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3.0   DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1   CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1   SUMMARY DESCRIPTION

This section contains an evaluation of the design bases of Perry Nuclear
Power Plant (PNPP) as measured against the NRC General Design Criteria
for Nuclear Power Plants, <10 CFR 50, Appendix A>, effective May 21,
1971, and subsequently amended July 7, 1971. The NRC General Design
Criteria which are divided into six groups and total 55 in number are
intended to establish minimum requirements for the design of nuclear
power plants.

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

Based on the content herein, the Perry Nuclear Power Plant fully
satisfies and is in compliance with the NRC General Design Criteria.

3.1.2   CRITERION CONFORMANCE

3.1.2.1   <u>Group I, Overall Requirements (Criteria 1-5)</u>

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
plant unit licensee throughout the life of the unit.

3.1.2.1.1.1   Evaluation Against Criterion 1

Structures, systems and components important to safety are listed in
<Table 3.2-1>. The total quality assurance program is described in
<Chapter 17> and is applied to the items identified by a quality group
classification in this table. The intent of the quality assurance
program is to assure sound engineering in all phases of design and
construction through conformance to regulatory requirements and design
bases described in the license application. In addition, the program
assures adherence to specified standards of workmanship and
implementation of recognized codes and standards in fabrication and
construction. It also includes the observance of proper preoperational
and operational testing and maintenance procedures as well as the

documentation of the foregoing by keeping appropriate records. The total quality assurance program of the applicant and its principal contractors is responsive to and satisfies the intent of the quality-related requirements of <10 CFR 50>, including <10 CFR 50, Appendix B>.

Structures, systems and components are first classified in Chapter 3 with respect to their relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment in these classifications as necessary to assure a quality product in keeping with the required safety function.

Records are maintained which demonstrate that all 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 used, 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 quality-related information is retrievable for reference. These records will be maintained during the life of the operating licenses.

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 <Section 1.2>

b.   Plant Description <Section 1.2>

c.   Classification of Structures, Components and Systems <Section 3.2>

d.   Quality Assurance Program <Section 17.0>

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

3.1.2.1.2.1   Evaluation Against Criterion 2

The design criteria adopted for structures, systems and components
depend on the magnitude and the probability of occurrence of natural
phenomena at this specific site. The designs are based on the most
severe of the 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 SAR Sections listed below.

The design criteria meet the requirements of Criterion 2.

For further discussion, see the following sections:

a.   Meteorology <Section 2.3>

b.   Hydrologic Engineering <Section 2.4>

c.   Geology and Seismology <Section 2.5>

d.   Classification of Structures, Components and Systems <Section 3.2>

e.   Wind and Tornado Loadings <Section 3.3>

f.   Water Level (Flood) Design <Section 3.4>

g.   Missile Protection <Section 3.5>

h.   Seismic Design <Section 3.7>

i.   Design of Seismic Category I Structures <Section 3.8>

j.   Mechanical Systems and Components <Section 3.9>

k.   Seismic and Dynamic Qualification of Mechanical and Electrical
      Equipment <Section 3.10>

3.1.2.1.3   Criterion 3 - Fire Protection

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

3.1.2.1.3.1   Evaluation Against Criterion 3

Structures are designed and located to avoid the propagation and minimize the effects of fire. This is done by the location of fire walls and by using noncombustible building materials where possible. Systems are designed to minimize their effect on initiating or contributing to the spread of a fire. Components are selected, where possible, to be of noncombustible materials, thus minimizing the possibility of fueling a fire.

The plant fire protection system includes the following provisions:

a.   Automatic fire detection equipment in those areas where fire danger is greatest.

b.   Fire fighting systems include sprinkler, water spray, preaction, carbon dioxide, and foam systems, installed to provide control and extinguishment of various fire hazards. In addition, hose stations and portable fire extinguishers are provided throughout the plant. These fire fighting systems are designed so that rupture or inadvertent operation will not significantly impair the safety capability of structures, systems and components.

The design of the fire protection system thus meets the requirements of Criterion 3.

For further discussion, see the following sections:

a.   Construction Materials <Section 3.8>

b.   Separation Criteria <Section 8.3>

c.   Instrumentation and Controls <Section 7.0>

d.   Electric Power <Section 8.0>

e.   Fire Protection <Section 9.5.1>

f.   Conduct of Operations <Section 13.0>

3.1.2.1.4   Criterion 4 - Environmental and Missiles 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.

3.1.2.1.4.1   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 loss-of-coolant accidents.

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 and from
events and conditions outside the plant.

The electrical equipment, instrumentation and associated cable for
protection of engineered safety features systems which are located
inside the containment are discussed in the SAR sections listed below

indicating the design requirements in terms of the time which each must survive the extreme environmental conditions following the loss-of-coolant accident.

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

For further discussion, see the following sections:

a.   Meteorology <Section 2.3>

b.   Hydrologic Engineering <Section 2.4>

c.   Geology and Seismology <Section 2.5>

d.   Classification of Structures, Components and Systems <Section 3.2>

e.   Wind and Tornado Loadings <Section 3.3>

f.   Water Level (Flood) Design <Section 3.4>

g.   Missile Protection <Section 3.5>

h.   Criteria for Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping <Section 3.6>

i.   Seismic Design <Section 3.7>

j.   Design of Seismic Category I Structures <Section 3.8>

k.   Mechanical Systems and Components <Section 3.9>

l.   Seismic and Dynamic Qualification of Mechanical and Electrical
     Equipment <Section 3.10>

m.   Environmental Design of Mechanical and Electrical Equipment
     <Section 3.11>

n.   Integrity of Reactor Coolant Pressure Boundary <Section 5.2>

o.   Engineered Safety Features <Section 6.0>

p.   Instrumentation and Controls <Section 7.0>

q.   Electric Power <Section 8.0>

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.

3.1.2.1.5.1   Evaluation Against Criterion 5

Originally, Perry Nuclear Power Plant Unit 1 and Unit 2 were designed to
share certain common facilities and to have some cross-ties between
systems. Since construction of Unit 2 was terminated prior to
completion of the unit, structures, systems, and components that were
originally to be shared by both units or intended for Unit 2 but are now
used to support Unit 1 are not shared by more than one unit, and GDC 5
does not apply.

(Deleted)

(Deleted)

(Deleted)

(Deleted)

3.1.2.2   Group II, Protection by Multiple Fission Product Barriers
            (Criteria 10-19)

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.

3.1.2.2.1.1   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 provide high integrity over a complete range of power levels including transient conditions. The core is sized with sufficient heat transfer area and coolant flow to ensure that fuel design limits are not exceeded under normal conditions or anticipated operational occurrences.

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

An analysis and evaluation has been made of the effects upon core fuel following adverse plant operating conditions. The results of abnormal operational transients are presented in <Chapter 15.0> of the SAR and show that the MCPR does not fall below the transient MCPR limit, thereby satisfying the transient design basis. Oscillation Power Range Monitor (OPRM) system provides compliance with Criterion 10 <Section 7.6.1.4.4>.

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Plant Description <Section 1.2>

c.   Fuel Mechanical Design <Section 4.2>

d.   Nuclear Design <Section 4.3>

e.   Thermal and Hydraulic Design <Section 4.4>

f.   Reactor Recirculation System <Section 5.4>

g.   Reactor Core Isolation Cooling System <Section 5.4>

h.   Residual Heat Removal System <Section 5.4>

i.   Accident Analysis <Section 15.0>

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 tend to compensate for a rapid increase in reactivity.

3.1.2.2.2.1   Evaluation Against Criterion 11

The reactor core is designed to have a reactivity response that regulates or dampens 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:

a.   Fuel temperature or Doppler coefficient

b.   Moderator void coefficient

c.   Moderator temperature coefficient

The combined effect of these coefficients in the power range is termed the power coefficient.

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

In a boiling water reactor, the 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 void coefficient of reactivity, the BWR has a number of inherent advantages, such as:

a.   Use of recirculation coolant flow as opposed to control rods for load following

b.   Inherent self-flattening of the radial power distribution

c.   Ease of control

d.   Spatial xenon stability.

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative. Typically, the power coefficient at full power is about -0.04 $\Delta k/k/\Delta P/P$ at the beginning of life and about -0.03 $\Delta k/k/\Delta P/P$ at 10,000 MWd/t. These values are well within the range required for adequate damping of power and spatial xenon disturbances.

The reactor core and associated coolant system is 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 <Section 1.2>

b.   Nuclear Design <Section 4.3>

c.   Thermal and Hydraulic Design <Section 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.

3.1.2.2.3.1   Evaluation Against Criterion 12

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity

coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large boiling water reactors, under-damped, unacceptable power distribution behavior could only be expected to occur with power coefficients more positive than about $-.01 \Delta k/k/\Delta P/P$. Operating experience has shown large boiling water reactors 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 reactor protection system design provides protection from excessive fuel cladding temperatures and protects the reactor coolant pressure boundary from excessive pressures which threaten the integrity of the system. Local abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation. Oscillation Power Range Monitor (OPRM) system provides compliance with Criterion 12 <Section 7.6.1.4.4>.

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 assure that Criterion 12 is met.

For further discussion see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Nuclear Design <Section 4.3>

c.   Thermal and Hydraulic Design <Section 4.4>

d.   Rod Control and Information System <Section 7.7>

e.   Accident Analysis <Section 15.0>

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.

3.1.2.2.4.1   Evaluation Against Criterion 13

The neutron flux in the reactor core is monitored by six subsystems. The Source Range Monitor (SRM) subsystem measures the flux from startup through criticality. The Intermediate Range Monitor (IRM) subsystem overlaps the SRM subsystem and extends into the power range. The power range is monitored by many detectors which make up the Local Power Range Monitor (LPRM) subsystem. The output from these detectors is used in many ways. Core-wide sets of detectors can be averaged to provide a core-average neutron flux. This output is called the Average Power Range Monitor (APRM) subsystem. Additionally, the LPRM detectors

provide input to the Oscillation Power Range Monitor (OPRM) subsystem to provide a profile of the reactor core for the detection of a reactor thermal-hydraulic instability. The Traversing Incore Probe (TIP) subsystem provides a means for calibrating the LPRM system. The IRM, APRM and OPRM subsystems generate scram trips to the reactor trip system. All subsystems but the OPRM and TIP subsystems generate rod-block trips. Additional information on the neutron monitoring system is given in <Chapter 7>.

The reactor protection system protects the fuel barriers and the 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 functions 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 reactor coolant pressure boundary, 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 reactor coolant pressure boundary. Nuclear system leakage rates are classified as identified and unidentified, which correspond respectively to equipment drain and floor drain sump flows. The permissible total leakage rate limit to these sumps is based upon the makeup capabilities of various reactor component systems. Flow integrators 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 integrity of the process barrier is threatened.

The process radiation monitoring system monitors radiation levels of various processes, providing trip signals to the reactor protection system and containment and reactor vessel isolation control system whenever pre-established limits are exceeded.

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Reactivity Control Mechanical Design <Section 4.2>

c.   Reactor Coolant Pressure Boundary <Section 5.2>

d.   Main Steam Isolation System <Section 5.4>

e.   Containment Systems <Section 6.2>

f.   Reactor Protection System <Section 7.2>

g.   Containment and Reactor Vessel Isolation Control System <Section 7.3>

h.   Neutron Monitoring System <Section 7.6>

i.   Reactor Vessel-Instrumentation and Control <Section 7.7>

j.   Reactor Manual Control System <Section 7.7>

k.   Recirculation Flow Control System <Section 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, rapidly propagating failure and gross rupture.

3.1.2.2.5.1   Evaluation Against Criterion 14

The piping and equipment pressure parts within the reactor coolant pressure boundary through the outer isolation valve(s) are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. <Section 3.2> classifies the systems and components with the reactor coolant pressure boundary as Quality Group A. 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.

In order to minimize the possibility of brittle fracture within the reactor coolant pressure boundary, 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 notch toughness properties by selecting fine-grained steels and limiting neutron exposure of materials. Materials are impact tested in accordance with ASME Boiler and Pressure Vessel Code, Section III where applicable. Where reactor coolant pressure boundary piping penetrates the containment, the fracture toughness temperature requirements of the reactor coolant pressure boundary materials apply.

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

<Section 5.2> and <Section 5.3> contain the detailed material and examination requirements for the piping and equipment of the reactor coolant pressure boundary prior to and after its assembly and erection. Leakage testing and surveillance is accomplished as described in the evaluation against Criterion 30 of the General Design Criteria.

The design, fabrication, erection, and testing of the reactor coolant pressure boundary assure 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 <Section 1.2>

b.   Design of Structures, Components, Equipment, and Systems
      <Section 3.0>

c.   Overpressurization Protection <Section 5.2>

d.   Reactor Vessel and Appurtenances <Section 5.3>

e.   Reactor Recirculation System <Section 5.4>

f.   Accident Analysis <Section 15.0>

g.   Quality Assurance Program <Section 17.0>

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.

3.1.2.2.6.1   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 system, and the residual heat removal system. These systems are
designed, fabricated, erected, and tested to stringent quality
requirements and appropriate codes and standards which assure high
integrity of the reactor coolant pressure boundary throughout the plant
lifetime. The reactor coolant system is designed and fabricated to meet
the requirements of the ASME Boiler and Pressure Vessel Code,
Section III as indicated in <Chapter 3.0>.

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

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

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Design of Structures, Components, Equipment, and Systems
     <Section 3.0>

c.   Overpressurization Protection <Section 5.2.2>

d.   Reactor Coolant Pressure Boundary Leakage Detection System
     <Section 5.2.5>

e.   Reactor Vessel <Section 5.3>

f.   Reactor Recirculation System <Section 5.4>

g.   Accident Analyses <Section 15.0>

3.1.2.2.7   Criterion 16 - Containment Design

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

3.1.2.2.7.1   Evaluation Against Criterion 16

The containment system consists of the following major components:

a.   A drywell enclosing the reactor pressure vessel, the reactor
      coolant recirculation loops and pumps and other branch connections
      of the reactor coolant system. The drywell is a cylindrical
      reinforced concrete structure with a removable steel head.

b.   A suppression pool containing a large amount of water used to
      rapidly condense steam from a reactor vessel blowdown or from a
      break in a major pipe.

c.   A leak tight steel containment vessel surrounding the drywell and
      the suppression pool.

d.   A reinforced concrete shield building completely enclosing the
      containment vessel.

The drywell, suppression pool and the containment vessel are designed to condense the steam and contain fission product releases from the postulated design bases accident, i.e., the double ended rupture of the largest pipe in the reactor coolant system. The leak tight containment vessel prevents the release of fission products to the environment. The shield building provides direct radiation shielding to protect operating personnel and/or the public and also protects the containment vessel from weather and external missiles.

Containment temperature and pressure following an accident are limited by using the residual heat removal system to cool the suppression pool water and to provide a supply of water to the containment spray system.

The design of the complete containment system meets the requirements of Criterion 16.

For further discussion, see the following sections:

a.   General Plant Description <Section 1.2>

b.   Design of Containment Structure <Section 3.8>

c.   Containment Systems <Section 6.2>

d.   Accident Analyses <Section 15.0>

3.1.2.2.8   Criterion 17 - Electrical 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 within a few seconds following a loss-of-coolant accident to assure that the core cooling, containment integrity and other vital safety functions are maintained.

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

3.1.2.2.8.1   Evaluation Against Criterion 17

Onsite and offsite power systems each independently provide the total
power requirements for essential systems to perform the required
engineered safety feature function. The onsite power required to
operate protective systems equipment is supplied by one 100 percent
capacity diesel generator per division. The offsite power required to
operate engineered safety feature safety systems is supplied by two
independent sources from the 345 kV switchyard. Each offsite source is
capable of supplying the total power requirements for engineered safety
feature systems.

Three onsite independent battery systems provide 125 Vdc motive power
and control power for three redundant and independent power distribution
systems for engineered safety systems as well as control power for other
onsite power systems. The vital instrumentation is powered from a
minimum of two independent, nominal 120 volt ac vital buses. Each bus
is supplied by its own associated inverter. Each battery system
supplies power to its associated static inverter(s).

The engineered safety features power supply bus is connected to the
switchyard through one of two independent circuits. Electric power from
the transmission network to the switchyard is provided by a number of
independent lines, thus minimizing the likelihood of simultaneous
failure.

These systems are designed in accordance with IEEE Standard No. 308,
"IEEE Criteria for Class 1E Power Systems for Nuclear Power Generating
Stations."

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

For further discussion, see the following sections:

a.   General Plant Description <Section 1.2>

b.   Seismic Qualification of Category I Instrumentation and Electrical
     Equipment <Section 3.10>

c.   Environmental Qualification of Mechanical and Electrical Equipment
     <Section 3.11>

d.   Offsite Power System <Section 8.2>

e.   Onsite Power Systems <Section 8.3>

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
operational 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.

3.1.2.2.9.1   Evaluation Against Criterion 18

The engineered safety features power supply buses and associated diesel
generators are arranged for periodic testing of each system
independently. The testing procedure will simulate a loss of bus
voltage to start the diesel, bring it to operating condition and
automatically connect it to the bus. Full load testing of the diesel
generator can be performed by manually synchronizing to the normal
supply. These tests, performed periodically in accordance with the
Technical Specifications, will prove the operability of the emergency
power supply system under conditions as close to design as practical to
assess the continuity of the system and condition of the components.

The design of the stand-by power systems provides for inspection and
testability in accordance with the requirements of Criterion 18.

For further discussion, see the following sections:

a.   Onsite Power Systems <Section 8.3>

b.   Initial Test Program <Section 14.0>

c.   Technical Specifications <Section 16.0>

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 (5 rem TEDE for the design basis loss-of-coolant 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.

3.1.2.2.10.1   Evaluation Against Criterion 19

The control room contains the following equipment: transmission station panel, electrical recording panels and control panels which contain those instruments and controls necessary for operation of the plant functions, such as the reactor and its auxiliary systems, engineered safety features, turbine generator, steam and power conversion systems, and station electric 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 provided in the design. Adequate shielding is provided to maintain acceptable radiation levels in the control room, in the event of a design bases accident, for the duration of the accident.

The control room ventilation system has redundant control loops which serve both Unit 1 and Unit 2 control rooms. Each ventilation loop is provided with radiation, and smoke detectors with appropriate alarms and interlocks. Provision is made for the control room air to be recirculated through HEPA and charcoal filters in the emergency recirculation mode.

The control room will be continuously occupied by qualified operating personnel 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 to safely perform a hot shutdown and a subsequent cold shutdown of the reactors.

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 <Section 1.2>

b.   Control Complex Building Design <Section 3.8>

c.   Habitability <Section 6.4>

d.   Instrumentation and Control <Section 7.0>

e.   Shutdown from Outside Control Room <Section 7.4>

f.   Fire Protection <Section 9.5.1>

3.1.2.3   Group III, Protection and Reactivity Control Systems (Criteria 20-29)

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.

3.1.2.3.1.1   Evaluation Against Criterion 20

The reactor protection system is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored nuclear system variables exceed pre-established limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The reactor protection system includes the high-inertia motor generator power system, sensors, 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, reactor vessel steam dome high pressure, turbine stop valve closure, turbine control valve fast closure, main steam line isolation valve closure and reactor vessel low water level-3, will prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Additional scram trips are initiated by drywell high pressure, scram discharge volume high water level and reactor vessel high water level-8. Response by the reactor protection system is prompt and the total scram time is short. Control rod scram motion starts in about 200 milliseconds after the high flux setpoint is exceeded.

In addition to the reactor protection system which provides for automatic shutdown of the reactor to prevent fuel damage, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to

safety. Systems such as the emergency core cooling system are initiated automatically to limit the extent of fuel damage following a loss-of-coolant accident. Other systems automatically isolate the reactor vessel or the containment to prevent the release of significant amounts of radioactive materials from the fuel and the reactor coolant pressure boundary. The controls and instrumentation for the emergency core cooling systems and the isolation systems are also initiated automatically when monitored variables exceed preselected operational limits.

The design of these protection systems satisfies Criterion 20.

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Reactivity Control <Section 4.6>

c.   Overpressurization Protection <Section 5.2>

d.   Main Steam Line Isolation System <Section 5.4>

e.   Emergency Core Cooling System <Section 6.3>

f.   Reactor Protection System <Section 7.2>

g.   Containment and Reactor Vessel Isolation Control System
     <Section 7.3>

h.   Emergency Core Cooling Systems Instrumentation and Control
     <Section 7.3>

i.   Neutron Monitoring System <Section 7.6>

j.   Process Radiation Monitoring System <Section 11.5>

k.   Leak Detection System <Section 7.6>

l.   Accident Analyses <Section 15.0>

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.

3.1.2.3.2.1   Evaluation Against Criterion 21

Reactor protection system design provides assurance that, through
redundancy, each channel has sufficient reliability to fulfill the
single-failure criterion. No single component failure, intentional
bypass maintenance operation, calibration operation, or test to verify
operational availability will prevent the system from performing its
intended safety function. Additionally, the system design assures that
when a scram trip point is exceeded there is a high scram probability.
However, should a scram not occur, other monitored components will scram
the reactor if their trip points are exceeded. There is sufficient
electrical and physical separation between channels and between logics

monitoring the same variable to prevent environmental factors, electrical transients and physical events from impairing the ability of the system to respond correctly.

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

The reactor protection system initiates an automatic reactor shutdown if the monitored plant variables exceed pre-established limits. This system is arranged as two separately powered trip systems. Each trip system has two trip logics, with one needed to produce an automatic trip signal. The logic scheme is a 1-out-of-2 twice arrangement. The reactor protection system can be tested during reactor operation. Manual scram testing is performed by operating one of the two manual scram controls. This tests one trip system. The total test verifies the ability to de-energize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive 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 withdrawn one notch and then reinserted to the original position without significantly perturbing the nuclear system at most power levels. One control rod is tested at a time. Control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. More importantly, the hydraulic control unit scram accumulator and the scram discharge volume level are designed for continuous monitoring. Availability of these instruments is controlled by technical specifications or plant procedures/instructions.

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

Residual heat removal system testing can be performed during normal operation. Main system pumps can be evaluated by taking suction from the suppression pool and discharging through test lines back to the suppression pool. System design and operating procedures also permit testing of the discharge valves to the reactor recirculation loops. The low pressure coolant injection mode can be tested after a reactor shutdown.

Each active component of the emergency core cooling systems provided to operate in a design basis accident is designed to be operable for test purposes during normal operation of the nuclear system.

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Reactivity Control System <Section 4.6>

c.   Main Steam Line Isolation System <Section 5.4.5>

d.   Residual Heat Removal System <Section 5.4.7>

e.   Containment Systems <Section 6.2>

f.   Emergency Core Cooling Systems <Section 6.3>

g.   Reactor Protection System <Section 7.2>

h.   Engineered Safety Feature Systems <Section 7.3>

i.   Accident Analyses <Section 15.0>

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 bases. 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.

3.1.2.3.3.1   Evaluation Against Criterion 22

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

The reactor protection system is designed to permit maintenance and diagnostic work while the reactor is operating (excepting sensors - <Section 7.2.2.2>) without restricting the plant operation or hindering the output of their safety functions. The flexibility in design afforded the protection system allows operational system testing by the use of an independent trip channel for each trip logic input. When an essential monitored variable exceeds its scram trip point, it is sensed by at least two independent sensors in each trip system. Maintenance operation, calibration operation, or test unless manually bypassed, will result in a single channel trip. This leaves at least two trip channels per monitored variable capable of initiating a scram. Thus, the arrangement of two trip channels per trip system assures that a scram will occur as each monitored variable exceeds its scram setting.

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Main Steam Line Isolation System <Section 5.4.5>

c.   Residual Heat Removal System <Section 5.4.7>

d.   Emergency Core Cooling Systems <Section 6.3>

e.   Reactor Protection System <Section 7.2>

f.   Engineered Safety Feature Systems <Section 7.3>

g.   Accident Analyses <Section 15.0>

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.

3.1.2.3.4.1   Evaluation Against Criterion 23

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

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

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Emergency Core Cooling Systems <Section 6.3>

c.   Reactor Protection System <Section 7.2>

d.   Engineered Safety Feature Systems <Section 7.3>

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.

3.1.2.3.5.1   Evaluation Against Criterion 24

There is separation between the reactor protection system and the
process control systems. Sensors, trip channels and trip logics of the
reactor protection system are not used directly for automatic control of
process systems. Therefore, failure in the controls and instrumentation
of process systems cannot induce failure of any portion of the
protection system. High scram reliability is designed into the reactor
protection system and hydraulic control unit for the control rod drive.
The scram signal and mode of operation override all other signals.

The containment and reactor vessel isolation control systems are designed so that any one failure, maintenance operation, calibration operation, or test to verify operability will not impair the functional ability of the isolation control system.

These protection systems are separated from control systems as required in Criterion 24.

For further discussions, see the following sections:

a. Principal Design Criteria <Section 1.2>

b. Emergency Core Cooling Systems <Section 6.3>

c. Reactor Protection System <Section 7.2>

d. Engineered Safety Feature Systems <Section 7.3>

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.

3.1.2.3.6.1   Evaluation Against Criterion 25

The reactor protection system provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Any monitored variable which exceeds the scram setpoint will initiate an automatic

scram and not impair the remaining variables from being monitored, and if one channel fails, the remaining portions of the reactor protection system will function.

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

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Reactivity Control System <Section 4.3>

c.   Nuclear Design <Section 4.3>

d.   Thermal and Hydraulic Design <Section 4.4>

e.   Reactor Protection System <Section 7.2>

f.   Rod Control and Information System <Section 7.7>

g.   Accident Analyses <Section 15.0>

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.

3.1.2.3.7.1   Evaluation Against Criterion 26

Two independent reactivity control systems utilizing different design
principles are provided. The normal method of reactivity control
employs control rod assemblies which contain a neutron observing
material, e.g., boron carbide (B_4C) powder and hafnium metal. Positive
insertion of these control rods is provided by means of the control rod
drive hydraulic system. The control rods are capable of reliably
controlling reactivity changes during normal operation (e.g., power
changes, power shaping, xenon burnout, normal startup, and shutdown) via
operator-controlled insertions and withdrawals. The control rods are
also capable of maintaining the core within acceptable fuel design
limits during anticipated operational occurrences via the automatic
scram function. The unlikely occurrence of a limited number of stuck
rods during a scram 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 independent sources of scram energy (accumulator pressure and reactor vessel pressure) provide needed scram performance over the entire range of reactor pressure, i.e., from operating conditions to cold shutdown. The design of the control rod system includes appropriate margin for malfunctions such as stuck rods in the highly unlikely event that they do occur. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance, and simultaneously, low individual rod worths. The operating procedures to accomplish such patterns are supplemented by the rod pattern control system, 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 effect the type of reactivity changes necessary for planned, normal power changes (including xenon burnout). In the unlikely event that reactor flow is suddenly increased to its maximum value (pump runout), the core 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 3.1.2.3.8.1>.

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Reactivity Control System <Section 4.3>

c.   Engineered Safety Feature Systems <Section 7.3>

d.   Standby Liquid Control System Instrumentation and Control
     <Section 7.4>

e.   Rod Control and Information System <Section 7.7>

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.

3.1.2.3.8.1   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 standby
liquid control system. 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 postulated accident conditions is the control rod
system. Abnormalities are sensed, and, if protection system limits are
reached, corrective action is initiated through an automatic insertion
of control rods. High integrity of the protection system is achieved
through the combination of logic arrangement, actuator redundancy, power
supply redundancy, and physical separation. High reliability of reactor
scram is further achieved by separation of scram and manual control
circuitry, individual control units for each control rod and fail-safe
design features built into the rod drive system. Response by the
reactor protection system is prompt and the total scram time is short.

In the very unlikely 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 system (SLCS) will be actuated to
insert soluble boron into the reactor core. The SLCS has sufficient
capacity to ensure that the reactor can always be maintained
subcritical; hence, only decay heat will be generated by the core which
can be removed by the residual heat removal system, thereby ensuring
that the core will always be coolable.

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Reactivity Control System <Section 4.3>

c.   Nuclear Design <Section 4.3>

d.   Thermal and Hydraulic Design <Section 4.4>

e.   Reactor Protection System <Section 7.2>

f.   Rod Control and Information System <Section 7.7>

g.   Accident Analyses <Section 15.0>

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.

3.1.2.3.9.1   Evaluation Against Criterion 28

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

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

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod which prevents rapid rod ejection. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity to less than five feet per second. Normal rod movement is limited to six inch increments and the rod withdrawal rate is limited through the hydraulic valve to three inches per second.

The accident analyses (<Chapter 15>) evaluate postulated reactivity accidents as well as abnormal operational transients in detail. Analyses are included for rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. The results of these analyses indicate that none of the postulated reactivity transients or accidents result in damage to the reactor coolant pressure boundary. In addition, the integrity of the core, its support structures or other reactor pressure vessel internals are maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident 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 <Section 1.2>

b.   Control Rod Drive System <Section 3.9.4>, and <Section 4.6>

c.   Reactor Core Support Structures and Internals Mechanical Design <Section 3.9.5>, <Section 4.1>

d.   Reactivity Control System <Section 4.1>

e.   Nuclear Design <Section 4.3>

f.   Overpressurization Protection <Section 5.2>

g.   Reactor Vessel and Appurtenances <Section 5.3>

h.   Main Steam Line Flow Restrictors <Section 5.4.4>

i.   Main Steam Line Isolation System <Section 5.4.5>

j.   Reactor Protection System <Section 7.2>

k.   Process Computer System <Section 7.7>

l.   Accident Analyses <Section 15.0>

3.1.2.3.10   Criterion 29 - Protection Against Anticipated Operational Occurrences

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

3.1.2.3.10.1   Evaluation Against Criterion 29

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

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

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Reactor Protection System <Section 7.2>

c.   Accident Analyses <Section 15.0>

3.1.2.4   Group IV, Fluid Systems (Criteria 30-46)

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.

3.1.2.4.1.1   Evaluation Against Criterion 30

By using conservative design practices and a quality control program,
the pressure retaining components of the reactor coolant pressure
boundary are designed and fabricated to retain their integrity during
normal and postulated accident conditions. Accordingly, components
which comprise the reactor coolant pressure boundary are designed,
fabricated, erected, and tested in accordance with recognized industry
codes and standards listed in <Chapter 5> and <Table 3.2-1>. Further, a
quality assurance program, described in <Chapter 17>, is implemented to
assure conformance with the applicable codes and standards, and to
retain appropriate documented evidence verifying compliance. Because
the subject matter of this criterion deals with aspects of the reactor
coolant pressure boundary, 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 reactor coolant pressure boundary from potentially 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 concomitant with a loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the leak detection system provides protection from small leaks, the emergency core cooling system network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

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

For further discussion, see the following sections:

a.   Principal Design Criteria <Section 1.2>

b.   Overpressurization Protection <Section 5.2>

c.   Reactor Coolant Pressure Boundary Leakage Detection System
     <Section 5.2>

d.   Reactor Vessel and Appurtenances <Section 5.3>

e.   Other Reactor Coolant Subsystems and Components <Section 5.4>

f.   Leak Detection Systems <Section 7.6>, and <Section 7.7>

g.   Quality Assurance Program <Section 17.0>

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.

3.1.2.4.2.1   Evaluation Against Criterion 31

Brittle fracture control of pressure-retaining ferritic materials is
provided to ensure protection against non-ductile fracture. To minimize
the possibility of brittle fracture failure of the reactor pressure
vessel, the reactor pressure vessel is designed to meet the requirements
of the 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 a
ductile manner. The NDT temperature increases as a function of neutron
exposure at integrated neutron exposures greater than about 1×10^{17} nvt
with neutrons of energies in excess of 1 MeV.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident upon the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies and reactor coolant. Assuming plant operation at rated power, and availability of 100 percent for the plant lifetime, 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 assure that NDT temperature shifts are accounted for in reactor operation.

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

For further discussion, see the following sections:

a.   Integrity of Reactor Coolant Pressure Boundary <Section 5.2>

b.   Reactor Vessel <Section 5.3>

c.   Technical Specifications

3.1.2.4.3   Criterion 32 - Inspection of Reactor Coolant Pressure
              Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important

areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

3.1.2.4.3.1   Evaluation Against Criterion 32

The reactor pressure vessel design and engineering effort include provisions for inservice inspection. Removable plugs in the biological shield and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system safety/relief valves, recirculation system and on the main steam and feedwater systems extending out to and including the first isolation valve outside containment. Inspection of the reactor coolant pressure boundary is in accordance with the ASME Boiler & Pressure Vessel Code Section XI. <Section 5.2.4> defines the Inservice Inspection plan, access provisions and areas of restricted access.

Vessel material surveillance samples will be located within the reactor pressure vessel. The program will include specimens of the base metal, weld metal and heat affected zone metal.

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

For further discussion, see the following sections:

a.   Mechanical Systems and Components <Section 3.9>

b.   Integrity of Reactor Coolant Pressure Boundary <Section 5.2>

c.   Reactor Vessel <Section 5.3>

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.

3.1.2.4.4.1   Response to Criterion 33

The plant is designed to provide ample reactor coolant makeup for
protection against small leaks in the reactor coolant pressure boundary
for anticipated operational occurrences and postulated accident
conditions.

Means are provided for detecting reactor coolant leakage as described in
the evaluation of Criterion 30. The total leakage rate limit is
established so that, in the absence of normal ac power coincident with
loss of feedwater supply, makeup capabilities are provided by the RCIC
system. Emergency core cooling system makeup capabilities for the
entire spectrum of liquid line breaks are described in response to
Criterion 35.

The design of these systems meets the requirements of Criterion 33.

For further discussion, see the following sections:

a.   Detection of Leakage Through Reactor Coolant Pressure Boundary
      <Section 5.2.5>

b.   Feedwater System <Section 10.4.7>

c.   Reactor Core Isolation Cooling System <Section 5.4.6>

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 offset 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.

3.1.2.4.5.1   Evaluation Against Criterion 34

The residual heat removal (RHR) system provides the means to:

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

b.   Remove decay heat and residual heat if the normal heat sink is
      unavailable.

The major equipment of the RHR system consists of heat exchangers, main system pumps, and emergency service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation.

Two independent loops are located in separate protected areas.

Both normal ac power and the auxiliary onsite power systems provide adequate power to operate all the auxiliary loads necessary for plant operation. The power sources for the plant 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.

Full capacity standby diesel generators are provided to supply a source of electrical power which is self-contained within the plant and is not dependent on external sources of supply. The standby generators produce ac power at a voltage and frequency compatible with the normal bus requirements for RHR and other essential equipment within the plant. Each of two diesel generators has sufficient capacity to start and carry the essential loads.

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

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

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

Provisions for leak detection are described in the evaluation of Criterion 30 <Section 3.1.2.4.1.1>.

The residual heat removal system is designed to remove residual heat from the reactor core to assure fuel and reactor coolant pressure boundary design limits are not exceeded. Redundant reactor coolant recirculation paths are available between the vessel and RHR system. Redundant onsite electric power systems are provided. The design of the residual heat removal system, including its power supplies, meets the requirements of Criterion 34.

For further discussion, see the following sections:

a.   Residual Heat Removal System <Section 5.4.7>

b.   Offsite Power System <Section 8.2>

c.   Onsite Power Systems <Section 8.3>

d.   Water Systems <Section 9.2>

e.   Accident Analyses <Section 15.0>

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.

3.1.2.4.6.1   Evaluation Against Criterion 35

The emergency core cooling systems (ECCS) consist of the following: (a) high pressure core spray system (HPCS), (b) automatic depressurization system (ADS), (c) low pressure core spray system (LPCS), and (d) low pressure coolant injection (LPCI) which is an operating mode of the RHR system. The emergency core cooling systems are designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the reactor coolant pressure boundary including a complete and sudden circumferential rupture of the pipe connected to the reactor vessel which represents the greatest challenge to fuel integrity.

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

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

The LPCS system consists of: a centrifugal pump that can be powered by normal power or the standby ac power system; a spray sparger in the reactor vessel above the core (separate from the HPCS sparger); piping and valves to convey water from the suppression pool to the sparger; and associated controls and instrumentation. In case of low water level in the reactor vessel or high pressure in the drywell, the LPCS system automatically sprays water onto the top of the fuel assemblies in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature.

The LPCI system starts from the same signals which initiate the LPCS system and operates independently to achieve the same objective by flooding the reactor vessel. LPCI is an operating mode of the RHR system. In case of low water level in the reactor or high pressure in the drywell, LPCI operation 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 automatic depressurization system has operated to lower the reactor vessel pressure.

ECCS performance results for the entire spectrum of liquid line breaks are discussed in <Section 6.3>. Peak cladding temperatures are well below the NRC acceptability limit.

Also provided in <Section 6.3> is an analysis to show that the ECCS conforms to <10 CFR 50, Appendix K>. This analysis shows compliance with the final acceptance criteria with the following results:

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

b.   The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1 percent 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 emergency core cooling systems provided are adequate to prevent fuel and clad damage which could interfere with effective core cooling and to limit clad metal-water reaction to a negligible amount. The design of the emergency core cooling systems, including their power supply, meet the requirements of Criterion 35.

For further discussion, see the following sections:

a.   Residual Heat Removal System <Section 5.4.7>

b.   Emergency Core Cooling Systems <Section 6.3>

c.   Onsite Power Systems <Section 8.3>

d.   Water Systems <Section 9.2>

e.   Accident Analysis <Section 15.0>

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.

3.1.2.4.7.1   Evaluation Against Criterion 36

The Emergency Core Cooling Systems discussed in Criterion 35 include
inservice inspection considerations. Reactor vessel piping nozzles and
spray spargers within the vessel are accessible for inspection during
refueling outages. Removable plugs in the biological shield and/or
panels in the insulation provide access for examination of nozzles.
Removable insulation is provided on the Emergency Core Cooling Systems
piping out to and including the first isolation valve outside the
drywell. Inspection of the Emergency Core Cooling Systems is in
accordance with the intent of Section XI of the ASME Code.
<Section 5.2.4> defines the Inservice Inspection Plan, access provisions
and areas of restricted access.

During plant operations, the pumps, valves, piping, instrumentation,
wiring, and other components outside the drywell can be visually
inspected at any time. Components inside the drywell can be inspected
when the drywell is open for access. When the reactor vessel is open,
for refueling or other purposes, the spargers and other internals can be
inspected. Portions of the ECCS which are part of the reactor coolant
pressure boundary are designed to specifications for inservice
inspection to detect defects which might affect the cooling performance.
Particular attention will be given to the reactor vessel nozzles, core
spray and feedwater spargers. The design of the reactor vessel and

internals for inservice inspection, and the plant testing and inspection program ensures that the requirements of Criterion 36 will be met.

For further discussion, see the following sections:

a.   Inservice Inspection and Testing of Reactor Coolant Pressure Boundary <Section 5.2.4>

b.   Reactor Vessel <Section 5.3>

c.   Emergency Core Cooling Systems <Section 6.3>

d.   Inservice Inspection of Class 2 and 3 Components <Section 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.

3.1.2.4.8.1   Evaluation Against Criterion 37

The emergency core cooling system consists of the high pressure core spray (HPCS) system, automatic depressurization system (ADS), low pressure coolant injection (LPCI) mode of the RHR system, and low pressure core spray (LPCS) system. Each of these systems is provided with sufficient test connections and isolation valves to permit

appropriate periodic pressure testing to assure the structural and leaktight integrity of its components.

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

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

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

The redundancy and capability of the electrical power systems for the ECCS are represented in the evaluation of Criterion 34.

For further discussion, see the following sections:

a.   Overpressurization Protection <Section 5.2.2>

b.   ECCS Inspection and Testing <Section 6.3.4>

c.   Technical Specifications <Section 16.0>

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, consistently 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 electric 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.

3.1.2.4.9.1   Evaluation Against Criterion 38

The containment heat removal function is accomplished by the residual heat removal system (RHR). Following a loss-of-coolant accident, one or both of the following operating modes of the RHR system would be initiated:

a.   Containment Spray - condenses steam within the containment.

b.   Suppression Pool Cooling - limits the temperature within the containment by removing heat from the suppression pool water by means 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 residual heat removal system are presented in the evaluation against Criterion 34.

For further discussion, see the following sections:

a.   Residual Heat Removal System <Section 5.4.7>

b.   Containment Systems <Section 6.2>

c.   Water Systems <Section 9.2>

d.   Accident Analyses <Section 15.0>

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
sumps, spray nozzles and piping to assure the integrity and capability
of the system.

3.1.2.4.10.1   Evaluation Against Criterion 39

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

The spray rings and nozzles of the RHR containment spray system are
located under the reactor building dome. An air connection is provided
on the supply piping to the spray rings for testing the spray nozzles.
The containment spray system spray nozzles will be verified unobstructed
following maintenance which could result in nozzle blockage. The test
may be performed using an inspection of the nozzle or an air or smoke
flow test.

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

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

For further discussion, see the following sections:

a.   Residual Heat Removal Systems <Section 5.4.7>

b.   Containment Systems <Section 6.2>

c.   Emergency Core Cooling Systems <Section 6.3>

d.   Water Systems <Section 9.2>

3.1.2.4.11   Criterion 40 - Testing of Containment Heat Removal
                System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system and (3) the operability and performance 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.

3.1.2.4.11.1   Evaluation Against Criterion 40

The containment heat removal function is accomplished by a cooling mode of the residual heat removal system (RHR).

The RHR System is provided with sufficient test connections and isolation valves to permit periodic pressure and flow rate testing. The pumps and valves of the RHR will be operated periodically to verify operability (the cooling mode is not automatically initiated).

The redundancy and capability of the offsite and onsite electrical power supplies for the RHR system are presented in the response to Criterion 34. 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.   Residual Heat Removal System <Section 5.4.7>

b.   ECCS Instrumentation and Control <Section 7.3>

c.   Technical Specifications <Section 16.0>

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 quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system

operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

3.1.2.4.12.1   Evaluation Against Criterion 41

A combustible gas control system is provided to control the concentration of combustible gases in the containment. The system has redundant equipment and power supplies. The containment atmosphere is continuously monitored for combustible gas concentration following a LOCA and the control system can be manually operated as required.

