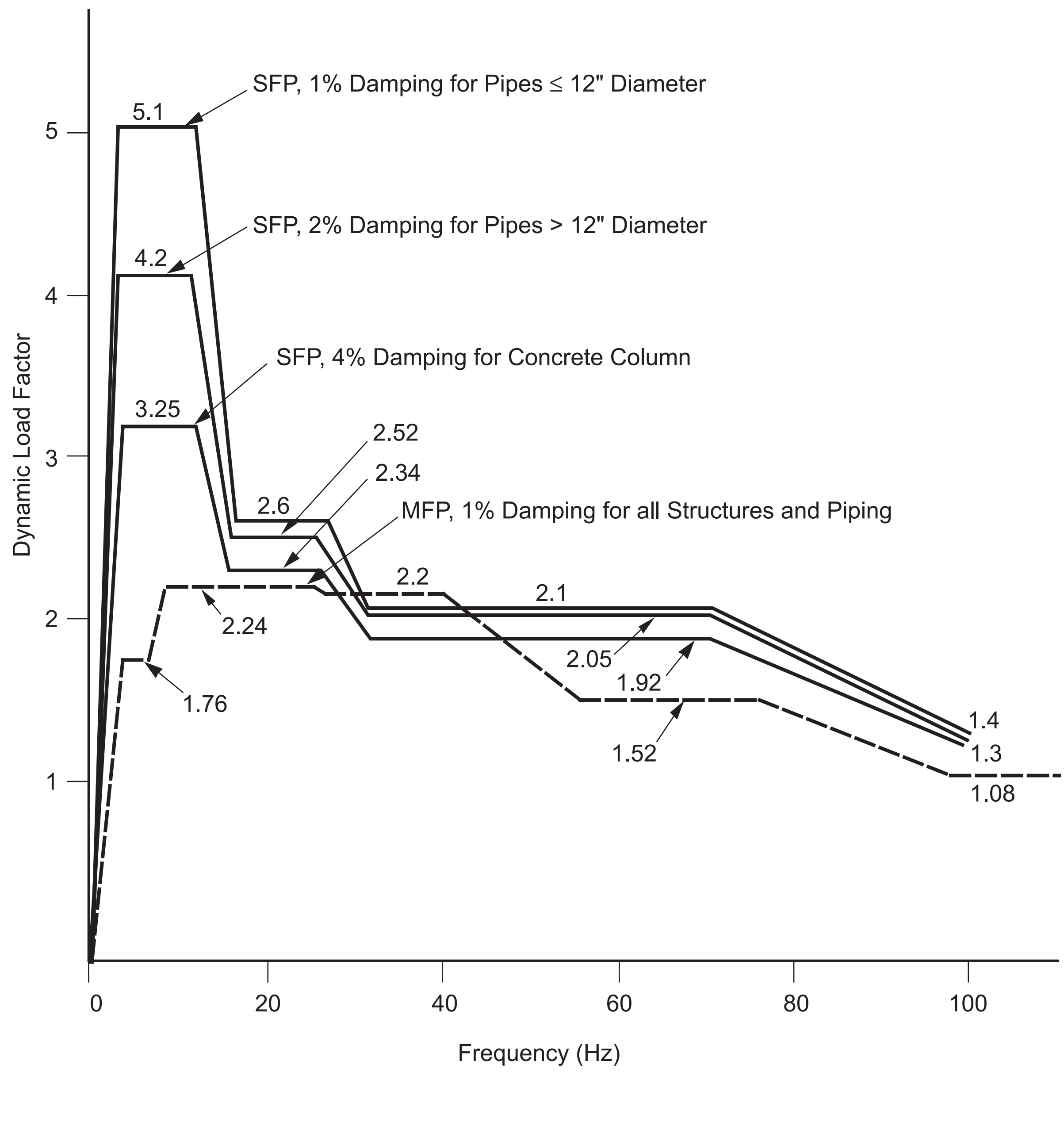
Notes

1. Dynamic load factors are obtained from Figure 3A.3.1-10.
2. Horizontal loads on SRV lines are applied in any horizontal direction producing maximum load effects.

SRV Air Clearing Load Distribution on an SRV Discharge Line and Quencher Support

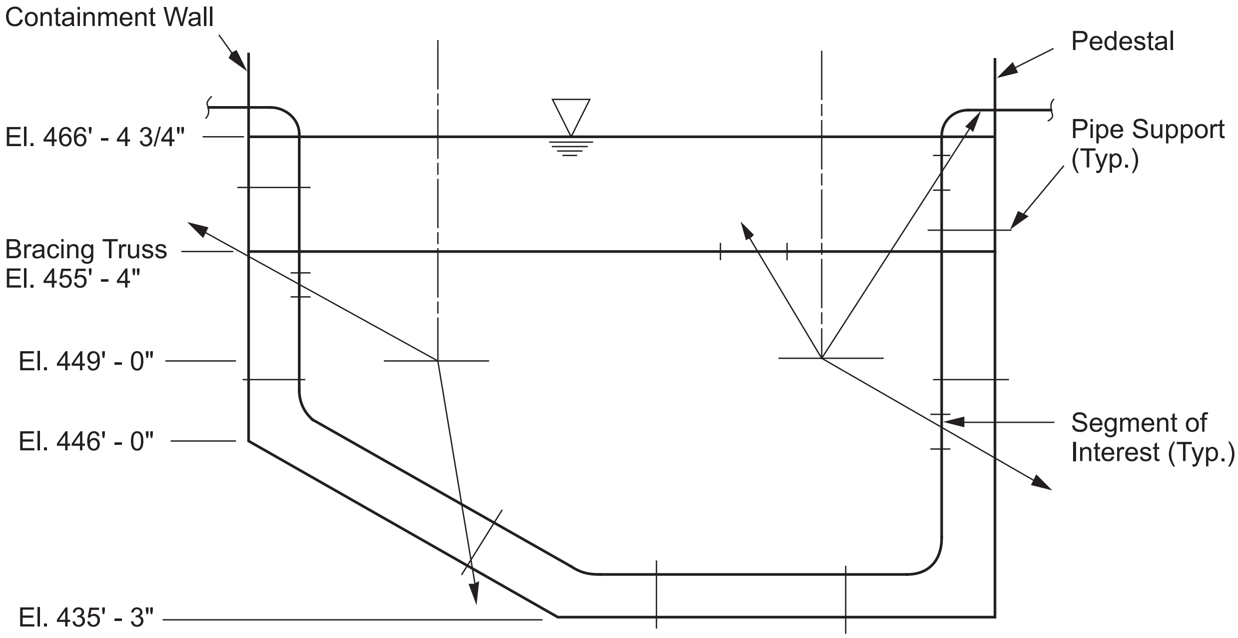
Note:

1. The curve labeled as MFP is used with initial actuation cases and the curves labeled as SFP are used with the subsequent actuation cases.

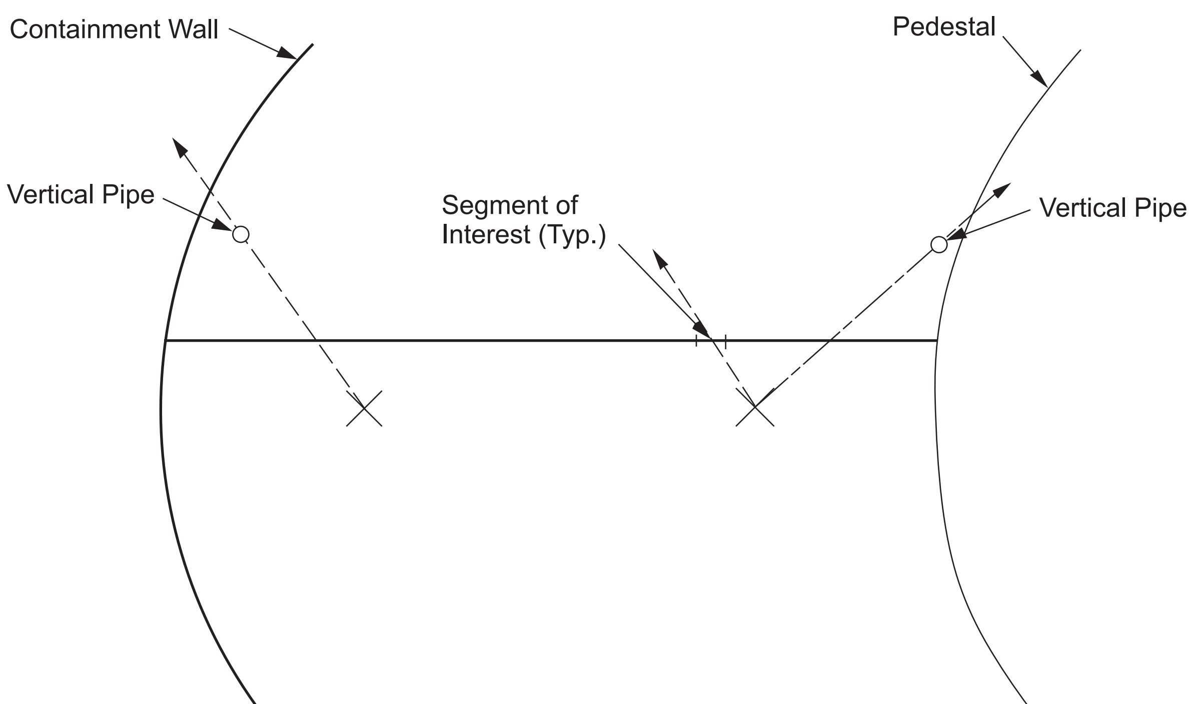


Dynamic Load Factor Versus Frequency to be Used for Defining SRV Load on Submerged Structures

Draw. No. 950021.73	Rev.	Figure 3A.3.1-10



ELEVATION



PLAN

Notes:

1. The load is applied along the line joining the center of the nearest actuating quencher and the geometric center of the pipe segment as shown in Fig. 3A.3.1-11a.

2. The equivalent static load ($p_1$lb) on piping, supports and bracing truss is:

$P = \pm 0.32D^2L$	for initial actuation.
$P = \pm 0.25KD^2L$	for subsequent actuation and for pipes near containment or pool bottom.
$P = \pm 0.64KD^2L$	for subsequent actuation and for pipes near pedestal.

   where:

   D = diameter of pipe or the cylinder circumscribing a non-circular cross-section, in units of inches.

   L = length of segment, in units of inches

   K = see note 5

3. For load on piping, supports and bracing truss, the load component parallel to the pipe is neglected.

4. Since the load direction on piping, supports and bracing truss generally varies from one point to another, segmentation of the piping or structural component may be needed.

5. If the fundamental natural frequency of the piping, support or bracing truss member is greater than or equal to 17HZ, K = 1. This is applicable in most cases.

   If the fundamental natural frequency is less than 17HZ, then $K = \frac{DLF}{2.6}$,

   where DLF is the dynamic load facor and is determined from Fig. 3A.3.1-10. If K, as calculated above, is less than 1, K = 1 is used.

3A.3.2 LOADS ASSOCIATED WITH LOSS-OF-COOLANT ACCIDENT

A loss-of-coolant accident (LOCA) occurs when the integrity of the reactor coolant pressure boundary is breached and coolant is released. In order to contain the coolant which flashes to steam, CGS utilizes a GE Mark II pressure suppression system. The LOCA loading phenomena are discussed in Section 3A.3.2.2. The short-term and long-term LOCA loads are discussed in detail in Sections 3A.3.2.3 and 3A.3.2.4, respectively. Section 3A.3.2.5 describes the LOCA pressure and temperature transients and Section 3A.3.2.6 describes the CGS building response to the LOCA loads.

3A.3.2.1 Description of Pressure Suppression System

The CGS primary containment utilizes a GE Mark II over/under pressure-suppression configuration (see Figure 3A.2.1-1). The drywell and suppression chamber (or wetwell) are large sealed volumes designed to contain and condense escaping reactor coolant. Both contain structures and piping systems with the suppression chamber approximately half filled with water (suppression pool) for steam quenching. The drywell is connected to the suppression pool by 99 downcomer pipes that channel steam released during a LOCA for quenching and pressure suppression. Details of the downcomers, other piping systems and structures in the suppression chamber are shown in *Figures 3A.2.1-2 through 3A.2.1-8*.

Originally, 102 downcomer pipes were provided. During the assessment of the wetwell piping for hydrodynamic loads, it was found that the LOCA water jet loads on three containment vessel penetrations and their supports were excessive. To eliminate these loads, three of the 102 downcomers are capped in the drywell region, leaving 99 for venting steam to the wetwell.

The piping systems involved are at penetration X-31, X-34, and X-36. The three capped downcomers are on the outer circle of downcomers closest to the containment vessel, at azimuths 95°, 63°, and 42° *(Figures 3A.2.1-2 and 3A.2.1-4)*.

3A.3.2.2 Description of the Phenomena and Resulting Loads

The sequence of LOCA-generated hydrodynamic events described below cause dynamic loads on the containment and on structures and components located in the wetwell. These transient dynamic forces (see Table 3A.3.2-1) are termed dynamic forcing functions and are discussed in detail in Reference 3A.3.2-2 and summarized below. Section 3A.3.4 discusses the sequence of LOCA generated loads.

Following a postulated LOCA, released coolant causes the drywell pressure to rise rapidly and to accelerate the column of water in each downcomer downward due to the pressure rise. As the water exits the downcomers and enters the suppression pool, it forms a jet-pool interface which rolls into a mushroom shaped vortex ring. Expulsion of water out of *each of the*

downcomers results in a water jet which produces loads on submerged structures and suppression pool boundary pressure loads. Because bulk pool velocities are small during vent clearing, the corresponding impact and induced flow field drag loads are generally small. However, locally, significant drag loads may result.

Immediately after vent clearing, air* in the downcomer vents from the drywell beings to flow into the suppression pool. The LOCA air* bubbles are formed at the exits of the vents which charge and expand under the entire pool surface causing three dimensional drag loads on submerged structures. On contact with each other, the individual bubbles coalesce and accelerate the pool water above the downcomer vent exit plane vertically with no significant horizontal water motion.

Pool swell is the upward movement of suppression pool water above the exit plane of the vents due to injection of drywell air* below the pool surface. The velocity and acceleration of the water slug associated with this phenomenon produces impact, drag, and lift forces on structures within the swell zone.

The containment boundaries also experience loads due to drywell pressurization, air bubble pressure and wetwell free airspace compression. The rising pool surface motion is slowed due to the compression of air in the suppression chamber airspace. At about this time the rising bubbles break up the remaining pool slug which falls back to its original position terminating pool swell.

During pool swell, the bottom of the rising water slug continually falls back to the suppression pool due to instabilities at the bubble/water slug interface. This phenomenon and the large scale falling back of the remaining water slug at pool swell termination is known as fallback and causes drag and lift loads on structures in the pool swell zone, but a negligible containment boundary load.

Pool swell and the subsequent fallback of the remaining water slug are followed by an air/steam mixture flow through the downcomers until the drywell air is completely purged and the mass flux becomes pure steam. The loading phenomena associated with a high or medium steam mass flux is termed steam condensation oscillation (CO).

The air content and steam mass flow rate along with the pool temperature determines the behavior of the steam/suppression pool water interface. At high steam flow the interface location is essentially constant. As the flow rate decreases, due to reactor depressurization and associated drywell pressure decrease, the interface takes on an oscillatory character. The rate

* During a LOCA an air (nitrogen)/steam mixture would be blown down the downcomer vents; however, the analytical models of LOCA phenomena conservatively assume that only noncondensables are injected into the suppression pool.

of change of the displacement of the interface is reflected in submerged structure and suppression pool boundary loads.

When the steam mass flux decreases below a critical level a hydrodynamic phenomenon termed chugging occurs. Chugging is associated with low steam mass flow and high suppression pool boundary pressure spikes relative to CO. The phenomenon appears to be random in time and is caused by the complex interaction of water/steam condensation surface instabilities with the physical properties of the downcomers, the suppression pool, and the suppression pool boundaries. Chugging causes loads on the suppression pool boundary and submerged structures.

To determine the effects of downcomer capping on the hydrodynamic load definition, displacement time histories due to chugging are obtained at selected nodes on the containment vessel. Two sets of time histories are obtained. The first includes the effects of downcomer capping and the second excludes these effects. A comparison of the maximum displacements indicates that capping results in a slight reduction in the maximum displacements. Since lower displacements are associated with lower loads and stresses, it is conservative to assume uncapped downcomers for establishing the chugging load definition. The Containment Functional Design Analysis described in Section 6.2 was based on the venting capacity of 99 downcomers.

3A.3.2.3 Short-Term Loss-of-Coolant Accident Loads

Short-term LOCA loads are associated with hydrodynamic related phenomena that occur within a few seconds after LOCA initiation. The short-term loading phenomena include downcomer vent clearing, LOCA bubble charging, pool swell, and fallback. Figure 3A.3.2-1 illustrates the short-term loading phenomena. The flow fields during downcomer vent clearing and LOCA bubble charging are three-dimensional. Flow fields during pool swell and fallback are vertical and exist only above the vent exit elevation. Loads on submerged structures due to downcomer vent clearing and LOCA bubble charging are compared and the larger loads are employed for subsequent evaluations.

Section 3A.3.2.3.1 and Table 3A.H-1 provide a detailed summary of the short-term LOCA loading phenomenon and load calculation procedures used to assess CGS structures.

3A.3.2.3.1 Analytical Models and Supporting Test Data

3A.3.2.3.1.1 Vent Clearing Jet and Induced Flow Field Model. To calculate the CGS vent clearing jet and induced flow field, a LOCA Water Jet Analytical Model was developed for CGS. The model development and supporting test data are documented in Reference 3A.3.2-9. The calculation is performed for a unit cell with a downcomer at the center. Input data included the downcomer vent water clearing time history.

In order to calculate the downcomer vent water clearing time history and to provide a continuous pool surface displacement time history, a VENT computer code was developed. The model development and supporting test data for VENT are discussed in detail in Attachment 3A.D.2.

CGS input data for the LOCA water jet model and the VENT computer code are shown in Tables 3A.3.2-2 and 3A.3.2-3, respectively. As appropriate, maximums or minimums of CGS parameters are used to make the input data conservative in order to maximize vent water clearing velocities.

The vent exit water velocity and acceleration calculated by VENT are increased by 10% as requested by the NRC in Reference 3A.3.2-1. The velocity and acceleration time histories (including the 10% increase) are shown in Figures 3A.3.2-2 and 3A.3.2-3, respectively. As indicated in Reference 3A.3.2-9, tests have shown that LOCA jet continues to propagate downward for a short duration beyond the vent clearing instant due to rapidly charging air. Although the vent clearing time for CGS is 0.654 sec, the jet tip reaches a maximum velocity of about 15.8 ft/sec at t = 0.704 sec.

Submerged boundary loads during downcomer vent water clearing is specified to be a static addition of an overpressure of 24 psi to the local hydrostatic pressure below the downcomer vent exit (walls and basemat) with a linear attenuation to zero at the pool surface.

3A.3.2.3.1.2 <u>Loss-of-Coolant Accident Bubble Charging Model.</u> In order to calculate the flow field associated with the LOCA bubble charging phenomenon, a LOCA bubble charging model was developed. The model development and supporting test data are discussed in detail in Attachment 3A.D.4.

As discussed in Attachment 3A.D.4, the LOCA bubble charging flow field is calculated using a numerical technique for potential flows in the exact CGS suppression pool geometry. The uniformly charging LOCA bubbles are modeled as equal strength point sources located one downcomer radius below the CGS vent exit at el. 453 ft-4.75 in. Figure 3A.B-2 shows the modeled geometry for the CGS LOCA bubble charging phenomenon and Figures 3A.3.2-4 through 3A.3.2-6 show contour plots of the maximum radial, tangential, and vertical components of the gradient of the velocity potential.

The growth rate of the CGS LOCA bubble and the corresponding source strengths of the point sources are determined by continuity from the rate of displacement of the pool surface after vent clearing. The rate of displacement of the pool surface is determined from the pool swell analytical model (PSAM) (see 3A.3.2.3.1.3). Figure 3A.3.2-7 shows the velocity source strength, Q(t), the acceleration source strength, Q(t), and the radius, R(t), of the CGS LOCA bubbles charging process. Section 3A.B.4.2 discusses how to determine the CGS LOCA bubble charging transient flow field using the data in Figures 3A.3.2-4 through

3A.3.2-7. Tables 3A.3.2-4 and 3A.D-5 summarize results from the LOCA bubble charging analysis for CGS.

Since LOCA bubble charging represents the early portion of pool swell, the associated suppression pool boundary loads are discussed in Section 3A.3.2.3.1.3.

3A.3.2.3.1.3 Pool Swell Analytical Model. The pool swell transient is conservatively defined for CGS by the computer code SWELL based on the model presented in Reference 3A.3.2-10. The model development and supporting test data are discussed in Section 3A.C.3.

The CGS input data for SWELL are given in Tables 3A.3.2-3 and 3A.3.2-5. As appropriate, maximums and minimums of CGS parameters are used to make the input data conservative in order to maximize pool swell velocities. The CGS pool swell velocity, acceleration, displacement, bubble pressure, and wetwell air pressure versus time are given in Figures 3A.3.2-8 through 3A.3.2-12, respectively. Figure 3A.3.2-13 shows the pool lug velocity vs. displacement and Table 3A.3.2-6 shows results from the pool swell analysis for CGS. Velocities and accelerations calculated by the computer code SWELL are multiplied by a factor of 1.1, as requested by the NRC in Reference 3A.3.2-1. The 10% increase in velocities and accelerations is included in Figures 3A.3.2-8, 3A.3.2-9, and 3A.3.2-13.

Test results indicate that no significant froth will occur when the water slug breaks up during pool swell in Mark II containments. Hence, the load due to froth is negligible (Reference 3A.3.2-11) and no froth impingement is considered in the assessment of the design of structures.

Loads on submerged boundaries during pool swell are calculated by specifying that basemat and wall loads be determined from static addition of the maximum bubble pressure predicted by the PSAM to the local hydrostatic pressure below the downcomer vent exit plane and with a linear attenuation to the maximum wetwell air space pressure at maximum pool swell elevation.

Wetwell wall loads due to air compression during pool swell are specified by the direct application of the PSAM calculated wetwell air compression pressure to the wetwell walls above the pool surface.

The short-term drywell pressure history during pool swell is specified as the drywell pressure transient presented in Table 3A.3.2-5.

Test data have shown that a small short duration upward pressure differential on the diaphragm floor may occur due to rapid pressurization of the wetwell air space during the pool swell transient. The 4T Test results, discussed in Section 4.4.6.6 of Reference 3A.3.2-11, show a net upward load on the diaphragm floor of less than 2.2 psi. These low value 4T test results

were confirmed by small scale pool swell tests which showed an average scaled up net upward pressure differential of 1.83 psi. However, as discussed in Section 4.4.6.6 of Reference 3A.3.2-11, the Bodega Bay test data, for a wide variety of blowdown conditions, shows that the wetwell air space pressure never exceeds the drywell pressure.

Conservatively, therefore, the CGS diaphragm floor is assessed for an uplift pressure, P, of 5.5 psi. This is a maximum dynamic load and is in agreement with the loads specified in References 3A.3.2-2 and 3A.3.2-8.

3A.3.2.3.1.4 Fallback Model. The model presented in Reference 3A.3.2-11 is used to calculate the CGS fallback phenomenon. The model development and supporting test data are discussed in Reference 3A.3.2-11. The fallback model conservatively assumes the pool swell slug remains intact during pool swell and falls back from its maximum height 1.5 x H_o, where H_o is the pre-LOCA downcomer submergence, at full water density and under the constant acceleration due to gravity. Figure 3A.3.2-14 shows a plot of the fallback water slug velocity vs. elevation of water slug top surface.

Reference 3A.3.2-10 indicates that fallback loads on submerged boundaries are negligible and are therefore not specified. This is based on review of existing fallback data.

3A.3.2.3.2 Boundary Loads

The analytical models used to describe containment boundary loads during short-term LOCA loading phenomena are discussed in Section 3A.3.2.3.1. Figures 3A.3.2-15 through 3A.3.2-17 show the duration and distribution of the short-term containment boundary loads.

3A.3.2.3.3 Structure Loads

The CGS structures affected by short-term LOCA loading phenomena are identified in Section 3A.2.1.1 and shown in *Figures 3A.2.1-2 through 3A.2.1-8*. Short-term LOCA loads on submerged structures are given in Tables 3A.3.2-8, 3A.3.2-9, and Figure 3A.3.2-18.

Piping and structural components below el. 454.4 ft are subjected to loads caused by water clearing/air charging. Piping and structural components between elevations 454.4 ft and 434.4 ft are subjected to drag and lift loads caused by pool swell/fallback. Also, piping and structural components between elevations 466.4 ft and 484.4 ft are subjected to impact loads caused by pool swell. Direct hydrodynamic loads do not exist for piping and structures above el. 484.4 ft.

The following is a description of how the short-term LOCA definition models, discussed in Section 3A.3.2.3.1, are applied to CGS structures.

a. Vent clearing jet load

The loads on submerged structures located inside the jet boundaries are calculated using the vent exit velocity and acceleration at t = 0.654 sec. The loads on submerged structures located outside the jet boundaries are calculated using velocity and acceleration fields at t = 0.704 sec.

For elbows of 24-in. diameter pipes, the impact load for full momentum transfer of the intercepted jet (K=2) (Reference 3A.3.2-10) is calculated to be smaller than the drag load. Hence, drag load is used in the assessment.

Submerged structures located below the vent exit elevation are subjected to drag loads caused by the vent clearing jet. The types of drag loads and the formulas for calculating them are presented in item c below and in Attachment 3A.C. For the case of the vent clearing jet, the standard drag coefficient, C_D, is equal to 1.2; the acceleration drag coefficient, C_M, is equal to 2.0; and the lift coefficient, C_L, is equal to 0. In order to account for the dynamic nature of the loading phenomenon, a multiplier of 2.0 is used to calculate the equivalent static pressure.

b. Loss-of-coolant accident bubble load

Submerged structures located below the vent exit elevation are subjected to drag loads caused by LOCA bubble charging. The types of drag loads and the formulas for calculating them are presented in Item c below and in Attachment 3A.C. The velocity and acceleration at any point in the pool are calculated by applying the methods outlined in 3A.3.2.3.1.2.

For the case of LOCA bubble charging, the standard drag coefficient, C_D, is equal to 1.2; the acceleration drag coefficient, C_M, is equal to 2.0; and the lift coefficient, C_L, is equal to 0. These numerical values are consistent with References 3A.3.2-4 and 3A.3.2-6. In order to account for the dynamic nature of the LOCA bubble charging load phenomenon, a DLF of 2.0 is used to calculate the equivalent static pressure.

c. Pool swell load

The structures above the initial pool surface and below the maximum pool swell height are subject to impact loads due to the rising pool surface. The maximum dynamic pressure due to pool swell impact is calculated by using the methodology outlined in Reference 3A.3.2-2 and in Section III.B.3.c.1 of Reference 3A.3.2-1. In order to account for the dynamic nature of the pool swell loading phenomenon, the equivalent static pressure is obtained by

multiplying the maximum dynamic impact pressure by a DLF. The DLF is given in Figure 5 of Reference 3A.3.2-3. No impact loads on grating surfaces are specified since the CGS grating bars are narrow (typically 3/16-in. wide).

Structures located above the vent exit elevation and below the maximum pool swell height are subjected to drag forces during the pool swell transient. Drag loads have three components: standard drag, acceleration drag, and lift. Formulas for calculating these three types of load are presented in Attachment 3A.C. Standard drag load is velocity square dependent and acts in the flow direction. Acceleration drag load is proportional to the flow acceleration and acts in the flow direction. Lift load is velocity square dependent and is normal to the flow direction.

In order to calculate the three components of drag load, three coefficients are determined. These are: standard drag coefficient, C_D; acceleration drag coefficient, C_M; and lift coefficient, C_L. For circular cross-sections, coefficients C_D, C_M, and C_L are calculated using Reference 3A.3.2-6. For non-circular cross-sections, coefficients C_D and C_M are calculated using Reference 3A.3.2-7, and C_L is assumed equal to 1.6 as indicated in Reference 3A.3.2-5. Interference effects from adjacent structures are taken into account by using the methodology presented in Reference 3A.3.2-4. In order to account for the dynamic nature of the pool swell loading phenomenon, appropriate DLFs are used to calculate the equivalent static loads.

The gratings are subject to drag loads during pool swell due to resistance to the flow through them. The dynamic drag load is determined by the product of the differential pressure across the grating and the total plan area of the grating. The open area fraction for the grating is greater than 60%. The peak dynamic differential pressure across the grating is obtained from Figure 4-1 of Reference 3A.3.2-2. This figure is based on a pool swell approach velocity of 40 fps. Since the maximum pool swell velocity for CGS is smaller than 40 fps, the differential pressure values in Figure 4-1 of Reference 3A.3.2-2 are multiplied by a factor equal to the square of the ratio given by the maximum velocity divided by 40 fps. The maximum pool swell velocity is given in Table 3A.3.2-6 (plus a velocity multiplier of 1.1). In order to account for the dynamic nature of the pool swell loading phenomenon, a multiplier of 2.0 is used to calculate the equivalent static pressure.

Vertical drag pressure and horizontal lift pressure are applied simultaneously to the projected area of a cylinder circumscribing the structural member under consideration. The projection is always made on a plane normal to the direction of pressure. Vertical impact load is applied separately (not in combination with drag and lift loads) to structures located between elevations 466.4 ft and

484.4 ft on the projected area of the structure on the horizontal plane. Impact load is not used to check local bending effects of flanges or webs or ovaling effects of a pipe. The full drag, lift, or impact load calculated as outlined above is applied normal to a pipe or a structural component. There are no loads parallel to the longitudinal axis of the structure.

d. Fallback load

Structures located above the vent exit elevation and below the maximum pool swell height are subjected to drag and lift forces during fallback. The same methodology and the same drag and lift coefficients used in calculating pool swell drag and lift loads reused to define fallback drag and lift loads. To account for the dynamic nature of the fallback loading phenomenon, a multiplier of 2.0 is used to calculate the equivalent static load. No impact loads are specified during fallback (Reference 3A.3.2-11).

Fallback drag loads on gratings are specified to be equal in magnitude to the pool swell drag loads but opposite in direction.

The method of applying drag and lift pressure on a structural member is the same as that outlined at the end of Item c above.

3A.3.2.3.3.1 Loads on Major Structures. The major vertical structures are shown in *Figure 3A.2.1-2*. They are: the 102 downcomers (three were capped), the 18 SRV lines including quenchers and support towers, and the 17 concrete columns supporting the diaphragm floor. The major horizontal structures are the steel truss shown in *Figure 3A.2.1-3*, which provides lateral support to the downcomers and the SRV lines, and the platform at el. 472 ft 4 in. shown in *Figures 3A.2.1-3 and 3A.2.1-8*. The truss is submerged and the platform is located in the pool swell zone.

Hydrodynamic loads on the vertically oriented columns may occur only due to horizontal flow across the columns. Since the columns are vertical, pool swell and fallback loads are negligible. The columns are located along circumferential and radial lines of symmetry between downcomers; however, some asymmetry is created because some of the downcomers have a diameter of 24 in. while others have a diameter of 28 in. This small asymmetry in the downcomer arrangement causes small loads on the columns during water clearing/air charging. The magnitude and distribution of loads on the columns are shown in Table 3A.3.2-8 and Figure 3A.3.2-18 respectively.

As with the columns, hydrodynamic loads on the vertically oriented downcomers may occur only due to horizontal flow across the downcomers since the downcomers are located along circumferential and radial lines of symmetry between other downcomers and the suppression pool boundaries, the LOCA water jet, LOCA bubble charging, pool swell, and fallback

horizontal flows are small. Because the horizontal flows are small the resulting loadings are negligible.

Table 3A.3.2-8 and Figure 3A.3.2-18 show the magnitude and distribution of loads on a vertical SRV line as well as quencher arms during the water clearing/air charging phases of LOCA. Since quencher supports are below el. 448 ft-0 in., they are not subjected to short-term LOCA (Figure 3A.3.2-18). Pool swell and fallback exert negligible loads on the SRV lines due to the vertical orientation of the SRV lines above the downcomer vent exit plane. Because of their location below the downcomer vent exit plane, no pool swell or fallback loads are experienced by the quencher supports and quencher arms.

The downcomer bracing truss is oriented in a horizontal plane one foot above the downcomer vent exit plane. Since the truss is located above the downcomer vent exit the induced flow field and, therefore, drag load due to LOCA water jet during vent clearing is small. Horizontal LOCA bubble load on the bracing truss is small. Because the truss is above the downcomer vent exit plane, pool swell and fallback loads are experienced.

The only major horizontal structure in the pool swell zone is a platform at el. 472 ft-4 in. The platform consists of a grating with an open area fraction of 0.776 and is supported by 0.5-in. thick vertical members. The dynamic drag load is calculated to be 4.2 psi for pool swell and fallback induced loads and is multiplied by a DLF of 2.0. The equivalent static load is then applied to the total plan area of the platform. Because the platform is above the initial pool surface it does not experience vent clearing or LOCA bubble loads.

3A.3.2.3.3.2 Loads on Fully Submerged Piping Systems Below Elevation 454.4 ft. The 24-in. high-pressure core spray (HPCS) (X-31), 24-in. residual heat removal (RHR)-B (X-32), 8-in. reactor core isolation cooling (RCIC) (X-33), 24-in. low-pressure core spray (LPCS) (X-34), 24-in. RHR-A (X-35), 24-in. RHR-C (X-36), 6-in. suppression pool cleanup (X-100) and the 4-in. fuel pool cooling and cleanup (FPC) are the eight piping systems fully submerged in the CGS suppression pool. Seven systems enter the pool through containment penetrations at el. 452 ft 0 in. as shown in *Figure 3A.2.1-2.* The 4-in. FPC enters the pool through the pedestal at el. 451 ft 6 in. and is also shown in *Figure 3A.2.1-2.* Each of these pipes runs along the CGS suppression pool boundaries with a maximum distance between pipe centerline and boundaries of 46 in. Since these pipes are below the downcomer vent exit el. 454 ft 4.75 in. they are not subjected to pool swell or fallback loads during a LOCA. The magnitude and distribution of short-term LOCA loads on submerged pipes are given in Table 3A.3.2-8 and Figure 3A.3.2-18 respectively.

3A.3.2.3.3.3 Loads On Partially Submerged Piping Systems. The 13 partially submerged piping systems enter the suppression chamber through containment penetrations at el. 467 ft 9 in. as shown in *Figure 3A.2.1-3* and enter the pool vertically within a 39 in. distance from the containment as shown in *Figures 3A.2.1-6 through 3A.2.1-8.*

The pipes' horizontal portions are subjected to vertical drag and horizontal lift loads caused by pool swell/fallback. Also, the pipes' horizontal portions experience vertical impact pressure due to pool swell. Inclined braces above the penetrations are not subjected to pool swell impact since they are shielded by the pipes. Horizontal supports below penetrations do not experience pool swell impact since these supports are below el. 466.4 ft.

3A.3.2.3.3.4 Loads on Piping Systems and Structural Components Between Elevations 454.4 ft and 484.4 ft. The portion of the wetwell between elevations 454.4 ft and 484.4 ft is affected by pool swell and fallback only. Piping systems located in this zone are subjected to drag, and lift loads. In addition piping between the normal pool surface, el. 466.4 ft, and el. 484.4 ft is subjected to impact loads unless shielded by pipe supports below. These piping systems include penetration sleeves, pipe stubs, pipes, and pipe supports.

3A.3.2.4 Long-Term Hydrodynamic Loads

Long-term hydrodynamic loads refer to the LOCA related loads exerted on the pool boundaries and on submerged wetwell structures following the fallback transient. These loads are associated with steam condensation at the downcomer vent exits as steam from the drywell flows into the suppression chamber via the vents. Depending on the steam mass flux through the vents, two types of loading conditions are anticipated. During high or medium mass flux so-called COs occur. During low mass flux, chugging loads occur. Chugging loads and CO loads are discussed in the following sections.

3A.3.2.4.1 Analytical Models and Supporting Test Data

3A.3.2.4.1.1 Chugging Loads. The design specification for chugging loads herein is based on the major test programs on the effects of CO and chugging, which were conducted for the Mark II Containment Program during the period of 1975 to 1981 (References 3A.3.2-16 and 3A.3.2-17). It takes account of the generic chugging load definition developed for the Mark II type plants (References 3A.3.2-18 and 3A.3.2-19), but departs from the generic definition in order to provide for important structural differences between the CGS plant and the generic plant. The design specification (Reference 3A.3.2-15) is called the Revised Definition as it represents a revision of an earlier definition called the Improved Definition (Reference 3A.3.2-20) which was based only on the first stage of the test results, the 4T Program (Reference 3A.3.2-16). The current design specification, i.e., the Revised Definition, was submitted to and approved by the NRC (Reference 3A.3.1-6). The scope of the design specification covers both single vent loads and multiple vent loads applicable to the CGS plant.

The single vent load definition is based on two series of full scale single vent tests, namely, the 4T tests and the 4TCO tests (References 3A.3.2-16 and 3A.3.2-17). The load is defined in terms of a series of source loads located at the vent exit with significant load features selected on the basis of the pressure readings at the test tank and the characteristics disclosed by the

vent-pool-tank system. Both test programs indicate the impulsive and random strength nature of chugging, and it is noted that the strength, although random, is related to system conditions. The greater chugs of the 4TCO program are controlling relative to pressure amplitude, frequency content, and spatial distribution of pressure. Thus, the Revised Definition is a group of seven source loads selected on the basis of the seven greatest chugs from the 4TCO data. A design load envelope is obtained by applying the source loads in turn at the vent exit; each load consists of an impulsive pressure gradient with appropriate system parameter values. To calculate results at the wetwell tank, the theory of acoustic fluids is used with a fully coupled model consisting of the vent steam, the pool water, and the wetwell tank. The calculated results, due to application of the defining source loads, envelope both the 4TCO data and the 4T data, relative to all pertinent characteristics.

The multiple vent load definition utilizes the preceding single vent definition in conjunction with the 102 vents in the CGS plant. It is described in Section 3A.3.2.4.2.1.

3A.3.2.4.1.2 Condensation Oscillation Loads. The generic definition of CO loads, as developed for the Mark II Containment Program, is given in Reference 3A.3.2-21. The definition is based on direct application of the bounding pressure measurements from full-scale single vent 4TCO tests to the Mark II containment boundaries.

A comparison has been made between this load definition for CO loads, the JAERI CO results, and the preceding definition for chugging loads. The comparison, as reported in Reference 3A.3.2-12 shows that the CO loads and the chugging loads are similar with respect to pertinent characteristics such as randomly varying amplitude, frequency content, and desynchronization of vent loads. However, it is also shown (References 3A.3.2-12 and 3A.3.2-23), that in a Mark II multivent containment, the controlling boundary pressures due to the CO load are less than those due to the chugging load. It is therefore concluded that the CO load does not represent a governing load and that it need not be considered in the assessment of the design adequacy of the CGS structures.

3A.3.2.4.2 Boundary Loads

3A.3.2.4.2.1 Chugging Loads. The current design specification, i.e., the Revised Definition (Reference 3A.3.2-15), includes definition of the chugging boundary loads due to multivent chugging together with the associated application methodology. The chugging pressures on the suppression pool boundary are defined as resulting from the application of the single vent design chugging loads at all 102 vent exits in the CGS pool-containment structure. To determine the boundary pressures, the analysis is based on a fully coupled model which accounts for all important plant parameters: vent length, three-dimensional multivent pool geometry, pool with sloped bottom, and flexibility of pool structural boundaries. Detailed analytical methods are described in Reference 3A.3.2-15.

The methodology for application of the source loads to the 102 vents is generally similar to that of the generic load definition (Reference 3A.3.2-19), and reflects the characteristics of multivent behavior disclosed by the JAERI and CREARE test programs. Two deterministic spatial distributions of chug strengths, similar to but not the same as the generic distributions, are specified to maximize axisymmetric and nonaxisymmetric responses. Random variation of chug initiation tines from vent to vent is recognized with desynchronization of the start times as in the generic definition. The seven basic single vent loads of the design specification are applied in turn at the 102 vents with variation of strength and initiation time between vents as previously noted; an envelope of containment pressures is calculated. The calculated results exceed the test results in the 4TC0 and the JAERI tests with respect to maximum containment pressures and the Fourier amplitude spectra for containment pressures.

3A.3.2.4.2.2 Condensation Oscillation Loads. The discussion in Section 3A.3.2.4.1.2 on the relative magnitudes of chugging loads and CO loads is applicable. As noted therein, the controlling boundary pressures due to chugging exceed those due to CO. Hence, the CO load is not a governing load and it is not considered in the assessment of the design adequacy of the CGS structure.

3A.3.2.4.3 Submerged Structure Loads

Loads on submerged structures caused by CO and chugging are presented in the following sections.

3A.3.2.4.3.1 Condensation Oscillation Loads. There is no need to assess the CGS plant for a CO load definition since CO loading is less critical and is bounded by the chugging load.

3A.3.2.4.3.2 Chugging Loads. The LOCA chugging loads on submerged structures are defined consistently with the load definition for the pool boundary. Pressure field in the fluid is obtained using chugging design sources developed for pool boundary loads (Reference 3A.3.2-15). From the pressure field, pressure gradients and loads on submerged structures are calculated.

In the method described above, it is assumed that the flow in the vicinity of the vent during chugging is unaffected by the presence of pool boundaries or other sources in the pool. Therefore, fluid pressures in a single cell/single vent pool are representative of pressures near the vent in the CGS pool.

For structures located beyond a distance of 4R (R being the downcomer radius) from the vent exit center, the chugging loads are negligible and need not be considered in the design assessment of the structures. Equivalent static chugging loads for structures located within a distance of 4R from the vent exit center are calculated by using the formula:

$$p = C_m \frac{\pi}{4} D \times DLF \times \frac{\partial p}{\partial n}$$

where

p = equivalent static pressure on structure (psi)

C_m = hydrodynamic mass coefficient = 2.0

DLF = dynamic load factor = 1.5

D = diameter in inches of the pipe or of the cylinder circumscribing the cross-section of a support member

$\frac{\partial p}{\partial n}$ = pressure gradient across a submerged structure (psi/in). Numerical values for this term are given in Figure 3A.3.2-19.

Chugging loads calculated as outlined above are listed as follows:

a. Bracing Truss Members: A load of 10 psi is applied vertically upward or downward. This load is applied simultaneously to all members connecting to a downcomer. Chugging event under each vent occurs with random phasing from events at other vents. Therefore, this load is not applied simultaneously to all members of the truss;

b. Inner Row SRV Line: A load of 12 psi is applied radially outward or inward from the vent axis of the adjacent inner row downcomer. The load distribution is assumed uniform between el. 454.4 ft and el. 452.4 ft and then linearly decreasing to zero at Point A (Figure 3A.3.2-19); and

c. Pipes and Supports: A pressure is applied vertically upward or downward. The magnitude and distribution of the vertical pressure are calculated using Equation 3A.3.2-1 and Figure 3A.3.2-19. The maximum vertical pressure is 2.4D psi. Simultaneously, a radial pressure is applied inward or outward from the vent axis of the adjacent outer row downcomer. The magnitude and distribution of the radial pressure are calculated using Equation 3A.3.2-1 and Figure 3A.3.2-19. The radial load is assumed zero above the vent exit.

The pressure loads listed above are multiplied by the projected area of the structure segment normal to the direction of the loading to obtain the total force on each segment. The total load on each segment of the structure is applied at the geometric center of the segment

3A.3.2.4.4 Lateral Loads on Downcomer Vents

This section provides the definition of the lateral loads which occur near the downcomer exits during chugging. The definition conforms with the requirements for such loads as prescribed in NUREG-0808 (Reference 3A.3.2-8). These lateral loads are defined herein in relation to the downcomer bracing system which is described and assessed in Section 3A.4.2.1. The principal features included in the definition are single vent loads, loads on multiple vents, overall loading of the bracing system, and loads and downcomer size.

a. Single Vent Load - The maximum lateral exit load on one 24-in. downcomer is defined as a dynamic single pulse load having a half sine wave shape with load amplitude of 65 klbf (kips) and duration of 3 msec. For the assessment of the bracing system, a dynamic analysis of the system acted on by this single vent load is made.

b. Loads on Multiple Vents - With multiple vent loading, the force on each loaded vent is also defined as a single pulse half sine wave dynamic load. However, in line with NUREG-0808, the pulse duration is taken to vary over a range of values and the force amplitude is taken to depend on the pulse duration, T, and on the number of loaded vents. The vent force, F(t), is evaluated by the expression:

$$F(t) = M\,A(T)\sin(t/T) \quad 0 \le t \le T \qquad \text{(Eq. 3A.3.2-2)}$$

where

$$A(T) = (50 - 20\,T/3)\ \text{klbf} \quad 3 \le T < 6\ \text{ms} \qquad \text{(Eq. 3A.3.2-3)}$$

The factor M is a load reduction factor which depends on the number of loaded vents and the required level of exceedence probability. For a given number of loaded downcomers, it is evaluated at the exceedence probability level of 10^{-4} using the diagram specified in NUREG-0808.

To determine the controlling value of the amplitude factor A, the bracing system was analyzed over the range of the duration, T. Thus, values of T in the range of 3 to 6 ms and the associated values of A were utilized. It was determined that the controlling value of T was always 3 ms. Consequently, the associated value of A in the above equation for F(t) is evaluated as 30 kips.

For the assessment under multiple vent loading, the bracing system is analyzed dynamically with lateral loads on a given set of vents. In the analysis, the force on each loaded vent is defined by Equation 3A.3.2-2 where M is obtained as previously described, A equals 30 klbf, and T equals 3 ms.

c. Overall Loading of the Bracing System - The design assessment of the downcomer bracing system under multiple lateral exit loads in 3A.4.2.1 describes the method used to determine the controlling number of loaded vents and the controlling direction of the loads. The methodology is generally similar to that of Section 2.3.2.2 of NUREG-0808.

d. Loads and Downcomer Size - The downcomer system consists of 102 downcomers of which 84 are 24-in. diameter and 18 are 28-in. diameter. The lateral loads on the 24-in. diameter vents have been defined above in paragraphs a and b for the cases of single vent loading and multiple vent loading. In line with NUREG-0808, the lateral loads on the 28-in. vents are defined as 1.34 times as great as those on the 24-in. vents.

3A.3.2.5 Pressure and Temperature Transients

A LOCA causes a pressure and temperature transient in the drywell and wetwell due to mass and energy released from the line break. The drywell and wetwell pressure and temperature histories are employed to establish the structural loading conditions in the containment. The response must be determined for a range of parameters such as break size, reactor pressure, and initial (preincident) containment conditions. The analytical models used to evaluate the pressure and temperature response of the containment have been developed by GE (References 3A.3.2-13 and 3A.3.2-14).

The assumptions made for analyzing the LOCA transients were based on conservatively predicting the blowdown mass and energy rates into the drywell and suppression pool. The following assumptions were made:

a. Initial drywell pressure of 0.75 psig,

b. Downcomer submergence at high water level,

c. Minimum drywell free air volume,

d. Minimum wetwell free air volume,

e. Minimum water volume in wetwell, and

f. Suppress ion pool initial temperature and service water temperature at the maximum Technical Specifications limit.

(Assumptions b, d, and e are inconsistent with each other but provide conservative results.)

For the intermediate size breaks, it was assumed that all the mass and energy releases from the broken pipe discharge via the drywell into the suppression pool. Normally, a portion of the mass and energy release will be dispersed over the drywell volume causing the drywell pressure and temperature to rise.

3A.3.2.5.1 Results for CGS

The drywell pressure transients have been calculated with inventory effects included in the analysis. The results for the recirculation and main steam line breaks are presented in Figures 3A.3.2-20 through 3A.3.2-28.

The plant parameters used are given in Table 3A.3.2-7. The spectrum of accident conditions covered are:

a. Large double-ended break of a recirculation line,
b. Large double-ended break of one main steamline, and
c. Intermediate recirculation line break (0.1 ft^2 break area).

Table 3A.3.2-5 provides the drywell pressure transient during the 2 sec period immediately following a recirculation line break. This table differs from the values plotted in Figure 3A.3.2-20, since it includes the influence of reactor subcooling on the initial blowdown flow rate from the break. This is a short-term effect that occurs during the first few seconds of the accident and does not influence the maximum drywell pressure. Since this LOCA leads to the most severe short-term pressure conditions, the data in Table 3A.3.2-5 are used to calculate the pool swell velocities and associated effects.

Figures 3A.3.2-20 through 3A.3.2-24 give the pressure/temperature transients resulting from a large recirculation line break. As shown in Figure 3A.3.2-20, the drywell pressure increases to a maximum value of 35 psig in about 20 sec. Also, the pressure of the wetwell air, while increasing with time, is less than the drywell pressure until the ECCS flow starts to spill from the break. The drywell temperature increases to about 280°F (Figure 3A.3.2-23), while the temperature of the suppression pool increases to about 220°F. Similar response curves for a main steam line break are also shown in Figures 3A.3.2-25 and 3A.3.2-26, and those for an intermediate break accident (IBA) are shown in Figures 3A.3.2-27 and 3A.3.2-28.

3A.3.2.5.2 Differential Pressure Load on the Diaphragm Floor

As illustrated in Figures 3A.3.2-20, 3A.3.2-21, 3A.3.2-25, and 3A.3.2-28, the net pressure on the diaphragm floor is downward throughout a LOCA transient (refer to FSAR Section 6.2.1). However, the diaphragm floor is conservatively designed for a net uplift pressure of 5.5 psi as discussed in 3A.3.2.3.1.3. The maximum net downward pressure on the diaphragm occurs during the short-term part of the large break LOCA transients and reaches a maximum value of 20 psi.

During the initial phase of a LOCA transient, drywell air is blown down through the downcomer vents into the wetwell by the steam from the break. The steam rapidly replaces the air in the drywell.

Initially, a steam-air fixture flows through the vents into the suppression pool and forms bubbles that rise to the pool's surface. Condensation of the steam in the bubbles allows only the air component to reach the wetwell airspace. As the air collects, the air space becomes pressurized and heated.

As the reactor vessel blowdown ends, the emergency core cooling system floods the core with water which starts to spill out of the break into the drywell. This results in rapid condensation of the steam, which has replaced the air, and consequent rapid drywell depressurization below that in the wetwell. Before the net upward pressure becomes large, however, nine 24-in. vacuum breakers, which are attached to nine selected downcomers, open to equalize the pressure difference by returning the air (nitrogen) collected in the wetwell to the drywell. These vacuum breakers are adjusted to open when the differential pressure across them is in the range of 0.15 to 0.35 psi.

3A.3.2.6 Building Response to Loss-of-Coolant Accident Loads

The analysis and response of the reactor building under the action of long-term LOCA loads is discussed in Section 3A.5.2.

3A.3.2.7 References

3A.3.2-1 "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," NUREG-0487, NRC, October 1978.

3A.3.2-2 Mark II Containment Dynamic Forcing Functions Information Report (DFFR), General Electric Company, NEDO-21061, Revision 4, November 1981.

3A.3.2-3 "Impact Loads on Structures Above Mark II Containment Pools," G. Maise, Brookhaven National Laboratory, February 28, 1978.

3A.3.2-4 "Submerged Structure Methodology," Appendix G, Zimmer Power Station, Unit 1, DAR Amendment No. 13, October 1980.

3A.3.2-5 "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," NUREG-0487, Supplement No.1, NRC, September 1980.

3A.3.2-6 "In-Line and Transverse Forces on Cylinders in Oscillatory Flow at High Reynolds Number," Journal of Ship Research, Volume 21, No. 4, pp. 200-216, December 1977.

3A.3.2-7 "Forces on Cylinders and Plates in an Oscillating Fluid," G. H. Keulegan and L. H. Carpenter, Journal of Research of the National Bureau of Standards, Volume 60, No. 5, May 1958.

3A.3.2-8 "Mark II Containment Program Load Evaluation and Acceptance Criteria," NUREG-0808, USNRC, August 1981.

3A.3.2-9 "An Analytical Model for LOCA Water Jet in Mark II Containments (The Ring Vortex Model)," Burns and Roe, Inc., September 1980.

3A.3.2-10 Mark II Containment Dynamic Forcing Functions Information Report (DFFR), General Electric Company, NEDO-21061, Revision 3, June 1978.

3A.3.2-11 Mark II Containment Dynamic Forcing Functions Information Report (DFFR), General Electric Company, NEDO-21061, Revision 2, September 1976.

3A.3.2-12 "Comparison of Condensation Oscillation and Chugging Loads for Assessment of WPPSS Nuclear Project No. 2," Summary Report, Proprietary, Burns and Roe, Inc., December 1981 transmitted to NRC by letter GO2-81-552 dated December 24, 1981.

3A.3.2-13 The General Electric Pressure Suppression Containment Analytical Model, General Electric Company, NEDO-10320, April 1971.

3A.3.2-14 Safe System Analysis for Standby Core Cooling Equipment, General Electric Company, NEDE-10169, Proprietary, September 1977.

3A.3.2-15 "Chugging Loads - Revised Definition and Application Methodology for Mark II Containments (Based on 4TCO Test Results)," Technical Report, Burns and Roe, Inc., July 1981. Transmitted to NRC by letter GO2-81-189 of July 22, 1981.

3A.3.2-16 "Mark II Pressure Suppression Test Program - Phase I, II, and III of the 4T Tests - Application Memo," NEDE-23678-P, Rev. 0, General Electric Company, January 1977.

3A.3.2-17 "4T Condensation Oscillation Test Program Final Test Report," NEDE-24811-P, General Electric Company, May 1980.

3A.3.2-18 "Mark II Improved Chugging Methodology," NEDE-24822-P, General Electric Company, May 1980.

3A.3.2-19 "Generic Chugging Load Definition Report," NEDE-24302-P, General Electric Company, April 1981.

3A.3.2-20 "Chugging Loads - Improved Definition and Application Methodology to Mark II Containments," Technical Report, Proprietary, Burns and Roe, Inc., June 1979.

3A.3.2-21 "Generic Condensation Oscillation Load Definition Report," NEDO-24288, General Electric Company, February 1981.

3A.3.2-22 "Mass Energy Report," General Electric Report, GEWP-2-77-533, March 15, 1977.

3A.3.2-23 Letter report on C.O. loads GO2-82-351 of April 1, 1982.

3A.3.2-24 "WPPSS Nuclear Project No. 2, Final Safety Analysis Report," Washington Public Power Supply System, Chapter 6.2.

3A.3.2-25 AEC-TR-6630, "Handbook of Hydraulic Resistance - Coefficients of Local Resistance And Friction," I. E. Idel'chik, 1960.

3A.3.2-26 "Flow of Fluids Through Valves, Fittings, and Pipe," Technical Paper No. 410, Crane Company, 1980.

3A.3.2-27 "CONTEMPT-LT--A Computer Program For Predicting Containment Pressure-Temperature Response to A Loss-of-Coolant Accident," Aerojet Nuclear Company, June 1975.

3A.3.2-28 "Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon," NEDE-21544-P, General Electric Company, December 1976.

3A.3.2-29 Response to NRC Question 020.071, transmitted via letter MFN-275-78 to Mr. J. F. Stolz, Chief, Light Water Reactor Branch No. 1, USNRC, from Mr. L. J. Sobon, Manager BWR Containment Licensing, General Electric Company on "Responses to NRC Request for Additional Information (Round 3 Questions)," dated June 30, 1978.

Table 3A.3.2-1

Summary of Loss-of-Coolant Accident Affected Structures

	Type of Loading Condition					
	Short-Term LOCA				Long-Term LOCA	
Structures Experiencing LOCA Loads	Water Jet	LOCA Bubble	Pool Swell	Fallback	Condensation Oscillations	Chugging
Fully submerged piping systems	3A.3.2.3.1.1/ 3A.3.2.3.3.2	3A.3.2.3.1.2/ 3A.3.2.3.3.2			3A.3.2.4.1.2/ 3A.3.2.4.3.1	3A.3.2.4.1.1/ 3A.3.2.4.3.2
Partially submerged piping systems	3A.3.2.3.1.1/ 3A.3.2.3.3.3	3A.3.2.3.1.2/ 3A.3.2.3.3.3	3A.3.2.3.1.3/ 3A.3.2.3.3.3	3A.3.2.3.1.4/ 3A.3.2.3.3.3	3A.3.2.4.1.2/ 3A.3.2.4.3.1	3A.3.2.4.1.1/ 3A.3.2.4.3.2
Piping systems fully above initial pool surface			3A.3.2.3.1.3/ 3A.3.2.3.3.4	3A.3.2.3.1.4/ 3A.3.2.3.3.4		
Grating			3A.3.2.3.1.3/ 3A.3.2.3.3.1	3A.3.2.3.1.4/ 3A.3.2.3.3.1		
Drywell floor			3A.3.2.3.2			
Containment wall	3A.3.2.3.1.1		3A.3.2.3.2	3A.3.2.3.2	3A.3.2.4.2.2	3A.3.2.4.2.1
Pedestal	3A.3.2.3.1.1		3A.3.2.3.2	3A.3.2.3.2	3A.3.2.4.2.2	3A.3.2.4.2.1
Basemat	3A.3.2.3.1.1	3A.3.2.3.2	3A.3.2.3.2	3A.3.2.3.2	3A.3.2.4.2.2	3A.3.2.4.2.1
Columns	3A.3.2.3.1.1/ 3A.3.2.3.3.1	3A.3.2.3.1.2/ 3A.3.2.3.3.1				
Downcomers	3A.3.2.3.1.1/ 3A.3.2.3.3.1	3A.3.2.3.1.2/ 3A.3.2.3.3.1			3A.3.2.4.1.2/ 3A.3.2.4.3.1	3A.3.2.4.1.1/ 3A.3.2.4.3.2
Downcomers bracing system			3A.3.2.3.1.3/ 3A.3.2.3.3.1	3A.3.2.3.1.4/ 3A.3.2.3.3.1	3A.3.2.4.1.2/ 3A.3.2.4.3.1	3A.3.2.4.1.1/ 3A.3.2.4.3.2
SRV system	3A.3.2.3.1.1/ 3A.3.2.3.3.1	3A.3.2.3.1.2/ 3A.3.2.3.3.1			3A.3.2.4.1.2/ 3A.3.2.4.3.1	3A.3.2.4.1.1/ 3A.3.2.4.3.2

Note: Numbers refer to section of Appendix 3A.

Table 3A.3.2-2

CGS Data for Loss-of-Coolant Accident
Water Jet Analysis

Unit cell diameter	40.87 in.
Unit cell depth	354.0 in.
Downcomer inner radius	13.625 in.
Downcomer submergence	144.0 in.
Downcomer water clearing velocity time history	Figure 3A.3.2-2

Table 3A.3.2-3

CGS Data for Vent Clearing And Pool Swell Analysis

Drywell	
1. Temperature (initial)	135°F
2. Drywell pressure transient	Table 3A.3.2-5
3. Relative humidity	0%
Suppression Chamber	
1. Free air volume (maximum)	147,290 ft^3
2. Net suppression pool surface area	4520 ft^2
3. Pressure (initial)	0.7 psig
4. Air specific heat ratio	1.4[a]/1.2[b]
Downcomer vent system	
1. Submergence (minimum/maximum)	11 ft-8 in./12 ft[a]
2. Nominal diameter	2 ft
3. Number of vents	102
4. Vent exit area	321 ft^2
5. Vent loss coefficient	1.9

[a] Value used to determine maximum pool swell elevation.

Table 3A.3.2-4

Results from Loss-of-Coolant Accident
Bubble Charging Analysis for CGS

	Downcomers		
	Inner Radius	Middle Radius	Outer Radius
Bubble coalescence time (sec)[a]	0.09	0.16	0.24
Bubble radius at coalescence (ft)	2.12	2.77	3.41

[a] Times represent time after vents have cleared.

Table 3A.3.2-5

CGS Drywell Pressure as a Function of Time for
Loss-of-Coolant Accident[a]
(Effects of Pipe Inventory and Subcooling Included)

Time After Loss-of-Coolant Accident (sec)	Drywell Pressure (psia)
0.0	15.45
0.00159	15.32
0.00171	15.30
0.00549	14.72
0.0641	17.61
0.127	20.18
0.252	24.83
0.502	33.27
0.720	35.69
0.740	35.42
1.099	35.11
1.537	35.84
1.568	35.92
2.037	36.00

[a] See Reference 3A.3.2-22.

Table 3A.3.2-6

Results of Pool Swell Analysis for CGS

Vent clearing time[a] - t_c	0.65 sec
Pool water surface velocity[b] at time t_c	5.3 ft/sec
Time of maximum pool swell velocity	1.12 sec
Maximum pool swell velocity[b]	28.7 ft/sec
Time of maximum pool height[a]	1.48 sec
Maximum pool swell height (H_{max})	18.0 ft
Ratio of H_{max} to H_o (downcomer submergence)	1.5
Maximum air bubble pressure	35.75 psia
Maximum wetwell airspace pressure	47.55 psia
Maximum pool swell elevation	484 ft 4.75 in. msl
Pool swell termination[a]	1.48 sec

[a] Times represent time after LOCA initiation.
[b] Does not include velocity multiplier of 1.1.

Table 3A.3.2-7

CGS Plant Parameters for
Loss-of-Coolant Accident Transient Analysis

1. Drywell	
a. Free air volume	200,540 ft^3
b. Temperature (initial)	135°F
c. Pressure (initial)	0.75 psig
d Relative humidity	50%
2. Wetwell	
a. Free air volume	144,184 ft^3
b. Water volume[a]	107,850 ft^3
c. Pool temperature (initial)	90°F
d. Pressure (initial)	0.75 psig
e. Relative humidity	100%
3. Break area	
a. Design basis accident (DBA) - recirculation line	3.106 ft^2
b. DBA - steam line	3.92 ft^2
c. Intermediate break	0.1 ft^2
4. Main vent	
a. Maximum submergence	12 ft
b. Nominal diameter	2 ft
c. Number of vents	102[b]
d. Vent entrance flow area	304.6 ft^2
e. Downcomer loss factor	1.9

[a] Water volume
[b] Refer to Section 3A.3.2.2 for a discussion of the effect of capping three downcomers.

Table 3A.3.2-8

Short-Term Loss-of-Coolant Accident Loads
on Structures Below Elevation 454.4 ft

	Radial Location r of Geometric Center of the Structure or Segment of Structure			
	Zone I $0 \leq r < 2.3R$		Zone II $2.3R \leq r < 5.0R$	
Structure	p_r max (psi)	p_v max (psi)	p_r max (psi)	p_v max (psi)
42 in. diameter vertical column			+2	
12 in. diameter vertical SRV line	±20 inner row		+2 outer row	
<u>Pipes and Supports</u>				
Diameter[a] > 12 in. (X-31,32,34,35,36)	±60	+212	+6	-45
Diameter[a] ≤ 12 in. (X-33,100,4 in. FPC, quencher arm)	±60	+100	+6	-25

[a] For noncylindrical structures, the diameter of a cylinder circumscribing the cross section of the structure is used.

<u>Notes</u>:

1. Vertical load is positive in the downward direction. Radial load is positive in the radially outward direction from the axis of symmetry of the downcomer.

2. The vertical distribution associated with the tabulated peak values is shown in Figure 3A.3.2-18. The distribution in the radial direction in each zone is uniform and is aximsymmetric in the circumferential direction.

3. Long structures are divided into smaller segments, L ≤ D; D being the diameter of the structure, and L being the segment length.

Table 3A.3.2-8

Short-Term Loss-of-Coolant Accident Loads
on Structures Below Elevation 454.4 ft (Continued)

4. Radial or vertical load is calculated at the geometric center of each structure or segment (see Figure 3A.3.2-18) by multiplying the pressure value at the geometric center (obtained from this table) with the length and the diameter of the structure.

   For noncylindrical structures, diameter of the circumscribing cylinder is used.