The annulus between the containment vessel and the shield building is maintained at a slight negative pressure to ensure that any leakage from the containment or through the shield building is into the annulus. In the event of an accident, noncondensible gases are contained within the containment vessel. Containment leakage is collected in the annulus; exhaust from the annulus volume is monitored and released in a controlled manner through HEPA and charcoal filters to minimize the release of radioactivity to the environment. The annulus exhaust gas treatment system is provided with redundant equipment and power supplies and is in continuous operation during normal plant operations. The exhaust air from the annulus is monitored for radioactivity level.

The above described systems meet the requirements of Criterion 41.

For further discussion, see the following sections:

a.   General Plant Description <Section 1.2>

b.   Containment Functional Design <Section 6.2.1>

c.   Secondary Containment Functional Design <Section 6.2.3>

d.   Combustible Gas Control in Containment <Section 6.2.5>

e.   Fission Product Removal and Control Systems <Section 6.5>

f.   Reactor Building Ventilation Systems <Section 9.4.6>

g.   Process and Effluent Monitoring <Section 11.5>

h.   Accident Analyses <Section 15.0>

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
system.

3.1.2.4.13.1   Evaluation Against Criterion 42

With the exception of ductwork and fans located in the drywell, all
equipment of the containment atmosphere cleanup systems, described in
the response to Criterion 41, can be inspected during normal plant
operation.

The annulus exhaust gas treatment system is operated continuously during
plant operation and can be monitored for satisfactory operation.
Components of the combustible gas control system will be periodically
inspected and tested to ensure continued availability. Redundant
components and power supplies are provided for these systems.

Inspection and testing of the containment spray system is described in
the response to Criterion 39.

The design of these systems meets the requirements of Criterion 42.

For further discussion, see the following sections:

a.   General Plant Description <Section 1.2>

b.   Containment Functional Design <Section 6.2.1>

c.   Secondary Containment Functional Design <Section 6.2.3>

d.   Combustible Gas Control in Containment <Section 6.2.5>

e.   Fission Product Removal and Control Systems <Section 6.5>

f.   Reactor Building Ventilation Systems <Section 9.4.6>

g.   Process and Effluent Monitoring <Section 11.5>

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 systems into operation, including operation of applicable portions
of the protection system, the transfer between normal and emergency
power sources and the operation of associated systems.

3.1.2.4.14.1   Evaluation Against Criterion 43

This requirement is addressed in response to Criterion 42.

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 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.

3.1.2.4.15.1   Evaluation Against Criterion 44

The systems provided to transfer heat from items of safety-related
importance to Lake Erie consist of the emergency service water system
and the emergency closed cooling system.

Redundancy and reliability of the cooling water supply from the lake is
provided by installing two independent Seismic Category I tunnels.
These tunnels are, in part, the intake and discharge tunnels for the
service water system built to Seismic Category I design extended to the
emergency service water intake structure. The emergency service water
structure is a Category I structure with 100 percent redundancy for all
mechanical equipment. The systems are operable either from offsite
power or from onsite emergency diesel generator power.

Redundancy, isolation capability and separation is provided such that no single failure will prevent safe shutdown of the unit. The lake itself is a reliable heat sink with an extremely low probability of not being available at all times. Reliability of the cooling water supply to the ESW intake structure is maintained, even though the physical redundancy of the ESW cooling water supplies may not be available during periods of elevated lake temperature when the sluice gate seals are inflated and the automatic opening feature is disabled. In compliance with <Regulatory Guide 1.27>, it has been demonstrated that there is an extremely low probability of normal intake failure during the time that the automatic opening feature of the sluice gates is disabled and the alternate intake tunnel is unavailable.

The design of these systems thus meets the requirements of Criterion 44.

For further discussion, see the following sections:

a.   General Plant Description <Section 1.2>

b.   Design of Category I Structures <Section 3.8>

c.   Water Systems <Section 9.2>

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.

3.1.2.4.16.1   Evaluation Against Criterion 45

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.   General Plant Description <Section 1.2>

b.   Water Systems <Section 9.2>

c.   Initial Tests and Operation <Section 14.0>

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 the
performance of the active components of the system and (3) the
operability of the system as a whole and, under conditions as close to
design as practical, the performance of the full operational sequence
that brings the system into operation for reactor shutdown and for
loss-of-coolant accidents, including operation of applicable portions of
the protection system and the transfer between normal and emergency
power sources.

3.1.2.4.17.1   Evaluation Against Criterion 46

Redundancy and isolation are provided to allow periodic pressure and
functional testing of the system as a whole, including the functional
sequence that initiates system operation. This also includes transfer
between the normal offsite power supply and the onsite emergency diesel
generator power supply. At least one of the redundant systems is in
service during normal plant operations.

The system design thus meets the requirements of Criterion 46.

For further discussion, see the following sections:

a.   General Plant Description <Section 1.2>

b.   Water Systems <Section 9.2>

c.   Initial Tests and Operation <Section 14.0>

d.   Technical Specifications <Section 16.0>

3.1.2.5   Group V, Reactor Containment (Criteria 50-57)

3.1.2.5.1   Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings,
penetrations and the containment heat removal system shall be designed
so that the containment structure and its internal compartments can
accommodate, without exceeding the design leakage rate and, with
sufficient margin, the calculated pressure and temperature conditions
resulting from any 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.

3.1.2.5.1.1   Evaluation Against Criterion 50

The design of the containment is based on natural phenomena postulated
to occur at the site and the design basis accident which assumes the

instantaneous circumferential rupture of a main steam line upstream of the main steam line flow restrictor. These conditions are coupled with a loss of offsite power and partial loss of the redundant engineered safety features systems (minimum engineered safety features).

The maximum pressure and temperature reached in the drywell and containment during this worst case accident are shown in <Section 6.2> and <Section 15.6> to be well below the design pressure and temperature of the structures. This provides an adequate margin for uncertainties in potential energy sources.

The design of the containment system thus meets the requirements of Criterion 50.

For further discussion, see the following sections:

a.   Classification of Structures, Components and Systems <Section 3.2>

b.   Wind and Tornado Design Criteria <Section 3.3>

c.   Missile Protection Criteria <Section 3.5>

d.   Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping <Section 3.6>

e.   Seismic Design <Section 3.7>

f.   Design of Containment Structure <Section 3.8.2>

g.   Containment Functional Design <Section 6.2.1>

h.   Containment Heat Removal System <Section 6.2.2>

i.   Accident Analyses <Section 15.0>

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.

3.1.2.5.2.1   Evaluation Against Criterion 51

This criterion applies to components of the containment system that
provide a pressure boundary in the performance of the containment
function under operating, maintenance, testing, and postulated accident
conditions. (<Section 3.11> provides the service temperatures for these
determinations.)

Determination of nonbrittle behavior (fracture toughness) of pressure
boundary materials is based on fracture toughness data and metallurgical
characterization of these materials developed from a review of how these
materials were fabricated and what thermal history they experienced
during fabrication. The metallurgical characterization of these
materials, when correlated with data presented in the ASME Code,
provided the technical basis for concluding that the reactor containment
pressure boundary will behave in a nonbrittle manner and the
requirements of this criterion are satisfied.

The preoperational test program and the ongoing quality assurance program will ensure the integrity of the containment.

The containment design thus meets the requirements of Criterion 51.

For further discussion, see the following sections:

a.   Steel Containment <Section 3.8.2>

b.   Preoperational Test Program <Section 14.2>

c.   Quality Assurance <Section 17.0>

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.

3.1.2.5.3.1   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 Option B of <10 CFR 50, Appendix J>.

The testing provisions provided and the test program meet the requirements of Criterion 52.

For further discussion, see the following sections:

a.   Containment Leakage Testing <Section 6.2.6>

b.   Technical Specifications <Section 16.0>

3.1.2.5.4   Criterion 53 - Provisions for Containment Testing and
               Inspection

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

3.1.2.5.4.1   Evaluation Against Criterion 53

There are special provisions for conducting individual leakage rate
tests on applicable penetrations. Penetrations will be visually
inspected and pressure tested for leak tightness at periodic intervals.

The provisions made for penetration testing meet the requirements of
Criterion 53.

For further discussion, see the following sections:

a.   Containment Isolation System <Section 6.2.4>

b.   Containment Leakage Testing <Section 6.2.6>

c.   Technical Specifications <Section 16.0>

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.

3.1.2.5.5.1   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 leakage tests to be
performed. The engineered safety features actuation system test
circuitry provides the means for testing isolation valve operability.

Conformance with Criterion 54 is further discussed in
<Section 3.1.2.5.6> (Criterion 55), <Section 3.1.2.5.7> (Criterion 56)
and <Section 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:

a.   One locked closed isolation valve inside and one locked closed
     isolation valve outside containment, or

b. One automatic isolation valve inside and one locked closed
    isolation valve outside containment, or

c. 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

d. 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 the 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.

3.1.2.5.6.1   Evaluation Against Criterion 55

The reactor coolant pressure boundary (as defined in <10 CFR 50.2>)
consists of the reactor pressure vessel, pressure retaining
appurtenances attached to the vessel, and valves and pipes which extend
from the reactor pressure vessel up to and including the outermost
isolation valve. The lines of the reactor coolant pressure boundary
which penetrate the containment have suitable isolation valves

capable of isolating the containment thereby precluding any significant release of radioactivity. Similarly for lines which do not penetrate the containment but which form a portion of the reactor coolant pressure boundary, the design ensures that isolation from the reactor coolant pressure boundary can be achieved.

The designs of the isolation systems detailed in sections listed below meet the requirements of Criterion 55.

a.   Integrity of Reactor Coolant Pressure Boundary <Section 5.2>

b.   Containment Isolation System <Section 6.2.4>

c.   Instrumentation and Controls <Section 7.0>

d.   Accident Analyses <Section 15.0>

e.   Technical Specifications <Section 16.0>

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:

a.   One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

b.   One automatic isolation valve inside and one locked closed isolation valve outside containment, or

c.   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

d.   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 the 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.

3.1.2.5.7.1   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 and the other outside containment.

<Section 6.2.4> discusses containment isolation. In general, the four
items listed in Criterion 56 are followed. In certain cases, other
acceptable defined bases for isolation which deviate from Items (1)
through (4) of this criterion are followed. Due to the number of
containment isolation valves, <Section 6.2.4> and the associated tables
and figures are referenced to show how Criterion 56 is met. The
isolation valving is shown schematically <Figure 6.2-60> and is
summarized, with valve actuation, in <Table 6.2-32>. Simple check
valves are not used as automatic isolation valves outside the
containment. Instrument lines satisfy the criteria set by <Regulatory
Guide 1.11>.

The manner in which the containment isolation system meets this requirement is detailed in the sections listed below:

a.   Containment Isolation System <Section 6.2.4>

b.   Instrumentation and Controls <Section 7.0>

c.   Accident Analyses <Section 15.0>

d.   Technical Specifications <Section 16.0>

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 the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

3.1.2.5.8.1   Evaluation Against Criterion 57

Each line that penetrates the containment and is not connected to the containment atmosphere and is not part of the reactor coolant pressure boundary has at least one isolation valve located outside the containment near the penetration.

Details demonstrating conformance with Criterion 57 are provided in <Section 6.2.4>.

3.1.2.6   Group VI, Fuel and Radioactivity Control (Criteria 60-64)

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.

3.1.2.6.1.1   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 and within the limits of <10 CFR 20>.
The plant is also designed with provisions to prevent radioactivity
releases during accidents from exceeding the limits of <10 CFR 100>
(superseded by <10 CFR 50.67> for future design basis analyses).

The principal gaseous effluents from the plant during normal operation
are the noncondensible gases from the air ejectors. These gases are
exhausted through a holdup system and a low temperature offgas treatment
system including charcoal absorbers. The effluent from this system is
continuously monitored and controlled, and the system will be shut down
and isolated in the event of abnormally high radiation levels.

Ventilation air from the various plant areas is exhausted through HEPA
and charcoal filters, and is continuously monitored and controlled.

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, and 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.   General Plant Description <Section 1.2>

b.   Detection of Leakage through Reactor Coolant Pressure Boundary
     <Section 5.2.5>

c.   Containment Functional Design <Section 6.2.1>

d.   Air Conditioning, Heating, Cooling and Ventilating Systems
     <Section 9.4>

e.   Liquid Waste Management Systems <Section 11.2>

f.   Gaseous Waste Management Systems <Section 11.3>

g.   Solid Waste Management System <Section 11.4>

h. Process and Effluent Radiological Monitoring and Sampling Systems
    <Section 11.5>

i. Accident Analyses <Section 15.0>

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 a residual heat removal
capability having reliability and testability that reflects the
importance to safety of decay heat and other residual heat removal, and
(5) to prevent significant reduction in fuel storage coolant inventory
under accident conditions.

3.1.2.6.2.1   Evaluation Against Criterion 61

3.1.2.6.2.1.1   New Fuel Storage

New fuel is placed in dry storage in the new fuel storage vault which is
located inside the fuel handling building. The storage vault within the
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.

3.1.2.6.2.1.2   Spent Fuel Handling and Storage

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

Reliable decay heat removal is provided by the closed loop FPCC system.
It consists of circulating pumps, heat exchangers, filter
demineralizers, and the required piping, valves and instrumentation.
The pool water is circulated through the system, suction is taken from
the pool, flow passes through the heat exchanger and filter
demineralizers, and it is discharged through diffusers below the pool
surface. Pool water temperature is maintained below 130°F when removing
the maximum normal heat load from the pool with the closed cooling water
temperature at its maximum. If it appears that the pool temperature
will exceed 150°F, the FPCC system can be connected to the RHR system.
This increases the cooling capacity of the FPCC system.

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

No tests are required because at least one pump and heat exchanger are
normally operating while fuel is stored in the pool. A heatup rate
evaluation shall be performed if it is necessary to secure the system
while fuel is stored in the pool. Routine visual inspection of the
system components, instrumentation and trouble alarms are adequate to
verify system operability.

3.1.2.6.2.1.3   Radioactive Waste Systems

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

Liquid radwastes are classified, contained and treated as high purity/low conductivity, low purity/high conductivity, chemical, detergent, filter sludges, or spent resin wastes. Processing includes filtration, ion exchange, evaporation, phase separation, analysis, and dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. Wet and dry solid wastes are packaged in accordance with <10 CFR 71>. Gaseous radwastes are monitored, processed, recorded, and controlled so that radiation doses to persons outside the Radiologically Restricted Area are below those allowed by applicable regulations.

Accessible portions of the fuel handling and radwaste buildings have sufficient shielding to maintain dose rates within the limits set forth in <10 CFR 20> and <10 CFR 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 assure operability. Performance is monitored by radiation monitors during operation.

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

For further discussion, see the following sections:

a.   Residual Heat Removal System <Section 5.4.7>

b.   Containment <Section 6.2>

c.   New Fuel Storage <Section 9.1.1>

d.   Spent Fuel Storage <Section 9.1.2>

e.   Fuel Pool Cooling and Cleanup System <Section 9.1.3>

f.   Heating, Ventilating and Air Conditioning System <Section 9.4>

g.   Radioactive Waste Management <Section 11.0>

h.   Radiation Protection <Section 12.0>

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.

3.1.2.6.3.1   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 storage racks. There is sufficient spacing between
the assemblies to assure that the array when fully loaded is
substantially subcritical. Fuel elements are limited by rack design to
only top loading and fuel assembly positions. The new and spent fuel
racks are Seismic Category I structures.

New fuel is placed in dry storage in the top-loaded new fuel storage vault. This vault contains a drain to prevent the accumulation of water. The new fuel storage vault racks (located inside the fuel handling building) are designed to prevent an accidental critical array, even in the event the vault becomes flooded or subjected to seismic loadings. The center-to-center new fuel assembly spacing limits the effective multiplication factor of the array to ≤0.95 if the new fuel is flooded.

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

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

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

For further discussion, see <Section 9.1>.

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.

3.1.2.6.4.1   Evaluation Against Criterion 63

Appropriate systems have been provided to meet the requirements of this
criterion. A malfunction of the fuel pool cooling and cleanup system
which could result in loss of residual heat removal capability and
excessive radiation levels is alarmed in the control room. Alarmed
conditions include low fuel pool cooling water pump discharge pressure
and high/low level in the fuel storage pool. System temperature is also
continuously monitored and alarmed in the control room. The containment
and fuel handling building ventilation radiation monitoring systems
detect abnormal amounts of radioactivity and initiate appropriate action
to control the release of radioactive material to the environs.

Area radiation and tank sump levels are monitored and alarmed to give
indication of conditions which 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.   Fuel Storage and Handling <Section 9.1>

b.   Liquid Radwaste Management Systems <Section 11.2>

c.   Gaseous Radwaste Management Systems <Section 11.3>

d.   Solid Radwaste Management System <Section 11.4>

e.   Process and Effluent Radiological Monitoring and Sampling Systems
      <Section 11.5>

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.

3.1.2.6.5.1   Evaluation Against Criterion 64

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

a.   Gaseous releases from the plant vents

b.   Liquid discharge to the circulating water tunnel

c.   Auxiliary Building ventilation

d.   Radwaste Building ventilation

e.   Fuel Handling Building ventilation

In addition, the drywell and containment atmosphere is monitored and
onsite and offsite monitors are provided.

Periodic reports of operation are submitted to the NRC. These reports include specific information on the quantities of the principal radionuclides released to the environs.

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

a.   Reactor Coolant Pressure Boundary Leakage Detection System <Section 5.2.5>

b.   Radioactive Waste Systems <Section 11.0>

c.   Area Radiation and Airborne Radioactivity Monitoring Instrumentation <Section 12.3.4>

3.2   CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

Certain structures, components and systems of the nuclear plant are
considered important to safety because they perform safety actions
required to avoid or mitigate the consequences of abnormal operational
transients or accidents. The purpose of this section is to classify
structures, components and systems according to the importance of the
safety function they perform. In addition, design requirements are
placed upon such equipment to assure the proper performance of safety
actions, when required.

3.2.1   SEISMIC CLASSIFICATION

Plant structures, systems and components important to safety are
designed to withstand the effects of a Safe Shutdown Earthquake (SSE)
and remain functional if they are necessary to assure:

a.   The integrity of the reactor coolant pressure boundary,

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

c.   The capability to prevent or mitigate the consequences of accidents
     that could result in potential offsite exposures comparable to the
     guideline exposures of <10 CFR 100> or <10 CFR 50.67> (future
     revisions to design basis analyses that compare consequences to
     10 CFR 100 will be updated to <10 CFR 50.67>).

Plant structures, systems 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>.

Structures, components, equipment, and systems designated as Safety Class 1, Safety Class 2 or Safety Class 3 (see <Section 3.2.3> for a discussion of safety classes) are classified as Seismic Category I except for (1) those noted in <Table 3.2-1> and (2) those portions of

the radioactive waste treatment handling and disposal systems whose postulated simultaneous failure would not result in conservatively calculated offsite exposures comparable to the guideline exposures of <10 CFR 100>.

All Seismic Category I structures, systems and components are analyzed under the loading conditions of the SSE and 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, and yield margins of safety appropriate for each earthquake. The margin of safety provided for safety class structures, components, equipment, and systems for the SSE is sufficiently large to assure that their design functions are not jeopardized.

During construction and plant operation, all nonsafety structures, systems or components which fall under <Regulatory Guide 1.29>, Paragraph c.2 are identified through a seismic clearance and anti-fall down inspection program. Such structures or components are then analyzed for seismic loads and, when necessary, additional supports are added or existing supports are modified to ensure structural integrity during a seismic event.

The seismic classification indicated in <Table 3.2-1> meets the requirements of <Regulatory Guide 1.29>, with exceptions as stated in Notes 19 and 24.

3.2.1.1   Unit 2 Auxiliary Building and Unit 2 Containment and Shield Building

The Unit 2 Auxiliary Building and Unit 2 Shield Building are Seismic Category I structures that provide structural design functions in support of Unit 1 operations, as indicated in <Table 3.2-1>. The structures must maintain structural integrity during postulated events

including seismic, tornado, and external missile, and provide barriers
to prevent propagation of flood or fire. The Unit 2 Containment vessel
is classified as Seismic II/I for preventing structural failure of the
Unit 2 Shield Building.

3.2.2   SYSTEM QUALITY GROUP CLASSIFICATIONS

System quality group classifications, as defined in <Regulatory
Guide 1.26>, have been determined for each water, steam or radioactive
waste containing component of those applicable fluid systems relied upon
to:

a.   prevent or mitigate the consequences of accidents and malfunctions
      originating within the reactor coolant pressure boundary,

b.   permit shutdown of the reactor and maintain it in a safe shutdown
      condition or

c.   contain radioactive material.

A tabulation of quality group classification for each component so
defined is shown in <Table 3.2-1> under the heading, "Quality Group
Classification." <Figure 3.2-1> is a diagram that depicts the relative
locations of these components along with their quality group
classification.

System Quality Group Classifications and design and fabrication
requirements as indicated in <Table 3.2-1> and <Table 3.2-2> meet the
guidelines of <Regulatory Guide 1.26>.

3.2.3   SYSTEM SAFETY CLASSIFICATIONS

Structures, systems and components are classified as Safety Class 1,
Safety Class 2, Safety Class 3, or nonsafety class (NSC) in accordance

with the importance to nuclear safety. Equipment is assigned a specific safety class, recognizing that components within a system may be of differing safety importance. A single system may thus have components in more than one safety class. Supports are appropriate for the component supported, as defined by the ASME Code Section III.

The safety classes are defined in this section and examples of their broad application are given. Because of specific design considerations, these general definitions are subject to interpretation and exceptions. <Table 3.2-1> provides a summary of the safety classes for the principal structures, systems and components of the plant.

Design requirements for components of safety classes are also delineated in this section. Where possible, reference is made to accepted industry codes and standards which define design requirements commensurate with the safety function(s) to be performed. In cases where industry codes and standards have no specific design requirements, the locations of the appropriate subsections that summarize the requirements to be implemented in the design are indicated.

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 reactor coolant pressure boundary or core support structure whose failure could cause a loss of reactor coolant. Structures, systems and components in Safety Class 1 are listed in <Table 3.2-1>.

3.2.3.1.2   Design Requirements for Safety Class 1

<Table 3.2-3> lists industry code requirements for Safety Class 1 mechanical components and structural supports and correlates these requirements with design condition categories.

3.2.3.2   Safety Class 2

3.2.3.2.1   Definition of Safety Class 2

Safety Class 2 applies to those structures, systems and components, other than cooling water systems, that are not safety Class 1 but are necessary to accomplish the following safety functions:

a.   Insert negative reactivity to shutdown the reactor.

b.   Prevent rapid insertion of positive reactivity.

c.   Maintain core geometry appropriate to all plant process conditions.

d.   Provide emergency core cooling.

e.   Provide and maintain containment.

f.   Remove residual heat from the reactor and reactor core.

g.   Store spent fuel.

Safety Class 2 includes the following:

a.   Reactor protection system.

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

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

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

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

f. Emergency core cooling systems.

g. Containment vessel.

h. Shield building and annulus gas treatment system.

i. Postaccident containment heat removal systems.

j. Containment combustible gas control system.

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

l. At least one of the systems which removes decay heat when the reactor is pressurized, and the system to remove decay heat when the reactor is not pressurized.

m. Spent fuel storage racks and spent fuel pool.

n. Electrical and instrument systems necessary for operation of the above.

o. Pipes having a nominal size of 3/4 inch or smaller, that are part of the reactor coolant pressure boundary.

Structures, systems and components in Safety Class 2 are listed in
<Table 3.2-1>.

3.2.3.2.2   Design Requirements for Safety Class 2

In applying industry codes to Safety Class 2 structures, systems and
components, the codes (except for mechanical equipment) do not fit
neatly and automatically into the safety class and design condition
designations developed in this section. Therefore, mechanical and
structural categories will be treated separately from electrical.

Note that temporary isolation devices associated with Reactor Pressure
Vessel Water Inventory Control per Technical Specification 3.5.2 may be
non-safety class and non-seismic but must be able to prevent an RPV
draining event. Reference Technical Specification 3.5.2 Bases for
additional detail. (see <Table 9.1-5>)

<Table 3.2-4> lists the code requirements for mechanical and structural
systems of Safety Class 2 designation, and correlates these requirements
with conditions of design.

Code requirements for protection and Class 1E electrical systems of
Safety Class 2 (as defined in IEEE-279 and IEEE-308) are shown in
<Table 3.2-6>.

3.2.3.3   Safety Class 3

3.2.3.3.1   Definition of Safety Class 3

Safety Class 3 applies to those structures, systems and components that
are not Safety Class 1 or Safety Class 2, but

a.   whose function is to process radioactive wastes and whose failure
      would result in release to the environment of gas, liquid or solids

resulting in a single event dose greater than the limits specified
in <10 CFR 100> to a person at the site boundary,

b.   which provide or support any safety system function, or

c.   which remove decay heat from the spent fuel.

Safety Class 3 includes the following:

a.   Portions of the gaseous waste disposal system.

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

c.   Cooling water systems required for the purpose of:

      1.   Removal of decay heat from the reactor or spent fuel pool.

      2.   Emergency core cooling.

      3.   Postaccident containment heat removal.

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

d.   Fuel supply for the onsite emergency electrical system.

e.   Emergency equipment area cooling.

f.   Compressed gas or hydraulic systems required to support control or
      operation of safety systems.

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

Structures, systems and components in Safety Class 3 are listed in
<Table 3.2-1>.

3.2.3.3.2   Design Requirements for Safety Class 3

The design requirements for Safety Class 3 mechanical and structural
categories are listed in <Table 3.2-5>. This table correlates these
requirements with design condition categories.

Code requirements for Safety Class 3 electrical equipment are shown in
<Table 3.2-6>.

3.2.3.4   Other Structures, Systems and Components

3.2.3.4.1   Definition of Other Structures, Systems and Components

A boiling water reactor has a number of structures, systems and
components in the power conversion or other portions of the facility
which have no direct safety function but which may be connected to or
influenced by the equipment within the safety classes defined above.
Such structures, systems and components are designated as nonsafety
class (NSC).

3.2.3.4.2   Design Requirements for NSC Structures, Systems and
               Components

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

3.2.4   QUALITY ASSURANCE

Structures, systems and components whose safety functions require
conformance to the quality assurance requirements of <10 CFR 50,
Appendix B>, are those designated as Safety Class 1, 2 and 3 in
<Table 3.2-1>. The Quality Assurance Program is described in
<Chapter 17>.

3.2.5   CORRELATION OF SAFETY CLASSES WITH INDUSTRY CODES

The design of plant equipment is commensurate with the safety importance
of the equipment. Hence, the various safety classes have a gradation of
design requirements. Application of code and addenda for the reactor
coolant pressure boundary Safety Class 1 equipment supplied by GE is
listed in <Table 3.2-7>.

TABLE 3.2-1

EQUIPMENT CLASSIFICATION

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi-cation	Principal[5] Construction Code	Seismic[6] Category	Comment
I. Reactor System					
1. Reactor vessel	1	A	III-1	I	
2. Reactor vessel support skirt	1	N/A	III-NF	I	
3. Reactor vessel appurtenances, pressure retaining portions	1	A	III-1	I	
4. CRD housing supports	2	N/A	None	I	
5. Reactor internal structures, engineered safety features	2	N/A	None	I	
6. Nonsafety class reactor internals	NSC	N/A	None	N/A	See Note[28]
7. Control rods	2	N/A	None	I	
8. Control rod drives	2	N/A	None	I	
9. Core support structure	2	N/A	III-NG	I	
10. Fuel assemblies	2	N/A	None	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
II. Nuclear Boiler System					
1. Vessels, level instrumentation condensing chambers	1	A	III-1	I	
2. Vessels, air accumulators	3	C	III-3	I	
3. Piping, relief valve discharge	3	C	III-3	I	See Note[7]
4. Piping, main steam, within outboard isolation valve (MSIV)	1	A	III-1	I	See Note[8]
5. Piping, feedwater within outboard isolation check valve	1	A	III-1	I	
6. Pipe supports, main steam	1	A	III-NF	I	
7. Pipe restraints, main steam	2	N/A	N/A	I	
8. Piping, main steam, between MSIV and M.O. stop valve	2	B	III-2	I	See Note[8]
9. Piping, main steam between M.O.stop valve and turbine stop valve	NSC	D	B31.1	N/A	See Note[8],[24]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
10. (Deleted)					
11. (Deleted)					
12. (Deleted)					
13. Safety/Relief Valves	1	A	III-1	I	
14. MSIV's	1	A	III-1	I	
15. (Deleted)					
16. M.O. stop valves	2	B	III-2	I	
17. (Deleted)					
18. (Deleted)					
19. Mechanical modules with safety function	2	N/A	None	I	
20. Electrical modules with safety function	2	N/A	IEEE	I	
21. Cable with safety function	2	N/A	IEEE	I	
22. (Deleted)					
23. (Deleted)					
III. Recirculation System					
1. Piping	1	A	III-1	I	See Note[8]
2. Pipe supports, recirculation line	1	A	III-NF	I	
3. Pipe restraints, recirculation line	2	B	None	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
4. Pumps	1	A	III-1	I	
5. Valves	1	A	III-1	I	
6. Pump motors	NSC	N/A	None	N/A	
7. Low frequency motor generator set	NSC	N/A	None	N/A	
8. Cable with safety function	2	N/A	IEEE	I	
IV. CRD Hydraulic System					
1. Valves, scram discharge volume lines	2	B	III-2	I	See Note[8]
2. Valves, insert and withdraw lines	2	B	III-2	I	See Note[10]
3. (Deleted)					
4. Piping, scram discharge volume lines	2	B	III-2	I	See Note[8]
5. Piping, insert and withdraw lines	2	B	III-2	I	
6. Piping, other	NSC	D	B31.1	N/A	See Note[8]
7. Hydraulic control unit	2	N/A	None	I	See Note[11]
8. Electrical modules with safety function	2	N/A	IEEE	I	
9. Cable with safety function	2	N/A	IEEE	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classification	Principal[5] Construction Code	Seismic[6] Category	Comment
V.a Standby Liquid Control System					
1. Standby liquid control storage tank	2	B	III-2	I	
2. Pump	2	B	III-2	I	
3. Pump motor	2	N/A	None	I	
4. Valves, explosive	1	A	III-1	I	
5. Valves, isolation and piping between valves	1	A	III-1	I	See Note[8]
6. (Deleted)					
7. Piping, beyond isolation valves	2	B	III-2	I	See Note[8]
8. Electrical modules with safety function	2	N/A	IEEE	I	
9. Cable with safety function	2	N/A	IEEE	I	
V.b Standby Liquid Control Transfer System					
1. Piping, between mixing tank and outboard containment isolation valve	3	C	III-3	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
2. Valves, containment isolation and piping between valves	2	B	III-2	I	
3. Piping, inboard to storage tank	3	C	III-3	I	
4. (Deleted)					
5. Transfer pump	3	C	III-3	I	
6. Transfer pump motor	5	N/A	N/A	N/A	
VI. Neutron Monitoring System					
1. Electrical modules, IRM, OPRM and APRM	2	N/A	IEEE	I	
2. Cable, IRM, OPRM and APRM	2	N/A	IEEE	I	
3. Detector dry tube assembly	2	B	III-1	I	
VII. Reactor Protection System					
1. Electrical modules	2	N/A	IEEE	I	
2. Cable	2	N/A	IEEE	I	See Note[33]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
VIII. Process Radiation Monitors					
1. Electrical modules and cable, main steam line and reactor building ventilation monitors	2	N/A	IEEE	I	
2. (Deleted)					
3. Main steam line	2	N/A	N/A	I	
4. Containment vent exhaust	2	N/A	N/A	I	
5. (Deleted)					
6. (Deleted)					
7. (Deleted)					
IX. RHR System					
1. Heat exchangers, primary side	2	B	III-2& TEMA-C	I	
2. Heat exchangers, secondary side	3	C	III-3& TEMA-C	I	
3. Piping, within outboard isolation valves	1,2	A,B	III-1/III-2	I	See Note[8]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
4. Piping, beyond outboard isolation valves	2	B	III-2	I	See Note[8]
5. Pumps	2	B	III-2	I	
6. Pump motors	2	N/A	None	I	
7. Valves, isolation and LPCI line between valves	1,2	A,B	III-1/III-2	I	See Note[8]
8. (Deleted)					
9. (Deleted)					
10. Mechanical modules	2	N/A	None	I	
11. Electrical modules with safety function	2	N/A	IEEE	I	
12. Cable with safety function	2	N/A	IEEE	I	
13. Suppression pool strainer	2	N/A	II, IX	I	See Note[36]
X. Low Pressure Core Spray					
1. Piping, within outboard isolation valves	1	A	III-1	I	See Note[8]
2. Piping, beyond outboard isolation valves	2	B	III-2	I	See Note[8]
3. Pumps	2	B	III-2	I	
4. Pump motors	2	N/A	None	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
5. Valves, isolation	1	A	III-1	I	See Note[8]
6. (Deleted)					
7. Electrical modules with safety function	2	N/A	IEEE	I	
8. Cable with safety function	2	N/A	IEEE	I	
9. Suppression Pool Strainer	2	N/A	II, IX	I	See Note[36]
XI. High Pressure Core Spray					
1. Day Supply Diesel Tank	3	C	III-3	I	
2. Piping, within outboard isolation valve	1	A	III-1	I	See Note[8]
3. Piping, diesel service water	3	C	III-3	I	
4. Piping, return test line to condensate storage tank beyond second isolation valve	NSC	D	B31.1	N/A	See Note[8]
5. Piping, beyond outboard isolation valve, other	2	B	III-2	I	See Note[8]
6. Piping, HPCS Diesel Skid-Mounted	3	C	III-3	I	See Note[34]
7. Pump, HPCS	2	B	III-2	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi-cation	Principal[5] Construction Code	Seismic[6] Category	Comment
8. Pump motor, HPCS	2	N/A	None	I	
9. Pump, diesel service water	3	C	III-3	I	
10. (Deleted)					
11. (Deleted)					
12. (Deleted)					
13. (Deleted)					
14. Electrical modules, with safety function	2	N/A	IEEE	I	
15. Electrical auxiliary equipment	3	N/A	IEEE	I	
16. Cable with safety function	2	N/A	IEEE	I	
17. (Deleted)					
18. Suppression Pool Strainer	2	N/A	II, IX	I	See Note[36]
XII. RCIC System					
1. Piping, within outboard isolation valves	1,2	A,B	III-1	I	See Note[8]
2. Piping, beyond outboard isolation valves	2	B	III-2	I	See Note[8]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
3. Piping, return test line to condensate storage tank beyond second isolation valve	NSC	D	ANSI B31.1	N/A	See Note[8]
4. Pumps	2	B	III-2	I	
5. Valves, isolation	1,2	A,B	III-1	I	See Note[8]
6. Valves, return test line to condensate storage beyond second isolation valve and vacuum pump discharge line to containment isolation valves	NSC	D	ANSI B31.1.0	N/A	
7. (Deleted)					
8. Turbine	2	N/A	None	I	See Note[12]
9. Electrical modules with safety function	2	N/A	IEEE	I	
10. Cable with safety function	2	N/A	IEEE	I	
11. Suppression Pool Strainer	2	N/A	II, IX	I	See Note[36]

XIII. Fuel Service Equipment

See <Table 9.1-3>

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
XIV. Reactor Vessel Service Equipment					
See <Table 9.1-5>					
XV. In-Vessel Service Equipment					
1. Control rod grapple	2	N/A	None		See Note[3]
XVI. Refueling Equipment					
1. Refueling equipment assembly platform	2	N/A	None	I	
2. Refueling bellows	NSC	N/A	None	N/A	
3. Fuel transfer system	See <Table 9.1-4>				See Note[13]
4. Auxiliary Platform	NSC	N/A	None	I	
XVII. Storage Equipment					
1. Fuel Storage Racks	2	N/A	None	I	
2. Defective fuel storage container	2	N/A	None	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
XVIII. Radwaste System					
1. Concentrated Waste Tanks	NSC	D	III-3 AWWA-D100 & API 650	N/A	See Note[15]
2. Heat exchangers	NSC	D	TEMA-C	N/A	See Note[15]
3. Piping					
a. Between containment isolation valves	2	B	III-2	I	See Note[8]
b. All others	NSC	D	III-3 & B 31.1	N/A	See Note[15]
4. Pumps	NSC	D		N/A	See Note[15]
5. Valves					
a. Containment isolation	2	B	III-2	I	See Note[37]
b. Discharge flow control	NSC	D	B31.1	N/A	See Note[15]
c. All others	NSC	D	III-3 & B31.1	N/A	See Note[15]
6. Mechanical modules	NSC	N/A	None	N/A	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
XIX. Reactor Water Cleanup System					
1. Vessels: Filter/ demineralizer	NSC	C	III-3	N/A	
2. Heat exchangers carrying reactor water	NSC	C	III-3,TEMA-C	N/A	
3. Pump suction piping, within outboard isolation valve	1	A	III-1	I	See Note[8],[16]
4. Pump discharge piping, to RHR and feedwater	2	B	III-2	I	See Note[8]
5. Pumps	NSC	C	III-3	N/A	
6. Valves, isolation valves and piping between valves	1/2	A/B	III-1,III-2	I	See Note[8],[16],[37]
7. (Deleted)					
8. Filter/demineralizer	NSC	C	III-3	N/A	
9. Filter/demineralizer precoat subsystem	NSC	D	B31.1	N/A	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
10. Nonregenerative heat exchanger shells and interconnecting piping carrying closed cooling water	NSC	D	III-3 & TEMA-C	N/A	
XX. Fuel Pool Cooling and Cleanup System					
1. Surge tanks	3	C	III-3	I	
2. Heat exchangers	3	C	III-3 & TEMA-C	I	
3. Piping	2,3,NSC	B,C,D	III-2,III-3, B31.1	I, N/A	See Note[8]
4. Pumps	3	C	III-3	I	
5. Valves	2,3,NSC	B,C,D	III-2,III-3, B31.1	I, N/A	See Note[37]
6. Filter Demineralizer (F/D) System, Cask Pit Drain Subsystem and Fuel Transfer Tube Subsystem					
a. Piping and valves	NSC	D	ANSI B31.1	N/A	
b. Tanks	NSC	D	AWWA D100 API 650 ANSI B96.1	N/A	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
c. Pumps	NSC	D		N/A	
d. Vessels (F/D)	NSC	D	VIII	N/A	
7. RHR Connection	3	C	III-3	I	See Note[17]
8. Pump motors	3				
XXI. Control Room Panels					
1. Electrical modules, with safety function	2	N/A	IEEE	I	
2. Cable with safety function	2	N/A	IEEE	I	
3. Safety Parameter Display Panel	NSC	N/A	N/A	N/A	
XXII. Local Panels and Racks					
1. Electrical modules with safety function	2		N/A	IEEE	I
2. Cable with safety function	2		N/A	IEEE	I
XXIII. Offgas System					
1. Tanks	NSC	N/A	API 650 AWWA D100 or ANSI B96.1	N/A	See Note[18]
2. Heat exchangers	NSC	N/A	TEMA-C VIII-1	N/A	See Note[8],[18]
3. Piping	NSC	N/A	ANSI B31.1	N/A	See Note[8],[18]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
4. Pumps	NSC	N/A	–	N/A	See Note[18]
5. Valves, flow control	NSC	N/A	ANSI B31.1	N/A	See Note[18]
6. Valves, other	NSC	N/A	ANSI B31.1	N/A	See Note[8],[18]
7. Mechanical Modules with safety function	NSC	N/A	None	N/A	See Note[8],[18]
8. Pressure Vessels	NSC	N/A	VIII-1	N/A	See Note[18]

XXIV. Main Steamline Isolation Valve Leakage Control System

MSIV Leakage Control System has been abandoned in place per DCP 96-0044.