5. Radial or vertical load has, in general, two components. One is parallel and the other is normal to the structure or segment. All components of load that are parallel to the structure or segment are neglected.

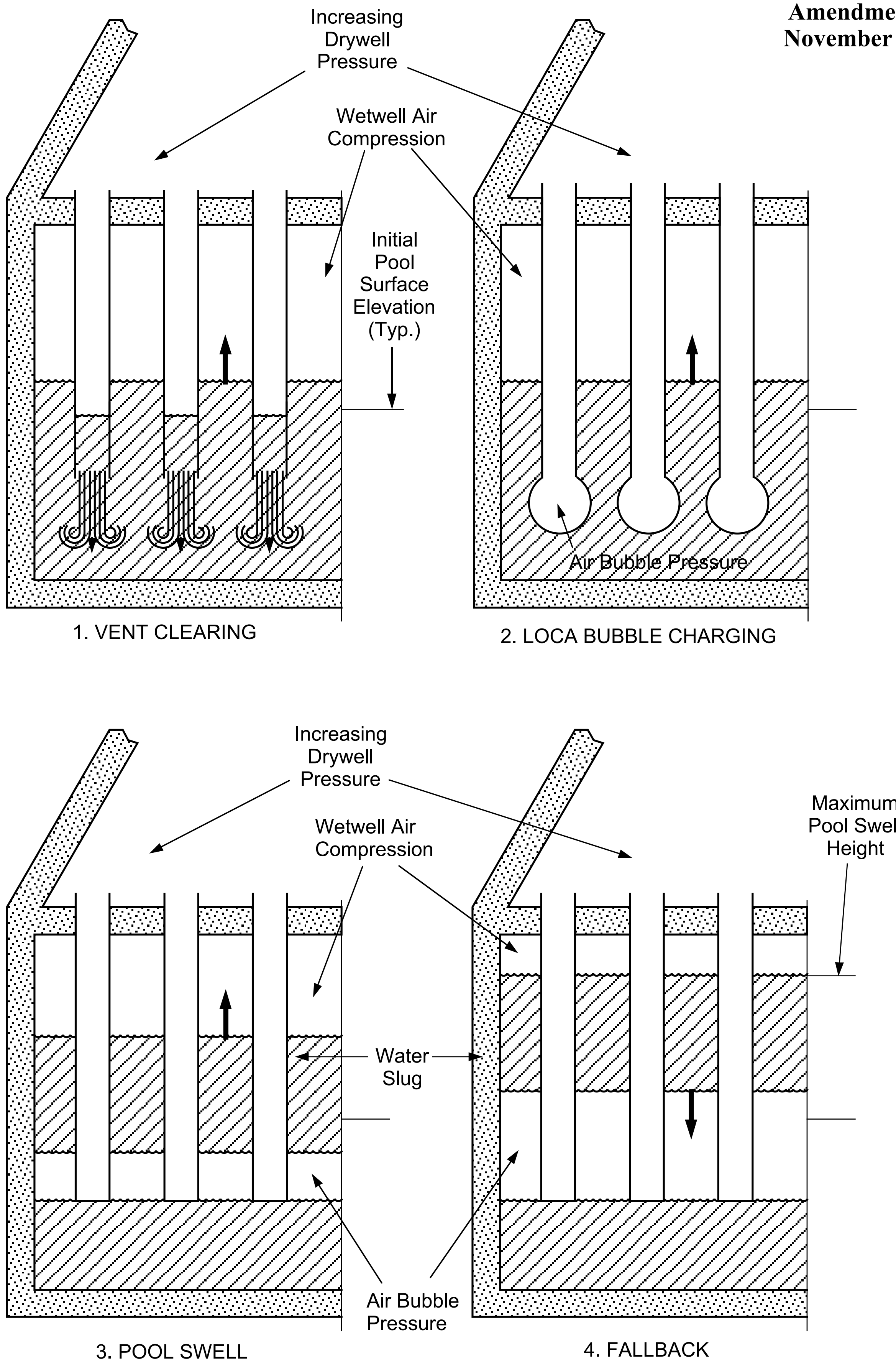
6. The radial location, r, of the structure or segment is taken from the nearest vent. Since the flow from vents occurs in-phase during water clearing/air charging phases of LOCA, the flow field calculations and the above specified loads have already accounted for the multi-vents effect. Therefore, effects of flow from other adjacent vents need not be added or subtracted.

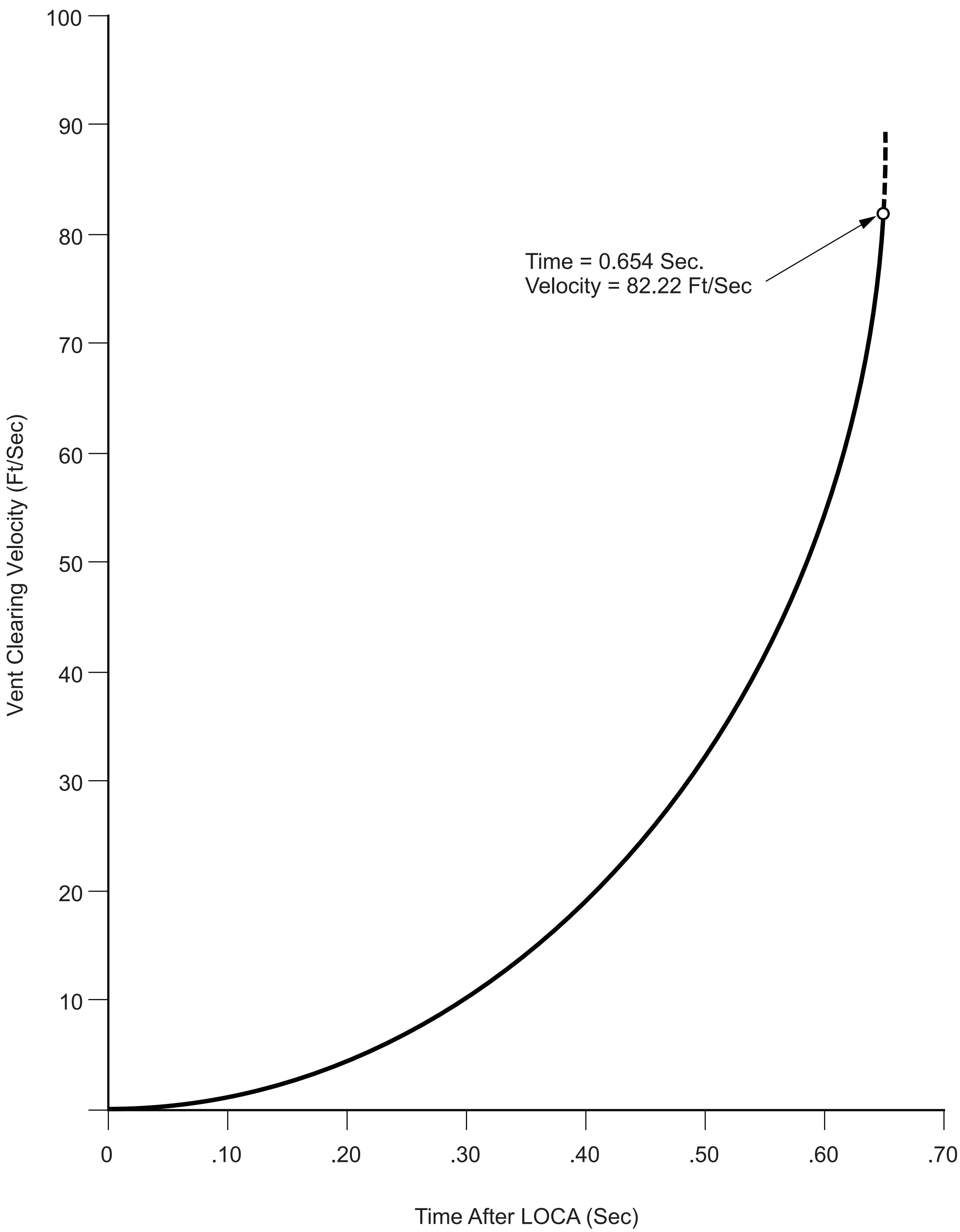
7. The loads specified are equivalent static and utilize a DLF equal to 2.

Table 3A.3.2-9

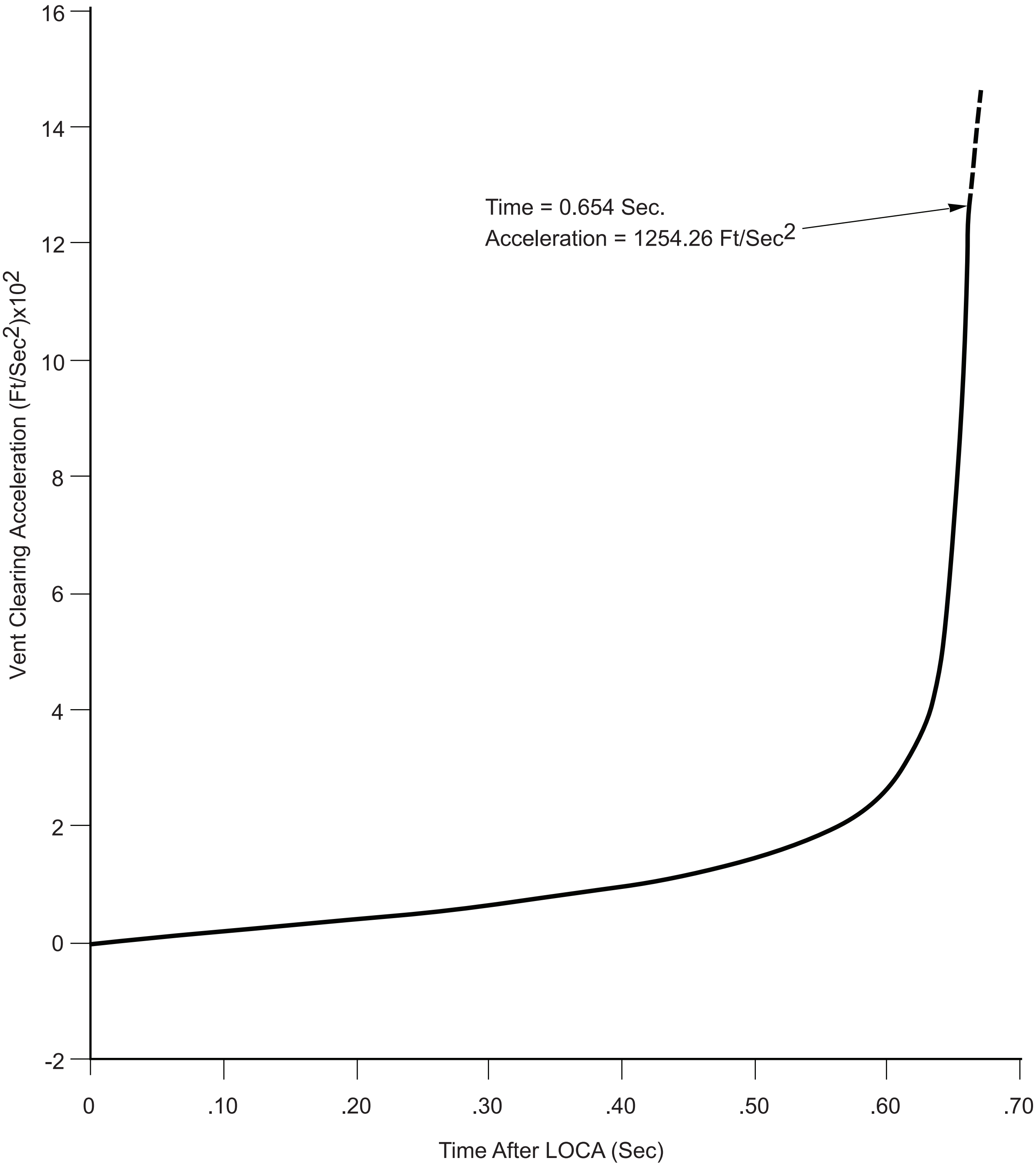
Short-Term Loss-of-Coolant Accident Loads on
Structures Between Elevations 454 ft 4.75 in. and 484 ft 4.75 in.

Structures Located Between El. 454 ft 4.75 in. and El. 484 ft 4.75 in.	Equivalent Static Load: Pool Swell/ Impact $\pm P\psi$ (psi)	Pool Swell Fallback: Drag $\pm P\psi$ (psi)	Pool Swell/ Fallback: Horizontal Lift $\pm Ph$ (psi)
I. Downcomer bracing truss		25	5
II. Platform at el. 472 ft 4 in.: grating perimeter members		8.4	25
III. Piping at 467 ft 9 in. and supports			
A. Horizontal portions of pipes			
a. X-48,118,117,47,26,63,49,101, X-23,24, X-4 (pipe and sleeve)	41	25	15
b. X-64,65	16	20	15
B. Inclined braces above penetrations		50	25
C. Horizontal supports below penetrations		60	25
IV. Penetration sleeves, pipe stubs, and electrical box protective structures			
a. X-51,66 sleeves for: X-81,82,83,84,116	209	25	5
b. X-87A	166	25	5
c. X-86A	62	25	5
d. X-88,87B,86B,116 (piping)		25	5
e. X-81,82,83,84 (piping except X-82e)	209	20	5
f. X-107A, X-107B (electrical box protective structures)	116	25	5

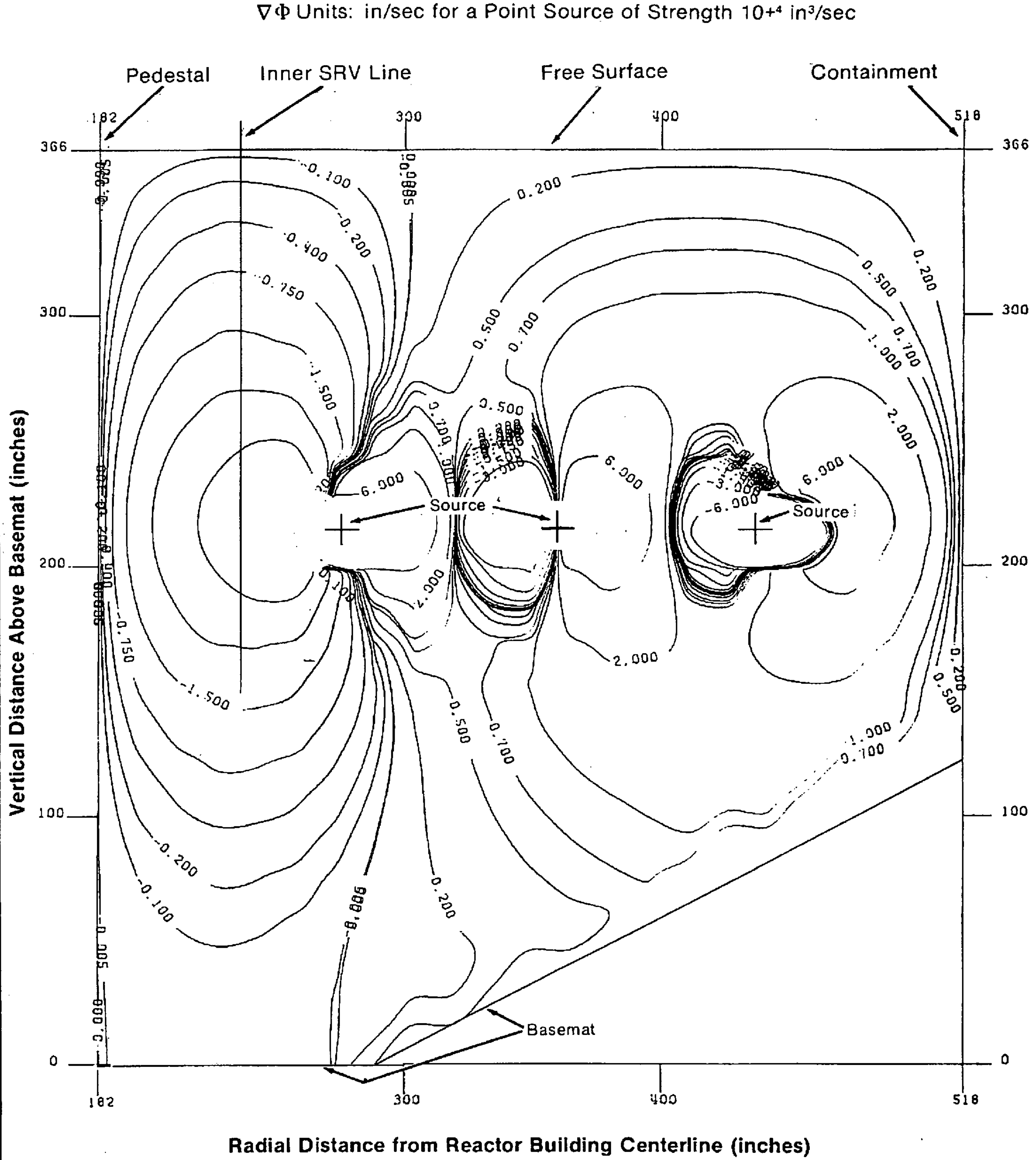

Increasing
Drywell
Pressure
Wetwell Air
Compression
Initial
Pool
Surface
Elevation
(Typ.)
Air Bubble Pressure
1. VENT CLEARING
2. LOCA BUBBLE CHARGING
Increasing
Drywell
Pressure
Wetwell Air
Compression
Maximum
Pool Swell
Height
Water
Slug
Air Bubble
Pressure
3. POOL SWELL
4. FALLBACK



Columbia Generating Station Final Safety Analysis Report	Downcomer Vent Water Clearing Velocity Versus Time	
	Draw. No. 950021.77	Rev.	Figure 3A.3.2-2



Columbia Generating Station **Final Safety Analysis Report**	**Downcomer Vent Water Clearing Acceleration Versus Time**
	Draw. No. **950021.78**	Rev.	Figure **3A.3.2-3**



Columbia Generating Station
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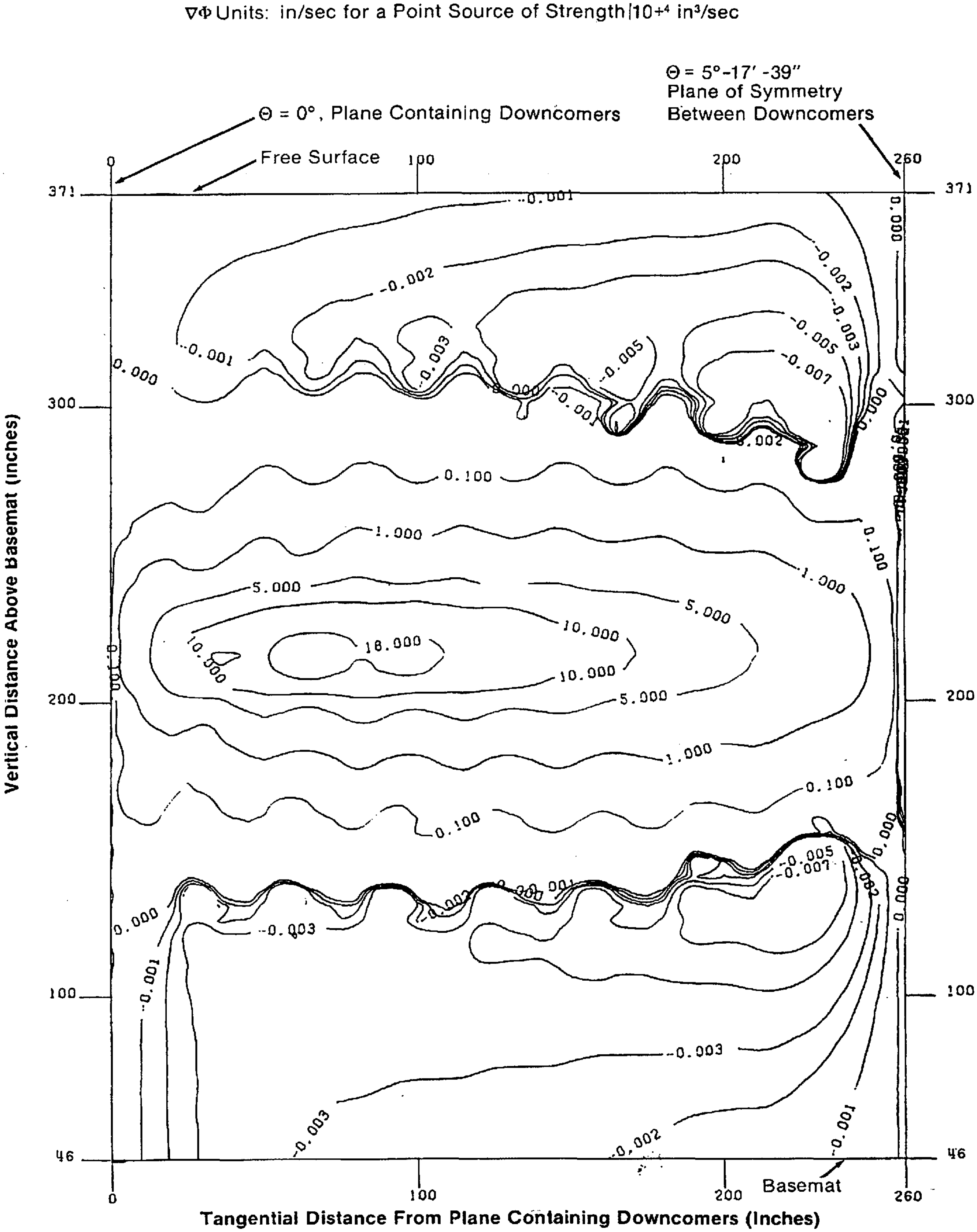
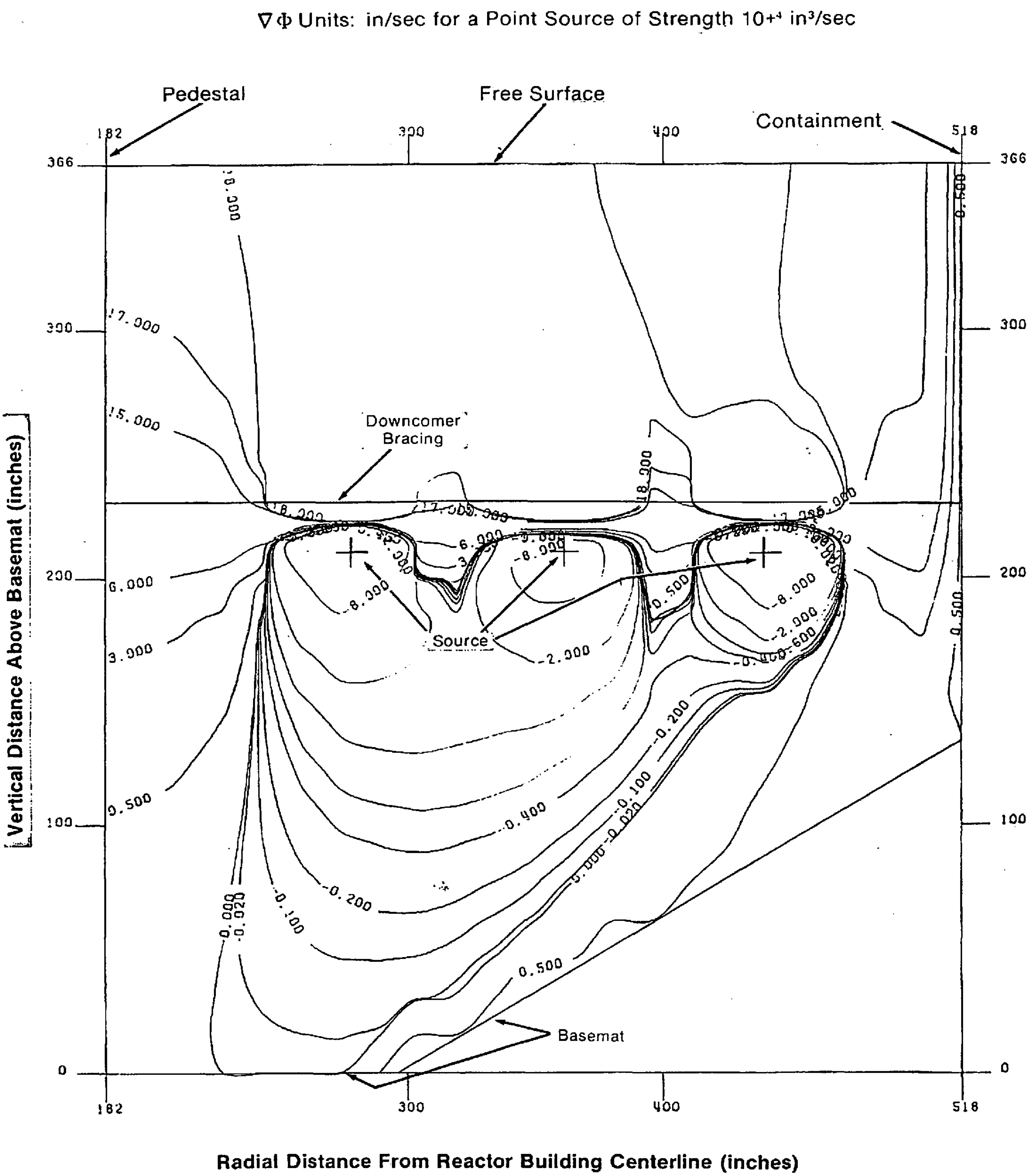
LOCA Bubble Charging Radial Component of ∇Φ in Radial Plane Containing Downcomers

Draw. No. 020361.16 | Rev. | Figure 3A.3.2-4



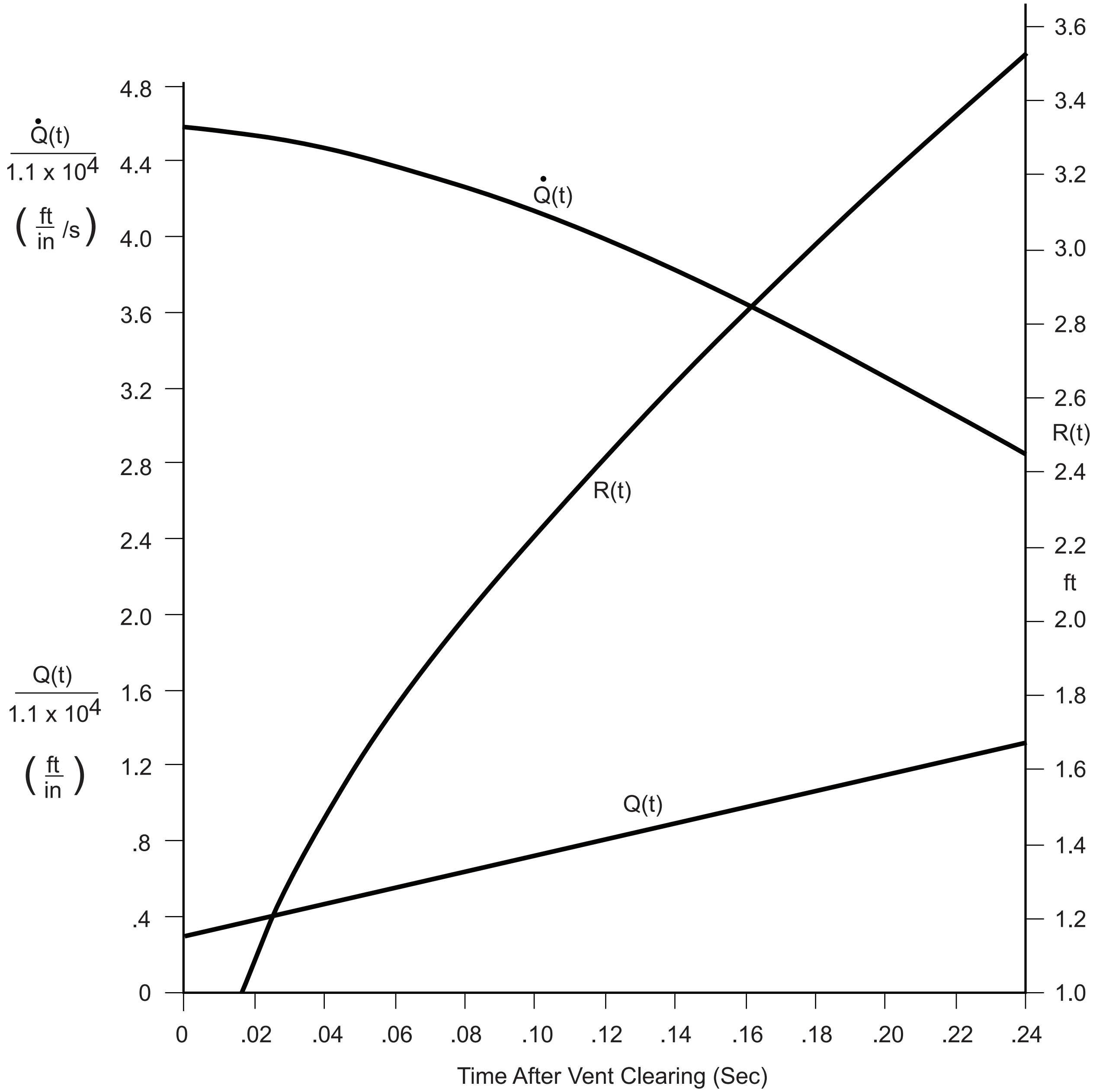
LOCA Bubble Charging Tangential Component of $\nabla\Phi$ in Vertical Cylindrical Surface Through Middle Downcomers

Draw. No. 020361.17	Rev.	Figure 3A.3.2-5	



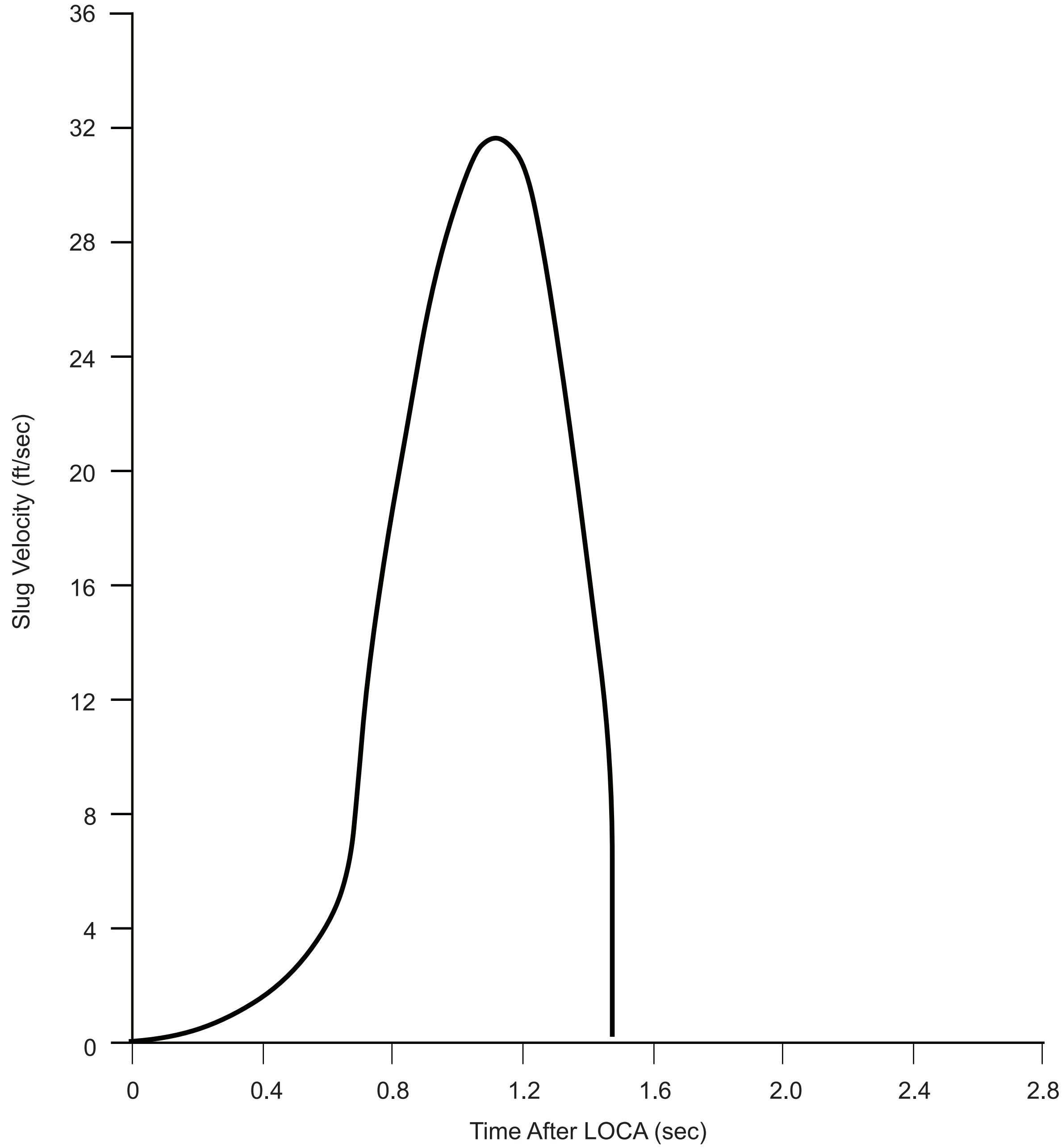
LOCA Bubble Charging Vertical Component of ∇Φ in Radial Plane Containing Downcomers

Draw. No. 020361.18	Rev.	Figure 3A.3.2-6

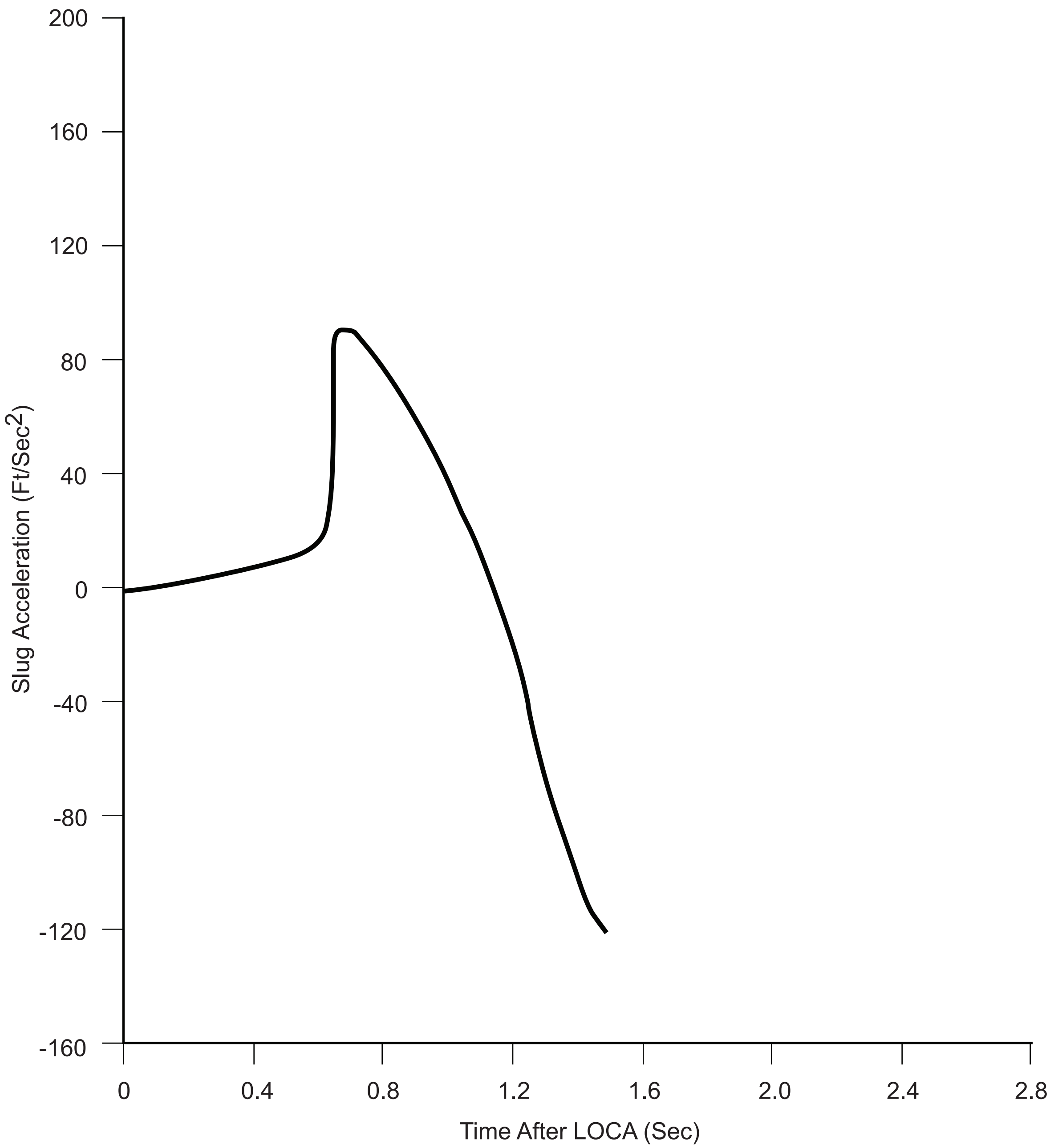


LOCA Bubble Radius and Source Strength Time Histories by PSAM Method

Draw. No. 950021.79 | Rev. | Figure 3A.3.2-7



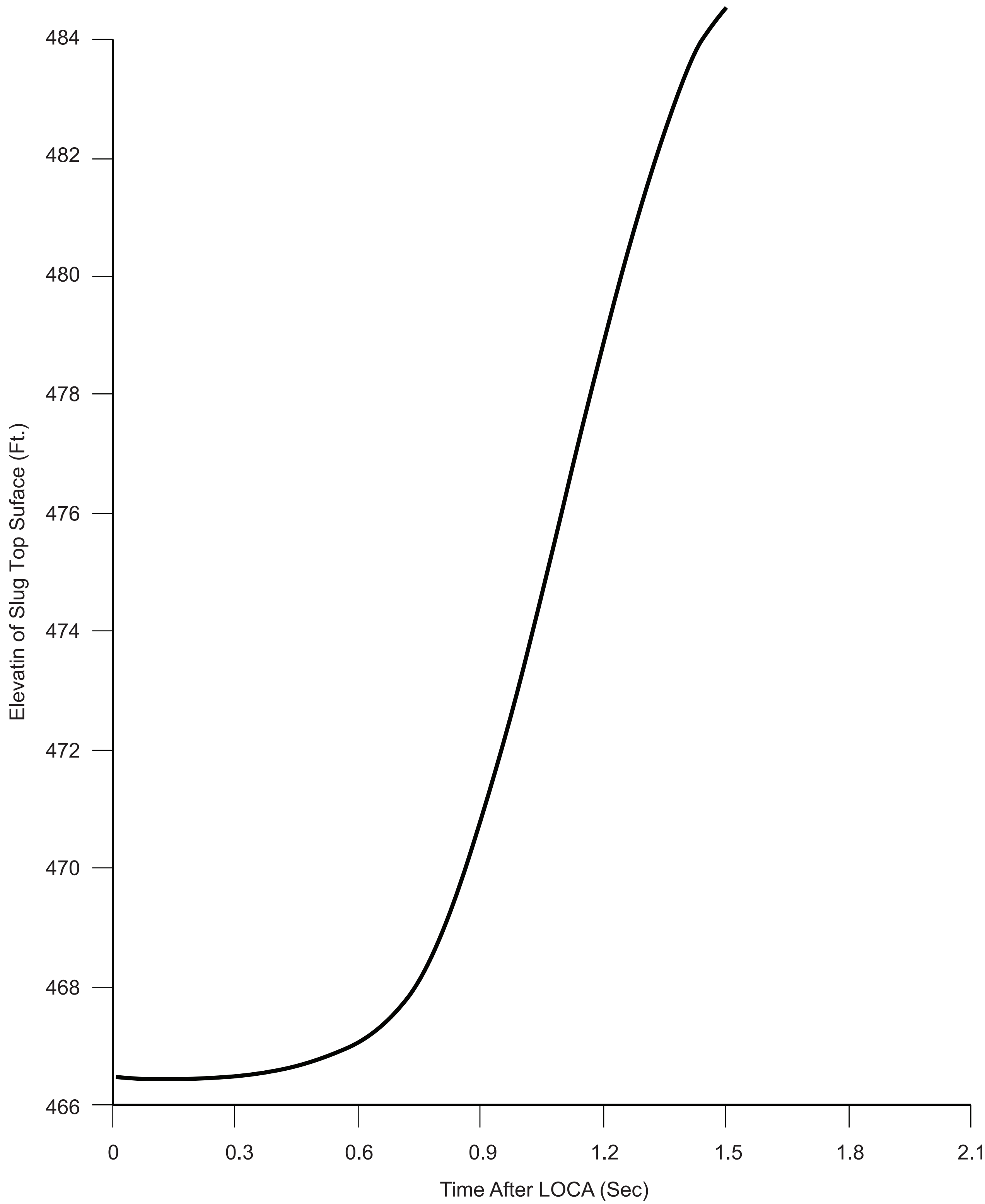
Pool Swell Water Slug Velocity Versus Time



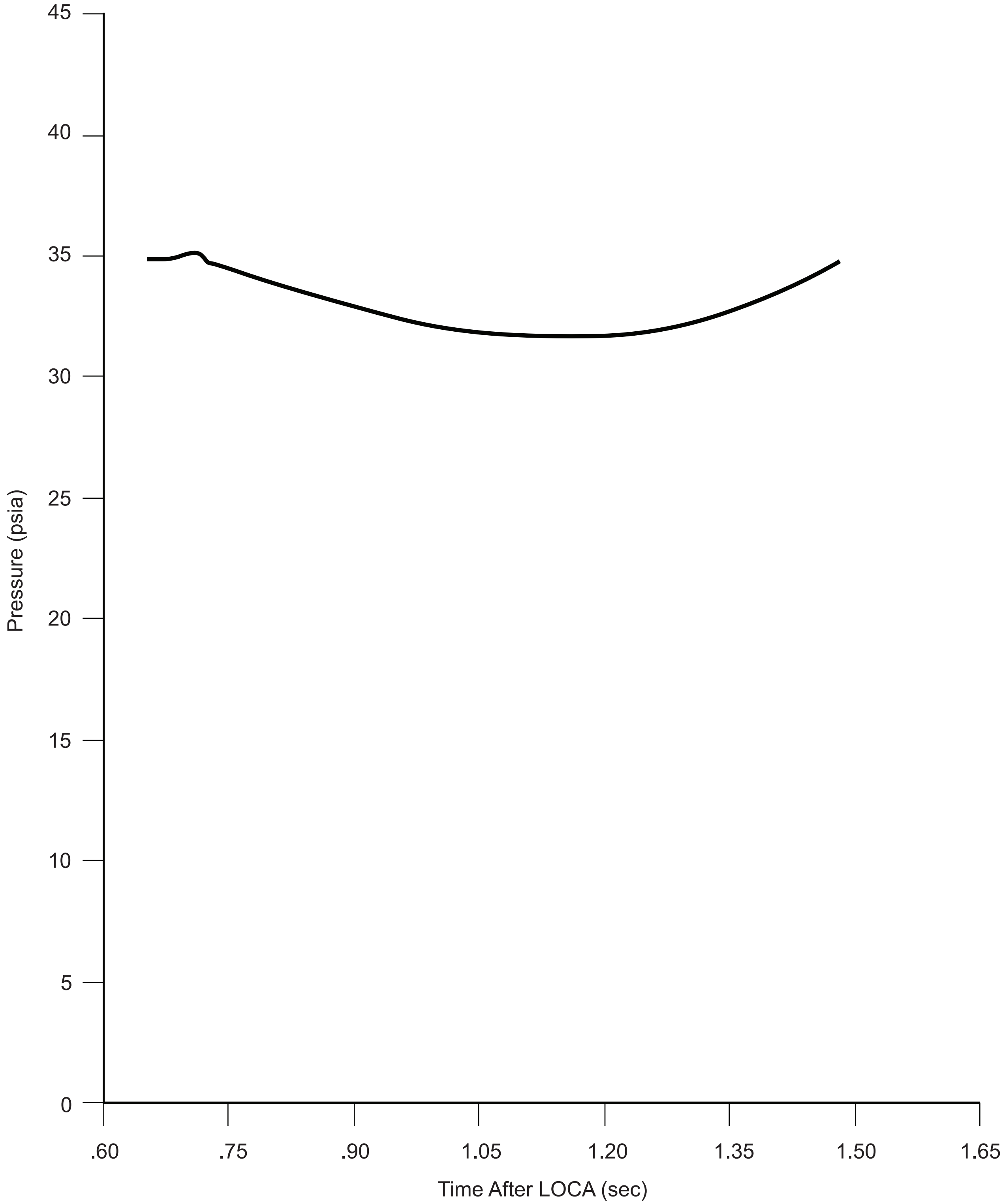
Columbia Generating Station
Final Safety Analysis Report

Pool Swell Water Slug Acceleration Versus Time

Draw. No. 950021.81	Rev.	Figure 3A.3.2-9



Pool Swell Water Slug Elevation
(Top Surface) Versus Time



Columbia Generating Station
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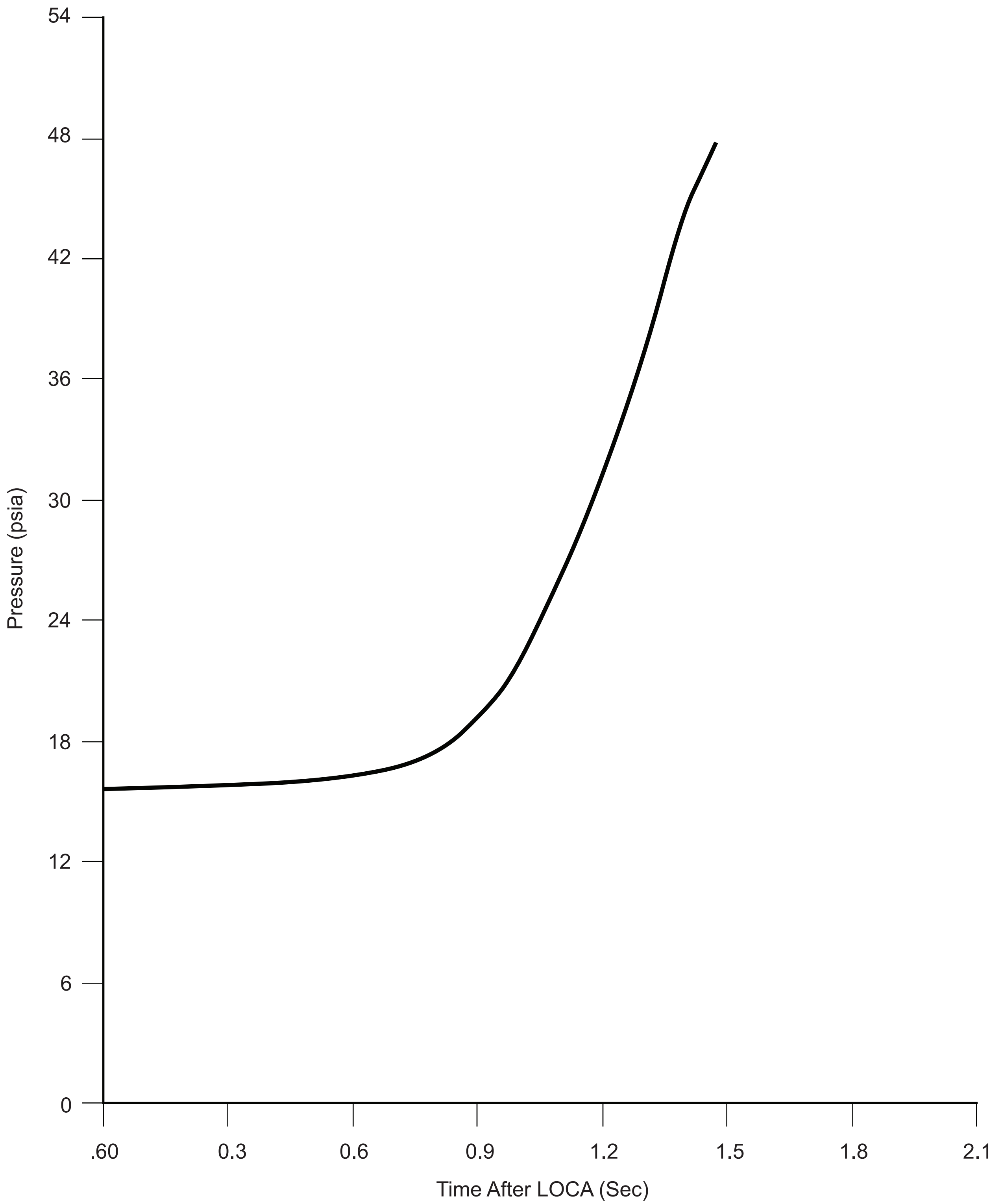
Pool Swell Air Bubble Pressure Versus Time

Draw. No. 950021.83 | Rev. | Figure 3A.3.2-11



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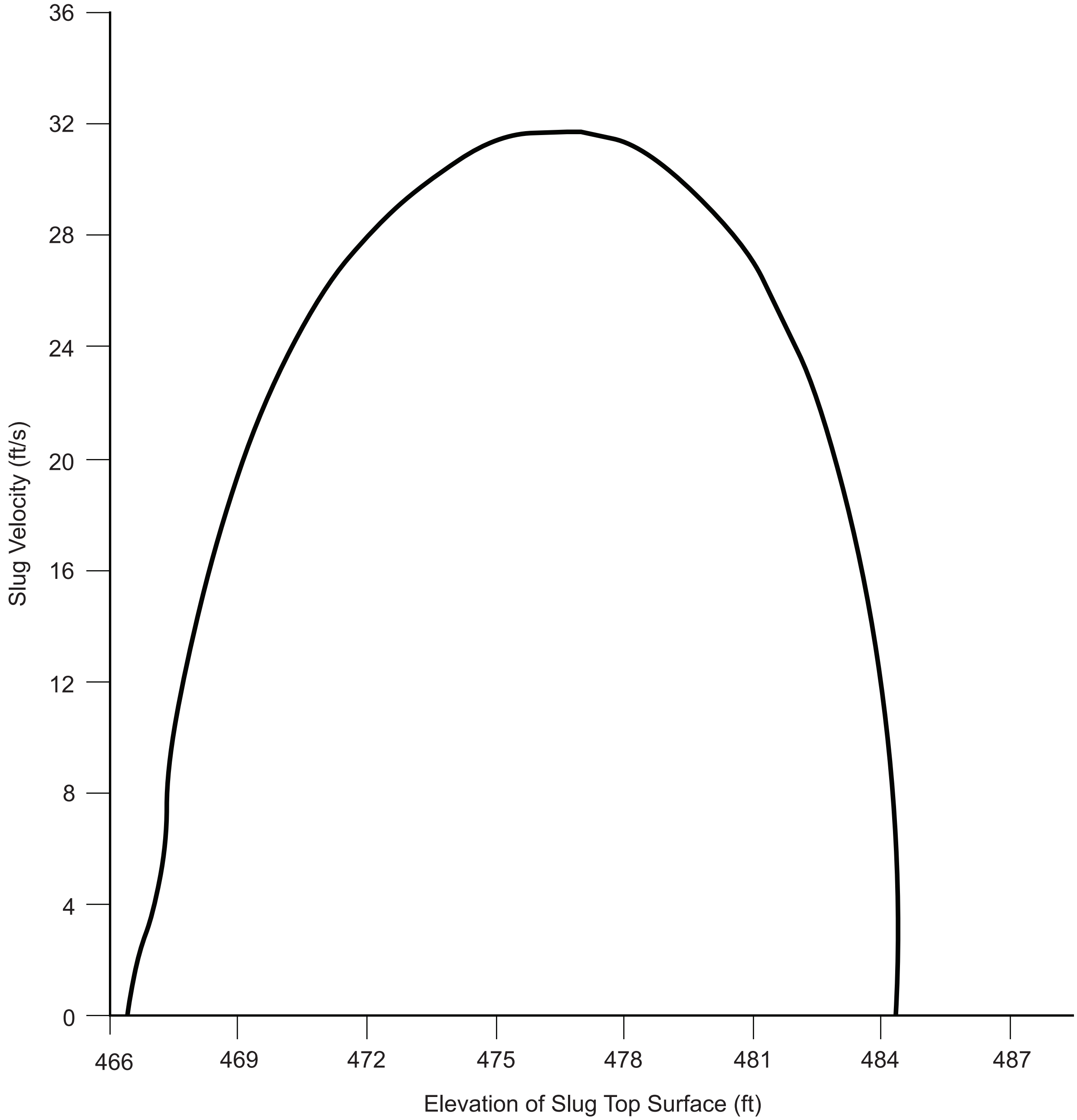
Pool Swell Wetwell Air Pressure Versus Time

Draw. No. 950021.84	Rev.	Figure 3A.3.2-12



Columbia Generating Station
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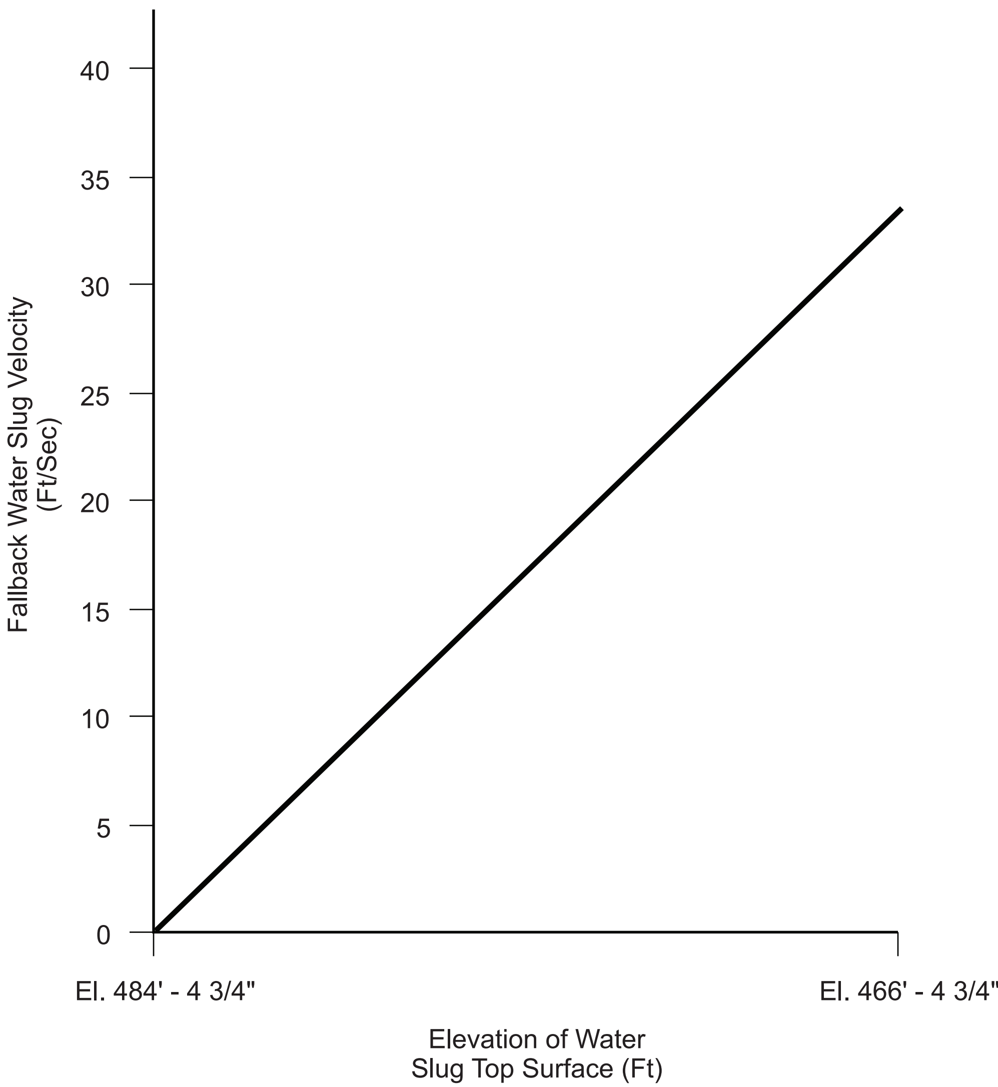
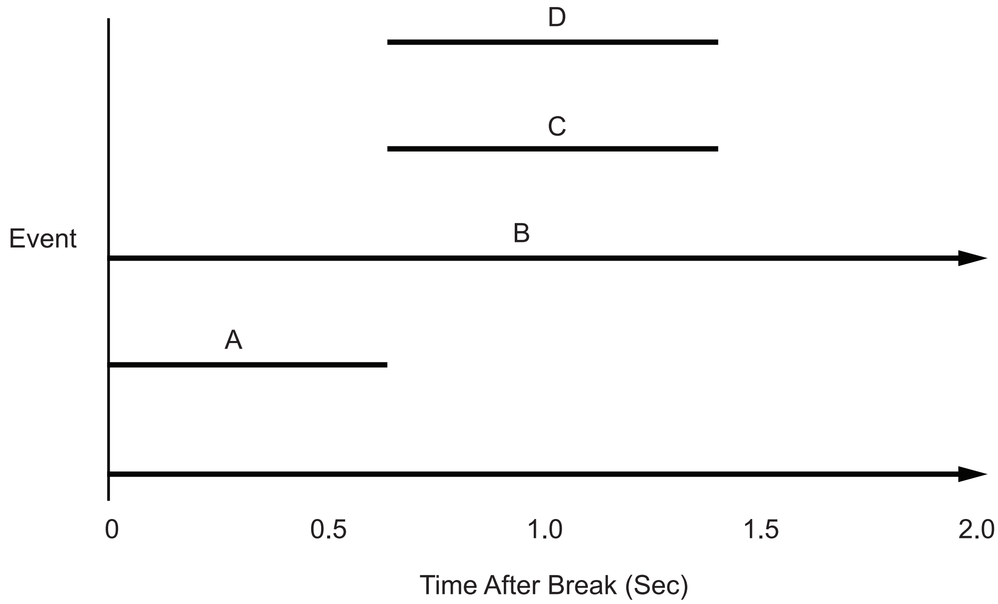
Pool Swell Water Slug Velocity Versus Elevation of Slug Top Surface

Draw. No. 950021.85	Rev.	Figure 3A.3.2-13

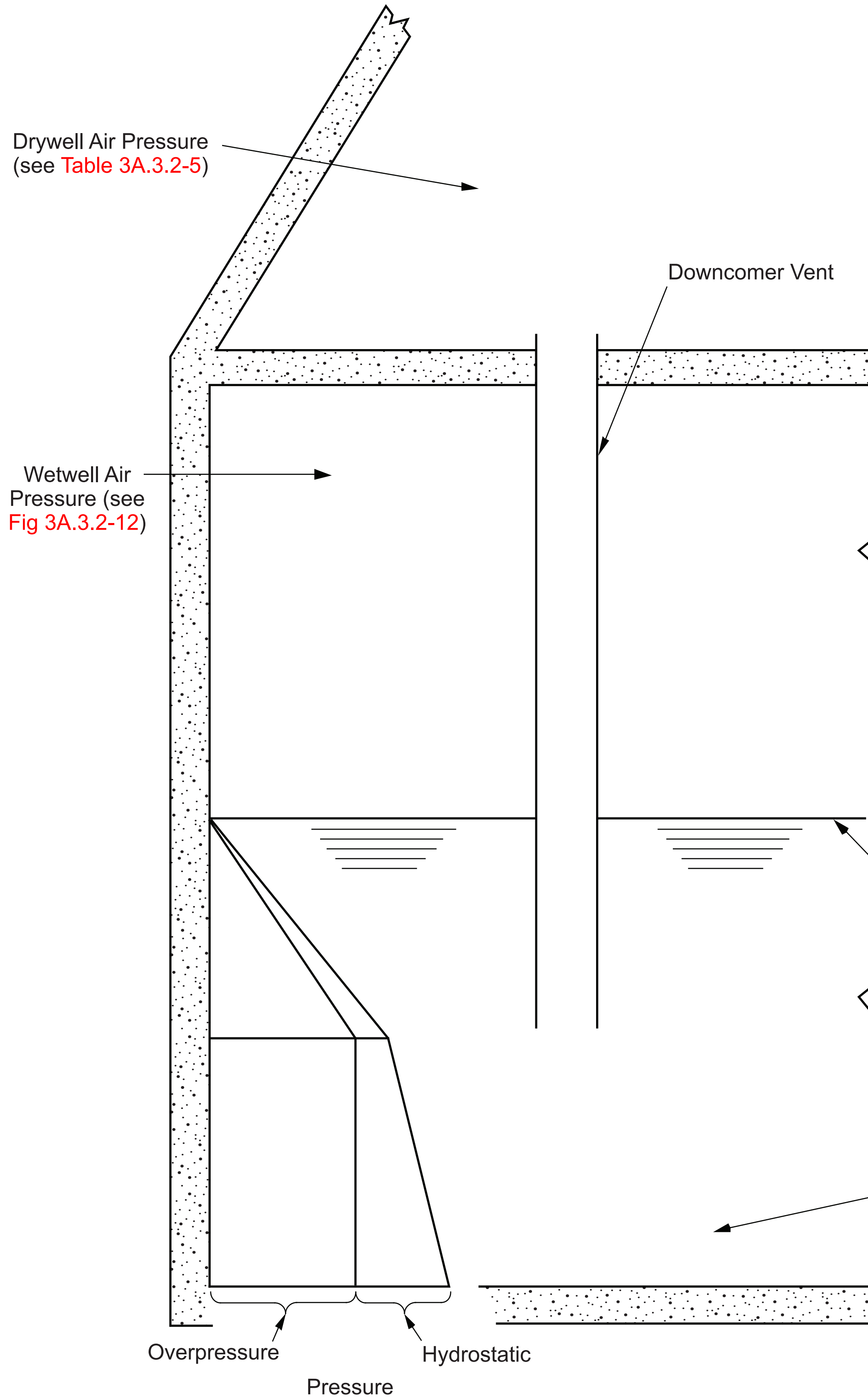


Fallback Water Slug Velocity Versus Elevation of Water Slug Top Surface

Draw. No. 950021.86 | Rev. | Figure 3A.3.2-14

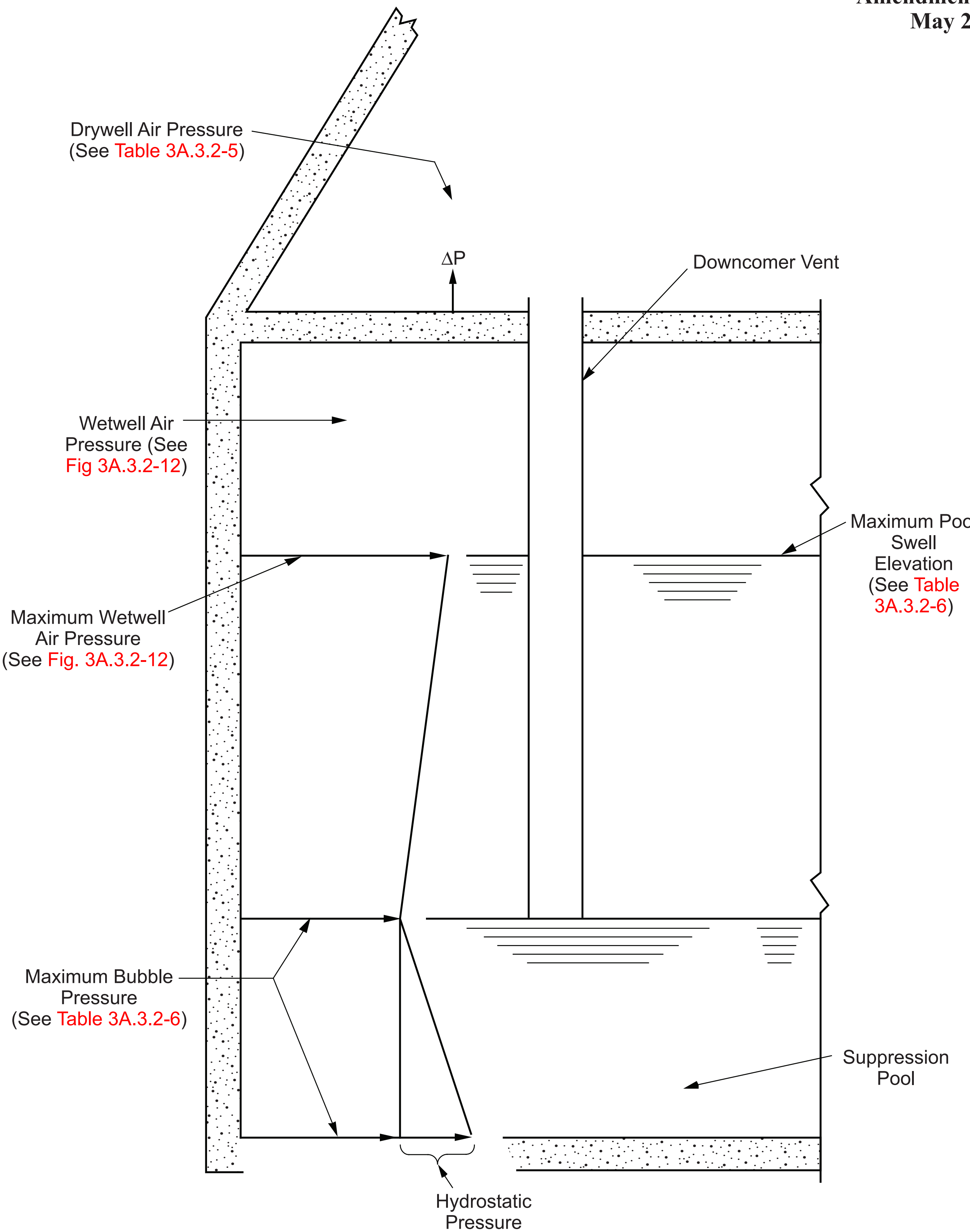


Event	Loading Phenomenon
A	Submerged Boundary Loads During Vent Clearing
B	Drywell Pressurization
C	Loads on Submerged Boundaries During Pool Swell
D	Wetwell Air Compression

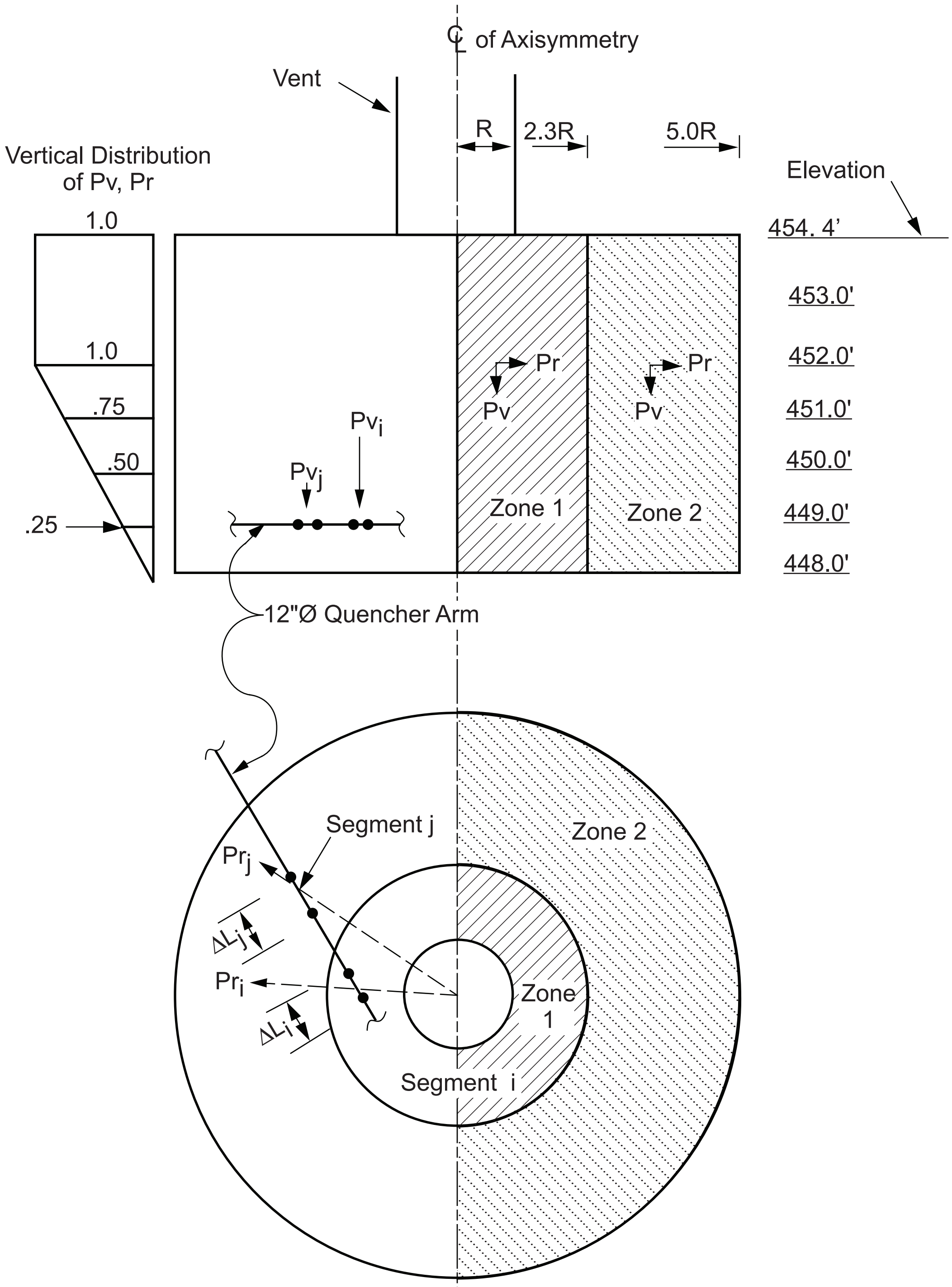


**LOCA Boundary Load Distribution
During Vent Clearing**

Draw. No. 950021.88 | Rev. | Figure 3A.3.2-16



LOCA Boundary Load Distribution
During Pool Swell



Example:

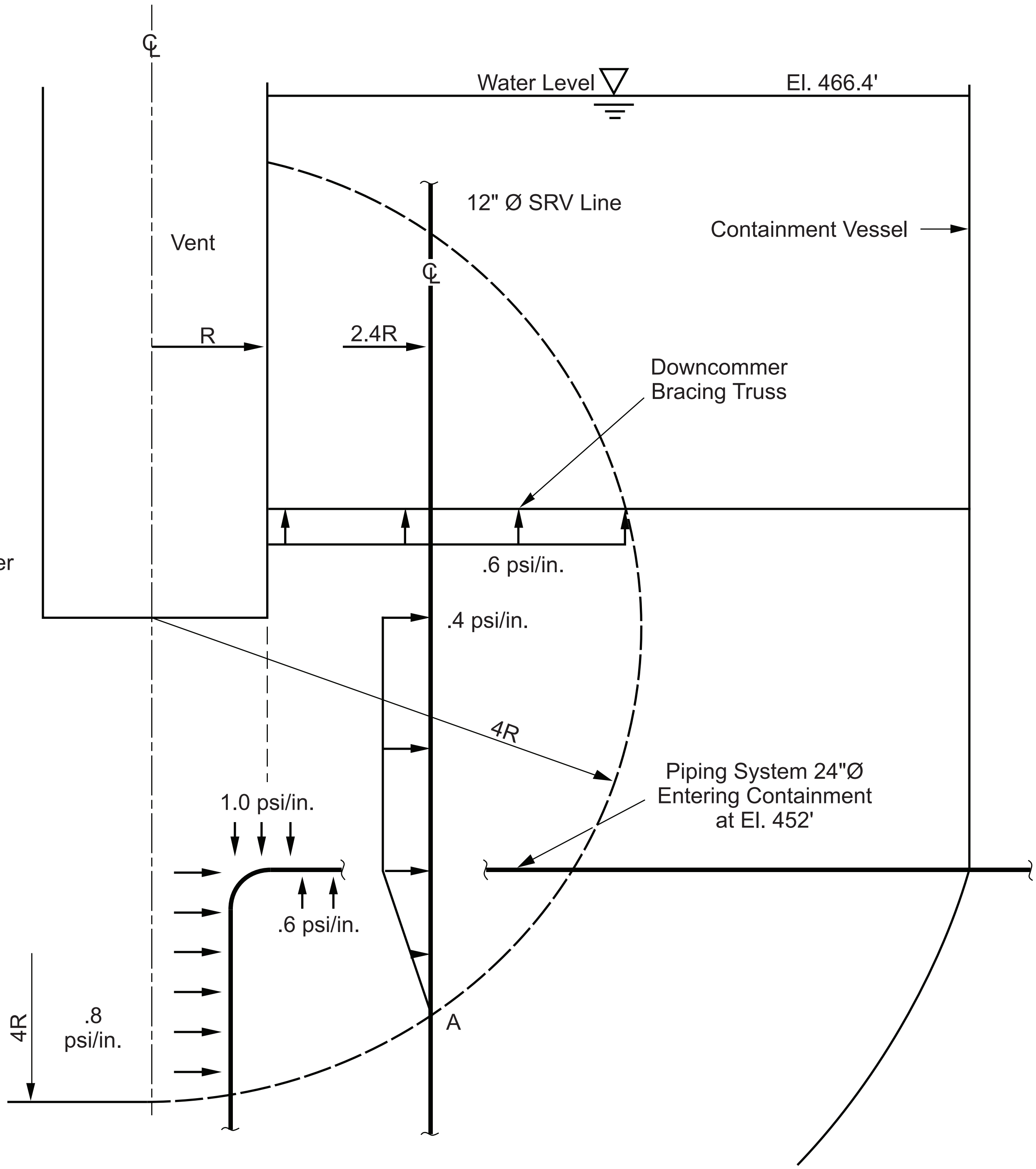
$Pv_i = 12\Delta L_i \times .25 \times (100)$

$Pr_i = 12\Delta L_i \times .25 \times (\pm 60)$

$Pv_j = 12\Delta L_j \times .25 \times (-25)$

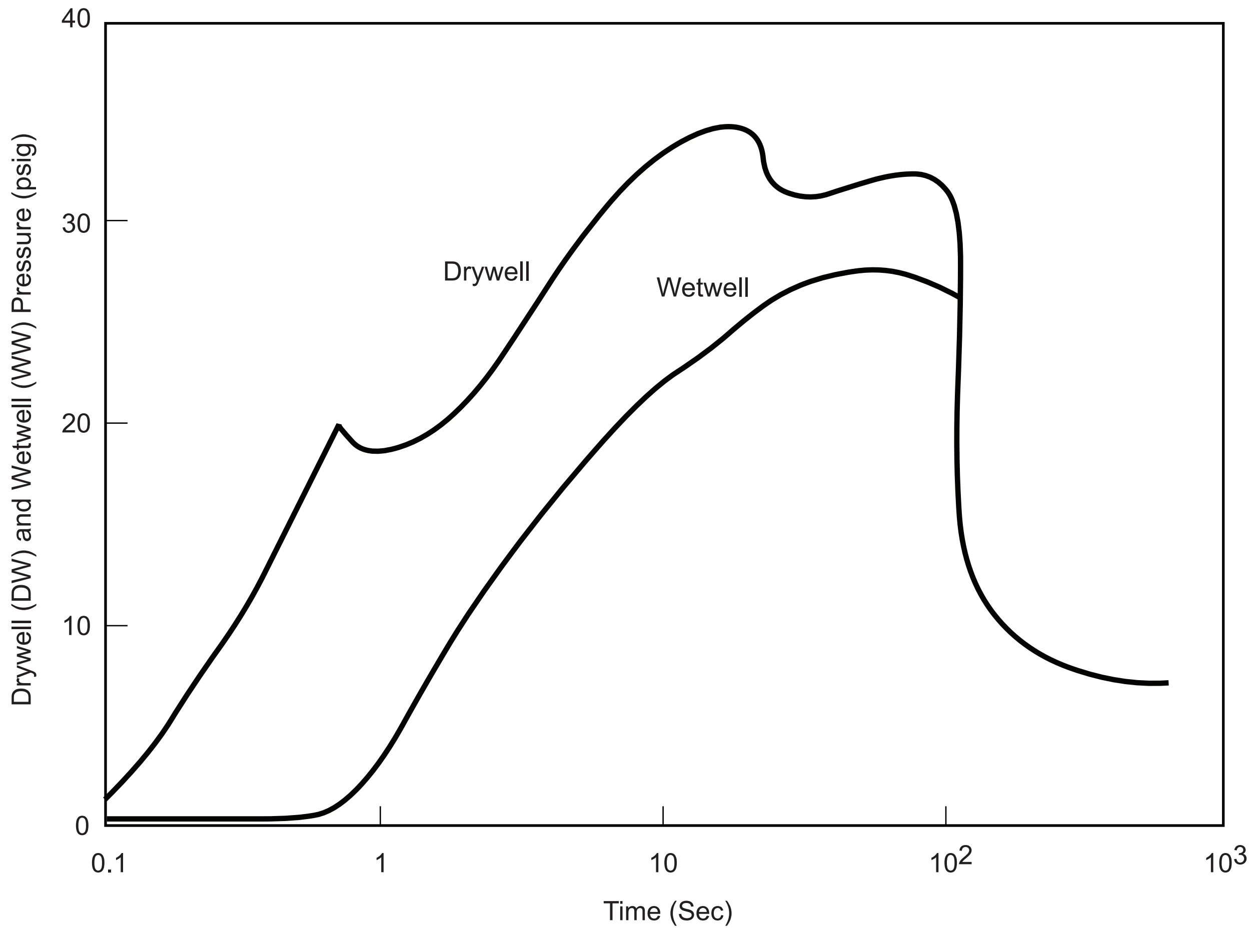
$Pr_j = 12\Delta L_j \times .25 \times (6)$

Columbia Generating Station Final Safety Analysis Report	Distribution of Short Term LOCA Loads on Structures Below El. 454.4 ft	
Draw. No. 950021.90	Rev.	Figure 3A.3.2-18

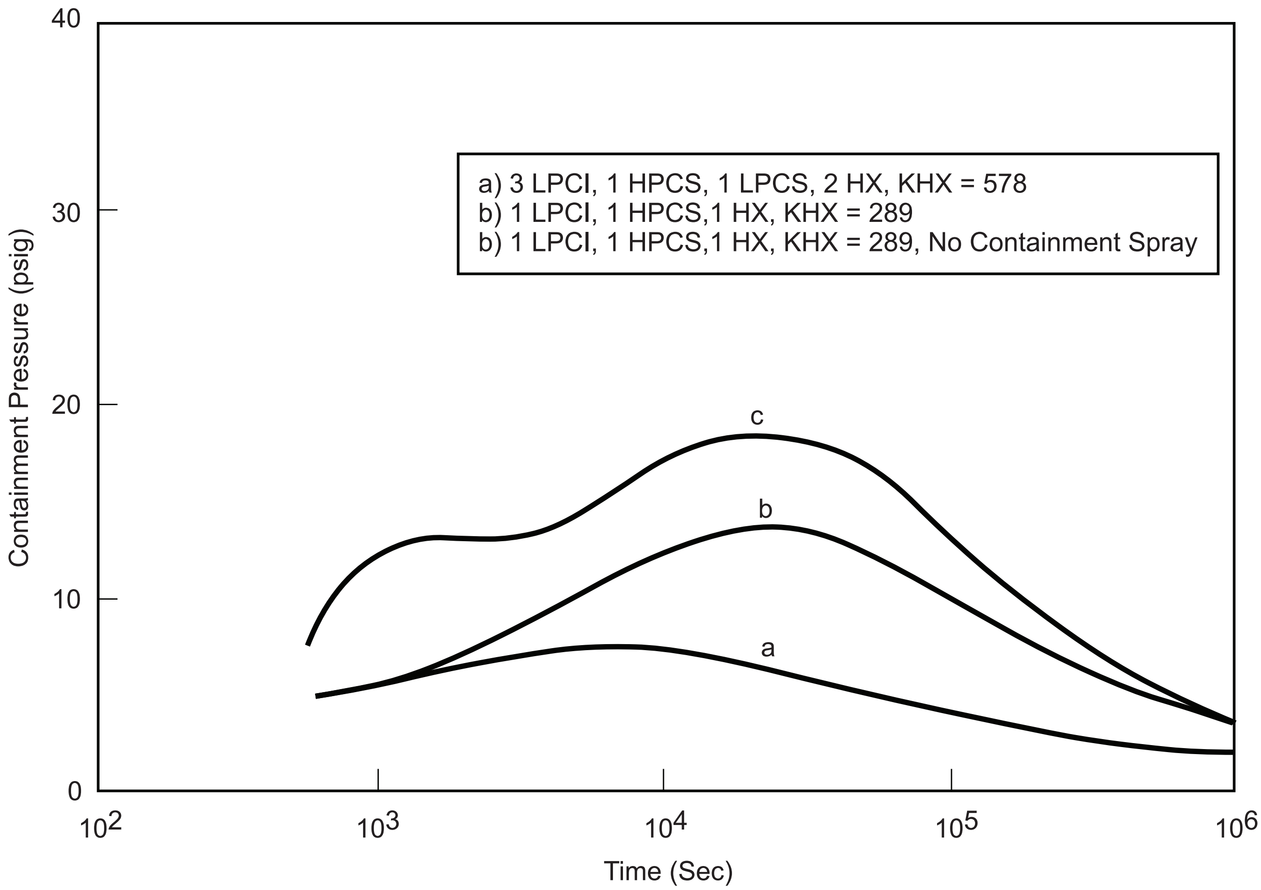


Pressure Gradients Across Submerged Structures Due to Chugging

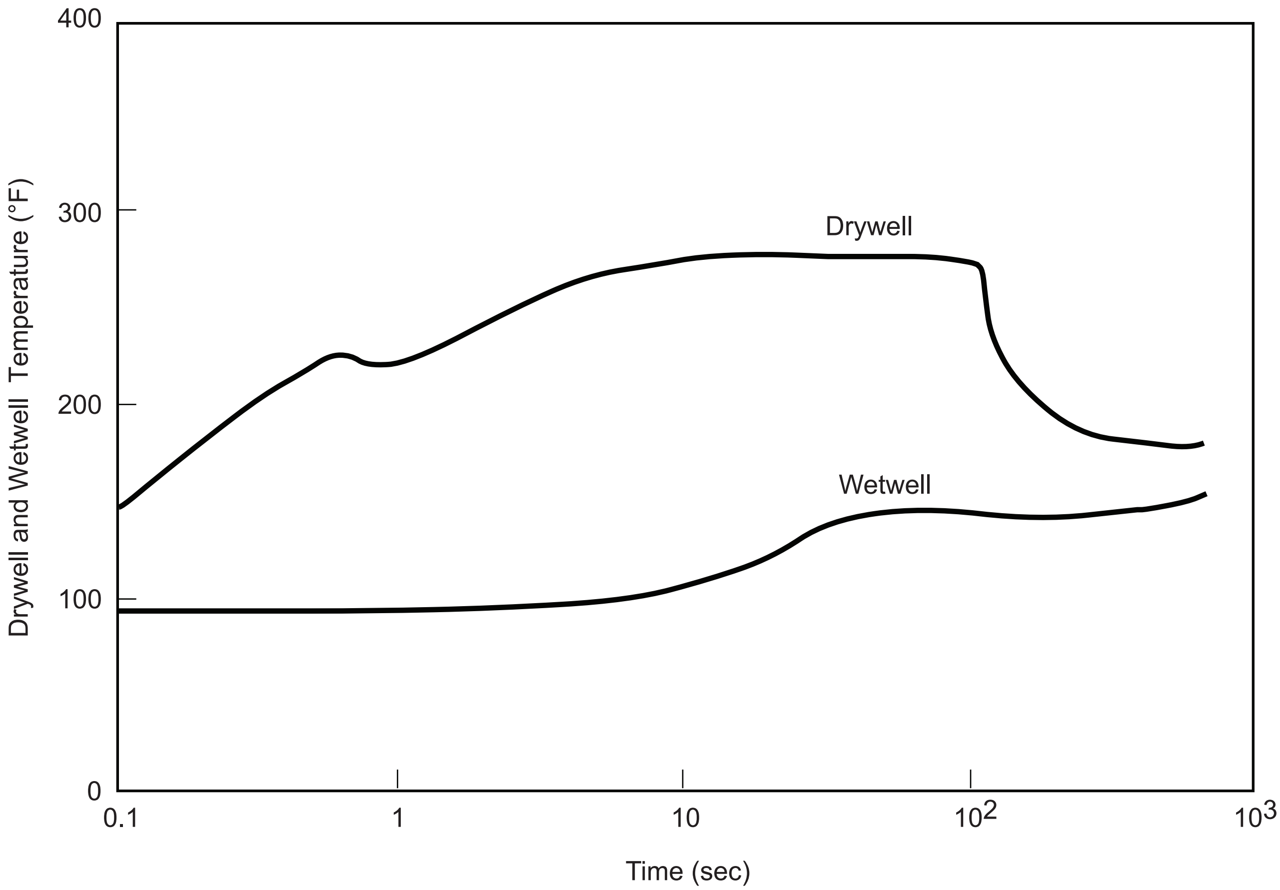
Draw. No. 950021.91 | Rev. | Figure 3A.3.2-19



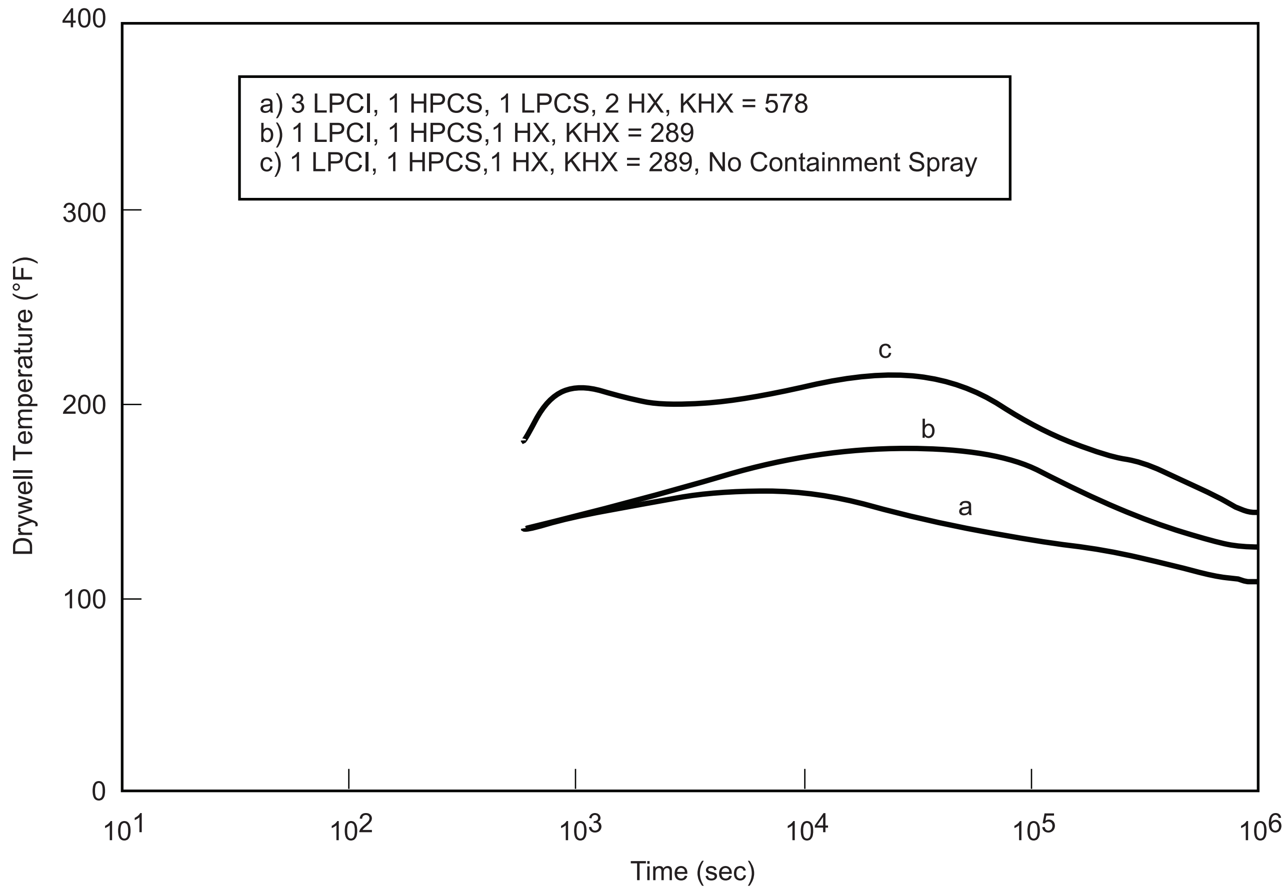
Large Recirculation Line Break - Pressure Response - Minimum ECCS



Columbia Generating Station Final Safety Analysis Report	Containment Pressure Response for Large Recirculation Line Break - Cases A, B, and C		
	Draw. No. 950021.93	Rev.	Figure 3A.3.2-21

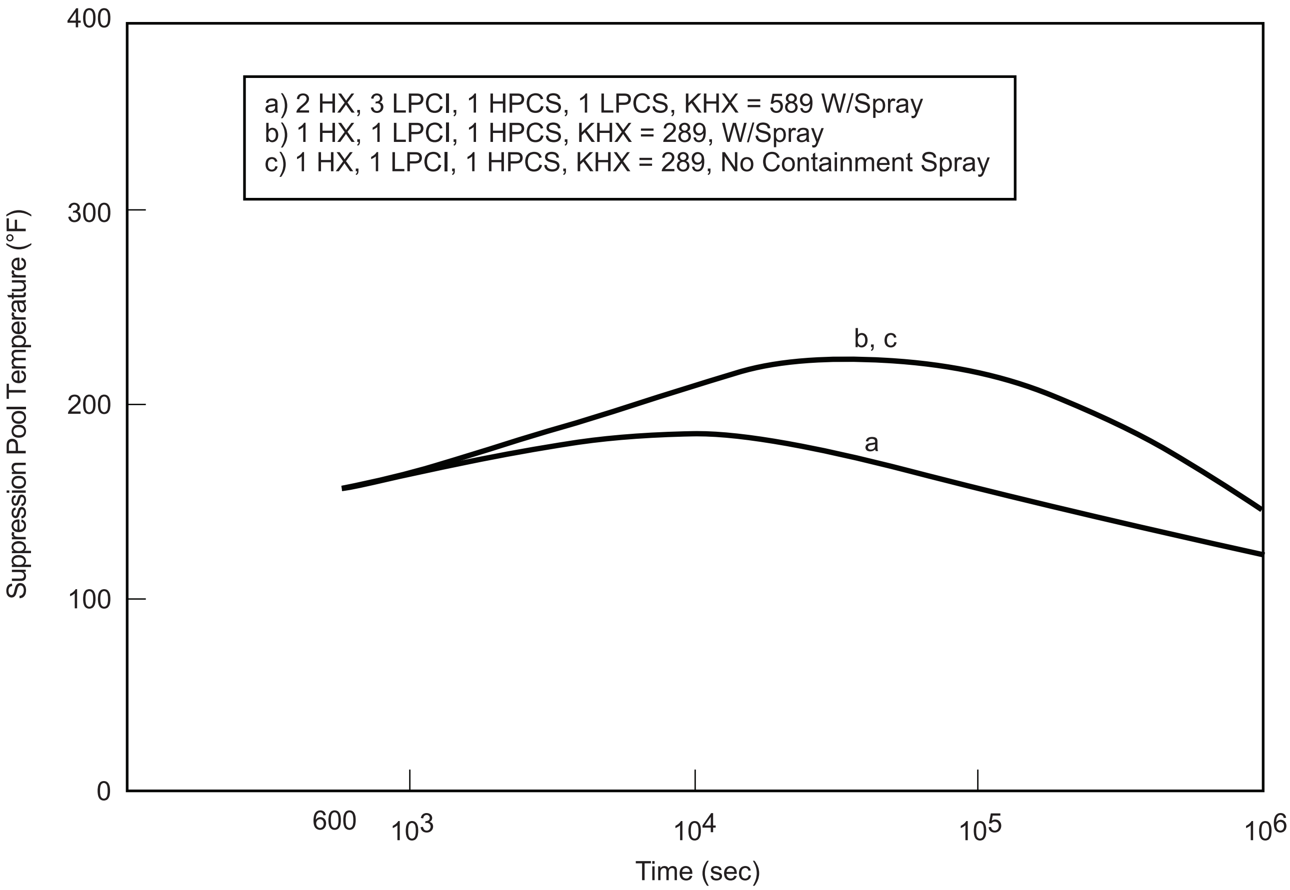


Large Recirculation Line Break - Temperature Response



Drywell Temperature Response for Large Recirculation Line Break - Cases A, B, and C

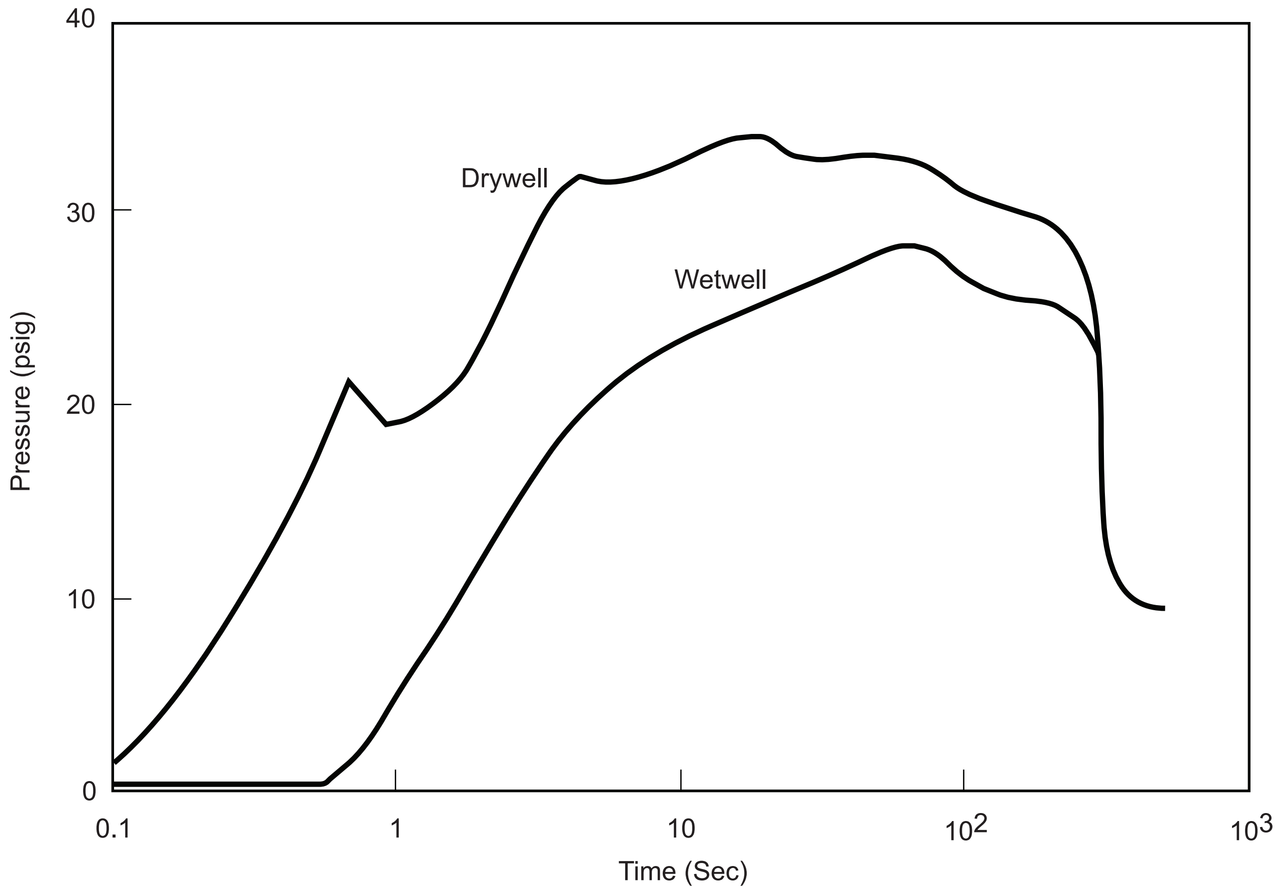
Draw. No. 950021.95 | Rev. | Figure 3A.3.2-23



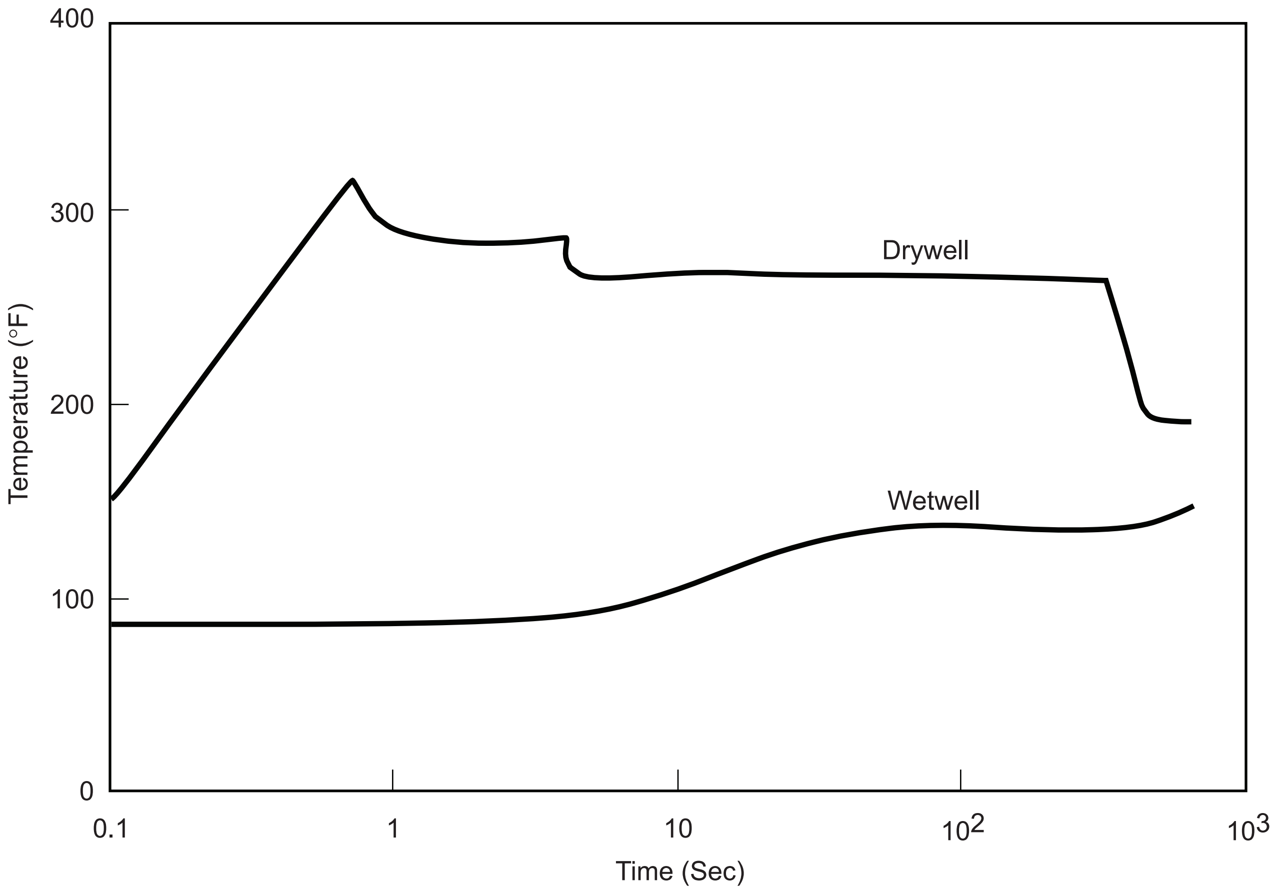
Columbia Generating Station
Final Safety Analysis Report

Suppression Pool Temperature Response for Large Recirculation Line Break - Long Term Response

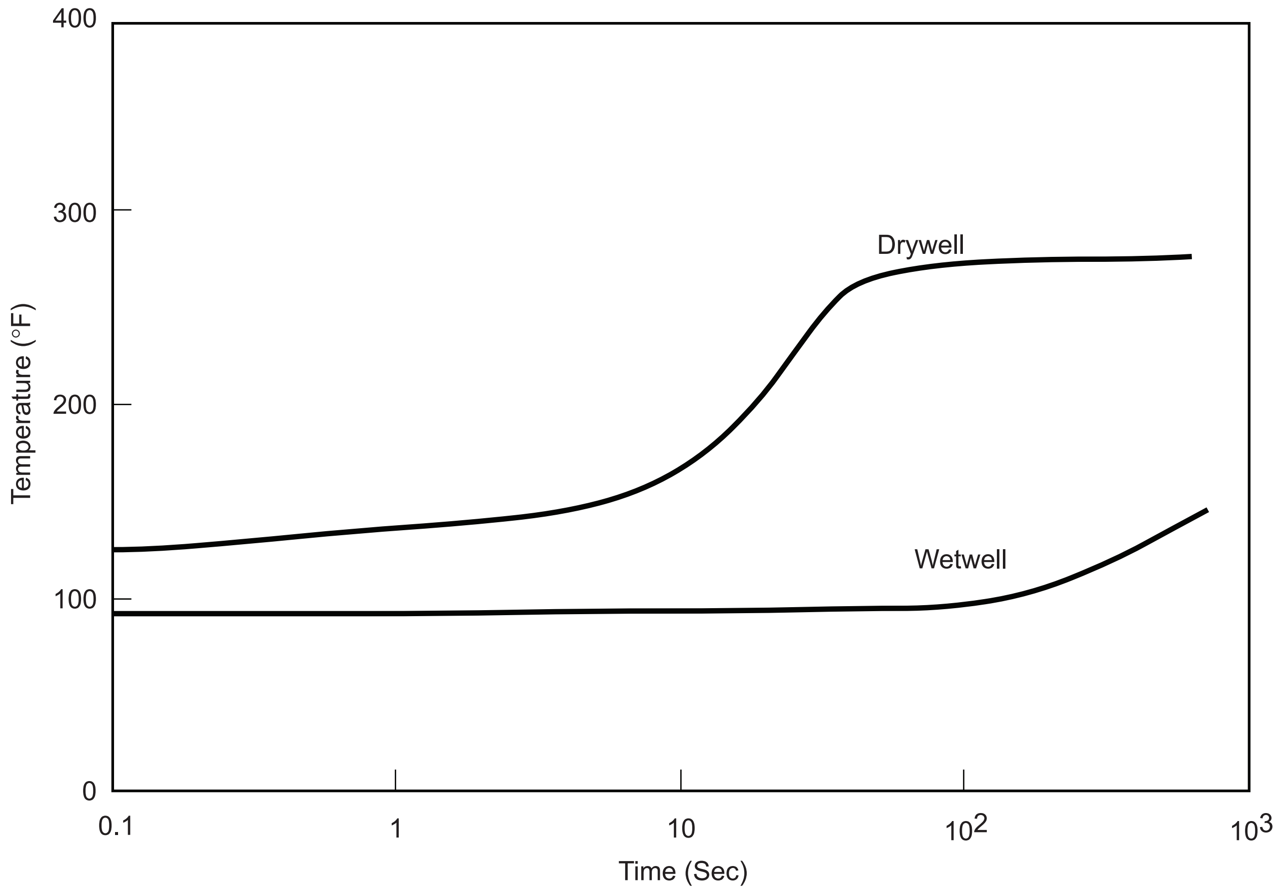
Draw. No. 950021.96	Rev.	Figure 3A.3.2-24



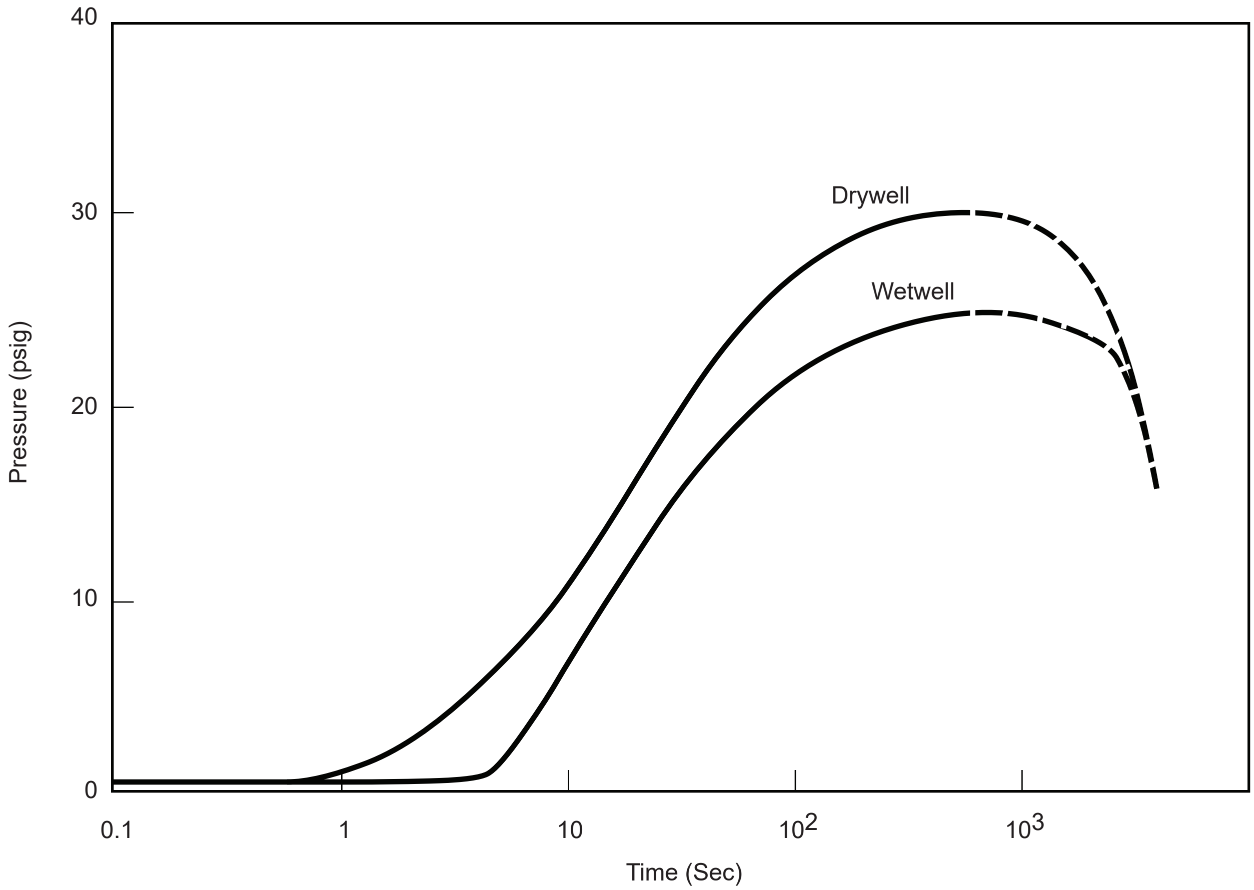
Pressure Response
Main Steam Line Break



Temperature Response -
Main Steam Line Break -
Minimum ECCS



Columbia Generating Station **Final Safety Analysis Report**	**Temperature Response - Recirculation Line Break (0.1 ft^2)**
	Draw. No. 950021.99 | Rev. | Figure 3A.3.2-27



Pressure Response -
Recirculation Line Break (0.1 ft^2)

Draw. No. 960222.84 | Rev. | Figure 3A.3.2-28

3A.3.3 LOAD SUMMARY

A load summary is given in Table 3A.3.3-1 to provide guidance in identifying loads and to provide references to more detailed discussions of them. The table lists the loads being evaluated and the peak load magnitude directly applied to each structure, or a reference to the DAR sections where the information is derived. It also lists the classification of the load as primary or secondary as defined by NRC.

Table 3A.3.3-1[a]

Summary of Hydrodynamic Loads on Wetwell Structures

												Piping System					
Load Category	Stl. Contain-ment	Vert. and Horiz. Tees (6)	Base-mat	Pedestal	Dia. Floor	Dia. Floor Seal	Down-comer Bracing	Column	Down-comer	SRV Piping System	Quencher	Plat-forms and Ladder	Fully Sub. (3)	Part. Sub.	In Pool Swell Zone (3)	DAR Ref. Sec.	Load Class
SRV																	
Water Clearing	sm	sm	sm	sm	N/A	N/A	sm	sm	sm	sm	sm	N/A	sm	sm	N/A	3.1.2.1	Sec.
Air Clearing	151	151	151	151	N/A	N/A	3.1.3.2	3.1.3.2	3.1.3.2	3.1.3.2	3.1.3.2	N/A	3.1.3.2 .	3.1.3.2	N/A	3.1.3	Prim.
Steam Cond.	sm	sm	sm	sm	N/A	N/A	sm	sm	sm	sm	sm	N/A	sm	sm	N/A	3.1.3.2	Sec.
LOCA																	
Vent Water Clearing	24	24	24	sm	N/A	N/A	sm	3.2.3.3	sm	3.2.3.3	3.2.3.3	N/A	3.2.3.3	sm	N/A	3.2.3	Prim.
Air Bubble Charging	21[b]	21[b]	21[b]	sm	N/A	N/A											

[a] Conformance to NRC acceptance criteria for each load in this table is presented in Attachment 3A.H.
[b] Values are in psig.

Notes:
1. Peak dynamic pressure.
2. Unless identified as dynamic pressure, the numerical value given is equivalent static peak pressure (i.e., includes a DLF).
3. All fully-submerged piping systems enter the pool below vent exit.
4. sm denotes load as small, not requiring consideration.
5. Between elevations 466 ft-4.75 in. and 484 ft-4.75 in.
6. LOCA induced vent, thrust loads are neglected as secondary loads (see Reference 3A.3.2-1).

3A.3.4 SEQUENCE OF DYNAMIC LOADS

The effects of various dynamic loads on structures are analyzed separately. It is important to establish the relative time sequence of all dynamic events to obtain a realistic assessment of design margins. The DFFR (Reference 3A.3.2-2) established relative sequence of dynamic loads during a single SRV discharge event and a postulated LOCA. It is noted that during an SRV actuation the water clearing loads, the air clearing loads, and the steam condensation loads occur in a sequence and the peak dynamic effects due to each need not be combined. Similarly, the LOCA water clearing, air clearing (bubble charging), pool swell impact, pool swell drag, pool fallback, and the high, intermediate, and low mass flux steam condensation occur in a sequence and the peak dynamic effects due to each need not be combined. During pool swell or fallback, drag pressure parallel to the flow and lift pressure normal to the flow occur simultaneously, and the two loads are combined. Also, for some submerged structures, pool swell impact occurs at the upper parts of the structure while pool swell drag and lift loads act on the lower parts of the structure. In this case, the three loads are combined. The continuity of short-term water and air clearing phases during an SRV discharge or during a LOCA and the pool swell impact and drag loads during a DBA LOCA is recognized either by specifying a conservative DLF for use with peak load value of the combined time history or by use of the combined time history in the dynamic analysis in assessing the structures.

The DFFR provides guidance about the SRV actuations occurring during the normal operation of the plant and during the small, intermediate, and large break LOCA. In absence of information about relative time sequence of the three events, the following conservative assumptions for submerged structures are made in combining the effects due to these events in this assessment:

a. Short-term LOCA and SRV loads

   For structures submerged in the pool, the direct hydrodynamic loads due to the LOCA jet, LOCA bubble, pool swell, and fallback are not combined with the direct hydrodynamic loads due to the SRV water jet or air bubbles. The presence in the pool of air bubbles from the SRV line is assumed to have negligible effect on short-term pool dynamics during a LOCA. The seismic effects are combined with the short-term LOCA load effects; and

b. Long-term LOCA and SRV loads

   For all miscellaneous submerged piping systems and major structures (columns, downcomer bracing, downcomers and SRV line with quencher) the worst case LOCA steam condensation loads are combined with seismic effects and with a single adjacent SRV actuation load or with the actuation of ADS valves.

3A.3.5 LOAD COMBINATIONS AND ACCEPTANCE CRITERIA

Load combinations and acceptance criteria for events which include suppression pool hydrodynamic loads are described in this section. Four categories of structural components affected by these events are identified and the applicable load combinations and acceptance for each category are listed. These structural categories are the steel containment structure (suppression chamber portion), reinforced concrete structures, steel structures, and piping and piping systems. Symbols representing generic load types are used in the load combinations; these symbols are defined below.

3A.3.5.1 Steel Containment Structure

3A.3.5.1.1 Definitions

D Dead loads

L Live loads

E_O Loads generated by operating basis earthquake

E_{SS} Loads generated by safe shutdown earthquake

H Loads associated with pool swell phenomenon following L the clearing of the downcomer vents including fallback

P Containment pressure associated with the large break A (DBA) LOCA

P_B Containment pressure associated with IBA or SBA

P Loads associated with chugging phenomena

P_E Design external pressure on the containment

P_o Normal operating pressure

P_{SR} Loads associated with main steam SRV actuation

P_V LOCA related hydrodynamic loads on suppression chamber, including H_L, P_c

R_A Pipe reactions under thermal conditions generated by the postulated accidents and including R_o

R_E Pipe reactions under thermal conditions during event causing external pressure

R_O Pipe reactions during startup, normal operating or shutdown conditions, based on the most critical transient or steady-state condition

R_R Reaction and jet forces associated with the pipe break

T_A Thermal loads under thermal conditions associated with LOCA

T_E Thermal loads under thermal conditions during event causing external pressure

T_O Thermal effects and loads during normal operation

3A.3.5.1.2 Load Combinations

The following load combinations for the CGS steel containment are in agreement with those specified in the NRC Standard Review Plan, 3.8.2, Revision 0, and properly include the new SRV and LOCA hydrodynamic loads.

(1) $D + L + P_O + T_O + R_O + P_{SR}$

(2) $D + L + E_O + P_O + T_O + R_O + P_{SR}$

(3) $D + L + E_O + T_A + R_A + (P_A \text{ or } P_B) + P_V + P_{SR}$

(4) $D + L + E_O + T_E + R_E + P_E + P_{SR}$

(5) $D + L + E_{ss} + P_O + T_O + R_O + P_{SR}$

(6) $D + L + E_{ss} + T_A + R_A + (P_A \text{ or } P_B) + P_V + P_{SR}$

(7) $D + L + E_{ss} + T_E + R_E + P_E + P_{SR}$

Notes:

1. In all combinations the hydrostatic pressure due to the presence of water in the pressure-suppression chamber pool is considered with dead and live loads.

2. Restraint due to the presence of filler material between the containment vessel and the biological shield wall (equivalent to an external pressure of 2 psi) is considered where critical.

3. For independent, short duration, vibratory loads such as seismic, SRV discharge, chugging loads, the peak dynamic responses due to the individual loads are combined by the square-root-of-sum-of-the-squares (SRSS) method.

4. For time relationship between concurrently applied loads in combinations (3) and (6) see Section 3A.3.4.

5. In the case of P_{SR} and P_C both the axisymmetric and nonaxisymmetric loading conditions are investigated.

6. Maximum equivalent static pressures for P_A, P_B, and P_V are given in Table 3A.3.5-1.

The design assessment of the containment vessel is made on the basis of the SRSS method as stated above. However, subsequent investigation indicates that if combination of peak dynamic responses due to seismic, SRV discharge loads, and chugging loads is done by the absolute sum method, the resultant design margins for the containment vessel are greater than 1.0.

3A.3.5.1.3 Acceptance Criteria

The design rules for the steel containment are in accordance with ASME Code Section III, 1971 Edition through the 1972 Summer addenda, Subsection NE, Class MC Components. The acceptance criteria for each load combination are summarized in Table 3A.3.5-2.

3A.3.5.2 Reinforced-Concrete Structures

Structures to which the criteria below apply include the basemat, the RPV pedestal, the columns supporting the diaphragm floor, and the diaphragm floor slab.

3A.3.5.2.1 Definitions

D Dead loads

E_O Loads generated by operating basis earthquake

E_{SS} Loads generated by safe shutdown earthquake

H_L Hydrodynamic forces associated with pool swell phenomenon following the clearing of the downcomer vents including fallback forces

L Live loads

P_A All loads associated with the large break (DBA) LOCA including drywell and suppression chamber transient pressure loads and H_L, P_c, and P_{co} as defined herein

P_B All loads associated with IBA or SBA type of LOCA including transient pressure loads and P_c

P_c Loads associated with chugging phenomena during LOCA

P_O Normal operating pressure

P_{SR} Loads associated with main steam SRV actuation

R_A Pipe reactions under thermal conditions generated by the postulated accidents and including R_O

R_O Pipe react ions during startup, normal operating or shutdown conditions, based on the most critical transient or steady-state condition

R_R Reaction and jet forces associated with the pipe break

T_A Thermal loads resulting from thermal conditions generated by postulated accidents and including T_O

T_O Thermal effects and loads during normal operation.

3A.3.5.2.2 Load Combinations

The load combinations for the basemat and for the reinforced concrete structures internal to the containment are listed in Table 3A.3.5-3. The following notes are applicable to Table 3A.3.5-3.

a. In combinations 4, 4a, 5, 5a, 7, and 7a, the maximum values of P_A, P_B, T_A, R_A, and P_{SR} including a DLF are used unless a dynamic analysis is performed.

b. Thermal loads may be neglected when it can be shown that they are secondary and self-limiting in nature.

c. All the loads listed are not necessarily applicable to all the structures.

d. For independent short duration vibratory loads such as seismic, SRV discharge loads and chugging loads, the peak dynamic responses are combined by the SRSS method. Also, peak responses due to SRV direct pressure loads and due to SRV building motion response spectra are combined by SRSS.

e. The design assessment of the structures of Section 3A.3.5.2 is made on the basis of the preceding notes. However, subsequent investigation indicates that if combination of peak dynamic responses due to seismic, SRV discharge loads and chugging loads is done by the absolute sum method, the resultant design margins for the structures of Section 3A.3.5.2 are greater than 1.0.

3A.3.5.2.3 Acceptance Criteria

For all load combinations in Table 3A.3.5-3, the allowable limit on section strength is the section strength required to resist design loads based on the strength design methods described in ACI 318-77 (Reference 3A.3.5-1).

3A.3.5.3 Steel Structures

Structures to which the criteria below apply include the downcomer bracing system, the diaphragm floor beams, and platforms and ladders attached to the containment shell.

3A.3.5.3.1 Definitions

Definitions of load symbols in Table 3A.3.5-4 are the same as those in 3A.3.5.2.1.

3A.3.5.3.2 Load Combinations

The load combinations for steel structures internal to the containment are listed in Table 3A.3.5-4. Notes a, b, c, and d listed for Table 3A.3.5-3 in 3A.3.5.2.2 are also applicable to Table 3A.3.5-4.

The design assessment of the structures of 3A.3.5.3 is made on the basis of the preceding notes. However, subsequent investigation indicates that if combination of peak dynamic responses due to seismic, SRV discharge loads and chugging loads is done by the absolute sum method, the resultant design margins for the structures of 3A.3.5.3 are greater than 1.0.

3A.3.5.3.3 Acceptance Criteria

The allowable limits for structural acceptance for the load combinations of Table 3A.3.5-4 using the elastic working stress method are defined as follows.

Combination	Limit
1	S
2,3	1.5S
4,4a,5,5a,6	1.6S
7,7a	1.7S

In the above, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC Specification (Reference 3A.3.5-2). The 33% increase in allowable stresses due to seismic loadings as allowed by AISC specification is not permitted in conjunction with the strength limits above.

The allowable limits for structural acceptance for the load combinations of Table 3A.3.5-4 using the plastic design method are defined as follows:

Combination	Limit
1,2,3	Y
4,4a,5,5a,6,7,7a	0.9Y

In the above, Y is the section strength required to resist design loads based on the plastic design methods described in Part 2 of the AISC Specification (Reference 3A.3.5-2).

3A.3.5.4 Piping Systems

3A.3.5.4.1 Definitions

The loads for the piping components are: dead weight, seismic, and loads associated with SRV actuation and LOCA effects. The SRV and LOCA loads have been described in detail in previous sections of this report. A description of the symbols as they appear in the piping and component load combination table (Table 3A.3.5-5) follows:

Load Symbol	Load Description
P	Operating pressure
D.W.	Dead weight
OBE	Loads due to operational basis earthquake
SSE	Loads due to safe shutdown earthquake

SRV	Loads due to sequential pressure setpoint actuation of all (18) SRVs a. SRV bubble loads on submerged piping and components in the suppression pool b. Building motion induced loads.
SBA/IBA	Loads associated with SBA/IBA: a. Chugging/CO loads on submerged structures. (Chugging bounds CO loads.) b. Building motion due to chugging loads. (Chugging bounds CO loads.)
DBA	Loads associated with DBA: a. Water jet b. LOCA bubble c. Pool swell d. Fallback e. Chugging loads on submerged structures. (Chugging bounds CO loads.) f. Building motion induced loads due to CO and chugging. (Chugging bounds CO.)

3A.3.5.4.2 Load Combinations

Load combinations for the loads listed in Section 3A.3.5.4.1 are given in Table 3A.3.5-5. These load combinations are based on Table 6.1 of the DFFR (Reference 3A.3.2-11) and modified conservatively. For independent short duration vibratory loads such as seismic, SRV discharge loads, and chugging loads, the peak dynamic responses are combined by the SRSS method as described in the DFFR. The time relationship for the loads are described in Section 3A.3.4.

3A.3.5.4.3 Acceptance Criteria

Piping and components are designed for normal, upset, emergency, and faulted plant conditions, as delineated in the Load Combination Table using the stress values for the respective normal, upset, emergency, and faulted limits as defined in the appropriate subsection of the ASME Boiler and Pressure Vessel Code (Reference 3A.3.5-3).

3A.3.5.5 References

3A.3.5-1 "Building Code Requirements for Reinforced Concrete," ACI 318-71/77.

3A.3.5-2 "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," American Institute of Steel Construction, February 12, 1969/ November 11, 1978.

3A.3.5-3 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC "Class 2 Components," American Society of Mechanical Engineers, 1971 through Winter 1973 Addenda.*

* Faulted conditions appeared for the first time in Winter 1976 addendum.

Table 3A.3.5-1

Equivalent Static Loads for Pressure Transients and
Loss-of-Coolant Accident Effects

Loading Combinations (See Section 3A.3.5.1.2)	P_A (Max) psig		P_B (Max) psig		H_L (Max) psig
	Drywell	Wetwell	Drywell	Wetwell	Vent Clearing Pressure
(1), (2), (4), (5), and (7)	-	-	-	-	-
(3) and (6)	+34	+28	+30	+25	+24

Table 3A.3.5-2

Acceptance Criteria for Containment Vessel
Allowable Stress Limits

Loading Combinations[a]	Primary Stresses: General Membrane (P_m)	Primary Stresses: Local Membrane (P_L)	Primary Stresses: Bending + Local Membrane ($P_B + P_L$)	Secondary Stresses	Peak Stresses	Buckling
(1) and (2)	S_m	1.5 S_m	1.5 S_m	3S_m	Consider for fatigue analysis	Allowable given by ASME III Section NE-3133
(3) and (4)	S_m	1.5 S_m	1.5 S_m	N/A	N/A	Allowable given by ASME III Section NE-3133
(5), (6), and (7)						
For elements not integral and continuous	S_m	1.5 S_m	1.5 S_m	N/A	N/A	Allowable given by ASME III Section NE-3133
For elements integral and continuous	The greater of 1.2S_m or S_y	The greater of 1.8S_m or 1.5S_y	The greater of 1.8S_m or 1.5S_y	N/A	N/A	120% of Allowable given by ASME III Section NE-31311

[a] For definition of loading combinations, see Section 3A.3.5.1.

Notes:

1. Thermal stresses need not be considered in computing P_m, P_L, and P_B.

2. Thermal effects are considered in:
   a. Specifying stress intensity limits as a function of temperature.
   b. Analyzing effects of cyclic operation ASME III Section NE-3222.4

Table 3A.3.5-3

Load Combinations - Reinforced-Concrete Structures

Number	Load Condition	D	L	P_O	T_O	R_O	E_O	E_{SS}	P_B	P_A	T_A	R_A	P_{SR}	R_R
	Service Load Conditions													
1	Normal w/o temperature	1.4	1.7	1.0									1.5	
2	Normal w/ temperature	1.0	1.3	1.0	1.0	1.0							1.3	
3	Normal severe environment	1.0	1.0	1.0	1.0	1.0	1.25						1.25	
	Factored Load Conditions													
4	Abnormal (IBA/SBA)	1.0	1.0						1.25		1.0	1.0	1.25	
4a	Abnormal (DBA)	1.0	1.0							1.25	1.0	1.0	1.0[a]	
5	Abnormal (IBA/SBA) severe environment	1.0	1.0				1.1		1.1		1.0	1.0	1.1	
5a	Abnormal (DBA) severe environment	1.0	1.0				1.1			1.1	1.0	1.0	1.0[a]	
6	Normal extreme environment	1.0	1.0					1.0					1.0	
7	Abnormal (IBA/SBA) extreme environment	1.0	1.0					1.0	1.0		1.0	1.0	1.0	1.0
7a	Abnormal (DBA) extreme environment	1.0	1.0					1.0		1.0	1.0	1.0	1.0[a]	1.0

[a] Single valve actuation.

Note: See Section 3A.3.5.2.2.

Table 3A.3.5-4

Load Combinations - Steel Structures

Number	Load Condition	D	L	P_O	T_O	R_O	E_O	E_{SS}	P_B	P_A	T_A	R_A	P_{SR}	R_R
	Using Elastic Working Stress Design Method - Part 1 of AISC Specs, 1969													
	<u>Service Load Conditions</u>													
1	Normal w/o temperature	1.0	1.0	1.0									1.0	
2	Normal w/ temperature	1.0	1.0	1.0	1.0	1.0							1.0	
3	Normal sev. envir.	1.0	1.0	1.0	1.0	1.0	1.0						1.0	
	<u>Factored Load Conditions</u>													
4	Abnormal (IBA)	1.0	1.0						1.0		1.0	1.0	1.0	
4a	Abnormal (DBA)	1.0	1.0							1.0	1.0	1.0	1.0[a]	
5	Abnormal (IBA) sev. env.	1.0	1.0				1.0		1.0		1.0	1.0	1.0	
5a	Abnormal (DBA) sev. env.	1.0	1.0				1.0			1.0	1.0	1.0	1.0[a]	
6	Normal ext. env.	1.0	1.0	1.0	1.0	1.0		1.0						
7	Abnormal (IBA) ext. env.	1.0	1.0					1.0	1.0		1.0	1.0	1.0	1.0
7a	Abnormal (DBA) ext. env.	1.0	1.0					1.0		1.0	1.0	1.0	1.0[a]	1.0
	Using Plastic Design Method - Part 2 of AISC Specs, 1969													
	<u>Service Load Conditions</u>													
1	Normal w/o temperature	1.7	1.7	1.7									1.5	
2	Normal w/temperature	1.0	1.3	1.0	1.0	1.0							1.3	
3	Normal sev. env.	1.0	1.0	1.0	1.0	1.0	1.25						1.25	
	<u>Factored Load Conditions</u>													
4	Abnormal (IBA)	1.0	1.0						1.25		1.0	1.0	1.25	
4a	Abnormal (DBA)	1.0	1.0							1.25	1.0	1.0	1.25[a]	
5	Abnormal (IBA) sev. env.	1.0	1.0				1.1		1.1		1.0	1.0	1.1	
5a	Abnormal (DBA) sev. env.	1.0	1.0				1.1			1.1	1.0	1.0	1.1[a]	
6	Normal ext. env.	1.0	1.0	1.0	1.0	1.0		1.0					1.0	
7	Abnormal (IBA) ext. env.	1.0	1.0					1.0	1.0		1.0	1.0	1.0	1.0
7a	Abnormal (DBA) ext. env.	1.0	1.0					1.0		1.0	1.0	1.0	1.0[a]	1.0

[a] Single valve actuation.