XXV. Emergency Service Water System

1. Piping	3	C	III-3	I	
2. Pumps	3	C	III-3	I	
3. Pump Motors	3	N/A	N/A	I	
4. Strainers	3	C	III-3	I	
5. Valves	3	C	III-3	I	
6. Electrical modules with safety function	2	N/A	IEEE	I	
7. Cable with safety function	2	N/A	IEEE	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
XXVI. Emergency Closed Cooling Water System					
1. Piping	3	C	III-3	I	
2. Pumps	3	C	III-3	I	
3. Pump Motors	3	N/A	N/A	I	
4. Valves	3	C	III-3	I	
5. Electric Modules with safety function	2	N/A	IEEE	I	
6. Cable with safety function	2	N/A	IEEE	I	
7. Heat Exchangers	3	C	III-3	I	
XXVII. Plant Service and Cooling Water Systems (Plant Service Water, Nuclear Closed Cooling & Turbine Building Closed Cooling)					
1. Piping and valves forming part of primary containment boundary	2	B	III-2	I	See Note[37]
2. Other equipment and piping	NSC	N/A	VIII-1 B31.1	N/A	See Note[19]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
XXVIII. Instrument and Service Air Systems					
1. Vessels, accumulators, supporting safety- related systems	3	C	III-3	I	
2. Piping and valves in lines between above accumulators and safety-related systems	3	C	III-3	I	
3. Other equipment and piping	NSC	N/A	VIII-1 B31.1	N/A	
XXIX. Diesel Generator Systems					
1. Day tanks and fuel storage tank	3	C	III-3	I	
2. Piping and valves, fuel oil system and diesel cooling water system	3	C	III-3	I	
3. Pumps, fuel oil system and diesel service water system	3	C	III-3	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
4. Pump motors, fuel oil system and diesel service water systems	3	N/A	N/A	I	
5. Diesel-generators	2	DEMA	N/A	I	
6. Diesel-generators starting air system	3	C	III-3	I	
7. Lube oil system	3	C	III-3	I	
8. Electrical modules with safety functions	2	N/A	IEEE	I	
9. Cable with safety functions	2	N/A	IEEE	I	
10. Diesel generator combustion air intake, exhaust system intake and exhaust systems intake air filter and valves	3	C	III-3	I	
11. Exhaust silencer and crankcase vent piping	NSC	N/A	N/A	N/A	
12. Jacket water cooling	3	C	III-3 TEMA-C	I	

XXX. Power Conversion System

1. Main Steam piping   (See Section II of this Table)

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
2. Main Steam Lines from Turbine Control Valves to Turbine Casing	NSC	N/A	ANSI B31.1	N/A	See Note[20]
3. Piping and valves, other	NSC	N/A	ANSI B31.1	N/A	See Note[21],[22]
4. Pressure Vessels	NSC	N/A	VIII-1	N/A	
5. Condensate & Feedwater Pumps	NSC	N/A	N/A	N/A	
6. All Other Pumps	NSC	N/A	N/A	N/A	
7. Tanks	NSC	N/A	API 650, AWWA 100 or ANSI B96.1	N/A	
8. Other Components	NSC	N/A	N/A	N/A	
XXXI. Condensate Storage and Transfer System					
1. Condensate storage tank	NSC	N/A	AWWA D100	N/A	
2. Piping and valves	NSC	N/A	ANSI B31.1	N/A	
3. Other components	NSC	N/A	N/A	N/A	
XXXII. Auxiliary AC Power System					
1. 4160V switch gear with safety functions	2	N/A	IEEE	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi-cation	Principal[5] Construction Code	Seismic[6] Category	Comment
2. 480V load centers with safety functions	2	N/A	IEEE	I	
3. 480V MCC's with safety functions	2	N/A	IEEE	I	
4. All cables with safety function (including under-ground cables, cable splices, connections, and terminal blocks)	2	N/A	IEEE	I	
5. Conduit, cable tray and supports containing Class 1E cables	2	N/A	IEEE	I	
6. Conduit, cable tray and supports containing non-1E cables in safety class structures where failure may damage other safety-related items.	NSC	N/A	N/A	I	See Note[28],[31]
7. 4160/480 transformer with safety functions	2	N/A	N/A	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
8. Protective relays and control panels with safety functions	2	N/A	N/A	I	
9. AC control power inverters with safety functions	2	N/A	N/A	I	
10. Containment electrical penetrations	2	N/A	III MC	I	
11. All components with safety functions	2	N/A	N/A	I	
XXXIII. 125 Volt DC Power System					
1. 125V batteries, racks, chargers, distribution equipment with safety functions	2	N/A	IEEE	I	
2. Cables with safety functions	2	N/A	IEEE	I	
3. Conduit and cable tray and supports containing Class 1E cables	2	N/A	IEEE	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
4. Conduit and cable tray and supports containing non-1E cables in safety class structures where failure may damage other safety-related items	NSC	N/A	N/A	I	See Note[28],[31]
5. Protective relays and control panels with safety functions	2	N/A	IEEE	I	
6. All components with safety functions	2	N/A	IEEE	I	

XXXIV. Structures

1. Reactor Building Complex					
a. Drywell/interior structure	2	N/A	<Section 3.8.3>	I	
b. Containment vessel	2	N/A	III-MC	I	
c. Shield Building	2	N/A	<Section 3.8.1>	I	
2. Auxiliary Building	2	N/A	<Section 3.8.4>	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
3. Steam Tunnel Between Aux Bldg and Turbine Bldg	NSC	N/A	N/A	N/A	
4. Fuel Handling and Intermediate Buildings	2	N/A	<Section 3.8.4>	I	
5. Radwaste Building	3	N/A	<Section 3.8.4>	I	See Note[23]
6. Turbine Building	NSC	N/A	N/A	N/A	
7. Control Building	2	N/A	<Section 3.8.4>	I	
8. Diesel Generator Building	2	N/A	<Section 3.8.4>	I	
9. Offgas Building	3	N/A	<Section 3.8.4>	I	
10. Emergency Service Water Pump House	3	N/A	<Section 3.8.4>	I	
11. Circulating and Service Water Pump House	NSC	N/A	N/A	N/A	
12. Intake structures and cooling water tunnels	3	N/A	<Section 3.8.4>	I	
13. Discharge tunnel entrance structure and downshaft	NSC	N/A	N/A	N/A	
14. Discharge tunnel and diffuser nozzle	3	N/A	<Section 3.8.4>	I	
15. Spent fuel pool and liner	2	N/A	AWS D1.1 RDT-F6-6T	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
16. Seismic Category I fill	N/A	N/A	\<Section 2.5.4.5\>	I	See Note[32]
17. Foundation for Seismic Category I electrical duct banks and manholes	2	N/A	ACI-318-71	I	
18. Condensate Storage Tank Foundation and Dike Walls	3	N/A	\<Section 3.8.4\>	I	
19. Unit 2 Reactor Building					
a. Containment vessel	3	N/A	III-MC	II/I	See Note[41]
b. Shield Building	3	N/A	See Note[41]	I	See Note[41]
20. Unit 2 Auxiliary Building	3	N/A	See Note[41]	I	See Note[41]
XXXV. Heating, Cooling and Ventilation Systems					
1. Annulus Exhaust Gas Treatment System Units	2	N/A	AMCA, ERDA 76-21, UL-586, ANSI N509, RDT M16-1T, ANSI N101.1	I	
a. Filter housing	2	N/A	AMCA, UL507 AISI	I	
b. Fans and motors	2	N/A	NEMA MG1	I	
c. Demisters	2	N/A	UL	I	
d. Heaters, electric	2	N/A	NEMA	I	
e. Ductwork	2	N/A	ASTM, SMACNA	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classification	Principal[5] Construction Code	Seismic[6] Category	Comment
f. Dampers, fire control	NSC	N/A	AMCA, ASTM	I	See Note[29]
g. Dampers, check	3	N/A	AMCA, ASTM	I	
2. Drywell Cooling Units	NSC	N/A	AMCA, ARI-410	N/A	
3. Containment Vessel Cooling Units	NSC	N/A	AMCA,	N/A	
4. Purge Supply Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29]
5. Purge Exhaust Units	NSC	N/A	AMCA, ERDA 76-21, UL 586, ANSI N509, RDT M16-1T, ANSI N101.1	I	See Note[29]
6. Piping & Isolation Valves from Containment Vessel through outer Isolation valves	2	B	III-2	I	See Note[37]
7. (ECCS) Pump Rooms Cooling Units	3	C	AMCA, ARI-410, III-3	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
8. Emergency Closed Cooling Pump Area Cooling Units	3	C	AMCA, ARI-410, III-3	I	See Note[29]
9. Radwaste Building Supply Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29]
10. Radwaste Building Exhaust Units	NSC	N/A	AMCA, ERDA 76-21, UL 586, ANSI N509, RDT M16-1T, ANSI N101.1	I	See Note[29]
11. Fuel Handling Building Supply Units					
a. Fans and motors	3	N/A	AMCA, NEMA MG1	I	
b. Coils, heating	Other	N/A	ARI-410	I	
c. Prefilter	3	N/A	UL	I	
d. Ductwork & dampers	3	N/A	AMCA, ASTM SMACNA	I	See Note[25] See Note[29]
e. Electrical modules with safety function	3	N/A	IEEE	I	
f. Cable with safety function	3	N/A	IEEE		

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
12. Fuel Handling Building Exhaust Units					
a. Fans and motors	3	N/A	AMCA, NEMA MG1	I	
b. Coils, electric	3	N/A	NEMA	I	
c. Prefilters	3	N/A	UL	I	
d. Hepa filters	3	N/A	MIL-F-51079	I	
e. Adsorber unit	3	N/A	ERDA 76-21 RDT M16-1T	I	
f. Demister	3	N/A	MSAR 71-45	I	
g. Charcoal filter housing	3	N/A	AISI	I	
h. Ductwork and dampers	3	N/A	AMCA, SMACNA ASTM	I	See Note[25] See Note[29]
i. Electrical modules with safety function	3	N/A	IEEE	I	
j. Cable with safety function	3	N/A	IEEE	N/A	
13. Auxiliary Building Supply Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
14. Auxiliary Building Exhaust Units	NSC	N/A	AMCA, ERDA 76-21, UL 586, ANSI N509, RDT M16-IT, ANSI N101.1	I	See Note[29]
15. Intermediate Building Supply Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29]
16. Intermediate Building Exhaust Units	NSC	N/A	AMCA	N/A	See Note[29]
17. Diesel Generator Bldg Ventilation Units		N/A			
a. Fans and motors	3		AMCA, NEMA MG1	I	
b. Ductwork and dampers	3		AMCA, SMACNA ASTM	I	See Note[29]
c. Electrical modules with safety function	3		IEEE	I	
d. Cable with safety function	3		IEEE	N/A	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
18. Emergency Service Water Pumphouse Ventilation Units			N/A		
a. Fans and motors	3		AMCA, NEMA MG1	I	
b. Ductwork and dampers	3		AMCA, SMACNA ASTM	I	See Note[29]
c. Electrical modules with safety function	3		IEEE	I	
d. Cable with safety function	3		IEEE	N/A	
19. Turbine Bldg Cooling and Ventilation Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29]
20. Offgas Bldg Exhaust Units					See Note[40]
a. Fans and motors	3	N/A	AMCA, NEMA MG1	I	
b. Coils, electric	3	N/A	NEMA	I	
c. Prefilters	3	N/A	UL	I	
d. HEPA filters	3	N/A	MIL-F-51079	I	
e. Adsorber unit	3	N/A	ERDA 76-21 RDT-M16-1T	I	
f. Charcoal filter housing	3	N/A	AISI	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
g. Ductwork and dampers	3	N/A	AMCA SMACNA ASTM	I	See Note[25],[26] See Note[29]
h. Electrical modules with safety function	3	N/A	IEEE	I	
i. Cable with safety function	3	N/A	IEEE	I	
j. Offgas holdup pipe room ventilating fan	NSC	N/A	AMCA NEMA MG1	N/A	
21. Turbine Building Power Complex Ventilation Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29]
22. Heater Bay Ventilation Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29]
23. Water Treatment Building Ventilation Units	NSC	N/A	AMCA	N/A	
24. Control Room Supply and Return Units					
a. Fans and motors	3	N/A	AMCA, NEMA MG1	I	
b. Coils, Cooling	3	N/A	ARI-410	I	
c. Filters	3	N/A	UL	I	
d. Humidifiers	NSC	N/A	-	N/A	
e. Ductwork and dampers	3	N/A	AMCA, SMACNA ASTM	I	See Note[25] See Note[29]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
f. Duct heaters, electrical	NSC	N/A	NEMA	N/A	
g. Electrical modules with safety function	3	N/A	IEEE	I	
h. Cable with safety function	3	N/A	IEEE	N/A	
25. Control Room Emergency Recirculation units					
a. Fans and motors	3	N/A	AMCA, NEMA MG1	I	
b. Coils, electric	3	N/A	NEMA	I	
c. Prefilters	3	N/A	UL	I	
d. Hepa filters	3	N/A	MIL-F-51079	I	
e. Adsorber unit	3	N/A	ERDA 76-21 RDT M16-1T	I	
f. Demister	3	N/A	MSAR 71-45	I	
g. Charcoal filter housing	3	N/A	AISI	I	
h. Ductwork and dampers	3	N/A	AMCA, SMACNA ASTM	I	See Note[25],[29]
i. Electrical modules with safety function	3	N/A	IEEE	I	
j. Cable with safety function	3	N/A	IEEE	N/A	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
26. MCC, Switchgear and Misc Elec. Equipment Areas Supply and Return Units					
a. Fans and motors	3	N/A	AMCA, NEMA MG1	I	
b. Coils, cooling	3	N/A	ARI 410-64 ASHRAE 33-64	I	
c. Filters	3	N/A	UL900-1965 ASHRAE 52-68	I	
d. Ductwork and dampers	3	N/A	UL, NEMA	I	See Note[29]
e. Duct heaters, electrical	NSC	N/A	UL, NEMA	N/A	See Note[25]
f. Electrical modules with safety function	3	N/A	IEEE	I	
g. Cable with safety function	3	N/A	IEEE	N/A	
27. Battery Room Exhaust Units					
a. Fans and motors	3	N/A	AMCA, NEMA MG1	I	
b. Ductwork and dampers	3	N/A	SMACNA, ASHRAE	I	See Note[25],[29]
c. Electrical modules with safety function	3	N/A	IEEE	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi-cation	Principal[5] Construction Code	Seismic[6] Category	Comment
d. Cable with safety function	3	N/A	IEEE	N/A	
28. Controlled Access Area & Misc Equipment Areas Supply, Return Units	NSC	N/A	AMCA, ARI-410	N/A	See Note[29],[35]
29. Controlled Access Area Exhaust Units	NSC	N/A	AMCA, ERDA 76-21, UL 586, ANSI N509, RDT M16-1T, ANSI N101.1	N/A	See Note[29]
30. Combustible Gas Control					
a. Compressors	2	B	III-2	I	
b. Valves	2	B	III-2	I	
c. Piping	2	B	III-2	I	
d. Electrical modules with safety function	2	N/A	IEEE	I	
e. Cable with safety function	2	N/A	IEEE	I	
f. Backup Purge System	2	B	III-2	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
g. Hydrogen Analyzer	2	N/A	IEEE, NEMA II, III-1, V, IX	I	
h. Hydrogen Recombiners	2	N/A	IEEE <10 CFR 50>	I	
i. Hydrogen Igniters	2	N/A	ANSI, IEEE, ASME II, IX	I	
31. Control Complex Chillers					
a. Motors	3	N/A	NEMA MG1	I	
b. Heat exchangers	3	C	III-3	I	
c. Piping	3	C	III-3	I	
d. Valves	3	C	III-3	I	
e. Electrical modules	3	N/A	IEEE	I	
f. Cable with safety function	3	N/A	IEEE	N/A	
32. Control Complex Chilled Water Pumps					
a. Motors	3	N/A	NEMA MG1	I	
b. Pumps	3	C	III-3	I	
c. Electrical modules	3	N/A	IEEE	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
d. Cable with safety function	3	N/A	IEEE	N/A	
33. Containment Vessel and Turbine Bldg Chillers	NSC	N/A	VIII-1	N/A	
34. Containment Vessel and Turbine Bldg Chiller Water Pumps	NSC	N/A	N/A	N/A	
35. Containment Spray Piping and Nozzles	2	B	III-2	I	
36. Drywell Vacuum Relief	2	B	III, NEMA, IEEE	I	
37. ADHR Room A/C Units	NSC	N/A	N/A	N/A	

XXXVI. Other Components

1. Containment Crane	3	N/A	N/A	I	
2. Refueling Cask Crane	3	N/A	N/A	I	
3. Containment isolation valves and piping between for all containment penetrations not listed above	2	B	III-2	I	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classification	Principal[5] Construction Code	Seismic[6] Category	Comment
XXXVII. Suppression Pool Make-up System					
1. Valves	2	B	III-2	I	
2. Piping	2	B	III-2	I	
3. Electrical modules with safety function	2	N/A	IEEE	I	
4. Cable with safety function	2	N/A	IEEE	I	
XXXVIII. Underdrain System					
1. Porous concrete	3	N/A	<Section 3.8.5>	I	
2. Porous concrete pipe	NSC	N/A	ASTM C654	N/A	
3. Pumps	NSC	N/A	N/A	N/A	
4. Pumped discharge piping	NSC	N/A	B31.1	N/A	
5. Gravity discharge piping	3	C	III-3; <Section 2.4.13.5>	I	
6. Manholes	3	N/A	ACI 318 & ACI 301	I	
XXXIX. Liquid System Radiation Monitors					
1. Emergency service water (ESW) systems	NSC	N/A	N/A	N/A	

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
2. Nuclear closed cooling system	NSC	N/A	N/A	N/A	
3. Radwaste effluent to ESW	NSC	N/A	N/A	N/A	
4. ADHR Heat Exchanger Service Water Outlet	NSC	N/A	N/A	N/A	
5. Underdrain system	NSC	N/A	N/A	N/A	
XL. Airborne and Atmospheric Radiation Monitors					
1. Radwaste Bldg. vent exhaust	NSC	N/A	N/A	N/A	See Note[26]
2. Aux Bldg. vent exhaust	NSC	N/A	N/A	N/A	
3. Intermediate Bldg. vent exhaust	NSC	N/A	N/A	N/A	See Note[26]
4. Fuel handling area vent exhaust	NSC	N/A	N/A	N/A	See Note[26]
5. Offgas Bldg. vent exhaust	NSC	N/A	N/A	N/A	See Note[26]
6. Drywell atmospheric	NSC	N/A	N/A	I (Con- tainment Only)	See Note[26],[30]
7. Containment atmospheric	NSC	N/A	N/A	I (Con- tainment Only)	See Note[26],[30]
8. (Deleted)					

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
9. (Deleted)					
10. (Deleted)					
11. (Deleted)					
12. Containment vessel (portable) and drywell purge exhaust	NSC	N/A	N/A	N/A	See Note [26],[30]
13. Control room airborne	NSC	N/A	N/A	N/A	See Note[26],[30]
14. Annulus exhaust trains A and B	NSC	N/A	N/A	N/A	See Note[26]
XLI. Area Radiation Monitors					
1. Personnel air lock	NSC	N/A	N/A	N/A	See Note[26],[30]
2. CRD HCU west	NSC	N/A	N/A	N/A	See Note[26],[30]
3. RWCU F/D rec. TK area	NSC	N/A	N/A	N/A	See Note[26],[30]
4. TIP drive area	NSC	N/A	N/A	N/A	See Note[26],[30]
5. RWCU F/D area	NSC	N/A	N/A	N/A	See Note[26],[30]
6. Upper pool area	NSC	N/A	N/A	N/A	See Note[26],[30]
7. Aux Bldg. Elev. 574' E	NSC	N/A	N/A	N/A	See Note[26],[30]
8. Aux Bldg. Elev. 574' W	NSC	N/A	N/A	N/A	See Note[26],[30]
9. Turbine room E	NSC	N/A	N/A	N/A	See Note[26],[30]
10. CRD HCU east	NSC	N/A	N/A	N/A	See Note[26],[30]
11. Turbine room W	NSC	N/A	N/A	N/A	See Note[26],[30]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
12. Turbine Bldg. Elev. 605'	NSC	N/A	N/A	N/A	See Note[26],[30]
13. Hotwell pump area	NSC	N/A	N/A	N/A	See Note[26],[30]
14. Turbine Bldg. sump area	NSC	N/A	N/A	N/A	See Note[26],[30]
15. Offgas Bldg. Elev. 584'	NSC	N/A	N/A	N/A	See Note[26],[30]
16. Condensate filter pump area	NSC	N/A	N/A	N/A	See Note[26],[30]
17. Offgas after-filter area	NSC	N/A	N/A	N/A	See Note[26],[30]
18. HP feedwater heater area	NSC	N/A	N/A	N/A	See Note[26],[30]
19. Feedpump area	NSC	N/A	N/A	N/A	See Note[26],[30]
20. Control room	NSC	N/A	N/A	N/A	See Note[26],[30]
21. Offgas holdup area	NSC	N/A	N/A	N/A	See Note[26],[30]
XLII. Primary Containment Area Monitors (High Range)	2	N/A	N/A	I	See Note[27]
XLIII. Secondary Containment Area Monitors (High Range)	2	N/A	N/A	I	See Note[27]
XLIV. Airborne Effluents Released from Plant (1.7×10^{-3} µCi/cc to 10^5 µCi/cc)					
1. Plant main vent	2	N/A	N/A	I	See Note[27]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
2. Offgas vent	2	N/A	N/A	I	See Note[27]
3. Turbine Bldg./heater bay vent	2	N/A	N/A	I	See Note[27]
XLV. Airborne Effluents Released from Plant (10^{-6} µCi/cc to 10^{-2} µCi/cc)					
1. Plant main vent	NSC	N/A	N/A	N/A	See Note[27],[30]
2. Offgas vent	NSC	N/A	N/A	N/A	See Note[27],[30]
3. Turbine Bldg./ heater bay vent	NSC	N/A	N/A	N/A	See Note[27],[30]
XLVI. Particulates and Halogens Collection (All identified release points)					
1. Plant main vent	NSC	N/A	N/A	N/A	See Note[27],[30]
2. Offgas vent	NSC	N/A	N/A	N/A	See Note[27],[30]
3. Turbine Bldg. heater bay	NSC	N/A	N/A	N/A	See Note[27],[30]
XLVII. Portable Radiation Monitoring Equipment					
1. Survey area monitors	NSC	N/A	N/A	N/A	See Note[27],[30]

TABLE 3.2-1 (Continued)

Principal Component[1]	Safety[2] Class	Quality Group[4] Classifi- cation	Principal[5] Construction Code	Seismic[6] Category	Comment
2. Airborne radio-halogen and particulates	NSC	N/A	N/A	N/A	See Note[27],[30]
3. Plant and environs equipment	NSC	N/A	N/A	N/A	See Note[27],[30]
4. Multichannel gammaray spectrometer	NSC	N/A	N/A	N/A	See Note[27],[30]
XLVIII. TSC and TC Airborne and Area Monitors	NSC	N/A	N/A	N/A	See Note[26],[30]
XLIX. Feedwater Leakage Control System					
1. Piping and valves of the Division 2 system	2	B	III-2	I	
2. Piping and valves of the Division 1 system	2	B	III-2	I	
L. ADHR System					
1. ADHR pump and motor	NSC	N/A	API 610 B 16.5	I	
2. ADHR heat exchanger	NSC	N/A	VIII-1	I	
3. Valves	NSC	N/A	N/A	I	See Note[39]
4. Piping	NSC	N/A	ANSI B31.1	I	See Note[39]
5. ADHR filter	NSC	N/A	VIII-1	I	

TABLE 3.2-1 (Continued)

NOTES:

(1) A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies and signal processors and mechanical modules include turbines, strainers and orifices. All valve operators are implicit on the entry of valves, except in those applications where the valve's sole function is maintaining a reactor coolant pressure boundary.

(2) 1, 2, 3 - safety classes defined in <Section 3.2.3>
NSC - No Safety Class
All Safety Class 1, 2 or 3 systems and components meet the quality assurance requirements of <10 CFR 50, Appendix B>. Additional details of the quality assurance program are provided in <Chapter 17>.

(3) The control rod grapple is classified as N/A, i.e., exempt from seismic evaluation because it is suspended from a cable which mitigates the seismic effect.

(4) A, B, C, D - NRC quality groups defined in <Regulatory Guide 1.26>. The equipment is constructed in accordance with the codes listed in <Table 3.2-2>. N/A - Quality Group Classification not applicable to this equipment.

(5) Notations for principal construction codes are:

ACI 301 American Concrete Institute "Specifications for Structural Concrete for Buildings"

ACI 318 "1973 Supplement to Building Code Requirements for Reinforced Concrete and Commentary on Same"

AISI   American Iron and Steel Institute

TABLE 3.2-1 (Continued)

ASHRAE   American Society of Heating, Refrigerating and
33-64   Air-Conditioning Engineers, Inc. "Methods of Testing for
           Rating Forced Circulation Air Cooling and Air Heating
           Coils"

ASHRAE   "Method of Testing Air Cleaning Devices used in General
52-64   Ventilation for Removing Particulate Matter"

AMCA   Air Moving and Conditioning Association

ANSI   American National Standards Institute "Power Piping"
B31.1

ANSI   "Welded Aluminum Alloy Field-Erected Storage Tanks"
B96.1

ANSI   "Efficiency testing of Air-Cleaning Systems Containing
N101.1   Devices for Removal of Particles"

ANSI
N509   "Nuclear Power Plant Air Cleaning Units and Components"

API 610   American Petroleum Institute "Centrifugal Pumps for
           Petroleum, Heavy Duty Chemical, and Gas Industry
           Services"

API 620   American Petroleum Institute "Large, Welded Low Pressure
           Storage Tanks"

API 650   American Petroleum Institute "Welded Steel Tanks for Oil
           Storage"

ARI 410   American Refrigerating Institute "Forced-Circulation
           Air-Cooling and Air-Heating Coils"

ASME Notations are as follows: (American Society of Mechanical
Engineers)

II   Section 2   Material Specifications

III-1, 2, 3, or MC   Boiler and Pressure Vessel Code, Section III,
                      Subsection NB, NC, ND, or NE respectively.
                      Specific editions and/or code cases used are in
                      compliance with <10 CFR 50.55a> and are
                      documented in the ASME Design Specifications,
                      Code Data Reports, or associated documents.
                      See <Chapter 5.2> for Reactor Coolant Pressure
                      Boundary Components.

TABLE 3.2-1 (Continued)

III-NF   Boiler and Pressure Vessel Code, Section III,
                  Subsection NF, Component Supports.

V   Section 5   Nondestructive Examination

TABLE 3.2-1 (Continued)

VIII-1	Section 8	Pressure Vessels
IX	Section 9	Welding and Brazing Qualifications
XI	Section 11	Rules for Inservice Inspection of Nuclear Power Plant Components
III-NG	Section III, Subsection NG, "Core Support Structures"	
ASTM	American Society for Testing and Materials	
ASTM C654	American Society for Testing & Materials "Standard Specification for Porous Concrete Pipe"	
AWWA D100	American Water Works Association "Welded Steel Elevated Tanks, Standpipes and Reservoirs for Water Storage"	
DEMA	Diesel Engine Manufacturer's Association	
ERDA 76-21	Energy Research and Development Administration "Nuclear Air Cleaning Handbook"	
MIL-F-51079	Military Specification "Fire Medium, Fire Resistant, High Efficiency"	
MSAR 71-45	Mine Safety Appliance Research Corp. "Entrained Moisture Separator for Fire, Water, Air, Steam Service, Their Performance, Development, and Status."	
NEMA MG-1	National Electric Manufacturers Association "Motors and Generators"	
RDT-M16-1T	Reactor Research and Development USAEC "Gas-Phase Absorbents for Trapping Radioactive Iodine and Iodine Compounds."	
RDT-F6-6T	Division of Reactor Development and Technology USAEC - Welding of Structural Components	
SMACNA	Sheet Metal Air & Conditioning Contractors National Association	
TEMA-C	Tubular Exchanger Manufactures Association, Class C	

TABLE 3.2-1 (Continued)

UL	Underwriters' Laboratories, Inc.
UL 507 (ANSI)	"Safety Standards for Electric Fans"
UL 586	"High Efficiency, Particulate, Air Filter Units"
UL 900	"Air Filter Units"
IEEE	The Institute of Electrical and Electronics Engineers, Inc.
<10 CFR 50>	Title 10, Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities"

6.   I - Constructed in accordance with the requirements of Seismic Category I structures and equipment as described in <Section 3.7>, Seismic Design. N/A - The seismic requirements are not applicable to the equipment.

7.   Safety/relief valve discharge line piping from the safety/relief valve to the suppression pool consists of two parts. The first is attached at one end to the safety relief valve and attached at its other end to the structural steel just below the main steam header through a pipe anchor. The main steam piping, including this portion of the safety/relief valve discharge piping, is analyzed as a complete system. The second part of the safety/relief valve discharge piping extends from the anchor (located below the mainsteam header) to the suppression pool. Because of the upstream anchor on this part of the line, it is physically decoupled from the main steam header and is, therefore, analyzed as a separate piping system.

8.   Components installed in safety class process lines have the same classification as the process line. Lines connected to safety class process lines have the same classification as the process line from the process line connection to and including the root or isolation valve except as noted in paragraph a below. Connecting lines that do not serve a safety function are classified NSC downstream of the root or isolation valve.

   a.   Lines 3/4 inch and smaller which are part of the reactor coolant pressure boundary are Safety Class 2 up to and including the root or isolation valve.

TABLE 3.2-1 (Continued)

b.   Instrument lines larger than 3/4 inch which are connected to
      Safety Class 1 (SC-1) process lines have a restricting orifice
      installed between the process connection and the root or
      isolation valve.

c.   (Deleted)

d.   The safety class of instrument lines that are SC-1 and serve a
      safety function is reduced to SC-2 downstream of the root or
      isolation valve, except as noted in Paragraph 8a above.

e.   Instrument lines that are connected to SC-2 or SC-3 process
      lines and serve a safety function retain this same safety
      class downstream of the root or isolation valve. The main
      steam SRV pressure sensing lines are Safety Class 3/Quality
      Group C up to and including the root valve, with a downstream
      transition to Safety Class 2/Quality Group B through a drywell
      penetration and including downstream instrument isolation
      valves, pressure switches and drain valves. SRV pressure
      sensing components are Seismic Category I.

f.   Instrument lines that are connected to process lines
      classified as NSC are designed to ANSI B31.1.

9. (Deleted)

10. The control rod drive insert and withdraw lines from the drive
     flange up to and including the first valve on the hydraulic control
     unit are Safety Class 2.

11. The hydraulic control unit (HCU) is a General Electric
     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 pressures and flows to
     accomplish slow insertion or withdrawal of the control rods for
     power control, and rapid insertion for reactor scram.

     Although the hydraulic control unit, as a unit, 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 Groups A, B, C, D pressure integrity quality
     levels clearly apply at all levels to the interfaces between the
     HCU and the conventional piping components (e.g., pipe nipples,
     fittings, simple hand valves, etc.), they do not apply to the
     specialty parts (e.g., solenoid valves, pneumatic components and
     instruments).

TABLE 3.2-1 (Continued)

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, (1) all welds are LP inspected, (2) all socket welds are inspected for gap between pipe and socket bottom, (3) all welding is performed by qualified welders, (4) all work is done per written procedures. Code 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 Code Groups A, B or C. This is supplemented by the QC technique described above.

12. The turbine does not fall within the available design codes. To assure that the turbine is fabricated to the standards commensurate with safety and performance requirements, General Electric 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 at 1.5 x design pressure,

c. All high pressure castings are radiographed according to:

ASTM E-94
    E-142 (maximum feasible volume)
    E-71, 186 or 280 (severity Level 3)

d. As-cast surfaces are magnetic particle or liquid penetrant tested according to ASME, Section III, Paragraph NB-2575, NB-2576 or NC-2576.

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

f. Butt-welds are radiographed according to ASME, and magnetic particle or liquid penetrant tested according to ASME Boiler and Pressure Vessel Code. Acceptance standards are in accordance with ASME Boiler and Pressure Vessel Code Section III, Paragraph NB-5340, NC-5340, NB-5350, NC-5350, respectively,

g. Notifications are made on major repairs, and records maintained thereof,

TABLE 3.2-1 (Continued)

h.   Record system and traceability according to ASME Section III, NA-4000,

i.   Control and identification according to ASME Section III, NA-4000,

j.   Procedures conform to ASME Section III, NB-5100, NC-5100,

k.   Inspection personnel are qualified according to ASME Section III, NB-5500, NC-5500.

13.   The portions of the transfer tube and the parts immediately adjacent and attached to that portion which are used for containment isolation are quality Group B. The remainder of the tube and the valves are quality Group D. For those parts of the fuel transfer system such as the sheave bore, fuel carrier up ender and similar parts for which no codes exist, quality group clarifications do not apply.

14.   (Deleted)

15.   Tanks, pipe, pumps, and valves in the NSC portions of the radwaste system are constructed to high industry standards. Tanks are radiographed per Appendix A of AWWA D-100.

16.   The reactor water cleanup system from the reactor vessel through the outermost isolation valves is Safety Class 1.

17.   To comply with <Regulatory Guide 1.26> and <Regulatory Guide 1.29>, the RHR system is interconnected to the fuel pool, thereby providing a redundant Seismic Category I source of coolant to the fuel pool. Additionally, systems for maintaining water quality and quantity are designed so that malfunction or failure will not cause significant loss of inventory.

18.   The equipment (including piping) in the offgas system is supplied and installed in accordance with the quality requirements recommended in <Regulatory Guide 1.143>. An additional description of the quality program for the offgas system is given in <Section 11.3.2>. The charcoal adsorbers and their supports are designed to the seismic criteria recommended in <Regulatory Guide 1.143>. All other components in the offgas system are non-seismic design as suggested in <Regulatory Guide 1.143>.

19.   Service water systems for emergency equipment cooling are classified as Safety Class 3 and Seismic Category I in <Table 3.2-1>.

TABLE 3.2-1 (Continued)

Recirculation pump seal cooling water is nonsafety and non-seismic. Loss of cooling water to the reactor coolant pumps (recirculation system pumps) is not considered to be a safety-related problem, as described in GE topical report NEDO-24083 and <Section 15.3>.

In addition, the response to TMI Action II.K.3.25 <Appendix 1A> concluded that the loss of pump seal cooling for 2 hours is not a safety problem and the total loss of recirculation pump seal cooling water would not result in a significant safety problem.

TABLE 3.2-1 (Continued)

(Deleted) |

TABLE 3.2-1 (Continued)

20. The main steam lines from the turbine control valves to the turbine casing will meet the following requirements:

a. All longitudinal and circumferential butt weld joints are radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination is substituted. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4.3 in ANSI B31.1 Code.

b. All fillet and socket welds are examined by either magnetic particle or liquid penetrant method. All structural attachment welds to pressure retaining materials are examined by either magnetic particle or liquid penetrant method. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4.3 in ANSI B31.1 Code.

c. Inspection records for the fabrication of these lines will be retained as described under Item 4(b), Page 24 of the General Electric Publication GEZ-4982A, "General Electric Large Steam Turbine-Generator Quality Control Program." Specifically, x-rays are retained for the life of the turbine and certificates of tests are held by GE for five (5) years after shipment. The inspection records of installation tests will be maintained for the life of the plant.

d. A statement is obtained from the manufacturer, certifying that the main steam lines have been manufactured in accordance with the Quality Assurance Plan described on Page 24 of GEZ 4982A.

TABLE 3.2-1 (Continued)

21. A certification is obtained from the vendors of the turbine stop valves and turbine bypass valves which shall certify compliance with the following:

    a. All cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods.

        Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examination in ANSI B31.1, 1967, Paragraph 136.4.3.

    b. The vendor of the turbine stop valves and turbine bypass valves utilizes quality control procedures equivalent to those defined for valves in the GEZ 4982A.

22. In addition to a piston-type check valve inside the drywell and a piston-type check valve outside containment, a third valve with high leak-tight integrity is provided in each feedwater line outside containment. Buffer pistons will be provided in the feedwater line check valves. These buffer pistons will slow valve closing to limit check valve slam and associated pressure transients. The high leak-tight integrity isolation valve will be remote-manually operated from the control room using signals which indicate loss of feedwater flow.

    The classification of the feedwater lines from the reactor vessel to and including the check valve outside containment is Safety Class 1; from this check valve to the high integrity isolation valve it is Safety Class 2; beyond the third valve it is nonsafety class <Figure 10.1-3>.

23. A nonsafety structural steel annex forms part of the radwaste building. See <Section 3.8.4> for further discussion.

24. Joseph M. Hendrie (Deputy Director for Technical Review, Director of Licensing, USAEC) to John A. Hinds, (Manager, Safety and Licensing, General Electric Company) letter of April 19, 1974, provides an appropriate standardized approach to MSL and MFL classification, which is acceptable as an alternate to the guidelines currently specified in <Regulatory Guide 1.29> (Revision 3/9/78).

25. Safety-related instrumentation and controls described in <Section 7.1>, <Section 7.2>, <Section 7.3>, <Section 7.4>, <Section 7.5>, and <Section 7.6> and safety-related instrumentation

TABLE 3.2-1 (Continued)

and controls for safety-related fluid systems will be subject to the pertinent requirements of the operational quality assurance program <Chapter 17>.

26. Diesel-backed power provided.

27. In accordance with <Regulatory Guide 1.97> (Revision 2) and clarified by <NUREG-0737>.

28. Will be subject to the pertinent provisions of <10 CFR 50, Appendix B> during the operations phase.

29. Fire control dampers, located in designated fire barriers, will be subject to the pertinent provisions of <10 CFR 50, Appendix B>, as discussed in the Perry Fire Protection Evaluation Report.

30. Nonsafety class instrumentation and controls specified for radiation or radioactivity monitoring will be subject to the pertinent provisions of <10 CFR 50, Appendix B>, during the operations phase.

31. The Seismic Category I classification is determined applicable based on the results of Perry plant's seismic clearance and anti-falldown inspection program <Section 3.2.1>.

32. Perry plant site fill is classified Seismic Category I when in place.

33. The Reactor Protection System (RPS) as described in <Section 7.2>, and the associated RPS power supplies as described in <Section 8.3.1.1.5>, are designed and will perform their intended safety function upon loss of power to either the control logic, or scram solenoids, or both. Although the C71 scram solenoid 120V ac power circuits are nonsafety-related, each group of RPS divisionalized circuits are routed in separate raceways to ensure adequate separation and independence as prescribed by the applicable General Electric design specifications.

34. The HPCS diesel engine skid-mounted piping and the associated components, such as valves, fabricated headers and fabricated special fittings are designed, manufactured and inspected in accordance with the guidelines and requirements of American National Standards Institute (ANSI) Standard B31.1, "Code for Pressure Piping;" ANSI Standard N45.2, "Quality Assurance Program Requirements for Nuclear Facilities;" and <10 CFR 50, Appendix B>.

TABLE 3.2-1 (Continued)

Design of the engine skid-mounted auxiliary system piping and components to the above standards is considered equivalent to a system designed to ASME Code, Section III, Class 3 requirements with regard to system functional operability and inservice reliability, provided that:

a.   A liquid penetrant examination for welds over 4 in. internal pipe size is performed.

b.   Diesel engine auxiliary system piping is hydrostatically tested to a minimum of 125% of design pressure.

35. The cooling coils and related piping of the controlled access area and miscellaneous equipment areas supply return units will maintain pressure integrity during a safe shutdown earthquake or an operating basis earthquake.

36. The ECCS suppression pool strainer (for RHR, LPCS, HPCS, and RCIC) is safety class 2, non ASME code. ASME II materials and IX welding does apply. Seismic Category I, <Section 3.9.2.2.3>, Item u. Quality group classification does not apply.

37. Pressure relief devices added to ASME piping systems as a result of <Generic Letter 96-06> "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions" are safety-related. <Generic Letter 89-09> "Replacement of ASME III Components" has been used to justify use of non-ASME safety-related components.

38. The auxiliary platform is Non-safety Related, Seismic Category I. Augmented Quality requirements apply to design, material and fabrication control of the load bearing structures.

39. The portions of the piping and valves in the ADHR system that are within the ASME boundaries of the RHR system (E12) and the LPCS system (E21) are safety related. The safety class and quality group of this safety related piping match those of interconnecting RHR and LPCS system piping (Safety Class 2/Quality Group B).

40. Those components located in the Turbine Building and Turbine Power Complex are Non-Safety related. Also, components located in the Off-Gas Building Exhaust Vent above roof elevation 660' are classified as Non-Safety Related.

41. The retired Unit 2 Shield Building and Auxiliary Building are Seismic Category I structures that provide structural design functions in support of Unit 1 operations <Section 3.2.1.1>. The Unit 2 Containment vessel is classified as Seismic Category II/I for its function to prevent structural failure of the Unit 2 Shield Building.

TABLE 3.2-2

QUALITY STANDARDS

Components	Group A	Group B	Group C	Group D
Pressure Vessels	ASME Section III Class I	ASME Section III Class 2	ASME Section III Class 3	ASME Section VIII Division 1
Piping	ASME Section III Class 1	ASME Section III Class 2	ASME Section III Class 3	ANSI B31.1.0 Power Piping
Pumps	ASME Section III Class 1	ASME Section III Class 2	ASME Section III Class 3	ASME Sec. VIII Division 1 See Note[1]
Valves	ASME Section III Class 1	ASME Section III Class 2	ASME Section III Class 3	ANSI B31.1
Atm. Storage Tanks	Not Applicable	ASME Section III Class 2	ASME Section III Class 3	API-650, AWWA D 100 or ANSI B96.1 or equivalent
0-15 psig Storage Tanks	Not Applicable	ASME Section III Class 2	ASME Section III Class 3	API 620 or equivalent

TABLE 3.2-2 (Continued)

Components	Group A	Group B	Group C	Group D
Heat Exchangers	ASME Section III Class 1 and TEMA-C	ASME Section III Class 2 and TEMA-C	ASME Section III Class 3 and TEMA-C	ASME Section VIII Division 1 and TEMA-C

NOTE:

[1] For pumps classified Group D and operating above 150 psi or 212°F, ASME, Section VIII, Div. 1 is used as a guide in calculating the wall thickness for pressure retaining parts and in sizing the cover bolting. For pumps operating below 150 psi and 212°F, manufacturers' standard pump for service intended is used.

TABLE 3.2-3

DESIGN REQUIREMENTS[1] FOR SAFETY CLASS 1
MECHANICAL SYSTEMS AND COMPONENTS

Design Condition Categories	Reactor Vessel	Pumps	Valves	Piping	Supports
NORMAL	ASME III NB3221 NB3222 See Note[2]	ASME III NB3400	ASME III NB3540	ASME III NB3640 & NB3652 NB3653	ASME III NF
UPSET	ASME III NB3221 NB3223 See Note[2]	ASME III NB3400	ASME III NB3500	ASME III NB3640 & NB3652 NB3654	ASME III NF
EMERGENCY	ASME III NB3221 NB3224 See Note[2]	ASME III NB3400	ASME III NB3500	ASME III NB3640 & NB3652 NB3655	ASME III NF
FAULTED	ASME III NB3221 NB3225 See Note[2]	ASME III NB3400	ASME III NB3500	ASME III NB3640 & NB3652 NB3656	ASME III NF

TABLE 3.2-3 (Continued)

NOTES:

(1) Design requirements are specified by three entries: the first entry is the code which contains the
design requirements, the second entry is the paragraph which describes the design requirements for the
particular component and the third entry specifies the paragraph containing the specific stress levels
for the component for the given condition of design.

(2) During RF07, a weld overlay was applied to the feedwater nozzle to safe-end weld 1B13-N4C-KB. The
overlay is designed as a full structural overlay in accordance with the recommendations of
<NUREG-0313>, Revision 2 (forwarded by <Generic Letter 88-01>), ASME Code Case N-504, and Section XI
of the ASME Boiler and Pressure Vessel Code, 1989 Edition (Paragraph IWB-3640), for which both normal
operating and emergency/faulted conditions were considered. Additionally, the overlay was sized to
meet the ASME Code Section III, 1989 edition, stress limits for primary local and bending stress and
secondary peak stress.