<u>Note</u>: See Section 3A.3.5.2.2.

Table 3A.3.5-5

Load Combinations and Acceptance Criteria for ASME Code Class 1, 2, and 3 Balance-of-Plant Piping and Equipment[a]

Load Cases	Load Combinations (1, 2, and 3)	Design Assessment Acceptance Criteria
1	P+D.W.	Normal (A)
2	$N+OBE+SRV_{ONE}$	Upset (B)
3	$N+OBE+SRV_{TWO}$	Upset (B)
4	$N+OBE+SRV_{ALL}$	Upset (B)
5	$N+OBE+SRV_{ALL}+SBA$	Emergency[b] (C)
6	$N+OBE+SRV_{TWO}+SBA$	Emergency[b] (C)
7	$N+SSE+SRV_{ADS}+SBA/IBA$	Faulted[b] (D)
8	$N+SSE+SRV_{TWO}+SBA/IBA$	Faulted[b] (D)
9	$N+SSE+SRV_{ONE}$	Faulted[b] (D)
10	$N+SSE+SRV_{TWO}$	Faulted[b] (D)
11	$N+SSE+SRV_{ALL}$	Faulted[b] (D)
12	N+SSE+DBA	Faulted[b] (D)

[a] Equipment includes pumps, valves, supports, and vessels. For bolting used in connection with the support of ASME Code Class 1, 2, and 3 components, vendor load capacity data sheets are used, or where design is by the architect engineer, stress levels are maintained less than specified minimum yield at temperature.

[b] All ASME Code Class 1, 2, and 3 piping systems which are required to function for safety shutdown under the postulated events shall meet the requirements of NRC's memorandum, "Evaluation of Topical Report - Piping Functional Capability Criteria," date July 17, 1980.

Table 3A.3.5-5

Load Combinations and Acceptance Criteria for ASME Code Class 1, 2, and 3 Balance-of-Plant Piping and Equipment[a] (Continued)

Notes:

1. As required by the appropriate subsection, i.e., NB, NC, or ND of ASME Section III, Division I. Other loads, such as thermal transient, thermal gradients, and anchor point displacement portion of the OBE or SRV, may require consideration in addition to those primary stress-producing loads listed.

2. SBA, IBA, and DBA include all event induced loads, as applicable, such as chugging, pool swell, drag loads, annulus pressurization, etc.

3. Seismic and hydrodynamic loads are combined by the SRSS technique and added to the applicable static loads.

Load Definition Legend

Normal (N)	Normal loads include internal pressure and dead weight
OBE	Operational basis earthquake loads
SSE	Safe shutdown earthquake loads
SRV_{TWO}	SRV discharge induced loads from two adjacent valves
SRV_{ALL}	The loads induced by actuation of all SRVs
SRV_{ADS}	The loads induced by the actuation of SRVs associated with the automatic depressurization system
SRV_{ONE}	The loads induced by the actuation of one SRV

3A.4 DESIGN ASSESSMENT

3A.4.1 SUPPRESSION POOL BOUNDARY STRUCTURES

3A.4.1.1 Assessment of Steel Containment Structure

The primary containment structure in the suppression chamber area, as shown in Figure 3A.4.1-1, is an orthogonally stiffened steel shell. See FSAR Section 3.8 for a description of the steel containment structure.

The thickness of the steel shell plate in the suppression chamber region is approximately 1.5 in. and varies with height. Vertical "Tee" stiffeners, at a spacing of 40.5 in., are welded to the inside face of the shell plate and extend about 8 ft beyond the knuckle (cylinder-cone inter-face) elevation. In the pool region of the suppression chamber, additional horizontal "Tee" stiffener rings at an approximate spacing of 36 in., are welded to the inside face of the shell plate. The purpose of adding these tees (Figure 3A.4.1-4) is to increase the load carrying capacity of the containment shell. The vertical tees are required for resisting compressive loads due, mainly, to seismic effects. The horizontal tees are intended to carry the hydrodynamic loads which were not considered in the original design of the containment shell. There are no heavy attachments to the containment in the suppression chamber region.

The drywell floor slab is radially separated from the containment and the gap is sealed by means of a radially and vertically flexible seal, as shown in Figure 3A.4.1-9. The slab is connected to the containment in the tangential direction by means of shear lugs.

3A.4.1.1.1 Loads Used For Assessment

The methods used for calculating hydrodynamic loads on the pool boundary, as described in Sections 3A.3.1 and 3A.3.2, provide a conservative definition of loads for design assessment.

3A.4.1.1.1.1 Safety/Relief Valve Loads. The suppression pool boundary pressure loading is determined in accordance with procedures described in Reference 3A.4.1-1 on the basis of operating conditions at CGS. A discussion on the derivation of the safety/relief valve (SRV) load definition is provided in 3A.3.1.

Several different incidents may occur which would cause one or more SRVs to actuate. For example, the valves may operate either manually, or on pressure setpoints following a turbine trip, automatically through the automatic depressurization system (ADS) system.

The critical modes of SRV actuations considered in the design are detailed in Reference 3A.4.1-1. For the purposes of design assessment, consideration is given to all the SRV discharge cases that are postulated to occur during the life of the CGS plant. A summary of the various cases is explained below.

3A.4.1.1.1.1.1 Single Valve Discharge Case. Actuation of any single SRV is postulated during a loss-of-coolant accident (LOCA) involving a large or intermediate break. Two possible cases of single SRV discharge are considered, the single inner quencher discharge and the single outer quencher discharge. A single inner quencher discharge is more likely to occur because of its lower pressure setpoint. However, a single outer quencher discharge is conservatively assumed for the assessment of the containment vessel.

3A.4.1.1.1.1.2 Two Valves Discharge Case. For this event, two SRVs are considered to discharge concurrently through two adjacent quenchers.

3A.4.1.1.1.1.3 Automatic Depressurization System Valves Discharge Case. The seven ADS valves for CGS are assigned to outer quenchers in a configuration that is nearly axisymmetrical. The ADS is characterized by an automatic and simultaneous actuation as discussed in the DFFR (Reference 3A.3.2-2). The ADS discharge is not considered to occur during a large pipe break LOCA. However, it is assumed that the ADS may discharge during an intermediate pipe break or a small pipe break LOCA.

3A.4.1.1.1.1.4 All Valves Discharge Case. Under certain plant conditions, the actuation of all 18 SRVs in CGS is assumed. Two different conditions may occur during this event: the axisymmetric all valves discharge conservatively assumes that all 18 SRVs discharge simultaneously; the nearly symmetric all valves discharge assumes that there is some imbalance during the discharge event. As discussed in the DFFR (Reference 3A.3.2-2), the all valves discharge is not considered to occur during a large pipe break LOCA. However, it is assumed that the all valves discharge may occur during an intermediate pipe break or a small pipe break LOCA.

3A.4.1.1.1.2 Loss-of-Coolant Accident Loads. Loss-of-coolant accidents are associated with postulated large pipe breaks [design basis accident (DBA)], intermediate pipe breaks [intermediate break accident (IBA)], or small pipe breaks [small break accident (SBA)]. Various transient LOCA pressure loads on the pool boundary considered in the assessment of the containment include vent water clearing jet loads, air bubble pressure loads, pool swell, fallback, drywell and wetwell pressure transients, and chugging loads.

3A.4.1.1.1.2.1 Chugging Loads. A general discussion of the chugging phenomenon is included in Section 3A.3.2.4. Design pool boundary loads discussed in Section 3A.3.2.4.2.1 are used for the structural assessment.

3A.4.1.1.1.2.2 High and Medium Mass Flux Condensation Oscillations. During the sequence of a LOCA event, condensation oscillations take place after pool swell and fallback. Depending on the steam mass flux rate, they are identified as either (a) high mass flux or (b) medium mass flux condensation oscillations. However, as noted in Section 3A.3.2.4.1.2, the controlling boundary pressure loads due to chugging exceed those due to condensation

oscillations. Therefore, condensation oscillation loads are not considered in the assessment of the containment vessel.

3A.4.1.1.1.2.3 Other Loss-of-Coolant Accident Loads. Loss-of-coolant accident loads, other than those due to condensation oscillations and chugging, include

a. Pressure and temperature transients

These transients represent symmetric loadings. An equivalent static loading is used for a LOCA pressure transient which takes due account of the time history of the pressure buildup. Thermal effects (T_A) on stress intensities and cyclic operation have been considered. Pressure and temperature transients considered in the assessment are shown in Figures 3A.3.2-20 through 3A.3.2-28;

b. Reaction from downcomer vent horizontal exit load

The horizontal loads acting at the downcomer exits are transmitted via the downcomer bracing system and result in tangential reactions at the steel containment structure;

c. Pool swell bubble pressure

The air slug pressure in the suppression pool during pool swell acts symmetrically around the inside of the containment structure. Its time history, in relation to the appropriate period of the containment structure, is used in determining the dynamic load factor (DLF).

In addition to the preceding case of symmetric loading, the case of an asymmetric bubble pressure acting on the submerged boundary in accordance with Reference 3A.3.2-5 is also included; and

d. Reactions from components supported by the steel containment structure

Drag and impact loads occur on structural components supported by the steel containment structure as a result of pool swell and fallback. Reactions from these structural components are carried by the containment structure.

3A.4.1.1.1.3 Other Significant Loads.

a. Seismic loads

Loads due to the operating basic earthquake (OBE) and the safe shutdown earthquake (SSE), developed in the project design, are applicable. Seismic

loads include the effect of the water in the suppression pool. The effects due to water sloshing (pressure load) have been accounted for in the containment pool swell assuming a concurrent seismic event (SSE) is insignificant;

b. Dead load, live load, and hydrostatic pressure

The hydrostatic pressure due to the suppression pool is included in the dead load; and

c. Design external pressure

An external pressure of 2 psi resulting from atmospheric conditions inside and outside the drywell is used. When external pressure governs the design, an additional external pressure of 2 psi due to the reaction of the compressible foam between the containment and the biological shield wall is used. This reaction results from the thermal expansion of the containment shell.

3A.4.1.1.2 Controlling Load Combinations

The applicable load combinations for the pressure-suppression chamber portion of the steel containment structure are defined in Section 3A.3.5.1.2. Combinations presented therein are also applicable to the horizontal and vertical stiffening tees. Load combination (3), stated below, is found to control the design of the stiffened steel containment structure.

Load Combination (3): $D + L + E_O + T_A + R_A + P_B + P_V + P_{SR}$

The interpretation and contribution of each of the terms depends on the event being considered. In considering the overall steel containment structure, the controlling combination of events involves ADS actuation during an intermediate break LOCA with clugging. Consequently, P_B and T_A refer to the IBA. In this load combination, T_A and R_A are relatively insignificant. The term P_{SR} refers to ADS pool boundary pressure loading. Thus, the effective controlling load combination involves:

$$D + L + E_O + P_B + P_{SR} + P_C$$

3A.4.1.1.3 Acceptance Criteria

The acceptance criteria for design of steel containment structure, including horizontal and vertical tees, is in compliance with the 1971 ASME Code, Edition through the 1972 Summer Addenda, Section III, Division 1, Subsection NE is given in Section 3A.3.5.1.3.

3A.4.1.1.4 Method of Analysis

3A.4.1.1.4.1 Formulation of the Problem. In accordance with the methods presented in Sections 3A.3.1 and 3A.3.2 for defining hydrodynamic loads, it is assumed that the incident pressures on the pool boundary, resulting from bubble oscillations and steam condensation, are acting on the boundary as externally applied loads. In computing the responses of the structure, the fluid-shell interaction effects are accounted for by solving the coupled partial differential equations governing the fluid and the shell structure using finite elements. The stresses due to chugging used for assessing the containment structure are obtained by using the building model and the method of analysis presented in Section 3A.5.2. For SRV loads, a refined containment model described in Section 3A.5.1 is used.

For pool boundary hydrodynamic loads, since the vertical and horizontal tees are integrated in the model of the suppression chamber portion of the containment, the fluid-shell interaction analysis also gives the dynamic stresses in the tees.

3A.4.1.1.4.2 Mathematical Model. The mathematical model used in the containment analysis for SRV loads is discussed in Section 3A.5.1.1.2 and shown in Figure 3A.5.1-2. Prominent features of this axisymmetric model are discussed in Reference 3A.4.1-1. The mathematical model used in the containment analysis for chugging loads is discussed in Section 3A.5.2.1 and shown in Figure 3A.5.2-1. A detailed description of the chugging model can be found in Reference 3A.4.1-4.

3A.4.1.1.4.3 Coupled Equations of Motion.

a. Shell equations

   The partial differential equations governing the motion of the containment shell are based on equations given in Reference 3A.4.1-3;

b. Fluid equations

   The partial differential equations governing the dynamics of compressible fluid are the continuity equation and the equation of motion establishing the relationship between the pressure and the velocity of a particle in the fluid. For reasons cited in Reference 3A.4.1-4, the water in the suppression pool is considered to be compressible for the chugging model. However, for reasons cited in Reference 3A.4.1-1, the water is considered to be incompressible for the SRV model;

c. Fluid boundary conditions

The fluid boundaries can be seen in Figure 3A.4.1-8. The pressure at the fluid surface is specified to be zero. The continuity requirement at the fluid-structure interface is satisfied by specifying the radial component of fluid motion at the interface to be equal to that of the shell;

d. Shell boundary conditions

As discussed in Section 3A.5.1.1.2, a refined containment model is used for the analysis of the containment structure under SRV loading. This refined model is connected to the overall building model at the following locations (see Reference 3A.4.1-1): basemat in the radial, vertical, and circumferential directions; diaphragm floor in the circumferential direction; and stabilizer truss and refueling bellows in the radial and circumferential directions.

For the analysis of the containment structure due to chugging loads, the overall building model discussed in Section 3A.5.2.1, is used;

e. Geometric symmetry

The axisymmetric geometry of the containment shell-fluid system is utilized in the solution of the equations in cylindrical coordinates. The azimuth coordinate is eliminated from the governing equations by representing azimuthal dependence of each variable by a Fourier series. The equations are thus solved for each Fourier term or harmonic of the series and the final solution is obtained by summation of solutions for each term.

3A.4.1.1.4.4 Numerical Solution. Numerical solutions to equations described in Section 3A.4.1.1.4.3 are obtained by using finite elements. An integration time step of 0.001 sec is used in the analyses for both SRV and chugging loads.

3A.4.1.1.4.5 Computer Program. A Burns and Roe computer program "HYDI-2" (Attachment 3A.F) was developed for Mark II containment configurations, and subsequently used in the analysis of the containment structure due to chugging loads. The numerical solutions were verified by the commercially available finite element program "NASTRAN" (Attachment 3A.F). The analysis of the refined containment model for SRV loads was made with the "NASTRAN" program.

3A.4.1.1.5 Results and Design Margin

3A.4.1.1.5.1 Results of Analysis. The hydrodynamic pressure loads, as described in Section 3A.4.1.1.1, are applied to the containment wall of the fluid-shell interaction models

(Figures 3A.5.1-2 and 3A.5.2-1). Utilizing the Burns and Roe, Inc. computer program HYDI-2 for chugging loads, and the NASTRAN program for SRV loads, the responses of the containment are computed. The maximum stresses (in time) are evaluated in the applicable load combinations for determining the controlling load combination and corresponding design margin for the containment structure. Maximum time-wise profiles of radial displacements are presented in Figures 3A.4.1-2 and 3A.4.1-3 for the containment structure.

Responses to various loads discussed in Section 3A.4.1.1.1 are summarized below:

a. ADS discharge case

   The ADS actuation combined with IBA or SBA is the most critical axisymmetric pressure load on the containment. For the purposes of assessment, the ADS pressure loading is conservatively assumed to be the same as the larger of each of the two all valves discharge pressure loadings, i.e., the design boundary pressure. Thus, the response of the containment to the ADS actuation is shown in Figure 3A.4.1-2. The resulting stresses are used in the controlling load combination for calculating the design margin of the containment structure;

b. All valve discharge case

   In considering the two different all valves discharge events, the nearly symmetric pressure loading is slightly greater than the axisymmetric pressure loading (see References 3A.4.1-1 and 3A.4.1-2);

c. Single valve discharge case

   Actuation of a single SRV combined with LOCA (DBA) loads results in the most critical non-axisymmetric pressure loading on the containment. However, responses of the containment to the resulting load combination is less severe than case (a), above;

d. Two valves discharge case

   Responses of the containment to this load are conservatively assumed to be the same as case (c) (see Reference 3A.4.1-1);

e. Chugging

   Reference 3A.4.1-4, described in Section 3A.3.2.4.2.2, presents the responses of the containment shell to the chugging pressure load. The resulting stresses are used in the controlling load combination for calculating the design margin of

the containment structure. As discussed in Section 3A.5.2.2, the nearly symmetric chugging load is used for assessment purposes. The response of the containment to these design chugging loads is shown in Figure 3A.4.1-3; and

f. High and medium mass flux condensation oscillations

Condensation oscillation loads are not considered in the assessment of the containment structure since they are bounded by chugging loads.

3A.4.1.1.5.2 Assessment Results. The containment assessment performed in accordance with Sections 3A.3.5.1.2 and 3A.3.5.1.3 shows that in load combination (3) the general membrane stress intensity controls the containment design. Based on this calculation, the design margin for the containment shell is 1.29.

The buckling strength of the CGS containment in resisting the external pressure and axial compression acting on the suppression pool region increases substantially as a result of adding the horizontal stiffening rings. The most critical load combination under events causing net external pressure and axial compression is load combination (7). Under this load combination the ratio of the allowable buckling pressure load of the containment to the applied external pressure load is 3.1. The ratio of the allowable buckling axial load of the containment to the axial compressive load is 1.37. In this analysis, the interaction effect is accounted for by assuming that the horizontal stiffening tees resist only external pressure while the vertical stiffening tees resist only axial compression.

For the containment tees, the stress intensities for load combination (3) are governed by primary bending plus local membrane stresses. These stresses occur at the webs of the horizontal tees and at the root of the flange for the vertical tees. Based on these stress values, the design margins for the vertical and horizontal tees are 2.23 and 1.26, respectively.

The combined stresses in both the containment and tees are calculated for the controlling load combination by adding the stress resulting from static loads algebraically and stresses due to dynamic oscillating loads by the square-root-of-sum-of-the-squares (SRSS) method. The resulting design margin for the containment structure (including the horizontal and vertical tees) is 1.26.

3A.4.1.2 Basemat

The assessment of the capacity of the basemat relative to load combinations involving suppression pool hydrodynamic loads is made in this section. The basemat and adjoining structures are shown in Figures 3A.4.1-5 and 3A.4.1-6.

3A.4.1.2.1 Loads Used for Assessment

A complete description of all the hydrodynamic loads used in the assessment of the basemat is provided in Section 3A.3. Symbols, equations, and load combinations referred to in this section are detailed in Section 3A.3.5.2.

3A.4.1.2.1.1 Safety/Relief Valve Loads. Loads on the suppression pool boundary due to SRV actuations are detailed in Section 3A.3.1. Specific SRV loading cases considered in the basemat assessment include symmetric loads due to the actuation of all 18 valves and asymmetric loads due to the actuation of a single SRV. For both cases, dynamic stresses in the basemat (bending and shear) are developed on the basis of a time-history application of the loads. The analytical model used for the assessment is shown in Figure 3A.5.1-lb. Prominent features of the model, including the use of axisymmetric shell elements is discussed in Section 3A.5.1.

3A.4.1.2.1.2 Loss-of-Coolant Accident Loads. Loads on the suppression pool boundary due to a LOCA are detailed in Section 3A.3.2. Of all the LOCA loads, chugging pressures are the most significant with respect to the basemat. Other LOCA loads, including jet loads and bubble pressures, are not significant with respect to the basemat assessment. Dynamic stresses in the basemat (bending and shear) are developed on the basis of a time-history application of the loads. The analytical model used for the assessment is shown in Figure 3A.5.2-1. Prominent features of the model, including the use of axisymmetric shell elements, are discussed in Section 3A.5.2.

3A.4.1.2.1.3 Other Significant Loads. Seismic loads constitute a principal loading in the basemat assessment. The seismic loads on the basemat from the superstructure (exterior walls, biological shield wall, and pedestal) are the same as those used in the original building design. In that original design, a dynamic analysis was made using a discrete mathematical idealization of the entire reactor building. The stress resultants at the base of the superstructure (overturning moment, axial force) due to the OBE and the SSE as developed in the original design are used.

Dead and live loads as developed in the original structural design are also used.

3A.4.1.2.2 Applicable Load Combinations and Acceptance Criteria

The load combination and acceptance criteria described in Section 3A.3.5.2 are applicable to the basemat.

3A.4.1.2.3 Method of Analysis

The structural capacity of the basemat is investigated for the applicable load combinations with loads as listed above. The general approach in the basemat assessment is to determine the

values of the controlling stress resultants in the basemat (bending and shear) on the basis of elastic analysis under applied design loads and to calculate the capacity of the basemat in terms of these stress resultants by the strength method of the ACI 318-71 Code (Reference 3A.4.1-5). Critical sections for bending and shear are located with respect to the face of the biological shield wall in compliance with code requirements.

3A.4.1.2.3.1 Effects of E_O, E_{SS}, D, L. The investigation of the basemat for the combined effect of dead, live, and seismic loads is based on the analysis performed in the original building design. Depicted as a plate on an elastic foundation, the basemat is modeled as a series of plate elements while the supporting soil is modeled as a group of elastic springs situated at designated nodes. The effects of the seismic overturning moment and the vertical acceleration of the dead and live loads are converted to nodal loads. Resulting stresses from the model and loads described above are calculated with the use of the computer program NASTRAN. Values of the controlling bending moment, beam shear, and punching shear due to combined dead, live, and SSE loads are tabulated in Table 3A.4.1-1 for the critical section of the basemat.

3A.4.1.2.3.2 Effect of P_{SR}, P_B. The values of bending moments and shears at the critical section in the basemat due to SRV actuation and chugging are available from the analysis of the reactor building models described in Section 3A.5. The symmetric mode of SRV actuation and the nearly symmetric mode of chugging result in the comparatively larger values of stress resultants. The controlling stress resultants for these loads are tabulated in Table 3A.4.1-1 for the critical section.

3A.4.1.2.3.3 Critical Load Combination. Review of the stress resultant values in connection with the applicable load combinations shows that the critical load combination for all stress resultants is load combination (7); this combination is noted below with only the significant terms included.

(7) $D + L + E_{SS} + P_B + P_{SR}$

3A.4.1.2.3.4 Capacity. The capacity of the basemat with respect to bending, beam shear, and punching shear is determined in accordance with the strength method of the ACI 318-71 Code (Reference 3A.4.1-5). These stress resultant capacities are listed in Table 3A.4.1-1.

3A.4.1.2.4 Results and Design Margins

Comparison of the design values for the stress resultants in Table 3A.4.1-1 with the capacity values in the table shows that the basemat provides adequate capacity. The ratio of bending capacity to design bending moment is 1.14. The ratio of beam shear capacity to design beam shear is 1.48. The ratio of punching shear capacity to design punching shear is 1.27.

3A.4.1.3 Pedestal

The assessment of the capacity of the reactor pressure vessel (RPV) pedestal relative to load combinations involving suppression pool hydrodynamic loads is made in this section. The pedestal and adjoining structures in the suppression chamber are shown in Figure 3A.4.1-7.

3A.4.1.3.1 Loads Used for Assessment

A complete description of all the hydrodynamic loads used in the assessment of the RPV pedestal is provided in Section 3A.3. Symbols, equations, and load combinations referred to in this section are detailed in Section 3A.3.5.2.

3A.4.1.3.1.1 Safety/Relief Valve Loads. Loads on the suppression pool boundary due to SRV actuations are detailed in Section 3A.3.1. Specific SRV loading cases considered in the pedestal assessment include symmetric loads due to the actuation of all 18 valves and asymmetric loads due to the actuation of a single SRV. For the asymmetric case, dynamic stresses in the pedestal are developed on the basis of time history application of the load. For the symmetric case, dynamic stresses are developed on the basis of applied pressures increased by an appropriate DLF which is determined from the time history analysis. The analytical model used for the assessment is shown in Figure 3A.5.1-lb. Prominent features of the model, including the use of axisymmetric shell elements, are discussed in Section 3A.5.1.

3A.4.1.3.1.2 Loss-of-Coolant Accident Loads. Loads on the suppression pool boundary due to LOCA are detailed in Section 3A.3.2. Of all the LOCA loads, chugging pressures are the most significant with respect to the pedestal. Other LOCA loads, including pool swell, jet loads, and bubble pressures are not significant with respect to the pedestal assessment. Dynamic stresses in the pedestal are developed on the basis of the model and load application described in Section 3A.4.1.2.1.2.

3A.4.1.3.1.3 Other Significant Loads. Seismic loads (E_o, E_{ss}) constitute a principal loading in the pedestal assessment. The seismic loadings and associated analysis in this assessment are the same as those used for the original design. A dynamic analysis was made using a discrete mathematical idealization of the entire reactor building including the pedestal. The stress resultants in the pedestal (overall bending moment, horizontal shear force, and axial force) due to the OBE (E_o) and the SSE (E_{ss}) as developed in the original design are used. Dead loads as developed in the original building design are also utilized.

3A.4.1.3.2 Applicable Load Combinations and Acceptance Criteria

The load combinations and acceptance criteria for internal reinforced concrete structures described in Section 3A.3.5.2 are applicable to the pedestal.

3A.4.1.3.3 Method of Analysis

The structural capacity of the pedestal is investigated for the load combinations with two types of loading, namely, asymmetric and symmetric. The general approach in the pedestal assessment is to determine the values of the controlling stress resultants on the basis of elastic analysis under design loadings and to calculate the capacity of the pedestal in terms of these stress resultants in accordance with the strength method of the ACI 318-71 Code (Reference 3A.4.1-5).

3A.4.1.3.3.1 Asymmetric Action. The loads which contribute to asymmetric action of the pedestal are seismic loads (E_O and E_{SS}), loads due to single SRV actuation (P_{SR}), and loads due to chugging phenomena (P_B). The significant stress resultants associated with these loads are overturning moment and total shear. For seismic loading, the values of these stress resultants in the original design are used. For P_{SR} and P_B, the stress resultants are obtained by integrating over the entire pedestal section the stresses obtained from the elastic analysis of the reactor building structural model. Controlling values of the stress resultants which occur at the base of the pedestal are tabulated in Table 3A.4.1-2.

Review of the stress resultant values in connection with the load combinations shows that the critical load combination for both moment and shear is load combination (7), stated below, with only the significant load terms included.

(7) $D + E_{SS} + P_B + P_{SR}$

The capacity of the pedestal relative to overturning moment and concurrent axial load is expressed by the interaction curve shown in Figure 3A.4.1-10. Points along the interaction curve representing different capacity combinations of axial load (P_u) and bending moment (M_u) have been calculated in line with the ACI 318-71 Code (Reference 3A.4.1-5). The minimum and controlling value of axial load occurs with upward seismic action; this axial load value (12,380 kips) and the controlling overturning moment from Table 3A.4.1-2 (212,380 ft kips) are also plotted in Figure 3A.4.1-10. As noted in the figure, the bending moment capacity coincident with the controlling axial load is 375,000 ft kips.

The capacity of the pedestal relative to overall horizontal (tangential) shear, calculated in accordance with the ACI 318-71 Code (Reference 3A.4.1-5), is 14,500 kips. From Table 3A.4.1-2, the controlling design shear is 2760 kips.

3A.4.1.3.3.2 Symmetric Action. Loads which contribute to symmetric action of the pedestal are due to actuation of all 18 SRVs (P_{SR}) and chugging phenomena (P_B). The hydrostatic pressure (D) also causes symmetric action. For P_{SR} and P_B, an appropriate DLF is included as noted in Section 3A.4.1.3.1.1. Symmetric action is investigated with respect to radial and circumferential normal stresses and with respect to the effect of the end fixity at the base.

To obtain the radial and circumferential normal stresses, the pedestal is analyzed as a thick walled cylinder. Maximum compressive stress (circumferential) occurs for load combination (1):

(1) $1.4D + 1.7L + 1.0P_O + 1.5P_{SR}$

In this load combination, terms L and P_O do not contribute to the stresses being considered and are omitted. For calculation purposes, the maximum values of D, and P_{SR} (positive value) are used. Maximum tensile stress (circumferential) occurs for load combination (4):

(4) $1.0D + 1.0L + 1.0T_A + 1.0R_A + 1.25 (P_B + P_{SR})$

In this load combination, L, T_A, and R_A are omitted as they do not affect the stress resultant under consideration.

Radial shear and moment occur under the symmetric loads due to the fixity of the pedestal at its junction with the basemat. The analysis is based on a general theory of the elastic behavior of cylindrical shells. Maximum values of radial shear and moment occur at the pedestal base with load combination (4).

In this load combination, L, T_A, and R_A are omitted as they do not affect the stress resultant under consideration.

3A.4.1.3.4 Results and Design Margins

Results for asymmetric loading are summarized below:

a. The controlling value of the pedestal overturning moment under design loadings is less than the pedestal moment capacity. The ratio of the moment capacity to the controlling overturning moment is 1.77; and

b. The controlling value of the pedestal base shear under design loadings is less than the pedestal shear capacity. The ratio of shear capacity to controlling applied shear is 5.25.

Results for symmetric loading are summarized below:

a. The calculated normal stresses occurring during symmetric action are found to be less than the allowable strength values. The ratio of pedestal capacity to stress under controlling loading is 5.53 for tensile circumferential stress and 15.83 for circumferential compressive stress; and

b. The calculated stresses due to radial shear and moment due to pedestal fixity at its base during symmetric action are found to be less than the allowable strength values. The ratio of radial shear capacity to maximum shear due to load is 1.11 and the corresponding ratio for radial moment is 5.06.

Review of the preceding results shows that the overall controlling design margin is 1.11 applicable to radial shear under symmetric loading.

3A.4.1.4 Diaphragm Floor

Assessment of the capacity of the diaphragm floor (see Figures 3A.4.1-6 and 3A.4.1-7) relative to load combinations involving suppression pool hydrodynamic loads is made in this section.

3A.4.1.4.1 Loads Used for Assessment

A complete description of all hydrodynamic loads is given in Section 3A.3. This subsection discusses the loads used for the assessment of the diaphragm floor.

3A.4.1.4.1.1 Safety/Relief Valve Actuation Loads. Safety/relief valve discharge does not result in pressure loads directly on the diaphragm floor, but causes dynamic horizontal pressure differentials across the downcomers, the SRV piping, and the columns, all of which are supported at the diaphragm floor, and dynamic vertical pressure differential across the downcomer bracing which is transferred to the diaphragm floor by the downcomers.

In addition, building response spectra from SRV discharge result in acceleration of the diaphragm floor which induces dynamic stresses in the components of the floor.

3A.4.1.4.1.2 Loss-of-Coolant Accident Loads. The maximum net downward pressure on the diaphragm floor during a DBA LOCA is 20 psi (see Section 3A.3.2.5.2). Since the time required to develop the maximum net downward differential pressure resulting from a recirculation line break is approximately 0.7 sec, dynamic effects are not significant and temperature transients at time of peak downward pressure differential do not contribute to the floor loading.

The maximum net upward pressure on the diaphragm floor is of short duration, and is due to wetwell atmosphere compression resulting from pool swell during a DBA LOCA (see Section 3A.3.2.5.2). A value of 5.5 psi maximum net upward pressure is considered in the assessment of the diaphragm floor (see Section 3A.3.2.3.1.3).

Pool swell and fallback drag loads on the downcomer bracing (see Section 3A.3.2.3.3.1) are transferred to the diaphragm floor by the downcomers.

Other significant LOCA loads include pipe break jet impingement and steam condensation accelerations obtained from the response spectra at the diaphragm floor support locations.

3A.4.1.4.1.3 Other Significant Loads. Other loads which result in significant stresses in the diaphragm floor are dead loads, live loads, and vertical seismic accelerations.

Dead leads include the weight of the diaphragm floor reinforced concrete slab and supporting steel beams, downcomers, horizontal run of SRV piping, including vertical supports, downcomer bracing supported vertically by the downcomers and, hence, the diaphragm floor. Live loads include personnel and equipment weights on the diaphragm floor. Seismic accelerations are obtained from the seismic response spectra at the support locations for the diaphragm floor.

3A.4.1.4.2 Controlling Load Combinations

The load combination criteria for structures internal to the pressure-suppression chamber (see Sections 3A.3.5.2 and 3A.3.5.3) are applicable to the diaphragm floor. In particular, the combinations for steel structures, using the elastic working stress design method with both service load conditions and factored load conditions, are investigated in the analysis for the structural steel components of the floor (see Section 3A.3.5.3). The combinations for reinforced concrete structures, using the ultimate strength design method with both service load conditions and factored load conditions, are investigated for the reinforced concrete slab component of the floor (see Section 3A.3.5.2). The controlling load combinations are specified in Section 3A.4.1.4.5.

3A.4.1.4.3 Acceptance Criteria

The acceptable stress levels for the steel components of the diaphragm floor are specified in Section 3A.3.5.3.3.

The acceptable allowable limit for the concrete components is the ultimate strength as determined by the ultimate strength design method of the ACI 318-71 Building Code (Reference 3A.4.1-5).

3A.4.1.4.4 Method of Analysis

The diaphragm floor components, consisting of the reinforced concrete slab, the structural steel circumferential and radial beams, and connections, were investigated individually for the effects of both the upward and downward loads. To determine design loads for each of the components, the diaphragm floor was analyzed as a slab (one-way); beam (circumferential beams), and girder (radial beams) structural system with the radial beams supported at the pedestal and on the columns, but not at the containment vessel shell.

3A.4.1.4.5 Results and Design Margins

The diaphragm floor reinforced concrete slab and the steel circumferential and radial beams, including connections, were found to have sufficient capacity to withstand the governing load combinations. The critical component under downward load, as defined by the governing load combination (7a) for steel structures, is the radial beam. The design margin (i.e., ratio of the allowable stress to the maximum absolute calculated stress) for the radial beam is 1.62, and is based on the following input into the loading combination (7a) for steel structures:

(7a) $1.7S \geq 1.0D + 1.0L + 1.0E_{SS} + 1.0P_A + 1.0T_A + 1.0R_A + 1.0P_{SR} + 1.0R_R$

D = 4.28 psi

L = 6.94 psi

E_{SS} = 1.52 psi

P_A = 20 psi

T_A = 0

R_A = 0

P_{SR} = 1.65 psi

R_R = 534,000 lb

where D, L, E_{SS}, P_A, T_A, R_A, P_{SR}, and R_R are as defined in Section 3A.3.5.2.

The critical components under upward load are the anchor bolts at the radial beam to column connection. The design margin (i.e., ratio of the allowable stress to the maximum absolute calculated stress) far the anchor bolts is 1.27, based on the following input into the governing loading combination (4a) for reinforced concrete structures:

(4a) $U \geq 1.0D + 1.0L + 1.25P_A + 1.0T_A + 1.0R_A + 1.0P_{SR}$

D = 4.28 psi

L = 0

P_A = 5.57 psi

T_A = 0

$R_A = 0$

$P_{SR} = 1.65$ psi

where D, L, P_A, T_A, R_A, and P_{SR}, are as defined in Section 3A.3.5.2.

3A.4.1.5 Diaphragm Floor Seal

The diaphragm floor seal is located at the inside surface of the primary containment vessel periphery at el. 493 ft 5 in. It provides a flexible, pressure tight seal between the primary containment vessel and the diaphragm floor and is capable of accommodating differential thermal expansion between them. The diaphragm floor seal is a 270° omega-shaped configuration of stainless steel and is drained to the floor drain system with four drain pipes, as shown in Figure 3A.4.1-9.

3A.4.1.5.1 Loads Used for Assessment

a. Normal plant condition

This condition is defined as reactor startup, operation at power, and normal reactor cold shutdown. These loads are due to thermal expansion of the component and thermal displacement between the concrete diaphragm floor and primary containment during normal plant operation;

b. SRV loads

These loads are not directly applied to the diaphragm floor seal, but do cause displacement of the primary containment shell relative to the diaphragm floor resulting in stress in the seal;

c. LOCA loads

The LOCA combination governing the design of the seal includes the effects of relative thermal displacement, differential pressure, and hydrodynamic effects. Other LOCA effects, given in Section 3A.3.2, include the temperature and pressure transients, pool swell air compression load, and primary containment displacement due to building response. A discussion of direct load, i.e., temperature and pressure transients and pool swell phenomenon is presented in Section 3A.4.1.5.2; and

d. Other loads

The effect of dead load, OBE, and SSE seismic loads, as applicable, are included in the analysis. Loads from the drain pipes are also included.

3A.4.1.5.2 Controlling Load Combination

The controlling load combination for the diaphragm floor seal is that which includes loads due to a DBA. The load due to pool swell air compression is given in Section 3A.3.2.3. The LOCA pressure and temperature transients are described in Section 3A.3.2.5.

The following individual loads were utilized for load combinations, per ASME Code Section III, Subsection NE in the design of the diaphragm floor seal. The load combinations used are defined in Table 3A.3.5-5.

a. Normal Plant Conditions,

b. OBE,

c. SSE,

d. LOCA,

e. SRV loads with all four valve actuations,

f. Piping loads due to all dynamic loads listed above,

g. The maximum differential pressure, and

h. Relative displacement due to the movement of the primary containment vessel and diaphragm floor.

The fatigue evaluation was performed conservatively using the cycles in Table 3A.4.1-3.

3A.4.1.5.3 Acceptance Criteria

The acceptance criteria for the analysis of the diaphragm floor seal is as follows:

a. Achieving a positive margin of safety on critical elastic buckling of the seal when considering the maximum convex pressure on the seal due to hydrodynamic loads;

b. Stress based on elastic analysis of the seal is not to exceed the following:

1. Average membrane stress intensity is not to exceed the allowable values defined in ASME Code Section III 1971 through Summer 1972 Addenda Paragraph NE 3320;

c. The cumulative usage factor as defined in ASME Code Section III, 1971 through Summer 1972 Addenda, Paragraph NB 3222.4(e) is not to exceed unity.

3A.4.1.5.4 Method of Analysis

The ANSYS (see Attachment 3A.F) finite element model of the diaphragm floor seal consists of a 5.2° segment of the omega shaped configuration and the weldolet welding fitting for the drain pipe. Refer to Figure 3A.4.1-10. Unit differential pressure, unit displacements in the vertical and radially horizontal directions to represent the differential displacement of the primary containment vessel and diaphragm floor, unit piping loads at the weldolet, and a linearized thermal gradient are specified load steps in the analysis. Stresses are calculated by applying scaling factors to the unit load analyses and superimposing the results. Note that circumferential differential displacement of the primary containment vessel and the diaphragm floor in the horizontal plane is prevented by shear lugs furnished along the outer periphery of the diaphragm floor.

3A.4.1.5.5 Results and Design Margins

The differential displacements, differential pressures, and the piping loads for the critical loading combinations, as tabulated in Section 3A.4.1.5.2, were used to perform the stress analysis. The design margin on the elastic buckling of the omega seal as described in Section 3A.4.1.5.3.a is 34.7. The calculated stress intensity and fatigue values are presented in Tables 3A.4.1-4 and 3A.4.1-5 respectively.

3A.4.1.6 References

3A.4.1-1 "SRV Loads - Improved Definition and Application Methodology for Mark II Containments," technical report, Burns and Roe, Inc. Transmitted to NRC by letter GO2-80-172 dated August 8, 1980.

3A.4.1-2 Letter GO2-82-35, "Responses to CSB Open Items 44 through 48," G. D. Bouchey (WPPSS) to A. Schwencer (NRC), January 13, 1982.

3A.4.1-3 Gosh, S. and Wilson, E., "Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading," University of California at Berkeley, EEEC, 69-September 10, 1969.

3A.4.1-4 "Chugging Loads - Revised Definition and Application Methodology for Mark II Containments (Based on 4TCO Test Results)," technical report, Burns and Roe, Inc. Transmitted to NRC by letter GO2-81-189 dated July 22, 1981.

3A.4.1-5 ACI Standard 318-71/77, Building Code Requirements for Reinforced Concrete, American Concrete Institute.

Table 3A.4.1-1

Basemat - Stress Resultants at Critical Sections

	Bending Moment (kips per ft)	Beam Shear (kips per ft)	Punching Shear (kips per ft)
$D+L+E_{SS}$	3132	125	315
P_{SR}	511	32	32
P_B	76	1	20
Comb. 7	3649	157	353
Capacity	4230	232	465

Table 3A.4.1-2

Pedestal - Stress Resultants at Base

Load	Overturning Moment - M (ft - kips)	Base Shear - H (kips)
E_O	123,600	1,530
E_{SS}	212,300	2,665
P_{SR}	2,825	719
P_B	5,088	29
(Comb. 7)	212,380	2,760
Capacity	375,000	14,500

Table 3A.4.1-3

Equivalent Stress Cycles for Fatigue Evaluation

Load	Number of Events	Number of Equivalent Stress Cycles per Event	Total Number of Stress Cycles
Operating basis earthquake	5	10	50
Safe shutdown earthquake	1	10	10
SRV[a]	4,478	3	13,434
Chugging	1	1,000	1,000

[a] This includes the cycles due to building motion, direct pressure, and fluid transients during SRV actuations.

Table 3A.4.1-4

Summary of Stress Intensities for Diaphragm Floor Seal

Loading Condition Table 3A.3.5-5	Primary Membrane Stress Intensity				Primary Membrane Plus Secondary Stress Intensity			
	Calculated Stress Int. Pm (ksi)	ASME Allowable Limit	ASME Allowable Stress (ksi)	Design Margin	Calculated Stress Int. Pm + Q Range (ksi)	ASME Allowable Limit	ASME Allowable Stress (ksi)	Design Margin/ Remarks
Normal	4.46	Sm	16.56	3.71	26.83	3 Sm	49.68	1.85
Upset	11.84	Sy	25.0	2.11	44.36	3 Sm	49.68	1.12
Emergency	12.89	1.2 Sm	19.9	1.54				Evaluation not required
Faulted	13.82	1.2 Sy	30.0	2.17				Evaluation not required

$$\text{Design Margin} = \frac{\text{Allowable Stress}}{\text{Calculated Stress}}$$

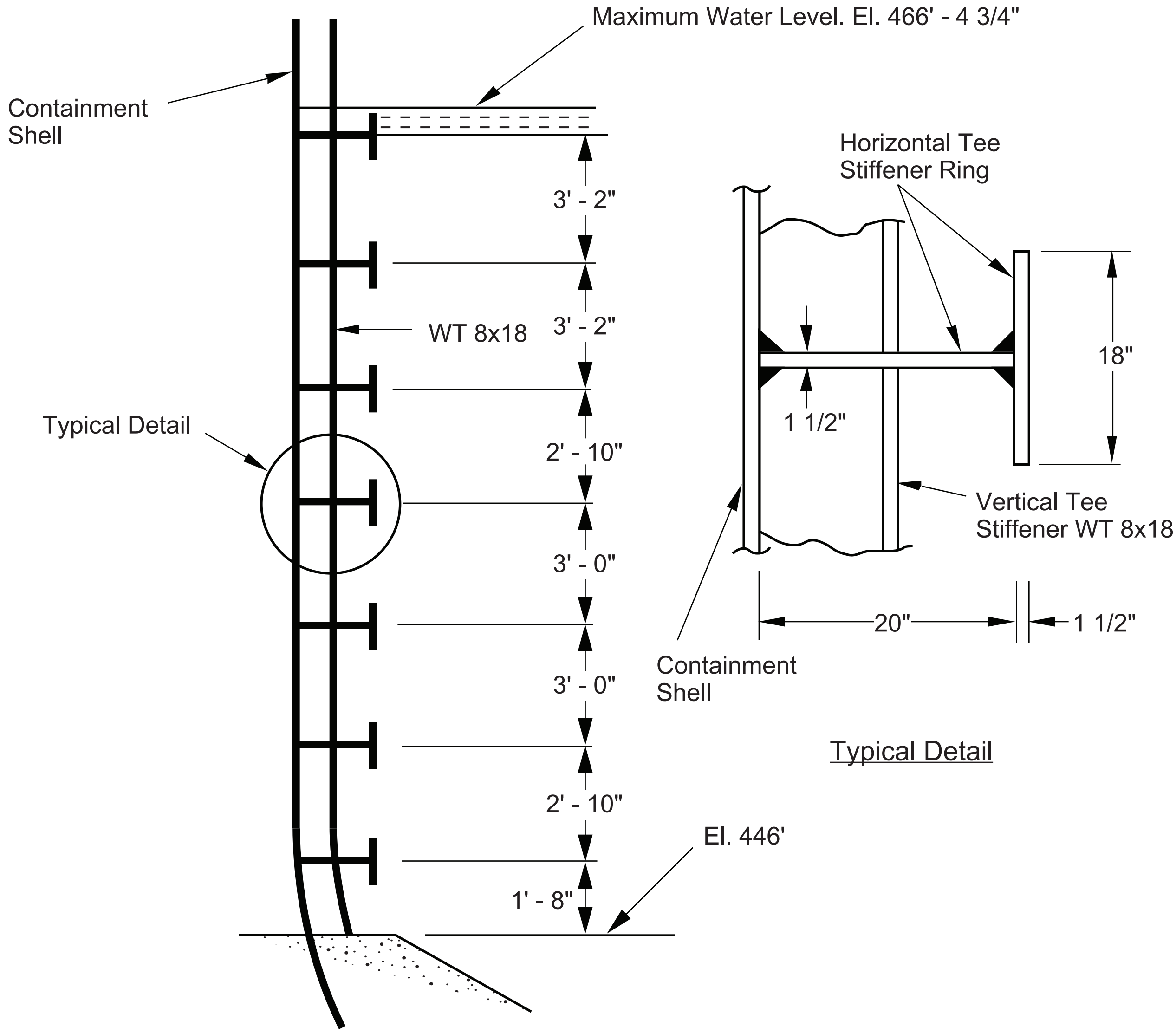
Table 3A.4.1-5

Cumulative Usage Factor Calculation
for Diaphragm Floor Seal

Load Combination Set[a]	Expected Number of Cycles (ni)	Alt. Stress Salt (psi)	Allowable Number of Cycles N	Calculated Usage Factor $Ui = \frac{ni}{Ni}$
1	1	387.0	33	0.03
2	9	204.0	150	0.06
3	50	178.5	250	0.2
4	60	153.4	350	0.172
5	880	74.0	3700	0.238
6	12,434	18.0		0.0
		Cumulative Usage Factor U = 0.7 < 1		

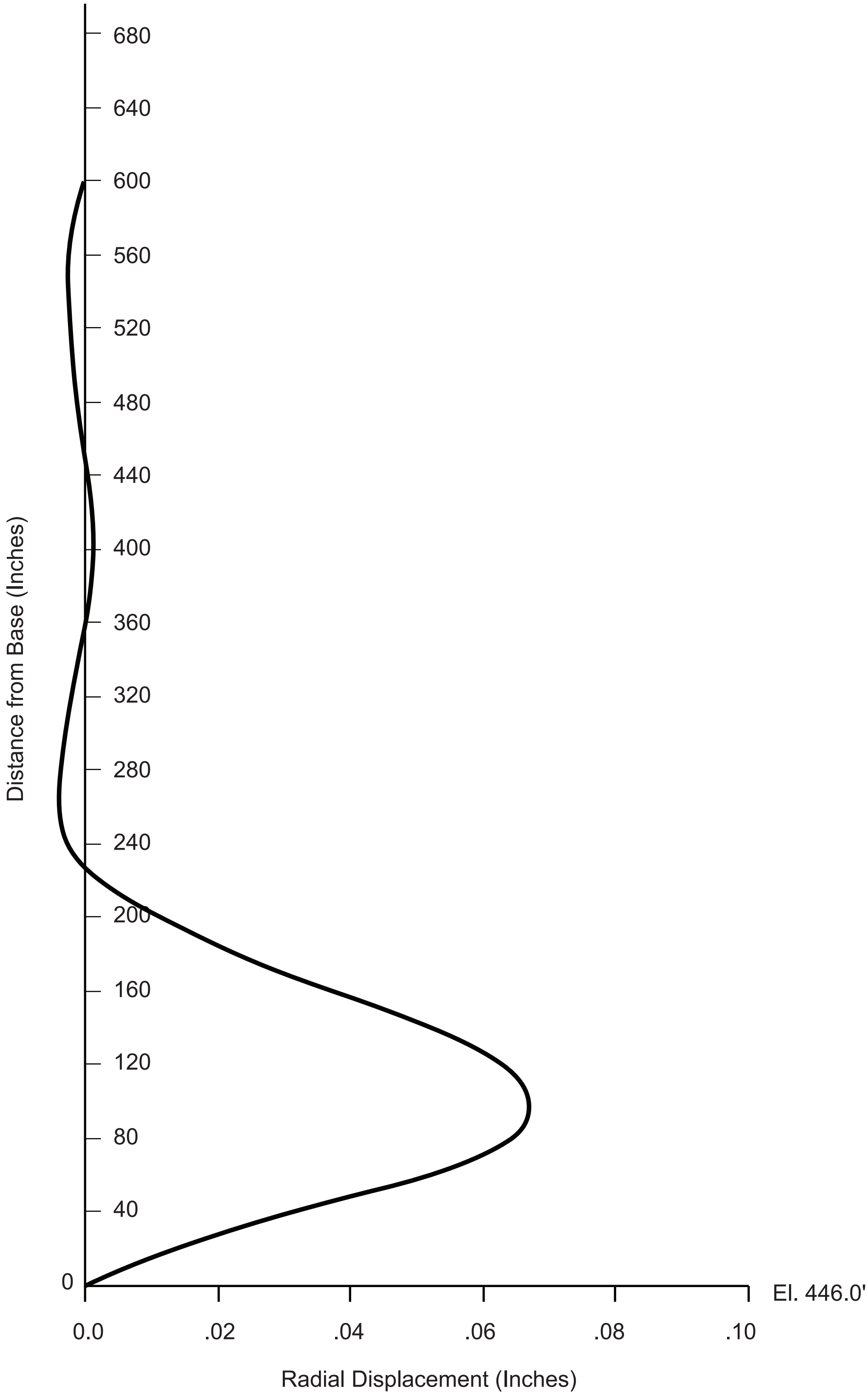
[a] Load combination set definition

1 - NPC + LOCA + SSE + SRV + CHUGGING
2 - NPC + SSE + SRV + CHUGGING
3 - NPC + OBE + SRV + CHUGGING
4 - NPC + SRV+ CHUGGING
5 - SRV + CHUGGING
6 - SRV



Stiffened Containment in Wetwell Region

Draw. No. 960222.85 | Rev. | Figure 3A.4.1-1



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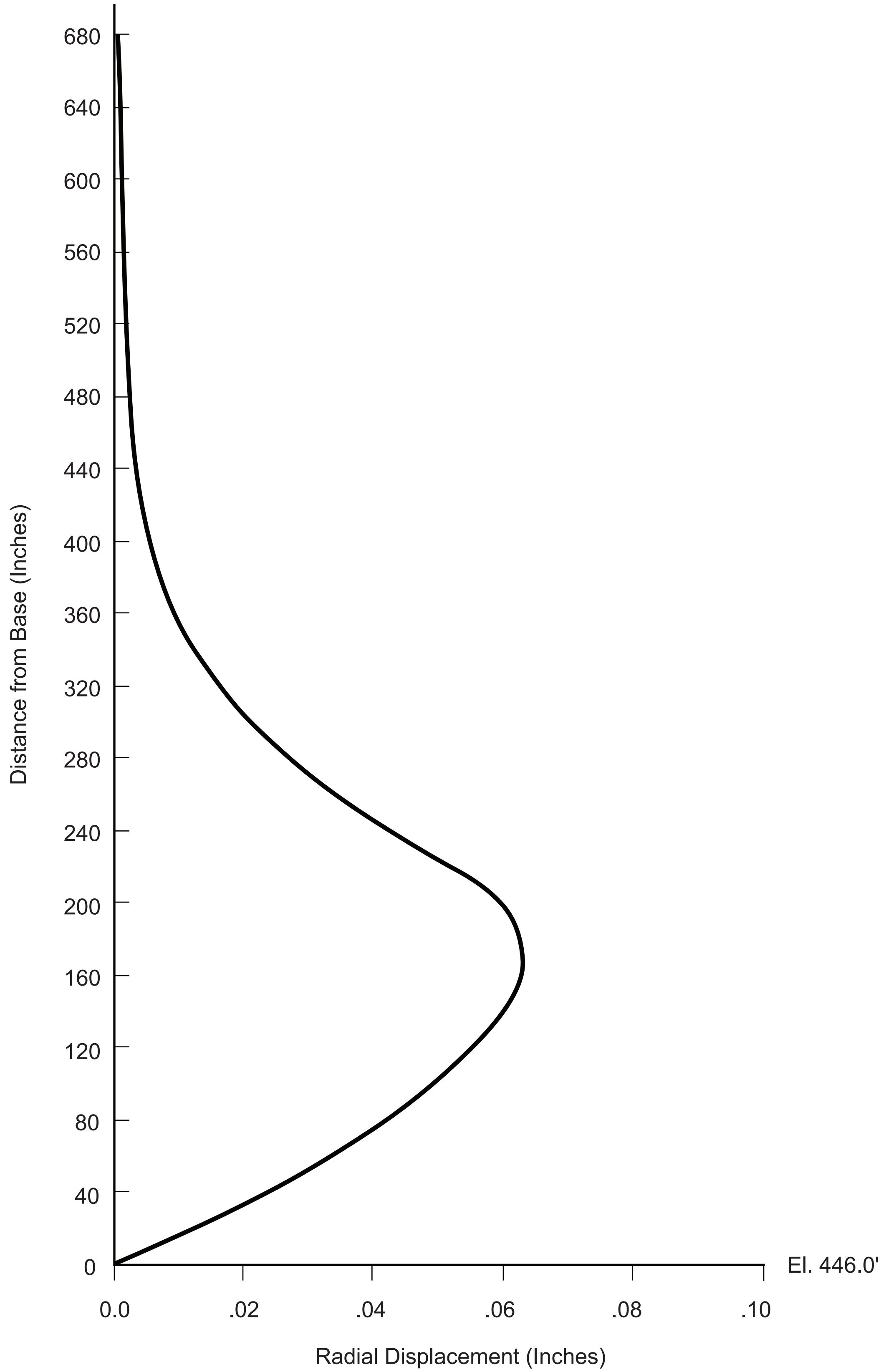
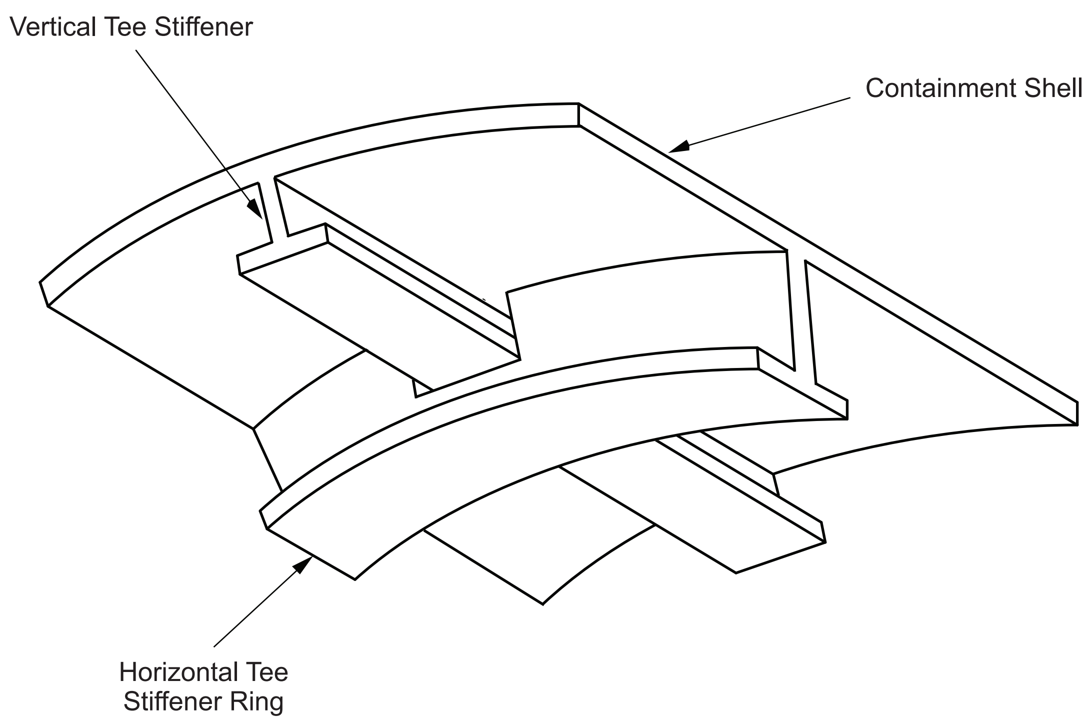
Displacement Profile
SRV Load - All Valves

Draw. No. 960222.86	Rev.	Figure 3A.4.1-2



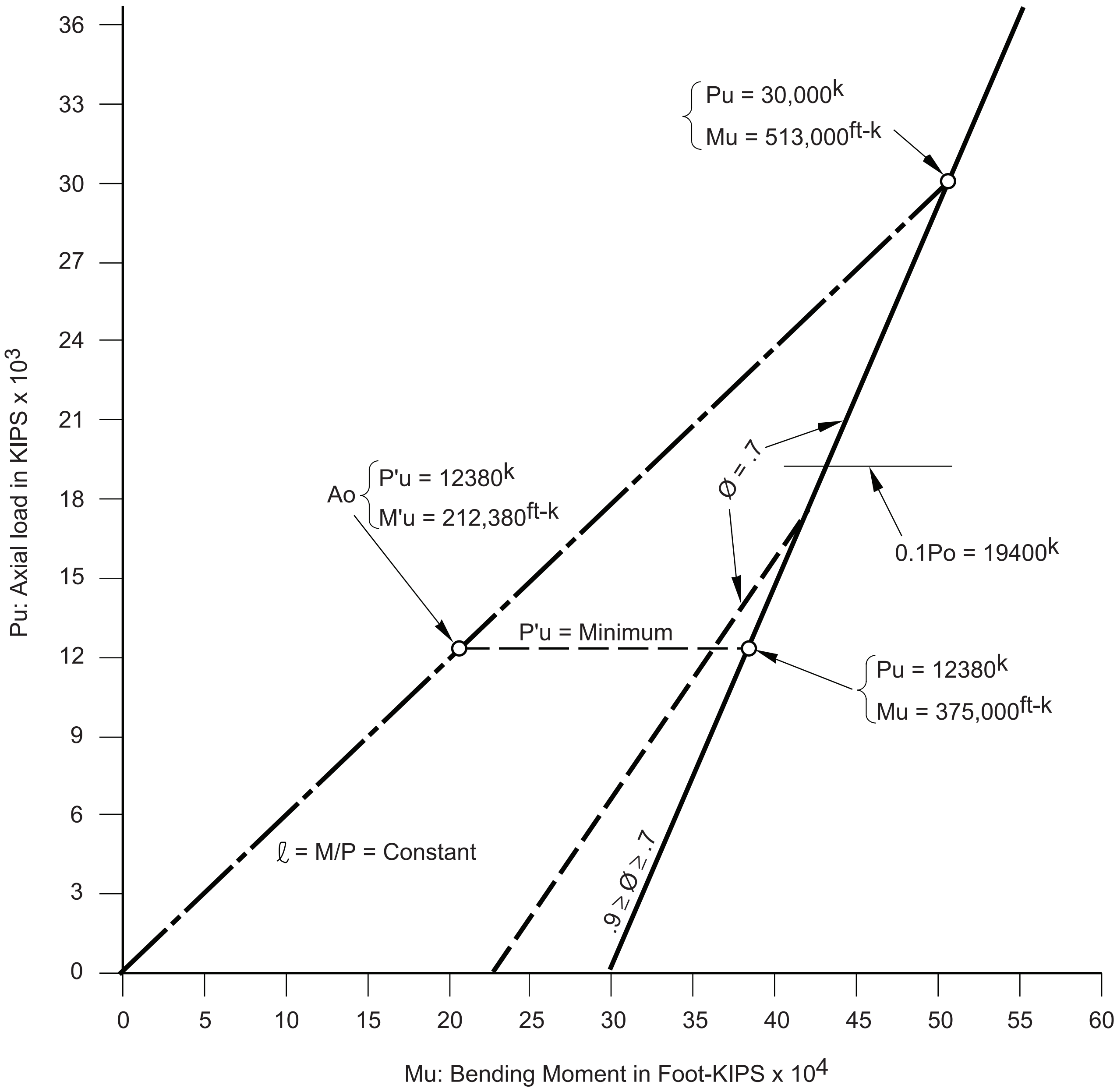
Displacement Profile
Nearly Symmetric Chugging

Draw. No. 960222.87 | Rev. | Figure 3A.4.1-3



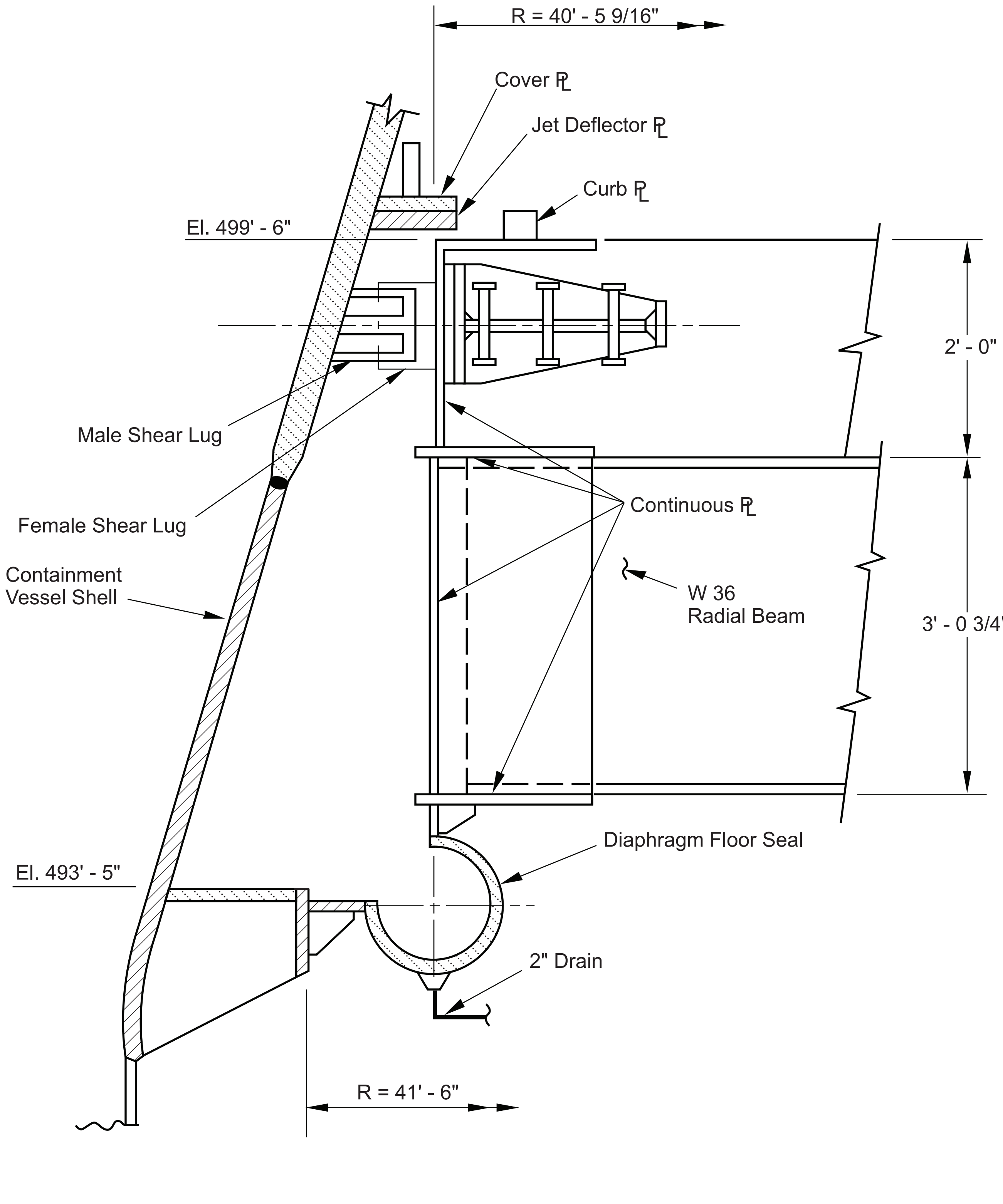
Stiffener Configuration

Draw. No. 970187.22 | Rev. | Figure 3A.4.1-4



Pedestal-Interaction Diagram
Axial Load Versus Moment

Draw. No. 960222.90 | Rev. | Figure 3A.4.1-8



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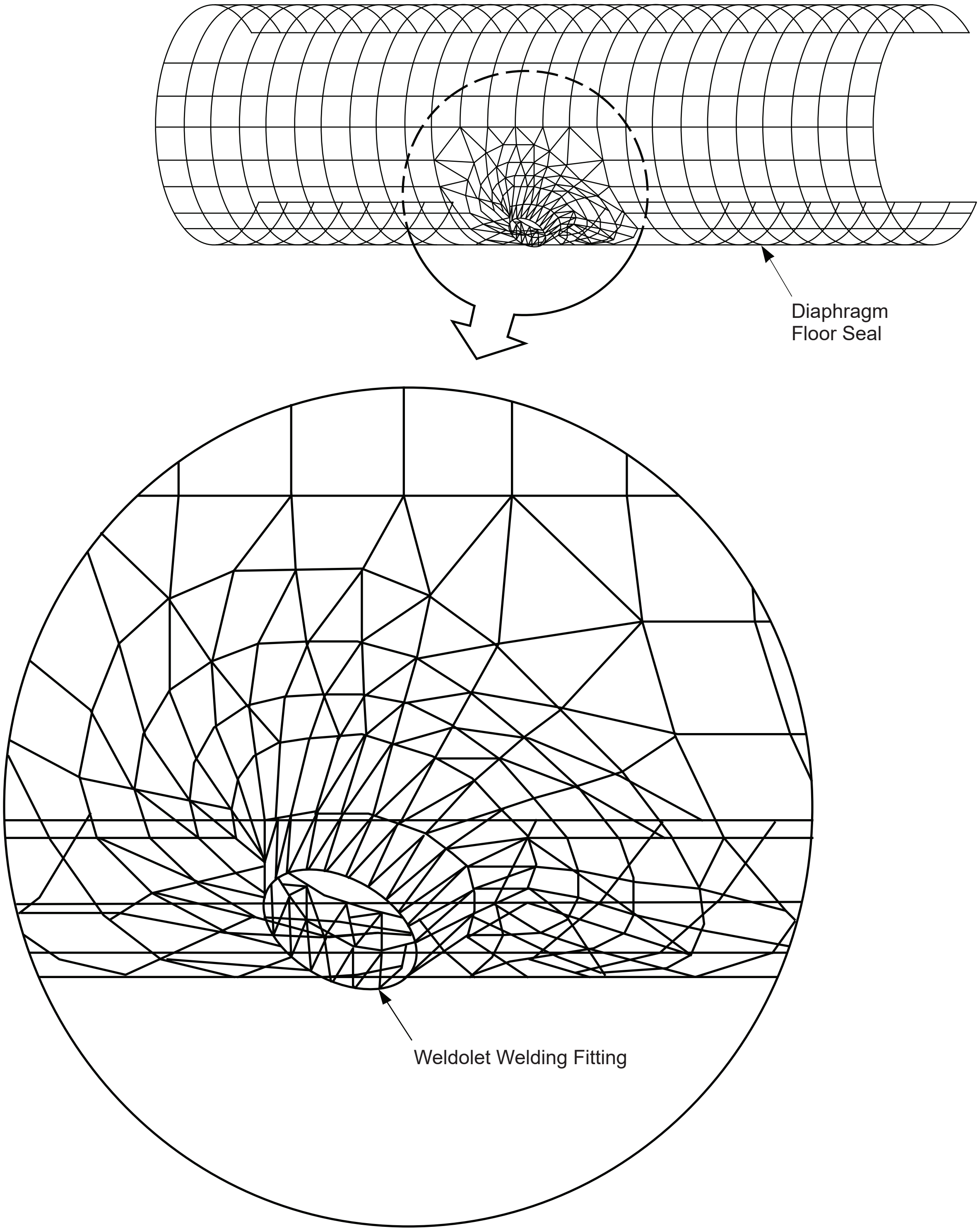
Drywell Diaphragm Floor Seal

Draw. No. 960222.88	Rev.	Figure 3A.4.1-9



Finite Element Model Diaphragm Floor Seal

Columbia Generating Station
Final Safety Analysis Report

Draw. No. 990578.01 | Rev. | Figure 3A.4.1-10

3A.4.2 SUPPRESSION POOL MAJOR STRUCTURES AND COMPONENTS

Assessment of the capacities of the major structures and components of the suppression pool chamber relative to load combinations involving suppression pool hydrodynamic loads is made in this section. The structures considered are downcomer bracing systems columns, downcomers, SRV piping system, quenchers, and platforms.

3A.4.2.1 Downcomer Bracing System

An assessment was made of the capacity of the original design of the downcomer bracing system relative to load combinations involving suppression pool hydrodynamic loads. It was determined that this original bracing, consisting of a system of radial beams, had inadequate capacity. Consequently, a replacement pipe truss bracing system was designed and installed. The assessment of the capacity of this pipe truss system relative to the load combinations involving suppression pool hydrodynamic loads is made in this section.

3A.4.2.1.1 Description of System

The pipe truss system of downcomer bracing is shown in Figure 3A.4.2-1. Like the original system, the function served by the pipe truss system is to provide horizontal support for the 102 downcomers, (three of which are capped as discussed in Sections 3A.3.2.1 and 3A.3.2.2) and the 18 SRV discharge pipes at a level near the lower end of the downcomers.

The pipe truss system consists of a horizontal planar truss located with center line at the same elevation as the original system. The model of the truss used in the structural analysis is shown in Figure 3A.4.2-2. In the truss system, the downcomers and the SRV lines are located at the truss nodes. Structural rings are provided around each downcomer and each SRV pipe for connections by the truss members. The truss members are 4-in. and 6-in. double extra strong steel pipes. Connections of the truss to the RPV pedestal and to the containment vessel are at the same connection points as the original radial beams.

As described in Section 3A.4.2.1.2, the pipe truss system is subjected to both horizontal and vertical loads. Horizontal reactions from the downcomers and from the SRV pipes are applied to the encircling structural rings which form the truss nodes. Horizontal forces applied directly to the truss members are also carried by the members to the truss nodes. By truss actions these horizontal loads are transmitted to the supports at the RPV pedestal and at the containment vessel. The pedestal connection can sustain both radial and circumferential reaction components due to horizontal loading; however, the vessel reaction is circumferential because the connection is free to move radially.

Vertical loadings, due to the various causes listed in Section 3A.4.2.1.2, act directly on the pipe truss system. To carry these vertical loadings, supports are provided against upward and downward motion at each of the downcomers; also the connections to the pedestal and vessel

are restrained vertically. Vertical forces acting on each truss member are carried to its ends at the structural rings, pedestal, or vessel connections. The structural rings around the downcomers are independent of the downcomers but stops are welded to the downcomers to prevent differential vertical motion. The structural rings around the SRV lines are independent of these lines and no restraint against differential vertical motion is provided. Vertical loads from the rings on the downcomers are transmitted by the downcomers to the drywell floor.

3A.4.2.1.2 Loads Used for Assessment

A complete description of all hydrodynamic loads is given in Section 3A.3. In this section only the loads used for the assessment of the bracing system are discussed. Symbols used in this section are defined in Section 3A.3.5.3 in connection with load combinations.

3A.4.2.1.2.1 Safety/Relief Valve Actuation Loads. Safety/relief valve actuation causes horizontal and vertical loading on the bracing system as unbalanced pressures and induced accelerations of supported components occur. The pressures and accelerations acting on the downcomers, the SRV pipe lines, and the bracing members cause the horizontal loading; the vertical loading is due to these actions on the bracing system alone. The spatial distribution of these loads is discussed under Methods of Analysis (Section 3A.4.2.1.4).

Loads due to the SRV pressures and induced accelerations are applied to the bracing system as equivalent static loads. The magnitudes of the loadings from the downcomers and SRV lines are based on analyses of each of these components as described in the assessments of the components in Sections 3A.4.2.3 and 3A.4.2.4 respectively. The pressure loadings on the bracing members proper are equivalent static pressures as defined in Section 3A.3. Reactions due to the pressures on the downcomers, SRV lines, and truss members are applied at the truss nodes.

The forces due to the induced accelerations of the downcomers, SRV lines, and bracing members are obtained by analysis of these structures using the response spectra developed for SRV actuation. These forces from downcomers, SRV lines, and bracing members are also applied as reactions at the truss nodes.

3A.4.2.1.2.2 Loss-of-Coolant Accident Loads. Loss-of-coolant accidents are characterized by several phenomena causing nonconcurrent loadings on the bracing system. The principal nonconcurrent loadings are short-term pool swell pressure and fallback pressure loads and long-term chugging loads. Other significant loads are short-term jet and bubble loads and long-term condensation oscillation loads. Pool swell and fallback are basically vertical motions and generally applied vertical pressures are associated with each. The chugging loads result from lateral forces at the downcomer exits, from horizontal and vertical accelerations induced by building motion, and from chugging pressures on the wetwell components supported by the bracing system. The horizontal accelerations of the downcomers, SRV lines,

and bracing members contribute to the horizontal loading of the bracing, whereas the vertical acceleration of only the bracing members causes vertical loading of the bracing.

a. Pressure due to pool swell

The pool swell pressures acting on the bracing system are determined to be 25 psi upward and 5 psi horizontal including the DLF, applied concurrently;

b. Pressure due to fallback

Fallback pressures acting on the bracing system are determined to be 25 psi downward and 5 psi horizontal including the DLF, applied concurrently;

c. Lateral forces at downcomer exits

The characteristics of these lateral exit loads are described in Section 3A.3.2.4.4. As noted therein, the lateral exit loads are dynamic loads whose amplitude depends on the size of the downcomer and the number of concurrently loaded downcomers. For an assumed number of concurrently loaded downcomers, the axial forces in the members of the bracing system and the reactions at the supports are determined by dynamic analysis of the system. Determination of the critical number of downcomer exits which are subjected simultaneously to lateral force is described subsequently under Methods of Analysis (Section 3A.4.2.1.4);

d. Induced accelerations due to chugging

The forces due to the induced accelerations of the downcomers, SRV lines, and bracing members are obtained by analyses of these structures using the response spectra developed for chugging action. Forces from these components are applied on the bracing system as reactions at truss nodes;

e. Chugging pressures

Chugging pressures are applied on the members of the bracing system and on the inner row SRV lines as described in Section 3A.3.2.4.3.2. Equivalent static loads constituting the end reactions due to the pressures on these components are applied at the truss nodes; and

f. Other LOCA loads

The other LOCA loads mentioned above are now considered. It is noted that the short-term loads due to LOCA jet and bubble are smaller in magnitude than

the previously considered SRV pressure loads which cannot occur simultaneously with them. Hence, these short-term loads are not controlling. Similarly, it is determined that the long-term condensation oscillation loads are smaller than loads due to chugging which occur subsequently. Therefore, the condensation oscillation loads also are not controlling.

3A.4.2.1.2.3 Other Significant Loads. Seismic forces represent a significant loading on the bracing system. The forces due to the seismic accelerations of the downcomers, SRV lines, and bracing members are obtained by analysis of these structures using the response spectra developed for OBE and SSE. The forces from downcomers, SRV lines, and bracing members are applied as reactions at the truss nodes.

Dead load of the bracing system and thermal loads are also included in the assessment. Thermal loads result from temperature change of the bracing system and from reactions on the bracing system from supported piping.

3A.4.2.1.3 Controlling Load Combinations and Acceptance Criteria

The load combinations and acceptance criteria pertinent to the downcomer bracing are those listed in Section 3A.3.5.3. Based on the results of the analysis, it is determined that, for the loading conditions involved, the controlling combinations with associated allowable stresses are those listed below. Only the significant load terms are included.

Service load conditions:

(1) $S \geq D + P_{SR}$

Factored load conditions:

(5) $1.6\ S \geq D+T_A+E_O+P_B+P_{SR}$

(7) $1.7\ S \geq D+T_A+E_{SS}+P_B+P_{SR}$

Further description as to the loads included in the design loading conditions is presented under Method of Analysis.

3A.4.2.1.4 Method of Analysis

Structurally, the pipe truss downcomer bracing system is treated as a plane truss with respect to horizontal loads and as an assembly of beams with respect to vertical loads. The horizontal planar pipe truss is supported externally at 17 equally spaced points around the pedestal and at an equal number of points at the containment vessel. Prior to the truss analysis, the reactions due to the distributed loads along the downcomers and along the SRV lines and bracing

members are calculated and these reactions are applied directly at the truss nodes. The analysis methods, design loading conditions, and principal results are described below.

3A.4.2.1.4.1 Analysis for Horizontal Loads.

a. The analysis uses the proprietary computer program "Mc Auto STRUDL".

b. The structural model used in the analysis is shown in Figure 3A.4.2-2. The model represents the actual configuration of the north half of the symmetrical structure except for adjustments along the omitted half structure.

c. The truss and its connections are treated as pin connected so that the truss members carry only axial load.

d. All loads are taken to act at the nodes in the truss analysis. The effect of truss member bending under distributed normal loads is considered in member design where combined axial load and member bending is provided for.

e. The STRUDL analysis furnishes truss member axial loads and reaction components as well as displacements of nodes.

3A.4.2.1.4.2 Analysis for Vertical Loads.

a. Vertical loads on the bracing system, i.e., on bracing members and on rings, are transmitted through bending and shear to supports on the downcomers and at the pedestal and vessel. Combined axial load and member bending is provided for in the design.

b. Downcomers and diaphragm floor are investigated for the resultant vertical loads.

3A.4.2.1.4.3 Design Load Conditions. The spatial distribution, direction, and magnitudes of possibly coincident loads as described below are used in the controlling load combinations; two critical loading conditions are adopted as a conservative basis of design. By these loading conditions it is intended to maximize stresses and reactions in specific portions of the half-structure. These critical stresses and reactions are then utilized to design all similar members and supports around the entire structure. The two conditions utilized in the design are noted below.

a. Condition 1: Maximum Horizontal Loading

This loading is intended to result in greatest stresses in the members and supports in the vicinity of the north-south and east-west axes. The directions of

building motions and applied pressures and loads are selected so as to maximize overall loading in the south to north and west to east directions. (Refer to truss layout in Figure 3A.4.2-2.) Associated vertical loads are also included.

1. SRV actuation

   a) Building motions due to all valve actuation and due to single valve actuation are considered. The building motions cause reactions at all downcomers and SRV lines and accelerations of all truss members. Radial forces and accelerations are directed outward; circumferential forces and accelerations are clockwise. Reactions of downcomers, SRV lines, and members act on the truss nodes.

   b) Unbalanced pressures due to all valve and single valve actuation are included. Varying pressures depending on distance from the active nodes are applied to all downcomers, SRV lines, and bracing members. The resulting reactions from these components act at the truss nodes. The direction of reactions from downcomers and SRV lines is generally along the line from the active node to the structure. For the bracing member, the pressures and reactions are normal to the bracing member.

   c) Vertical loads on the pipe truss are due to dead load, building motion induced acceleration, and unbalanced pressure.

2. Long-term LOCA loads (chugging)

   a) In the analysis, the number of downcomers subjected to lateral exit loads is varied from one to all 51 downcomers in the model structure. Two directions of load application are considered: south to north and west to east. It is determined that controlling effects are always due to loading of a single vent. In this regard it is noted that the lateral load definition in Section 3A.3.2.4.4 provides a high intensity load for single vent loading as compared to multiple vent loading.

   b) Building motion due to chugging causes reactions at all downcomers and SRV lines and accelerations of all truss members. Radial forces and accelerations are directed outward; circumferential forces and accelerations are clockwise. Reactions of the downcomers, SRV lines, and truss members act on the truss nodes.

c) Chugging pressures on the interior SRV lines result in radially outward reactions applied at the truss nodes.

d) Vertical loads on bracing members are due to building motion induced accelerations and chugging pressure. In addition, inclination of bracing members during LOCA caused by downcomer temperature growth results in vertical component of member axial force as an additional reaction on the downcomer.

3. Seismic

a) Eastward, northward, and vertical seismic actions are included. Both OBE and SSE are considered.

b) In west-east seismic action, all downcomers and SRV lines react against the bracing towards the east. All bracing members have accelerations eastward. Corresponding northward effects result from south-north seismic action.

c) Vertical loads on bracing members are due to vertical accelerations;

b. Condition 2: Maximum Vertical Loading

The vertical loads associated with Condition 1 which involve controlling horizontal loads have been previously stated. To determine the maximum vertical loading, short-term LOCA events are investigated. Loads included under Condition 2 are described below.

1. SRV actuation

The acceleration and pressure loads due to single valve actuation are included.

2. Pool swell or fallback

The vertical statically equivalent pressures due to pool swell or fallback are 25 psi on pipe truss members and on truss rings at downcomers and SRV lines.

3. Seismic

Horizontal and vertical loadings are the same as described above for Condition l.

4. Additional DBA effects

Inclination of bracing members during DBA caused by downcomer temperature growth results in a vertical component of member axial force as an additional reaction on the downcomers. Also, building motion results in vertical acceleration of the bracing members. However, this acceleration has relatively small magnitude.

5. Coincident member axial loads

Such axial loads are caused by single SRV actuation and by seismic loading.

3A.4.2.1.5 Results and Design Margin

The principal results of the analysis and the resultant design margins are stated below.

3A.4.2.1.5.1 Principal Results.

a. Bracing members

These members are designed for combined stress involving axial force and biaxial bending. Members connecting to the pedestal are 6-in. double extra strong steel pipe; all other members are 4-in. double extra strong steel pipe;

b. Node rings

These rings are designed for a combination of radial loads which is conservative based on the design loading conditions. As a result of this analysis, built-up H-sections 5 in. wide by 7 in. high at the downcomers and 4 in. wide by 6 in. high at the SRV lines are used. Material is high strength structural steel (ASTM A 572, Fy = 60 k.s.i.);

c. Pedestal connections

The existing embedded plates for the original radial beam system are utilized in the pipe truss system. Additional strengthening was installed based on the design loading conditions. Six concrete anchors were added at each pedestal connection; and

d. Containment vessel connection

The pipe truss end bearing at the containment vessel fits into the existing socket provided for the original radial beams. The containment vessel redesign includes the reactions due to the downcomer bracing design loading conditions.

3A.4.2.1.5.2 Design Margins. The design margins provided by the pipe truss bracing system are discussed in this section. The design margin for a structural component or system refers to the controlling design parameter such as bending stress for a flexural member and sum of the amplified stress ratios for the case of combined axial and bending stress. The design margin is then defined as the ratio of the permissible value of the design parameter to the value of the design parameter under design loading.

Table 3A.4.2-1 lists the controlling design margins for each of the six principal structural components of the pipe truss bracing system, namely, 6-in. pipe members, 4-in. pipe members, pedestal connection, vessel connection, downcomer ring, and SRV line ring. The calculated design margins are listed in the table for the cases of maximum horizontal or vertical loading (Conditions 1 or 2), under both service load and factored load conditions.

3A.4.2.2 Columns

The assessment of the capacity of the columns relative to load combinations involving suppression pool hydrodynamic loads is presented in this section. The columns and adjoining structures are shown in Figure 3A.4.2-3.

3A.4.2.2.1 Loads Used for Assessment

A complete description of all hydrodynamic loads is given in Section 3A.3. Seismic loads are described in Section 3A.3.5. In this section, only the loads used for the assessment of the columns are discussed. Symbols used in this section are defined in Section 3A.3.5.2 in connection with load combinations.

3A.4.2.2.1.1 Safety/Relief Valve Actuation Loads. Actuation of the SRVs results in four different load effects on the columns. These are unbalanced bubble pressure on the columns, column accelerations associated with resultant building motions, quencher discharge water jet loads, and secondary effects from pressure loading of the downcomer bracing truss.

Maximum bubble pressures applied laterally on the column are defined for two cases, namely, initial actuation and subsequent actuation. The spatial distributions of the maximum pressure loading on the column corresponding to these two cases are shown in Figure 3A.3.1-8. The DLFs associated with these maximum pressure loadings are functions of the column modal frequencies as shown in Figure 3A.3.1-10. The maximum dynamic column reactor loads for the subsequent actuation load case are documented by Reference 3A.3.1-9 and are tabulated in Table 3A.4.2-2. The structural model used for the dynamic analysis of the column under lateral load is described in Section 3A.4.2.2.3 and shown in Figure 3A.4.2-4.

The horizontal and vertical accelerations due to building motion are based on the response spectra developed for SRV actuation in Section 3A.5.1. The horizontal (lateral) accelerations are used in the dynamic analysis of the column by the response spectrum method in conjunction with the structural model described in Section 3A.4.2.2.3 and shown in Figure 3A.4.2-4. The axial (vertical) forces in the column are obtained from the vertical response spectra by a dynamic analysis by the response spectrum method; a structural model, which includes both the diaphragm floor beam and the column, is used and is shown in Figure 3A.4.2-5.

The quencher jet drag load on the column is small in comparison with the bubble load. Since the two loadings do not occur simultaneously, the jet loading is not a controlling loading. The direct vertical pressure loading on the downcomer bracing members is transmitted into the diaphragm floor by the downcomers. The diaphragm floor in turn loads the column. The net loading on the column during the phase of maximum bubble pressure on the column is small and is included in the assessment.

3A.4.2.2.1.2 Loss-of-Coolant Accident Loads. Both short-term and long-term LOCA loads are significant in the assessment of the columns.

Several axial loads on the columns due to short-term LOCA events are considered in the analysis. The overall pressure transient in the drywell and wetwell results in a net downward pressure (20.55 psi equivalent static) on the diaphragm floor and hence an axial compressive column load. The design net upward pressure (5.50 psi) on the diaphragm floor during the pool swell transient causes an upward load on the columns. During LOCA, vertical pressures act on the downcomer bracing system due to bubble charging and due to pool swell and fallback. The resultant vertical forces are transmitted via the downcomers and the diaphragm floor into the columns. However, these forces are not controlling in comparison with the aforementioned net downward and upward pressures on the diaphragm floor. In addition to the pressure loads, the load on the column, due to pipe break/jet impingement on the diaphragm floor, is included in the assessment.

The significant lateral loads associated with LOCA, which are included in the column assessment, are those due to chugging. Lateral loads due to water clearing and due to air

clearing are negligible. The loads due to condensation oscillation are less critical than those due to chugging, and consequently, are omitted from the assessment. The effects of chugging include both direct horizontal pressures on the columns and induced horizontal and vertical accelerations associated with building motions, but only the effects due to building accelerations are significant.

The horizontal and vertical accelerations due to chugging are based on the response spectra described in Section 3A.5.2. Column loading is then developed by dynamic analysis of the column by the response spectrum method in the manner described above for SRV induced building motion. The analysis for horizontal loading uses the structural model of Figure 3A.4.2-4, and the analysis for vertical loading uses the structural model of Figure 3A.4.2-5.

3A.4.2.2.1.3 Other Significant Loads. Dead load, live load, seismic loads, and loads due to annulus pressurization constitute additional significant loads included in the column analysis. The dead load of the diaphragm floor and columns is carried as an axial column load. The live load on the diaphragm floor is transmitted to the columns as an axial load. Horizontal and vertical column loadings are obtained from the seismic response spectra in the same manner as described above for the SRV and LOCA response spectra.

Pressurization of the annulus between the RPV and the sacrificial shield wall, due to a postulated break in either the circulation line or the feedwater line, as described in 6.2.1.2 of the FSAR, results in building motions with associated response spectra. Horizontal and vertical column loadings are obtained from the response spectra in the same manner as described above for the SRV and LOCA response spectra.

3A.4.2.2.2 Applicable Load Combinations and Acceptance Criteria

The load combinations and acceptance criteria for internal reinforced-concrete structures described in Section 3A.3.5.2 are applicable to the columns.

3A.4.2.2.3 Method of Analysis

The structural capacity of the column is investigated for the applicable load combinations with loads as listed above. The general approach in the column assessment is to determine the values of the controlling stress resultants in the column (bending moment, axial force, and shear) on the basis of elastic analysis under design loads and to calculate the capacity of the column in terms of these stress resultants by the strength method of the ACI 318-71 Code (Reference 3A.4.1-5). Three loading cases are investigated. Because of the shape of the column interaction curve for bending and axial load, maximum bending in the column is checked first with minimum coincident axial load and then with maximum coincident axial load. The third loading case involves maximum shears.

The moments, shears, and axial forces are obtained for each of the significant loads. The column is analyzed dynamically via a modal time-history technique (Reference 3A.3.1-9) for the directly applied SRV pressures and for the loads due to the building motion accelerations as defined by the SRV chugging, seismic, and annulus pressurization response spectra; dynamic analysis is by the response spectrum method. In the analysis, the column is treated as fixed at its base and simply supported at its top in accordance with actual construction. For the dynamic analysis under lateral (horizontal) loading, the actual structure is modeled as a 17-node beam as shown in Figure 3A.4.2-4. Masses are lumped at the nodes with additional mass provided below pool level to account for the effect of the suppression pool. The axial (vertical) column forces are determined by the dynamic analysis using a structural model which represents the diaphragm floor beam - column system. As shown in Figure 3A.4.2-5, the diaphragm floor mass is included in 26 nodes along the beam and the column mass is 14 nodes along the column. Dynamic analysis for lateral loads and for vertical loads are made using the commercial computer programs STRUDL and IMAGES-3D.

The critical locations along the column for bending moment and shear are determined from the analysis results. Maximum bending moments and shear occur at the column base. The magnitude of the horizontal shear at the top of the column is also noted to verify the adequacy of the top connection. The values of the controlling bending moments, shears, and axial force are listed for each significant loading in Table 3A.4.2-2.

a. Maximum bending moment with minimum axial load

It is determined that the controlling values of maximum bending moment with coincident minimum axial load occur for load combination (1). This combination is stated below for minimum axial load with only the significant load terms listed.

(1) $1.0\ D + 1.5\ P_{SR}$

The maximum bending moment is caused by the direct lateral pressure on the column, due to SRV subsequent actuation, and the associated horizontal building motion acceleration. The coincident minimum compression axial load occurs with dead load (D) and upward load due to the vertical building motion acceleration caused by the SRV actuation (P_{SR}). The controlling values of bending moment and axial load are 17.67 ft kips and 173.3 kips (compression), respectively.

The column capacity for combined loading and axial load is determined in accordance with the strength design method of the ACI 318-71 Code (Reference 3A.4.1-5). From the applicable interaction curves, the bending moment capacity of the column, with the above axial load of 173.3 kips, is found to be 2182.0 ft kips;

b. Maximum bending moment with maximum axial load

The controlling coincident values of bending moment and axial load occur for load combination (1). This combination is stated below for maximum axial load with only the significant load terms listed.

(1) $1.4\ D + 1.7\ L + 1.5\ P_{SR}$

The maximum bending moment is caused by SRV subsequent actuation as in (a) above. The coincident maximum compressive axial load is due to dead load (D), live load (L), and downward load due to the SRV actuated building motion (P_{SR}). The controlling value of the axial load is 745.2 kips. As a result of moment magnification due to the axial load, the controlling value of the bending moment is increased from 1700 ft kips to 1869 ft kips. Utilizing the column interaction curves, the bending moment capacity is found to be 2230.5 ft kips;

c. Maximum shear

As noted previously, maximum column shear occurs at the column base. Column shear capacity is affected by the axial force at the section. The controlling load combination for column shear is load combination (1) which is stated below for minimum axial load with only the significant loads listed.

(1) $1.0\ D + 1.5\ P_{SR}$

The maximum shear is caused by the same event as in (a) above, namely, SRV subsequent actuation (P_{SR}). The coincident minimum compressive axial load occurs with dead load (D) and SRV actuation (P_{SR}). The controlling values of shear and axial load are 142.4 kips and 173.3 kips (compression), respectively. Utilizing the strength design method of the ACI 318-71 Code (Reference 3A.4.1-5), the column shear capacity is calculated to be 376.9 kips; and

d. Column top shear connection

The controlling top horizontal reaction in relation to the capacity of the top connection in shear occurs with load combination (1) with minimum axial (compressive) load. This combination is stated below.

(1) $1.0\ D + 1.5\ P_{SR}$

The top horizontal reaction is caused in this case by SRV initial actuation (P_{SR}); its magnitude is 62.1 kips.

The shear capacity of the top connection is due to shear friction associated with the connecting anchor bolts and the superimposed axial load. Conservatively, the superimposed load is due to dead load (D) reduced by the upward load due to SRV actuation (P_{SR}). Utilizing the strength design method of the ACI 318-71 Code (Reference 3A.4.1-5), the shear capacity of the top connection is 75.5 kips.

3A.4.2.2.4 Results and Design Margins

It is determined that the columns have adequate capacity with regard to the applicable load combinations involving suppression pool hydrodynamic loads. The design margins with respect to the significant stress resultants are noted below:

a. Maximum bending moment with minimum axial load - The smallest design margin representing the ratio of column bending capacity to applied bending moment, with minimum axial load, is 1.54;

b. Maximum bending moment with maximum axial load - The smallest design margin representing the ratio of column bending capacity to applied bending moment, with maximum axial load, is 1.49;

c. Maximum column shear - The smallest design margin representing the ratio of the column shear capacity to the applied shear is 3.23; and

d. Column top shear connection - The smallest design margin representing the ratio of the capacity of the column top connection in shear to the applied top shear is 1.18.

3A.4.2.3 Downcomers

The primary function of the downcomer vent system is to channel the steam accumulating in the drywell chamber during a LOCA into the wetwell chamber to accomplish pressure suppression (see Section 3A.3.2.1).

The downcomer vent system consists of eighty-three 24-in. OD and sixteen 28-in. OD standard schedule carbon steel pipes running vertically downward from the diaphragm floor (except that the ends of the downcomers are stainless steel as described in FSAR Section 3.8.3.4). Originally 102 downcomers were provided, but three (one 24-in. and two 28-in.) were capped, as discussed in Sections 3A.3.2.1 and 3A.3.2.2. The downcomers are embedded in the diaphragm floor and extend down to el. 454 ft 4.75 in. All downcomers

are restrained laterally at el. 455 ft 4 in. by the downcomer bracing system, which is vertically restrained by the downcomers. Vertical loads are imposed by the bracing system onto the downcomers and transmitted to the diaphragm floor. *(See Figures 3A.2.1-5 and 3A.2.1-6.)*

Nine of the 24-in. OD downcomers have an extra strong welding tee at el. 491 ft 11 in. to accommodate 24-in. dual inline vacuum breaker valves. In addition, to provide extra strength, the eighteen 28-in. downcomers have been stiffened by the insertion of a 4-ft 8-in. long by 2-in. thick spool piece. This piece accommodates the penetration for the main steam SRV (MSRV) piping which is welded to these downcomers at el. 493 ft 0 in.

Figure 3A.2.1-4 shows locations where the vacuum breaker valve assemblies and the MSRV piping penetrate the downcomers just below the diaphragm floor.

3A.4.2.3.1 Loads Used for Assessment

The downcomer piping is subjected to static, dynamic, and hydrodynamic loads under the various plant operating conditions identified as normal, upset, emergency, and faulted. Each of these loads in various combinations is identified in Section 3A.3.5.4.

The individual loads acting on the downcomers are identified below:

a. Deadweight (W)

b. Thermal expansion and thermal transient

c. Pressure (P)

   The pressure differential between the drywell and suppression chamber atmospheres produces loads on the downcomer walls since it acts as a pressure retaining boundary during a LOCA.

d. OBE

e. SSE

f. SRV discharge dynamic loads

   The spatial distribution of the maximum direct bubble pressure loading used for downcomer assessment was obtained by multiplying a dynamic pressure load (Figure 3A.3.1-7) by the maximum DLF.

   A maximum DLF of 4.2 was conservatively obtained from the response spectrum of DLFs shown in Figure 3A.3.1-10.

The inertia loading effects due to the acceleration of the structure are described in Section 3A.5.1. The response spectra used for downcomer assessment are the enveloped spectra which were developed by enveloping the spectra due to four SRV actuation cases at the appropriate locations for the downcomers.

g. LOCA loads

The loads on the downcomer associated with LOCA are chugging pressure, condensation oscillation pressure, and the building response loading during LOCA event.

The spatial distribution of condensation oscillation and chugging pressure loadings on downcomers are considered as equivalent static pressure loads.

The pressure distribution for the condensation oscillation on downcomers is bounded by chugging load and therefore is not a controlling load.

The LOCA jet, LOCA bubble, pool swell, and fallback loads are identified to be negligible on downcomers as described in Section 3A.3.2, hence, these loads are not considered in downcomer analysis.

The input response spectra for chugging and condensation oscillation are based on the spectra described in Section 3A.5.2 and are enveloped in the same manner as described in the previous section for SRV load.

3A.4.2.3.2 Load Combination and Acceptance Criteria

The resultant stresses experienced by the downcomers are considered acceptable if they satisfy the ASME Boiler and Pressure Code, Section III, Subsection NC (Reference 3A.4.2-1).

The allowable stress "S_h" for both the 24-in. and 28-in. OD downcomers is 15,000 psi. This value was obtained from the tabulated values in Section III, Attachment 3A.I for "S_h" at a design temperature of 340°F, for carbon steel SA 155 KCF 70 and SA 106, GR.C.

<u>Allowable Stress Limits (Equation 9 of NC-3652 and NC-3611, Reference 3A.4.2-1)</u>

The stress includes the primary membrane plus the primary bending stresses. The limits of these stresses depend on the loading conditions as follows :

a. The limit of stress under the upset condition is $1.2S_h = 18{,}000$ psi,
b. The limit of stress under the emergency condition is $1.8S_h = 27{,}000$ psi, and
c. The limit of stress under the faulted condition is $2.4S_h = 35{,}000$ psi.

3A.4.2.3.3 Method of Analysis

Downcomers were analyzed for the appropriate loading combination using the computer program ADLPIPE (Attachment 3A.F).

The mathematical model for the downcomer is a vertical pipe anchored at the underside of the drywell floor and guided at the downcomer bracing system. The inertia effect of water surrounding the submerged portion of the downcomer was obtained by the addition of a virtual mass of water distributed along the submerged portion. The mass of water inside the submerged portion of the downcomers was conservatively considered in the model for all dynamic loadings. The SRV discharge lines were incorporated in the model of the 28-in. downcomer.

3A.4.2.3.3.1 Static Analysis. Static analysis techniques were used to determine the stresses due to dead weight, internal pressure, thermal and hydrodynamic loads using an equivalent static pressure load, or an appropriate DLF, as shown in Figure 3A.3.1-10.

3A.4.2.3.3.2 Response Spectrum Analysis. The response spectrum method of analysis was performed, for seismic and hydrodynamic loads, using the ADLPIPE program. Modal responses were combined in accordance with Regulatory Guide 1.92 while damping values were selected per Regulatory Guide 1.61.

Spatial components were combined by the SRSS method, with the exception of seismic which used the higher of the absolute sum of (a) north-south and vertical or (b) east-west and vertical (see Section 3.7 of the FSAR).

3A.4.2.3.4 Results and Design Margin

The downcomers were analyzed for all load combinations described in Section 3A.3.5.4. The stresses in the 24-in. OD and 23-in. OD downcomers pipe show that they are structurally adequate for all plant operating conditions. The design margins for the 24-in. and 28-in. downcomers in each criteria category are summarized below. The lowest design margin is shown to be 1.08.

24-in. Downcomer

Acceptance Criteria From Table 3A.3.5-5	Allowable Stress	Calculated Stress	Design Margin
Upset	18,000	15,513	1.16
Emergency	27,000	15,709	1.72
Faulted	36,000	18,810	1.91

28-in. Downcomer

Acceptance Criteria From Table 3A.3.5-5	Allowable Stress	Calculated Stress	Design Margin
Upset	18,000	16,654	1.08
Emergency	27,000	16,744	1.61
Faults	36,000	18,371	1.96

Design margin is defined as follows:

$$DM = \frac{\text{Allowable Stress}}{\text{Calculated Stress}}$$

3A.4.2.3.5 Fatigue Evaluations

The fatigue evaluation presented below was an NRC request and is not an ASME requirement.

The fatigue evaluation of 24-in. and 28-in. downcomer lines in the wetwell air volume was performed using ASME Section III, Class 1 rules (NB-3600). A governing loading scenario, based on the DFFR (Reference 3A.3.2-1), was developed. The loadings which were evaluated are:

a. Internal pressure,
b. Thermal expansion and transients,
c. Seismic,
d. Pressure differential effects between drywell and suppression chamber,
e. SRV pool load and building response, and
f. Chugging pool load and building response.

Equivalent numbers of fatigue cycles were determined for dynamic loads. The 24-in. and 28-in. downcomers were analyzed for the appropriate load combinations and their associated number of cycles as presented in Table 3A.4.1-3. The combined stresses and corresponding equivalent stress cycles were computed to obtain the fatigue usage factor. The maximum fatigue usage factor for both downcomers are presented in Tables 3A.4.2-4 and 3A.4.2-5.

3A.4.2.4 Safety/Relief Valve Piping Systems

The MSRV piping in the suppression chamber consists of 18 independent piping systems, each comprised of 10-in. and 19-in. OD Schedule 80 carbon steel pipe. The wetwell portion of each SRV piping system in the wetwell originates from a 28-in. downcomer (anchor point),

which penetrates at el. 493 ft 0 in., and then runs horizontally for a sufficient length to provide enough thermal flexibility. The horizontal run also allows the quenchers to be distributed evenly about the suppression pool. The piping then drops vertically downward to the quencher body which is bolted to the quencher support at el. 447 ft 0 in. The quencher support is modeled as an integral part of the SRV piping system and as such has flexibility taken into account. A schematic showing a typical SRV piping layout is shown in Figure 3A.4.2-6. Lateral guides are provided at the downcomer bracing.

3A.4.2.4.1 Loads Used for Assessment

The SRV piping systems are subjected to static, dynamic, and hydrodynamic loads due to normal, upset, emergency, and faulted plant operating conditions. The loading cases and combinations are described in Section 3A.3.5.4.

The hydrodynamic loads resulting in significant effects on the SRV piping are listed below. For a description of these loads see Section 3A.4.2.3.1.

a. Deadweight (w),
b. Thermal,
c. Pressure (P),
d. OBE,
e. SSE,
f. SRV pressure (Figure 3A.3.1-9),
g. SRV building response,
h. SRV blowdown,
i. LOCA jet (Figure 3A.3.2-18 and Table 3A.3.2-8),
j. LOCA bubble (Figure 3A.3.2-18 and Table 3A.3.2-8),
k. Chugging drag (Figure 3A.3.2-19), and
l. Chugging building response.

The SRV pressure used in the analysis is applied in the same manner as described in Section 3A.4.2.3.1 for the downcomers. The building response loads are based on the spectra described in Section 3A.5.

3A.4.2.4.2 Load Combination and Acceptance Criteria

The stresses within the SRV piping are acceptable if they satisfy the ASME Boiler and Pressure Vessel Code, Subsection NC. The allowable stress "S_h" used for primary stress evaluation is 15,000 psi. This value was obtained from the tabulated values in Section III, Attachment 3A.I for "S_h" at a design temperature of 475°F for carbon steel.

3A.4.2.4.3 Allowable Stress Limits (Equation 9 of NC-3652 and NC-3611, Reference 3A.4.2-1)

The stress for the SRV piping includes the primary membrane plus the primary bending stresses. The limits of these stresses depend on the loading conditions as follows:

a. The limit of stress under the upset condition is $1.2S_h$ = 18,000 psi,
b. The limit of stress under the emergency condition is $1.8S_h$ = 27,000 psi, and
c. The limit of stress under the faulted condition is $2.4S_h$ = 36,000 psi.

3A.4.2.4.4 Method of Analysis

The SRV piping was analyzed for the appropriate loading combinations (Table 3A.3.5-5) using the computer programs ADLPIPE and ANSYS (Attachment 3A.F). Analysis was performed for all 18 SRV lines. The quencher towers were included in the piping models to account for quencher tower flexibility. The inertial effects of water were accounted for in the same matter as described in Section 3A.4.2.3.3. Analysis results are summarized in Section 3A.4.2.4.5.

3A.4.2.4.4.1 Static Analysis. Static analysis techniques were used to determine the stresses due to dead weight, internal pressure, thermal and hydrodynamic loads using an equivalent static pressure load or an appropriate DLF, as shown in Figure 3A.3.1-10.

3A.4.2.4.4.2 Response Spectrum Analysis. The ADLPIPE program was utilized to perform dynamic response spectra analyses. Modal responses were combined in accordance with Regulatory Guide 1.92 while damping values were selected per Regulatory Guide 1.61.

3A.4.2.4.4.3 Time History Analysis. The ANSYS program was utilized as required for critical SRV lines in order to obtain more realistic piping response for SRV building response loads. For SRV blowdown transient loads, ADLPIPE was used to perform a force time history analysis.

3A.4.2.4.5 Results and Design Margin

The calculated stresses for the design configuration of all SRV piping systems show that the piping is structurally adequate for all plant operating conditions. The maximum calculated stresses and the resulting minimum design margins for the SRV piping systems for the controlling load combinations from Section 3A.3.5.4 are shown below.

Acceptance Criteria from Table 3A.3.5-5	Allowable Stress (psi)	Calculated Stress (psi)	Design Margin
Upset	18,000	17,206	1.05
Emergency	27,000	17,207	1.57
Faulted	36,000	17,280	2.08

Design margin is defined as follows:

$$DM = \frac{\text{Allowable Stress}}{\text{Calculated Stress}}$$

3A.4.2.4.6 Fatigue Evaluations

The fatigue evaluation presented below was an NRC request and is not an ASME requirement.

The fatigue evaluation on all 18 SRV lines in the wetwell air volume was performed using ASME Section III, Class 1 rules (NB-3600). A governing loading scenario, based on the DFFR (Reference 3A.3.2-11), was developed. The loadings which were evaluated are:

a. Internal pressure,
b. Thermal expansion and transients,
c. Seismic,
d. SRV blowdown,
e. SRV pool load and building response, and
f. Chugging pool load and building response.

Equivalent numbers of fatigue cycles were determined for dynamic loads. All 18 SRV discharge lines in the wetwell region were analyzed for the appropriate load combinations and their associated number of cycles as presented on Table 3A.4.1-3. The combined stresses and corresponding equivalent stress cycles were computed to obtain the fatigue usage factor. The maximum fatigue usage factor for all 18 SRV discharge lines in the wetwell air volume was found to be below ASME allowable limits. The results of the maximum usage factors is presented on Table 3A.4.2-6.

3A.4.2.5 Quencher

Quenchers have been installed on the discharge end of the SRV lines to reduce air clearing loads and to promote effective heat transfer between the suppression pool water and the discharging steam-air mixture during SRV actuation. The quenchers are an integral part of the SRV piping system and are bolted to the quencher support at the base plate. The quencher support assessment is included in the SRV piping system assessment, Section 3A.4.2.4.

3A.4.2.5.1 Loads Used for Assessment

The quenchers, in common with the other piping components, are subjected to static, dynamic, and hydrodynamic loads due to normal, upset, emergency, and faulted plant conditions. The loading cases arc described in Section 3A.3.5.4.

The mechanical loads are from Table 3-16 of the DFFR (Reference 3A.4.2-2) and modified to account for CGS plant specific conditions. The load from DFFR Table 3-16, which are not plant specific, are the quencher arm and body loads arising from the SRV water and air clearing transients. These generic bounding loads are described in the DFFR (Reference 3A.4.2-2). The other loads on Table 3A.3-16 of the DFFR are modified to account for (a) quencher arm loads caused by the various submerged hydrodynamic loading described in Section 3A.3, and (b) the static and dynamic loads resulting from the SRV piping system analysis described in Section 3A.4.2.4.

3A.4.2.5.2 Load Combination Acceptance Criteria

The assessment of the quenchers for the plant loads is performed in accordance with ASME Section III, Subsection NC (Reference 3A.4.2-1). The code stamp and hydrotest requirements are not applicable since the quencher is not a pressure retaining component. The code jurisdiction ends at the weld between the SRV discharge piping and the quencher inlet nozzle (12 in. x 24 in. reducer).

3A.4.2.5.3 Evaluation

The quencher body and the quencher arms are examined to determine their adequacy for conditions of loading described above.

The quencher body together with the quencher arms were modeled through finite element program ANSYS. The model uses a quadrilateral shell element which has both bending and membrane capabilities. The element has six degrees of freedom at each node. This element also has an option for variable thickness. The modeled structure is shown in Figures 3A.4.2-7 and 3A.4.2-8.

Element loading capabilities include surface temperatures and pressures. Also, concentrated loads can be applied at each node point. The significant loads affecting quencher body and the quencher arms are

a. The loads arising from the SRV water and air clearing transients,

b. The SRV loads caused by pool velocity and acceleration fields from an adjacent firing quencher, and

c. SRV induced building response loads.

The quencher assembly was analyzed statically with the load vectors related to type of loads as discussed above and for load combinations presented in Section 3A.4.2.5.2. Since the quencher has linear properties the superposition of load combinations was used for evaluation of results.

The calculated stress intensities for the various loading conditions are presented in Table 3A.4.2-3. The tabulated design margins for the governing upset condition indicate the quencher assembly is structurally adequate.

3A.4.2.6 Platforms and Ladders

The assessment of the capacities of the platforms and ladder relative to load combinations involving suppression pool hydrodynamic loads is made in this section.

3A.4.2.6.1 Loads Used for Assessment

Complete description of all hydrodynamic loads has been given in Section 3A.3. The loads used for the assessment of the platforms and ladder are discussed below.

3A.4.2.6.1.1 Safety/Relief Valve Operation Loads. No direct loads on the platforms at el. 472 ft 4 in. and el. 486 ft 8 in. or the ladder between el. 472 ft 4 in. and 490 ft 1 in. result from operation of the SRV system. However, building response to dynamic pressures at the pool boundary during SRV discharge result in dynamic stresses in the platform and ladder components which are supported from the steel containment structure.

3A.4.2.6.1.2 Loss-of-Coolant Accident Loads. As discussed in Section 3A.3.2.3, a LOCA results in pool swell impact and drag loads and fallback drag loads on the platform at el. 472 ft 4 in. and the ladder between el. 472 ft 4 in. and el. 484 ft 4.75 in. Also, the ladder and the knee braces below the platform at el. 472 ft 4 in. are subjected to a horizontal lift load caused by both pool swell and fallback.

Additional LOCA loads include building responses to condensation oscillation and chugging which are obtained from response spectra at tile points of attachment of the platforms and ladder to the containment vessel. Since the condensation oscillation load is bounded by the chugging load, no separate platform assessment has been performed for the condensation oscillation loading condition.

3A.4.2.6.1.3 Other Significant Loads. Other loads which result in significant stresses in the platform and ladder components are dead loads, live loads, and seismic accelerations.

Dead loads include the weights of the platforms and ladder. Live loads include personnel on the platforms and ladder, equipment weights on the platforms, and the monorail loads on the platform at el. 486 ft 8 in. Seismic accelerations are obtained from seismic response spectra at the points of attachment of the platforms and ladder to the containment vessel.

3A.4.2.6.2 Controlling Load Combinations

The load combination criteria for structures internal to the suppression chamber (see Section 3A.3.5.3) are applicable to the platforms and ladder. In particular, the combinations for steel structures using the elastic working stress method are investigated for both service load conditions and factored load conditions.

3A.4.2.6.3 Acceptance Criteria

The acceptable stress level for the platform and ladder components using the elastic working stress method is as defined for the loading combinations in Section 3A.3.5.3.

3A.4.2.6.4 Method of Analysis

Both platforms and the ladder were investigated for the effects of the loading combinations in Section 3A.3.5.3. The components of the platform at el. 472 ft 4 in. and the ladder were analyzed individually, whereas the platform at el. 486 ft 8 in. is above the effects of pool shell and therefore was not reanalyzed. As noted in Section 3A.3.2.3, the grating does not sustain impact loads. Therefore, the only significant load on the grating is the drag load component. The supporting beams and bracing members were analyzed for drag and lift pool swell/fallback loads. The portion of the ladder between el. 472 ft 4 in. and 404 ft 4.75 in. was analyzed for impact loading on the highest rung in the pool swell region and drag and lift on all rungs below the rung with impact.

3A.4.2.6.5 Results

The governing load combination (5a) for steel structures is given below with corresponding input for the platform at el. 472 ft 4 in. and the portion of the ladder between el. 472 ft 4 in. and 484 ft 4.75 in.:

(5a) $1.6S \geq 1.0D + 1.0L + 1.0E_o + 1.0P_A + 1.0T_A + 1.R_A + 1.0P_{SR}$

$D = 0.10$ psi

$L = 0$

$E_O = 0.11$ psi

P_A = 8.4 psi (drag on gross area of grating)

= 25.0 psi (drag on ladder rungs and bracing members)

= 44.0 psi (impact on ladder rungs)

= small (horizontal lift pressure on supporting bars and handrail members)

= 11.0 psi (horizontal lift pressure on bracing members)

= 5.0 psi (horizontal lift pressure on ladder rungs)

= small (impact and drag on supporting bars and handrail members)

T_A = 0

R_A = 0

P_{SR} = 0.05 psi

where D, L, E_o, P_A, T_A, R_A, and P_{SR} are as defined in Section 3A.3.5.2.

The portion of the ladder between el. 472 ft 4 in. and 484 ft 4.75 in. was found to have sufficient capacity to withstand the governing load combination with a design margin (i.e., ratio of the allowable stress to the maximum absolute calculated stress) of 2.15. However, the existing platform at el. 472 ft 4 in. was found to be deficient (under the above load combination. The critical component is the grating under upward load. To make the grating sufficient, every other bearing bar (instead of every fourth as is in the original design) has been welded to all supporting members including the platform member supporting the ladder. With this reinforcement a design margin for the grating of 1.80 is attained. Also, two platform members have been reinforced, and the rectangular bracing members have been replaced with circular members. With this reinforcement the critical supporting member has a design margin of 1.18.

3A.4.2.7 References

3A.4.2-1 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, “Class 2 Components,” American Society of Mechanical Engineers, 1971 through Winter 1973 Addenda.*

* Faulted conditions appeared first in Winter 1976 Addenda.

3A.4.2-2 Mark II Containment Dynamic Forcing Functions Information Report (DFFR), NEDO-21061, Revision 2, September 1976.

Table 3A.4.2-1

Downcomer Bracing System Controlling Design Margins

Component	Service Load Conditions Maximum Horizontal Loading	Factored Load Maximum Horizontal Loading	Conditions Maximum Vertical Loading
6 in. D.E.S. pipe	3.45		2.96
4 in. D.E.S. pipe	2.5		1.68
Pedestal connection	2.00	2.02	
Vessel connection	5.26		2.75
Downcomer ring	1.92	1.95	
SRV line ring	3.39	4.12	

Notes:

1. For service load conditions, load combination (1) controls. Only maximum horizontal loading is applicable.

2. For factored load conditions, load combination (5) controls. Only the design margin for the controlling loading is listed.

Table 3A.4.2-2

Controlling Stress Resultants in Column

	Base Moment (ft kips)	Shear (kips)		Axial Force[a] (kips)	
		Base	Top	Base	Top
Safety/Relief Valve Actuation					
All valves initial	771.1	70.2	41.4	±102.1	±86.4
1 valve subsequential	1133.0	94.9	48.9	±37.1	±37.1
Loss-of-Coolant Accident					
Chugging	13.5	2.5	1.4	±11.7	±10.2
Annulus pressurization	136.4	11.1	6.5	±6.4	±5.3
Pipe break jet				±745.0	±745.0
Pool swell				-172.0	-172.0
Transient pressure difference (drywell-wetwell)				+642.0	+642.0
Seismic					
Operating basis earthquake	127.1	9.9	5.7	±31.2	±27.1
Safe shutdown earthquake	164.2	12.8	7.4	±46.1	±39.6
Dead				+229.0	+134.0
Live				+217.0	+217.0

[a] Positive axial force is compressive; negative is tensile.

Table 3A.4.2-3

Results Summary - Safety/Relief Valve Quencher

Loading Condition	Stress Category	Principal Stress (psi) 1	2	3	Critical Stress Intensity (psi)	Allowable Stress (psi)	D.M. (1)
Normal	P_m	2525	1390	-275	2800	15,978	5.71
	P_L	11,092	1204	-275	11,367	23,967	2.11
	$P_L + Q$	13,678	1052	0	13,678	47,934	3.50
Upset	$P_L + Q$	28,399	6985	0	38,399	47,934	1.25
	P_m	4102	535	-275	4377	17,576	4.01
	P_L	13,480	2540	-275	13,755	26,364	1.92
Emerge	P_m	3989	1392	-275	4264	23,967	5.62
	P_L	16,964	1711	-275	17,239	28,761	1.67
Fault (1)	P_m	3994	1395	-275	4267	31,956	7.48
	P_L	16,982	1712	-275	17,257	38,347	1.99
Fault (2)	P_m	3334	1223	-275	3609	31,956	8.85
	P_L	14,738	1633	-275	15,013	33,347	2.55

[a] Generally the maximum stresses occurs between the quencher arms, where the arms meet the body.