TABLE 3.2-4

DESIGN REQUIREMENTS[1] FOR SAFETY CLASS 2
MECHANICAL SYSTEMS AND COMPONENTS

Design Condition Categories	Vessels	Pumps	Valves	Piping	MECHANICAL Tanks Atmos- pheric	Tanks 0-15 psig	Heat[2] Exchangers	Supports	Duct- work and Valves	Fuel Assemb	REACTOR ASSEMBLY INTERNALS Core Support Structures	Reactor Internal Structures	Other Core Protection Components
NORMAL	ASME III NC3300 X[2]	ASME III NC3400 X	ASME III NC3500 X	ASME III NC3600	ASME III NC3800 X	ASME III NC3800 X	ASME III NC3300 X	X	X	X	See Note[3]	See Note[3]	X
UPSET	ASME III NC3300 X	ASME III NC3400 X	ASME III NC3500 X	ASME III NC3600	ASME III NC3800 X	ASME III NC3800 X	ASME III NC3300 X	X	X	X	See Note[3]	See Note[3]	X
EMERGENCY[4]	ASME III NC3300 X	ASME III NC3400 X	ASME III NC3500 X	ASME III NC3600	ASME III NC3800 X	ASME III NC3800 X	ASME III NC3300 X	X	X	X	See Note[3]	See Note[3]	X
FAULTED[4]	ASME III NC3300	ASME III NC3400	ASME III NC3500	ASME III NC3600	ASME III NC3800	ASME III NC3800	ASME III NC3300	X	X	X	See Note[3]	See Note[3]	X

NOTES:

[1] Design requirements are specified by three entries. The first entry is the code which contains the design requirements, the second is the paragraph which describes the design requirements for the particular component and the third entry specifies the paragraph containing the specific stress levels for the component for the given condition of design.

[2] X - Design requirements have not yet been developed by the applicable code.

[3] Design requirements for these components are contained in <Section 4.2.2>.

[4] Components which are loaded due to the direct effects of an emergency or faulted condition need only be designed for no loss of safety function. This means that stress and deformation limits similar to those for Safety Class 1 may be used.

TABLE 3.2-5

DESIGN REQUIREMENTS[1] FOR SAFETY CLASS 3 MECHANICAL SYSTEMS AND COMPONENTS

Design Condition Categories	Pressure	Pumps	Valves	Piping	Tanks Atmospheric	Tanks 0-15 psig	Heat[2] Exchangers	Supports	Duct work and Valves
NORMAL	ASME III ND3300 X[3]	ASME III ND3400 X	ASME III ND3500 X	ASME III ND3600 X	ASME III ND3800 X	ASME III ND3800 X	ASME III ND3300 X	X	X
UPSET	ASME III ND3300 X	ASME III ND3400 X	ASME III ND3500 X	ASME III ND3600 X	ASME III ND3800 X	ASME III ND3800 X	ASME III ND3300 X	X	X
EMERGENCY[4]	ASME III ND3300 X	ASME III ND3400 X	ASME III ND3500 X	ASME III ND3600 X	ASME III ND3800 X	ASME III ND3800 X	ASME III ND3300 X	X	X
FAULTED[4]	ASME III ND3300 X	ASME III ND3400 X	ASME III ND3500 X	ASME III ND3600 X	ASME III ND3800 X	ASME III ND3800 X	ASME III ND3300 X	X	X

TABLE 3.2-5 (Continued)

NOTES:

(1) Design requirements are specified by three entries. The first entry is the code which contains the design requirements, the second is the paragraph which describes the design requirements for the particular component and the third entry specifies the paragraph containing the specific stress levels for the component for the given condition of design.

(2) The rules of Tubular Exchanger Manufacturers Association, Class C, also apply.

(3) X-Design requirements have not yet been developed by the applicable code.

(4) Components which are loaded due to the direct effects of an emergency or faulted condition need only be designed for no loss of safety function. This means that stress and deformation limits similar to those for Safety Class 1 may be used.

TABLE 3.2-6

IEEE REQUIREMENTS FOR SAFETY CLASS 2 & 3 (CLASS 1E) ELECTRICAL SYSTEMS AND COMPONENTS

Systems	Components	RPS[2] Modules	RPS[2] Sensors	RPS[2] Systems[1]	Switch Gear	Transformers	Diesel	Motors	Valve Actuators
IEEE-308	IEEE-323	IEEE-323	IEEE-323	IEEE-279	IEEE-344	IEEE-344	IEEE-387	IEEE-323	IEEE-323
		IEEE-344	IEEE-344					IEEE-334	IEEE-344
								IEEE-344	

NOTES:

[1]   IEEE-279 shall apply only to those Safety Class 2 or 3 systems and components which actuate reactor trip or, in the event of a serious reactor accident, actuate engineered safety features.

[2]   RPS = Reactor Protection System

TABLE 3.2-7

CODE AND ADDENDA APPLICATION FOR REACTOR COOLANT
PRESSURE BOUNDARY SAFETY CLASS 1 EQUIPMENT[1]

Equipment	Master Part No.	Code and Addenda	Purchase Order Date
RECIRCULATION SYSTEM			
1. Gate valves			
a. Suction	B33-F023	ASME-Section III 1974 Edition (No Addendas)	10/9/74
b. Discharge	B33-F067	ASME-Section III 1974 Edition (No Addendas)	10/9/74
2. Flow control valves			
a. Body	B33-F060	ASME Section III 1971 Edition up to and including Winter 1973 Addenda	PO No. AG833 3/21/74
b. Topworks			AJ 530 11/20/76
3. Pump	B33-C001	ASME Section III 1974 Edition up to and including Winter 1974 Addenda	5/1/74
4. Piping	–	ASME Section III 1983 Edition up to and including Winter 1984 Addenda	3/5/75
MAIN STEAM SYSTEM			
1. Main steam isolation valves	B21-F022 B21-F028	ASME Section III 1974 Edition (No Addenda)[2]	12/10/74
2. Safety/relief valves	B21-F041 B21-F047 B21-F051	ASME B&PV Code Section III, 1974 Edition up to and including Summer 1976 Addenda	11/17/76

TABLE 3.2-7 (Continued)

Equipment	Master Part No.	Code and Addenda	Purchase Order Date
3. Piping	N/A	ASME Section III 1983 Edition up to and including Winter 1984 Addenda	3/75
HIGH PRESSURE CORE SPRAY SYSTEM			
Valves	E22-F004	ASME Section III Class 1, 1971 Edition up to and including Winter 1973 Addenda. Also used Code Case 1637.	8/16/74
REACTOR PRESSURE VESSEL	283x 218 CA B13-D003	ASME B&PV Code 1971 Edition with Addenda up to and including Winter 1972	3/23/73
		During RF07, a weld overlay was applied to the feedwater nozzle to safe-end weld 1B13-N4C-KB. The overlay is designed as a full structural overlay in accordance with recommendations of <NUREG-0313>, Revision 2 (forwarded by <Generic Letter 88-01>), ASME Code Case N-504, and Section XI of the ASME Boiler and Pressure Vessel Code, 1989 Edition (Paragraph IWB-3640).	N/A

NOTES:

[1] GE supplied

[2] The original ASME Code edition was invoked. The vendor was permitted to reconcile later Editions/Addenda as required by ASME Section XI IWA-7000.

3.3   WIND AND TORNADO LOADS

3.3.1   WIND LOADINGS

3.3.1.1   Design Wind Velocity

The basic wind speed used in the design of safety class structures is
90 mph at 30 feet above grade. This velocity is in agreement with
Figure 2 of ANSI A 58.1 (Reference 1) which gives 80-90 mph as the
fastest mile wind speed expected in a 100 year interval. The 90 mph is
higher than the predicted fastest mile wind from meteorological data
given in <Section 2.3>.

Increased wind velocities are considered in the case of wind being
channeled through the cooling towers and impinging on safety class
structures <Section 2.3>. The maximum postulated increased wind
velocity using the conservative approach of <Section 2.3> is 190 mph at
Elevation 150'-0" (top of safety class structures).

3.3.1.2   Determination of Applied Forces

Wind pressures corresponding to the basic wind speed are given by
Tables 5 and 6 of ANSI A 58.1 (Reference 1). The pressure values
indicated consider the effects of height, terrain and gusts.

For the postulated wind velocities due to cooling tower channeling, effective pressures are calculated by using:

$$q = 0.00256V^2$$

where:

V = Velocity in mph

q = Pressure in psf

No reduction of wind forces due to shielding is taken except for safety class structures which are immediately adjacent to each other.

The effective pressures are distributed on rectangular plane structures by the use of coefficients in accordance with Section 6.4, Section 6.5, and Section 6.6 of ANSI A 58.1 (Reference 1). Common coefficients used are:

a. Windward wall   0.8

b. Leeward wall   -0.6

c. Side wall   -0.5

d. Roofs   -0.7

e. Corners   -2.0

The effective pressures on the shield building are distributed in accordance with Table 4(f) of ASCE Paper 3269 (Reference 2) using a length over diameter ratio of 2. Pressure coefficients are given in <Table 3.3-1>.

The drag coefficient used in determining the total wind shear on the shield building is 0.45.

Calculated wind pressures are applied as equivalent static loads using the load factors and load combinations given in <Section 3.8>.

3.3.2 TORNADO LOADINGS

Tornadoes can be characterized as vortices possessing tangential, radial and translational velocities, whose net effect is a strong wind force which varies from a small value at the center to a maximum at the wall of the funnel cloud, and then decreases with increase in distance from the center. Three potential sources of damage exist: the wind pressure loading, the atmospheric pressure difference between the inside and outside of a building and the possible impact of wind generated missiles.

3.3.2.1 <u>Applicable Design Parameters</u>

3.3.2.1.1 Probability of Tornado Occurrence

Studies indicate that the probability of occurrence of any tornado at the plant site is very low. Page 408 of Reference 3 shows an isogram map of the United States giving lines of equal annual probability of tornado occurrence. This plant site would fall under a 5 in 10,000 annual probability of having a tornado occurrence. More recent studies reported in <Section 2.3> indicate an annual probability of 3.11 in 10,000. Not withstanding these low probabilities, the plant is designed assuming that a severe tornado will occur during its operational lifetime. See <Section 3.5.1.4.2.1> for annual probability of tornado strike used in the "TORMIS" analysis.

3.3.2.1.2 Design Basis Tornado

Based on the available damage information from tornadoes within a 50 mile radius of PNPP during the years 1951-1972, wind speeds of less than 260 mph are estimated for tornadoes <Section 2.3>.

The design basis tornado specified by <Regulatory Guide 1.76> for
Region I is used for Perry. The characteristics of this tornado are:

a.   Maximum wind speed   360 mph

b.   Rotational speed   290 mph

c.   Maximum translational speed   70 mph

d.   Minimum translational speed   5 mph

e.   Radius of maximum rotational speed   150 feet

f.   Maximum Pressure drop   3.0 psi

g.   Rate of pressure drop   2.0 psi/sec

The maximum wind speed is the sum of rotational and maximum
translational wind speed. The maximum pressure drop occurs in or near
the vortex center of a tornado.

3.3.2.2   Determination of Forces on Structures

The velocity pressure corresponding to the 360 mph wind velocity is
calculated using the equation in <Section 3.3.1.2> as follows:

$$q = 0.00256 \, V^2 = 0.00256 \, (360)^2$$
$$q = 332 \, psf$$

Effective velocity pressures on structures are calculated using the
following criteria:

a.   Horizontal wind distribution is used based on the fact that the
     rotational velocity of the tornado varies with the distance from

the vortex center in accordance with (Reference 3). The maximum pressure of 332 psf applies to small structures or components. Lower average pressures apply in determining total load on larger structures.

b.   Pressures are assumed to be constant at any elevation.

c.   Coefficients from Section 6.4, Section 6.5, and Section 6.6 of ANSI A 58.1 (Reference 1) are used for distributing the effective velocity pressure to walls and roofs of rectangular plan buildings.

d.   Tornado wind pressure is distributed on the shield building cylinder and dome using the coefficients for wind described in <Section 3.3.1>.

e.   Total tornado drag force on the shield building is determined by applying the drag coefficient of 0.45 <Section 3.3.1.2> to the average pressure for the building.

All Category I structures are considered to be non-vented and, therefore, are designed for a tornado pressure drop. ASCE Paper 7949 (Reference 3) states that the maximum pressure drop occurs at the center of the tornado vortex and diminishes with distance from the vortex center. Therefore, the maximum tornado rotational speed and maximum pressure drop are not coincidental. For design purposes, the following occurrences of tornado wind and atmospheric pressure loads are applied in determining tornado loads:

a.   Wind speed of 360 mph and pressure drop of 1.5 psi.

b.   Wind speed of 0 mph and maximum pressure drop of 3.0 psi.

Tornado wind forces on structures are treated as static loads. The
combination of the tornado wind loads with other applied loads
(including tornado missile loads) on safety class structures is
discussed in <Section 3.8>.

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

Failure of parts of structures not designed for tornado loads, such as
panels, piping or nonsafety class systems, will not impair safety class
structures that have been designed for tornado missiles. Safety class
structures are designed for a wide range of postulated missiles as
stated in <Section 3.5>.

Total failure of a nonsafety class building, i.e., completely collapsing
and falling against a safety class building structure, is precluded by
design. The design approach used for the service building and the
turbine buildings is based on a building cladding that blows off at a
predictable wind load, i.e., considerably below maximum tornado wind
pressures. Therefore, the critical load on these buildings is the
tornado wind pressures on the structural skeleton of the building. The
structural components of these buildings, including the foundations, are
designed for these tornado wind pressures. A somewhat different
approach is used on the nonsafety portion of the radwaste building,
which is a steel framed, steel cladded annex to the radwaste building.
Blow-off cladding is not used on this nonsafety portion and full tornado
pressures are assumed in the design. The nonsafety portion is braced in
the north-south direction to withstand these loads. Tornado loads on
the nonsafety portion in the east-west direction are transmitted to the
safety class portion of the radwaste building through structural ties.
These loads are considered in the design of the safety class portion of
the radwaste building. The allowable stresses for tornado loadings on
the service building, turbine building and the nonsafety portion of the
radwaste building are specified in <Section 3.8.4>.

3.3.3   REFERENCES FOR SECTION 3.3

1.   American National Standards Institute, "Building Code Requirements
      for Minimum Design Loads in Buildings and Other Structures,"
      ANSI A 58.1, 1972.

2.   ASCE Paper No. 3269, "Wind Forces on Structures," American Society
      of Civil Engineers, <u>Transactions</u>, Vol. 126, Part II, 1961.

3.   ASCE Paper No. 7949, "Tornado Design Considerations," <u>Journal,
      Power Division</u>, American Society of Civil Engineers, March 1971.

TABLE 3.3-1

<u>SHIELD BUILDING PRESSURE COEFFICIENTS</u>

Angle From Wind Direction	Pressure Coefficient
0°	1.0
15°	0.8
30°	0.1
45°	-0.7
60°	-1.2
75°	-1.6
90°	-1.7
105°	-1.2
120°	-0.7
135°	-0.5
150°	-0.4
165°	-0.4
180°	-0.4

3.4   WATER LEVEL (FLOOD) DESIGN

3.4.1   FLOOD PROTECTION

3.4.1.1   Flood Protection Measures for Seismic Category I Structures

<Section 2.4> describes the design flood and the grading of the site
around the safety class structures, and substantiates that these
structures will not be subjected to water levels greater than 6 inches
above the finished plant grade of about Elevation 620'-0".

The portions of land safety class structures located below finished
grade are protected on their outside surfaces by a continuous
waterproofing membrane. Waterstops are provided at construction joints.
To ensure shear transfer from the foundation media to the reinforced
concrete building foundations, the shear strength of the waterproofing
membrane is greater than that required for the applicable loading
conditions.

In the unlikely event that the waterproofing of the structures is
insufficient, additional flood protection for safety class components,
equipment and systems located below grade is provided; this is
accomplished by floors that slope to floor drains. Details of the floor
drain systems are discussed in <Section 9.3.3>.

3.4.1.2   Permanent Dewatering System

A permanent underdrain system is constructed under the main plant area
as discussed in <Section 2.4.13.5>. This system ensures that the
groundwater elevation will not exceed Elevation 590'-0". All the safety
class structures in the main plant area are designed to withstand
groundwater to Elevation 590'-0".

A more detailed discussion of the flooding possibilities and consequences of a circulating water yard pipe break, or an expansion joint rupture within the turbine building via flow through a base mat fracture, is presented in <Section 2.4.13.5> and <Section 10.4.5>.

The design criteria for ensuring the prevention of damage to safety-related equipment and systems by internal flooding due to the failure of non-Category I components and piping are:

a.   The plant layout uses separation of Seismic Category I and non-Seismic Category I components by locating them, to the maximum extent possible, in separate buildings.

b.   The ECCS equipment is located in separate, water tight compartments.

c.   Small leaks will be handled by the floor drain system.

A review of the layout of systems and components was performed to ensure that Items a and b, above, have accomplished the prevention of flooding damage. Details of that analysis are discussed in <Section 3.6.2>.

The flood elevation pertaining to land structures is not a factor since the site is graded to carry all surface water away from the safety class structures. Refer to <Figure 2.4-3> for details of plant grading and the storm drainage system. The site storm drainage system and plant grading are designed to preclude ponding of water greater than six inches above the plant grades shown. The plant underdrain system discussed in <Section 2.4.13.5> will maintain a groundwater level below Elevation 568'-6" and will ensure that groundwater level will not exceed Elevation 590'-0".

The maximum recorded Lake Erie Elevation of 575.4' (USGS), which occurred in June 1973, was used in the design of the following submerged safety class structures:

a.   Cooling water tunnels.

b.   Offshore intake structures.

c.   Offshore discharge structure.

3.4.2   ANALYTICAL AND TEST PROCEDURES

The Safety Class I structures will not be subjected to flood current, wind wave, hurricane, tsunami, seiche, or dynamic water force. Refer to <Section 2.4> for a description of the site hydrology. The submerged offshore structures noted in <Section 3.4.1.2> were designed for the effect of dynamic wave action. Consideration of pressure oscillations in the cooling water tunnel system resulting from waves, as described in <Section 2.4.5>, shows that the oscillations are insignificant.

To arrive at a range of wave heights to be used in determining the dynamic forces on the submerged intake structures, wind speeds generated by the probable maximum storm (PMS) <Section 2.4.5> were applied from different directions to give different fetches. The characteristics of waves approaching the structures are affected by wave generation, refraction and shoaling. These factors are in turn dependent on the water depth so that the seiche at the site, which is coincident with the waves, is important. <Section 2.4.5> indicates that the waves generated by the PMS have a significant height of 17 feet (the corresponding H_{max} = 30 ft). The 31.2-foot depth of water (at maximum stillwater level of 580.5 feet) near the intake structures, limits the wave height to 24.3 feet. All larger waves will break (based on wave breaking criteria of 0.78 of the depth).

Alternately, when the lake is at the recorded low monthly mean level of 569.3' (USGS), resulting in a 20.0 foot average water depth at the intakes, waves higher than 15.6 feet will break. For these reasons, the applicable dynamic forces for the design of the structures are due to the incipient breaking wave height corresponding to water depths between 20.0 feet (15.6 foot wave height) and 31.2 feet (24.3 foot waves). Since the intakes are axisymmetric, the direction of wave approach is not important. Aspects of the wave force analysis are discussed in <Section 2.4.5>.

In addition to the wave loads, the offshore intake and discharge structures are designed for ice loads as discussed in <Section 3.8.4>. These loads are several times larger in magnitude than the loads produced by wave action. Therefore, the results of wave loads were not critical in the structural design of the offshore structures.

3.4.3   FLOOD FORCE APPLICATION

The land safety class structures are not designed for flood force application, since the grading and topography of the site precludes the development of floods.

3.5   MISSILE PROTECTION

3.5.1   MISSILE SELECTION AND DESCRIPTION

3.5.1.1   Internally Generated Missiles (Outside Containment)

3.5.1.1.1   Criteria

The following criteria have been adopted to assess the plant's
capability to ensure that, if a generated missile of any postulated type
occurs, there is:

a.   No loss of containment function.

b.   No direct loss of reactor coolant.

c.   No loss of function (assuming offsite power is not available during
     the shutdown of the plant) to systems required to shut down the
     reactor and maintain it in a safe shutdown condition, or mitigate
     the consequences of the missile damage, thereby ensuring:

     1.   No equipment will be damaged in one safety-related division,
          e.g., Division 1, from internally generated missiles
          originating from another safety-related division, e.g.,
          Division 2.

     2.   No damage will occur to any safe shutdown equipment by
          missiles generated from nonsafety-related equipment.

d.   No offsite exposure will exceed the guidelines of <10 CFR 100> or
     <10 CFR 50.67> (future revisions to design basis analyses that
     compare consequences to 10 CFR 100 will be updated to
     <10 CFR 50.67>.

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

No failures to Class 1E sensors to the reactor protection system (RPS) located on the turbine control valve and stop valve will prevent the reactor from being safely shutdown since other RPS sensors (high pressure scram or high flux scram) located in safety-related buildings provide sufficient backup. Therefore, these sensors are not protected from missile hazards inside the turbine building.

Systems to be protected from missiles are listed in <Table 3.5-1>, along with a cross reference to the USAR section that describes the items and identifies the applicable drawings or piping and instrumentation diagrams. System and component seismic category and quality group classification are listed in <Table 3.2-1>.

3.5.1.1.2   Missile Selection

Rotating and high pressure (greater than 275 psig) system components located inside the following buildings and areas were examined to identify and classify potential missiles:

a.   Auxiliary building.

b.   Intermediate building.

c.   Control building.

d.   Steam tunnel.

e.   Diesel generator building.

f.   Emergency service water pumphouse.

g.   Radwaste building.

Pumps located outside the containment were evaluated for missiles associated with potential failure due to overspeed. The maximum no-load speed of these pumps is equivalent to associated motor no-load operating speed. No pipe break or other single failure in a pump suction line results in pump speeds exceeding the no-load speed. Pump casings are designed to contain impeller fragments should an impeller fail. Therefore, missiles generated by pumps outside containment are not postulated. The RCIC drive turbine is not a credible source of missiles. It is provided with mechanical overspeed protection as well as automatic governing; very extensive industrial and nuclear experience with this model of turbine has never resulted in a missile which penetrated the turbine casing.

Valves of ANSI 900 psig rating and above, constructed in accordance with Section III of the ASME Code, are pressure seal bonnet type valves. Valve bonnets 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. Bonnet ejection is highly improbable because of the very conservative design of the retaining rings of these valves. Therefore, bonnets are not considered credible missiles.

Most valves of ANSI rating 600 psig and below are valves with bolted bonnets. Valves that are two inch and smaller are valves with screwed 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 Code, Section III, and by designing flanges in accordance with applicable code requirements. Even if bolt failure occurs, the likelihood of all bolts experiencing simultaneous complete severance failure is remote. The widespread use of valves with bolted bonnets, and the low historical incidence of complete severance failure of bonnets, confirm that bolted valve bonnets need not be considered as credible missiles.

Valve stems are not considered potential missiles if at least one feature, in addition to the stem threads, is included in the design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air or motor-operated valve stems are effectively restrained by the valve operators.

Nuts, bolts, nut and bolt combinations, and nut and stud combinations have little stored energy and are of no concern as potential missiles.

Temperature or other detectors installed on piping or in wells are evaluated as potential missiles if a single circumferential weld failure would cause their ejection.

Thicknesses of barriers, including walls, slabs and specially designed barriers, which protect safety class equipment or systems, satisfy the criteria discussed in <Section 3.5.3>. Therefore, scabbing or the generation of secondary missiles from the non-impacted face of such a barrier is precluded. Concrete fragments ejected from the impacted face (spalling effect), if any, have energies too low for consideration as missiles due to the small weight and velocity of such fragments. Fragments and the initial missile constitute no threat to safety class equipment or systems as secondary missiles during the drop following impact.

3.5.1.1.3   Missile Protection Methods

Protection of safety-related systems and components from postulated missiles is accomplished by one or more of the following methods:

a.   Compartmentalization

     Equipment is enclosed in missile protected compartments.

b.   Barriers

     Barriers are erected to stop missiles either at the source or at
     equipment locations.

c.   Separation

     Redundant components of vital systems are separated by one or a
     combination of the following methods:

     1.   Components are located within separate cubicles.

     2.   Adequate spatial separation between redundant components and
          electric circuits.

d.   Equipment Design

     Structures or components can, by virtue of design, withstand impact
     of postulated missiles without loss of function.

e.   Strategic Orientation

     Equipment or components are so oriented that postulated missile
     paths are directed away from equipment and components requiring
     protection.

f.   Distance

     Equipment is located out of range of postulated missiles.

Safety-related instrument and control components and instrument pulse
lines outside the reactor building, which are required for safe plant
shutdown, are not in the paths of postulated missiles.

<Table 3.5-2> lists all postulated missiles outside containment and summarizes the method used to protect any of the nearby structures, systems or components listed in <Table 3.5-1>.

3.5.1.2   Internally Generated Missiles (Inside Containment)

3.5.1.2.1   Criteria

The criteria for internally generated missiles, inside containment, are identical to those listed in <Section 3.5.1.1.1>, Items a through f, for internally generated missiles, outside containment.

Systems to be protected from missiles are listed in <Table 3.5-1>, along with the USAR section that describes the items and references the applicable drawings or piping and instrumentation diagrams. System and component location, seismic category and quality group classification are listed in <Table 3.2-1>.

3.5.1.2.2   Missile Selection

Rotating and high pressure (greater than 275 psig) system components located inside containment were examined to identify and classify potential missiles.

The most substantial piece of NSSS rotating equipment is the recirculation pump and motor. This potential missile source is discussed in the report, "Analysis of Recirculation Pump Under Accident Conditions," (Reference 1). It is concluded in this report that destructive pump overspeed can result in certain types of missiles. A careful examination of shaft and coupling failures shows that the fragments will not result in damage to the containment or to vital equipment.

a. Missiles from motor

Missiles may be created at excessive motor speeds through failure
of the end structure of the rotor. The structure consists of the
retaining ring, the end ring and the fans. Missiles potentially
generated in this manner will strike the overhanging ends of the
stator coils, the stator coil bracing, support structures, and two
walls of one-half inch thick steel plate. Due to the ability of
these structures to absorb energy, it is concluded that missiles
would not escape this structure. At this point, frictional forces
would tend to bring the overspeed sequence to a stop.

In the postulated event that the body of the rotor were to burst,
missiles could be created. The likelihood that these missiles
would escape the motor is less than the likelihood of escape for
the missiles described above; this is due to the additional amount
of material constraining missile escape, such as the stator coil,
field coils and stator frame directly adjacent to the rotor.

Thus, it is concluded that a decoupler is not needed between the
pump and the motor.

Since pump-to-motor hold down bolting is capable of carrying
greater torque loads than the pump shaft, bolt failure is
precluded. Because pump shaft failure decouples the rotor for the
overspeed driving blowdown force, only those cases with peak torque
less than that required to fail the pump shaft (five times rated)
will have the capability to drive the motor to overspeed. When
missile generation probabilities are considered along with a
discussion of the actual load bearing capabilities of the system,
it is evident that these considerations support the conclusion that
it is unrealistic that the motor would become a missile.

b.   Missiles from pump impeller

Overspeed of the recirculation pump due to a LOCA can result in
certain types of postulated missiles consisting of pump impeller
fragments. However, in (Reference 1) it was concluded that the
fragments will not penetrate the pump case.

Concerning other missiles, the control rod drive (CRD) mechanism is
located inside of the pedestal. The pedestal is a composite steel and
concrete structure which provides support for the reactor pressure
vessel. The pedestal consists of two concentric steel liners tied
together by diaphragms with the annulus area filled with concrete. A
support structure for the control rod drive mechanism is located inside
of the inner steel shell of the pedestal. This structure consists of
supporting bars, grid plates and beams.

In the event of a postulated failure of the CRD housing, the CRD housing
would be ejected downward toward the floor of the pedestal when the
housing has moved approximately one inch; it would then contact the
support structure and be stopped. Since the CRD housings are located
within the pedestal and are further restrained by the support structure,
they cannot be considered as credible missiles.

Valves of ANSI 900 psig rating and above, constructed in accordance with
Section III of the ASME Code, are pressure seal bonnet type valves.
Valve bonnets 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.

3.5.1.2.3   Missile Protection Methods

Protection of safety-related systems and components from postulated missiles is accomplished by one or more of the following methods:

a.   Compartmentalization

     Equipment is enclosed in missile protected compartments.

b.   Barriers

     Barriers are erected to stop missiles either at the source or at equipment locations.

c.   Separation

     Redundant components of vital systems are separated by one or a combination of the following methods:

     1.   Components are located within separate cubicles.

     2.   Adequate spatial separation between redundant components and electric circuits.

d.   Equipment Design

     Structures or components can, by virtue of design, withstand impact of postulated missiles without loss of function.

e.   Strategic Orientation

     Equipment or components are so oriented that postulated missile paths are directed away from equipment and components requiring protection.

f.   Distance

     Equipment is located out of range of postulated missiles.

Safety-related instrument and control components and instrument impulse
lines inside the reactor building, which are required for safe plant
shutdown, are not in the paths of postulated missiles.

<Table 3.5-3> lists all postulated missiles inside containment and
summarizes the method used to protect any of the nearby structures,
systems or components listed in <Table 3.5-1>.

3.5.1.2.4   Evaluation of Potential Gravitational Missiles Inside
               Containment

Gravitational missiles inside the containment have been considered as
follows:

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

Non-seismic items and systems inside containment are classified as
follows:

a.   Cable Tray

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

b.   Conduit and Nonsafety Pipe

     Class 1E conduit and associated hardware are seismically designed
     and supported. Non-Class 1E conduit and associated raceways, and
     nonsafety class pipe are seismically designed and supported where
     these systems are a potential hazard to safety-related equipment.

c.   Equipment for Maintenance

All other equipment, such as hoists, required during maintenance
will either be removed during operation, moved to a location where
it is not a potential hazard to safety-related equipment or
seismically restrained to prevent it from becoming a missile.

3.5.1.3   Turbine Missiles

Turbine missiles are discussed in (Reference 2) and (Reference 11).
Additional information can be found in <Section 10.2.3>.

3.5.1.4   Missiles Generated by Natural Phenomena

3.5.1.4.1   Missile Selection

<Table 3.5-4> lists potential tornado missiles with corresponding design
parameters considered in the plant design. This table represents the
spectrum of potential missiles which would be generated on or near the
site by the design basis tornado described in <Section 3.3>.

Missile selection is based on the potential for the element to exist on
or near the site and the potential for the element to be lifted and
accelerated by the tornado wind. Each of the missiles listed is capable
of being lifted off the ground using an aerodynamic lift force for an
assumed 0.2 second duration and the force of the vertical wind. Once
aloft, the vertical tornado wind continues to act. Design velocity for
the missile is determined by considering the force of the tornado's
tangential wind on the element for the period of time during which the
element is maintained aloft. Maximum velocity is limited to that which
would cause the element to exit from the tornado. The wood plank, eight
inch wood pole, and the two automobile missiles reach exit velocity,
while the remaining missiles cannot be sustained aloft for a sufficient
time to achieve exit velocity. Further discussion of tornado missiles
is contained in (Reference 3).

As discussed in <Section 3.4>, flood missiles are not a consideration. Therefore, tornado generated missiles are considered as the limiting natural phenomena hazard in the design of all structures which are required for ensuring the integrity of the reactor coolant pressure boundary, ensuring the capability to shutdown the reactor and maintain it in a safe shutdown condition, and those whose failure could lead to radioactive releases which would exceed offsite radiation exposure limits [25% of <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>) guideline exposures], as discussed in <Regulatory Guide 1.117>.

3.5.1.4.2   Missile Protection Methods

System and component safety classification and seismic category are given in <Table 3.2-1>. Specific location within the building is provided by the layout drawings <Figure 1.2-2>, <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>, <Figure 1.2-11>, <Figure 1.2-12>, <Figure 1.2-13>, <Figure 1.2-14>, <Figure 1.2-15>, <Figure 1.2-16>, and <Figure 1.2-17>. Those systems or components listed in <Table 3.2-1> that are required to ensure the integrity of the reactor coolant pressure boundary, maintain safe shutdown conditions or prevent release of radiation which would exceed offsite radiation exposure limits, are provided with tornado missile protection by location within Seismic Category I structures, by unique missile barriers, by the shielding of an adjacent Seismic Category I structure, or, they have been analyzed as discussed in <Section 3.5.1.4.2.1>.

<Table 3.2-1> also identifies Seismic Category I structures. The exterior walls and roof of these structures are required, by definition, to withstand the effects of the design basis tornado including tornado

missiles. These elements are two foot thick (minimum) reinforced concrete having a 28 day compressive strength of 3,000 psi. Design approach is discussed in <Section 3.5.3>. Systems or components located wholly within these structures are considered protected from external

missiles unless they are located near building penetrations.
Penetrations in the exterior structural elements of these structures:

- are shielded due to adjacent Seismic Category I structures,
- were strategically located to preclude damage to any safety-related
  system (if a missile entered) due to internal walls and slabs which
  provide shielding,
- were provided with unique missile barriers, or
- were evaluated as discussed in <Section 3.5.1.4.2.1>.

<Table 3.5-5> lists the barriers which were provided to protect
penetrations in Seismic Category I structures.

Safety-related systems and components which are located outside of
Seismic Category I structures are provided with unique missile barriers,
or were evaluated as described in <Section 3.5.1.4.2.1>. <Table 3.5-5>
also lists the barriers which were provided to protect systems and
components which are located outside of Seismic Category I structures.

A unique situation occurs for the condensate storage tank. The
500,000 gallon steel tank is a non-seismic, nonsafety design. However,
to prevent an uncontrolled release of radioactive liquid the tank is
located within a Seismic Category I concrete dike. Missile impact has
been considered in the design of the two foot thick (3,000 psi) concrete
dike.

3.5.1.4.2.1   Systems/components not requiring unique tornado missile
               protection

A limited amount of safety-related systems and components located near
penetrations in Seismic Category I structures or located outside of such

structures are evaluated as not requiring unique tornado missile protection barriers. Two approaches were used in this evaluation:

a. Certain safety-related systems and components are screened out using the criteria of <Regulatory Guide 1.117> "Tornado Design Classification", including its Appendix, which together detail the items that should be protected from the effects of tornadoes. The criteria in the Regulatory Guide are summarized as "important" systems and components required to ensure the integrity of the reactor coolant pressure boundary; ensure the capability to shut down the reactor and maintain it in a safe shutdown condition; and those whose failure could lead to radioactive releases resulting in calculated offsite exposures greater than 25% of the guideline exposures of <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>) using appropriately conservative analytical methods and assumptions. The safety-related systems and components not required to support these <Regulatory Guide 1.117> guidelines are evaluated as not requiring unique tornado missile protection.

b. "Important" systems and components (as discussed in <Regulatory Guide 1.117>) are generally protected. The limited amount of unprotected portions of important systems and components are analyzed using a probabilistic missile strike analysis as permitted in Standard Review Plan 3.5.1.4 "Missiles Generated by Natural Phenomena". This analysis is conducted to determine the total (cumulative) probability per year of missiles striking important structures, systems and components due to postulated tornadoes. This information is then utilized to determine the specific design provisions that must be provided to maintain the estimate of strike probability below an allowable level.

The allowable level established for the protection of such systems
and components at PNPP is consistent with the acceptance criteria
in Standard Review Plan 2.2.3 "Evaluation of Potential Accidents",

i.e., that a probability of occurrence of initiating events [those that could lead to potential consequences in excess of <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>) guidelines] of "approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower." The PNPP-specific acceptance criteria is that the total probability of tornado missiles simply striking an important system or component must be shown by analysis to be less than 10^{-6} per year. This PNPP-specific criteria contains the following inherent conservatisms:

- It is assumed that an important system or component simply being struck by a tornado missile will result in damage sufficient to preclude it from performing its intended safety function, although this is not realistic for all cases.
- It is assumed that the damage to the important system or component results in damage to fuel sufficient to result in conservatively calculated radiological release values in excess of the <10 CFR 50.67> guidelines, although this is not realistic for all cases.
- There are no missiles that can directly impact on irradiated fuel, even on the spent fuel stored in the Fuel Handling Area of the Intermediate Building. Any missiles postulated to enter this area either miss the pools entirely, are stopped by internal walls, or strike the far side of the pool above the level of the fuel.
- While residing in the FHB rail bay, the HI-STORM and HI-TRAC transfer cask with spent fuel are vulnerable to a tornado missile strike due to the FHB roll up door, which is identified as not being an adequate barrier against a tornado

missile per USAR <Table 3.5-6>. The HI-TRAC transfer cask and HI-STORM storage cask have been evaluated for tornado missiles while being located on the ISFSI Pad, (Reference 12) and (Reference 13). Consequently, the analysis for overturning of the HI-TRAC/HI-STORM stacked cask and HI-TRAC located on the Decon Pad configurations due to a potential tornado missile impact are required. PNPP (Reference 14) analyzed the configurations and concluded the casks are stable in the event of a potential impact by a PNPP design basis tornado missile.

The analysis uses an NRC approved (Reference 9) methodology developed by the Electric Power Research Institute (EPRI) (Reference 10). The methodology is implemented using the computer program TORMIS, which is further described in <Section 3.5.1.4.2.1.1>.

Should PNPP evaluations using the TORMIS methodology provide results indicating that the plant configuration equals or exceeds PNPP's 10^{-6} acceptance criteria, then unique protective barriers will be utilized to reduce the total probability value to below the acceptance criteria. Temporary removal of a protective feature is permitted under administrative controls, if removal is determined to be necessary.

The unprotected "important" systems and components are listed in <Table 3.5-8>. These systems/components have been analyzed by the TORMIS program, and it was determined that no unique tornado missile protection is required.

3.5.1.4.2.1.1 TORMIS Description

TORMIS implements a methodology developed by the Electric Power Research Institute (Reference 10). TORMIS determines the probability of striking walls and roofs of buildings on which penetrations or exposed portions of systems/components are located. The probability is calculated by simulating a large number of tornado strike events at the site for each tornado wind speed intensity scale. After the probability of striking the walls or roof is calculated, the exposed surface area of the particular components are factored in to compute the probability of striking a particular item.

The TORMIS analysis for the PNPP site is in accordance with the TORMIS program, as described in (Reference 10), except for the following deviations:

1.   As described in USAR <Section 2.3.1.2.2>, the probability of a tornado strike at PNPP is 3.11×10^{-4} per year based on local analysis. As part of the PNPP analysis, the annual probability of a tornado will be determined for the wind speeds in item 2 below, using regional data available in TORMIS for NRC Region I. This

value will be compared to the 3.11×10^{-4} value from
<Section 2.3.1.2.2>. The more conservative of these two values
will be compared to the occurrence rate for the PNPP area provided
in Reference 11, and the most conservative value will be utilized
in the PNPP analysis.

2. The Fujita scale (F-scale) wind speeds will be used in lieu of the
   TORMIS wind speeds (F'-scale) for the F0 through F5 intensities.
   In addition, a wind speed range from 313 to 360 mph will be used
   for the F6 intensity to correspond to the tornado wind speed
   described in <Section 3.3.2.1.2>.

3. The PNPP analysis addresses the TORMIS reduction in tornado wind
   speed near the ground due to surface friction by injecting the
   potential tornado missiles into the tornado wind field at
   elevations above the surface of the ground. The increased
   injection height will increase the wind speed acting on the
   missile. The height of injection will ensure that the missiles are
   subjected to wind speeds greater than or equal to 246 MPH (a V_0/V_{33}
   value of 0.82).

4. The number of missiles used in the PNPP TORMIS analysis is a
   conservative value for site specific sources, such as laydown,
   parking, and warehouse areas. These are postulated by general
   walkdown information at PNPP.

See <Section 3.5.2> for additional considerations of protection from
external missiles.

3.5.1.5   Missiles Generated by Events Near the Site

Due to the considerable distance between the potential sources of
missiles from accidental explosions in the vicinity of the site to the
safety-related structures on the site, no credible events can be
postulated to occur.

3.5.1.6   Aircraft Hazards

The approach and methodology outlined in Standard Review Plan 3.5.1.6,
"Aircraft Hazards," have been used in the calculation of the probability
of an aircraft crash into the effective plant areas of the Perry Nuclear
Power Plant. The aircraft handling facilities and air routes are
described in <Section 2.2.2.5>. The aircraft hazard probability
developed from the total probability of an aircraft crash into the
effective areas of the plant does not constitute a design basis event.
The calculation of probabilities is based on the following:

a.   Three federal airways pass within two miles of the plant
     site: V-188 and V-10 are low altitude airways and are primarily
     used by general aviation aircraft; J-584 is a high altitude airway
     and is primarily used by commercial aviation. No data on the
     annual traffic for these airways was readily available, so the peak
     daily traffic was assumed to exist throughout the year. This
     results in a maximum annual usage of 4,745, 5,475 and
     34,310 flights for airways V-188, V-10 and J-584, respectively.

b.   No airports are located within five miles of the plant center. One
     small private airstrip is located 4.5 miles ESE of the plant. Five
     single engine propeller driven planes may be based there. Average
     operations are estimated at one flight per week.

c.   No airports having more than $500d^2$ movements per year are located
     within 10 miles of the site and no airports having more than
     $1,000d^2$ movements per year are located beyond 10 miles of the site
     (d = the distance from the site in miles).

d.   No military airports are within five miles of the site. The only
     other airspace usage in the plant vicinity is a secondary backup
     holding pattern that is only infrequently used by commercial
     aircraft. Its closest approach to the plant site is greater than

10 miles. The probability of air crash into effective plant areas due to the location and usage of this holding pattern is extremely small.

Utilizing the above data, the total probability of an aircraft crash into the plant was determined to be 1.44×10^{-6} crashes per year. This probability was developed conservatively and the realistic probability can be shown to be substantially smaller.