Notes:

1. Design margin is defined as follows:

$$DM = \frac{\text{Allowable Stress}}{\text{Calculated Stress}}$$

2. The definition of the above terms is proved in ASME Code Section III, Paragraph NC-3217 of Winter 1976 Addenda.

Table 3A.4.2-4

Cumulative Usage Factor Calculation at 24 in. Downcomer Anchor

Load Combination[a] Set	Expected Number of Cycles (ni)	Bending Moment Mi lb/ft	Peak Stress Sp (psi)	Alt. Stress Salt (psi)	Allowable Number of Cycles N	Calculated Usage Factor $U_i = \frac{n_i}{N_i}$
1	1	244,008	60,116	30,058	20,000	0.0001
2	9	244,008	32,575	16,288	200,000	0.0001
3	50	202,180	26,991	13,496	300,000	0.0002
4	940	216,065	28,845	14,422	300,000	0.0031
5	12,434	183,244	24,463	12,232	400,000	0.0311
				Cumulative Usage Factor U= 0.0346 < 1.0		

[a] Load combination set definition

1 - NPC + LOCA + SSE + SRV + CHUGGING

2 - SSE + SRV + CHUGGING

3 - OBE + SRV

4 - SRV + CHUGGING

5 - SRV

Table 3A.4.2-5

Cumulative Usage Factor Calculation at 28 in. Downcomer Anchor

Load Combination[a] Set	Expected Number of Cycles (ni)	Bending Moment Mi lb/ft	Peak Stress Sp (psi)	Alt. Stress Salt (psi)	Allowable Number of Cycles N	Calculated Usage Factor $Ui = \frac{ni}{Ni}$
1	1	346,771	61,562	30,781	18,000	0.0001
2	9	346,771	33,775	16,888	150,000	0.0001
3	50	307,359	29,937	14,968	200,000	0.0002
4	940	557,714	54,321	27,161	30,000	0.0313
5	12,434	275,671	26,850	13,425	400,000	0.0311
				Cumulative Usage Factor U= 0.0629 < 1.0		

[a] Load combination set definition

1 - NPC + LOCA + SSE + SRV + CHUGGING

2 - SSE + SRV + CHUGGING

3 - OBE + SRV

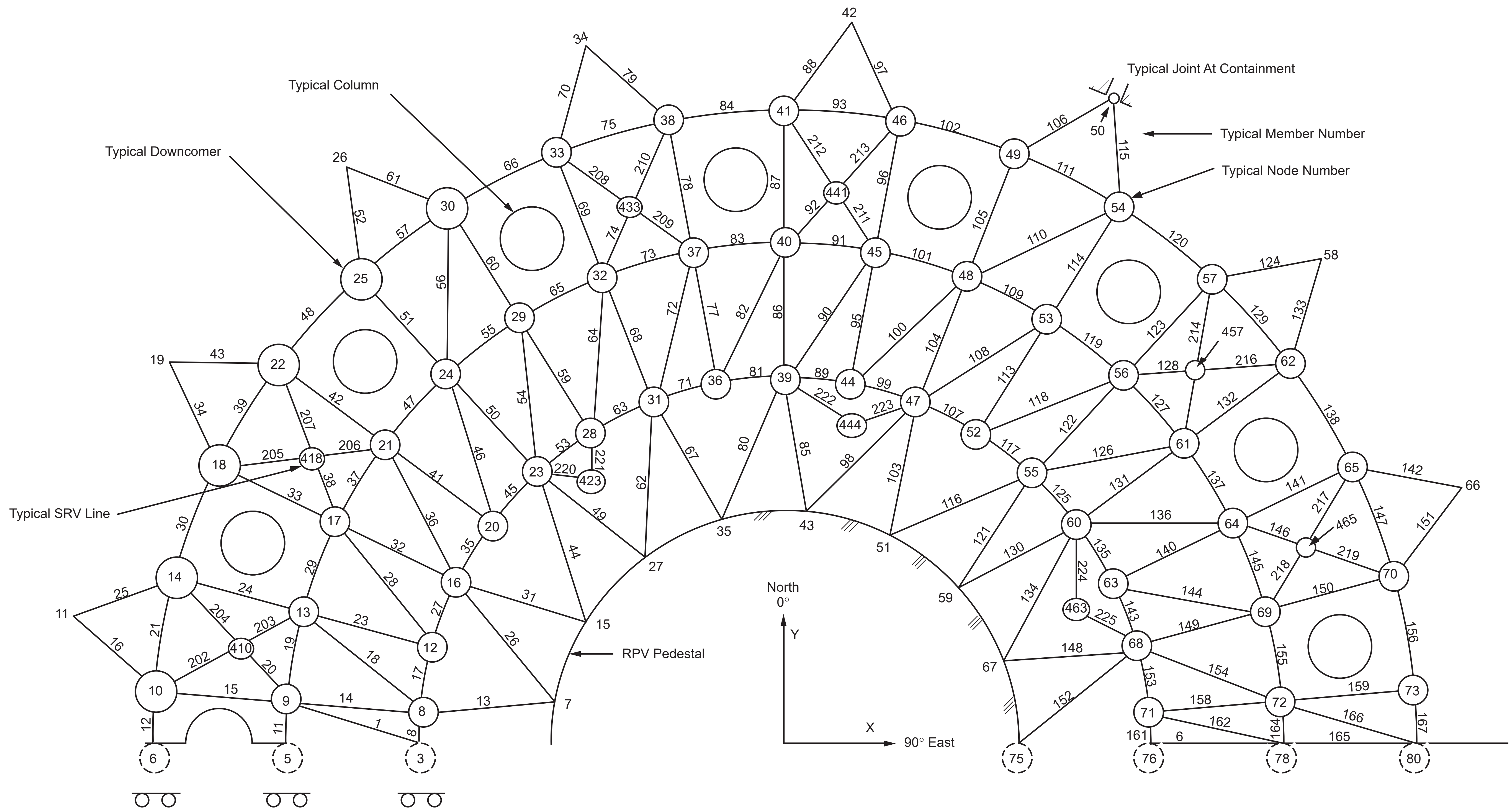
4 - SRV + CHUGGING

5 - SRV

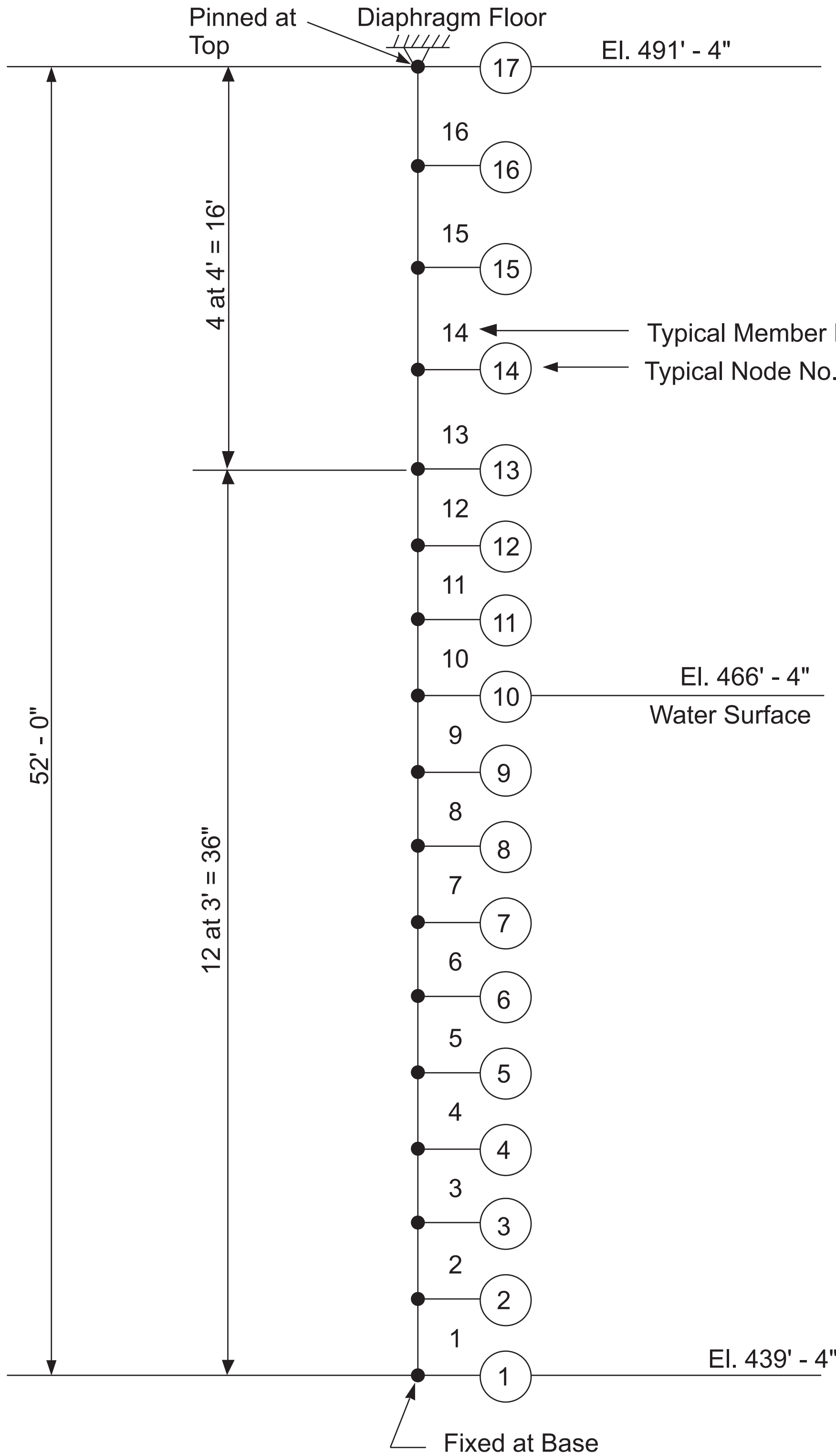
Table 3A.4.2-6

Maximum Usage Factors Table

	First Actuation	Second Actuation	Subsequent Actuations	Total Cumulative Usage Factor
Low set SRV lines (MSRV-1C) (MSRV-1B)	0.626	0.033	0.237	0.896
Non-low-set SRV lines	0.527	0.202	N/A	0.729



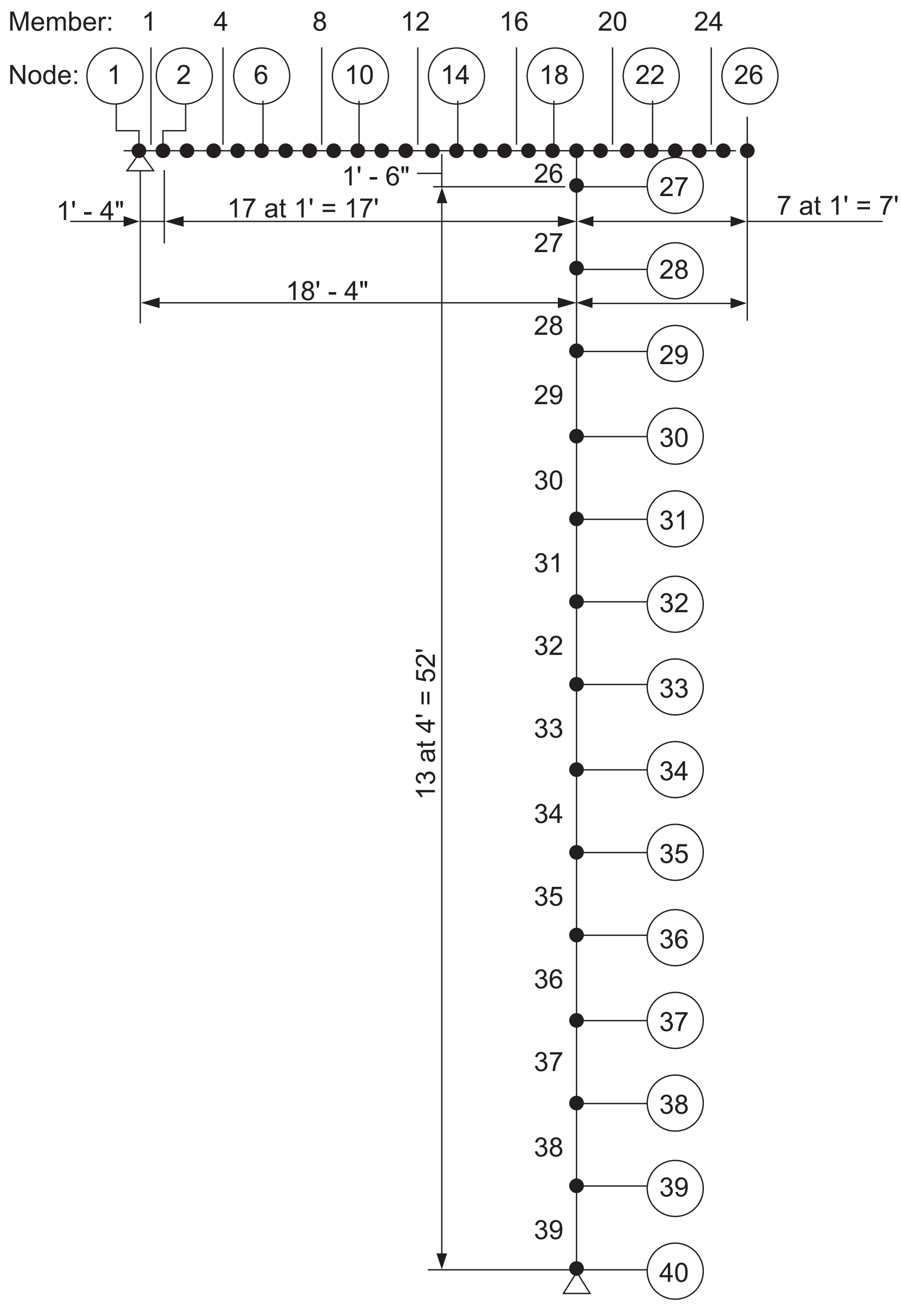
Columbia Generating Station Final Safety Analysis Report	**Downcomer Bracing System - Model For Structural Analysis**		
	Draw. No. 950021.61	Rev.	Figure 3A.4.2-2



Model for Horizontal Loads

Diaphragm Floor Column Model for Dynamic Analysis

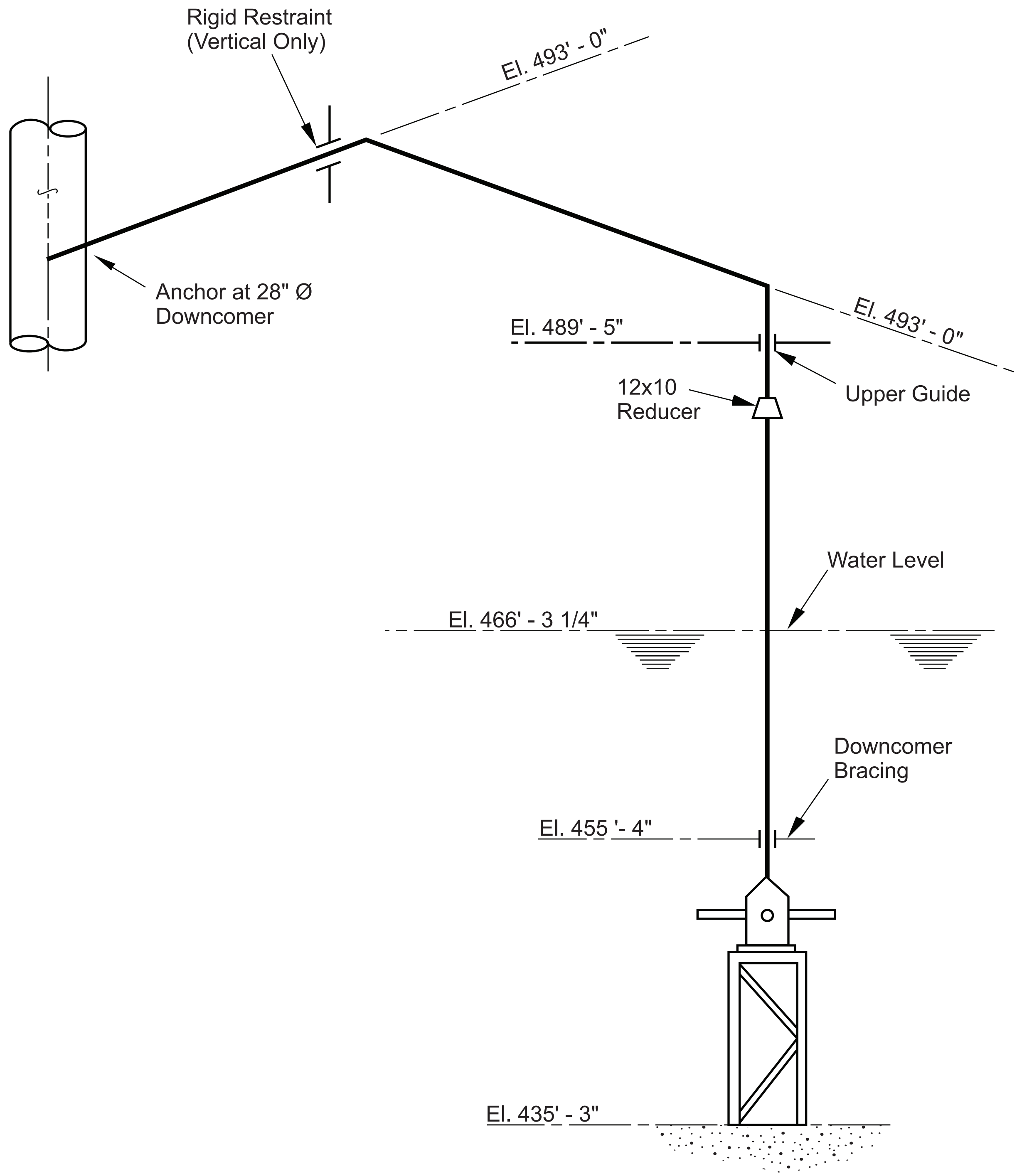
Draw. No. 950021.62	Rev.	Figure 3A.4.2-4



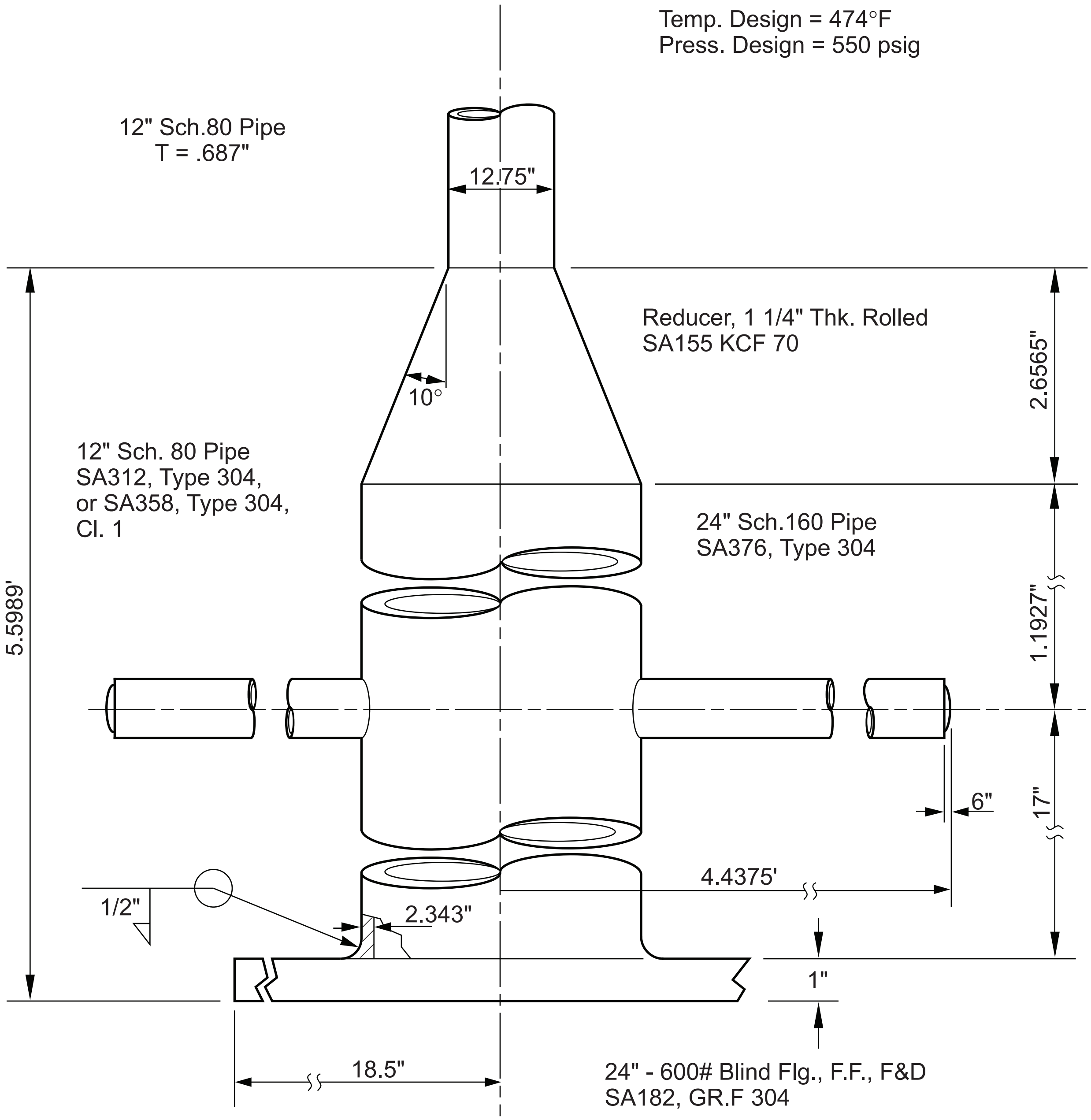
Model for Vertical Loads

Structural Model of Diaphragm Floor Beam and Column

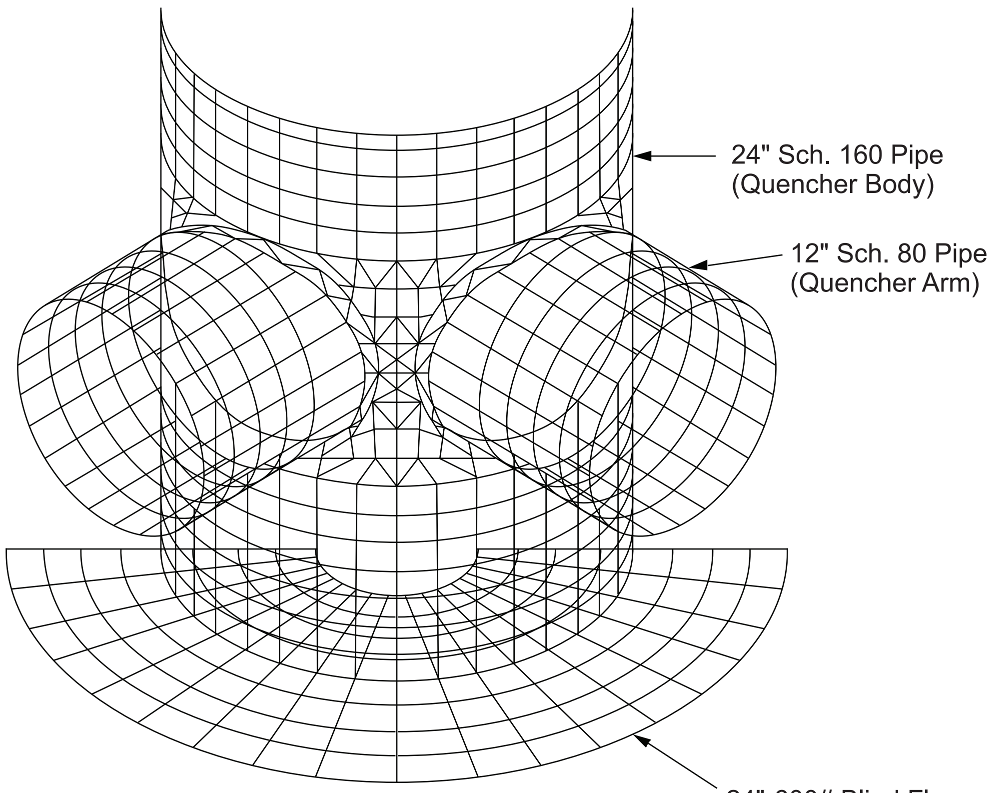
Draw. No. 950021.63 | Rev. | Figure 3A.4.2-5



Columbia Generating Station Final Safety Analysis Report	**SRV Piping System Inner Ring Quencher Support**
	Draw. No. 960222.91 | Rev. | Figure 3A.4.2-6



Quencher Assembly



SRV Quencher Assembly

Draw. No. 960222.93 | Rev. | Figure 3A.4.2-8

3A.4.3 MISCELLANEOUS SUPPRESSION POOL PIPING SYSTEMS

A tabulation of the miscellaneous systems located in the suppression chamber is given in Table 3A.4.3-1. The results of the stress analysis for suppression pool piping other than SRV piping and downcomers is presented in Table 3A.4.3-3.

Depending upon the location in the wetwell, the suppression pool piping will be subjected to different loading associated with SRV discharge and LOCA. For identification purposes, the miscellaneous wetwell piping has been broken into four zones, i.e., fully submerged piping, partially submerged piping, piping in the pool swell zone, and piping above the pool swell zone (Figure 3A.4.3-1). Piping in each zone is noted in Table 3A.4.3-1.

3A.4.3.1 Loads Used for Assessment

The wetwell piping systems are subjected to static, dynamic, and hydrodynamic loads due to normal, upset, emergency, and faulted plant operating conditions. Each zone is characterized by certain applicable loads shown in Table 3A.4.3-2. A description of each of these loads is provided in Sections 3A.3 and 3A.5.

3A.4.3.2 Load Combination and Acceptance Criteria

The design limits, as set forth in the ASME Boiler and Pressure Vessel Code (ASME Code) Section III, Subsection NC (Reference 3A.4.2-1) were utilized for the assessment of the suppression pool piping systems. The various piping systems are considered acceptable if they satisfy the equations of Paragraph NC-3652 of Section III of the ASME Code.

The piping, as listed on Table 3A.4.3-1, is fabricated of low alloy carbon steel pipe having an allowable stress "S_h" primary evaluation of 15,000 psi up to a temperature of 275°F.

Allowable Stress Limits (Equation 9 of NC-3652 and NC-3611, Reference 3A.4.2-1)

The stress for the miscellaneous wetwell piping includes the primary membrane plus bending stresses. The limits of these stresses, depending on loading conditions, are the same as those listed in Section 3A.4.2.4.3.

3A.4.3.3 Method of Analysis

The miscellaneous suppression pool piping systems were analyzed for the appropriate loading combinations using ADLPIPE and ANSYS computer programs (Attachment 3A.F). Hand calculation methods were used to analyze short cantilevered pipes attached to the primary containment. Static and response spectrum analyses were handled in the same manner as described in Section 3A.4.2.3.3. However, displacement time history analyses were performed, as required, to obtain more realistic piping responses due to SRV building response

loads. For residual heat removal (RHR) blowdown transient loads, ADLPIPE was used to perform force time history analysis.

3A.4.3.4 Results and Design Margins

Table 3A.4.3-3 presents pipe stress results and design margins for piping systems in the wetwell. The pipe supports for the wetwell were also assessed and determined to be within acceptable limits.

Table 3A.4.3-1

Miscellaneous Wetwell Piping

Ref. Penetration Number X-	Line/Description	Penetration Elevations	Penetration Azimuths	Remarks
Zone I - Fully Submerged Piping				
33	8 in. RCIC Pump Suction	452 ft-0 in.	336°	
100	6 in. Fuel pool cleanup	452 ft-0 in.	295°	
35	24 in. RHR pump "A" suction	452 ft-0 in.	276° - 49 ft	Similar in config. to X-32
32	24 in. RHR pump "B" suction	452 ft-0 in.	263° - 11 ft	
31	24 in. HPCS pump suction	452 ft-0 in.	97° - 36 ft	
34	24 in. LPCS pump suction	452 ft-0 in.	66° - 05 ft	Similar in config. to X-36
36	24 in. RHR pump C suction	452 ft-0 in.	45° - 39 ft	
88	3 in. instrumentation	455 ft-0 in.	180°	
86B	2 in. instrumentation	462 ft-0 in.	45°	Short length of pipe
87B	2 in. instrumentation	462 ft-0 in.	245°	Short length of pipe
N/A	4 in. FPC - SYPHON	N/A	N/A	
Zone II - Partially Submerged Piping				
64	1.5 in. RCIC vacuum pump discharge	467 ft-9 in.	345°	Similar in config. to X-65
4	10 in. RCIC line (24 in. header)	467 ft-9 in.	318°	
65	2 in. RCIC pump min. flow	467 ft-9 in.	312°	
101	6 in. RPC test	467 ft-9 in.	0°	
47	18 in. RHR pump A test	467 ft-9 in.	288.7°	Similar in config. to X-48
117	18 in. RHR pressure relief	467 ft-9 in.	279°	Similar in config. to X-11B
118	18 in. RHR pressure relief	467 ft-9 in.	261°	
48	18 in. RHR pump B test	467 ft-9 in.	251.29°	
23	3 in. EDR equipment drain	467 ft-9 in.	132°	
24	3 in. FDR floor drain	467 ft-9 in.	111°	
49	12 in. HPCS pump test	467 ft-9 in.	103° - 31 ft	
63	12 in. LPCS pump test	467 ft-9 in.	60°	Similar in config. to X-49
26	18 in. RHR pump "C" test	467 ft-9 in.	20°	

Table 3A.4.3-1

Miscellaneous Wetwell Piping (Continued)

Ref. Penetration Number X-	Line/Description	Penetration Elevations	Penetration Azimuths	Remarks
Zone III - Pool Swell Zone				
87A	2 in. instrumentation	471 ft-6 in.	245°	Short length of pipe
86A	2 in. instrumentation	471 ft-6 in.	45°	Short length of pipe
51	42 in. HATCH	473 ft-6 in.	155°	Short length of pipe
107B	12 in. electrical line	475 ft-0 in.	240°	Short length of pipe
107A	12 in. electrical line	475 ft-0 in.	52° - 30 ft	Short length of pipe
66	24 in. CSP	475 ft-0 in.	222° - 30 ft	Short length of pipe
116	12 in. RCIC	477 ft-6 in.		Short length of pipe
81	14 in. instrumentation	479 ft-4 in.	235°	Short length of pipe
82	10 in. CAS	479 ft-4 in.	230°	
83	10 in. instrumentation	479 ft-4 in.	240°	
84	10 in. instrumentation	479 ft-4 in.	40°	Similar in config. to X-83
Zone IV - Above Pool Swell Zone				
67	24 in. CEP	491 ft-0 in.	0°	Short length of pipe
102	6 in. Plugged (4 in. CAC pipe)		240°	Short length of pipe
103	6 in. Plugged (4 in. CAC pipe)		180°	Short length of pipe
104	6 in. Plugged (4 in. CAC pipe)		323°	Short length of pipe
105	6 in. Plugged (4 in. CAC pipe)		140°	Short length of pipe
57	12 in. SPARE		47°	Short length of pipe
58	12 in. Hardened Containment Vent		132°	
59	12 in. SPARE		258°	Short length of pipe
60	12 in. SPARE		280°	Short length of pipe
25A	6 in. RHR		70°	
25B	6 in. RHR		235° - 30 ft	
119	24 in. vacuum breaker		151°	Short length of pipe

Table 3A.4.3-2

Piping Zone Versus Loads

Loading	Zone			
	Fully Submerged	Partially Submerged	Pool Swell Zone	Above Pool Swell Zone
Deadweight	X	X	X	X
Thermal	X	X	X	X
Pressure	X	X	X	X
Operating basis earthquake	X	X	X	X
Safe shutdown earthquake	X	X	X	X
SRV pressure	X	X		
SRV response spectra	X	X	X	X
LOCA jet	X	X		
LOCA bubble	X	X		
Pool swell and fallback	X	X	X	
Chugging pressure	X	X		
Chugging response spectra	X	X	X	X

Table 3A.4.3-3

Summary of Results and Design Margins for
Miscellaneous Wetwell Piping

Penetration Number	Controlling Load Case	Allowable Stress (psi)	Calculated Stress (psi)	Design Margin
X-33	Upset	18,000	12,264	1.46
X-100	Upset	18,000	5,231	3.44
X-32	Upset	18,000	9,923	1.81
X-31	Upset	18,000	14,096	1.28
X-36	Upset	18,000	11,687	1.54
X-4	Emergency	27,000	14,453	1.86
X-101	Faulted	36,000	19,826	1.81
X-47 and X-117	Emergency	27,000	16,952	1.59
X-23	Upset	18,000	13,963	1.28
X-49	Emergency	27,000	13,001	2.07
X-26	Emergency	27,000	19,580	1.37
X-24	Emergency	27,000	19,344	1.39
X-35	Upset	18,000	9,923	1.81
X-34	Upset	18,000	17,939	1.003
X-48 and X-118	Emergency	27,000	24,400	1.03[a]
X-58	Upset	18,000	7,948	2.26
		For Most Severe Condition		
X-104	Emergency	27,000	17,687	1.53
X-82	Upset	18,000	14,388	1.25
X-25A and X-25B	Upset	18,000	17,500	1.03[b]

Table 3A.4.3-3

Summary of Results and Design Margins for
Miscellaneous Wetwell Piping (Continued)

Penetration Number	Controlling Load Case	Allowable Stress (psi)	Calculated Stress (psi)	Design Margin
X-83	Faulted	44,640	33,084	1.35
X-84	Faulted	44,640	33,084	1.35
X-88	Upset	22,700	16,508	1.37
X-63	Faulted	36,000	6,220	5.78
X-64	Upset	18,000	15,193	1.19
X-65	Upset	18,000	14,271	1.26
4 in. FPC	Upset	18,000	2,774	6.49

[a] The design margin for this line in reality is larger because of the conservative approach taken, that the flexibility of the containment was not considered for the governing load in this case - RHR transient.

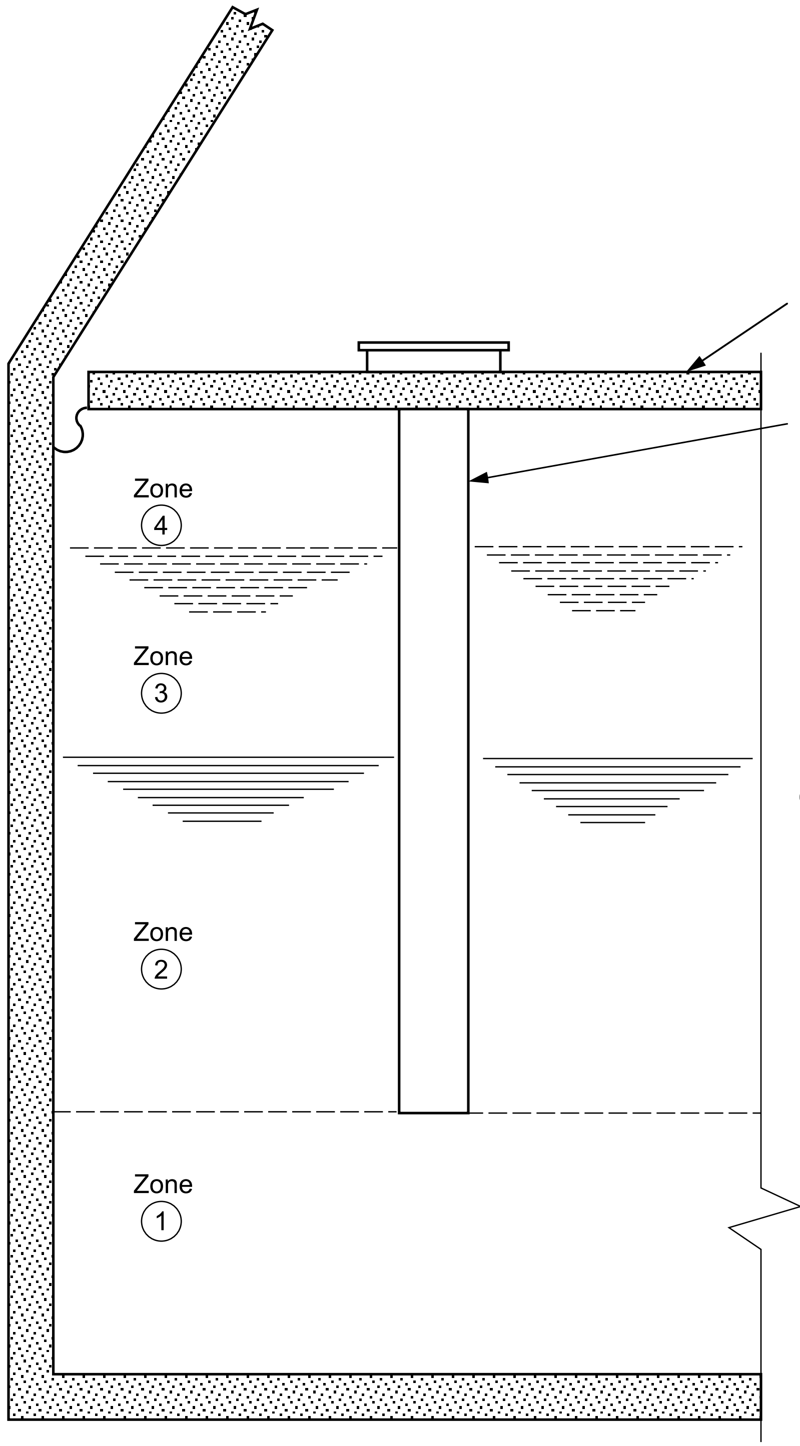
[b] The design margin for this line is also larger due to conservatism in the envelope spectrum analysis for SRV and chugging as compared to multi-input or time-history analysis that could have been utilized.

Notes:

1. The effects of short pipes cantilevered from the containment to hydrodynamic loads is considered minimal. These short stubs are listed in Table 3A.4.3-1.

2. The information provided within this table are the maximum stresses between the nozzle and the termination of the pipe within the wetwell.

3. Results of the stresses at the shell will be provided later.



Columbia Generating Station
Final Safety Analysis Report

Hydrodynamic Loading Zones

Draw. No. 960222.94 | Rev. | Figure 3A.4.3-1

Form No. 960690

3A.5 EFFECTS DUE TO BUILDING RESPONSES TO SAFETY/RELIEF VALVE DISCHARGE AND LOSS-OF-COOLANT ACCIDENT LOADS

Response spectra generated at each floor location define the input motions used for qualification and assessment of all the safety-related piping and equipment. In load combinations which include safety/relief valve (SRV) discharge and seismic loads, or loss-of-coolant accident (LOCA) steam condensation and seismic loads, seismic response spectra based upon a finite element soil-structure model were used in design and plant assessments for structures, piping, and equipment. The use of the finite element soil-structure model generally results in lower structural responses than the soil spring and dashpot model used in the original seismic design.

3A.5.1 BUILDING RESPONSES TO SAFETY/RELIEF VALVE DISCHARGE LOADS

This section presents the dynamic responses of the containment and internal structures subjected to loads resulting from SRV discharge as defined in Section 3A.3.1.3. The analytical model and method of analysis for determining the building structural response to SRV discharge loads are described in the following sections.

3A.5.1.1 Analytical Model

3A.5.1.1.1 Overall Building Model

Figure 3A.5.1-1a presents the soil-structure model. Figure 3A.5.1-1b presents the axisymmetric overall building model of the CGS reactor building. It should be noted that the thick reactor pressure vessel (RPV) pedestal and building mat were accounted for in the model by utilizing multiple layer axisymmetric solid elements.

These models were utilized to determine the response to the loading conditions stated in Section 3A.3, and provided responses at the RPV pedestal, foundation mat, biological shield wall, and in the reactor building walls and floors outside of primary containment.

3A.5.1.1.2 Steel Containment Shell Model

Figure 3A.5.1-2 shows the more refined three-dimensional finite element model of the containment shell. This model is interconnected to the rest of the building at the basemat (el. 446 ft), diaphragm floor (el. 503 ft), stabilizer truss (el. 565 ft), and the refueling bellows (el. 583 ft).

3A.5.1.2 Method of Analysis

The containment shell model was analyzed for SRV rigid wall pressure loads acting on the wetwell boundary and specified displacements at all the points where the shell is connected to

the rest of the building. The displacements were obtained from the solutions of the overall building model. The refined containment shell model, as discussed in Section 3A.5.1.1.2, was used to determine the response of the steel shell.

A more detailed discussion of the two models, analytical approach, and results are found in "SRV Loads - Improved Definition and Application Methodology" (Reference 3A.3.1-1).

3A.5.1.3 Safety/Relief Valve Discharge Load Cases

Several SRV discharge cases were considered for CGS design evaluation as discussed below.

3A.5.1.3.1 Response to All Valve Discharge

All 18 SRVs are conservatively assumed to discharge during certain plant conditions.

Two design conditions are associated with all the valves discharge case - the "axisymmetric" condition and the "nearly symmetric" condition. Each of these loading conditions is applied in load combinations involving the all valves discharge case. The axisymmetric loading condition assumes all valves will discharge simultaneously in the pool, thus maximizing the response of the axisymmetric features of the containment and reactor pedestal. The nearly symmetric loading condition assumes some imbalance may occur during actuation of all SRVs. The imbalance may occur from sequential discharging at different set points, variability in valve opening time, differences in SRV discharge line geometry, etc.

Listed below is a sample of the response spectra used in plant assessments involving the axisymmetric loading condition. For the "nearly symmetric" loading condition, the response spectra used are constructed by adding 0.5 times the axisymmetric response spectra to 0.6 times the single valve response spectra.

Location	Direction	Figure
Top of RPV Pedestal, el. 520 ft Mass No. 44	Radial	3A.5.1-3a, 3A.5.1-3b
Top of RPV Pedestal, el. 520 ft Mass No. 44	Vertical	3A.5.1-4a, 3A.5.1-4b
Basemat at RPV Pedestal, el. 435 ft Mass No. 141	Radial	3A.5.1-5a, 3A.5.1-5b
Basemat at RPV Pedestal, el. 435 ft Mass No. 141	Vertical	3A.5.1-6a, 3A.5.1-6b

Top of Sac. Shield Wall, el. 567 ft Mass No. 14	Radial	3A.5.1-7a, 3A.5.1-7b
Top of Sac. Shield Wall, el. 567 ft Mass No. 14	Vertical	3A.5.1-8a, 3A.5.1-8b
RPV, el. 545 ft Mass No. 27	Radial	3A.5.1-9a, 3A.5.1-9b
RPV, el. 545 ft Mass No. 27	Vertical	3A.5.1-10a, 3A.5.1-10b
Containment Vessel, el. 547 ft Mass No. 60600	Radial	3A.5.1-11a, 3A.5.1-11b
Containment Vessel, el. 547 ft Mass No. 60600	Vertical	3A.5.1-12a, 3A.5.1-12b
Containment Vessel, el. 448 ft Mass No. 50100	Radial	3A.5.1-13a, 3A.5.1-13b
Containment Vessel, el. 448 ft Mass No. 50100	Vertical	3A.5.1-14a, 3A.5.1-14b

Notes:

1. Figures denoted "a" refer to the conventional [single frequency pressure (SFP)] load case while the figures denoted "b" refer to the multiple frequency pressure (MFP) load case (see Reference 3A.3.1-1 for details).

2. The tangential loads were also utilized in the assessment of CGS. The tangential values are not included in this submittal as they are much smaller in magnitude then the presented valves.

3A.5.1.3.2 Automatic Depressurization System Valves Discharge Case

This case corresponds to the discharge of the SRVs of the automatic depressurization system (ADS) which are automatically actuated. For assessment purposes, the more conservative all valves response spectra values were utilized.

3A.5.1.3.3 Two Valves Discharge Case

In this case, two SRVs are considered to discharge concurrently through two adjacently located quenchers. For assessment purposes, the more conservative single outer valve discharge response spectra were utilized.

3A.5.1.3.4 Single Valve Discharge

The single outer quencher discharge, which is less likely to happen, is considered in the assessment because it is found to be more conservative, both for containment and pedestal response.

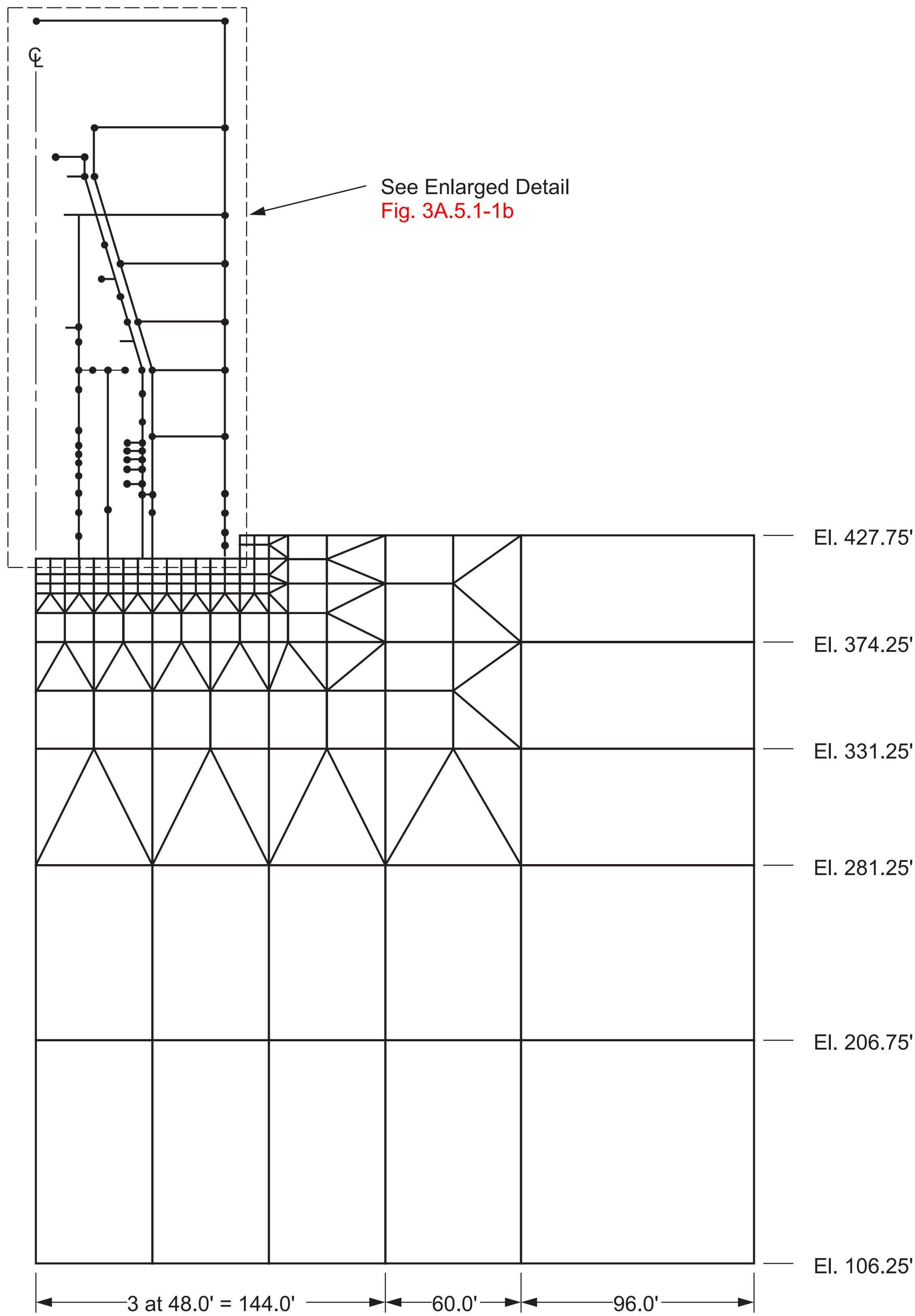
Location	Direction	Figure
Top of RPV Pedestal, el. 520 ft Mass No. 44	Radial	3A.5.1-15a, 3A.5.1-15b
Top of RPV Pedestal, el. 520 ft Mass No. 44	Vertical	3A.5.1-16a, 3A.5.1-16b
Basemat at RPV Pedestal, el. 435 ft Mass No. 141	Radial	3A.5.1-17a, 3A.5.1-17b
Basemat at RPV Pedestal, el. 435 ft Mass No. 141	Vertical	3A.5.1-18a, 3A.5.1-18b
Top of Sac. Shield Wall, el. 567 ft Mass No. 14	Radial	3A.5.1-19a, 3A.5.1-19b
Top of Sac. Shield Wall, el. 567 ft Mass No. 14	Vertical	3A.5.1-20a, 3A.5.1-20b
RPV, el. 545 ft Mass No. 27	Radial	3A.5.1-21a, 3A.5.1-21b
RPV, el. 545 ft Mass No. 27	Vertical	3A.5.1-22a, 3A.5.1-22b
Containment Vessel, el. 547 ft Mass No. 60600	Radial	3A.5.1-23a, 3A.5.1.23b
Containment Vessel, el. 547 ft Mass No. 60600	Vertical	3A.5.1-24a, 3A.5.1-24b

Containment Vessel, el. 448 ft Mass No. 50100	Radial	3A.5.1-25a, 3A.5.1-25b
Containment Vessel, el. 448 ft Mass No. 50100	Vertical	3A.5.1-26a, 3A.5.1.26b

Notes:

1. Figures denoted "a" refer to the conventional (SFP) load case while the figures denoted "b" refer to the MFP load case (see Reference 3A.3.1-1 for details).

2. The tangential loads were also utilized in the assessment of CGS. The tangential values are not included in this submittal as they are much smaller in magnitude then the presented valves.



Axisymmetric Model of the Reactor Building and Soil Foundation

Draw. No. 960222.95 | Rev. | Figure 3A.5.1-1a

3A.5.2 BUILDING RESPONSES TO LOSS-OF-COOLANT ACCIDENT LOADS

The analysis of the containment under the action of long-term LOCA loads and the resultant responses are described in this section. The LOCA loads considered are those described in Section 3A.3.2.4, namely, chugging and condensation oscillation. However, the condensation oscillation load is not a governing load as compared to the chugging load, therefore it was not considered in the design assessment. The discussion in this section applies to the effects of chugging only. A complete definition of the chugging loads and the methodology of their application to the reactor building is contained in Reference 3A.3.2-15.

3A.5.2.1 Analytical Model

The mathematical model used for the analysis of the structure subjected to chugging loads includes the reactor building and the supporting soil. The model of the reactor building is shown in Figure 3A.5.2-1. The model of the supporting soil is the same as that shown in Figure 3A.5.1-1a in connection with the analysis for SRV discharge loads. The soil is represented by solid axisymmetric elements with asymmetric load capability. Spatial variation of soil shear modulus and unit weight and soil-structure interaction are accounted for.

As shown in Figure 3A.5.2-1, two types of finite elements have been used in modeling of the building, namely, axisymmetric conical shell elements and axisymmetric solid elements; both types of elements have asymmetric loading capability. The wetwell columns, stabilizer trusses, bellows, and shear lugs between the diaphragm floor and containment are modeled using springs.

In the suppression pool region, where the hydrodynamic loads are applied and where a more accurate representation is required, the node locations are closely spaced. The horizontal rings attached to the containment vessel are treated as discrete rings and the additional stiffness due to the vertical stiffeners is included with the vessel properties.

The RPV and internals are represented by axisymmetric shell elements. The dynamic coupling effect of the fluid in the RPV is accounted for by adding hydrodynamic masses to the nodal points of the mathematical model.

3A.5.2.2 Method of Analysis and Building Response

The structural response of the reactor building, when subjected to the chugging phenomenon, is determined from the application of the seven distributions of the chugging pressures on the wetwell pool boundary in the analytical model. These seven distributions of pool boundary pressures result from the seven design chugging sources developed for the multivent chugging definition described in Section 3A.3.2.4.2.1. Two loading cases are considered for each of the seven design sources, namely, nearly symmetric and asymmetric conditions. Detailed

analytic methods for determination of the structural response are given in Reference 3A.3.2-15.

As described in Reference 3A.3.2-15, the nearly symmetric and asymmetric results are comparable. For the purpose of this assessment, the nearly symmetric loading condition is used.

The response of the building is obtained in terms of acceleration response spectra. These were calculated for significant locations in the reactor building for the nearly symmetric and asymmetric loading conditions. The envelope spectrum curves were plotted with peaks spread by ±15% for damping values of 0.5%, 1.0%, 2.0%, and 4.0% of critical damping. Nearly symmetric response spectra plots for different locations in the building are illustrated in the figures listed below.

It should be noted that the design values presented in Figures 3A.5.2-2 through 3A.5.2-10 were increased by a factor of 1.10 to account for the differences in vent size (28 in. for Columbia Generating Station as compared to 24 in. for the 4T and 4TCO test) and an additional factor of 1.16 over the values presented in the chugging report.

3A.5.2.2.1 Reactor Building Response, Nearly Symmetric Loading - Acceleration Response Spectra

Location	Direction	Figure
Containment Vessel, el. 448 ft Mass No. 152	Radial	3A.5.2-2
Containment Vessel, el. 448 ft Mass No. 152	Vertical	3A.5.2-3
Containment Vessel, el. 459 ft Mass No. 123	Radial	3A.5.2-4
Containment Vessel, el. 459 ft Mass No. 123	Vertical	3A.5.2-5
RPV Support on Pedestal, el. 519 ft Mass No. 57	Radial	3A.5.2-6
RPV Support on Pedestal, el. 519 ft Mass No. 57	Vertical	3A.5.2-7

Containment Vessel, el. 583 ft Mass No. 12	Radial	3A.5.2-8
Building Wall, el. 521 ft Mass No. 55	Radial	3A.5.2-9
Building Wall, el. 521 ft Mass No. 55	Vertical	3A.5.2-10

Note that the tangential loads were also utilized in the assessment of CGS. The tangential values are not included in this submittal as they are much smaller in magnitude then the presented values.

Attachment 3A.B

THREE-DIMENSIONAL SOURCE FLOWS IN EXACT CONTAINMENT GEOMETRY

3A.B.1 INTRODUCTION AND SUMMARY

The method specified in Reference 3A.B-1 to calculate the flow field due to bubbles [caused by a loss-of-coolant accident (LOCA) or safety/relief valve (SRV) actuation] in a Mark II suppression pool is the method of images (MOI). The MOI is a potential flow technique and uses point source(s) or sink(s) to represent the bubble(s) and a number of image sources and/or sinks to simulate flow at the pool boundaries to satisfy the kinematical boundary conditions. The MOI has the following limitations with regard to its application to CGS:

a. The annular suppression pool geometry needs to be idealized into a rectangular pool using an "equivalent radius" concept,

b. The sloping suppression pool bottom needs to be idealized into a flat bottom, and

c. Computer flow field calculation costs are high.

In view of these limitations, the MOI is not used to calculate the potential flow field due to stationary source(s) in the CGS suppression pool. Instead, a numerical method is used to determine three-dimensional potential flows induced by source(s) in the exact CGS containment geometry.

The principle of the method can be stated rather simply. The three-dimensional potential flow induced by sources in any arbitrary suppression pool geometry is provided by solution of Laplace's equation for the velocity potential, ϕ. This is done by splitting the function ϕ into two components: ϕ_s which is due to all the sources and sinks (representing expanding or contracting bubbles) and is calculated analytically, and $\hat{\phi}$, which is a smooth function that is calculated numerically. The function $\hat{\phi}$ calculated from the original boundary conditions for ϕ with the boundary values of ϕ_s subracted off. Single ϕ_s analytically known elsewhere and $\hat{\phi}$ can be determined from its boundary conditions by iteration, the total value of the velocity potential, ϕ, can be easily determined.

This attachment describes the formulation of the problem, the method of solution, the numerical procedures utilized, and presents and analyzes some results. Section 3A.B.2 discusses the problem formulation and the solution method, Section 3A.B.3 discusses the numerical procedure, Section 3A.B.4 discusses how the flow field is calculated,

Section 3A.B.5 discusses the initial estimate to start the iteration solution for $\hat{\phi}$. Section 3A.B.6 discusses the convergence and accuracy of the method, and Section 3A.B.7 presents and analyzes some results for single and multiple bubble cases.

3A.B.2 PROBLEM FORMULATION AND SOLUTION METHOD

For a potential flow, the fluid velocity components may be expressed in terms of a velocity potential function, as shown below:

$$V_x = -\frac{\partial\phi}{\partial x}, \qquad V_y = -\frac{\partial\phi}{\partial y}, \qquad V_z = -\frac{\partial\phi}{\partial z}$$

The negative sign is purely a convention and means that the fluid flows in the direction of potential drop (Reference 3A.B-2).

The formulation of the boundary value problem for the potential is:

$\nabla^2\phi = 0$ everywhere in the fluid except appropriate delta functions at sources and sinks,

$\frac{\partial 0}{\partial n} = 0$ on the rigid boundaries (containment wall, basemat, and pedestal), and

$\phi = 0$ on the free surface (follows from pressure = 0).

The velocity potential, ϕ is split into two parts:

$$\phi = \phi_s + \hat{\phi}$$

The function ϕ_s represents contributions from the singularities (sources and/or sinks) and its analytical expression at any point within the fluid is given below:

$$\phi_S = \sum_{\nu} \phi_S$$

$$o_S = \frac{Q_\nu}{4\pi_{\rho\nu}}$$

where Q_ν represents the source strength of the νth source or sink and ρ_ν represents the distance from the νth source or sink to the point in question (Reference 3A.B-1).

The function $\hat{\phi}$ is a smooth function and satisfies the boundary value problem:

$$\nabla^2 \hat{\phi} = 0 \qquad \text{everywhere,}$$

$$\frac{\partial \phi}{\partial n} = -\frac{\partial \phi_s}{\partial n} \qquad \text{on the rigid boundaries and any planes of symmetry, and}$$

$$\hat{\phi} = -\phi_s \qquad \text{on the free surface.}$$

The boundary values for $\hat{\phi}$ were derived by subtracting the boundary values of ϕ_s from the original boundary value conditions far the total velocity potential, ϕ. Because $\hat{\phi}$ is extremely smooth, it can be accurately calculated by finite differences.

3A.B.3 NUMERICAL PROCEDURE

The geometry of the Columbia Generating Station suppression pool and the location of quenchers and downcomers are shown in *Figures 3A.2.1-1 through 3A.2.1-8*, inclusive. A cylindrical grid of points is overlayed such that all boundaries with $\partial\phi / \partial n = 0$ (i.e., walls and symmetry planes) are all centered between planes of grid points, while the water surface $z = z_{max}$ is a plane of grid points. Figures 3A.B-1 and 3A.B-2 show how the suppression pool geometry has been modeled for a single SRV actuation case and a LOCA bubble case to take advantage of symmetry.

The lengths between grid points in the r, ϕ, and z direction are denoted by Δr, $\Delta\phi$, and Δz and are indexed in the three directions by i, j, and k, respectively.

$$r_i = r_{min} = (i - 1.5)\,\Delta r$$

$$\phi_j = (j - 1.5)\,\Delta\phi$$

$$z_k = (z - 1.5)\,\Delta z$$

Any function, f, when regarded as a function defined on the grid points will be denoted as $f(r_i, \phi_j, z_k) = f_{i,j,k}$. The coordinates of the source(s) are indicated by the subscript s_v.

As mentioned before, the potential ϕ is split into $\phi_s + \hat{\phi}$. ϕ_s is the source potential (or the sum of each source's contribution in the multi-bubble cases):

$$\phi_s = \sum_v \frac{Q_v}{4\pi\rho_v}$$

where

$$\rho_v = ((r \cos \theta - r_v)^2 + (r \sin\theta)^2 = (z-z))\ 1/2$$

ϕ_s is evaluated exactly on all grid points. The numerical procedure of calculating ϕ is by the standard cover-relaxation method, which is summarized succinctly here. $\hat{\phi}$ satisfied the Laplace equation which, in cylindrical coordinates, is (Reference 3A.B-2):

$$\nabla^2 0 = \frac{1}{r}\frac{\partial}{\partial r}\frac{(r\partial 0)}{\partial r} + \frac{1}{r^2}\frac{\partial^2 0}{\partial \theta^2} + \frac{\partial^2 \hat{\phi}}{\partial z^2} = 0 \qquad (3A.B\text{-}1)$$

The finite difference approximation of equation 3A.B-1 is chosen as
(∇_f^2 = finite aprrox. of ∇^2 2):

$$\nabla_f^2 \hat{\phi} = \frac{(r_{i+1} + r_i)(\hat{\phi}_{i+1,j,k} - \hat{\phi}_{i,j,k}) - (r_i - r_{i-1})(\hat{\phi}_{i,j,k} - \hat{\phi}_{i-1,j,k})}{2r_i(\Delta r)^2} +$$
$$\frac{\hat{\phi}_{i,j+1,k} + \hat{\phi}_{i,j-1,k} - 2\hat{\phi}_{i,j,k}}{r_i^{\,2}(\Delta\theta)^2} + \frac{\hat{\phi}_{i,j,k+1} + \hat{\phi}_{i,j,k-1} - 2\hat{\phi}_{i,j,k}}{(\Delta z)^2} = 0 \qquad (3A.B\text{-}2)$$

where:

$$i=2,3\ldots i_{max}-1,\ j=2,3\ldots j_{max}-1,\ k=2,3\ldots k_{max}-1$$

Equation 3A.B-2 constitutes a linear system of (i_{max}-2) x (j_{max}-2) x (k_{max}-2) equations. The number of unknowns is larger, since they are unknowns at the boundaries, which are to be related by the boundary conditions.

As stated before, the boundary conditions for ϕ are:

$\hat{\phi} = -\phi_s$ on the water surface and $\dfrac{\partial\hat{\phi}}{\partial n} = -\dfrac{\partial\phi_s}{\partial n}$ on walls, bottom, and planes of symmetry.

For planes of symmetry a simplification from $\partial\hat{\phi} / \partial n = -\partial\phi_s / \partial n$ to $\partial\hat{\phi} / \partial n = 0$ occurs.

The finite difference equations have corresponding boundary conditions for flat bottom containments:

$$\hat{\phi}_{i,j,k_{max}} = -\phi_{s\,i,j,k_{max}} \qquad \text{(on the free surface),}$$

$$\hat{\phi}_{i,j,1} - \hat{\phi}_{i,j,2} = -\phi_{s\,i,j,1} + \phi_{s\,i,j,2} \qquad \text{(on basement floor),}$$

$$\hat{\phi}_{1,j,k} - \hat{\phi}_{2,j,k} = -\phi_{s\,1,j,k} + \phi_{s\,2,j,k} \qquad \text{(on the pedestal wall),}$$

$$\phi_{i_{max},j,k} - \phi_{i_{max}-1,j,k} = -\phi_{i_{max},j,k} + \phi_{i_{max}-1,j,k} \qquad \text{(on the containment wall),}$$

$$\hat{\phi}_{i,1,k} = \hat{\phi}_{i,2,k} \qquad \text{(on the surface } \theta = 0^{o}\text{), and}$$

$$\hat{\phi}_{i,j_{max},k} - \hat{\phi}_{i,j_{max}-1,k} = -\phi_{s\,i,j_{max},k} + \phi_{s\,i,j_{max}-1,k} \qquad \text{(on the surface } \theta = \theta_{max}\text{).}$$

These equations, coupled with equation 3A.B-2, now provide a complete system of i_{max}, k_{max} equations for the same number of unknowns.

There are many techniques available to solve the finite-difference equations generated from the Laplace equation. Some, using fast Fourier transform techniques or direct elimination techniques, are indeed very fast. The successive over-relaxation procedure (SOR) was selected because it is adequate in speed; in addition, it can handle general boundaries, whereas, use of other (possibly faster) methods require rectangular domains, periodic boundary conditions, or other restrictions.

Equation 3A.B-2 is solved for the center point in terms of the values at the neighboring points by iterating: (Superscript (n) indicates the n^{th} iteration, (n+1) the next iteration, etc.)

$$\hat{\phi}^{(n+1)}_{i,j,k} = \frac{\hat{\phi}_{i,j+1,k} + \hat{\phi}_{i,j-1,k}}{r_i^{\,2}(\Delta\theta)^2} + \frac{\overleftarrow{\phi}_{i,j,k+1} = \hat{\phi}_{i,j,k-1}}{(\Delta z)^2} + \frac{(r_{i+1} + r_i)}{2r_i(r)^2}\hat{\phi}_{i+1,j,k} +$$

$$\frac{(r_{i-1} + r_i)}{2r_i(\Delta\theta)^2}\hat{\phi}_{i-1,j,k} - \frac{2}{2r_i(\Delta\theta)^2} + \frac{2}{(\Delta z)^2} + \frac{2}{(\Delta r)^2}$$

or $\quad \hat{\phi}^{(n+1)} = V_o\hat{\phi}^{(n)}$

where:

V_o denotes an "averaging" operator.

Whenever a point is updated, the new value is to be used in calculating its neighboring points $\left(\hat{\phi}^{(n+1)} = V_1\hat{\phi}^{(n)} + V_2\hat{\phi}^{(n+1)}\right)$. This is straightforward iteration or relaxation. In SOR, which is much faster, the change is anticipated by using an acceleration parameter, (Reference 3A.B-3):

$$\hat{\phi}^{(n+1)} = \hat{\phi}^{(n)} + \omega\left(V_{0\,\phi^{(n)}-\phi^{(n)}}\right) \qquad (3A.B\text{-}3)$$

Numerical analysis theory shows that convergence occurs for $0 < \omega < 2$, but the optimal depends on the geometry. Various test runs for the single SRV bubble, and for the three bubble LOCA geometries were run with $\omega = 1.98$ chosen for rapid convergence (Section 3A.B.6 gives more details on convergence).

(Note: for $\omega = 1$, equation 3A.B-3 becomes $\hat{\phi}^{(n+1)} = V_0\hat{\phi}^{(n)}$ which is the straightforward iteration case).

3A.B.4 CALCULATION OF FLOW FIELD

3A.B.4.1 Steady State Flow Field Calculation

The calculation of the velocity due to a potential function is performed by taking the negative gradient of the total potential, ϕ. As discussed earlier, the total potential is determined by summing ϕ_s, the potential due to the singularities plus $\hat{\phi}$, which is a smooth function. To ensure accuracy near singularities the velocity field is also calculated as the sum of two components:

$$V = -\nabla\phi = -\nabla\left(\phi_s + \hat{\phi}\right) = V_s + \hat{V}$$

The velocity field due to the smooth function $\hat{\phi}$ is determined numerically at points one-half a grid distance away from the velocity potential grid system as shown in Figure 3A.B-3a. Essentially this scheme averages values of $\hat{\phi}$ in the two neighboring grid planes that are normal to the direction of the desired velocity component. Once these two averages are established, the velocity component is determined by subtracting the two values and dividing by the grid width (Δr, $r\Delta\theta$, Δz) in the appropriate direction (a representative example is given in Figure 3A.B-3). The r, θ, and z velocity components due to $\hat{\phi}$ are shown below:

$$V_{r\,i,j,k} = \left(\hat{\phi}_{i,j,k} + \hat{\phi}_{i,j+1,k} + \hat{\phi}_{i,j,k+1} + \hat{\phi}_{i,j+1,k+1}\right) - \left(\hat{\phi}_{i+1,j,k} + \hat{\phi}_{i+1,j+1,k} + \hat{\phi}_{i+1,j,k+1} + \hat{\phi}_{i+1,j+1,k+1}\right) \;/\; (4 \cdot \Delta r)$$

$$V_{0\ i,j,k} = \left(\hat{\phi}_{i,j,k} + \hat{\phi}_{i+1,j,k} + \hat{\phi}_{i,j,k+1} + \hat{\phi}_{i+1,j,k+1}\right) - \left(\hat{\phi}_{i,j+1,k} + \hat{\phi}_{i+1,j+1,k} + \hat{\phi}_{i,j+1,k+1} + \hat{\phi}_{i+1,j+1,k+1}\right) / (4 \cdot r \cdot \Delta 0)$$

$$V_{z\ i,j,k} = \left(\hat{\phi}_{i,j,k} + \hat{\phi}_{i+1,j,k} + \hat{\phi}_{i,j+1,k} + \hat{\phi}_{i+1,j+1,k}\right) - \left(\hat{\phi}_{i,j,k+1} + \hat{\phi}_{i+1,j,k+1} + \hat{\phi}_{i,j+1,k+1} + \hat{\phi}_{i+1,j+1,k+1}\right) / (4 \cdot \Delta z)$$

The velocity field due to a source is determined analytically at the same grid points used for the velocity field due to the smooth function. The magnitude of the total velocity at a point due to a source is defined as:

$$\left|V_s\right| = \frac{Q}{4\pi\rho^2}$$

The x, y, z velocity components are defined by:

$$Vx_s = \left|V_s\right| \cdot \frac{x - x_s}{\rho}$$

$$Vy_s = \left|V_s\right| \cdot \frac{y - y_s}{\rho}$$

$$Vz_s = \left|V_s\right| \cdot \frac{z - z_s}{\rho}$$

and the cylindrical components are defined by:

$$Vr_s = Vx_s \cos\theta + Vy_s \sin\theta$$
$$V\theta_s = Vy_s \cos\theta - Vs_s \sin\theta$$
$$Vz_s = Vz_s$$

For multiple sources, the velocity component are determined by summing the contribution of each of the sources. The total velocity magnitude is determined by the square root of the sum of the squared of the three components.

The usual way of calculating the velocity field is by differentiating the total potential function ϕ by the use of finite differences in exactly the same manner as was describe for the smooth function's velocity calculation. The defect here is that the resultant velocity field becomes inaccurate as one approaches the source(s). This is due to the singularity in ϕ and the resultant inability of a finite difference scheme to approximate a derivative. Also, finite differences taken across the source(s) will be totally inaccurate and meaningless. The above method of calculating the velocity field avoids this problem because (1) the velocities due to

ϕ_s are an exact solution and, (2) $\hat{\phi}$ is a smooth function which allows for a good approximation of a derivative by finite differences.

3A.B.4.2 Transient Flow Field Calculation

The flow field due to a time varying source can be calculated by assuming that the source strength varies with time. Since the velocity field is now a function of time and space, an acceleration field can also be calculated:

$$V(r, \theta, z, t) = -Q(t)\nabla\phi(r, \theta, z)$$
$$V(r, \theta, z, t) = -Q(t)\nabla\phi(r, \theta, z)$$

where:

$Q(t)$ is the equivalent time varying point source strength representing the hydrodynamic source.

$\dot{Q}(t)$ is the time rate of change of $Q(t)$.

$\nabla\phi(4, \theta, z)$ = gradient of the velocity potential at a given point due to a normalized point source strength.

$V(r, \theta, z, t)$ = total velocity at a given point.

$\dot{V}(r, \theta, z, t)$ = total acceleration at a given point.

3A.B.5 INITIAL ESTIMATE FOR ITERATION

Since $\partial\hat{\phi}/\partial n$ connects two grid planes which make up a rigid boundary, it is a "soft" boundary condition and "clamps" down the solution less than does a $\hat{\phi}$ boundary condition. With $\hat{\phi}$ known at only one out-of-six possible boundaries, it is necessary to start the iterations with as good an estimate as possible for the initial function $\hat{\phi}^{(0)}$. For single source cases, the initial function $\hat{\phi}^{(0)}$ is easy to construct: simply take $\hat{\phi}$ - ϕ_s on the surface, and $\hat{\phi} = 0$ on the basemat, and interpolate linearly in z along each vertical line. Subsequent convergence for $\hat{\phi}$ is not too far from zero on all the boundaries.

For the multi-bubble LOCA cases, the situation is quite different. Above the sources the flow is essentially a vertical slug with uniform velocity up to the surface, while below the sources the flow is very small. Thus ϕ rises from zero on the surface to some large value near the

bubbles. From this knowledge of the phenomenon the following construction for the initial function was developed:

(1) Calculate the average surface velocity:

$$V_{surf} = \Sigma Q/\text{area of surface}$$

(2) The total ϕ behaves like:

$\phi \cong V_{surf}(z_{surf} - z)$	far above the sources
$\phi \cong \phi$ sources	near the source
$\phi \cong V_{surf}(z_{surf} - z_s)$	far below the source

(3) From (2) above use:

$\hat{\phi} = V_{surf}(z_{surf} - z) - \phi_s$	above the source
$\hat{\phi} = V_{surf}(z_{surf} - z_s) - \phi_{s'}$,	$\lvert z - z_s \rvert < 12''$
$\hat{\phi} = V_{surf}(z_{surf} - z_s)$,	$\lvert z - z_s \rvert < -12''$

The prescription is designed to obtain the correct $\hat{\phi}$ away from the sources quickly, and not to introduce any singular parts into $\hat{\phi}$ near the sources. Naturally, such estimates are arbitrary, but absolute accuracy of the initial estimates are not required anyway.

3A.B.6 CONVERGENCE AND ACCURACY

This section is concerned with the accuracy of the finite difference solution. A good numerical solution is one that accurately approximates the exact solution. The approach of the numerical solution to the exact solution as the grid is refined is called convergence of the numerical solution. If an iterative scheme is used, as in our case, convergence of the iterations is defined by the difference between any two iterations approaching zero as the number of iterations is increased.

For this scheme, theory assures us that both kinds of convergence take place (Reference 3A.B-3) provided $0 < \omega < 2$ (see Section 3A.B.3) and provided small enough grids and large enough number of iterations are used. The real questions, of course, are whether small enough grids and enough iterations have been used to guarantee an accurate enough approximate solution.