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

Structures, systems and components requiring protection from externally generated missiles are listed in <Table 3.2-1>. The general approach of locating safety systems within Seismic Category I structures is discussed in <Section 3.5.1.4>.

Penetrations in exterior walls or roofs of Seismic Category I structures create the potential for a missile to enter and contact safety-related equipment inside. Therefore, special shield barriers have been used, where required, to prevent this possibility. Penetrations between adjacent safety-related structures are considered shielded. Unique missile shields on penetrations and the associated system and/or component are listed in <Table 3.5-5>. Barrier design procedures are discussed in <Section 3.5.3>.

The remaining penetrations have been determined as not requiring a special shield for one or more of the following reasons:

a.   Internal walls and slabs create compartments which shield safety-related systems from missiles.

b.   Structural walls and slabs in conjunction with equipment location
     create a labyrinth effect which precludes missile contact with the
     equipment.

c.   Adjacent structures prevent missile incidence at an angle which
     would permit contact with safety-related components.

d.   Certain penetrations have been evaluated for tornado generated
     missiles as discussed in <Section 3.5.1.4.2.1>.

Penetrations which have not been provided with unique missile shields
are listed in <Table 3.5-6>, along with the method considered for
protection of safety-related systems. Walls and slabs considered as
missile shields are two foot thick (minimum) reinforced concrete with
minimum 3,000 psi compressive strength at 28 days, unless noted
otherwise. Barrier design approach is discussed in <Section 3.5.3>.

Systems or components located outside Seismic Category I structures and
not provided with a unique concrete missile shield (see also
<Table 3.5-5>) are identified in <Table 3.5-7>, along with the method of
providing missile protection.

3.5.3   BARRIER DESIGN PROCEDURES

The exposed walls and roofs of Seismic Category I structures have a
minimum concrete thickness of 24 inches and are reinforced each way on
each face with a minimum of No. 8 bars at 12 inch center-to-center for
walls, and No. 9 bars at 12 inch center-to-center for slabs. Typical
elements were evaluated for local effects of penetration, perforation
and scabbing as well as overall structural response.

Local effects were determined to be of a noncritical nature by
comparison of the Perry missile spectrum and Perry missile barriers to
full scale test results (Reference 4) (Reference 5). Overall structural

response was evaluated for typical concrete structural elements by demonstrating that actual design loads as established by design loading combinations. The pipe missile impact force is estimated from full scale test results (Reference 4) to be 350 kips applied as a triangular pulse (Reference 6). The wooden and automobile missiles were considered as deformable missiles since the wood pole splinters and the automobile crushes as evidenced by full scale tests (Reference 4) (Reference 5). Peak impact force was imparted by the automobile as determined by the equation: (Reference 8)

$$P_B + \mu V^2$$

where:

P_B = crushing strength of the missile (uncrushed mass times deceleration)

μ = mass density per unit length

V = residual velocity at instant being considered

For a rectangular pulse (Reference 6) and limiting ductility ratio of 10, (Reference 7) required target strength was 420 kips. Barriers having the minimum thickness with equal positive and negative reinforcing were evaluated using a limiting ductility ratio of 10 for flexural elements. Capacity of the Perry missile barrier and its supporting elements was sufficient to resist the assumed loadings.

The lightest steel column section supporting a missile barrier was evaluated for a peak impact load of 640 kips based on a ductility factor of 1.3. The section was determined adequate to resist the assumed load in combination with design dead and live loads supported.

Seismic events and tornado generation are assumed not to occur simultaneously. Each safety class building is designed to withstand seismic loading which is more severe than tornado wind plus missile impact loads; therefore, structures are sufficient to resist the tornado loading.

Interior structural elements used as barriers to contain internally generated missiles are also of reinforced concrete of the same thickness and minimum reinforcement as exterior walls and roofs. Internally generated missiles are lighter than, and travel at lower velocities than, the postulated tornado missiles. Thus, the internal barriers are adequate to serve the intended purpose of protecting safety class equipment.

3.5.4   REFERENCES FOR SECTION 3.5

1.   Letter, Robase (GE) to Vassallo (NRC) on "GE Recirculation Pump Potential Overspeed," March 1979, with GE report, "Analysis of Recirculation Pump Under Accident Conditions," Rev. 2, March 1979.

2.   Gilbert Associates, Inc., "An Analysis Of Low Trajectory Turbine Missile Hazards, Perry Nuclear Power Plant, Units 1 and 2," GAI Report No. 1848, October 1976.

3.   Gilbert Associates, Inc., Topical Report, "Design Parameters for Tornado Generated Missiles," GAI-TR-102, January 1974, revised January 1975.

4.   "Full-Scale Tornado-Missile Impact Tests;" Sundia Laboratories Report SAND 77-1166, July 1977.

5.   Vassallo, F. A., "Missile Impact Testing of Reinforced Concrete Panels;" Calspan Report No. HC-5609-D-1.

6. McMahon, P. M., Sen, S. K. Meyers, B. L., and Buchert, K. P. (Bechtel Power Corporation), "Structural Response of Concrete Walls Impacted by Tornado Missiles."

7. Biggs, J. M., "Introduction to Structural Dynamics;" McGraw-Hill, 1964.

8. Haley, J. L., Jr., and Turnbow, J. W., "Total Reaction Force Due to an Aircraft Impact into a Rigid Barrier," Final Report AVSER 68-3, 29 April 68, Dynamic Science, The AVSER Facility, Phoenix, Arizona.

9. Letter, Rubenstein (NRC) to Miraglia (NRC) entitled "Safety Evaluation Report - Electric Power Research Institute (EPRI) Topical Reports Concerning Tornado Missile Probabilistic Risk Assessment (PRA) Methodology", dated October 26, 1983.

10. Twisdale, L. A. and Dunn, W. L. EPRI NP-2005, "Tornado Missile Simulation and Design Methodology," Volumes I and II, Final Report dated August 1981.

11. "Safety Analysis Report for Perry 5% Thermal Power Uprate," NEDC-32907P, September 1999.

12. HOLTEC International Report No. HI-2002444, Docket 72-1014, Final Safety Analysis Report for the HOLTEC International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Revision 7, August 2008.

13. PNPP Calculation G58-H-HI-2094384, Missile Penetration Analyses for Perry HI-STORM.

14. PNPP Calculation G58-P-002, Stability Analysis of HI-TRAC/HI-STORM Stack-up for Tornado Missile Impact.

TABLE 3.5-1

SYSTEMS TO BE PROTECTED FROM INTERNALLY GENERATED MISSILES

System	Applicable USAR Section(s)
Containment	<Section 6.2>
Reactor	<Section 5.1>
Reactor Water Recirculation	<Section 5.4>
Nuclear Boiler	<Section 5.1>
Emergency Closed Cooling	<Section 9.2.2>
Reactor Core Isolation Cooling	<Section 5.4.6>
Residual Heat Removal	<Section 5.4.7>
Fuel Pool Cooling and Cleanup	<Section 9.1.3>
Containment Spray	<Section 6.2.2>
Standby Liquid Control	<Section 9.3.5>
High Pressure Core Spray	<Section 6.3>
Low Pressure Core Spray	<Section 6.3>
Emergency Service Water	<Section 9.2.1>
Combustible Gas Control	
Control Rod Drive Hydraulic (Scram Section)	
Containment Atmosphere Monitoring	
Leak Detection	
Main Steam	<Section 10.3>
Feedwater	<Section 10.4.7>
Main Steam Drains	<Section 10.3>
Emergency Service Water Screen Wash	<Section 9.2.1>
Safety-Related Instrument Air	<Section 9.3.1>
Standby Diesel Generator and	<Section 8.3>,
Subsystems	<Section 9.5.4>
	<Section 9.5.5>,
	<Section 9.5.6>,
	<Section 9.5.7>,
	<Section 9.5.8>
Remote Shutdown Panel	
Reactor Protection System	<Section 7.2>
HVAC Systems (required during above operations)	<Section 9.4>
Electrical and Control Systems, and Wiring (required for above operations)	<Section 7.0>, <Section 8.0>

TABLE 3.5-2

POSTULATED MISSILES OUTSIDE CONTAINMENT

Component[1] as Postulated Missile	Location	Barrier
B21-TE-N040	Steam Tunnel at Elev. 633'-0"	Steam Tunnel Ceiling
B21-TE-N041A	Steam Tunnel at Elev. 625'-3 1/2"	Steam Tunnel Wall
B21-TE-N041B	Steam Tunnel at Elev. 625'-3 1/2"	Feedwater Line A (Nonsafety) and Steam Tunnel Wall
B21-TE-N041C	Steam Tunnel at Elev. 625'-3 1/2"	Feedwater Line B (Nonsafety) and Steam Tunnel Wall
B21-TE-N041D	Steam Tunnel at Elev. 625'-3 1/2"	Steam Tunnel Wall
B33-TE-N029	Steam Tunnel at Elev. 633'-0"	Steam Tunnel Ceiling
B33-TE-N030	Steam Tunnel at Elev. 633'-0"	Steam Tunnel Ceiling
B21-TE-N059 (Abandoned-in-Place)	Steam Tunnel at Elev. 626'-0"	Steam Tunnel Wall

NOTE:

[1] Temperature elements (TE) at specific locations.

TABLE 3.5-3

POSTULATED MISSILES INSIDE CONTAINMENT

Component[1] as Postulated Missile	Location	Barrier
B33-C001A	Elev. 619'-4"	Biological Shield Wall
Pump Impeller Fragment, Suction Line Safe End Nozzle Rupture		
TE-H[2]	Elev. 589'-5"	Recirc. Pump A Motor
TE-J[3]	Elev. 589'-1"	Reactor Vessel Pedestal
B33-C001B	Elev. 619'-4"	Biological Shield Wall
Pump Impeller Fragment, Suction Line Safe End Nozzle Rupture		
TE-H	Elev. 589'-5"	Recirc. Pump B Motor
TE-J	Elev. 589'-1"	Reactor Vessel Pedestal
B33-TE-N021	Elev. 588'-9"	Reactor Vessel Pedestal
B33-TE-N022	Elev. 588'-9"	Reactor Vessel Pedestal
B33-TE-N023A	Elev. 587'-5"	Recirc. Pump A Motor
B33-TE-N023B	Elev. 587'-5"	Recirc. Pump B Motor
B33-TE-N028A	Elev. 586'-6"	Weir Wall
B33-TE-N028B	Elev. 586'-6"	Weir Wall
G33-TE-N004	Elev. 626'-6"	Biological Shield Wall
G33-TE-N006	Elev. 656'-4"	Compartment Floor
G33-TE-N007	Elev. 655'-1"	Compartment Wall
G33-TE-N015	Elev. 631'-0"	Compartment Wall
G33-TE-N019	Elev. 655'-1"	Compartment Wall
G33-TS-N019	Elev. 655'-1"	Compartment Wall

TABLE 3.5-3 (Continued)

<u>NOTES</u>:

[1] Temperature elements (TE), temperature switch (TS) at specific locations.

[2] Temperature element/thermocouple off No. 2 seal cavity.

[3] Temperature element/thermocouple off No. 1 seal cavity.

TABLE 3.5-4

POTENTIAL TORNADO MISSILES

Element	Geometric Properties[1]	Total Weight (lb)	Design Velocity (ft/sec)
Wood Plank	Am = 4 in. x 12 in. L = 12 ft	125	280
Wood Pole	D = 8 in. L = 12 ft	209	180
Utility Pole	D = 14 in. L = 35 ft	1,880	91
Solid Steel Rod	D = 1 in. L = 3 ft	8	200
6 in. Sch 40 Pipe	D = 6.625 in. L = 15 ft	285	155
12 in. Sch 40 Pipe	D = 12.75 in. L = 15 ft	800	75
Compact Auto	Am = 1.5 ft x 4.9 ft	2,000	180
Passenger Auto	Am = 1.67 ft x 6.0 ft	4,000	170

NOTE:

[1] D = diameter
L = length
Am = impact area

TABLE 3.5-5

TORNADO MISSILE SHIELDS AND BARRIERS

Structure/System/ Component Protected	Shield/Barrier	Structural Concrete Thickness	Structural Concrete Design Strength	Location Shown on USAR Figure
Diesel Generator Building Doors	Manway Labyrinth (6 places)	2'-0"	3,000 psi at 28 days	<Figure 1.2-5>
Diesel Generator Building - Exhaust Air Vents	External Shields	2'-0"	3,000 psi at 28 days	<Figure 1.2-13>
Diesel Generator Exhaust	Shield Wall and Slab and External Shields	2'-0"	3,000 psi at 28 days	<Figure 1.2-13>
Diesel Generator Intake Air Vents (4)	Control Building Wall	2'-0"	3,000 psi at 28 days	<Figure 1.2-5>
Diesel Generator Fuel Storage Tank - Maintenance Structure Area	Shield Structure/ Hatch Plug	2'-0"	3,000 psi at 28 days	<Figure 9.5-22>
Emergency Service Water Pumphouse - Intake Air Vents	External Shields	2'-0"	3,000 psi at 28 days	<Figure 1.2-16>
Emergency Service Water Pumphouse - Exhaust Air Vents	External Shields	2'-0"	3,000 psi at 28 days	<Figure 1.2-16>
Emergency Service Water Pumphouse - Exhaust Vent (Diesel Driven Fire Service Pump Room)	External Shields	2'-0"	3,000 psi at 28 days	<Figure 1.2-16>

TABLE 3.5-5 (Continued)

Structure/System/ Component Protected	Shield/Barrier	Structural Concrete		Location Shown on USAR Figure
		Thickness	Design Strength	
Emergency Service Water Pumphouse Door	Manway Labyrinth	2'-0"	3,000 psi at 28 days	\<Figure 1.2-16\>
Intermediate Building Plant Vent	Concrete Stack	1'-0"	3,000 psi at 28 days	\<Figure 1.2-10\>
Intermediate/Fuel Han- dling Bldg Intake Air Vent	External Shield	1'-0"	3,000 psi at 28 days	\<Figure 1.2-6\>
Radwaste Bldg Air Intake Unit	External Enclosure	2'-0"	3,000 psi at 28 days	\<Figure 1.2-8\>
Auxiliary Bldg Air Intake Unit	External Enclosure	2'-0"	3,000 psi at 28 days	\<Figure 1.2-8\>
Control Complex Intake Air Vents (2 places)	External Shield	2'-0"	3,000 psi at 28 days	\<Figure 1.2-9\>
Control Complex Exhaust Air Vents (2 places)	External Shield	2'-0"	3,000 psi at 28 days	\<Figure 1.2-10\>
Underground Duct Bank Manholes	Manhole Cover	2'-6"	3,000 psi at 28 days	\<Figure 3.8-76\>
Intermediate Bldg: Reactor Bldg Supply Air Intakes (2)	External Shield	2'-0"	3,000 psi at 28 days	\<Figure 1.2-9\>

TABLE 3.5-5 (Continued)

Structure/System/ Component Protected	Shield/Barrier	Structural Concrete Thickness	Structural Concrete Design Strength	Location Shown on USAR Figure
Intermediate Bldg: Intake Air Vent	Internal Shield Wall	2'-0"	3,000 psi at 28 days	<Figure 1.2-9>
Condensate Storage tank and associated conduit, instrumentation, and piping	Dike/shield wall, protects the CST and associated components from horizontal tornado missiles	2'-0"	3,000 psi at 28 days	-
Instrument missile shield for condensate storage tank level instrumentation	External walls and roof	2'-0"	3,000 psi at 28 days	-

TABLE 3.5-6

EXISTING OPENINGS WITHOUT UNIQUE MISSILE SHIELDS

Building	Opening Description	Approximate Location	Shown On USAR Figure	Missile Protection Provided By
Intermediate Building	Man doors (3' x 7'-2"): Fuel Handling area to exterior (2)	Elev. 620'-6" east	<Figure 1.2-5> internal shields	Location and
	Intermediate building to exterior (opening only, no door)	Elev. 620'-6" west	<Figure 1.2-5>	
	Intermediate building to exterior	Elev. 620'-6" east	<Figure 1.2-9> interior walls	Shield building,
	Stair tower to exterior	Elev. 707'-6" west	<Figure 1.2-10>	Interior walls
	Elev. room to exterior	Elev. 707'-6" south	<Figure 1.2-10>	Interior walls
	Rolling steel doors: To exterior (16' x 17')	Elev. 620'-6" south	<Figure 1.2-5>	<Section 3.5.1.4.2.1>
	To exterior (15' x 14')	Elev. 620'-6" west	<Figure 1.2-5>	Location, adjacent structures and interior walls
	Misc. Pipe & Conduit Penetrations	Various – see TORMIS Analysis for details	-	<Section 3.5.1.4.2.1>

TABLE 3.5-6 (Continued)

Building	Opening Description	Approximate Location	Shown On USAR Figure	Missile Protection Provided By
Auxiliary Building	Man doors (3' x 7'-2"):			
	Auxiliary building to exterior	Elev. 620'-6" east	<Figure 1.2-5>	<Section 3.5.1.4.2.1>
	Stair tower to exterior	Elev. 620'-6" east	<Figure 1.2-5>	Location
	Stair tower to exterior	Elev. 620'-6" west	<Figure 1.2-5>	Adjacent structures, interior walls
	Aux. building to turbine power complex	Elev. 620'-6" north	<Figure 1.2-5>	Steam tunnel, offgas, turbine building
	Stair tower to exterior	Elev. 652'-0" east		Location
	Stair tower to exterior	Elev. 652'-0" west		Location
	Rolling steel doors:			
	To exterior (15' x 14')	Elev. 620'-6" east	<Figure 1.2-5>	<Section 3.5.1.4.2.1>
	To exterior (15' x 14')	Elev. 620'-6" west	<Figure 1.2-5>	Adjacent structures, interior walls
	Electrical openings to turbine power complex (2)	Elev. 633'-0" north		Adjacent structures
	Misc. Pipe & Conduit Penetrations	Varies - See TORMIS Analysis for details	-	<Section 3.5.1.4.2.1>
Diesel Generator Building	Rolling steel door to control complex (12' x 10') - DG114	Elev. 620'-6" east	-	<Section 3.5.1.4.2.1>
	Misc. Pipe & Conduit Penetrations	Various - See TORMIS Analysis for details	-	<Section 3.5.1.4.2.1>

TABLE 3.5-6 (Continued)

Building	Opening Description	Approximate Location	Shown On USAR Figure	Missile Protection Provided By
Radwaste Building	Doors (3' x 7'-2"):			
	Stair tower to exterior	Elev. 620'-6" east	<Figure 1.2-5>	Interior walls
		Elev. 623'-6" north	<Figure 1.2-5>	Interior walls
	Radwaste to exterior	Elev. 646'-6" east (2)	<Figure 1.2-6>	
		Elev. 646'-6" west		
	Elev. tower to exterior	Elev. 664'-6" south	<Figure 1.2-7>	Interior walls
	Opening to radwaste storage area (12' x 15')	Elev. 620'-6"	<Figure 1.2-5>	Interior walls
Offgas Building	Doors (3' x 7'-2"):			
	Offgas to exterior	Elev. 620'-6" west	<Figure 1.2-5>	Location, adjacent structures, interior walls
	Stair tower to exterior	Elev. 620'-6" south	<Figure 1.2-5>	Location, adjacent structures
	Offgas to exterior	Elev. 647'-6" north		
	Door to exterior (6' x 7'-2")	Elev. 620'-6" south	<Figure 1.2-5>	Location, adjacent structures
	Cowl opening – plant vent (3')	Elev. 660'-0" (roof)	<Figure 1.2-7>	Location

TABLE 3.5-6 (Continued)

Building	Opening Description	Approximate Location	Shown On USAR Figure	Missile Protection Provided By
Offgas Building (Continued)	Relief louver (4' x 4')	Elev. 628'-0" west		Location, adjacent structures, interior walls
	Opening with hood (8' x 8')	Elev. 645'-5" east		Adjacent structures
Control Complex	Bullet resistant door to exterior (6' x 7'-2")	Elev. 620'-6" south	<Figure 1.2-5>	<Section 3.5.1.4.2.1>
	Bullet resistant door to service building (6' x 7'-2")	Elev. 654'-6" south	<Figure 1.2-7>	<Section 3.5.1.4.2.1>
	Door – stair tower and elevator (3' x 7'-2")	Elev. 707'-6" south	<Figure 1.2-10>	Interior walls/ slab
	Misc. Pipe & Conduit Penetrations	Various – See TORMIS Analysis for details	–	<Section 3.5.1.4.2.1>

TABLE 3.5-7

SAFETY-RELATED SYSTEMS/COMPONENTS LOCATED OUTSIDE
SEISMIC CATEGORY I STRUCTURES[1]

System/Component	Shown on USAR Figure	Protection From External Missiles
Emergency Service Water Intake Structure	<Figure 3.8-68>	(2) Multi-Port structures – crushing can be tolerated <Section 9.2>
Emergency Service Water Discharge Structure	<Figure 3.8-67>	On lake bottom, 13.3' min. water cover plus 2' min. concrete
Emergency Service Water Piping, except Swale Discharge Piping (for Swale discharge piping, <Table 3.5-8>)	<Figure 9.2-2>	Underground, covered with 4.0' min. compacted earth fill or controlled low strength material, as described in <Section 2.5.4.5.5>
Duct and Conduit to Yard Areas		Underground, covered with 5" concrete and 2.0' min. compacted earth fill
HPCS, RCIC Piping to Reserve Water in Cond. Stg. Tank	<Figure 9.2-13>	Underground, covered with 4.5' min. compacted earth fill, except for above ground piping at CST connection. <Section 5.4.6.1>, <Section 6.3.2.2.1>, and <Section 7.4.1.1>
HPCS, RCIC Instrumen- tation	<Figure 9.2-13>	Instrumentation missile shield and as described in <Section 5.4.6.1>, <Section 6.3.2.2.1>, and <Section 7.4.1.1>
Diesel Fuel Storage Tank	<Figure 1.2-5>	Underground, covered with 5.0' compacted earth fill

TABLE 3.5-7 (Continued)

System/Component	Shown on USAR Figure	Protection From External Missiles
Diesel Fuel Storage Tank - Maintenance Structure Area	<Figure 9.5-22>	2' min. concrete shield structure and concrete hatch plug

NOTE:

[1] For information on safety-related systems/components requiring Tornado Missile Analysis (TORMIS) or protection, see <Section 3.5.1.4.2.1>.

TABLE 3.5-8

TORMIS ANALYSIS TORNADO MISSILE TARGETS[1]

LOCATION	TARGETS
Auxiliary Bldg. Roll-Up and Man Doors - East Wall - Elev. 620'-6"	ESW Discharge Piping
Exterior East Wall of Auxiliary Bldg.	ESW Swale Discharge Piping
Control Complex Door South Wall - Elev. 654'-6"	Unit 1 Control Room
Control Complex Door South Wall - Elev. 620'-6"	Divisional Cable Chases
Misc. Pipe and Conduit Penetrations into Control Complex	Penetration & Depressurization Vent Path
Misc. Pipe and Conduit Penetrations into Diesel Generator Bldg.	Division 1, Division 2, & Division 3 Diesel Generators
Diesel Generator Bldg. Unit 2 East Wall of Bldg.	MCC Switchgear Room and Divisional Cable Chases
ESW Pumphouse Misc. Pipe and Conduit Penetrations	Misc. Systems Supporting the ESW Pumps
Fuel Handling Bldg. Roll-Up South Wall - Elev. 620'-6"	Spent Fuel Pool Dry Cask: HI-TRAC/MPC loaded with spent fuel on decontamination pad[2] Dry Cask: HI-TRAC/MPC loaded with spent fuel in the cask pit pool[2] Dry Cask: HI-TRAC/HI-STORM/MPC loaded with spent fuel at stack-up location[2] Unit 1 Containment Equipment Hatch[2]
Control Complex Door East Wall - Elev. 707'-6"	Door Tornado Depressurization Vent Path
Control Complex East Penthouse Wall – Elev. 709'-2"	Intake Louver Barrier Tornado Depressurization Vent Path
Control Complex Penthouse Roof - Elev. 719'-2"	Exhaust Fan Barrier Tornado Depressurization Vent Path

NOTES:

[1] See <Section 3.5.1.4.2.1> for the description of the TORMIS analysis.

[2] Each target is evaluated but only one is a target at any given time.

3.6   PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE

          POSTULATED RUPTURE OF PIPING

3.6.1   POSTULATED PIPING FAILURES IN FLUID SYSTEMS

3.6.1.1   Design Bases

The foremost requirement for protection against the effects of a
postulated pipe rupture event is conformance to licensing basis offsite
dose limits. This objective is achieved by plant arrangements which
permit habitability of the control room, ensure essential shutdown
system operation and mitigate the consequences of the piping failure.
The plant design provides this protection primarily by physical
separation through spatial arrangement or enclosure within structures or
compartments. Structural barriers and jet impingement shields are
employed as required to further mitigate the consequences of pipe break
events. Where separation or barriers are either insufficient or
impractical, protection by means of pipe whip restraints (high energy)
or environmental qualification of equipment (moderate energy) is used to
ensure the operability of equipment and structures essential for safe
plant shutdown.

A comprehensive review of plant fluid systems with respect to postulated
rupture of piping is presented in <Section 3.6.1.2>. Spatial
arrangements of high and moderate energy lines are shown relative to
equipment required for safe shutdown. <Section 3.6.1.3> provides a
discussion of the effects of postulated pipe ruptures coincident with
single active failures in required systems. The ability to safely shut
down the plant is discussed with respect to the various combinations of
pipe rupture and single active failure. Environmental conditions for
which equipment is designed to operate in the accident mode are
addressed in <Section 3.11>.

3.6.1.2   Description

High and moderate energy lines are listed in <Table 3.6-1> and
<Table 3.6-2>, respectively. These lines are illustrated by
<Figure 3.6-1>, <Figure 3.6-2>, <Figure 3.6-3>, <Figure 3.6-4>,
<Figure 3.6-5>, <Figure 3.6-6>, <Figure 3.6-7>, <Figure 3.6-8>,
<Figure 3.6-9>, <Figure 3.6-10>, <Figure 3.6-11>, <Figure 3.6-12>,
<Figure 3.6-13>, <Figure 3.6-14>, <Figure 3.6-15>, <Figure 3.6-16>,
<Figure 3.6-17>, <Figure 3.6-18>, <Figure 3.6-19>, <Figure 3.6-20>,
<Figure 3.6-21>, <Figure 3.6-22>, <Figure 3.6-23>, <Figure 3.6-24>,
<Figure 3.6-25>, <Figure 3.6-26>, <Figure 3.6-27>, <Figure 3.6-28>,
<Figure 3.6-29>, <Figure 3.6-30>, <Figure 3.6-31>, <Figure 3.6-32>,
<Figure 3.6-33>, <Figure 3.6-34>, <Figure 3.6-35>, <Figure 3.6-36>,
<Figure 3.6-37>, <Figure 3.6-38>, <Figure 3.6-39>, <Figure 3.6-40>,
<Figure 3.6-41>, <Figure 3.6-42>, <Figure 3.6-43>, <Figure 3.6-44>,
<Figure 3.6-45>, <Figure 3.6-46>, and <Figure 3.6-47> in relation to
plant layout. These figures also identify systems and components
required for safe shutdown <Table 3.6-3>. As illustrated by
Figures 3.6-1 through 3.6-47, systems and components required for safe
shutdown are protected from postulated pipe rupture, to a large extent,
by physical arrangement. Detailed descriptions of these physical
arrangements are presented in the sections that follow.

3.6.1.2.1   Physical Arrangement Inside the Reactor Building

a.   Inside the Drywell

     To the greatest possible extent, the piping, the electrical and
     structural arrangement within the drywell, by means of spatial
     separation, provides for safe shutdown capability in the event of
     high energy pipe rupture. Both the main coolant piping
     (recirculation and feedwater) and the ECCS piping are evenly
     distributed around the reactor. Furthermore, the electrical power
     divisions serving the various ECCS systems govern the location of

system pipe routing to prevent any single high energy pipe break from jeopardizing any additional ECCS. A limited number of postulated ruptures could potentially jeopardize the functioning of an adequately redundant number of ECCS due to limitations of spatial and barrier separation. Each such case is discussed in <Section 3.6.2> and resolved either by means of a jet shield or by analytically establishing the adequacy of separation. These high energy lines within the drywell are restrained from whipping by elastic/plastic pipe whip restraints which prevent pipe whip damage to essential systems and limit structural loads.

b.   Between the Drywell and the Reactor Building Wall

Between the drywell and the reactor building wall, portions of two high energy systems constitute potential pipe rupture sources: the reactor water cleanup system and control rod drive supply line. In all cases postulated ruptures are located so that spatial separation provides protection to ECCS from the effects of postulated ruptures.

High energy lines between the drywell and the reactor building wall are restrained from whipping by pipe whip restraints in all cases in which damage could have occurred to structures, systems or components necessary for safe shutdown.

3.6.1.2.2   Physical Arrangement Outside the Reactor Building

Building arrangements outside the reactor building are characterized by the following areas for purposes of the pipe rupture analysis:

a.   Inside the Steam Tunnel

A significant design feature of the plant with regard to postulated rupture of high energy piping is the provision of the steam tunnel.

This structure serves as a conduit for essentially all high energy piping between the reactor building and turbine building. The steam tunnel is designed to contain the environmental effects (pressure and temperature) resulting from a full circumferential pipe break (double ended rupture) of either a main steam or feedwater pipe. Following such a postulated event, the steam tunnel vents the blowdown from the break to the turbine building. Rapid closing isolation valves close to limit the release of mass and energy from the break. These valves and their operation are discussed in <Section 5.4.5> and <Section 6.2.4>. The pressure rise analysis for this design basis event is discussed in <Section 3.6.2>.

High energy piping routed through the steam tunnel is shown in <Figure 3.6-24>. Pipe whip restraints and shields are provided to prevent consequential damage following a postulated pipe break.

b.   Inside the Fuel Handling Building

The fuel handling building is free of high energy lines, except for one 2-1/2 inch nominal OD control rod drive (CRD) line which conveys cold water at approximately 1,900 psig. This line is prevented from damaging surrounding structures by means of piping supports of sufficient capacity. No equipment required for safe shutdown is located in the vicinity of the route of this line in the lowest elevation of the fuel handling building. The consequences of a postulated rupture of this line are limited to local flooding of the lowest elevation in the fuel handling building to a depth of approximately six inches.

Consequences of a moderate energy line crack of the fuel pool cooling and cleanup system results in a flood depth of 10.7 inches on the lowest level. No equipment required for safe shutdown is affected by this flood.

c.   Inside the Intermediate Building

      The intermediate building contains no high energy piping.

      Moderate energy lines whose failure could result in limited (less
      than six inches in depth) flooding of the lowest level of the
      intermediate building present no hazard to the operation of any
      systems essential to safe plant shutdown.

d.   Inside the Auxiliary Building

      The auxiliary building, excluding the structurally separated steam
      tunnel addressed in Item a, above, contains four sources of high
      energy pipe ruptures. The reactor water cleanup system piping and
      pumps are located in a compartment which is vented to a corridor
      containing safety-related electrical cabling. Analysis of the
      conditions following the pipe rupture event indicate the safe
      shutdown capability of the plant is not jeopardized. The second
      source of high energy pipe rupture occurs in a main steam drain
      line routed through the same corridor which communicated with the
      RWCU pump room. The piping configuration of this drain line is
      such that the postulated break occurs within a guard pipe which
      also vents to the steam tunnel. Analysis of the effects of this
      event indicate the safe shutdown capability of the plant is not
      jeopardized. A third source of high energy pipe rupture is the
      RCIC steam drain in the RHR "A" room. This line operates at low
      flow and pressure and presents no significant hazard. The fourth
      source of high energy pipe rupture is the auxiliary steam system.
      The main auxiliary steam piping is routed to enter the steam tunnel
      from the turbine building and exit through the steam tunnel roof
      where it proceeds over the auxiliary building.

      Breaks in the 10 inch auxiliary steam main in the auxiliary
      building are confined to the steam tunnel. A four inch test line

to the RCIC turbine is normally isolated. Condensate lines through the auxiliary building are maintained below 275 psig and below 200°F by condensate coolers.

Two 10-inch RCIC steam supply lines pressurized from the reactor vessel are located in two RHR heat exchanger rooms. A four inch branch to the RCIC turbine is located in the RHR "A" room. Analysis of the conditions following a pipe rupture in each of these lines indicates that the safe shutdown capability of the plant is not jeopardized. Shields and restraints provide required protection from transient jet and rupture loads.

e.   Inside the Control Complex

The control complex is isolated from adjacent structures by three foot thick concrete walls and pressure tight doors where required.

A portion of moderate energy piping is concentrated in two areas of the control complex. One area, at Elevation 599'-0", houses the nuclear closed cooling water (NCCW) heat exchangers served by service water piping. The piping and heat exchangers are in a single, enclosed room. Water flowing from a postulated leakage crack in NCCW or service water piping would either drain through sleeves in the floor to the next lower Elevation at 574'-10" or discharge directly into that elevation. The area below the NCCW heat exchanger room at Elevation 574'-10" houses the service and instrument air receiver tanks. Elsewhere at this elevation are essential shutdown systems. The water would drain to the floor of this space, and from there to floor drain sumps equipped with redundant nonsafety-related seismically mounted instrumentation that actuates alarms to the control room upon detection of high level.

The maximum leakage rate is from a through-the-wall crack in service water piping and is calculated to be 1.8 ft^3/sec. Pipe size is 42 inches, nominal OD, with a wall thickness of 1/2 inch and a system head of 45 psi; this piping is seismically supported. Considering the inventory of water in the service water piping, an overall time in excess of 30 minutes is available from first sump level alarm to required isolation in order to prevent a final flood level less than 18 inches.

Equipment required for safe shutdown or for maintaining control room habitability is located at Elevation 574'-10". This equipment includes three water chillers and the emergency closed cooling water pumps. This equipment is protected from moderate energy line crack flooding by mounting it on 6 inch foundation pads, and the lowest height of flood sensitive safety-related components is 21 inches above floor.

The area, at Elevation 679'-6" above the control room, houses chilled water piping (CCCW) that provides cooling for the control room HVAC equipment. A crack in this moderate energy system is not limiting from a flood perspective. The area at Elevation 679'-6" is sealed off from the control room and is provided with completely embedded drain piping sized to carry water issuing from the design basis leakage crack to drain sumps outside the control complex. This area also houses an electric boiler capable of producing saturated steam at 5 to 10 psig. This boiler supplies a low pressure humidification system, with piping defined as high energy piping.

An analysis of possible effects of jets and pipe whip due to humidification system breaks shows that safe shutdown is not jeopardized. The low power rating of the boiler and the small energy reservoir of the system preclude any rapid environmental

effects. Redundant leak detection sensors are provided to ensure
that any failure is detected with ample time to shut off the boiler
before environmental effects could compromise safe shutdown
components.

f.   (Deleted)

g.   Inside the Diesel Generator Building

Only moderate energy piping is located in the diesel generator
building. Each diesel generator is separated from adjacent diesel
generators by 18 inch thick concrete walls. Therefore, any
postulated event that might disable one diesel generator is
prevented from adversely affecting the others.

h.   Inside the Emergency Service Water Pumphouse Building

Only moderate energy piping is located in the emergency service
water pumphouse building. In the event of a postulated moderate
energy line break resulting in localized wetting and 100% relative
humidity, the essential electrical equipment is qualified and will
not adversely affect the operation of safe shutdown systems.

i.   Yard Piping and Other Structures

     Piping failures in yard piping and in piping in structures not
     addressed in Items a through h, above, have been found to result in
     conditions that do not jeopardize safe plant shutdown or adversely
     affect operation of safe shutdown systems. The flooding effects of
     the above ground service water piping installed in Unit 2
     Transformer Alley have been discussed in <Section 3.6.2.3.5.9.3>.

3.6.1.3   Safety Evaluation

Failures which could affect the ability to bring the plant to a safe
shutdown condition are analyzed in <Chapter 15>. These analyses include
consideration of the occurrence of a single active component failure in
required systems coincident with postulated pipe rupture. The pipe
rupture analysis clearly demonstrates that no system or component
required for safe plant shutdown is rendered inoperable as a consequence
of any postulated pipe rupture.

3.6.2   DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS
           ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Pipe break and crack location criteria and methods of analysis are
needed to evaluate the dynamic effects associated with postulated breaks
and cracks in high and moderate energy fluid system piping inside and
outside of primary containment. This information confirms that the
requirements for protection of structures, systems and components relied
upon for safe plant shutdown or to mitigate the consequences of a
postulated pipe break have been satisfied.

3.6.2.1   Criteria Used to Define Break and Crack Location and
           Configuration

The sections that follow establish the criteria for the location and
configuration of postulated breaks and cracks.

3.6.2.1.1   Definition of High Energy Fluid System

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

a.   Maximum temperature exceeds 200°F.

b.   Maximum pressure exceeds 275 psig.

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

3.6.2.1.2   Definition of Moderate Energy Fluid System

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

a.   Maximum temperature is 200°F or less.

b.   Maximum pressure is 275 psig or less.

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

3.6.2.1.3   Postulated Pipe Breaks and Cracks

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

All high energy piping systems (or portions thereof) are considered as
potential initiators for a postulated pipe break under normal plant
conditions and are analyzed for potentially damaging dynamic effects.

Portions of piping systems isolated from the source of the high energy
fluid under normal plant conditions are exempted from consideration of
postulated pipe breaks. This exemption includes portions of piping
systems beyond a normally closed valve. Pump and valve bodies are also
exempted from consideration of pipe break because of the greater wall
thickness of such components.

A high energy pipe break is not postulated to occur simultaneously with
a moderate energy piping system crack, nor is any pipe break or crack
outside containment postulated to occur concurrently with a postulated
pipe break inside containment.

3.6.2.1.4   Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip is not provided where any one of the following
conditions exist:

a.   The postulated pipe break is in a moderate energy piping system.

b.	The unrestrained movement of either end of the ruptured pipe in any feasible direction about a plastic hinge formed within the piping, following a single postulated pipe break, cannot impact any structure, system or component required for safe shutdown.

c.	Reaction forces on the broken pipe are insufficient to impart sufficient energy to the broken pipe to cause unacceptable damage to any structure, component or system required for safe shutdown. Any line restrictions (e.g., flow limiters) between the pressure source and the break location, and the effects of either a single ended or double ended flow condition may be considered in the determination of the reaction forces. The energy of the broken pipe is considered insufficient to cause unacceptable damage if any of the following criteria are met:

1.	The energy level in a whipping pipe is considered insufficient to damage another pipe of equal or greater nominal pipe size, and equal or heavier wall thickness in accordance with NRC Branch Technical Position APCSB 3-1, Item B.2.b(2).

2.	The reaction force, applied to the broken pipe, is insufficient to stress the piping to the elastic limit at any point, and the limits of deflection of the broken pipe, in any direction, do not allow impact of any structure, system or component required for safe shutdown. Cases where this criterion and method are used are listed in <Table 3.6-17>.

3.	The impacting energy of the broken pipe, determined by the strain energy method, does not impair the essential safety function of any impacted component.

3.6.2.1.5   Locations for Postulated Pipe Breaks

Postulated pipe break locations are selected in accordance with NRC
Branch Technical Position APCSB 3-1, Appendix B, and NRC Branch
Technical Position MEB 3-1, November 24, 1975.

a.   For piping systems classified as high energy, postulated break
     locations are as follows:

     1.   The terminal ends of pressurized portions of the run.
          Terminal ends are the extremities of piping runs that connect
          to structures, components or pipe anchors that act as rigid
          constraints to free thermal expansion.

          On Safety Class I piping systems inside containment where the
          break exclusion region has been extended out to the first
          restraint beyond the break exclusion zone, a break will not be
          postulated (in the region beyond the actual break exclusion
          zone) unless required by <Section 3.6.2.5.1>.

     2.   For ASME Code, Section III, Class I, Seismic I piping, breaks
          are postulated to occur at intermediate locations between
          terminal ends whenever the following stress and fatigue limits
          are exceeded:

          (a) The maximum stress range between any two load sets
               (including the zero load set) shall be calculated
               according to Equation (10) of Paragraph NB-3653 of the
               ASME Code, Section III for normal and upset plant
               conditions, including safety relief valve (SRV) loads,
               and an operating basis earthquake (OBE) event transient.
               If this value is less than 2.4 Sm, no break need be
               postulated.

(b) If Equation (10) exceeds 2.4 Sm but is less than 3.0 Sm
     and the cumulative usage factor U of
     Subparagraph NB-3653.5 is less than 0.1, no break need be
     postulated.

(c) If for a given load set, Equation (10) exceeds 3.0 Sm,
     but the maximum stress ranges calculated according to
     Equations (12) and (13) of Subparagraph NB-3653.6 for
     that load set are each less than 2.4 Sm and the
     cumulative usage factor calculated according to
     Subparagraph NB-3653.6 (using Equation (14) or
     Subparagraph NB-3653.3 for S_{alt}) does not exceed 0.1, no
     break need be postulated.

     In accordance with Subparagraph NB-3653.6 and
     BTP-MEB 3.1, Section B.1.b.(1)(b), Equations (12) and
     (13) need be evaluated only for those load sets for which
     Equation (10) exceeds 3.0 Sm.

     A list of Class 1 analysis nodes where the cumulative
     usage factor exceeds 0.1 or where either Equation (12) or
     (13) exceed 2.4 Sm is provided in <Table 3.6-18>.

3. For ASME Code, Section III, Class 2 and 3 piping, breaks are
    postulated to occur at all locations where the sum of
    Equations (9) and (10) of ASME Code Section III,
    Paragraph NC-3652, calculated under all normal and upset plant
    conditions, including SRV loads, and an independent OBE event
    transient, is greater than 0.8 (1.2 Sh + S_A), except where
    break exclusion rules apply.

4. Breaks are assumed in piping designed to the power piping
    code, ANSI B31.1, at each fitting-to-pipe weld, including
    welds to pumps, flanges, attachments, and valves.

Breaks are also postulated at worst locations in high energy B31.1 piping not supported to Seismic Category I and not analyzed for seismic loads. Cracks are also postulated at worst locations in high energy B31.1 piping supported to Seismic Category I or analyzed for seismic loads.

5. In the event that two or more intermediate locations cannot be determined by stress or usage factor limits, at least two intermediate locations are identified on a reasonable basis for each piping run or branch run, unless the piping is a straight run without fittings, attachments and valves; in this case only one intermediate location is chosen. A reasonable basis is one or more of the following:

    (a) Each fitting-to-pipe weld, including welds to pumps, flanges, attachments, and valves.

    (b) Highest stress or usage factor locations.

Where more than two such intermediate locations are possible using the application of the above reasonable basis, those two locations having the greatest damage potential may be used. A break at each end of a fitting may be classified as two discrete break locations where the stress analysis is sufficiently detailed to differentiate stresses at each postulated break.

The Standard Review Plan <NUREG-0800>, Section 3.6.2 was revised in 1987 to eliminate the requirement for postulation of arbitrary intermediate line breaks. Therefore, consideration of these breaks will no longer be required for design changes to the plant. Additionally, previously postulated arbitrary intermediate line breaks may also be eliminated if conditions warrant their removal.

Hardware associated with arbitrary intermediate breaks, which is physically eliminated, will be reflected as appropriate in updated USAR revisions.

b. For piping systems which contain moderate energy fluids, through-wall leakage cracks are postulated at locations as follows:

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

2. In moderate energy fluid system piping located within structures and compartments containing required systems and components. The through wall leakage cracks are postulated to occur individually at locations appropriate to form the basis for providing required protection from the hazards of fluid spraying, flooding, pressurization, and other environmental conditions. Cracks need not be postulated in portions of Seismic Category I, Class I piping (ASME Code Section III) where the maximum stress range is less than 1.2 Sm, and Class 2 or 3 or nonsafety class piping where the maximum stress range calculated as the sum of Equations (9) and (10) of Paragraph NC/ND-3642 of the ASME Code, Section III, under normal and upset plant conditions including SRV discharge loads, and an OBE event transient does not exceed 0.4 (1.2 Sh + Sa).

3. Moderate energy fluid system piping or portions thereof that are located within a compartment or confined area containing a postulated break in high energy fluid system piping are considered acceptable without postulation of through wall leakage cracks, except where a postulated leakage crack in the moderate energy fluid system piping results in more severe

environmental conditions than the break in the proximate high energy fluid piping system. In such cases the provisions of this section, Item b, apply.

3.6.2.1.6   Types of Breaks to Be Postulated in Fluid System Piping

The following types of breaks are postulated in high energy fluid system piping:

a.   No breaks are postulated in piping having a nominal diameter less than or equal to one inch.

b.   Circumferential breaks are postulated only in piping exceeding a one inch nominal pipe diameter, except for CRD insert lines (1-1/4 inch) which do not have postulated breaks.

c.   Longitudinal splits are postulated only in piping having a nominal diameter equal to or greater than four inches.

d.   Circumferential breaks are assumed at all terminal ends and at intermediate locations identified by the criteria stated in <Section 3.6.2.1.5>. At each of the postulated break locations identified, in piping four inches nominal diameter or greater, either a circumferential or a longitudinal break, or both, is postulated according to the following criteria.

     1.   If the maximum stress range exceeds the applicable limits and the maximum stress range in the longitudinal direction is greater than 1.5 times the maximum stress range in the circumferential direction, only the circumferential break need be postulated.

     2.   If the maximum stress range exceeds the limit of <Section 3.6.2.1.5.a.2.a> or <Section 3.6.2.1.5.a.3> and the

maximum stress range in the circumferential direction is
greater than 1.5 times the maximum stress range in the
longitudinal direction, only the longitudinal break need be
postulated.

3. If the maximum circumferential and longitudinal stress ranges
    are within a factor of 1.5 of each other, or if the analysis
    does not differentiate between circumferential and
    longitudinal stress ranges, then both types of breaks are
    postulated.

4. Circumferential breaks are postulated at fitting joints.

5. Longitudinal breaks are postulated in the center of the
    fitting at two diametrically opposed points (but not
    concurrently) located so that the reaction force is
    perpendicular to the plane of the piping and produces
    out-of-plane bending.

6. At arbitrary intermediate break locations only circumferential
    breaks are postulated in accordance with BTP-MEB 3.1,
    B.3.b(2)(b).

7. At terminal ends without longitudinal welds, only
    circumferential breaks are postulated. At terminal ends with
    longitudinal welds, the criteria of Paragraphs 1, 2 and 3 of
    Item d apply, according to BTP-MEB 3-1, Section B.3.b.(2)(a).

e. For design purposes, a longitudinal break area is assumed to be the
    equivalent of one circumferential pipe area.

f. For both longitudinal and circumferential breaks, after assessing
    the contribution of upstream piping flexibilities, pipe whipping is
    assumed to occur in the plane defined by the piping geometry and

configuration for circumferential breaks and out of plane for longitudinal breaks, and to cause pipe movement in the direction of the jet reaction.

g. The dynamic force of the jet discharge at the break location is based upon the effective cross sectional flow area of the pipe and upon a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Line restrictions, flow limiters and the absence of energy reservoirs have been used, as applicable, in the calculation of jet discharge.

The following through wall leakage cracks are postulated in moderate energy fluid systems (or portions of systems):

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

b. Crack openings are assumed as a circular orifice of cross sectional piping flow area equal to that of a rectangle of length equal to one-half pipe diameter and width of one-half pipe wall thickness.

c. Flow from the crack opening is assumed to result in an environment that wets all unprotected components within the structure or compartment, with consequent flooding in the compartment and communicating compartments (based upon a conservatively estimated time flow period to effect corrective actions) and not the environmental effects of a direct spray as is required for a high energy line break (HELB) designated as a "harsh" environmental zone.

d. Through-wall leakage cracks instead of breaks are postulated in the piping of those high energy fluid systems that qualify as moderate due to short operational periods.

3.6.2.1.7   Criteria for High Energy Piping Systems in the Area of
                Containment Isolation Valves

3.6.2.1.7.1   Allowable Piping Stresses in Area of Containment
                 Isolation Valves

No pipe break is postulated in the portions of high energy piping in the
containment penetration break exclusion region.

The containment penetration break exclusion region is defined as that
section of piping between (1) the outboard weld of the outboard
containment isolation valve, and (2) the inboard weld of the inboard
containment isolation valve. Where a torsional and moment restraint is
required to meet the containment penetration stress criteria under
rupture loads, and where breaks between the containment isolation valve
and the restraint would also cause these criteria to be exceeded, the
containment penetration break exclusion region is extended to the
restraint.

The following high energy systems have break exclusion areas:

    1.   Feedwater

    2.   Main Steam

    3.   Reactor Water Cleanup

    4.   Main Steam Drain

    5.   RCIC Steam Line

    6.   CRD Hydraulic Supply

These systems are shown on <Figure 3.6-4>, <Figure 3.6-5>, <Figure 3.6-6>, <Figure 3.6-7>, <Figure 3.6-10>, <Figure 3.6-24>, and <Figure 3.6-33>.

No safe shutdown components other than containment isolation valves and their auxiliaries are located in break exclusion regions.

The containment penetration region of high energy piping meets the following criteria for break exclusion regions:

a. For high energy ASME Code, Section III, Class 1 Piping:

1. Piping in this region shall meet the requirements of Subarticle NE-1120 of the Code.

2. The stress criteria of <Section 3.6.2.1.5.a.2> shall not be exceeded.

3. The maximum stress in the break exclusion region due to a postulated rupture of the affected line outside the break exclusion region, as calculated by Equation (9) of Subparagraph NB-3653 of the Code, shall not exceed 2.25 Sm, except that following a failure outside containment higher stresses are allowed between the outboard containment isolation valve and the outboard torsional and moment restraint, provided a plastic hinge is not formed and the operability of the valve is ensured under this loading in accordance with Standard Review Plan 3.9.3. Primary loads for Equation (9) include those loads which are deflection limited by restraints.

For most piping systems the B_1, B_2 indices are taken from Table NB-3683.2-1 from the ASME Code, SIII 1974 Edition, Winter 1975 Addenda (Design code of record). For some butt

welding elbows, the B_1, B_2 indices used are taken from the
equivalent Table NB-3681(a)-1 from the 1980 Edition,
Winter 1981 Addenda of the ASME Code. This addenda
acknowledged that internal pressure does not detract from the
moment carrying capacity of elbows. This is especially
relevant for a pipe rupture analysis.

A 10 percent increase to allowable stress is permitted
reflecting the 10 percent increase of "minimum specified
design yield strength" (Sy) due to strain rate effects.

b.   For high energy ASME Code, Section III, Class 2 piping:

1.   Piping in this region shall meet the requirements of
     Subsubarticle NE-1120 of the Code.

2.   The stress criteria of <Section 3.6.2.1.5.a.3> shall be met.

3.   The maximum stress in the break exclusion region due to a
     postulated rupture of the affected line outside the break
     exclusion region, as calculated by Equation (9) of
     Paragraph NC-3652 of the Code, shall not exceed 1.8 Sh. The
     exceptions permitted for Class 1 piping under
     <Section 3.6.2.1.7.1.a.(3)>, above, may be applied to piping
     outboard of the outboard containment isolation valve, provided
     that any such piping between the valve and outboard torsional
     and moment restraint constructed to the ANSI B31.1 power
     piping code shall be provided with full radiography of all
     welds, both circumferential and longitudinal. Primary loads
     for Equation (9) include those loads which are
     deflection-limited by restraints.

For most piping systems the "i" stress intensifiers given in
Figure NC-3673.2(b)1, (ASME Code; SIII, 1974 Edition,
Winter 1974 Addenda) are used. A number of components,
primarily tapered transitions, are evaluated using Equation 9
of Paragraph NC-3653 and the B_1, B_2 indices of
Subsubarticle NB-3680 of the 1980 Edition, Winter 1981 Addenda
of Subsection NC of the ASME Code. This addenda acknowledged
that the "i" factor is not an appropriate factor for most
components for the evaluation of primary loads such as due to
pipe rupture.

A 10 percent increase to the allowable stress is permitted
reflecting the 10 percent increase of "minimum specified
design yield strength" (Sy) due to strain rate effects.

c. For high energy ANSI/ASME B31.1 power piping code piping between
the outboard containment isolation valve, and the outboard
torsional and moment restraint:

1. The stress criteria of <Section 3.6.2.1.5.a.3> shall be met.

2. All longitudinal and circumferential welds shall be provided
with 100 percent radiography.

3.6.2.1.7.2   Welded Attachments to the Process Pipe

Welded attachments, for pipe supports or other purposes, to these
portions of piping are designed by means of detailed stress analyses to
demonstrate compliance with applicable limits. A typical attachment is
a welded lug for torsional and moment restraints. In addition, the
number of circumferential and longitudinal piping welds are minimized.
There are no branch connections in these portions of the process pipe,
with the exception of the RHR-RWCU connections to the feedwater lines in
the steam tunnel, and drains and vents. Where guard pipes are used, the

enclosed portion of fluid system piping is of seamless construction.
The length of these portions of piping is the minimum practical. The
analysis of the head fitting, including the welds to the main steam pipe
and the guard pipe, is in accordance with the GE report NEDO-23652.

3.6.2.1.7.3   Design of Pipe

Pipe anchors are designed to be 100 percent volumetrically examinable in
service and a detailed stress analysis is performed to demonstrate
compliance with the applicable limits.

3.6.2.1.7.4   Guard Pipe Design

Design criteria for guard pipe assembly are as follows:

a.   Construction requirements satisfy Subsection NE of Section III of
      the ASME Code.

b.   The guard pipe is designed to a temperature and pressure equal to
      or greater than the normal operating temperature and pressure of
      the process pipe.

c.   The guard pipe is pressure tested in accordance with SA-530-5 of
      the ASME Code, either by the materials manufacturer or the guard
      pipe fabricator. This test may be performed prior to fabrication
      of the complete assembly.

d.   A 100 percent volumetric examination is performed in accordance
      with the requirements of the ASME Code, Section III, Subsection NE,
      for all longitudinal welds (Category A) and all circumferential
      welds (Category B) in the guard pipe.

e.   A 100 percent volumetric examination is performed in accordance
      with the requirements of the ASME Code, Section III, Subsection NB

or NC, depending upon class, for the head fitting to process pipe
weld as a full penetration Category C weld.

3.6.2.1.7.5   Augmented Inservice Inspection

Augmented volumetric inservice inspection for high energy piping systems
in the containment penetration break exclusion is described in
<Section 5.2.4.9>.

All high energy lines over one inch in diameter which have a break
exclusion area will undergo an augmented inservice inspection during
each inspection interval as defined in ASME Section XI, IWA-2400. The
augmented program will include a 100 percent UT examination of all welds
except for socket welds in all line sizes and piping welds in line sizes
less than four inches in diameter. These welds will be examined using
the liquid penetrant method. Alternatively, the number of welds to be
inspected and method of examination can be determined using the
Risk-Informed Program, as documented in the Inservice Examination Plan.

3.6.2.1.8   Criteria for Moderate Energy Piping Systems in the Area
                 of Containment Isolation Valves

No through-wall leakage cracks are postulated in the portions of
moderate energy piping between containment isolation valves which meet
the following criteria:

a.   The requirements of Subsubarticle NE-1120 of the ASME Code,
      Section III.

b.   The maximum stress range calculated as the sum of Equations (9) and
      (10) of Paragraph NC-3642 of the ASME Code, Section III, under
      normal and upset plant conditions including SRV discharge loads,
      and an OBE event transient, does not exceed 0.4 (1.2 Sh + S_A).

Where an approved design allows placement of both isolation valves on
the same side of containment, the piping between containment isolation

valves shall be taken to include the pipe extending from the valves to the weld defining the Class 2 to Class MC boundary.

3.6.2.2   Analytical Methods to Define Blowdown Forcing Functions and Response Models

3.6.2.2.1   Analytical Methods to Define Blowdown Forcing Functions

Rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces which can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other parameters. A more detailed description of the analytical computer model used in defining the blowdown forces is presented in GAI Topical Report 104P (Reference 1).

Criteria used for calculation of fluid blowdown forcing functions include the following:

a.   Circumferential breaks are assumed to result in pipe severance with a break area equivalent to the pipe's cross sectional area and separation amounting to at least one diameter lateral displacement of the ruptured piping sections.

      Longitudinal breaks are equal to a full circumferential break.

b.   The dynamic force of the jet discharge at the break location is based upon the effective cross sectional flow area of the pipe and upon a calculated fluid pressure as modified by an analytically determined thrust coefficient. Line restrictions, flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account in the reduction of jet discharge.

Sections of broken piping without connecting pumps or tanks, containing only cold water or a negligible volume of steam or compressed water above 212°F, compared to the cross sectional area of the break, and separated from other pressurized sources by normally closed valves or check valves, are considered to contain insufficient energy to develop a jet. Frictional effects of piping, components, flow limiters, filters, and metering orifices and venturis may be included in determination of the steady-state portion of the blowdown, as described below. Frictional effects and flow limiters are considered for all blowdown calculations, other than assumptions of 2.0 P_oA for cold water and 1.26 P_oA for steam and flashing water. No mechanical devices have been added solely to reduce jet discharge.

c.   NSSS analyses assume instantaneous breaks. For the balance-of-plant analyses, a rise time of one millisecond is used for the initial pulse, except for main steam line longitudinal breaks outside containment. Break opening times greater than one millisecond are used only for main steam longitudinal breaks outside containment, as reflected in <Table 3.6-12>.

The break opening time for a longitudinal break of main steam lines was calculated using the BMI relationship (Reference 9):

$$A = 4.4.3\, t \left[\left(\frac{t^2 + 1.96}{1.213} \right)^{1/2} - 1.27 \right]$$

where:

A = opening area, $in.^2$

t = time, milliseconds

Based on this equation, the longitudinal break in the 28 inch main steam line would reach one pipe flow area in 0.0118 seconds.

Blowdown forcing functions are determined by either of the two following methods:

a.   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 upon methods described in (Reference 1). These may be described as follows:

    1.   The transient forcing functions at points along the pipe results from the propagation of waves (wave thrust) along the pipe and from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

    2.   The waves cause various sections of the pipe to be loaded with time dependent forces. 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 will occur at the break end, changes in direction of piping and the pressure vessel until a steady flow condition is established. Vessel and free space conditions are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the pipe break.

    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 required for a wave to reach the first change in direction), the force is assumed to decrease to the value of the blowdown thrust ($0.7\ P_oA$).

4.   Time histories of transient pressure, flow rate and other
     thermodynamic properties of the fluid can be used to calculate
     the blowdown force on the pipe using the following equation:

$$F = \left[\left(P - P_a \right) + \frac{\rho u^2}{g_c} \right] A$$

   Where:

   F = Blowdown force.

   P = Pressure at exit plane.

   P_a = Ambient pressure.

   u = Velocity at exit plane.

   ρ = Density at exit plane.

   A = Area of the break.

   g_c = Newton's constant.

5.   Following the transient period, a steady-state period is
     assumed to exist. Steady-state blowdown forces, including
     frictional effects, are calculated. 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 (Reference 2). For subcooled water a
     reduction from the theoretical maximum of 2.0 P_oA is found
     through use of Bernoulli's equation and standard equations,
     such as Darcy's equation, which account for friction.

b. An alternate method for calculating blowdown forcing functions is
    as follows:

    The computer codes, RELAP3 (for main steam and recirculation
    systems, see Reference 3) and RELAP4 (for remaining high energy
    systems, see Reference 1), are used to obtain exit plane
    thermodynamic states for postulated ruptures. Specifically, these
    programs supply exit pressure, specific volume and mass rate. From
    these data the blowdown reaction load is calculated using the
    following equation:

$$\frac{T}{A_E} = P_E - P_\infty + \frac{G_E^2 \ \overline{V}_E}{g_c}$$

$$R = -\frac{T}{A_E} \ x \ A_E$$

    Where:

$\dfrac{T}{A_E}$ = Thrust per unit break area, lbf/ft^2

P_E = Exit pressure, lbf/ft^2

P_E = Receiver pressure, lbf/ft^2

G_E = Exit mass flux, $lbm/sec\text{-}ft^2$

$\overline{V}_E$ = Exit specific volume, ft^3/lbm

g_c = Newton's constant, 32.174 $ft\text{-}lbm/lbf\text{-}sec^2$

R = Reaction force on the pipe, lbf

3.6.2.2.2   Pipe Whip Dynamic Response Analyses

The prediction of time dependent and steady thrust reaction loads caused by blowdown of subcooled, saturated and two phase fluid from a ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads is presented in the following paragraphs.

Criteria used in performing pipe whip dynamic response analyses for the design of pipe whip restraints include the following:

a.   A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other locations.

b.   The analysis includes the dynamic response of the pipe in question and the pipe whip restraints which transmit loading to the structures.

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 within the broken loop is no longer considered part of the reactor coolant pressure boundary. Plastic deformation in the broken pipe is considered as a potential energy absorber. The following strain limits are used:

1.   Fifty percent of the minimum actual ultimate uniform strain
     (at the maximum stress on an engineering stress-strain curve)
     based upon restraint material tests; or

2.   One-half of minimum percent elongation as specified in the
     ASME Code, Section II, or ASTM Specifications, as applicable,
     when demonstrated to be as, or more, conservative than
     Item (1), above.

These limits are the same as those imposed on the energy absorbing,
plastically deforming pipe whip restraints.

Piping systems are designed so that plastic instability does not
occur in the pipe from the design dynamic and static loads.

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
     the failure of which would not further escalate the consequences of
     the accident, are not designed to satisfy the ASME Code imposed
     limits for essential components under faulted loading. However,
     where components are required for safe shutdown or serve a safety
     function to protect the structural integrity of an essential
     component, limits to satisfy the ASME Code requirements for faulted
     conditions and limits to ensure operability, if required, are met.

The pipe whip analysis was performed using the PDA computer program
(Reference 4). 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
independent stress/strain relations are used for the pipe and the

restraint. Similar to the popular plastic hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using an energy consideration and the equations are numerically integrated in small time steps to yield time history information concerning the deformed pipe.

A comprehensive verification program has been performed to demonstrate the conservatism inherent in the PDA pipe whip computer program and the analytical methods used. Part of this verification program included an independent analysis by Nuclear Services Corporation (NSC), under contract to GE, of the recirculation piping for the 1969 Standard Plant Design. The recirculation piping system was chosen for study due to the complex piping arrangement and assorted pipe sizes. The NSC analysis included elastic-plastic pipe properties, elastic-plastic restraint properties and gaps between the restraint and pipe. This analysis is documented in (Reference 5). The piping/restraint system geometry and properties and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by NSC for the restraint load-deflection curve supplied by GE. This approximation is illustrated by <Figure 3.6-48>. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint to structure loads than the GE analysis. The deflection limit used by NSC is the design deflection at one-half the ultimate uniform strain for the GE restraint design. The restraint properties used for both analyses are presented by <Table 3.6-4>. Break locations and restraints analyzed are shown by <Figure 3.6-49>.

A comparison of the NSC analysis with the PDA analysis, as presented by
<Table 3.6-5>, shows that PDA predicts higher loads in 13 of the
16 restraints analyzed with comparable data. This is due to the NSC
model including energy absorbing effects in secondary pipe elements and
structural members. However, PDA predicts higher restraint deflections
in 50 percent of the restraints. The higher deflections predicted by
NSC for the lower loads are caused by the linear approximation used for
the force-deflection curve, rather than by differences in computer
techniques. This comparison demonstrates that the simplified modeling
system used in PDA is adequate for pipe rupture loading, restraint
performance and pipe movement predictions within the meaningful design
requirements for these low probability postulated accidents.

3.6.2.2.3   Asymmetric Dynamic Loading from Postulated Pipe Break at
                 the RPV Nozzle Safe-End

a.   Introduction

     In conformance with the intent of <Regulatory Guide 1.46>, PNPP is
     designed to accommodate the asymmetric loading resulting from a
     postulated high energy pipe break at the RPV nozzle safe-end. The
     analyses of pipe breaks of the reactor nozzle safe-end and
     subsequent transient effects on reactor components and other piping
     and equipment are generally termed "Asymmetric Loading Analyses."
     The transient loads associated with an instantaneous full
     circumferential break at the reactor nozzle safe-end can be
     characterized by:

     1.   Annulus pressurization (when break is in the biological shield
          wall annulus).

     2.   Jet impingement on the shield wall and/or on the RPV.

     3.   Jet reaction (thrust force) at the RPV broken nozzle.

4.   Impact load on the biological shield wall at the pipe whip
     restraint anchor.

The methodology developed to evaluate the effects of asymmetric
loading consists of the following steps:

1.   Determination of the time history mass and energy flow into
     the annular region and the resultant pressure time history in
     the annulus.

2.   Determination of jet reaction and jet impingement forces.

3.   Determination of pipe reaction force time history at the pipe
     whip restraint anchor.

The forces and pressure mentioned above are employed in a
structural dynamic analysis to evaluate the dynamic responses of
the RPV, RPV pedestal, RPV internals, the biological shield wall
and attached piping systems, and components.

b.   Mass Energy Release in Annulus - Annulus Pressurization

     <Section 6.2.1.2> gives a description of the break locations, the
     time history of the mass releases and the resulting containment
     subcompartment pressurization. The computer code RELAP 4/MOD 3 was
     used in the analysis of the blowdown and pressurization.
     RELAP 4/MOD 3 is a general computer program used to analyze the
     thermal hydraulic transient behavior of a water cooled nuclear
     reactor under postulated loss of coolant accident conditions. The
     program simultaneously solves the fluid flow, heat transfer and
     reactor kinetics equations.

The annulus is divided into zones or subcompartments and RELAP 4 generates a pressure time history for each zone. Further detailed information is provided in <Section 6.2>.

c.   Jet Forces

1.   Jet Impingement - Time History

The time history of the jet impingement load on the reactor pressure vessel and shield wall is conservatively calculated and typical results are illustrated in <Figure 3.6-99>.

2.   Jet Reaction (Thrust) Forces

The jet reaction or thrust acting on the pressure vessel from either a feedwater, recirculation or main steam line break was determined from:

$$F = K P_o A$$

where:

K = Thrust coefficient related to thermodynamic state of fluid at the exit

P_o = RPV internal reservoir pressure

A = Cross sectional area of the break

3.   Impact Force at Pipe Whip Restraint Anchor

The impact force at the pipe whip restraint anchor (of the biological shield wall) can be computed from the normal pipe whip analysis with a given break and the details of the pipe

whip restraint. Typical pipe whip restraint forces at the anchor are illustrated in <Figure 3.6-100>.

d.   Dynamic Structural Analysis

1.   Models

The pressure and jet related force-time histories are combined in a dynamic structural analysis to provide forces, moments, accelerations, and displacements for the pedestal, RPV, RPV internals, and shield wall.

2.   Analysis and Results

The dynamic structural analysis will produce resultant forces (or stresses) at the nodes of the analytical model. This analysis can also produce acceleration time history for each node. Further analysis based on these acceleration time histories can then be performed for any component which is attached to the model <Figure 3.6-101>. The results of this asymmetric loading analysis can be combined with other loads on the components to ensure structural integrity is maintained in accordance with applicable codes and standards. The final results from the appropriate load condition are summarized in <Table 3.9-2>.

3.6.2.3   <u>Dynamic Analysis Methods to Verify Integrity and Operability</u>

3.6.2.3.1   Jet Impingement Analyses and Effects on Safety-Related Components

Criteria used for evaluating the effects of fluid jets on safety-related structures, systems and components are as follows:

a. Safety-related structures, systems and components should not be so impaired as to preclude essential functions.

b. Safety-related structures, systems and components which are not necessary to safely shut down the plant for a given postulated pipe break need not be protected from the consequences of the fluid jet.

c. Safe shutdown of the plant following postulated pipe rupture in the reactor coolant pressure boundary must not be aggravated by sequential failures of safety-related piping, and required ECCS performance must be maintained.

d. Licensing basis offsite dose limits must be met.

e. Postulated design basis breaks resulting in jet impingement loads are assumed to occur in high energy lines at full (100 percent) power operation of the plant.

f. Through-wall leakage cracks are postulated to occur in moderate energy lines and are assumed to result in wetting and spraying of safety-related structures, systems and components.

g. (Deleted)

Jet impingement loads are calculated using the following assumptions:

a. For NSSS piping systems, the direction of the fluid jet for purposes of determining impingement loads is based upon the position of the pipe during steady-state blowdown. In BOP piping systems, for purposes of determining the direction of the fluid jet for impingement loads, circumferential breaks are assumed to result in pipe severance and separation amounting to at least one diameter lateral displacement of the ruptured piping sections unless

physically limited by piping restraints, structural members or
piping stiffness, as may be demonstrated by inelastic limit
analysis.

b.   The impinging jet proceeds along a straight path.

c.   The total impingement force acting upon any cross sectional area of
the jet is time and distance invariant, with a total magnitude
equivalent to the fluid blowdown force as defined below.

d.   The jet impingement force is uniformly distributed across the cross
sectional area of the jet and only the portion intercepted by the
target is considered.

e.   The break opening is assumed to be a circular orifice of cross
sectional flow area equal to the effective flow area of the break.

f.   The jet impingement force is equal to the steady-state value of the
fluid blowdown force as calculated using the methods described in
<Section 3.6.2.2.1>.

g.   The distance of jet travel is divided into two or three regions.
Region 1 <Figure 3.6-50> extends from the break to the asymptotic
area. Within this region the discharging fluid flashes and
undergoes expansion from the break area pressure to atmospheric
pressure. In Region 2 the jet remains at a constant diameter. For
partial separation circumferential breaks the area increases as the
jet expands. Therefore, it is assumed that Region 3 never occurs.
In Region 3 (except in partial separation circumferential breaks)
interaction with the surrounding environment is assumed to start
and the jet expands at a half angle of 10 degrees.

h.   Moody (Reference 6) has developed a simple analytical model for
estimating the asymptotic jet area for steam, saturated water and

steam-water blowdown conditions. For fluids discharging from a
break and which are below the saturation temperature at the
corresponding room pressure or have a pressure at the break area
equal to the room pressure, expansion does not occur.

i.   The distance downstream from the break where the asymptotic area is
      reached (Region 1) has been found by Moody (for circumferential and
      longitudinal breaks) to be approximately equal to five pipe
      diameters. Assuming a linear expansion from the break area to the
      asymptotic area, the jet shape can be defined for Region 1, as well
      as for Regions 2 and 3. <Figure 3.6-51> is used to determine the
      asymptotic area.

j.   Both longitudinal and fully separated circumferential breaks are
      treated similarly. The value of fl/D used in the blowdown
      calculation is also used for jet impingement.

k.   Circumferential breaks with partial (i.e., h <D/2) separation
      between the two ends of the broken pipe, not significantly offset
      (i.e., no more than one pipe wall thickness lateral displacement),
      are more difficult to quantify. For such cases the following
      assumptions are made:

      1.   The jet is uniformly distributed around the periphery.

      2.   The jet cross section at any cut through the pipe axis has the
            configuration depicted by part (B) of <Figure 3.6-50> and the
            jet regions are as delineated by this figure.

      3.   The jet force, F_j, equals total blowdown force, F.

4.   The pressure at any point intersected by the jet is determined
      using the following equation:

$$P_j = \frac{F_J}{A_R}$$

      Where:

          P_j = Pressure.

          F_j = Jet force.

          A_R = The total 360 degree area of the jet at a radius
                equal to the distance from the pipe centerline to
                the target.

5.   The pressure of the jet is then multiplied by the area of the
      target submerged within the jet in the manner explained by
      Item 7, below.

6.   Region 1 in part (B) of <Figure 3.6-50> is assumed to be 2.5
      pipe diameters.

7.   The area, A_R, of the jet at target intersection, r_T, from pipe
      centerline is calculated using <Figure 3.6-52> and
      <Figure 3.6-53> (to determine H, width of jet at the
      asymptotic plane) and the relationship:

$$A_R = 2\pi r_T H$$

      <Figure 3.6-52> and <Figure 3.6-53> were developed using
      <Figure 3.6-51> and the following equations:

          A_A = Asympototic area = $2\pi rH$

$$A_B = \text{Break area} = 2\pi Rh$$

$$A_A \,/\, A_B = \frac{rH}{Rh}$$

Where:

$A_A \,/\, A_B$ = is from <Figure 3.6-51>.

r = 6R = (as explained by Item k.6, above).

l. Target loads are determined using the following procedures and assumptions:

1. For both the fully separated circumferential break and the longitudinal break the jet is assumed to reach its asymptote at five pipe diameters from the break; see Region 1 in part (A) of <Figure 3.6-50>. For design purposes, within this region the jet is assumed to have a linear expansion half angle, θ:

$$\tan \Theta = \frac{0.5 \, (D_A - D)}{5D} = \frac{D_A - D}{10D}$$

Where:

D_A = Diameter of the fully expanded jet or:

$$D_A = \left(\frac{4A_A}{\pi} \right)^{1/2}$$

(A_A is determined from <Figure 3.6-51>).

2. The area within Region 2 can be assumed to be constant to the beginning of Region 3 which starts at the intersection of a line drawn at a 10 degree half angle (see dotted line in parts (A) and (C) of <Figure 3.6-50>) and the boundary of the

jet. In Region 3 the area expands at a constant 10 degree half angle.

3. After determination of the total area of the jet at the target, the jet pressure is calculated using the following equation:

$$P_i = \frac{F_j}{A_x}$$

Where:

P_i = Incident pressure.

A_x = Area of the expanded jet at the target intersection.

If the effective target area, A_{te}, is less than the expanded jet area ($A_{te} \leq A_x$), the target is fully submerged in the jet and the impingement load is equal to P_i times A_{te}. If the effective target area is greater than the expanded jet area ($A_{te} > A_x$), the target intercepts the entire jet and the impingement load is equal to P_i times A_x which equals F_j. The effective target areas for various geometries are outlined as follows:

(a) Flat Surface

For a case where a target with physical area, A_t, is oriented at an angle, ϕ, with respect to the jet axis and with no flow reversal, the effective target area, A_{te}, is as follows:

$$A_{te} = A_t \sin \phi$$

(b) Pipe Surface

As the jet hits the convex surface of the pipe, the forward momentum of the jet is decreased rather than stopped. The jet impingement load on the impacted area is therefore reduced. The analytically determined shape factor for a cylindrical surface is 0.5. The effective target area, A_{te}, is as follows:

$$A_{te} = D_A(D)$$

Where:

D_A = Diameter of the jet at the target interface.

D = OD of the target pipe for a fully submerged pipe.

When the target pipe is larger than the area of the jet, the effective target area equals the expanded jet area:

$$A_{te} = A_x$$

(c) For all cases the jet area, A_x, is assumed to be uniform and the load is uniformly distributed on the impinged target area, A_{te}.

m. For the partial separation circumferential break described in Item k, above, the target loads are calculated in a similar manner, except that the jet cross section appears as shown by part (B) of <Figure 3.6-50> and A_R equals A_x, and D_A equals H and is calculated in accordance with Item k.7, above.

Evaluation of the ability of potential targets to withstand the jet impingement loads is performed using the following methods:

a. Evaluation of Piping Systems under Jet Impingement Loads

1. General Electric piping systems:

    (a) The stresses due to jet impingement loads on piping are considered primary stress and are evaluated using ASME Code Section III limits for Service Level D.

    (b) The motion of piping due to jet impingement loads is limited by structural steel, pipe whip restraints, snubbers, or other equipment capable of providing support. This effect is accounted for in the analysis.

2. Balance-of-plant piping systems:

    (a) Jet impingement loads on piping are considered emergency or faulted loads and are evaluated as primary stresses in ASME Code Section III piping analyses. Level C or D service limits are used. Functional check calculations are performed for piping whose function is required for the given event.

    (b) Each jet impingement load is applied independently to the piping system and the load which supplies the largest bending moment for each particular component is used for the evaluations of the pressure retaining capability or functionality of that component.

    (c) Jet impingement load can be characterized as a two part load application on a piping system as follows:

(1) Dynamic Load Portion

A conservative dynamic load factor of 2.0 is
generally assumed. If the design margin of some
targeted components warrants a detailed analysis, a
refined DLF is determined by a representative
dynamic load analysis. A dynamic load factor less
than 2.0 will be accompanied by proper
justifications.

(2) Static Load Portion

Where a steady-state static load is being applied to
the piping system, snubbers are not activated and
the calculated moments or stresses are combined with
other simultaneous loads using the absolute sum
method.

b.   Evaluation of Structural Components under Jet Impingement Loads

1.   Each jet impingement load is applied independently to the
structure and the load which results in the largest internal
stress is used for evaluation of the structural component.

2.   Specifically designed jet impingement barriers, wherever
installed, are considered structural components.

3.   Jet impingement load can be characterized as a two part load
application as follows:

(a) Dynamic Load Portion

Where static analysis methods are used, a suitable DLF is
applied to the static load. Dynamic load factors are

conservatively estimated using (Reference 10). The ratio of the duration of the applied load and the period of the structure in the direction of the applied load is used in the appropriate response curves shown in this reference. Stresses are combined with concurrent vibratory dynamic load cases by the SRSS method.

(b) Static Load Portion

The static load portion of the impingement is combined arithmetically with other simultaneous loads by the absolute sum method.

c. Evaluation of Jet Impingement Loads on Mechanical System Components

1. The physical configuration of valves, pumps, etc., is approximated by rectangular and cylindrical solid shapes enveloping the component elements for the purposes of determining angular deflection coefficients and shape factors.

2. Loads are considered to be part of piping or structural loads due to jet impingement, according to the physical arrangement of the target. Moments are included in the piping loads for jet impingement on valve operators where a component of the valve loading is normal to the pipe axis.

d. Evaluation of Jet Impingement Loads on Electrical Cable Trays

The only safety-related electrical cable trays subject to impingement by high energy jets are located in the RWCU heat exchanger rooms inside containment and the Unit 1 RCIC pump room outside containment. The trays in the RWCU pump room are fully protected by jet shields. The tray in the RCIC pump room contains RCIC and RCIC room leakage detection circuits only, which are not

required to survive the postulated breaks, since the equipment in the rooms is assumed to be lost as a result of the postulated break.

e.   Evaluation of Jet Impingement Loads on Electrical Conduit and Instrumentation Impulse Lines

The design criteria for routing of electrical conduit and instrument impulse lines is intended to ensure that impingement by high energy jets does not occur. However, this is not always feasible. Support of conduit and impulse lines subject to jet impingement is established by adjustments to spacing criteria which assure conduit integrity under governing load conditions. Design of special supports for rigid conduit and routings of flexible connections to equipment consider individual load conditions from impacting jets. Jet shields are used to protect against jet impingement if protection by support design and by routing is not feasible. Circuits not required for safe shutdown for a given pipe break do not require protection.

f.   Evaluation of Jet Impingement Loads on Instrument Racks and Panels

No safety-related instrumentation racks or panels are subject to impingement by high energy jets.

3.6.2.3.2   Pipe Whip Effects on Safety-Related Components

Potential pipe whip effects are identified by comprehensive reviews of all areas of the plant which contain high energy piping. Potential whips due to each postulated circumferential break, and displacement due to each longitudinal break, are evaluated. All structures, systems and components that can possibly be struck by each whipping or displacing pipe are reviewed to determine which are safety-related.

Pipe whip (displacement) effects on safety-related structures, systems and components can be placed into two categories: pipe displacement effects on components (nozzles, valves, tees, etc.) located in the piping run in which the break occurred and pipe whip or controlled displacements onto external components, such as building structure or other piping systems.

a.   Pipe Displacement Effects on Components in the Same Piping Run

     Criteria used for determining the effects of pipe displacements on inline components are as follows:

     1.   Components, such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or the failure of which would not further escalate the consequences of the accident, need not be designed to satisfy limits imposed by ASME Code, Section III, for essential components under emergency loading.

     2.   If these components are required for safe shutdown or serve a safety function to protect the structural integrity of an essential component, limits to satisfy ASME Code requirements for component emergency conditions and limits to ensure operability, if required, are satisfied.

b.   Pipe Displacement Effects on Structures, Other Systems and Components

     The criteria used to ensure the mitigation of the effects of high energy pipe whip on structures, systems and components require that the arrangement of pipe whip restraints, supporting structures and piping system components preclude impact of whipping pipe on any structure, system or component essential to the safe shutdown of the plant in the event of occurrence of a given postulated pipe

rupture. In exceptional cases a damage study as described in
<Section 3.6.2.1.4.c> is used to show that pipe whip impact on any
structures, systems or components essential to safe shutdown does
not compromise the safe shutdown function of those structures,
systems or components.

3.6.2.3.3   Loading Combinations and Design Criteria for Pipe Whip
                 Restraints

Pipe whip restraints (i.e., those devices which serve only to control
movement of a ruptured pipe following gross failure) and torsional and
moment restraints, as differentiated from simple piping supports, are
designed to function and carry load for an extremely low probability
gross failure in a piping system containing high energy fluid. Piping
integrity does not depend upon the pipe whip restraints for any loading
combination. Piping integrity in high energy containment penetration
regions is assured by torsional and moment restraints. When piping
integrity is lost as a result of the occurrence of a postulated break,
the pipe whip restraints act to limit movement of the broken pipe to an
acceptable distance. Pipe whip restraints are subject to once in a
lifetime loading. For purposes of design the pipe break event is
considered to be a faulted plant condition and the pipe, associated
restraints and structures to which restraints are attached are analyzed
and designed accordingly. Pipe whip restraints are designed to strain
limits as described below in this section.

Torsional and moment restraints are also designed for piping support
loads. A typical pipe whip restraint is illustrated by <Figure 3.6-54>.
A principal feature of these restraints is that they are installed with
several inches of annular clearance between the restraint and the
process pipe. This arrangement allows for installation of normal piping
insulation and unrestricted thermal movement of the piping. These
loading combinations and design criteria are applicable to piping
systems such as main steam, recirculation, feedwater, and ECCS (inside

containment), as well as other high energy piping systems inside and outside of containment.

3.6.2.3.3.1   Restraint Design Objectives

Specific design objectives for the restraints are as follows:

a.   The restraints must in no way increase reactor coolant pressure boundary stresses during any normal mode of reactor operation or condition.

b.   The restraint system must function to stop movement of a failed pipe (gross loss of piping integrity) without allowing damage to critical components or missile development.

c.   The restraints should produce minimum hindrance to performance of inservice inspection of process piping.

3.6.2.3.3.2   Restraint Dynamic Loads

For purposes of design the pipe whip restraints are designed for the following dynamic loads:

a.   Blowdown thrust of the pipe section that impacts the restraint.

b.   Dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust and subsequent impact on the restraint.

Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in <Section 3.6.2.2.2>. Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

3.6.2.3.3.3   Restraint Components

The main steam pipe whip restraints are composed of several components, each of which performs a different function. These components are categorized as Types I, II, III, and IV, as described below:

a.   Type I - Restraint Energy Absorption Members

     Restraint energy absorption members, under the influence of impacting pipe (pipe whip), absorb energy by significant plastic deformation (e.g., U-rods).

b.   Type II - Restraint Connecting Members

     Restraint connecting members are those components which form a direct link between the restraint plastic members and the structure (e.g., clevises, brackets, pins).

c.   Type III - Restraint Connecting Member Structural Attachments

     Restraint connecting member structural attachments are those fasteners which provide the method of securing the restraint connecting members to the structure (e.g., weld attachments, bolts).

d.   Type IV - Structural and Civil Components

     Structural and civil components are the steel and concrete structures which ultimately must carry the restraint load (e.g., biological shield, trusses).

Each component Type (I through IV) is typically constructed of a different material, with a different design objective, to perform the overall design function. Therefore, the material and inspection

requirements and design limits for each type of component are somewhat different. The requirements for each type of component are as follows:

a.   Type I Restraint

  1.   Materials

     All materials used to absorb energy through significant plastic deformation must conform to the following:

     (a)   ASME Code, Section III, Subsection NB, for Class 1 components; or

     (b)   ASTM Specifications, with consideration for brittle fracture control; or

     (c)   ASME Code, Section III, Subsection NF, if applicable; or

     (d)   GE Material Specifications.