To check convergence of the numerical solution LOCA bubble charging, SRV, etc. were recalculated with several different grid sizes until the change in the solution was insignificant. Figure 3A.B-4 shows the results of several different grids. Figure 3A.B-1 shows the geometry for a single SRV case.

Convergence of iterations is easier to check by printing the total percent change between two successive iterations, and ascertaining that this change is not more than 0.001% (for single bubble, 300 iterations, for LOCA, 2000 iterations). Note that this suggests accuracy, but does not guarantee it, for even such small changes can be accumulated over a large number of steps and can cause divergence of the solution.

(For example: $\overset{M}{\Sigma} \frac{1}{n}$ diverges as M approaches infinity but the percent change between successive terms n=1 approaches zero.)

There are additional checks made, namely, overall conservation. To do this, the flow $\iint (\partial\phi/\partial n)ds$ over all the surface areas due to the singular part of the potential, ϕ_s and the smooth part of the potential, $\hat{\phi}$ are calculated. On the solid walls, they are equal and opposite and add up to zero (this is exactly so on vertical walls and on the flat bottom, but only closely so on the slant bottom). On the free surface, the sum of the two should equal exactly the total of all the sources. For the number of iterations performed the solutions are always within 0.1%.

In addition, the integral $\iiint \overline{V}^2\phi dV$ is calculated. For the smooth part, this should equal zero, while for the singular part, this should equal the total flow of N sources, $(N \cdot 4\pi)$. The first is accurate always to within 0.1 in.^{3}/sec, the latter agrees with the sum of the free surface flows to within 0.1%. This however, is not coincidental, since the finite difference approximation of the Laplacian operator has been chosen to be "conservative", i.e., the Gauss theorem holds for the difference approximation.

Finally, the integral, $\iiint_3 \left|\overline{V}^2\phi\right| dV$, is also calculated and vanished to within 0.1 in.^{3}/sec for the smooth part. This same integral is calculated for the singular part and should also equal the total source flows which in fact it does not. This is unimportant, since the singular part is never calculated by finite difference except at the boundary, and the finite difference Laplacian is not expected to be accurate for the singular functions.

3A.B.7 RESULTS

In this section, some computed results for two representative cases are presented and analyzed: (a) slanted bottom pool, LOCA bubble charging event and, (b) flat bottom pool, LOCA bubble charging event.

In all the velocity results presented, the source strengths have been normalized to:

$$Q/4\pi = 10{,}000 \text{ in.}^3/\text{sec}$$

i.e., the flow velocity is 100 in./sec at a radius of 10 in. from the source center, or the total flow is $4\pi \times 10^4$ in.^{3}/sec. However, to keep the potentials in a more manageable range, the printed values are 1/10 of the potentials corresponding to the above normalized source strength (i.e., the reader should multiply all printed potentials by 10 to get the correct value).

3A.B.7.1 Loss-of-Coolant Accident Bubble Charging

Loss-of-coolant accident bubble charging calculations were made under the assumption that all bubbles are of the same strength ($Q/4\pi = 10{,}000$). Then by symmetry, it suffices to perform the calculation for a wedge of angle $2\pi/34 = 10.6^\circ$ radius (since there are three rows of 34 downcomers). By further symmetry, it suffices to consider just a half-wedge, or 5.29°, since the plane exactly halfway between two downcomer planes is a symmetry plane. Figure 3A.B-2 gives the geometry of the LOCA bubble charging calculations. The flat bottom calculations used 16 points in r, 26 points in z, and 10 points in θ; corresponding to grid sizes of $\Delta r = 23.75$ in., $\Delta\theta = 0.01155$ rad, and $\Delta z = 15.26$ in. Figure 3A.B-5 shows the potential distribution at a plane near the sources and Figure 3A.B-6 shows the same at the plane midway between two source planes. Figures 3A.B-7 and 3A.B-8 show the velocity in the planes $\theta = 0^\circ$ and $\theta = 5.29^\circ$, respectively.

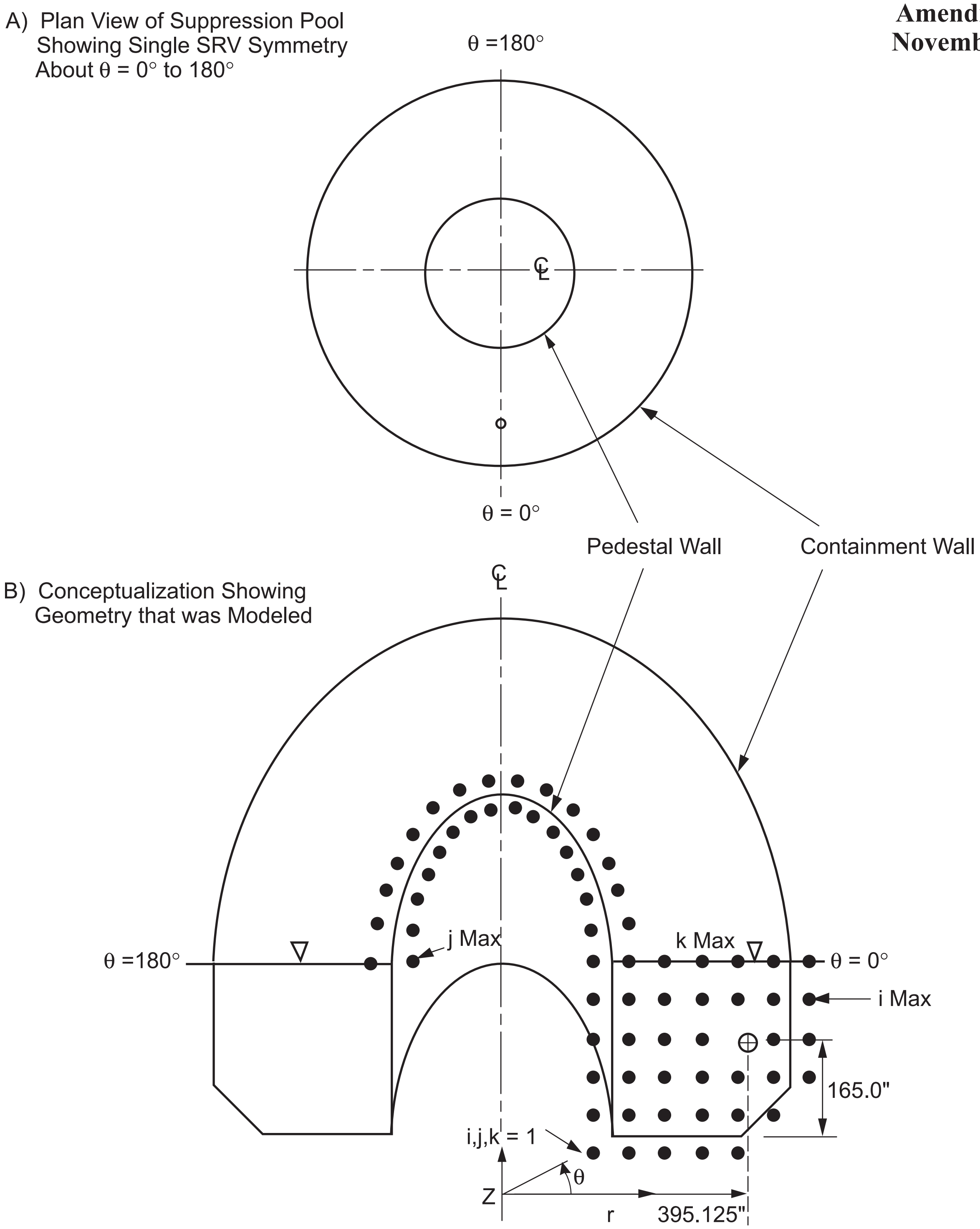
There are some physically interesting features revealed in these calculations. (a) In contrast to one or a few single bubbles, when LOCA bubbles grow in phase, flow below the source is essentially negligible. (b) There is very little variation in the flow pattern in the azimuthal (θ) direction, except of course, at the exact elevation of the sources. (c) Although the sources have the same strength, the innermost source has a stronger influence on the flow than does the outermost one, simply because of the increased density of the bubbles as one approaches the pedestal. (d) From calculations for three-dimensional LOCA flow for the flat bottom case (Figure 3A.B-9) the smallness of the velocities (or essentially constant ϕ values) below the source levels, indicates that the effect of the slant bottom is negligible.

3A.B.8 REFERENCES

3A.B-1 Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and SRV Ramshead Air Discharges, General Electric Company, NEDE-21471 (Proprietary), September 1977.

3A.B-2 Theoretical Hydrodynamics, by L. M. Milne-Thompson, the Macmillan Company, N.Y., 1950.

3A.B-3 Analysis of Numerical Methods, by E. Isaacson and H. B. Keller, John Wiley & Sons, Inc., N.Y., 1966.



Notes:

1) o Indicates Downcomer Location

2) ⊕ Indicates Source/Bubble Location

3) ℄ Indicates Suppression Pool Centerline

4) ● Indicates Velocity Potential Grid Points

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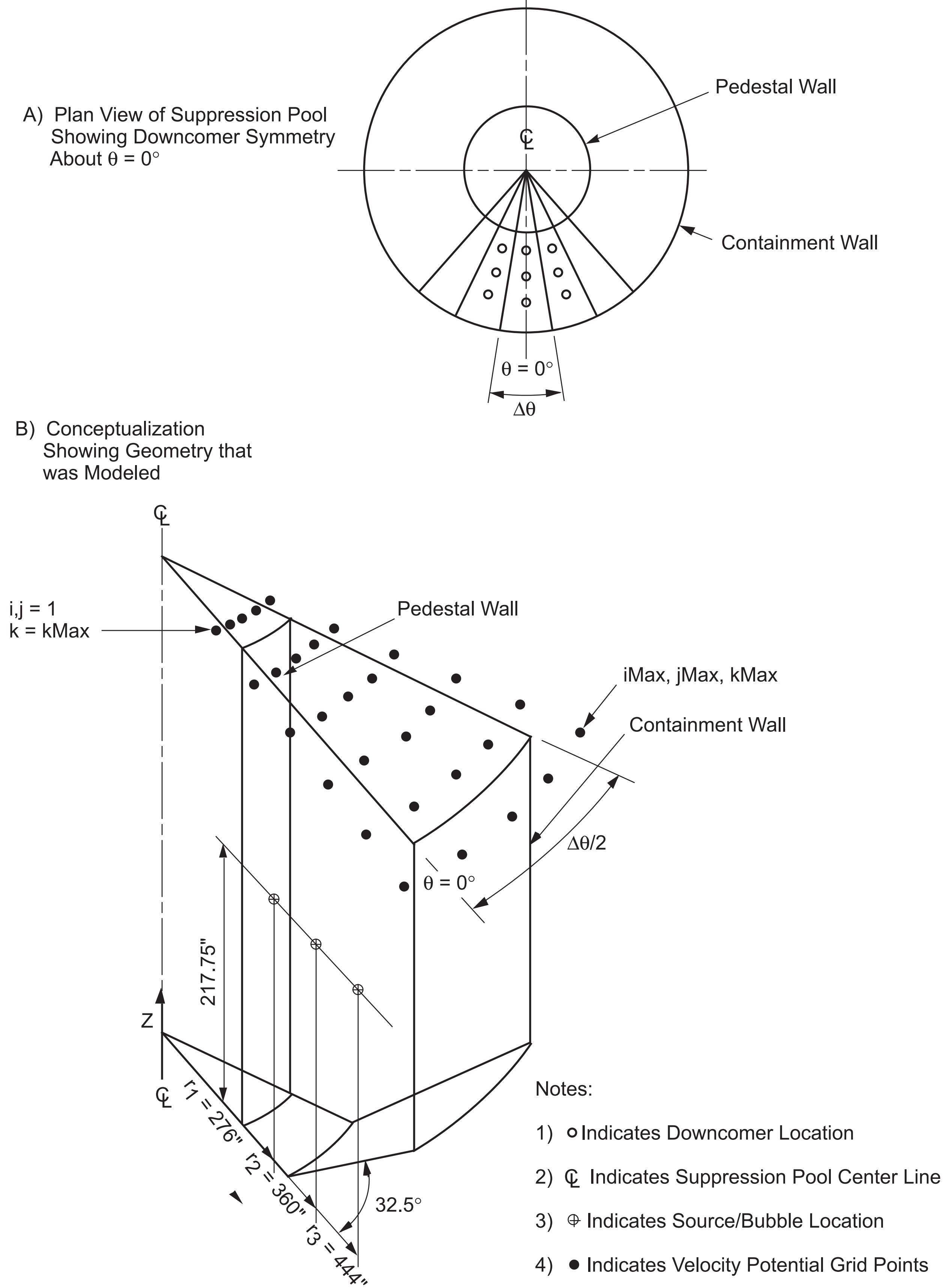
Geometry for Single SRV Case

Draw. No. 900547.82	Rev.	Figure 3A.B-1



Notes:

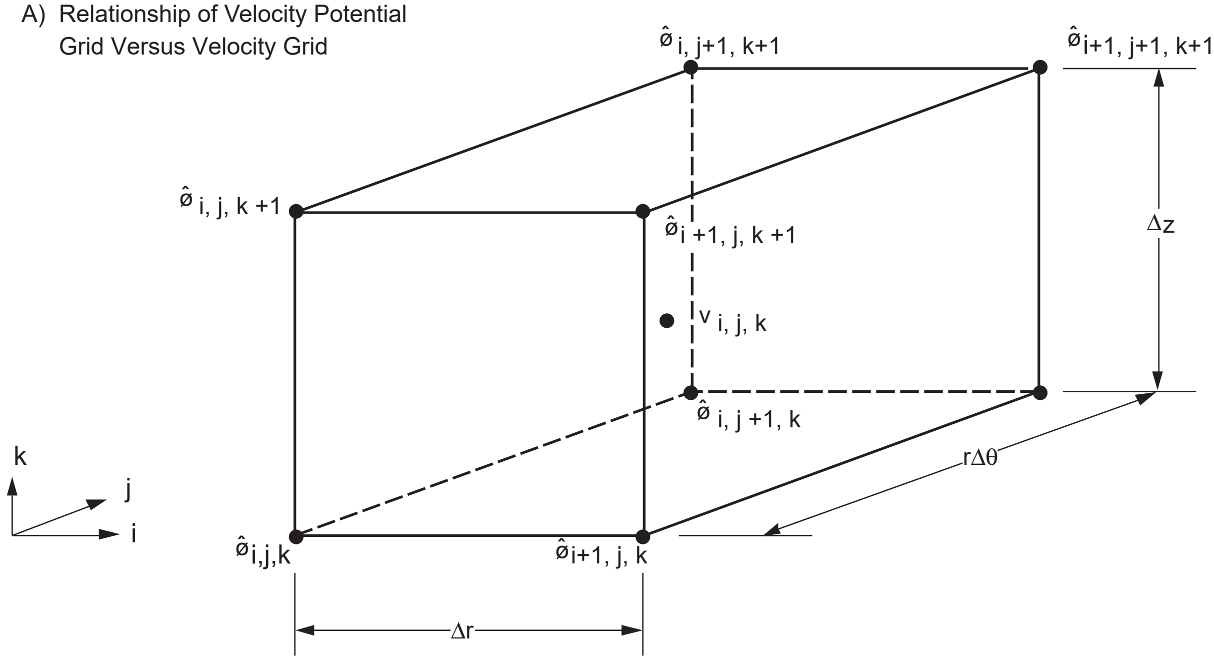
1) ○ Indicates Downcomer Location

2) ℄ Indicates Suppression Pool Center Line

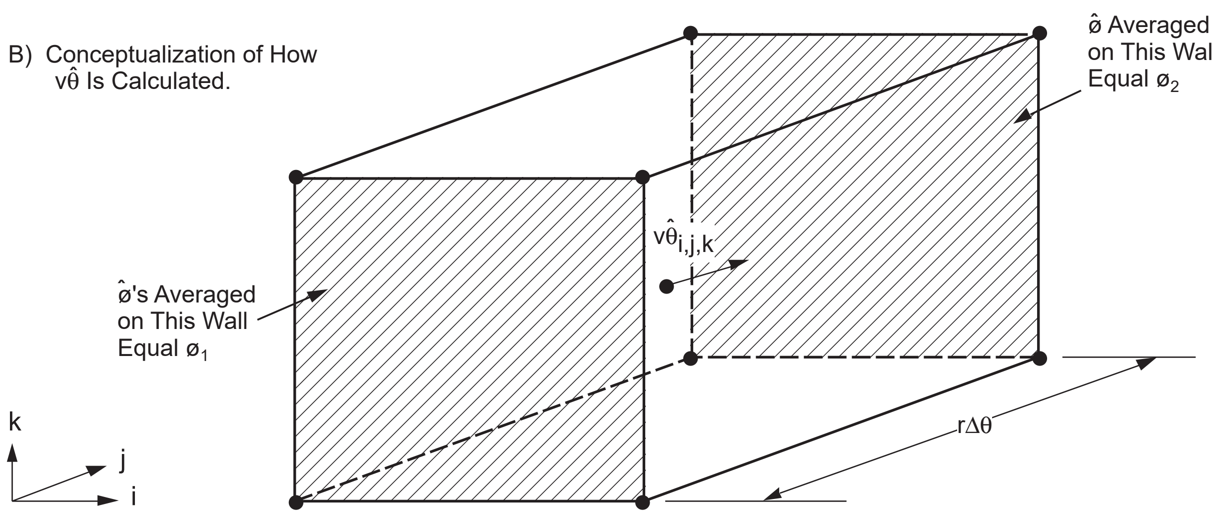
3) ⊕ Indicates Source/Bubble Location

4) ● Indicates Velocity Potential Grid Points

A) Relationship of Velocity Potential Grid Versus Velocity Grid



B) Conceptualization of How $v\hat{\theta}$ Is Calculated.



$$\phi_1 = (\hat{\phi}_{i, j, k} + \hat{\phi}_{i+1, j, k} + \hat{\phi}_{i+1, j, k+1} + \hat{\phi}_{i, j, k+1})/4$$

$$\phi_2 = (\hat{\phi}_{i, j+1, k} + \hat{\phi}_{i, j+1, k+1} + \hat{\phi}_{i+1, j+1, k+1} + \hat{\phi}_{i+1, j+1, k})/4$$

$$v\hat{\theta} = \frac{\hat{\phi}_1 - \hat{\phi}_2}{r\Delta\theta}$$

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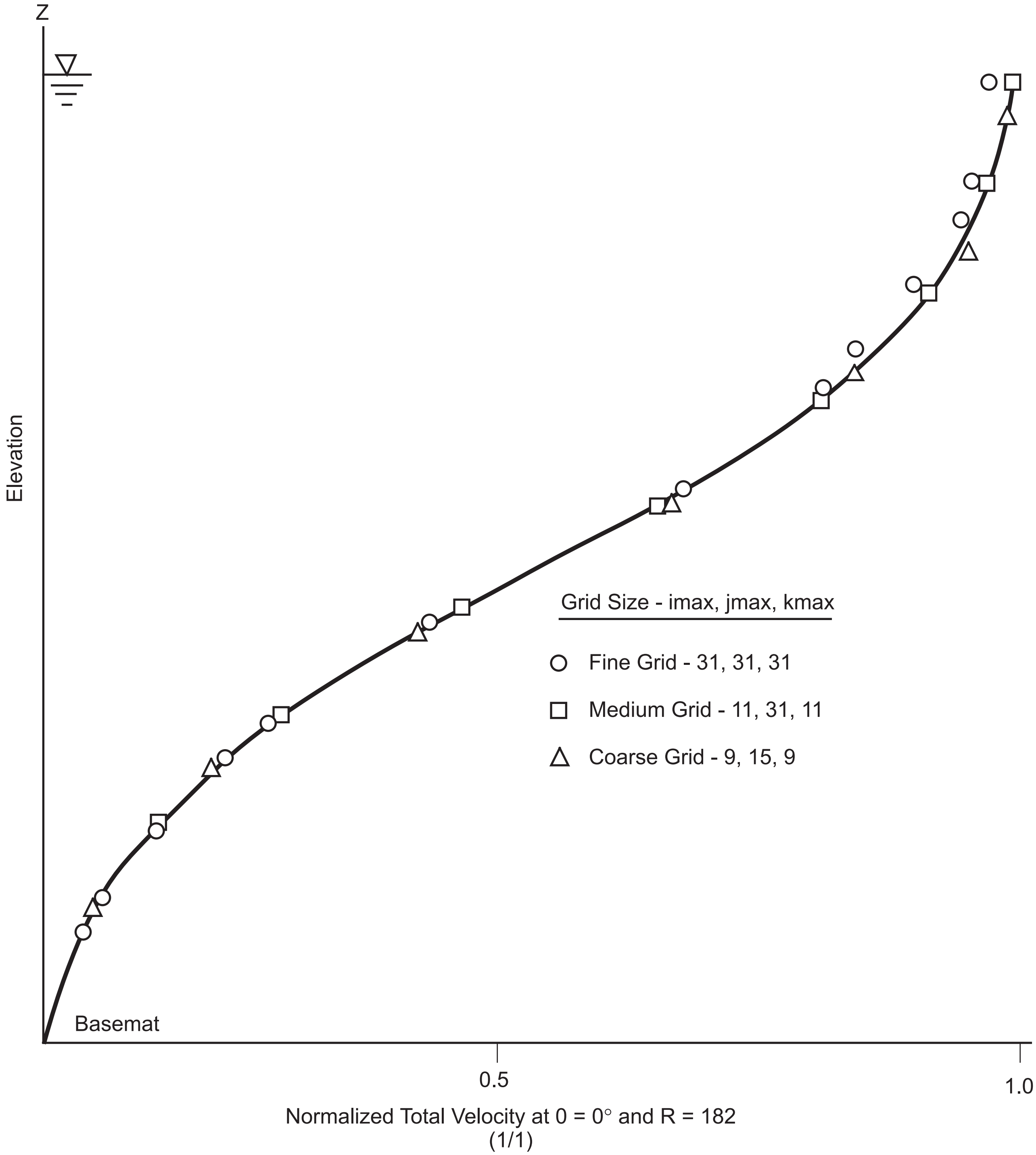
Velocity Grid

Draw. No. 900547.83	Rev.	Figure 3A.B-3



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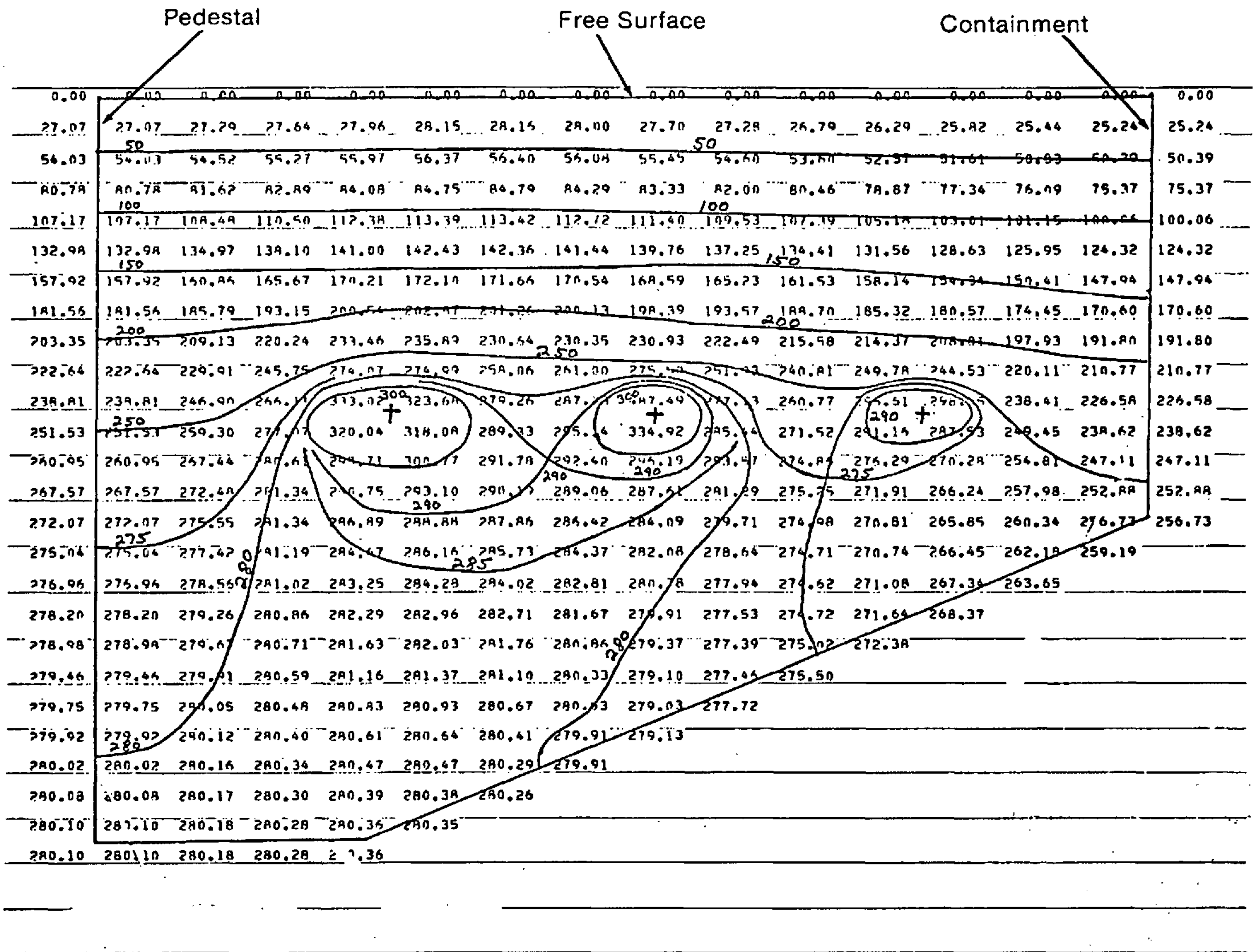
Grid Convergence Test

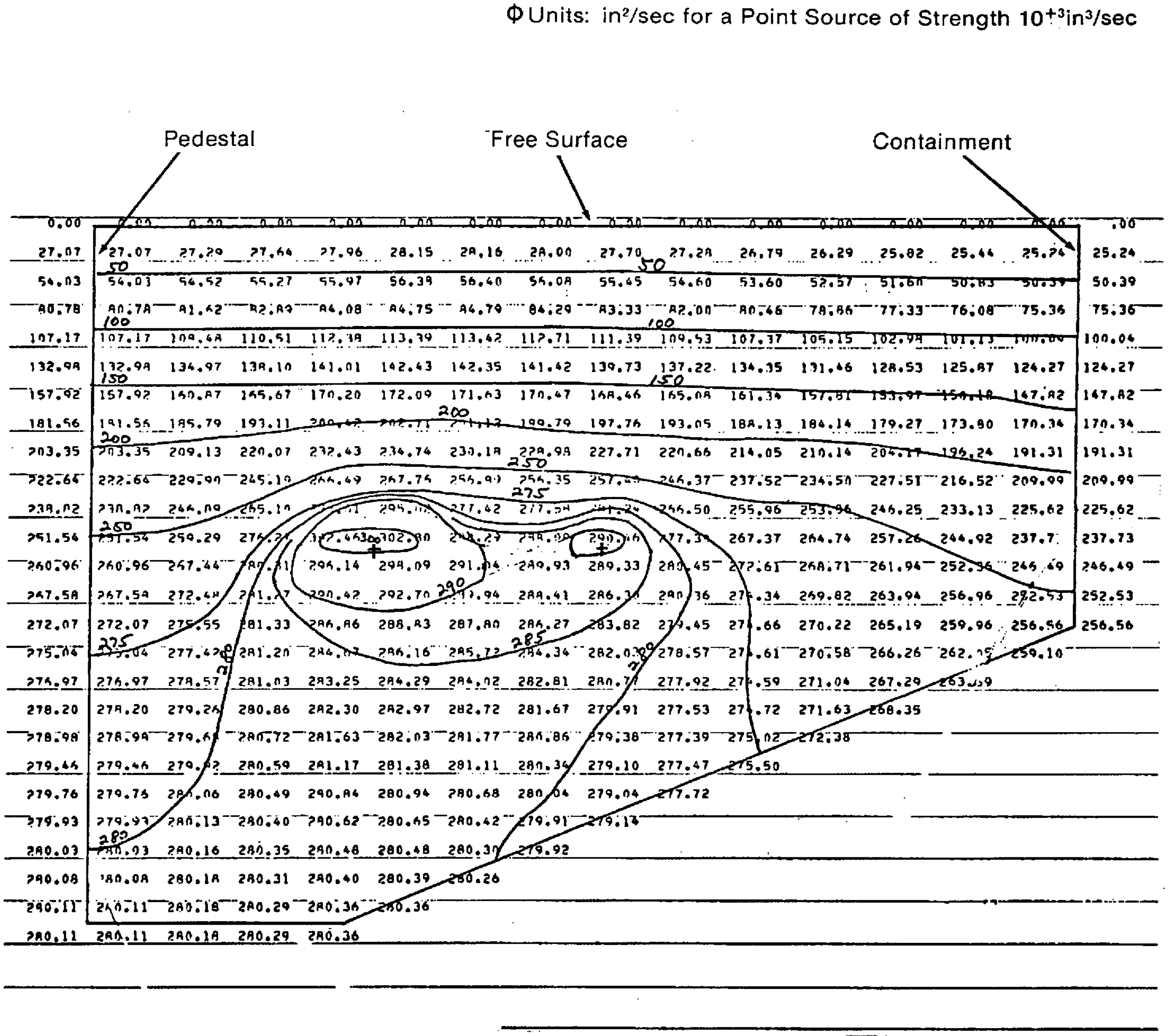
Draw. No. 900547.80 | Rev. | Figure 3A.B-4

ΦUnits: in²/sec for a Point Source of Strength 10^{+3} in³/sec



Notes:
1. + indicates radial and vertical locations of the sources.
2. See Figure 3A.B-2 for geometry.

Velocity Potential, Φ, in Plane 0.33° Away from Sources for LOCA Bubble Slant Bottom Case

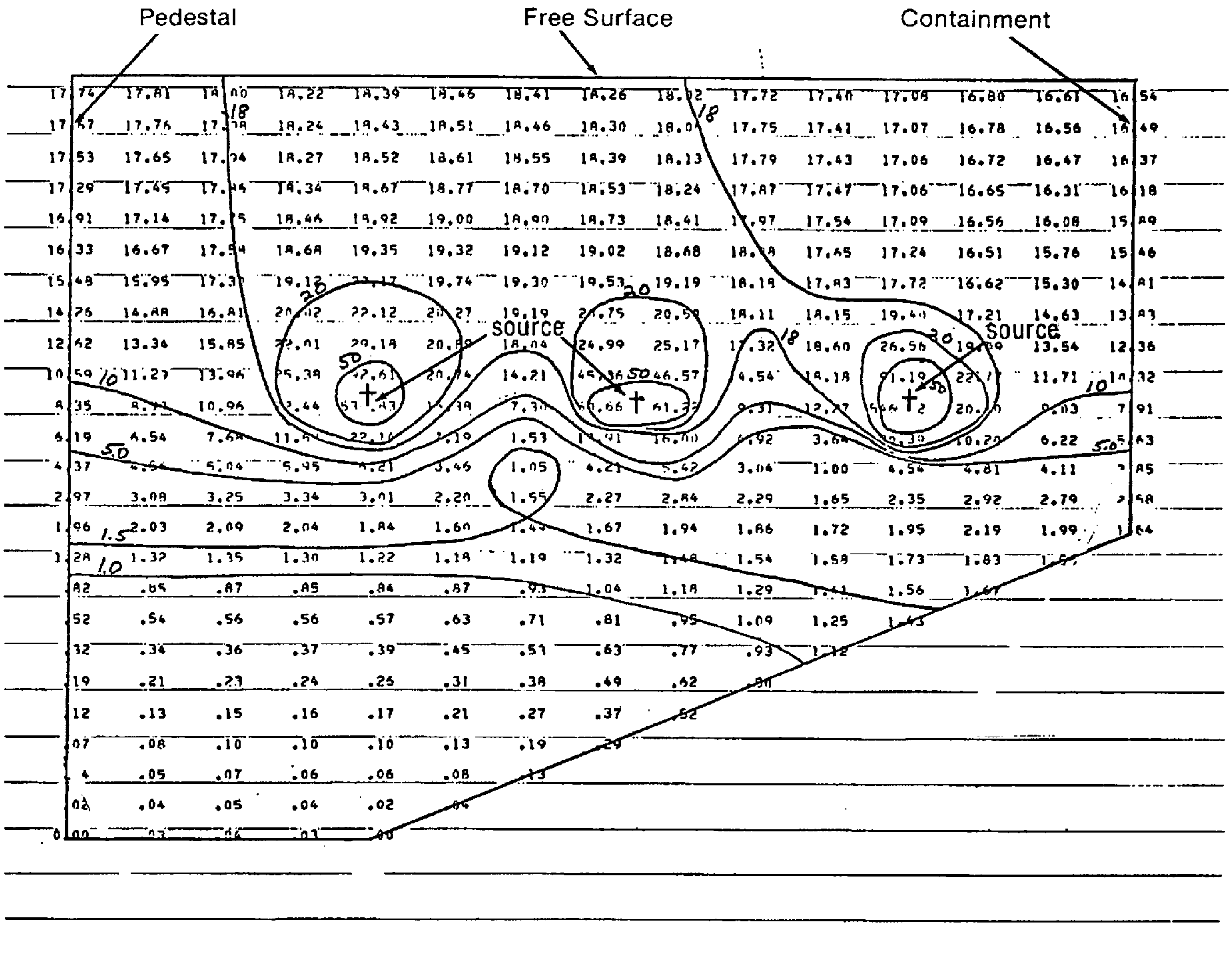


Notes:
1. + indicates radial and vertical locations of the sources.
2. See Figure 3A.B-2 for geometry.

Velocity Potential, Φ, in Plane 4.96° Away from Sources for LOCA Bubble Slant Bottom Case

Draw. No. 020361.24 | Rev. | Figure 3A.B-6

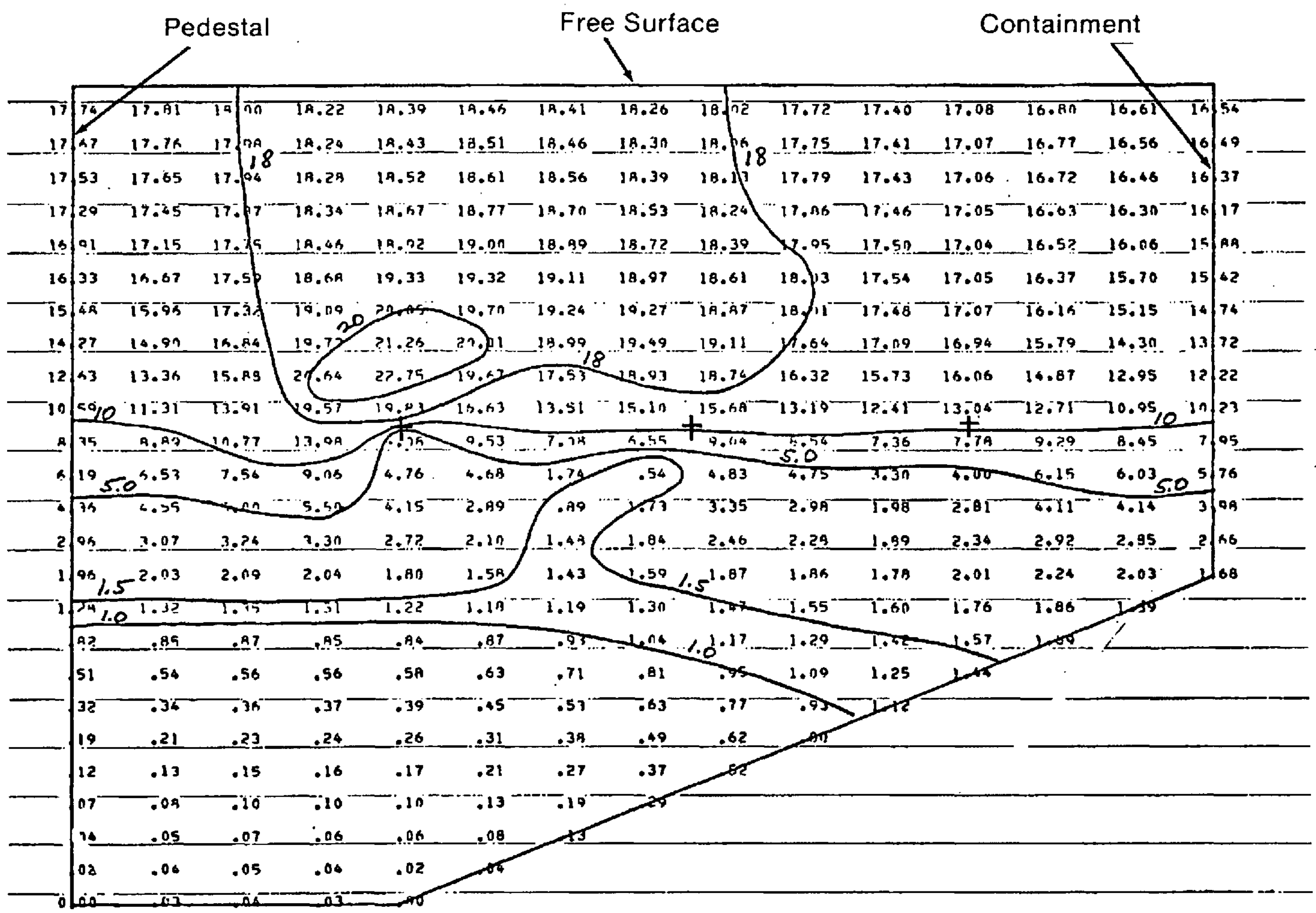
$\nabla\Phi$ Units: in/sec for a Point Source of Strength 10^{+4}in^3/sec



Note:
1. Figure 3A.B-2 shows geometry.

Total Velocity Gradient, $\nabla\Phi$, in Source Plane for LOCA Bubble Slant Bottom Case

$\nabla\Phi$ Units: in/sec for a Point Source of Strength $10^{+4} in^3/sec$

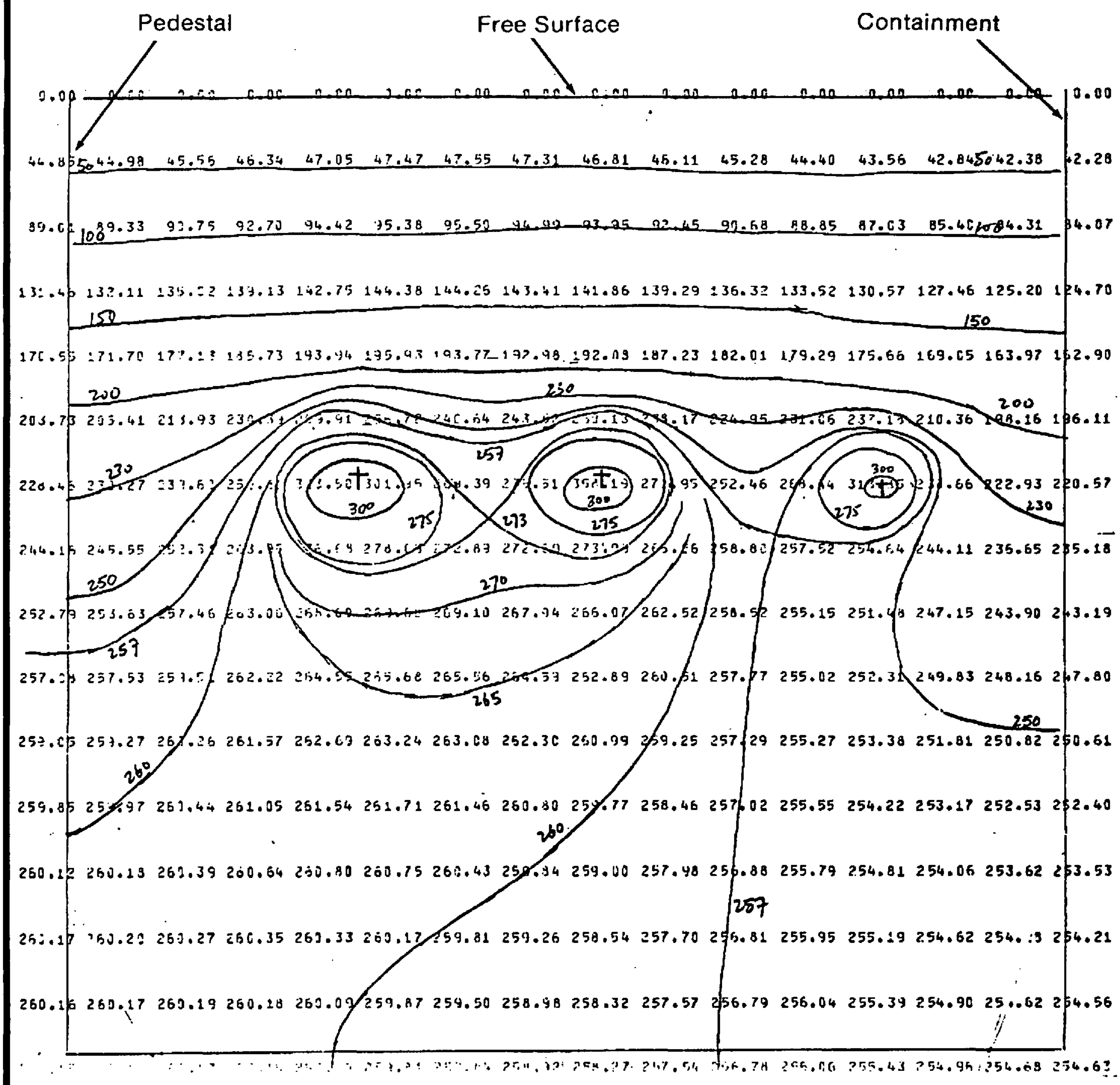


Note:
1. † indicates radial and vertical locations of the sources.

Total Velocity Gradient, $\nabla\Phi$, in Source Plane 5.29° Away from Sources for LOCA Bubble Slant Bottom Case

Draw. No. 020361.26 | Rev. | Figure 3A.B-8

Φ Units: in²/sec of a Point Source of Strength 10^{+3}in³/sec



Notes:
1. + indicates radial and vertical locations of the sources.
2. Figure 3A.B-2 shows approximate geometry.

Columbia Generating Station Final Safety Analysis Report	Velocity Potential, Φ, in Plane 0.33° Away from Sources for LOCA Bubble Flat Bottom Case		
	Draw. No. 020361.27	Rev.	Figure 3A.B-9

Attachment 3A.C

CONCEPT OF DRAG FORCES DUE TO HYDRODYNAMIC FLOW FIELDS

3A.C.1 CONCEPT

The concept of drag forces is described in Reference 3A.C-1 as a means to estimate loads on submerged structures due to flow fields created in a Mark II containment suppression pool by the hydrodynamic events described in Section 3A.3. Loads resulting from the actual distorted flow around a structure may be estimated by postulating an equivalent locally uniform flow field due to the safe forcing function in the pool without any structures. This uniform flow is characterized by the velocity and acceleration fields present at the geometric center of the structure or structural segment. The loads on submerged structures are characterized by drag forces due to locally uniform velocity and acceleration fields. The velocity field causes a standard drag force and a lift force, and the acceleration field causes an acceleration drag force. The total load on the structure or structural component is obtained by the vectorial summation of these forces.

Information essential for calculating the drag loads is identified below.

3A.C.2 FORMULAS FOR DRAG LOADS

In the following three sections, formulas are presented to calculate velocity drag load, acceleration drag load, and lift load. The methodology described below is in general agreement with Reference 3A.3.2-4.

Long structures are divided into segments for more precise evaluation. This is done to account for the variations of the velocity and acceleration along the structure.

3A.C.2.1 <u>Velocity Drag Load</u>

The velocity drag load is calculated using the following formula:

$$P_s = \frac{1}{2}\rho\, C_D\, V_{max}^2$$

where:

P_s = velocity drag pressure amplitude (psi). This pressure acts in the flow direction.

ρ = mass density of water (lb sec^2/in.^{4}).

C_D = standard drag coefficient. Numerical values for C_D are given in the applicable sections of Section 3A.3.

V_{max} = maximum velocity in the direction of flow (in./sec).

3A.C.2.2 Acceleration Drag Load

The acceleration drag load is calculated using the following formula:

$$P_A = \rho \, C_M \frac{\pi}{4} D \, V_{max}^2$$

where:

A = acceleration drag pressure amplitude (psi). This pressure acts in the flow direction.

C_M = acceleration drag coefficient. Numerical values for C_M are given in the applicable sections of Section 3A.3.

D = diameter of cylindrical structure (in.). If the structure is not cylindrical, D is the diameter of a cylinder circumscribing the structure.

V_{max} = maximum acceleration in the direction of flow (in./sec^2).

3A.C.2.3 Lift Load

The lift load is calculated using the following formula:

$$P_L = \frac{1}{2} \rho \, C_L \, V_{max}{}^2$$

where:

P_L = lift pressure amplitude (psi). This pressure is normal to the flow direction.

C_L = lift coefficient. Numerical values for C_L are given in the applicable sections of Section 3A.3.

3A.C.3 REFERENCES

3A.C-1 "Analytical Model for Estimating Drag Forces on Rigid Submerged Structures caused by LOCA and Safety/Relief Valve Ramshead Air Discharges," General Electric Company, NEDO-21471, September 1977.

Attachment 3A.D

CALCULATION MODELS FOR SHORT-TERM LOSS-OF-COOLANT ACCIDENT PHENOMENA

3A.D.1 INTRODUCTION

This attachment provides additional information concerning the numerical techniques used to model short term hydrodynamic phenomena. The vent clearing, pool swell, and loss-of-coolant accident (LOCA) bubble numerical models are discussed in Sections 3A.D.2, 3A.D.3, and 3A.D.4 respectively. In each section, the model assumptions, equations, numerical techniques, and verification are either discussed in detail or referenced to the appropriate General Electric document.

3A.D.2 DOWNCOMER VENT CLEARING MODEL

The pool swell analytical model (PSAM), (Reference 3A.D-1), models the pool swell event subsequent to downcomer vent water clearance. Initial conditions required to start the PSAM include the time of vent clearing and the pool surface displacement, velocity, acceleration, and wetwell pressure at the time of vent clearing. In order to provide a time history of the suppression pool surface during the downcomer vent clearing process and a conservative input to the PSAM, the computer code VENT was developed. VENT is a subroutine for the PSAM computer code SWELL (see Section 3A.D.3). By continuity the downcomer vent exit water velocity and acceleration time histories are also calculated. These transients are used as conservative input to the LOCA water jet code (see Section 3A.3.2.3.1.1).

Section 3A.D.2.1 discusses the downcomer vent clearing model development and Section 3A.D.2.2 discusses the experimental verification and the conservatism of the model.

3A.D.2.1 Model Development

Assumptions used in developing the vent clearing model are as follows:

a. The frictional losses of the pool system are conservatively neglected,

b. The wetwell free air volume is isentropically compressed by the upward moving water slug,

c. Heat losses are neglected,

d. The air velocity within the downcomers is small, therefore the air pressure in the vents is conservatively assumed to equal the current drywell pressure,

e. Downcomer vent losses are conservatively neglected, and

f. Viscous effects are neglected.

Figure 3A.D-1 shows a schematic of the vent clearing model. The mathematical derivation of this model is similar to the model in Reference 3A.D-2 with the exception that the vent clearing model described here couples the equation of motion for the vent system with the equation of motion for the pool system.

3A.D.2.1.1 Drywell Pressure

The time varying drywell pressure, P_D, is the driving function for the vent clearing analysis. P_D is not calculated by VENT but is input as data.

3A.D.2.1.2 Water in the Downcomer Vents

The mass of water within the downcomer vents, m', that is being accelerated downward by the increasing drywell pressure transient is given by:

$$m' = \rho_w (H_o - h) A_V \qquad \text{(3A.D-1)}$$

where

ρ_w = the density of water

A_V = the total downcomer vent exit area

H_o = the initial submergence of the downcomers

h = the displacement of the internal downcomer water surface.

3A.D.2.1.3 Water Slug in the Suppression Pool

The mass of the water slug in the suppression pool, m, which is being accelerated upward by the increasing drywell pressure is given by:

$$m = \rho_w (H_o + z) A_p \qquad \text{(3A.D-2)}$$

where

A_p = the net suppression pool water surface area

z = the displacement of the pool surface.

3A.D.2.1.4 Suppression Chamber Air Space

From assumption 2, the transient pressure in the suppression chamber air space, P_s, is calculated from:

$$P_s = P_{so}(V_{so}/V_s)^k$$

where

V_s = V_{so} - $A_v h$

V_{so} = initial wetwell free air space volume

P_{so} = initial wetwell air pressure

k = specific heat ratio of air.

Combining and solving yields:

$$P_s = P_{so}\left(V_{so}/\left(V_{so} - A_v h\right)\right)^k$$

3A.D.2.1.5 Fluid Dynamics

Refer to Figure 3A.D-1. From Newton's second law, MA = F, the equation of motion for the water inside of the vents is:

$$m'\frac{d^2h}{dt^2} = (P_D - P_\infty)A_v = \rho_w A_v (H_o - h)g$$

where

P_∞ = the pressure at the downcomer vent exit.

Combining with equation 3A.C-1 and solving for $\frac{P_D - P_\infty}{\rho_w}$ yields:

$$\frac{P_D - P_\infty}{\rho_w} = (H_o - h)\frac{d^2n}{dt^2} - (H_o - h)g \quad \text{(3A.D-3)}$$

Also from Newton's second law, the equation of motion for the water outside of the vents is:

$$m\frac{d^2z}{dt^2} = (P_\infty - P_s)A_p - \rho_w A_p(H_o + z)g$$

Combining with equation 3A.D-2 and solving for $\frac{P_\infty - P_s}{\rho_w}$ yields:

$$\frac{P_\infty - P_s}{\rho_w} = (H_o + z)\frac{d^2z}{dt^2} + (H_o + z)g \quad \text{(3A.D-4)}$$

Substituting h A_v/A_p for z (see Figure 3A.D-1) in equation 3A.D-4 and then summing equations 3A.D-3 and 3A.D-4 and solving for d^2h/dt^2 yields:

$$\frac{d^2h}{dt^2} = \frac{\frac{P_D - P_s}{\rho_w} - g\,h\left(1 + A_v/A_p\right)}{H_o\left(1 + A_v/A_p\right) - h\left(1 - \left(A_v/A_p\right)^2\right)} \quad \text{(3A.D-5)}$$

Integration of d^2h/dt^2 yields the downcomer vent water velocity, dh/dt, and displacement, h, transients:

$$\frac{dh}{d5} = \int_0^t \frac{d^2h}{dt^2}\,dt \quad \text{(3A.D-6)}$$

$$h = \int_0^t \frac{dh}{dt}\,dt \quad \text{(3A.D-7)}$$

where

t = time after LOCA initiation.

The pool surface acceleration, velocity, and displacement transients are related by continuity to the vent water transients by a factor of A_v/A_p (see Figure 3A.D-1). It is important to note that at $h = H_o$, equation 3A.D-5 is the same as the equation of motion used in the PSAM (Reference 3A.D-7). This implies that equation 3A.D-5 plus the PSAM will provide a continuous and consistent time history of the suppression pool surface displacement during a LOCA.

3A.D.2.1.6 Numerical Integration

Sections 3A.D.2.1.1 and 3A.D.2.1.4 show that the drywell pressure, P_D, and the wetwell air space pressure, P_S are functions of time and vent water displacement, respectively. From equation 3A.D-5 this shows that the downcomer vent water acceleration, d^2h/dt^2, is a function of time and vent water displacement only. This means that equation 3A.D-5 is a second order differential equation of the functional form: $d^2h/dt^2 = f(t,h)$, where $h = h(t)$ by equation 3A.D-7. This allows numerical integration of equation 3A.D-5 using a fourth order Runge-Kutta technique given by equation 25.5.22 of Reference 3A.D-3. Integration of d^2h/dt^2 gives dh/dt and h (equations 3A.D-6 and 3A.D-7, respectively).

3A.D.2.1.7 Termination of Vent Clearing Analysis

Termination of the vent clearing analysis occurs when $H_o - h \leq O$. This is the moment that the vent clearing is completed; or t_o, vent clearance time.

3A.D.2.1.8 Input Data and Results

Input to the VENT subroutine requires data on the following plant characteristics: net pool area (A_p), total downcomer vent exit area (A_v), initial submergence of the downcomers (H_o), initial wetwell pressure (P_{so}), initial wetwell free air volume (V_{so}), and the drywell pressure transient (P_D). Table 3A.3.2-3 shows the CGS input data for the VENT computer code.

The vent exit water velocity and acceleration calculated by VENT are increased by 10% as indicated by the NRC in Reference 3A.3.2-1. The velocity and acceleration time histories (including the 10% increase) are shown in Figures 3A.3.2-2 and 3A.3.2-3, respectively.

3A.D.2.2 Experimental Verification

VENT has been verified against downcomer vent water displacement data from the 4T Test Series 5101, runs 21, 22, 24, and 37 (Reference 3A.D-4). These runs were chosen because they ran the full range of Mark II submergences and drywell pressurization rates. In each run, three conductivity probes were used to determine the displacement of the downcomer vent air/water interface. These probes, shown in Figure 3-3 of Reference 3A.D-5, sense the difference in conductivity between air and water. For each test, the three probes were located

in each downcomer at 0.5 ft, 6.5 ft, and 9.5 ft above the downcomer vent exits (see Figure 3-2 of Reference 3A.D-5).

Tables 3A.D-1 and 3A.D-2 show the input data to VENT for the verification runs. Table 3A.D-3 shows the measured and calculated 4T Test downcomer vent probe water clearing times. Figure 3A.D-2 summarizes the data in Table 3A.D-3 and shows that the comparison is excellent.

3A.D.3 POOL SWELL ANALYTICAL MODEL

In order to conservatively calculate the pool swell transient, the computer code SWELL was developed after the model discussed in References 3A.D-1, 3A.D-6, and 3A.D-7. The equations used in the SWELL computer code are documented in those references. The PSAM is schematically shown in Figure 3A.D-3 and its verification against empirical data and its conservatism is discussed in References 3A.D-1, 3A.D-6, 3A.D-7, and 3A.D-8. Input to SWELL requires data on the following plant characteristics: net pool area, total downcomer vent exit area, initial submergence of the downcomers, initial drywell air pressure, initial wetwell air pressure, initial drywell air temperature, initial wetwell free air volume, initial drywell humidity, downcomers loss coefficient, time of vent clearing, vent clearing velocity, and drywell air pressure transient.

Figures 3A.D-4 through 3A.D-9 are plots of pool surface velocity and pool surface elevation obtained with SWELL for the three benchmark plants presented in Reference 3A.D-7. (Note: for these verification runs, the vent clearing subroutine described in Section 3A.D.2 is not used since the vent clearing velocity and time are given in Reference 3A.D-7.) These plots are provided for comparison with the data included in Reference 3A.D-7 as benchmark problems for the SWELL code and show good agreement.

3A.D.4 LOSS-OF-COOLANT ACCIDENT BUBBLE CHARGING MODEL

The one-dimensional PSAM (see Section 3A.D.3) describes the bulk flow process in the suppression pool during a postulated pool swell event. This model assumes a flow field in the vertical direction only. The assumption of predominately vertical flow has been verified by small scale multivent pool swell tests. However, observation of these tests have shown that prior to LOCA bubble coalescence and the forming of an air blanket under the pool water slug, a significant three-dimensional flow field is developed. In response to these observations, new analytical techniques were developed in order to model the LOCA bubble charging event. The purpose of the LOCA bubble charging model, therefore, is to describe the three-dimensional flow fields during the early portion of the pool swell phenomenon.

In order to calculate the transient flow fields in the CGS suppression pool during the LOCA bubble charging portion of a postulated pool swell event, the computer code SOURCE was developed (see Attachment 3A.B). The application of the SOURCE code to the LOCA bubble

charging phenomenon is schematically shown in Figure 3A.B-2. This method uses point sources with the appropriate source strength to represent the LOCA bubble charging event in the exact CGS suppression pool geometry. In using this method it is assumed that all vents uniformly charge air into spherical bubbles with centers one downcomer radius below the vent exits. To calculate the transient LOCA bubble charging flow field, the rate of bubble growth is determined by continuity from the pool surface rise obtained from the PSAM.
A comparison of the similarity between this method and the method discussed in Reference 3A.D-9 is provided in Table 3A.D-4.

3A.D.4.1 Potential Flow Field

The three-dimensional potential flow field calculation method that is the basis for the SOURCE computer code is described in Attachment 3A.B. Also described in Attachment 3A.B are the numerical techniques, the flow field calculation procedure, and the conservation, convergence, and accuracy checks of the method. Results for the CGS LOCA bubble charging case is discussed in Sections 3A.B.7 and 3A.3.2.

Although both methods use the same assumptions of potential flow, point sources to represent bubbles, and uniformly charging spherical bubbles, the SOURCE code is used instead of the method of images (MOI) (Reference 3A.D-9) to determine finite pool effects. This is because the SOURCE code models the exact CGS suppression pool geometry, whereas the MOI has to idealize the pool's annular boundaries and sloping floor characteristics.

3A.D.4.2 Source Strength Calculation

The rate of air charging (and, therefore, the rate of bubble radius growth) is determined by continuity from the pool surface rise obtained from the PSAM. In using this method, it is assumed that all vents uniformly charge air into spherical bubbles with centers 1 ft 0 in. below the vent exits. The PSAM method of calculating a transient source strength for use in determining the flow field during LOCA bubble charging is used for the CGS instead of the method presented in Reference 3A.D-9. Source strengths calculated using both methods are presented in the following sections where it is shown that the PSAM method is preferable to the method of Reference 3A.D-9 in that it is more conservative and has experimental verification. General Electric has developed a method to calculate the equivalent bubble charging velocity and acceleration source strengths for a point source in a finite pool. It is described in Reference 3A.D-9. Air bubbles at the downcomer vents during the LOCA bubble charging process are assumed to be spherical. The bubble radius growth time history is obtained by assuming the bubble dynamics are represented by the Rayleigh equation coupled with a mass and energy balance for the bubble. Because the Rayleigh equation models the bubble dynamics in an infinite pool, a factor "K" must be solved for each bubble to correct for finite pool boundary effects. This factor is then multiplied by the Rayleigh bubble velocity and acceleration source strengths to solve for the finite pool velocity and acceleration source strengths at each bubble as shown in Figure 3A.D-10. Table 3A.D-5 provides some CGS

LOCA bubble charging pool surface velocities and accelerations obtained using the source strength method or Reference 3A.D-9 along with the SOURCE computer code. Extensive small scale multivent pool swell tests have shown that the pool surface remains relatively flat during the LOCA bubble charging process. For CGS, the pool surface transient is calculated by the PSAM (see Section 3A.D.3). It is evident in References 3A.D-1 and 3A.D-7 that the PSAM estimates of the pool surface transient during the early portion of pool swell (which is LOCA bubble charging) consistently bounds all experimental data. The small scale multivent pool swell tests also indicate bubble sphericity during the early portion of the transient.

With these observations it is possible to obtain the bubble velocity and acceleration source strengths from the PSAM calculated pool surface transient by continuity as shown in Figure 3A.D-11. Table 3A.D-5 provides some LOCA bubble charging pool surface velocities and accelerations obtained for CGS using the Reference 3A.D-9 method and the PSAM methods for comparison. It is seen that while pool surface accelerations are similar, the pool surface velocities from the Reference 3A.D-9 method are less than 50% of the PSAM values. Therefore, for CGS the PSAM method is conservatively accepted for LOCA bubble source strength definitions.

Figure 3A.3.2-7 shows the velocity, Q(t) and acceleration, $\dot{Q}(t)$, source strengths which are used in CGS load calculations.

3A.D.5 REFERENCES

3A.D-1 Ernst, R. J., Ward, M. G., Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon, General Electric Company, NEDE-21544-P (Proprietary), December 1976.

3A.D-2 The General Electric Pressure Suppression Containment Analytical Model, General Electric Company, NEDM-10320, April 1971.

3A.D-3 Handbook of Mathematical Functions with Formulas, Graphs, and Mathematical Tables, National Bureau of Standards - Applied Mathematics Series .55, May 1968.

3A.D-4 Phases I, II, and III of the Temporary Tall Tank Test (4T) Program, an Applications Memorandum, Preliminary Draft, General Electric Company, December 1976.

3A.D-5 Mark II - Pressure Suppression Test Program, General Electric Company, NEDE-13442-P-01, May 1976.

3A.D-6 Response to NRC Question 020.68, Appendix A, WNP-2, DAR, Revision 2, August 1979.

3A.D-7 Mark II Containment Dynamic Forcing Functions Information Report (DFFR), General Electric Company, NEDE-21061, Revision 3, June 1978.

3A.D-8 Comparison of the 1/13 Scale Mark II Containment Multivent Pool Swell Data with Analytical Methods, General Electric Company, NEDO-21667, August 1977.

3A.D-9 Moody, F. J., Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety Relief Valve Ramshead Air Discharges, General Electric Company, NEDE-21471 (Proprietary), September 1977.

Table 3A.D-1

Vent Clearing Analytical Model Input Data

Parameter	21	22	24	37
Net pool area[a], ft^2	35.17	35.17	35.17	35.17
Downcomer vent flow area, ft^2	2.0211	2.0211	2.0211	2.0211
Downcomer submergency, ft	13.15	9.0	13.5	11.0
Integration time step, sec	0.0001	0.0001	0.0001	0.0001
Initial wetwell free air volume, ft^3	950	1108	950	1038
Drywell pressure time history, psia	See Table 3A.D-2			

[a] Excludes area of downcomer.

Table 3A.D-2

Measured 4T Test Series 5101 Drywell Dome
Pressure Time History

Run 21		Run 22		Run 24		Run 37	
Time (sec)	Pressure (psia)	Time (sec)	Pressure (psia)	Time (sec)	Pressure (psia)	Time (sec)	Pressure (psia)
0.	14.63	0.	14.57	0.	14.55	0.	14.54
0.037	14.65	0.037	14.80	0.040	14.53	0.041	14.52
0.055	15.12	0.058	15.30	0.058	14.66	0.060	14.60
0.076	15.61	0.079	15.87	0.076	15.35	0.078	14.93
0.097	15.73	0.097	16.16	0.097	16.12	0.096	16.04
0.115	15.91	0.115	16.42	0.113	16.52	0.114	17.28
0.134	16.18	0.133	16.70	0.133	17.11	0.133	17.91
0.152	16.42	0.152	17.01	0.154	17.80	0.153	18.46
0.170	16.54	0.170	17.47	0.173	18.41	0.174	19.35
0.189	16.80	0.191	17.98	0.191	19.04	0.193	19.94
0.210	17.27	0.212	18.47	0.210	19.54	0.211	20.44
0.230	17.80	0.230	10.89	0.228	20.03	0.230	20.97
0.249	13.10	0.249	19.42	0.246	20.52	0.248	21.54
0.267	18.41	0.267	19.79	0.267	21.07	0.266	22.21
0.285	13.79	0.285	20.33	0.288	21.67	0.287	23.00
0.303	19.08	0.304	20.80	0.306	22.22	0.308	23.77
0.322	19.32	0.324	21.29	0.324	22.73	0.326	24.40
0.343	19.70	0.345	21.87	0.343	23.19	0.344	24.99
0.363	20.03	0.364	22.36	0.361	23.64	0.363	25.54
0.382	20.37	0.382	22.95	0.380	24.11	0.381	26.16
0.400	20.64	0.400	23.29	0.400	24.68	0.400	26.77
0.419	20.94	0.419	23.72	0.421	25.12	0.420	27.40
0.437	21.27	0.437	24.23	0.440	25.61	0.441	27.91
0.455	21.69	0.458	24.75	0.458	26.10	0.460	28.40
0.476	22.06	0.479	25.34	0.476	26.68	0.478	29.00
0.497	22.46	0.497	25.85	0.494	27.21	0.496	29.63
0.515	22.81	0.515	26.30	0.513	27.78	0.514	30.24
0.534	23.15	0.533	26.78	0.533	28.33	0.533	30.81
0.552	23.46	0.552	27.23	0.554	28.87	0.553	31.40
0.570	23.80	0.570	27.76	0.573	29.32	0.574	31.95
0.539	24.13	0.591	28.14	0.591	29.79	0.593	32.43
0.610	24.51	0.612	28.61	0.610	30.21	0.611	32.90
0.630	24.88	0.630	28.99	0.628	30.70	0.630	33.37
0.649	25.26	0.649	29.40	0.646	31.10	0.668	33.79
0.667	25.59	0.667	29.81	0.667	31.63	0.666	34.22
0.685	25.91	0.685	30.21	0.688	32.14	0.687	34.80
0.703	26.22	0.704	30.68	0.706	32.57	0.708	35.35
0.722	26.58	0.724	31.16	0.724	33.01	0.726	35.78

Table 3A.D-2

Measured 4T Test Series 5101 Drywell Dome Pressure
Time History (Continued)

Run 21		Run 22		Run 24		Run 37	
Time (sec)	Pressure (psia)	Time (sec)	Pressure (psia)	Time (sec)	Pressure (psia)	Time (sec)	Pressure (psia)
0.743	27.01	0.745	31.61	0.743	33.44	0.744	36.23
0.763	27.33	0.764	31.98	0.761	33.90	0.763	36.71
0.782	27.61	0.782	32.32	0.780	34.37	0.781	37.08
0.800	27.86	0.800	32.58	0.800	34.78	0.800	37.46
0.819	29.18	0.819	32.87	0.821	35.20	0.820	37.79
0.837	28.39	0.837	33.11	0.840	35.49	0.841	38.05
0.855	28.81	0.858	33.31	0.858	35.81	0.860	38.25
0.876	29.12	0.878	33.52	0.876	36.13	0.878	38.35
0.897	29.48	0.897	33.68	0.894	36.52	0.896	38.52
0.915	29.78	0.915	33.80	0.913	36.84	0.914	38.68
0.934	30.07	0.933	33.88	0.933	37.17	0.933	38.78
0.952	30.39	0.952	33.96	0.954	37.47	0.953	33.84
0.970	30.66	0.970	34.36	0.973	37.74	0.974	38.88
0.989	30.92	0.991	34.10	0.991	37.96	0.993	38.98
1.010	31.29	1.102	34.15	1.010	38.18	1.011	39.00
1.030	31.57	1.030	34.21	1.028	38.31	1.030	39.04
1.049	31.87	1.049	34.19	1.046	38.41	1.048	39.06
1.067	32.08	1.067	34.17	1.067	38.57	1.066	39.08
1.085	32.26	1.085	34.17	1.088	38.59	1.087	39.11
1.103	32.54	1.104	34.18	1.106	38.61	1.108	39.10
1.122	32.67	1.124	34.08	1.124	38.65	1.126	39.08
1.142	32.85	1.145	34.38	1.143	38.65	1.144	39.08
1.163	33.07	1.169	34.32	1.161	38.63	1.163	39.06
1.182	33.19	1.182	33.90	1.180	38.61	1.181	39.06
1.200	33.33	1.200	33.88	1.200	38.61	1.200	39.06
1.219	33.38	1.218	33.80	1.221	38.55	1.220	39.02
1.237	33.44	1.237	33.76	1.240	38.49	1.241	38.94
1.255	33.50	1.258	33.64	1.258	38.43	1.260	38.92
1.276	33.60	1.278	33.54	1.276	38.35	1.278	38.90
1.297	33.60	1.297	33.50	1.294	38.26	1.296	38.88
1.315	33.66	1.315	33.44	1.313	38.20	1.314	38.84
1.333	33.62	1.333	33.35	1.333	38.08	1.353	38.89
1.352	33.56	1.352	33.29	1.354	37.98	1.353	38.89
1.370	33.46	1.370	33.19	1.373	37.92	1.374	38.82
1.389	33.42	1.391	33.13	1.391	37.78	1.393	38.82

Table 3A.D-3

Comparison of Measured and Calculated 4T Test
Probe Water Clearing Times

Probe Elevation Above Downcomer Exit (ft)	Probe Water Clearing Time (sec)			
	21[a]	22[a]	24[a]	37[a]
9.5	0.667 (0.687)	Initially dry	0.573 (0.582)	0.381 (0.388)
6.5	0.800 (0.811)	0.458 (0.467)	0.688 (0.688)	0.533 (0.533)
0.5	0.952 (0.961)	0.667 (0.654)	0.800 (0.813)	0.687 (0.670)

[a] 4T run number.

Note: Unbracketed numbers are measured data. Bracketed Numbers are calculated data.

Table 3A.D-4

Comparison Between Source Method
and the Reference 3A.D-9 Method for Calculation
of the Loss-of-Coolant Accident Bubble Charging Event

Item	Source	Reference 3A.D-9	Comments
1. Uses potential flow assumption	Yes	Yes	Same
2. Uses point source to represent charging LOCA bubbles	Yes	Yes	Same
3. All vents assumed to charge uniformly into spherical bubbles one radius below downcomers	Yes	Yes	Same
4. Finite pool effects	Uses numerical scheme discussion in Attachment 3A.B of this Report	Uses MOI	-
5. Models CGS annular suppression pool geometry	Yes	No	For MOI, the CGS annular pool geometry must be idealized into a rectangular pool using an "equivalent radius" concept.
6. Models CGS sloping pool bottom	Yes	No	For MOI, the CGS sloping pool bottom needs to be idealized into a flat pool bottom.
7. Transient source strength	Determined by continuity from pool surface rise obtained from the PSAM	Determined from Rayleigh bubble dynamics equation in an infinite pool and a finite pool correction factor "K"	Source method results in the same conservatism as PSAM flow field calculations. Source method yields higher velocities and accelerations than the method discussed in Reference 3A.D-9.
8. Experimental verification of source strength	Yes	No	Source method determined directly from PSAM. PSAM has extensive experimental verification as to its conservatism.

Table 3A.D-5

Comparison of Results of Source and Method of Reference 3A.D-9 Bubble Charging Source Strength Methods

Time After Vent Clearing (sec)	GE Method[a]		PSAM Method[b]	
	V(ft/sec)	$\dot{V}$ (ft/sec^2)	V(ft/sec)	$\dot{V}$ (ft/sec^2)
0.	1.215	83.136	5.308	80.977
0.066[c]	4.872	81.767	10.613	79.772
0.24[d]	-	-	22.074	53.983

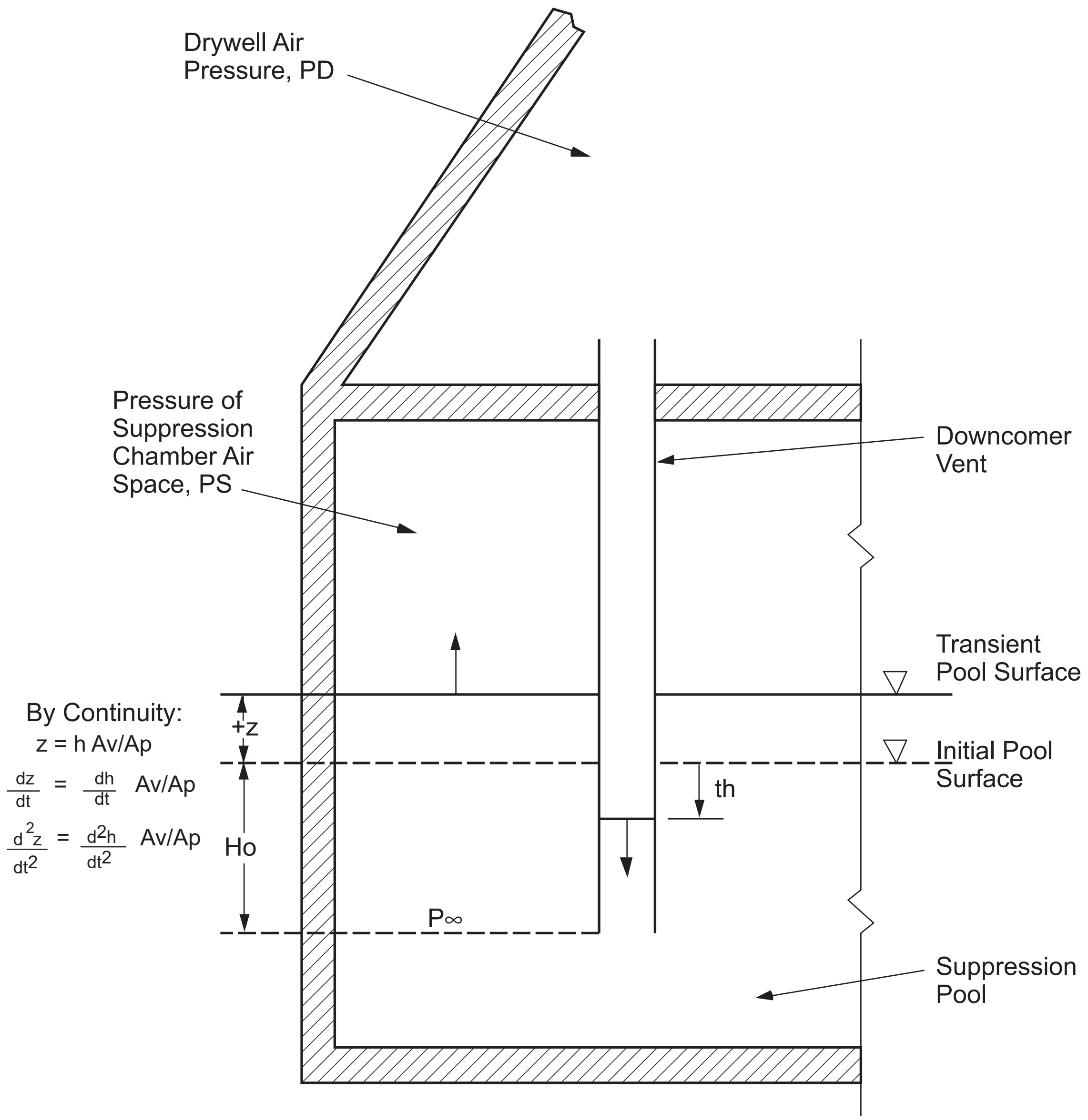
V = average pool surface velocity
$\dot{V}$ = average pool surface acceleration

[a] Source strength method documented in Reference 3A.D-9. Flow field data from SOURCE code.

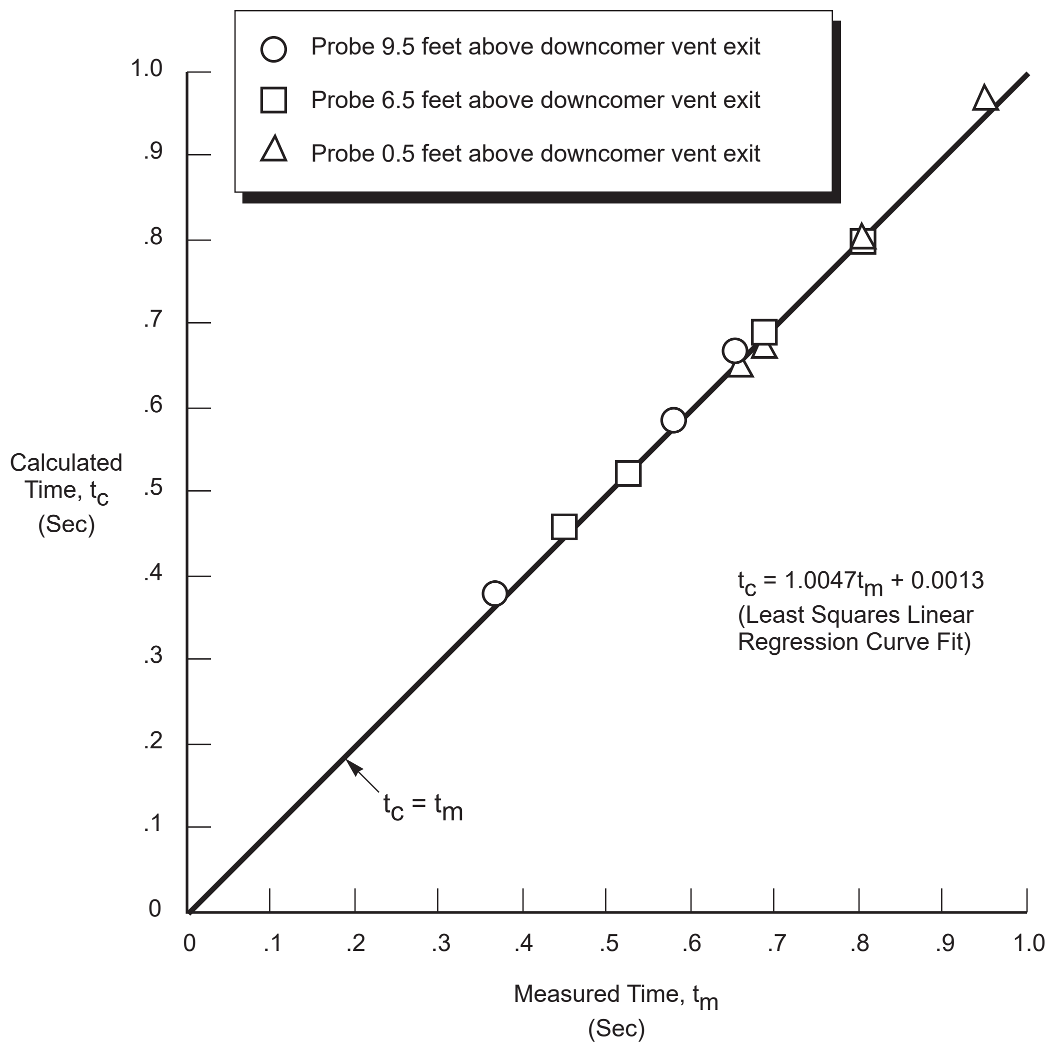
[b] Data obtained from Figures 3A.3.2-4 through 3A.3.2-7.

[c] Near bubble coalescence by Reference 3A.D-9 method.

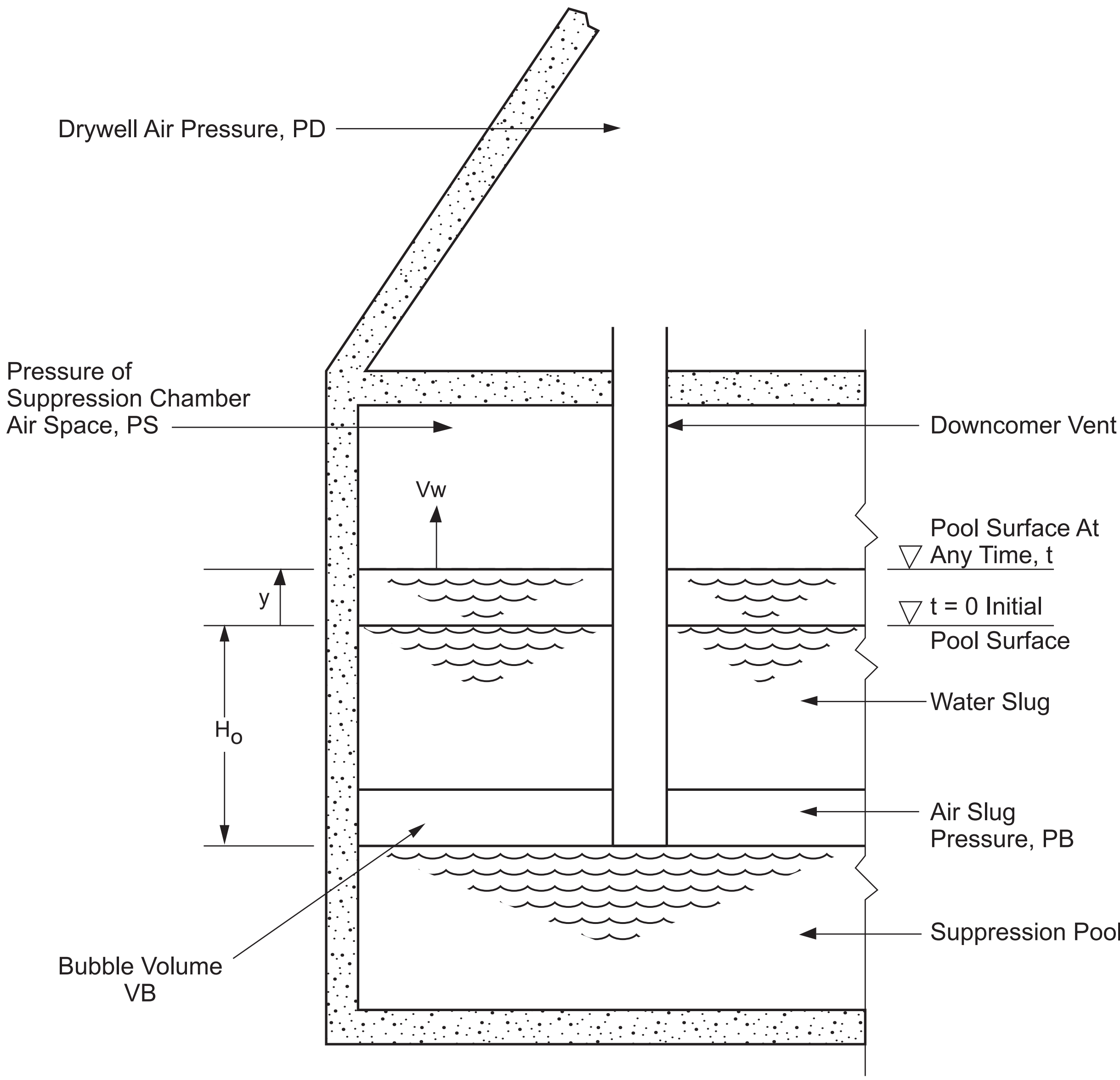
[d] Bubble coalescence time by SOURCE.



Schematic Representation of the Vent
Clearing Model



Comparison of Measured and Calculated 4T Test Probe Water Clearing Times



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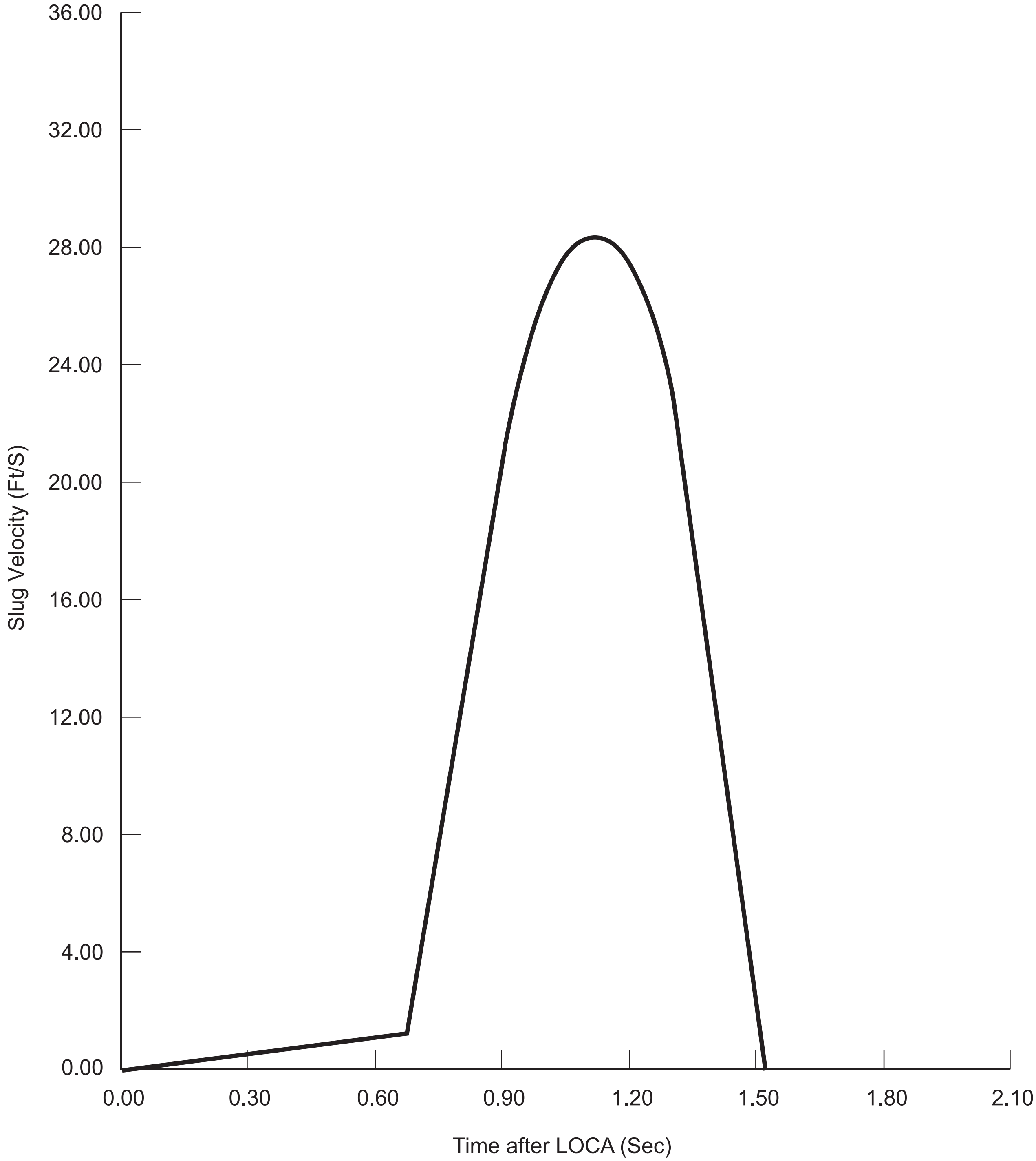
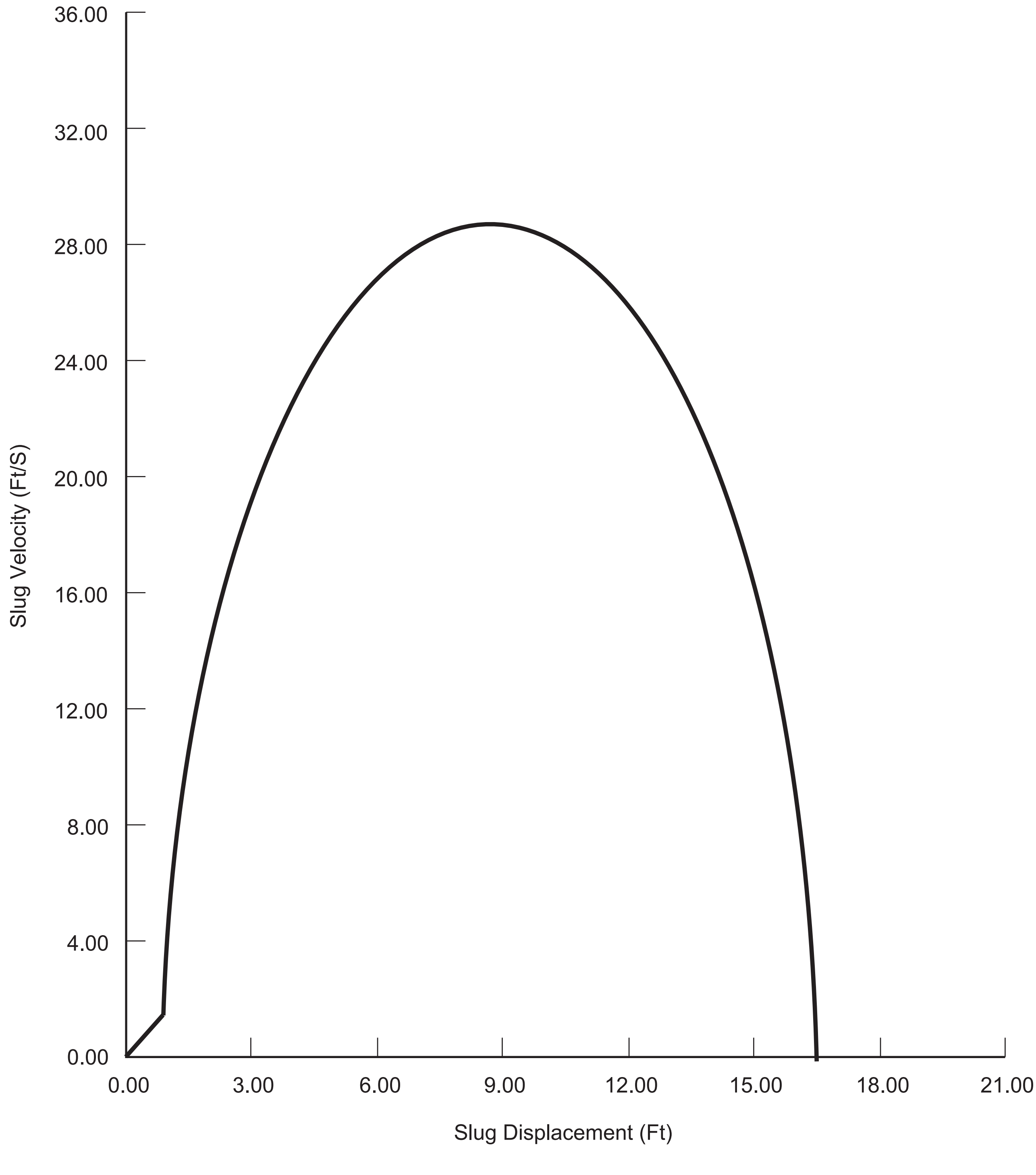
Schematic Representation of the
Pool Swell Model

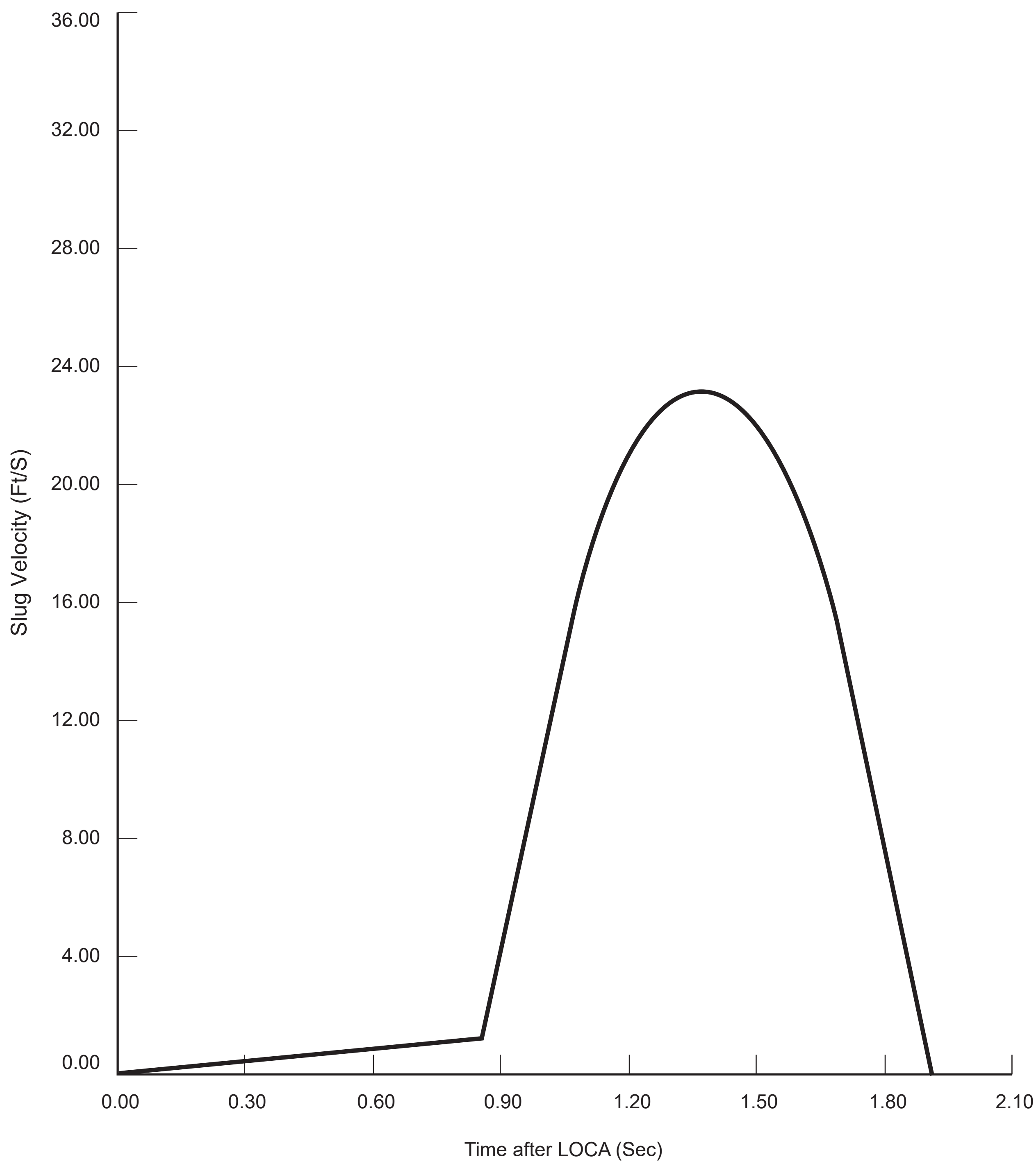
Draw. No. 900547.86 | Rev. | Figure 3A.D-3



**Benchmark I Plant -
Slug Velocity Versus Time**



Benchmark I Plant -
Slug Velocity Versus Displacement



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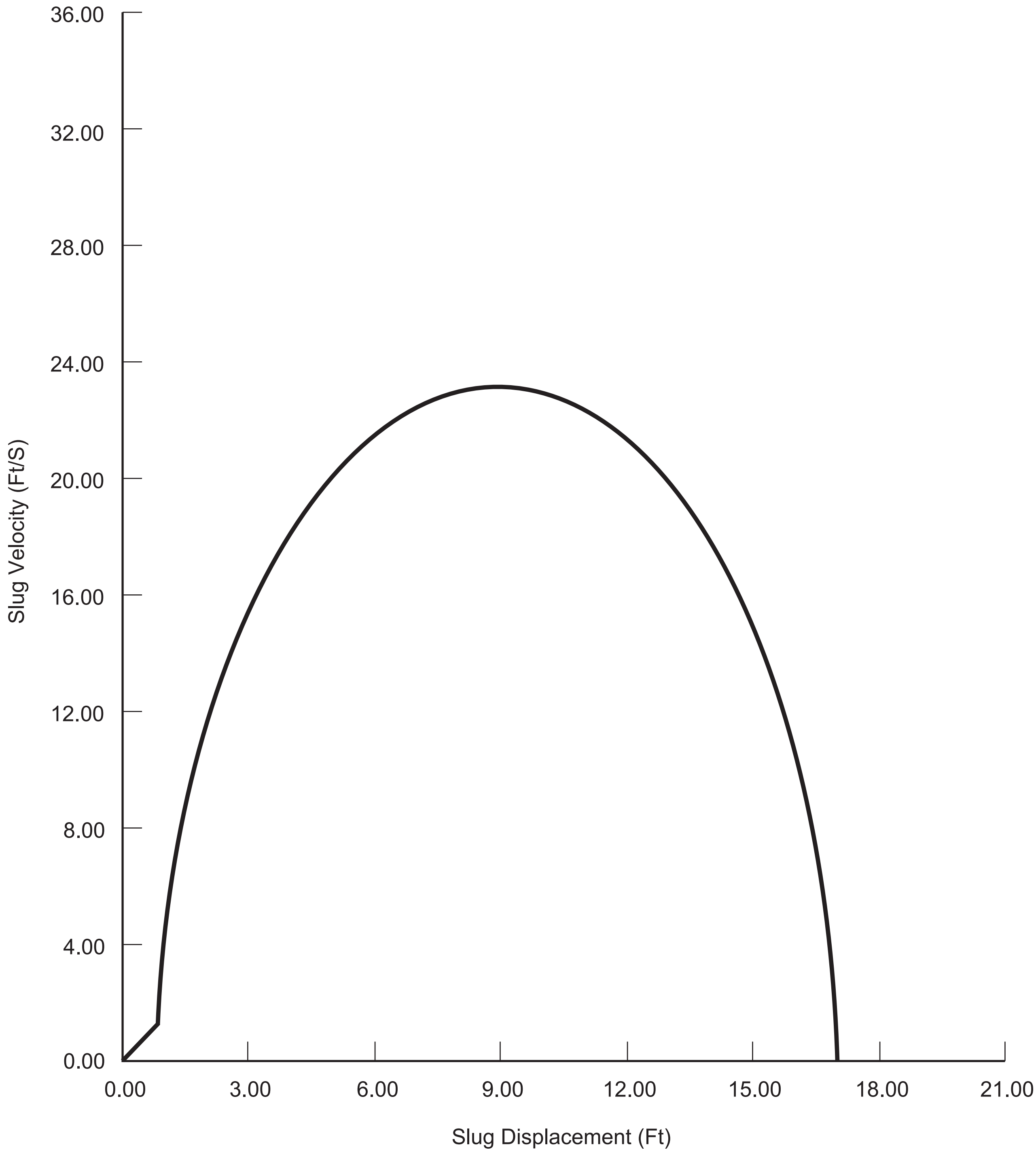
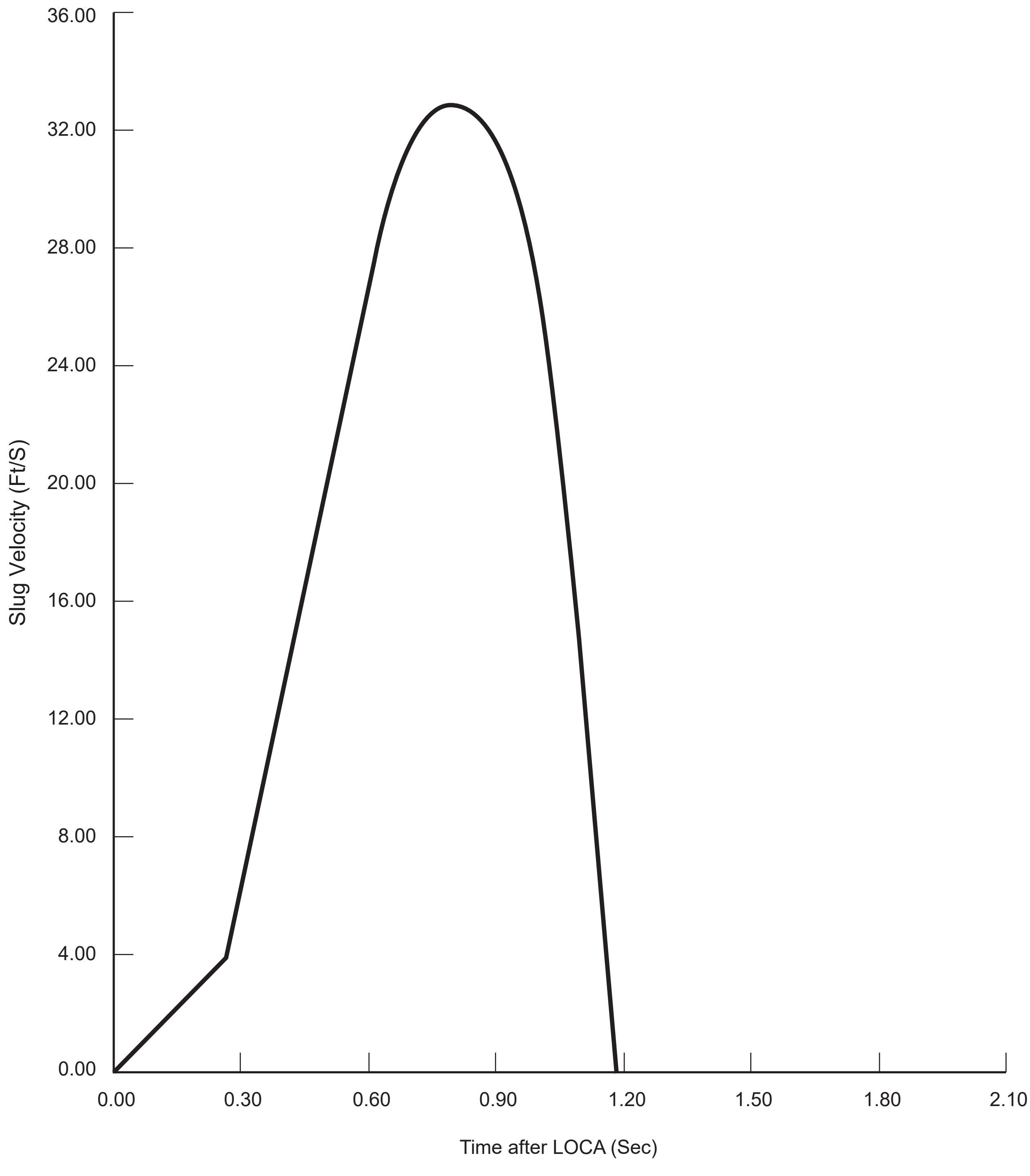
Final Safety Analysis Report

Benchmark II Plant - Slug Velocity Versus Time

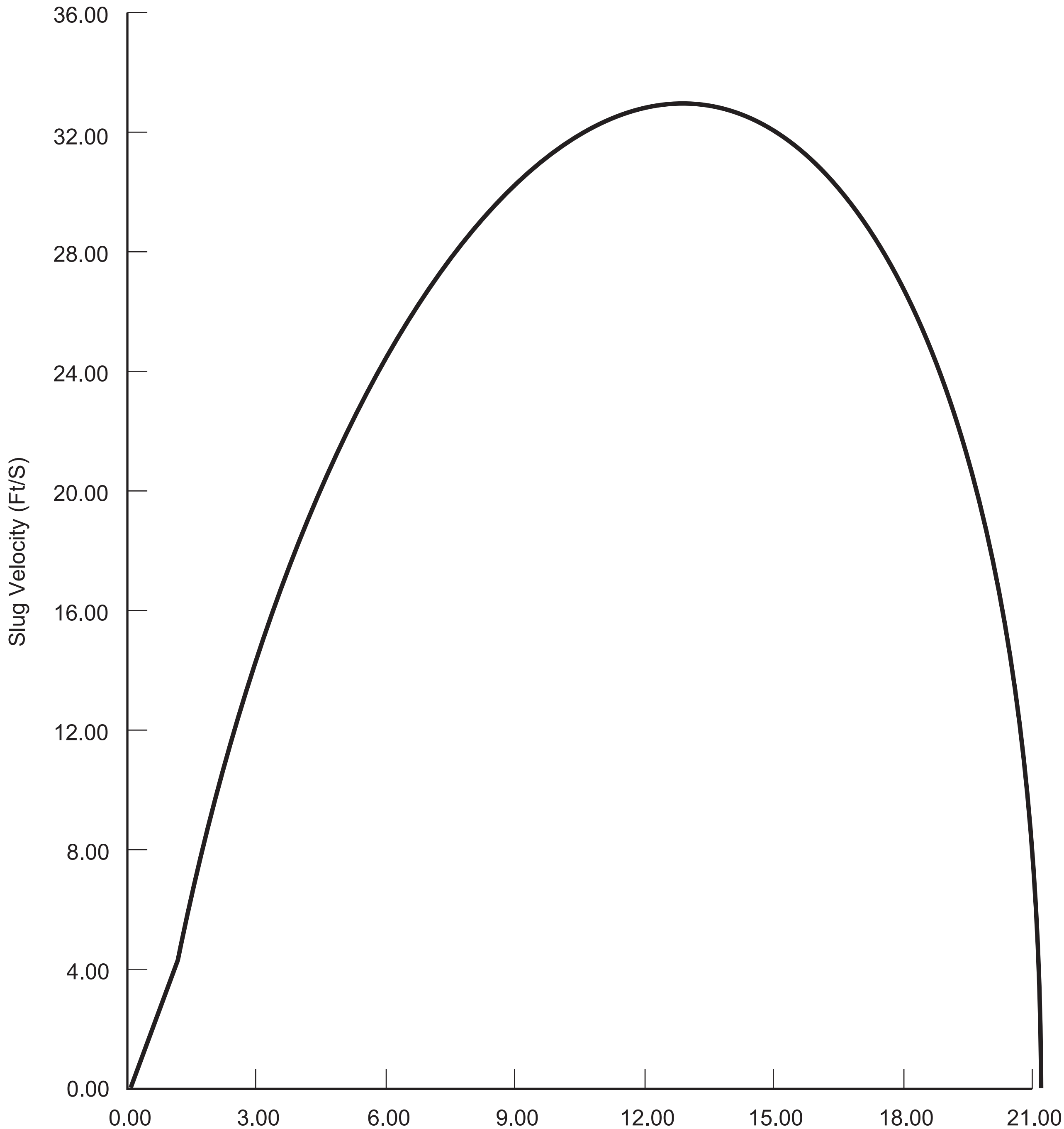
Draw. No. 900547.89	Rev.	Figure 3A.D-6



Benchmark II Plant -
Slug Velocity Versus Displacement



Benchmark III Plant -
Slug Velocity Versus Time



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Final Safety Analysis Report

Benchmark III Plant -
Slug Velocity Versus Displacement

Draw. No. **900547.92** | Rev. | Figure **3A.D-9**

Infinite Pool:

Steps

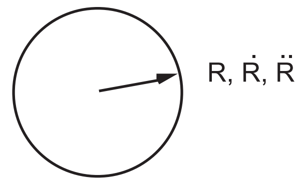
i) Spherical Bubble Dynamics Governed By Rayleigh Equation:

$$R\ddot{R} + \frac{3}{2}\dot{R}^2 = \frac{gc}{\ell}\left(P_B - P_\infty\right)$$

ii) From R, $\dot{R}$, And $\ddot{R}$ Get Source Strengths For Each Bubble

$$Q_i(t) = R^2\dot{R}$$

$$\dot{Q}_i(t) = R^2\ddot{R} + 2R\dot{R}^2$$



Note: $R \equiv R(t)$

$\dot{R} \equiv \frac{dR}{dt}$

$\ddot{R} \equiv \frac{d^2R}{dt^2}$

Application To Finite Pool:

iii) Determine "K_i" Factor for Each Bubble to Account for Finite Pool and Other Bubble Effects

iv) Modify Source Strength for Each Bubble to Account for Finite Pool and Other Bubble Effects:

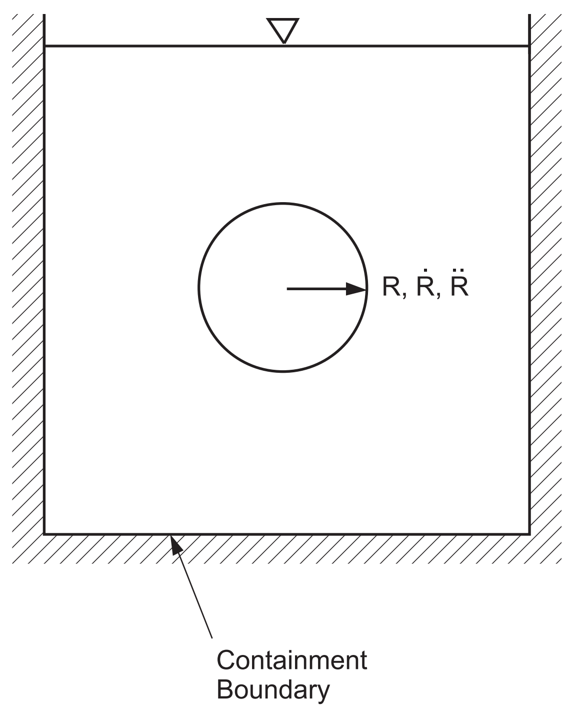
$$Q_i(t) = R^2\dot{R}K_i$$

$$Q_i(t) = R^2\dot{R} + 2R\dot{R}^2)K_i$$

v) Superposition Of All Bubbles To Get Flow Field: Velocity V, and Acceleration $\dot{V}$

$$V = -\sum_{i=1}^{102} Q_i(t)\,\nabla\phi_i$$

$$\dot{V} = -\sum_{i=1}^{102} \dot{Q}_i(t)\,\nabla\phi_i$$



Source: Reference 3A.D-9

Attachment 3A.E

SUPPRESSION POOL TEMPERATURE MONITORING SYSTEM

3A.E.1 DESIGN BASES

The suppression pool temperature monitoring (SPTM) system monitors the suppression pool bulk temperature with sensors distributed around the suppression pool. This system provides the main control room operator with the information necessary to avoid the conditions which might lead to the high-temperature steam quenching vibration phenomena mentioned below and discussed in detail in Section 3.5 of Reference 3A.E-1. This phenomena is not expected to occur when using a quencher discharge device. However, precautions using the SPTM system are designed to further make the occurrence of the vibrations impossible. Temperatures in the suppression pool are recorded and alarmed in the main control room.

The design basis for the SPTM system alarm setpoints provides the operator with adequate time to take the necessary action required to ensure that the conditions which are postulated to lead to high-temperature steam quenching vibrations do not occur. The design also provides the operator with the necessary information regarding localized heat-up of the pool water while the reactor vessel is being depressurized using the safety/relief valves (SRVs) when the SRVs are selected for actuation, they may be chosen so as to ensure mixing and uniformity of heat energy injection to the pool by monitoring the temperature sensors.

3A.E.1.1 High-Temperature Steam Quenching Vibrations

Boiling water reactor plants take advantage of the large thermal capacity of the suppression pool during plant transients which require SRV actuation. The discharge steam from each SRV is directed through a discharge line and a quencher device to the suppression pool where it is condensed. This results in an increase in pool water temperature, but a negligible increase in containment pressure.

However, certain events such as small pipe break have the potential for substantial energy addition to the suppression pool and could result in a high local pool temperature and the phenomenon of steam quenching vibration if timely corrective action is not taken. Suppression pool structural vibrations would occur during this condensing mode which would be forced by the periodic pulsation of the steam jet at the discharge.

The onset of the high-temperature steam quenching vibration phenomenon is a function of both the local suppression pool water temperature and the steam mass flux rate. The steam mass flux in the SRV piping in turn, is a function of the reactor vessel pressure.

3A.E.2 SYSTEM DESCRIPTION

The CGS SPTM system conformance to the criteria set by paragraph III.c of Reference 3A.E-2 is as discussed below. Specifically, the criteria for the upper ring of the sensors are:

1. Each monitoring location has two redundant type thermocouples monitored in the control room;

2. There are eight monitoring locations equally spaced about the outer containment perimeter;

3. The sensors are mounted 7 in. below the minimum technical specification water level;

4. All sensors are monitored and recorded in the control room;

5. Instrument setpoints for alarms will be set at or below the technical specification temperature values such that the plant can be shutdown and depressurized prior to the water in the suppression pool reaching a temperature at which condensation instabilities are postulated to occur; and

6. The SPTM system sensors are Seismic Category 1, Quality Class 1. The electrical power is Class 1E. Divisional separation is maintained.

In addition to the sensors described above which monitor bulk temperature, there is a second, lower ring of sensors. The lower ring meets Criteria 2, 4, 5, and 6, above. The degree of conformance to Criteria 1 and 3 for the lower ring of sensors is as follows:

1. Each monitoring location has one thermocouple Division 1 and 2 at alternate locations, and

3. The sensors are mounted at el. 447 ft 10.25 in., the approximate elevation of the quencher discharge devices.

Since warmer water is more buoyant, the upper ring provides a more conservative value for bulk temperature than the lower ring. The lower ring of sensors is provided to allow the operator to assess if significant vertical thermal stratification occurs.

3A.E.3 REFERENCES

3A.E-1 Mark II Containment Dynamic Forcing Function Information Report (DFFR), NEDO-21061, Revision 3, June 1978.

3A.E-2 Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria, Nuclear Regulatory Commission, NUREG-0487, October 1978.

Attachment 3A.F

COMPUTER PROGRAMS

The following are the programs referenced in this report:

3A.F.1 COMMERCIALLY AVAILABLE PROGRAMS

ANSYS

ANSYS is a large scale general purpose finite element computer program used for the solution of several classes of engineering analysis problems. Analytical capabilities include: static and dynamic analyses; plastic, creep and swelling analyses; and steady state and transient heat transfer analyses.

NASTRAN

NASTRAN is a large scale general purpose finite element computer program used for the solution of several classes of engineering analysis problems. Analytical capabilities include: static and dynamic analyses; thermal analyses; and the determination of eigenvalues for use in vibration analyses.

MCAUTO-STRUDL

MCAUTO-STRUDL is a commercially available computer program with general capability for the static and dynamic analysis of structures. The program, which is serviced and maintained by the McDonnell Douglas Automation Company, St. Louis, Missouri, has had wide commercial usage for many years.

ADLPIPE

ADLPIPE is a commercially available program used for the analysis of piping systems. Analytical capabilities include: static and dynamic analyses; and thermal analyses including thermal transient and fatigue evaluations for Class 1 piping systems.

FLUSH

FLUSH is a non-linear plain strain finite element seismic analysis program for soil-structure interaction analysis.

3A.F.2 BURNS AND ROE DEVELOPED PROGRAMS

BESSEL

BESSEL is a computer program which computes semi-analytical hydrodynamic added masses (incompressible fluids) for cylindrical and annular geometries

HYDI-1

HYDI-1 is a finite element program which computes hydrodynamic added masses and incident pressures for compressible fluids in three dimensional geometries.

FOX/HYDI-2

FOX/HYDI-2 is a finite element computer program used for the dynamic analysis for axisymmetric structures. The program performs the analysis by determining structural displacements in the frequency domain.

SWELL

This program is discussed in detail in Attachment 3A.D.

VENT

This program is discussed in detail in Attachment 3A.D.

SOURCE

This program is discussed in detail in Attachment 3A.B.

Attachment 3A.H

CONFORMANCE OF CGS DESIGN TO NRC ACCEPTANCE CRITERIA

Table 3A.H-1 is a summary of the CGS position for each of the pool dynamic loads. This table provides a description of each load or phenomenon, the Mark II Owner's Group load specification, the NRC evaluation reference, and the CGS position on the acceptance criteria for each load.

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
I. Loss-of-coolant accident (LOCA)-related hydrodynamic loads			
A. Submerged boundary loads during vent clearing	24 psi over-pressure added to local hydrostatic below vent exit (walls and basemat) - linear attenuation to pool surface.	II.A.1 [a]	Acceptable
B. Pool swell loads			
1. Pool swell analytical model (PSAM)			
a) Air bubble pressure	Calculated by the PSAM used in calculation of submerged boundary loads.	III.B.3.a [b]	Acceptable
b) Pool swell elevation	Use PSAM with polytropic exponent of 1.2 to a maximum swell height which is the greater of 1.5 vent submergence or the elevation corresponding to the drywell floor uplift P per NUREG 0487 criteria I.A.4. The associated maximum wetwell air compression is used for design assessment.	II.A.2 [c]	Acceptable

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
c) Pool swell velocity	Velocity history vs. pool elevation predicted by the PSAM used to compute impact loading on small structures and drag on gratings between initial pool surface and maximum pool elevation and steady-state drag between vent exit and maximum pool elevation. Analytical velocity variation is used up to maximum velocity. Maximum velocity applies thereafter up to maximum pool swell. PSAM predicted velocities multiplied by a factor of 1.1.	III.B.3.a.3 [a]	Acceptable
d) Pool swell acceleration	Acceleration predicted by the PSAM. Pool acceleration is utilized in the calculation of acceleration loads on submerged components during pool swell.	III.B.3.a.4 [b]	Acceptable
e) Wetwell air compression	Wetwell air compression is calculated by PSAM.	II.A.2 [c]	Acceptable
f) Drywell pressure	Methods of NEDM-10320 and NEDO-20533 Appendix B. Utilized in PSAM to calculate pool swell loads.	III.B.3.a.6 [b]	Acceptable

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
2. Loads on submerged boundaries	Maximum bubble pressure predicted by the PSAM added uniformly to local hydrostatic below vent exit (walls and basemat) liner attenuation to pool surface. Applied to walls up to maximum pool elevation.	III.B.3.b [b]	Acceptable
3. Impact loads			
a) Small structures	1.35 x pressure-velocity correlation for pipes and I beams based on PSTF impulse data and flat pool assumption. Variable pulse duration.	III.B.3.c.1 [b]	Acceptable
b) Large structures	None - Plant-unique load where applicable.	III.B.3.c.6 [b] Criteria A.5 [a]	Acceptable. CGS has no large structures in the pool swell zone
c) Grating	No impact load specified. P drag vs. open area correlation and velocity vs. elevation history from the PSAM. P drag multiplied by dynamic load factor.	III.B.3.c.3 [b] Criteria A.3 [a]	Acceptable

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
4. Wetwell air compression			
a) Wall loads	Direct application to the PSAM calculated pressure due to wetwell compression.	III.B.3.d.1 [b]	Acceptable
b) Diaphragm floor upward loads	5.5 psid for diaphragm floor loadings only.	2.12.7 [a]	Acceptable
5. Asymmetric pool LOCA	Use 20% of maximum pressure statistically applied to 1/2 of the submerged bubble.	II.A.3 [c] Criteria A-4 [a]	Acceptable
C. Steam condensation and chugging loads			
1. Downcomer lateral loads			
a) Single vent loads (24 in.)	Use single vent dynamic lateral load developed in NEDE-23806.	2.3.3.2 [a] Criteria B.1.a [a]	Acceptable
b) Multiple vent loads (24 in.)	Use multivent dynamic lateral load developed in NEDE-24106-P and NEDE-24794-P.	2.3.3.3 [a]	Acceptable
c) Single/multiple vent loads (28 in.)	Multiply basic vent loads by factor f=1.34	2.3.2.1 [a] B.1.b [a]	Acceptable

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
2. Submerged boundary loads			
a) High/medium steam flux condensation oscillation (CO) load	Use method described in NEDE-24288-P[d]	2.2.2.1.3 [a]	CO loads are not governing design condition for CGS
b) Low steam flux chugging loads	Representative pressure fluctuation taken from 4TCO (NEDE-24285-P) test added to local hydrostatic	2.2.2.3 [a]	Plant unique. Chugging report entitled "Chugging Loads-Revised. Definition and Application Methodology for Mark II Containments" submitted July 1981
- Uniform loading conditions	Use method described in NEDE-24302-P[d]		See above
- Asymmetric loading	Representative pressure fluctuation taken from 4TCO test [NEDE-24285-P] applied as described in NEDE-24822-P.		See above

Table 3AH-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
II. Safety/relief valve (SRV)-related hydrodynamic loads			
A. Pool temperature limits for X-quencher	20°F subcooling at quencher elevation for steam mass flux of 42 lb/ft^2-sec or less. 200°F for steam flux greater than 94 lb/ft^{2}-sec.	6.2.1.8.8 (5) A (4)	Acceptable
B. Quencher air clearing loads	Mark II plants utilizing the four arm quencher, use quencher load methodology described in DFFR.	Criteria II.2 [b]	CGS Plant unique SRV (x-quencher) load report entitled "SRV Loads - Definition and Application Methodology for Mark II Containments" submitted August 1980
C. Quencher arm and tie-down loads	Includes vertical and lateral arm load transmitted to the basemat via the tie-down.	III.C.2.e.2 [b]	Acceptable
1) X-quencher arm loads	Vertical and lateral loads developed on the basis of bounding assumption for air/water discharge from the quencher and conservative combinations of maximum/ minimum bubble pressure acting on the quencher.	III.C.2.e.1	Acceptable

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
2. X-quencher tie-down loads	II.C.1 above plus vertical transient wave and thrust loads. Thrust load calculated using a standard momentum balance. Vertical and lateral moments for air or water clearing are calculated based on conservative clearing assumptions.	III.C.2.e.2 [b]	Acceptable
III. <u>LOCA/SRV submerged structure loads</u>			
A. SRV air bubble loads			
1. Standard drag in Accelerating flow fields	Drag Coefficients are presented in Attachment 1.k of the Zimmer FSAR.	Acceptable with the following modification: 1) Use $C_H = C_M$-1 in the F_A formula 2) For noncylindrical structures use lift coefficient for appropriate shape or $C_L = 1.6$	Generic methodology acceptable. (Amplitudes for SRV loads verified by CAORSO data on submerged structures).

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
		3) The standard drag coefficient for pool swell and SRV oscillating bubbles should be based on data for structures with sharp edges	
2. Equivalent uniform flow velocity and acceleration	Structures are segmented into small sections such that $1.0 \leq L/D \leq 1.5$. The loads are then applied to the geometric center of each segment.	Acceptable	See III. A.1. above
3. Interference effects	A detailed methodology is presented in Attachment 1.k of the Zimmer FSAR.	Acceptable	See III. A.1 above
B. LOCA jet loads	Calculated by the Ring Vortex Model.	2.2.4.3 [a]	Acceptable

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
C. Steam condensation drag loads	No generic load methodology provided	CGS load specification and NRC review is addressed in CGS SER	Generic “drag load” methodology acceptable Plant unique flow fields are consistent with I.C.2.a and I.C.2.b of this table. (See DAR Attachment 3A.I)
IV. <u>Secondary loads</u>			
A. Sonic wave load	Negligible load - none specified	Acceptable	Acceptable
B. Compressive wave load	Negligible load - none specified	Acceptable	Acceptable
C. Post swell wave load	No generic load provided	Plant unique load specification addressed in CGS SER	
D. Seismic slosh load	No generic load provided	Plant unique load specification addressed in CGS SER	

Table 3A.H-1

Conformance of CGS Design to NRC Acceptance Criteria (Continued)

Load or Phenomenon	Mark II Owners Group Load Specification	NRC Evaluation	CGS Position on Acceptance Criteria
E. Fallback load on submerged boundary	Negligible load - none specified	Acceptable	Acceptable
F. Thrust loads	Momentum balance	Acceptable	Acceptable
G. Friction drag loads on vents	Standard friction drag calculations	Acceptable	Acceptable
H. Vent clearing loads	Negligible load - none specified	Acceptable	Acceptable

[a] NRC Acceptance Criteria set forth in NUREG-0808.

[b] NRC Acceptance Criteria set forth in NUREG-0487.

[c] NRC Acceptance Criteria set forth in Supplement 1 of NUREG-0487.

[d] NRC Acceptance Criteria set forth in WNP-2 SER NUREG (0892).

Attachment 3A.I

SAFETY/RELIEF VALVE AND LOSS-OF-COOLANT ACCIDENT LOADS ON SUBMERGED STRUCTURES

3A.I.1 INTRODUCTION

The loss-of-coolant accident (LOCA)/safety/relief valve (SRV) discharge devices and other submerged structures are shown in Figures *3A.2.1-2, 3A.2.1-6, 3A.2.1-7, and 3A.2.1-8* and identified in Table 3A.I-1.

The most significant hydrodynamic load for each structure is identified in Table 3A.I-1.

3A.I.2 SUMMARY OF METHODOLOGY USED FOR DEFINING LOSS-OF-COOLANT ACCIDENT JET/BUBBLE LOADS

Loss-of-coolant accident jet/bubble loads are defined using the ring vortex model. The pool is divided into zones and to ensure conservatism in design, the largest velocity and acceleration values seen by a submerged structure are assumed equal to the maximum calculated values anywhere in the applicable zone. The LOCA bubble charging model is used to verify/ensure that the design values are conservative.

3A.I.3 SUMMARY OF METHODOLOGY USED FOR DEFINING LOSS-OF-COOLANT ACCIDENT STEAM CONDENSATION LOADS

Generic "drag load" methodology and plant unique flow fields are used for LOCA steam condensation loads on submerged structures in compliance with the NRC acceptance criteria. Plant unique flow fields are defined consistently with steam condensation boundary loads.

The generic methodology identifies three components of flow induced loads on submerged structures: acceleration dependent and velocity square dependent in-line loads, velocity square dependent lift load (normal to the direction of flow).

Representative plant unique chugging flow fields show that the chugging loads on submerged structures are due to acceleration or pressure gradients established in the pool during the impulsive chugging phenomenon, i.e., velocity dependent loads are small.

3A.I.4 SUMMARY OF METHODOLOGY USED FOR DEFINING SAFETY/RELIEF VALVE LOADS

Caorso SRV test data on submerged structures are examined to supplement theoretical approaches of the acceptance criteria. The data and their correlation with theoretical approaches of the acceptance criteria confirm that SRV loads are primarily due to pressure

gradients established in the pool during the SRV discharge, i.e., velocity dependent loads are small.

The dynamic pressure gradients measured across Caorso column, vent and SRV line are used to define the peak load values (at quencher elevation), the spatial distribution of the load and its time dependence.

The pressure time histories recorded on submerged structures show waveform characteristics similar to those recorded at pool boundary. The SRV loads on submerged structures are defined consistently with the plant unique boundary loads.

The SRV loads on Columbia Generating Station structures are calculated using the following formula:

$$P = \left[\frac{\pi D^2}{4}\right]\left[\frac{d^2_{Caorso}}{d^2_{WNP2}}\right] \alpha\, P_b\, L$$

where:

P = load on a structure (force/unit length)

D = diameter of the structure

α = a load gradient factor established using Caorso SRV test data on submerged structures. The method to calculate (α) is explained in the notes for and in Figure 3A.I-1

d_{Caorso} = horizontal distance of the structure from the nearest actuating quencher in Caorso plant

d_{WNP2} = horizontal distance of the structure from the nearest actuating quencher

P_b = boundary pressure load definition from Reference 3A.I-1 including any modifications agreed upon with the NRC

L = load margin = minimum value of 1.4 is used for all piping which are adequately braced and a value of 2.0 is used for the column which is the only unbraced structure and is closest to the nearest quencher

Notes on Figure 3A.I-1

1. The SRV load gradient is obtained from Caorso data as follows:

$$A = \frac{P_f - P_{ba}}{D} = \alpha P_{19}$$

where:

A = measured gradient across the cylindrical structure

P_f = P_{front}

P_{ba} = P_{back}

2. P_{19}, P_f, P_{ba} waveform characteristics are similar.

3. The value of (α) for each set of P_f (P_{42}, P_{41}, P_{33}, P_{24}) and P_{ba} (P_{40}, P_{39}, P_{53}) is obtained from Caorso SRV test data (single and multiple valve actuations).

4. For miscellaneous piping which run along the suppression pool boundary, the load gradient factor (α) equal to that for the column is specified.

3A.I.5 REFERENCES

3A.I-1 "SRV Loads - Improved Definition and Application Methodology for Mark II Containments," Technical Report (Proprietary), prepared by Burns and Roe, Inc. for application to Washington Public Power Supply System Nuclear Project No. 2, submitted to the Nuclear Regulatory Commission on 7/29/80.

Table 3A.I-1

Loss-of-Coolant Accident/Safety/Relief Valve Loads on Submerged Structures

Identification of Structures	Identification of Most Significant Hydrodynamic Load
1. (a) SRV line	SRV (due to actuation of adjacent SRV) [a]
(b) Quencher [b]	LOCA jet on arms
(c) Quencher Support [b]	None significant
2. Downcomer vents	SRV
3. Concrete columns	SRV
4. Bracing truss [b] at vent exit	Pool swell drag
5. Platform with grating (at el. 472 ft 4 in., 78% open area)	Pool swell drag
6. Miscellaneous piping, penetrations and supports along containment boundary	
(a) Below vent exit (el. 454 ft 4.75 in.)	LOCA jet and SRV [a]
(b) Above vent exit, below initial pool surface (el. 466 ft 4.75 in.)	Pool swell drag
(c) Above initial pool surface, below maximum pool swell (el. (484 ft 4.75 in.)	

[a] See also discussion presented in Reference 3A.3.1-8.

[b] Loads on discharge devices and their supports during discharge through the devices are addressed elsewhere.