  2.   Inspection

     Inspection and identification of material must conform to the following:

     (a)   ASME Code, Section III, Subsection NB, for Class 1 components (Section V, "Non-Destructive Examination Methods"); or

     (b)   ASTM Specifications, procedures, including volumetric and surface inspection; or

     (c)   ASME Code, Section III, Subsection NF, if applicable; or

(d) GE Methods and Acceptance Standards

3.   Design Limits

(a) Design Local Strain

Permanent strain in metallic ductile materials must be limited to the following:

(1) Fifty percent of the minimum actual ultimate uniform strain (at the maximum stress on an engineering stress-strain curve) based upon restraint material tests; or

(2) One-half of minimum percent elongation as specified in the ASME Code, Section III, or ASTM Specifications, as applicable, when demonstrated to be as, or more, conservative than Item (1), above.

(b) Design Steady-State Load

The maximum restraint load will be limited to 80 percent of the maximum calculated static ultimate restraint strength at the drywell design temperature. This strain is less than 50 percent of the ultimate uniform strain for all materials used for Type I components.

(c) Dynamic Material Mechanical Properties

The material selected must exhibit tensile impact properties which are not less than the following:

(1) Seventy percent of the static percent elongation; or

(2) Eighty percent of the statically determined minimum total energy absorption.

b. Type II Restraint

1. Materials

Material selection must conform to the following:

(a) ASTM Specifications, including consideration for brittle fracture control; or

(b) ASME Code, Section III, Subsection NF, if applicable; or

(c) GE Material Specifications

2. Inspection

Inspection must conform to the following:

(a) ASME/ASTM requirements or process qualification and finished part surface inspection in accordance with ASTM methods; or

(b) ASME Code, Section III, Subsection NF, if applicable; or

(c) GE Methods and Acceptance Standards.

3. Design Limits

Design limits will be based upon the following stress limits:

(a) Primary stresses (in accordance with definitions in the
     ASME Code, Section III) are limited to the higher of:

   (1) θ = 70 percent of Su, where Su equals minimum
        ultimate strength by test or ASTM specification; or

   (2) Sy + 1/3 (Su - Sy) where Sy equals minimum yield
        strength by test or ASTM specification.

(b) Recommended stress limits in accordance with ASME Code,
     Section III, Subsection NF, for faulted conditions, if
     applicable.

c. Type III Restraint

  1. Fasteners

    (a) Materials must conform to ASTM, ASME or MIL requirements.

    (b) All fasteners must be inspected or certified in
         accordance with applicable ASTM, ASME or MIL
         specifications.

    (c) Design limit will be the same as for Type II materials.

  2. Welds

    (a) Materials for attachment to carbon steel structures will
         be limited to low hydrogen type.

    (b) Liquid penetrant or magnetic particle surface inspection
         must be performed in accordance with the following:

(1) ASME Section III, Subsection NF; or

(2) American Welding Society (AWS D1.1)

(c) Design limits will be based upon the following stress limits:

The maximum primary weld stress intensity (two times maximum shear stress) will be limited to three times AWS or AISC building allowable weld shear stress.

(d) Procedures and welders will be qualified in accordance with the latest AWS Code for welding in building structures.

d. Type IV Restraint

Material, inspection and design requirements for structural and civil components are provided by industry standards, such as AISC, ACI and ASME (ASME Code, Section III, Division II), along with appropriate requirements imposed for similar loading events. These components are also designed for other operational and accident loadings, seismic loadings, wind loadings, and tornado loadings.

The design basis approach of categorizing components is consistent in allowing less stringent requirements for those components subject to lower stresses. Considerable strength margins exist in Type II through IV components even to the limit of load capacity (fracture) of a Type I component. Impact properties in all components are considered since brittle type failures could reduce restraint system effectiveness.

3.6.2.3.3.4   Restraint Material Allowables

In addition to the design considerations discussed above, strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties used in design have included one or more of the following methods:

a.   Code minimum or specification yield and ultimate strength values for the affected components and structures are used for both the dynamic and steady-state events; or

b.   Not more than a 10 percent increase in code or specification values is used when designing Type IV components or structures for the dynamic event. Code minimum or specification yield and ultimate strength values are used for the steady-state loads.

3.6.2.3.3.5   Structural Steel Pipe Rupture Restraints

a.   Location of Structural Steel Pipe Rupture Restraints

Sufficient pipe rupture restraints are provided to prevent damage to structures, systems and components from pipe rupture events. Criteria for selection of locations are as follows:

1.   Structural steel restraints are located and oriented to prevent pipe material failure and subsequent missile formation from pipe rupture events, where the U-rod type restraint offers insufficient pipe motion limiting ability.

2.   Structural steel restraints are located as close to postulated rupture points as practicable without interfering with inservice inspection of pipe welds. Removable restraints are used where interference is unavoidable.

b.   Design of Structural Steel Pipe Rupture Restraints

Structural steel restraints are used where use of U-rod type restraints were impractical because of support structure geometry and/or multidirectional breaks. ASTM A 36 steel is used for frame restraints; ASTM A 516 steel for plate restraint structures.

To allow access for inservice inspection, bolted connections are used wherever frame members are less than four inches from a pipe weld.

c.   Analytical Criteria

Where pipe restraints are employed, they are designed using the principles of the equations of motion.

Since the forces due to dead, live, seismic, and thermal loads are considered negligible or self-relieving, only the dynamic effect of the whipping pipe is considered in the design.

The equations of motion method (dynamic analyses) use the computer programs DYREC (S061) (Reference 7) and DYNAL (S085) (Reference 8).

The computer programs use thrust versus time data. The pipe and the restraint are modeled as lumped mass systems. Nodal masses and element spring properties are determined and gaps are input. Using direct numerical integration of the equations of motion, the dynamic response of the pipe and restraint are calculated at specific time points.

The following criteria are used for material properties:

1.   Minimum yield strength of pipe steel is reduced in accordance with operating temperature.

2. Minimum tensile strength of the pipe material, as listed in the material specification, is used as the ultimate strength of the pipe. Refinement of assumed ultimate strength for changes due to operating temperatures would not result in a substantive change in pipe restraint design or reactions.

3. Ultimate tensile strain of both piping and restraint material is one-half of guaranteed minimum percent elongation. Ultimate shear strain is equal to guaranteed percent elongation.

4. Minimum values of yield strength, ultimate strength and modulus of elasticity for pipe are taken from the ASME Code. Values for restraint material are taken from the applicable ASTM specification.

5. A 10 percent increase in material properties is applied to allow for strain rate effect.

Acceptability of the restraint design is based upon the results of the dynamic analysis. Neither the pipe nor the restraint stresses and strains exceed the following limits:

1. Tensile strains are limited to 50 percent of the assumed ultimate tensile strain. This is equal to 0.25 times percent elongation.

2. Bending and axial tensile stresses are limited to the values at the above strain limit as determined appropriate from the stress/strain, moment/curvature or P/Δ curves.

3. Shearing strains are limited to 50 percent of the assumed ultimate shear strain. This is equal to 0.5 times percent elongation.

4.   Shearing stresses are limited to the value at the above strain
     limit as determined from the shear/shear strain curve.

Restraints, connections, anchorages, and the supporting structure
are designed for the maximum reactions obtained from the dynamic
analysis.

3.6.2.3.3.6   Pipe Whip Restraints as Guides (NSSS Scope)

Pipe whip restraints which also serve as guides are as follows:

a.   Main steam line restraint/guides inside drywell including guide
     numbers G101B and G101D, as shown in <Figure 3.6-65>, and guide
     numbers G101A and G101C, not shown. These restraint guides are
     designed and supplied by GE. GE has classified these
     restraint/guides as a plate and shell type support under
     Subsection NF of the ASME Code. These restraints attach to the
     building steel which is classified AISC.

b.   RCIC restraint/guide inside drywell, G201A, as shown in
     <Figure 3.6-70>. This restraint consists of a structural steel
     frame with an insert around the pipe which serves as a guide. The
     insert/guide is classified as a linear type support under
     Subsection NF of ASME Code. The supporting framework is classified
     as building structure and designed in accordance with AISC. This
     approach is consistent with that of pipe supports on steel
     platforms.

c.   Main steam and feedwater restraint/support structures in the
     auxiliary building steam tunnel as shown in <Figure 3.6-75> and
     <Figure 3.6-76>. These restraints consist of a structural steel
     frame with an insert around the pipe which serves as a guide. The
     insert/guide is classified as a linear type support under
     Subsection NF of ASME Code. The supporting framework is classified

as building structure and designed in accordance with AISC. This
approach is consistent with that of pipe support on steel
platforms.

3.6.2.3.4   Compartment Pressurization Analysis

Based on the blowdown time histories generated using the criteria of
<Section 3.6.2.2.1>, all compartments containing high energy lines have
been analyzed for the highest energy release rate pipe rupture event to
determine the maximum loadings of the compartment. The results of these
analyses are presented in <Figure 3.11-10>, <Figure 3.11-11>,
<Figure 3.11-12>, <Figure 3.11-13>, <Figure 3.11-14>, <Figure 3.11-15>,
<Figure 3.11-16>, <Figure 3.11-17>, <Figure 3.11-18>, <Figure 3.11-19>,
<Figure 3.11-20>, <Figure 3.11-21>, <Figure 3.11-22>, <Figure 3.11-23>,
<Figure 3.11-24>, <Figure 3.11-25>, <Figure 3.11-26>, <Figure 3.11-27>,
<Figure 3.11-28>, <Figure 3.11-29>, <Figure 3.11-30>, <Figure 3.11-31>,
<Figure 3.11-32>, <Figure 3.11-33>, <Figure 3.11-34>, <Figure 3.11-35>,
<Figure 3.11-36>, <Figure 3.11-37>, and <Figure 3.11-38>, which include
peak pressures, temperatures, duration, and means of terminating the
blowdowns. These conditions then become the design parameters
<Section 3.8>.

3.6.2.3.5   Flooding Analysis

Based on blowdown analyses from high energy line breaks discussed in
<Section 3.6.2.2.1> and leakage from moderate energy line cracks, the
potential for flooding of safety-related structures has been determined.
In no case are safe shutdown systems jeopardized by the effects of such
flooding when allowance is given for adequate means of detecting the
event and a reasonable time period for corrective action to be taken.

The sections that follow discuss areas subject to flooding due to breaks in high and moderate energy lines.

3.6.2.3.5.1   Inside the Reactor Building

Loss of reactor coolant, including ECCS injection of the entire contents of the condensate storage tank, could result in flooding the drywell and RPV pedestal to the height of the weir wall. The containment is

designed for flooding above this level as required for accident recovery <Section 3.8.3>. Equipment required for safe shutdown is designed to withstand these effects as described in <Section 3.11>.

A break in the RWCU system in the containment steam tunnel could conceivably flood containment isolation valve operators. Sleeves in the floor of this room drain directly to the suppression pool, and are more than adequate for the maximum break.

Breaks in other areas of the containment building drain directly to the suppression pool or contain no items necessary for safe shutdown that are subject to flood damage.

3.6.2.3.5.2   Inside the Auxiliary Building

Hallways within the auxiliary building at Elevations 568'-4" and 574'-10" are subject to maximum flooding from a through-wall crack in a 24-inch emergency service water pipe from an above elevation. Allowing 30 minutes after a high level alarm to isolate the crack, a depth of less than 20 inches will exist on Elevation 568'-4". This alarm is provided by redundant nonsafety grade level switches set to alarm in the control room at a water level 2 inches above the floor. The ECCS racks are located within the auxiliary building hallways. These racks are mounted on 6-inch concrete pads and have 14 inches of clearance between rack bottom and first level instrumentation. Therefore, any depth less than 20 inches will not jeopardize the operability of the ECCS racks.

In both units, the service water piping is routed above the 599'-0" floor elevation. This piping is of 24-inch nominal diameter in Unit 1 and of 42-inch nominal diameter in Unit 2. This pipe has been supported to Seismic Category I. No leakage cracks need to be postulated in this piping since the applicable stress criteria for Class 2/3 piping have been met.

On Elevation 620'-6", the major threat of flooding exists due to a
24 inch emergency service water pipe routed to the swale. A flow of
less than 230 gallons per minute will exit a through-wall crack in this
pipe. Water will drain either by the floor drain system or by the
stairwells down to Elevation 568'-4". The total volume during the
duration of the leak, will result in a final depth on Elevation 568'-4"
that is significantly less than 20". Since this depth is acceptable,
operability of any essential components is not compromised.

Within the steam tunnel, breaks in the main steam, feedwater and reactor
water cleanup systems have been considered. These breaks do not release
sufficient volume to endanger any isolation valves or other components
required to achieve and maintain safe shutdown.

During normal operations (non-accident), the ECCS pump rooms are subject
to an unisolable leak resulting from the failure of suction piping
upstream of a suction isolation valve. However, the watertight
construction of the ECCS pump rooms protects against mass flooding of
redundant ECCS pump rooms due to a break within any one cubicle. In
addition, it has been demonstrated by analysis that assuming manual
initiation of makeup from the upper containment pool, an unisolable
suppression pool leak into the largest ECCS pump room would not result
in an equilibrium suppression pool water level below that required for
minimum vent submergence or for adequate ECCS pump NPSH.

During an accident, unisolable leaks from piping failures are not
considered except as the initiating event for the LOCA, since the ECCS
is assumed to be operating within allowable design values established
for the pressure boundary. The only passive mechanical failures
considered concurrent with an accident are valve packing leakage and
pump seal leakage. See <Section 6.3.2.6> for additional passive
mechanical failure information.

3.6.2.3.5.3   Inside the Intermediate Building

The intermediate building is subject to flooding from breaks in the intermediate and control complex buildings. A sump high alarm from a level switch and a standby sump pump running alarm are provided to the radwaste control room, allowing ample time to isolate breaks before the depth exceeds six inches. No safety-related equipment in the intermediate building is threatened by floods of this depth.

3.6.2.3.5.4   Inside the Fuel Handling Building

Flooding hazards and protective measures are described in <Section 3.6.1.2.2.b>.

3.6.2.3.5.5   Inside the Control Building

Flooding hazards and protective measures are described in <Section 3.6.1.2.2.e>.

3.6.2.3.5.6   Inside the Diesel Generator Building

Diesel generators are located in separate cubicles. A flood in any one cubicle due to a crack in the diesel generator cooling water system will affect only that diesel generator. Minor wetting of floors in adjacent buildings is of no consequence.

3.6.2.3.5.7   Inside the Emergency Service Water Pumphouse

Discharge from cracks in emergency or fire service water lines drains directly through floor grating into the suction pit. The floor grating has been sized to sufficiently handle the expected flow. No significant flooding potential exists.

3.6.2.3.5.8   Flooding in Other Plant Buildings

The auxiliary boiler building, heater bays, offgas buildings, radwaste
building, condensate demineralizer buildings, circulating water pump
house, service water pumphouse, and other detached structures contain no
components essential to safe shutdown. The turbine buildings contain
instruments and controls supplying inputs to safe shutdown systems.
These systems trip to the safest position on loss of signal. No flood
protection is required for any of these areas.

3.6.2.3.5.9   Flooding in Unit 2 Buildings and Yard Areas

Service water piping is routed through the Auxiliary Building, Turbine
Power Complex/Steam Tunnel, and Transformer Alley of Unit 2. Since
service water piping is a moderate energy system <Table 3.6-2>, impact
on the safety-related equipment in the area due to jet impingement and
flooding has been considered. Jet impingement loading is not a concern
for these areas since the replaced piping is not located in the vicinity
of any safety-related equipment. The installation of buried piping has
been performed in the same manner as the existing pipe and will not have
any affect on the present buried piping flooding analysis. The effects
of flooding have been considered for the above ground piping in the
Unit 2 Transformer Alley.

3.6.2.3.5.9.1   Unit 2 Auxiliary Building

Service water piping in the Unit 2 Auxiliary Building is addressed in
<Section 3.6.2.3.5.2>.

3.6.2.3.5.9.2   Unit 2 Turbine Power Complex and Steam Tunnel

A full guillotine pipe break coincident with a seismic event (not
required to be postulated but was evaluated for conservatism) and a
MEB 3.1 critical crack has been postulated in the service water piping

routed through the Unit 2 turbine power complex and steam tunnel. To increase the storage volume in the TPC, all open penetrations between the turbine power complex and auxiliary buildings below the 599' elevation have been sealed.

Results of the flooding analysis indicate that an uncontrolled guillotine service water line break for 45 minutes after a seismic event (at which time operations will have been able to identify and isolate the leak) will result in the maximum flood height of 595'- 8 ½". Unit 2 turbine power complex/steam tunnel can provide sufficient storage capacity (up to elevation 598'- 11") for the Operators to respond to the event without jeopardizing safety-related equipment.

The flooding event due to a MEB 3.1 crack may occur over several days and this event will be discovered by the Operator during routine rounds. Therefore, a MEB 3.1 crack or a guillotine break of the service water line in Unit 2 turbine power complex or steam tunnel does not create a flooding concern.

3.6.2.3.5.9.3   Unit 2 Transformer Alley

The above ground SW System piping in the Transformer Alley has been evaluated for the effects of a non-mechanistic crack Moderate Energy Line Break (MELB), a seismic event, tornado missiles, and a malevolent vehicle blast during normal operation.

The above ground service water system piping in the Transformer Alley has been evaluated for the effects of tornado missiles as a result of the impact of a 12" X 4" wood plank (considered to be the worst case tornado missile). The results of the analysis indicate that the maximum flood elevation will be less than the ground floor elevation of the power block structures (620.5'). Hence, flooding in the Transformer Alley due to a tornado missile induced break is not a concern.

The evaluation for the seismic event is based on a review of historical data and on an assessment of seismic anchor movement of the above ground piping. Also, the evaluation addressed interfaces, interferences and surrounding hazards. This assessment included a walkdown of the area of the proposed routing for the piping and the surrounding structures and components. The evaluation shows that the piping will maintain its pressure boundary integrity during a safe shutdown earthquake or an operating basis earthquake. Thus, flooding in the Transformer Alley due to a seismically induced break is not a concern.

The effects of a non-mechanistic (MELB) crack required for moderate energy systems are bounded by the crack size caused by the tornado missiles already discussed and is not a concern. In the event of a malevolent vehicle blast, the distance between the above ground SW System piping and the Vehicle Barrier System is sufficient to preclude damage to the piping. Thus, flooding in the Transformer Alley due to a malevolent vehicle blast or an MEB 3.1 crack is not a concern.

3.6.2.3.6   Electrical Protection Criteria

An evaluation was made of effects of pipe break on plant electrical control and instrumentation systems. The evaluation considered effects of physical damage to equipment, cabling, penetrations, and instrument piping resulting from the break. Specific events included are:

a.   Dynamic motion and impact of identified pipe whips, if any.

b.   Fluid jet impingements identification.

c.   Temperature, pressure humidity and flooding conditions determined in the thermal-hydraulic analyses.

d.   Effects of activation of fire protection systems caused by pipe break.

The acceptability of damage resulting from a break is governed by the following criteria:

a. Preserve redundancy in the control, instrumentation, protection, and Class 1E electrical systems required for safe plant shutdown in accordance with IEEE Standard 279.

b. Maintain the ability to deal with consequences of a break, despite a loss of offsite power.

c. Preserve the ability to shut the plant down from the control room or the remote shutdown panel.

d. Breaks which do not result in automatic protective action may induce loss of redundancy, but not loss of function in protective, safety and shutdown systems.

3.6.2.4   Guard Pipe Assembly Design Criteria

3.6.2.4.1   Main Steam Piping System Guard Pipe

The guard pipe assembly for the main steam piping system extends from the drywell to the shield building. It serves as an extension of containment in the annular space between the containment and the shield building to prevent any pressurization of this area due to a postulated break of the main steam piping. In the event of a pipe break between the drywell and the containment, this guard pipe also serves to redirect the released steam back into the drywell where it passes through vents into the suppression pool. This limits containment pressure resulting from the postulated steam pipe break.

Design criteria for the guard pipe and main steam piping located within the guard pipe are presented in <Section 3.6.2.1.7>.

Design details of the main steam piping system guard pipe are shown by
<Figure 3.6-55>. The process pipe used for main steam piping within the
guard pipe is a continuous length of seamless pipe. This eliminates all
inservice inspection access requirements relative to piping within the
guard pipe.

3.6.2.4.2   Other Guard Pipes

Other systems for which guard pipes are used at the containment boundary
are listed in <Table 3.6-6>. All such guard pipes serve as an extension
of containment in the annular space between the containment and the
shield building. Details of guard pipe designs are illustrated by
<Figure 3.6-55>, <Figure 3.6-56>, <Figure 3.6-57>, <Figure 3.6-58>,
<Figure 3.6-59>, <Figure 3.6-60>, <Figure 3.6-61>, <Figure 3.6-62>,
<Figure 3.6-63>, and <Figure 3.6-64> for the Type K, Type Q and Type J
designs. Design criteria for guard pipes, process pipe, fixed heads,
and bellows, as well as inservice inspection requirements, are addressed
by <Section 3.6.2.4.1>.

3.6.2.5   Material Submitted for the Operating License Review

3.6.2.5.1   Implementation of Criteria for Pipe Break and Crack
              Location and Orientation

3.6.2.5.1.1   Postulated Pipe Breaks in Main Steam Piping System,
                Inside Containment

The criteria for selection of postulated pipe breaks in the main steam
piping system inside containment are presented in <Section 3.6.2.1>.
Postulated pipe break locations and types selected in accordance with
these criteria for main steam lines A through D are shown by
<Figure 3.6-65>. Conformance with these criteria is demonstrated by
<Table 3.6-7>.

For each line, no breaks are postulated in that portion of the main steam piping near the containment isolation valves in accordance with the criteria stated in <Section 3.6.2.1.7>. Conformance with these criteria is demonstrated by <Table 3.6-8>.

3.6.2.5.1.2   Postulated Pipe Breaks in Recirculation Piping System,
                   Including Residual Heat Removal Piping - Inside
                   Containment

The criteria for selection of postulated pipe breaks in the recirculation piping system inside containment are presented in <Section 3.6.2.1>. Postulated pipe break locations and types selected in accordance with these criteria are shown by <Figure 3.6-66> and <Figure 3.6-66a>. Conformance with these criteria is demonstrated by <Table 3.6-9>.

3.6.2.5.1.3   Postulated Pipe Breaks in Feedwater Piping System -
                   Inside Containment

The criteria for selection of postulated pipe breaks in the feedwater piping system inside containment are presented in <Section 3.6.2.1>. Postulated pipe break locations and types selected in accordance with these criteria are shown by <Figure 3.6-67>. Conformance with these criteria is demonstrated by <Table 3.6-10>.

3.6.2.5.1.4   Postulated Pipe Breaks in Emergency Core Cooling Piping
                   System - Inside Containment

The criteria for selection of postulated pipe breaks in the ECCS piping system inside containment are presented in <Section 3.6.2.1>. Postulated pipe break locations and types selected in accordance with these criteria are shown by <Figure 3.6-68>, <Figure 3.6-69a>, and <Figure 3.6-69b>. Conformance with these criteria is demonstrated by <Table 3.6-11>.

3.6.2.5.1.5   Postulated Pipe Breaks for Other Piping Systems –
               Inside Containment

The criteria for selection of postulated pipe breaks in other high
energy piping systems inside containment are presented in
<Section 3.6.2.1>. Postulated break locations and types selected in
accordance with these criteria are shown by <Figure 3.6-71>,
 <Figure 3.6-72>, <Figure 3.6-73>, and <Figure 3.6-74> for the RCIC head
spray, CRD supply line, RWCU system, and main steam drains,
respectively.

Small high energy piping may be assumed to break at each weld and
fitting, and each terminal end. Review of the jet impact and pipe whip
hazards is then done on this basis. If protection from breaks at every
such location is not practical, the stress analysis is reviewed to the
break location criteria of <Section 3.6.2.1>, and breaks are postulated
accordingly.

Restraints, shields and other measures necessary to ensure safe shutdown
in the event of each postulated break are provided, regardless of the
criterion by which the break was postulated.

3.6.2.5.1.6   Postulated Pipe Breaks for Piping Systems – Outside
               Containment

The criteria for selection of postulated pipe breaks in high energy
piping systems outside containment are presented in <Section 3.6.2.1>.
Postulated break locations and types selected in accordance with these
criteria are shown by <Figure 3.6-70a>, <Figure 3.6-75>,
<Figure 3.6-76>, <Figure 3.6-77>, <Figure 3.6-78>, <Figure 3.6-79>, and
<Figure 3.6-80>.

Small high energy piping may be assumed to break at each weld and
fitting, and each terminal end. Review of the jet impact and pipe whip

hazards is then done on this basis. If protection from breaks at every
such location is not practical, the stress analysis is reviewed to the
break location criteria and breaks are postulated accordingly.

Restraints, shields and other measures necessary to ensure safe shutdown
in the event of each postulated break are provided, regardless of the
criterion by which the break was postulated.

3.6.2.5.2   Implementation of Special Protection Criteria

3.6.2.5.2.1   Pipe Whip Restraints for Main Steam Piping System,
                   Including Reactor Core Isolation Cooling Piping -
                   Inside Containment

Pipe whip restraints provided for the main steam piping system are shown
by <Figure 3.6-65> and <Figure 3.6-70>. Using the analytical methods
described in <Section 3.6.2.2.1>, this system of restraints has been
found to prevent unrestrained pipe whip resulting from a postulated
rupture at any of the identified break locations. These restraints were
visually inspected during preoperational and startup testing for
adequate clearances to accommodate thermal expansion.

3.6.2.5.2.2   Pipe Whip Restraints for Recirculation Piping System,
                   Including Residual Heat Removal Piping - Inside
                   Containment

Pipe whip restraints provided for the recirculation system piping are
shown by <Figure 3.6-66> and <Figure 3.6-66a>. This system of
restraints has been found to prevent unrestrained pipe whip resulting
from a postulated rupture at any of the identified break locations.
These restraints were visually inspected during preoperational and
startup testing for adequate clearances to accommodate thermal
expansion.

3.6.2.5.2.3   Pipe Whip Restraints for Feedwater Piping
                System - Inside Containment

Pipe whip restraints provided for the feedwater piping system inside
containment are shown by <Figure 3.6-67>. This system of restraints has
been found to prevent unrestrained pipe whip resulting from a postulated
rupture at any of the identified break locations. These restraints were
visually inspected during preoperational and startup testing for
adequate clearance to accommodate thermal expansion.

3.6.2.5.2.4   Pipe Whip Restraints for Emergency Core Cooling Piping
                System - Inside Containment

Pipe whip restraints provided for the ECCS piping system inside
containment are shown by <Figure 3.6-68>, <Figure 3.6-69a>, and
<Figure 3.6-69b>. This system of restraints has been found to prevent
unrestrained pipe whip resulting from a postulated rupture at any of the
identified break locations. These restraints were visually inspected
during preoperational and startup testing for adequate clearances to
accommodate thermal expansion.

3.6.2.5.2.5   Pipe Whip Restraints for Other High Energy Piping
                Systems - Inside Containment

Pipe whip restraints are provided for other high energy piping systems
inside containment to prevent unrestrained pipe whip resulting from a
postulated rupture at any of the identified break locations. These
restraints were visually inspected during preoperational and startup
testing for adequate clearances to accommodate thermal expansion.

3.6.2.5.2.6   Pipe Whip Restraints for High Energy Piping Systems -
               Outside Containment

Pipe whip restraints provided for main steam and feedwater piping
systems outside containment in the safety-related steam tunnel area of
the auxiliary building are shown by <Figure 3.6-75> and <Figure 3.6-76>.
These restraints have been found to prevent unrestrained pipe whip in
the vicinity of safety-related structures, systems and components
resulting from a postulated rupture at any of the identified break
locations.

Pipe whip restraints are provided for other high energy systems (other
than main steam and feedwater) in the steam tunnel and auxiliary
building to prevent unrestrained pipe whip which would jeopardize
safety-related structures, systems and components. The system has been
found to prevent unrestrained pipe whip resulting from a postulated pipe
rupture at any of the identified break locations.

3.6.2.5.3   Summary of Results of Jet Effects Analysis

3.6.2.5.3.1   Jet Effects for Postulated Ruptures of Main Steam
               Piping System - Inside Containment

Fluid jet thrust for each of the postulated break locations in the main
steam piping are listed in <Table 3.6-12>. Structures, systems or
components essential to safe shutdown of the plant in the case of a
particular pipe break, and subject to impact by the steam jet from the
particular break, are discussed in the following paragraphs:

a.   High Pressure Core Spray Injection Pipe

     Main Steam line breaks which had impinged upon this line are no
     longer postulated.

b.   Standby Liquid Control System Injection Line Azimuth 240°.

The standby liquid control system is required to ensure the
capability to achieve less than k=1.0 reactivity, at cold shutdown
conditions, in the event of failure of control rod drives.

c.   Low Pressure Core Injection, Line A (LPCI A) Azimuth 45°.

The LPCI A line is required for a main steam break, loss of offsite
power and loss of Division 2 diesel.

The main steam jet impact loads noted in Items a, b and c, above, are
resolved as follows:

a.   High Pressure Core Spray Injection Pipe

Main steam line breaks which had impinged upon this line are no
longer postulated.

b.   Standby Liquid Control System Injection Line Azimuth 240°.

Shields were provided to protect this line from the effects of main
steam breaks. However, main steam line breaks which had impinged
upon this line are no longer postulated.

c.   Low Pressure Core Injection Line A (LPCI A) Azimuth 45°.

Main steam line breaks which had impinged upon this line are no
longer postulated.

3.6.2.5.3.2   Jet Effects for Postulated Ruptures of Recirculation
                  Piping System - Inside Containment

Fluid jet thrusts for each of the postulated break locations in the
recirculation piping system are listed in <Table 3.6-13>. Structures,
systems or components essential to safe plant shutdown in the case of a
particular pipe break, and subject to impact by the steam jet from the
particular break, are discussed in the following paragraphs:

a.   High Pressure Core Spray Piping

     <Figure 3.6-84> illustrates the physical arrangement, impact loads
     and jet shields.

b.   Control Rod Drive Bundles - Longitudinal Break

     Similar jets from four different postulated recirculation line
     breaks could impact any of the four CRD bundles at reactor pressure
     vessel Azimuths 74°, 106°, 254°, or 286°. <Figure 3.6-85>
     illustrates the physical arrangement and impact loads and jet
     shields.

The recirculation line jet impact loads noted in Items a and b, above,
are resolved as follows:

a.   High Pressure Core Spray Piping

     The impact on HPCS pipe of a jet from a ruptured recirculation
     discharge header was analyzed as both an impact load and as a
     steady-state load in combination with thermal, deadweight and
     seismic loads acting on the pipe simultaneously. At no time was
     the maximum allowable stress exceeded at any point in the impacted
     HPCS piping run.

Shields have also been provided to intercept these jets at the source.

b.   Control Rod Drive Bundles - Circumferential Break

A circumferential break in the recirculation discharge header connection at any of four locations, 60°, 120°, 240°, or 300°, was found to result in a jet impact that caused overstress of individual withdraw lines and exceeded the design capacity of the entire impacted CRD tube bundle supports if a jet shield were supported from the bundle. Jet shields are provided to intercept these jets at the source to prevent overstress of individual withdraw pipes or tube bundle supports as a result of the postulated event.

3.6.2.5.3.3   Jet Effects for Postulated Ruptures of Feedwater Piping
                 System - Inside Containment

Fluid jet thrusts for each of the postulated break locations in the feedwater piping system are listed in <Table 3.6-14>. Structures, systems or components essential to safe shutdown of the plant in the case of a particular pipe break, and which are jeopardized by the jet resulting from a particular break, are discussed in the following paragraphs:

a.   Control Rod Drive Bundles at Reactor Pressure Vessel Azimuths 74° and 286°

A jet shield is provided around the CRD bundle arrangement to prevent overstress of individual withdraw lines or tube bundle supports as a result of the postulated event.

A jet shield for the CRD bundle at Azimuth 74° protects against a
feedwater loop A rupture; a shield for the CRD bundle at
Azimuth 286° protects against a loop B rupture.

b.   Low Pressure Core Injection B Piping

Jet impact loading on LPCI B piping and valve operator was analyzed
both as an impact load and as a steady-state load in combination
with thermal, deadweight and seismic loads acting simultaneously.
At no time was the maximum allowable stress exceeded at any point
in the impacted LPCI B piping.

3.6.2.5.3.4   Jet Effects for Postulated Ruptures of Emergency Core
                 Cooling Piping System - Inside Containment

Fluid jet thrusts for each of the postulated break locations in the ECCS
piping system are listed in <Table 3.6-15>. Structures, systems or
components essential to safe plant shutdown in the case of a particular
pipe break, and which are jeopardized by the jet resulting from a
particular break, are discussed in the following paragraphs:

a.   High Pressure Core Spray Pipe

HPCS piping is subject to jet impact resulting from postulated
rupture of LPCI B piping. Both the HPCS isolation valve and pipe
support elements add to the total load resulting from jet
impingement. <Figure 3.6-90> illustrates physical arrangement and
jet shields.

b.   Low Pressure Core Injection B Pipe

LPCI B piping is subject to jet impact resulting from postulated
rupture of HPCS piping. The LPCI piping, isolation valve and pipe
supports are struck by the conically expanding jet as shown by

<Figure 3.6-91> which illustrates physical arrangement, impact loads and jet shields.

c.   Automatic Depressurization System (ADS) Valves, Air Lines and Accumulators.

Longitudinal breaks at the valve welds and upper elbows could cause jet impact loading on the ADS valves, or their air lines and accumulators, sufficient to compromise the required ADS capacity for the break size.

The ECCS line jet impact loads noted in Items a and b, above, are resolved as follows:

a.   High Pressure Core Spray Pipe

Jet impact loading on the HPCS piping was analyzed both as an impact load and as a steady-state load in combination with thermal, deadweight and seismic loads acting simultaneously. Jet shields were provided to intercept these jets at the source if the maximum allowable stress was exceeded at any point in the impacted HPCS piping.

b.   Low Pressure Core Injection B Pipe

Jet impact loading on the LPCI B piping was analyzed both as an impact load and as a steady-state load in combination with thermal, deadweight and seismic loads acting simultaneously. Jet shields were provided to intercept these jets at the source if the maximum allowable stress was exceeded at any point in the impacted LPCI B piping.

c.   Automatic Depressurization System (ADS) Valves, Air Lines and
     Accumulators.

     Shields were provided to intercept these jets at the source.

3.6.2.5.3.5   Jet Effects from Postulated Ruptures of Piping
                  Systems - Outside Containment

Fluid jet thrusts for each of the postulated high energy pipe breaks
outside containment are listed in <Table 3.6-16>. Structures, systems
or components essential to safe plant shutdown in the case of a
particular pipe break, and which could potentially be jeopardized by the
jet resulting from a particular break, are discussed in the following
paragraphs:

a.   Shield Building Wall

     A short term dynamic loading of the portion of the shield building
     wall forming the end wall of the auxiliary building steam tunnel
     results from the postulated full circumferential rupture of a main
     steam line outside the outermost moment and torsion limiting
     restraint. The turbine side of such a rupture is an unrestrained
     whipping pipe located within a nonsafety category structure, the
     steam tunnel. Motion of the whipping pipe is such that, for a
     portion of its movement, the jet strikes the shield building.
     <Figure 3.6-92> shows the approximate physical arrangement and the
     area struck by the jet at the position of maximum impact. Duration
     of the jet impact is less than 100 msec and the shape of the
     force-time curve is approximately sinusoidal.

b.   Main Steam Isolation Valve

     Should any one of the four main steam lines rupture immediately
     outside the outer pipe whip restraint, the outer main steam

isolation valve in the affected main steam line or in an adjacent
line may be impacted by the resulting jet. A short duration
impact, similar to the load time history described in Item a,
above, results from the whipping motion of the broken pipe as it
rises above the normal pipe centerline. <Figure 3.6-92>
illustrates the approximate physical arrangement for a typical
break and the loads involved.

c.   RCIC Steam Supply Piping

A short term dynamic loading of the RHR piping (Loop A or B)
downstream of E12-F052A and B results from the postulated full
circumferential rupture of the 10-inch RCIC steam supply line in
the auxiliary building. Motion of the whipping pipe is limited by
restraints RHR-6 and RHR-7 <Figure 3.6-70a>.

The jet impact loads noted in Items a through c, above, are resolved as
follows:

a.   Shield Building Wall

Dynamic loading of the shield building wall was analyzed in
conjunction with other thermal, deadweight and seismic loads acting
simultaneously. The loading was found to be within the capacity of
the structure.

b.   Main Steam Isolation Valve

Dynamic loading of a main steam isolation valve was analyzed as a
steady-state maximum loading and was superimposed on the equivalent
thermal, deadweight and seismic loads acting on both the valve
elements and the piping system. Resultant component and pipe
stresses were found to be acceptable.

c.   RCIC Steam Supply Piping

      Dynamic loading of the RHR and other piping and components was
      analyzed in conjunction with other thermal, deadweight and seismic
      loads acting simultaneously. The resultant loads were found to be
      acceptable.

3.6.2.5.4   Implementation of Guard Pipe Criteria

Systems for which guard pipes are used are listed in <Table 3.6-6>.
Details of guard pipe designs are shown by <Figure 3.6-55>,
<Figure 3.6-56>, <Figure 3.6-57>, <Figure 3.6-58>, <Figure 3.6-59>,
<Figure 3.6-60>, <Figure 3.6-61>, <Figure 3.6-62>, <Figure 3.6-63>, and
<Figure 3.6-64>. Guard pipes are discussed in <Section 3.6.2.4>.

3.6.3   REFERENCES FOR SECTION 3.6

1.   Gilbert Associates, Inc., "Analysis of Thrust and Jet Impingement
      Forces Using the JIP Program," Report No. TR 104P.

2.   GE Spec. No. 22A2625, "System Criteria and Applications for
      Protection Against the Dynamic Effects of Pipe Break."

3.   RELAP 3, "A Computer Program for Reactor Blowdown
      Analysis IN-1321," Reactor Technology TID-4500, issued June 1970.

4.   GE Report NEDE-10313, "PDA - Pipe Dynamic Analysis Program for Pipe
      Rupture Movement." (Proprietary Filing)

5.   Nuclear Services Corporation Report No. GEN-02-02, "Final Report
      Pipe Rupture Analysis of Recirculation System for 1969 Standard
      Plant Design."

6. Moody, F. J., "Fluid Reactor and Impingement Loads," Vol. 1, ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, pp 219-262, dated December 1973.

7. Gilbert Associates, Inc., "DYREC Program User's Manual," GAI Report No. 1866, July 1975; revised March 1976.

8. McDonnell-Douglas Automation Co., "ICES DYNAL User's Manual," dated September 1971.

9. Eiber, R. J., et. al., "Investigation of the Initiation and Extent of Ductile Pipe Rupture - Phase 1 Final Report - Task 17," Battelle Memorial Institute, BMI-1866, dated July 1969.

10. Biggs, J. M., "Introduction to Structural Dynamics," McGraw-Hill, dated 1964.

TABLE 3.6-1

HIGH ENERGY LINES[1][2]

System Number	System Designation
B-21	Main Steam - inside containment (SRV discharge piping excluded)
N-11	Main Steam - outside containment
N-27	Feedwater System
B-33	Recirculation System
N-22	Main Steam System Drains - including RCIC steam drain
E-51	Reactor Core Isolation Cooling System - steam supply from main steam Line "A" out to E51-F045 and E12-D0505
E-51	RCIC Head Spray - from RPV to E51-A0F066
G-33	Reactor Water Cleanup System
G-36	RWCU Filter/Demineralizer System
E-12	Low Pressure Core Injection Loops "A", "B" and "C" (RHR) - from RPV to E12-F041A, B & C
E-21	Low Pressure Core Spray - from RPV to E21-F006
E-22	High Pressure Core Spray - from RPV to E22-F005
C-11	Control Rod Drive Hydraulic System - Pump discharge side only
C-41	Standby Liquid Control Supply Line - from RPV to C41-F007
B-21	RPV Head Vent to Main Steam Line "A"
P-61	Auxiliary Steam System
M-29	Control and Computer Room Humidification System
E-12	Normal Shutdown - (from connection to the B33 System to E12-F009 and return from E12-F053 to the connection to the N27 System)

TABLE 3.6-1 (Continued)

NOTES:

[1] Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

    a.   maximum operating temperature exceeds 200°F, or

    b.   maximum operating pressure exceeds 275 psig

[2] High energy lines that are located in the turbine building, yard or other areas free from safety-related equipment and structures are not included in this list.

TABLE 3.6-2

MODERATE ENERGY LINES[1]

System Number	System Designation
P-43	Nuclear Closed Cooling System
P-50	Containment Vessel Chilled Water System
P-54	Fire Protection System
P-11	Condensate Transfer - Storage System
P-46	Turbine Building Chilled Water
P-47	Control Complex Chilled Water
R-44	Diesel Generator Starting Air (from receiver tank to start air admission valves)
N-71	Circulating Water System
N-26	Low Pressure Heater Drain System
N-23	Condensate Filtration System
N-24	Condensate Demineralizer System
G-50	Liquid Radwaste System
G-41	Fuel Pool Cooling and Cleanup System
P-71	Potable Water System
P-41	Service Water System
P-20	Makeup Water System
E-12	Residual Heat Removal System - except high energy <Table 3.6-1>
P-55	Building Heating Hot Water System
N-21	Condensate System
N-11	Condenser Air Ejector Steam System
P-12	Condensate Seal Water System
G-40	Alternate Decay Heat Removal System

TABLE 3.6-2 (Continued)

System Number	System Designation
P-21	Two Bed Demineralizer Water System
P-22	Mixed Bed Demineralizer Water System
E-51	RCIC - Except High Energy Lines <Table 3.6-1>
P-45	Emergency Service Water System
P-48	Service Water and Emergency Service Water Chlorination System
P-49	Emergency Service Water Screen Wash System

NOTE:

[1] Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a. maximum operating temperature is 200°F or less, and

b. maximum operating pressure is 275 psig or less

TABLE 3.6-3

<u>SYSTEMS TO BE PROTECTED AGAINST
PIPING FAILURES</u>

a.   Systems Required to Achieve Safe Cold Shutdown:

<u>System No.</u>	<u>System Designation</u>
B-21	Nuclear Boiler-reactor coolant pressure boundary
B-21C	Nuclear Boiler - Automatic Depressurization System, including local accumulators
B-21H	Nuclear Boiler - Nuclear Steam Supply Shutoff System
B-21	Nuclear Boiler - Main Steam Containment Isolation Valves
B-33	Recirculation System - reactor coolant pressure boundary only
C-11	Control Rod Drive System - withdraw lines
C-41	Standby Liquid Control System
C-51	Neutron Monitoring System - trip logic interfacing with Reactor Protection System only
C-61	Remote Shutdown System
C-71	Reactor Protection System
E-12	Low Pressure Coolant Injection "A", "B" and "C"
E-12	RHR System - Shutdown Cooling Mode
E-21	Low Pressure Core Spray System
E-22	High Pressure Core Spray System
E-31	Leak Detection System
E-51	Reactor Core Isolation Cooling (available for safe shutdown, but not required)
E-53	Containment Isolation System - reactor coolant pressure boundary isolation portions

TABLE 3.6-3 (Continued)

System No.	System Designation
G-33	Reactor Water Cleanup System - reactor coolant pressure boundary, through outboard containment isolation valve on suction line
G-43	Suppression Pool Makeup System
M-23	MCC, Switchgear and Miscellaneous Electrical Equipment Area HVAC Systems
M-24	Battery Room Exhaust System
M-28	Emergency Closed Cooling Pump Area Cooling System
M-32	Emergency Service Water Pump House Ventilation System
M-39	ECCS Pump Room Cooling System
M-43	Diesel Generator Building Ventilation System
N27, B-21	Feedwater Containment Isolation Valves
N-27	Feedwater - reactor coolant pressure boundary
P-42	Emergency Closed Cooling System
P-45	Emergency Service Water System
P-47	Control Complex Chilled Water System
P-49	Emergency Service Water Screen Wash System
R-42	D.C. System - batteries, chargers and switchboards
R-43 (R44, R45, R46, R47, & R48)	Standby Diesel Generators - and auxiliary systems

b. Systems Required to Mitigate the Consequences of Pipe Ruptures (to within licensing basis offsite dose limits):

System No.	System Designation
E-15	Containment Spray (backup for suppression pool only)
E-53	Containment Isolation System
M-15	Annulus Exhaust Gas Treatment System

TABLE 3.6-3 (Continued)

System No.	System Designation
M-16	Drywell Vacuum Relief System
M-17	Containment Vacuum Relief System
N-27	Feedwater Leakage Control System
P-57	Safety-Related Instrument Air
T-23	Reactor Building Containment System
-	Containment Isolation Valves on all systems

c. Systems Required to Maintain Control Room Habitability and Access to Other Areas Required to Achieve and Maintain Safe Shutdown:

System No.	System Designation
M-25	Control Room HVAC System
M-26	Control Room Emergency Recirculation System
P-47	Control Complex Chilled Water System

d. Systems Required to Maintain Safe Long Term Cold Shutdown:

System No.	System Designation
G-41	Fuel Pool Cooling and Cleanup System
M-51	Combustible Gas Control System
P-57	Safety-Related Instrument Air

e. Other Items Requiring Protection:

- Instrument lines, control wiring and electrical cabling serving the above systems including main control room, remote shutdown panel, panels, and racks.

Safety-related structures housing the above systems and components

TABLE 3.6-4

RESTRAINT DATA[1]

Pipe Size (Inches)	Rest Load Direction	C_2	n	Limit Δ Restraint	Initial Clearance	Effective Clearance	Total Clearance
12	0°	27,733	0.24	6.129	4	1.941	5.941
12	90°	14,795	0.401	9.063	4	12.247	16.247
16	0°	109,265	0.24	6.278	4	1.934	5.934
16	90°	62,599	0.377	8.978	4	12.187	16.187
24	0°	102,228	0.24	8.222	4	1.984	5.084
24	90°	55,531	0.375	11.972	4	13.685	17.685
24	38°[2]	109,888	0.24	5.588	4	5.698	9.698
24	52°[2]	109,835	0.24	5.473	4	8.462	12.462

NOTES:

[1] General restraint data for 1 bar of a restraint.

   $F = C_2 (\Delta \text{ restraint})^n$, where Δ restraint = σ pipe - total clearance.

[2] Applies to restraint RCR 3 only.

TABLE 3.6-5

COMPARISON OF PDA AND NUCLEAR SERVICES CORPORATION CODE

Break ID No. [1]	Restraint ID No. [1]	Force Vector (degrees) [2]	No. Bars		Load (Kips)		Restraint Deflection (in.)		Percent of Design Restraint Deflection		Pipe Deflection (in.)	
			PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC
RC1J	RCR1	0	5	5	803.3	788.3	6.6	7.9	79.9	96.4	17.7	15.6
RC2LL	RCR1	90	5	5	766.4	458.4	15.0	7.5	125.2	62.6	35.8	24.5
RC3LL	RCR2	0	6	6	747.0	639.7	2.3	3.7	27.7	45.4	17.2	20.1
RC3LL	RCR2	90	6	6	796.6	780.3	10.2	10.5	85.4	88.1	41.5	43.0
RC4LL	RHR3	0	5	5	846.0	838.4	8.2	8.1	99.2	98.0	18.9	16.4
RC4LL	RCR3	52	8	8	1,319.0	1,073.9	5.4	4.2	99.2	76.9	23.4	17.3
RC4CV	RCR3	38	8	8	1,260.7	1,275.0	4.5	5.6	80.4	99.9	22.6	18.7
RC6AV	RCR3	38	8	8	928.5	722.5	1.3	1.8	22.5	31.7	23.7	95.4
RC7J	RCR7	0	6	6	953.3	801.6	6.3	5.8	76.4	70.1	16.5	21.6
RC8LL	RCR6	90	4	4	599.0	NA[3]	8.3	NA[3]	69.2	NA[3]	26.8	NA[3]
RC8LL	RCR7	90	6	6	895.0	NA[3]	8.2	NA[3]	68.2	NA[3]	29.3	NA[3]
RC9CV	RCR6	0	4	4	575.8	520.2	4.2	5.5	50.6	67.3	13.2	14.6
RC9LL	RCR8	90	6	6	830.2	546.8	11.4	6.8	95.3	56.9	36.7	26.2
RC11A	RCR8	90	6	6	818.3	493.6	11.0	6.0	91.7	50.1	31.4	23.7
RC12	RCR9	0	6	6	NA[3]	832.9	NA[3]	6.3	NA[3]	76.9	NA[3]	15.7
RC13	RCR10	0	4	4	668.4	478.0	5.9	3.7	93.5	58.4	13.4	10.4
RC14CV	RCR20	0	8	8	285.0	309.6	2.8	5.9	46.3	95.9	15.5	14.0
RC14LL	RCR20	90	8	8	116.3	129.9	1.0	3.4	10.5	37.1	22.0	23.6
RC16	RCR11	0	4	4	687.4	518.4	6.6	4.4	105.1	69.9	15.4	10.2

NOTES:

[1] See <Figure 3.6-49>
[3] NA - Data Not Available

[2] Force Vector Represented as -

TABLE 3.6-6

SYSTEMS REQUIRING CONTAINMENT BOUNDARY GUARD PIPES

System	Nominal Pipe Size (in.)	Penetration Type
Main Steam <Section 3.6.2.4.1>	26	K
Feedwater	20	Q
RHR Suction	20	J
RCIC Steam	10	J
RCIC Head Spray	6	J
RWCU Suction	6	J
Main Steam Drains	3	J

TABLE 3.6-7

SUMMARY OF MAIN STEAM PIPING DESIGN ANALYSIS STRESSES AT BREAK LOCATIONS[2]

a.   Main Steam "A" Piping:

Break I.D. No.[2][4]	Node No.[3]	STRESS RATIOS[1]			Usage Factor U	Break Type	Break Basis Section No.[5]
		EQ. (10) $2.4S_m$	EQ. (12) $2.4S_m$	EQ. (13) $2.4S_m$			
SA1	001	0.560	0.275	0.354	0.00	Circ.	Terminal End
SA2A (SA1A)	002	1.408	1.024	0.479	0.02	Circ.	<Section 3.6.2.1.5.a.2.(c)>
SA2LL (SA1LL)	002	1.408	1.024	0.479	0.02	Long.	<Section 3.6.2.1.5.a.2.(c)>

b.   Main Steam "C" Piping ("B" is a mirror image of Main Steam "C"):

Break I.D. No.[2]	Node No.[3]	STRESS RATIOS[1]			Usage Factor U	Break Type	Break Basis Section No.
		EQ. (10) $2.4S_m$	EQ. (12) $2.4S_m$	EQ. (13) $2.4S_m$			
SC1	001	0.512	0.202	0.363	0.00	Circ.	Terminal End

c. Main Steam "D" Piping:

Break I.D. No.[2]	Node No.[3]	STRESS RATIOS[1]			Usage Factor U	Break Type	Break Basis Section No.
		EQ. (10) $2.4S_m$	EQ. (12) $2.4S_m$	EQ. (13) $2.4S_m$			
SD1	001	0.542	0.245	0.352	0.00	circ.	Terminal End

TABLE 3.6-7 (Continued)

<u>NOTES</u>:

[1] These are the ratios of calculated stresses (by code equation) over allowable stresses.
[2] See <Figure 3.6-65> for postulated break locations and break identification.
[3] See <Figure 3.6-65a>, <Figure 3.6-65b>, <Figure 3.6-65c>, <Figure 3.6-65d>, <Figure 3.6-65e>, and <Figure 3.6-65f> for node locations.
[4] Perry unique break location designations in parentheses.
[5] Terminal end as defined in <Section 3.6.2.1.5.a.1>.

TABLE 3.6-8

SUMMARY OF PIPE DESIGN ANALYSIS STRESSES IN PORTION OF MAIN STEAM
LINES BETWEEN PRIMARY CONTAINMENT ISOLATION VALVES

| Line | Node[1] | Stress (psi)[2] | | | Cumula-tive Usage Factor | $2.4S_m$[3] (psi) | $3.0S_m$[3] (psi) |
		EQ. (10)	EQ. (12)	EQ. (13)			
Main Steam Line A	28	25,726	8,380	21,294	0.01	42,480	53,100
Main Steam Lines B and C	28	26,956	10,347	21,906	0.01	42,480	53,100
Main Steam Line D	29	27,867	11,697	21,209	0.01	42,480	53,100

NOTES:

[1] See <Figure 3.6-65a>, <Figure 3.6-65c>, and <Figure 3.6-65e>, for pipe node locations.

[2] Equation (10), (12) and (13) stresses and cumulative usage factors calculated in accordance with ASME Code, Section III, Subarticle NB-3650.

[3] Design stress intensity values, S_m, selected in accordance with Appendix 1 to ASME Code, Section III.

TABLE 3.6-9

SUMMARY OF RECIRCULATION PIPING DESIGN ANALYSIS STRESSES AT BREAK LOCATIONS[2]

Break I.D. No.[2]	Node No.[3]	STRESS RATIOS[1]			Usage Factor U	Break Type	Break Basis Section No.[4]
		EQ. (10) 2.4 S_m	EQ. (12) 2.4 S_m	EQ. (13) 2.4 S_m			
RS1	5	0.82	0.21	0.50	0.00	Circ.	Terminal End
RD1	308	0.86	0.19	0.51	0.00	Circ.	Terminal End
RD2	328	1.08	0.29	0.51	0.00	Circ.	Terminal End
RD3	348	1.18	0.15	0.55	0.00	Circ.	Terminal End
RD4	368	1.07	0.25	0.50	0.00	Circ.	Terminal End
RD5	388	0.95	0.29	0.49	0.00	Circ.	Terminal End
RD7LL	216	1.64	0.52	0.82	0.12	Long.	<Section 3.6.2.1.5>
RD7	216	1.64	0.52	0.82	0.12	Circ.	<Section 3.6.2.1.5>

NOTES:

[1] These are the ratios for calculated stresses (by code equation) over allowable stresses.
[2] See <Figure 3.6-66> and <Figure 3.6-66a> for postulated break locations and break identification.
[3] See <Figure 3.6-66b> for node locations.
[4] Terminal ends as defined in <Section 3.6.2.1.5.a.1>.

TABLE 3.6-10

FEEDWATER PIPING SYSTEM OPERATING STRESSES[1] AT BREAK LOCATIONS INSIDE CONTAINMENT

Break I.D. No.[2]	Node No.[3]	EQ. (10) $2.4\ S_m$	EQ. (12) $2.4\ S_m$	EQ. (13) $2.4\ S_m$	Usage Factor	Break Type	Break Basis Section No.[4]
W1	435	1.117	0.572	0.595	0.03	Circ.	Terminal End
W3LL	125	1.698	0.745	1.015	0.20	Long.	<Section 3.6.2.1.5.a.2.c>
W4	335	0.935	0.479	0.665	0.02	Circ.	Terminal End
W5A	333	1.579	0.998	0.580	0.10	Circ.	<Section 3.6.2.1.5.a.2.c>
W5LL	333	1.579	0.998	0.580	0.10	Long.	<Section 3.6.2.1.5.a.2.c>
W6LL	110	2.309	1.137	1.158	0.64	Long.	<Section 3.6.2.1.5.a.2.c>
W7	275	1.092	0.539	0.535	0.06	Circ.	Terminal End
W10LL	95	1.640	0.650	1.192	0.20	Long.	<Section 3.6.2.1.5.a.2.c>

NOTES:

[1] Ratio of calculated to allowable stress.
[2] Terminal end as defined in <Section 3.6.2.1.5.a.1>.

TABLE 3.6-11

EMERGENCY CORE COOLING SYSTEM OPERATING STRESSES AT BREAK LOCATIONS

Break[2] I.D. No.	Node No.	EQ. (10)[1] PSI	EQ. (12)[1] PSI	EQ. (13)[1] PSI	Usage Factor	Break Type	Break Basis Section No. [3]
LPA 1	1	118,285	14,497	59,171	0.46	Circ.	Terminal End
LPA3B	8	103,641	35,358	26,373	0.06	Circ.	\<Section 3.6.2.1.5.a.5.b>
LPB4B	25	110,489	8,380	58,064	0.21	Circ.	\<Section 3.6.2.1.5.a.2.c>
LPB4BLL	25	110,489	8,380	58,064	0.21	Long.	\<Section 3.6.2.1.5.a.2.c>
LPB4A	28	104,168	6,004	55,771	0.15	Circ.	\<Section 3.6.2.1.5.a.2.c>
LPB4ALL	28	104,168	6,004	55,771	0.15	Long.	\<Section 3.6.2.1.5.a.2.c>
LPB1	32	121,234	11,489	57,640	0.63	Circ.	Terminal End
LPC1	29	121,100	6,835	58,164	0.48	Circ.	Terminal End
LPC4A	26	100,721	3,949	55,446	0.12	Circ.	\<Section 3.6.2.1.5.a.2.c>
LPC4ALL	26	100,721	3,949	55,446	0.12	Long.	\<Section 3.6.2.1.5.a.2.c>
LPC4B	23	115,229	5,277	56,683	0.23	Circ.	\<Section 3.6.2.1.5.a.2.c>
LPC4BLL	23	115,229	5,277	56,683	0.23	Long.	\<Section 3.6.2.1.5.a.2.c>
LPA5	4	59,905	6,796	26,674	0.14	Circ.	\<Section 3.6.2.1.5.a.2.c>
LPA5LL	4	59,905	6,796	26,674	0.14	Long.	\<Section 3.6.2.1.5.a.2.c>
LPA6LL	SW	82,924	10,583	27,985	0.20	Long.	\<Section 3.6.2.1.5.a.2.c>

TABLE 3.6-11 (Continued)

Break[2] I.D. No.	Node No.	EQ. (10)[1] PSI	EQ. (12)[1] PSI	EQ. (13)[1] PSI	Usage Factor	Break Type	Break Basis Section No. [3]
HC1	27	90,326	6,956	41,302	0.37	Circ.	Terminal End
HC4	C 24	84,930	5,673	48,155	0.19	Circ.	<Section 3.6.2.1.5.a.2.c>
HC4LL	C24	84,930	5,673	48,155	0.19	Long.	<Section 3.6.2.1.5.a.2.c>
HC5A	24	80,029	1,889	54,268	0.14	Circ.	<Section 3.6.2.1.5.a.2.c>
HC5ALL	24	80,029	1,889	54,268	0.14	Long.	<Section 3.6.2.1.5.a.2.c>
HC5B	21	89,729	5,463	55,213	0.16	Circ.	<Section 3.6.2.1.5.a.2.c>
HC5BLL	21	89,729	5,463	55,213	0.16	Long.	<Section 3.6.2.1.5.a.2.c>
LC1	27	62,851	11,628	33,041	0.07	Circ.	Terminal End
LC2A	26	68,654	28,549	16,470	0.10	Circ.	<Section 3.6.2.1.5.a.2.c>
LC2LL	26	68,070	28,100	15,858	0.09	Long.	<Section 3.6.2.1.5.a.2.c>
LC6	21	69,439	2,965	39,245	0.05	Circ.	<Section 3.6.2.1.5.a.2.c>
LC6LL	21	69,439	2,965	39,245	0.05	Long.	<Section 3.6.2.1.5.a.2.c>

NOTES:

[1] 3 S_m = 53,100 psi.

[2] See <Figure 3.6-18>, <Figure 3.6-69a>, and <Figure 3.6-69b> for postulated break locations and break identification in LPCI, LPCS and HPCS, respectively.

[3] Terminal end is as defined in <Section 3.6.2.1.5.a.1>.

TABLE 3.6-12

FLUID BLOWDOWN THRUST TIME HISTORIES FOR
MAIN STEAM PIPING SYSTEM

a.   Line A - Inside Containment (for NSSS Design and Analysis):

Break Location[1][2]	Type Of Break	Side Of Break	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
SA1	Circ.	Turbine	446	312	208	.0037	.0988
SA2A (SA1A)	Circ.	Vessel	446	446	497	.00187	.01227
SA2A (SA1A)	Circ.	Turbine	446	312	208	.0037	.0988
SA2LL (SA1A)	Long.	-	446	446	519	.00122	.00263

TABLE 3.6-12 (Continued)

b.   Lines B, C - Inside Containment (For NSSS Design and Analysis):

Break Location[1][2]	Type Of Break	Side Of Break	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
SC1	Circ.	Turbine	446	312	208	.0037	.0948

c.   Line D - Inside Containment (for NSSS Design and Analysis):

Break Location[1][2]	Type Of Break	Side Of Break	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
SD1	Circ.	Turbine	446	312	208	.0037	.0988

TABLE 3.6-12 (Continued)

d. 26″ Breaks - Inside Containment (for BOP Design Analysis):[3]

Flow Element Side of Break[4]

Time (seconds)	Thrust (Kips)
0-.001	0-450
.001-.003	450
.003-.009	315
.009-00	186

Unrestricted Side of Break[2]

Time (seconds)	Thrust (Kips)
0-.001	0-450.
.001-.091[5]	305.

e. 28″ Breaks - Outside Containment[6]

Break	Time (sec)	Thrust (Kips)
SA-2, SA-3, SA-4, SA-5, SA-6, SA-7, SA-8, SA-9		
Longitudinal Break (25.15″ I.D.)	0.-.0118	0 - 509
	.0118 - .24	509
	.25 - 00	439
Double-Ended Break		
Reactor Side (25.15″ I.D.)	0.-.001	0 - 509
	.001 - .31	509
	.31 - 00	425
Turbine Side (25.15″ I.D.)	0.-.001	0 - 509
	.001 - .31	509
	.31 - 00	366

NOTES:

[1] See <Figure 3.6-65> for postulated break locations and break identifications.

[2] Perry unique break location designations in parenthesis.

[3] See <Figure 3.6-65> for identification of postulated break locations.

[4] I.D. of piping is 23.65″. Credit is taken for the main steam flow element (I.D. = 12.125″) on one side of full circumferential breaks.

[5] Will decrease after this time.

[6] See <Figure 3.6-75> for identification of postulated break locations.

TABLE 3.6-13

FLUID BLOWDOWN THRUST TIME HISTORIES FOR
RECIRCULATION PIPING SYSTEM

Break Location[1]	Type Of Break	Side Of Break	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
RD1	Circ.	Pump	135	115	128	.00183	.02733
RD2	Circ.	Pump	135	115	128	.00183	.02733
RD3	Circ.	Pump	135	115	128	.00183	.02733
RD4	Circ.	Pump	135	115	128	.00183	.02733
RD5	Circ.	Pump	135	115	128	.00183	.02733
RD7LL	Long.	-	214	189	196	.00036	.03348
RD7	Circ.	Pump	135	101	115	.00036	.0238
RD7	Circ.	Vessel	135	93	52	.00212	.01978
RS1	Circ.	Pump	323	285	149	.00178	.08104

NOTE:

[1] See <Figure 3.6-66> for identification of postulated break locations.

TABLE 3.6-14

<u>FLUID BLOWDOWN THRUST TIME HISTORIES FOR
FEEDWATER PIPING SYSTEM</u>

This information is presented by <Figure 3.6-95>, <Figure 3.6-96>,
<Figure 3.6-97>, and <Figure 3.6-98>.

TABLE 3.6-15

FLUID BLOWDOWN THRUST TIME HISTORIES FOR
EMERGENCY CORE COOLING PIPING SYSTEM

Time (sec)	Thrust (kips)
0-0.001	0 - 128
0.001-∞	128

TABLE 3.6-16

BLOWDOWN THRUSTS - HIGH ENERGY PIPE
BREAKS OUTSIDE CONTAINMENT

System	Line Size (in.)	Break Type	Initial Blowdown Thrust (lbs)	Steady-State Thrust (lbs)	Remarks
Main Steam	28	Long.	509,000	439,000	See Note[2]
Main Steam	28	Circ.	509,000	425,000	Reactor side[3]
Main Steam	28	Circ.	509,000	366,000	Turbine side[3]
Feedwater	20	Circ.	265,000	99,000	Pump side[4]
Feedwater	20	Long.	265,000	99,400	See Note[5]
Feedwater	36	Long.	1,200,000	390,000	See Note[6]
Feedwater	36	Circ.	1,200,000	390,000	See Note[8]
Main Steam Drains	1-1/2	Circ.	1,720	See Note[7]	
Main Steam Drains	2	Circ.	2,740	See Note[7]	
Main Steam Drains	3	Circ.	6,610	See Note[7]	
RWCU	4	Circ.	14,000	See Note[7]	
RWCU	4	Long.	14,000	See Note[7]	
RWCU	6	Circ.	32,250	See Note[7]	
RWCU	6	Long.	32,250	See Note[7]	
CRD Supply	2-1/2	Circ.	13,000	100	
Auxiliary Steam	1-1/2	Circ.	340	See Note[7]	
Auxiliary Steam	3	Circ.	1,422	See Note[7]	
Auxiliary Steam	4	Circ.	2,450	See Note[7]	
Auxiliary Steam	4	Long.	2,450	See Note[7]	
Auxiliary Steam	10	Circ.	15,181	See Note[7]	
Auxiliary Steam	10	Long.	15,181	See Note[7]	
RCIC Steam Supply	10	Circ.	147,600	88,000	
Reactor Core Isolation Cooling	4	Circ.	17,000	13,400	

TABLE 3.6-16 (Continued)

NOTES:

[1] (Deleted)
[2] Break number SA2LL, <Figure 3.6-75> is typical.
[3] Break number SA3, <Figure 3.6-75> is typical.
[4] Break number WA1 and WB1, <Figure 3.6-76> is typical.
[5] Break number WA2LL and WB2LL, <Figure 3.6-76> is typical.
[6] Break number W9LL, <Figure 3.6-76> is typical.
[7] Magnitude of the steady-state blowdown thrust is a function of the location of the break relative to the pressure reservoir. For design pressure, the initial thrust is assumed to govern jet effects analysis.
[8] Break number W9, <Figure 3.6-76> is typical.

TABLE 3.6-17

PIPE WHIP ANALYSIS EMPLOYING STRAIN
ENERGY METHODS OR COMPONENT DAMAGE STUDIES

Impacting System	Impacted System or Component	Description of Protection or Qualification
B-21 Main Steam	Restraints	Energy-absorbing U-bolt or frame restraints inside containment - <Section 3.6.2.3.3>
B-33 Recirculation	Restraints	Energy-absorbing U-bolt or frame restraints inside containment - <Section 3.6.2.3.3>
N-27 Feedwater	Restraints	Energy-absorbing U-bolt or frame restraints inside and outside containment - <Section 3.6.2.3.3>
E-12, 21, 22 ECCS Lines	Restraints	Energy-absorbing U-bolt or frame restraints inside and outside containment - <Section 3.6.2.3.3>
E-51 RCIC Steam	Restraints	Energy-absorbing U-bolt or frame restraints inside and outside containment - <Section 3.6.2.3.3>
Other High-Energy Lines	Restraints	Energy-absorbing U-bolt restraints inside containment - <Section 3.6.2.3.3>
M-29 Control Rm. & Computer Humidifier	M-23 HVAC Duct M-25 HVAC Duct	Low pressure steam line impacts safety-related duct with negligible energy
N-11 Main Steam in Turbine Building Steam Tunnel	Nonsafety Steam Tunnel	No impacted safe shutdown components
N-27 Feedwater in Turbine Building Steam Tunnel	Nonsafety Steam Tunnel	No impacted safe shutdown components

TABLE 3.6-18

SUMMARY OF SAFETY CLASS 1 HIGH ENERGY PIPING SYSTEM
OPERATING STRESS EXCEEDING BREAK POSTULATION CRITERIA

Sub System	Node	Break[2] I.D. No.	Stress Ratio[1]			Usage Factor
			EQ. (10)	EQ. (12)	EQ. (13)	
N27						
	125	W3LL	1.698	0.745	1.015	0.20
	333	W5A	1.579	0.998	0.580	0.10
	333	W5ALL	1.579	0.998	0.580	0.10
	110	W6LL	2.309	1.137	1.158	0.64
	95	W10LL	1.640	0.650	1.192	0.20
	67	W11	0.839	0.072	1.184	0.40
	67	W11LL	0.839	0.072	1.184	0.40
B21	02	SA2A(SA1A)	1.408	1.024	0.479	0.02
	02	SA2LL(SA1LL)	1.408	1.024	0.479	0.02
B33	216	RD7LL	1.64	0.52	0.82	0.12
	216	RD7	1.64	0.52	0.82	0.12
1E12G09	01	LPA1	2.784	0.341	1.393	0.46
	04	LPA5	1.410	0.160	0.628	0.14
	SW	LPA6LL	1.952	0.249	0.659	0.20
1E12G10	25	LPB4B	2.601	0.197	1.367	0.21
	28	LPB4A	2.452	0.141	1.313	0.15
	32	LPB1	2.854	0.270	1.357	0.63
1E12G11	29	LPC1	2.851	0.161	1.369	0.48
	26	LPC4A	2.371	0.093	1.305	0.12
	23	LPC4B	2.713	0.124	1.334	0.23
1E22G04	27	HC1	2.126	0.164	0.966	0.37
	C24	HC4	1.999	0.134	1.134	0.19
	24	HC5A	1.884	0.044	1.277	0.14
	21	HC5B	2.112	0.129	1.300	0.16
1C41G05	7	SL5	2.240	0.314	1.098	0.32
	9	SL7	1.878	0.161	1.114	0.11
	10	SL14	1.786	0.101	1.134	0.07
	8	SL6	1.879	0.216	1.031	0.11
	27	SL13	1.917	0.160	0.980	0.11

TABLE 3.6-18 (Continued)

Sub System	Node	Break I.D. No.	Stress Ratio[1] EQ. (10)	EQ. (12)	EQ. (13)	Usage Factor
1E51G07	46	HS1	1.958	0.106	0.921	0.27
	1	HS2	2.185	0.091	1.262	0.79
	4	HS3	2.176	0.055	1.263	0.75
	99	HVL7	1.648	0.056	0.860	0.19
	101	HVL8	1.342	0.053	0.942	0.18
	81	HVL3	0.987	0.00	0.00	0.26
	82	HVL16	0.843	0.00	0.00	0.10
1G33G01						
	800	RW9, RW10 RW128	1.87	0.34	1.92	0.05
	5110	RW229	1.75	0.90	1.69	0.07
	5145	RW238	1.10	0.90	0.36	0.14
	1210	RW303	1.46	0.79	1.50	0.12

NOTES:

[1] Stress ratio is the calculated stress divided by 2.4 S_m

[2] Perry unique break locations designations in parenthesis.

3.7   SEISMIC DESIGN

3.7.1   SEISMIC INPUT

Geologic and seismologic surveys of the site were conducted to establish
two design earthquakes with different intensities of ground motion.
These are the operating basis earthquake (OBE) and the safe shutdown
earthquake (SSE).

The OBE is postulated to be the earthquake which could reasonably be
expected to affect the plant site during the operating life of the
plant. The OBE produces the vibratory ground motion for which the
Seismic Category I structures, systems and components are designed to
remain operational without undue risk to the health and safety of the
public. The OBE is considered to be a modified Mercalli Intensity VI as
measured at the site <Section 2.5.2.7>.

The SSE represents the strongest vibratory ground motion earthquake for
which these features (as mentioned for OBE) are, as a minimum, designed
to remain functional. The SSE is considered to be a modified Mercalli
Intensity VII as measured at the site <Section 2.5.2.4> and
<Section 2.5.2.6>.

These Seismic Category I structures, systems and components, and the
seismically analyzed systems and components of the plant are necessary
to assure: (1) the integrity of the reactor coolant pressure boundary,
(2) the capability to shut down the reactor and maintain it in a safe
shutdown condition or (3) the capability to prevent or mitigate the
consequences of accidents which could result in potential offsite
exposures comparable to the guideline exposure of <10 CFR 100> or
<10 CFR 50.67> (future revisions to design basis analyses that compare
consequences to 10 CFR 100 will be updated to <10 CFR 50.67>).

The design earthquakes, OBE and SSE, for the plant are specified by OBE and SSE design response spectra. These criteria are based on the plant site geologic investigations and seismologic recommendations as

discussed in <Section 2.5>. These spectra represent earthquake ground
motions which are potentially damaging to structures. While these
spectra could be exceeded by ground motion "spikes" above 10 Hz such as
those caused by the January 31, 1986, earthquake, extensive
investigations concerning the effects of these high-frequency motions,
both from structure/equipment evaluations as well as seismological
considerations, demonstrate the adequacy of the spectra used for design.
Further discussion on this subject is included in <Section 3.7.4.4.2>.

3.7.1.1   Design Response Spectra

Design response spectra for the SSE and OBE, as shown in <Figure 3.7-1>,
<Figure 3.7-2>, <Figure 3.7-3>, and <Figure 3.7-4>, comply with
<Regulatory Guide 1.60>. As shown in these figures, the vertical
component is 2/3 of the horizontal component in the frequency region
lower than 2.5 cps and the vertical and horizontal components are equal
in the frequency regions higher than 3.5 cps. These have been developed
as described in <Section 2.5.2>, which also contains the following
information:

a.   Historical data showing the distance and depth of reported
     earthquakes relative to the plant site.

b.   No earthquake record exists at or near the plant site for
     calculating amplification factors, other than as discussed in
     <Section 3.7.1> for high frequency ground motions.

c.   Earthquake duration is estimated as 10 seconds.

Safety class structures are founded in shale or on materials with
equivalent seismic properties (see <Section 3.7.1.4> for further
discussion), and hence, no site dependent analysis is used. A 12-inch
layer of porous concrete is used between the shale and foundation slabs

of safety class structures. The porous concrete has a modulus of elasticity in excess of 1.2×10^6 psi and a minimum shear wave velocity of 4,400 fps, which is equivalent to that of the underlying shale.

3.7.1.2   Design Time History

The response spectra derived from synthetic earthquake time-motion records are as shown on <Figure 3.7-5>, <Figure 3.7-6>, <Figure 3.7-7>, <Figure 3.7-8>, <Figure 3.7-9>, and <Figure 3.7-10>. The synthesized OBE time-history accelograms are shown in <Figure 2.5-92>, <Figure 2.5-93>, and <Figure 2.5-94>. Since a time history simulating ground design response spectrum with 1 percent damping is not used, the comparison is not shown in these figures.

The method of time history generation is as described in <Section 2.5.2.6.1>. The spectrum values were calculated for each damping value at 200 points equally spaced on a logarithmic scale between 0.02 sec and 4 sec. The same time history was generated to match spectra with 2, 5, 7, and 10% damping values. Thus, many iterations were required to match all four damping values at 800 calculated points. With so many points calculated to match the target design response spectra, there should be no concern about values in between calculated points.

3.7.1.3   Critical Damping Values

The specific percentage of critical damping values for safety class structures and soil-structure interaction are as shown in <Table 3.7-1>. These values comply with <Regulatory Guide 1.61>. A weighted damping technique described in (Reference 1) is used. Theoretically, the weighted damping technique considers the hysteretic and the viscous soil damping contributions. The hysteretic soil damping is strain dependent and is determined from tests as reported in <Section 2.5.2>. The viscous soil damping is due to radiational energy lost to the

semi-infinite half space. For the seismic analysis used in this plant, the hysteretic damping in combination with a small percentage of theoretical viscous damping is used as a conservative approach.

The justification of using viscous damping was demonstrated in the NRC <NUREG/CR-1665> report, which compares the calculated responses vs. observed responses. The calculated response value was based on 100% of the viscous damping and was 2 to 9 times higher than the original design value, yet there was essentially no damage to the El Centro Steam Power Plant due to the 1979 El Centro earthquake.

3.7.1.4   Supporting Media for Seismic Category I Structures

Seismic Category I structures that are supported on soil are as follows:

Structure	Approximate Distance Between Rock and Bottom of Foundation (ft)	Soil Layer at Bottom of Foundation
Diesel Generator Building	50	Upper Till
Offgas Buildings	15	Upper Till
Radwaste Building	6	Fill Concrete

The lower till is a dense granular material with material properties similar to the underlying rock for seismic design, therefore, no amplification of seismic input will occur due to the thickness of lower till between the foundations and rock. See <Section 2.5.4> for a discussion of the properties of soil as affecting seismic analysis. Design of foundations is discussed in <Section 3.8.5>.

For foundations extending down only into upper till, the upper till material was removed down to the lower till and backfill was placed.

The backfill consisted of compacted granular material described as Class A fill in <Section 2.5.4>. The material and dynamic properties of Class A fill are given in <Section 2.5.4>. The amplification of seismic responses due to Class A fill was taken into account by utilizing a two dimensional plane strain finite element model which includes the structure and foundation bearing materials. The structure was modeled using beam elements and the Class A fill was modeled using plane strain elements. The thickness of the plane strain element was taken as the width of the structure. Time history excitation was input at the base of the model. Therefore, any amplification at the foundation level is reflected by the dynamic analysis of the structure.

A linear finite element method is used to evaluate the soil structure interaction for the diesel generator and offgas buildings sitting on Class A fill. For this analysis, the upper and lower soil shear moduli and strain dependent soil properties were used in two separate analyses to cover any significant variation of soil properties.

In the finite element analysis, the deconvolution of the SSE value from the grade to the base rock was not applied. The SSE value was conservatively applied at the base rock and amplified through the soil. Furthermore, the energy absorbing boundary was not applied to the model. Due to the above two conservative steps used, the analytical results are expected to be higher than actual responses.

3.7.2   SEISMIC SYSTEM ANALYSIS

3.7.2.1   Seismic Analysis Methods

3.7.2.1.1   Balance of Plant

Seismic Category I structures, except for the cooling water tunnels, were analyzed by using the simulated time history as described in <Section 3.7.1.2> as input.

In the analysis, system stiffness matrix, frequencies, structural responses, and floor responses were obtained by use of the computer program DYNAL (Reference 2). A description of DYNAL is given in <Section 3.8.1.4.3> and <Section 3.8.4.4.1>. The Householder-Ortega-Wielandt method described in DYNAL was used in the modal analysis to obtain the frequencies and mode shapes. Structural responses were obtained by superposition of modal responses of all significant modes as described in <Section 3.7.3.5>. The significant modes are as shown in <Table 3.7-4>. Since higher modes have smaller participation factors and smaller spectral values, the inclusion of additional modes do not result in more than 10% increase in response. Time histories of structural responses, at mass points of interest, were used to generate the floor response spectra. A response spectra scaling method was used in lieu of the square-root-of-the-sum-of-the-squares (SRSS) method to account for coupling between horizontal and vertical responses. This method was shown to provide conservative results when compared to the SRSS spectra and the design spectra. Analysis and design of structures for dynamic loads including stress criteria are discussed in <Section 3.8>. The concrete structural elements were assumed to have linear elastic properties.

The responses due to two horizontal and one vertical input were combined by taking the square root of the sum of the squares, i.e.,:

$$\text{Total Response} = \sqrt{R_x^2 + R_y^2 + R_z^2}$$

Where:

R_x, R_y, and R_z = Responses due to the x, y and z inputs, respectively.

Examples of lumped mass models are shown in <Figure 3.7-11> and <Figure 3.7-14>. The mass points of a building are chosen at the points of physical mass concentration (e.g., heavy floors, and include the masses of floors, equipment and walls as required).

For details of the analysis of the tunnels for the cooling water system, see <Section 3.8.4.4>.

Floor response spectra are used in the seismic specification and issued to vendors of purchased safety class equipment and components. The vendors are responsible for equipment qualification based on the issued floor response spectrum. See <Section 3.7.3> for more information.

A summary listing of various methods for seismic analysis of Seismic Category I structures is given in <Table 3.7-2>.

<Figure 3.7-11>, <Figure 3.7-12>, <Figure 3.7-13>, and <Figure 3.7-14> show the mathematical models used in the seismic analysis of the reactor building, auxiliary building, intermediate/fuel handling buildings, and control complex, respectively.

3.7.2.1.2   NSSS

Analysis of Seismic Category I NSSS systems and components is accomplished using the response spectrum or time-history approach. Either approach utilizes the natural period, mode shapes and appropriate damping factors of the particular system. Certain pieces of equipment having very high natural frequencies were analyzed statically. In some cases, dynamic testing of equipment were used for seismic qualification.

The time history analyses involve the solution of the equations of the dynamic equilibrium <Section 3.7.2.1.2.1> by means of the methods discussed in <Section 3.7.2.1.2.2>. In this case, the duration of motion is of sufficient length to ensure that the maximum values of response have been obtained.

A response spectrum analysis involves the solution of the equations of motion <Section 3.7.2.1.2.1> by the method discussed in <Section 3.7.2.1.2.3>.

Analytical results for some selected large Seismic Category I equipment are given in <Table 3.7-3>.

3.7.2.1.2.1   The Equations of Dynamic Equilibrium

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as:

$$[M] \ \{\ddot{u}(t)\} + [C] \ \{\dot{u}(t)\} + [K] \ \{u(t)\} = \{P(t)\}$$

Where:

$u(t)$ =   Time dependent displacement of nonsupport points relative to the supports.

$\dot{u}(t)$ =   Time dependent velocity of nonsupport points relative to the supports.

$\ddot{u}(t)$ =   Time dependent acceleration of nonsupport points relative to the supports.

$[M]$ =   Diagonal matrix of lumped masses.

$[C]$ =   Damping matrix.

$[K]$ =   Stiffness matrix.

$P(t)$ =   Time dependent inertial forces acting at nonsupport points.

3.7.2.1.2.2   Solution of the Equations of Motion by Mode Superposition

The first technique used for the solution of the equations of motion is the method of mode superposition.

The set of homogenous equations represented by the undamped free vibration of the system is:

$$[M] \; \{\ddot{u}(t)\} + [K] \{u(t)\} = \{O\} \qquad (3.7\text{-}1)$$

Since the free oscillations are assumed to be harmonic, the displacements can be written as:

$$\{u(t)\} \qquad = \qquad \{\phi\} \; e^{i\omega t} \qquad (3.7\text{-}2)$$

Where:

$\{u(t)\}$ =   Column matrix of the amplitude of displacements.

ω =   Circular frequency of oscillation.

t =   Time.

$\{\phi\}$ =   Column matrix of maximum nodal displacements.

Substituting Equation 3.7-2 and its derivatives in Equation 3.7-1, noting that $e^{i\omega t}$ is not necessarily zero for all values of ωt, yields:

$$\left[-\omega^2[M] + [K]\right] \{\phi\} = \{O\} \qquad (3.7\text{-}3)$$

Equation 3.7-3 is the classical algebraic eigenvalue problem wherein the eigenvalues are related to the frequencies of vibrations ω_i and the eigenvectors are the mode shapes, $\{\phi\}_i$.

3.7.2.1.2.3   Analysis by Response Spectrum Method

As an alternative to the step by step mode superposition method
described in <Section 3.7.2.1.2.2>, the response spectrum method may be
used. The response spectrum method is based on the fact that the modal
responses can be expressed as a set of integral equations, rather than a
set of differential equations. The advantage of this form of solution
is that for a given ground motion the only variables under the integral
are the damping factor and the frequency. Thus, for a specified damping
factor it is possible to construct a curve which gives a maximum value
of the integral as a function of frequency. This curve is called a
response spectrum for the particular input motion and the specified
damping factor. The integral has units of velocity. Consequently, the
maximum of the integral is called the spectral velocity.

Using the calculated natural frequencies of vibration of the system, the
maximum values of the modal responses are determined directly from the
appropriate response spectrum. The modal maxima are then combined as
discussed in <Section 3.7.3.7>.

3.7.2.1.2.4   Support Displacement in Multi-Supported Structures

The preceding sections have discussed analysis procedures for forces and
displacements induced by time dependent support accelerations. In a
multi-supported structure there are, in addition, time dependent support
displacements which produce additional displacements at non-support
points and pseudo-static forces at both support and non-support points.

The governing equation of motion of a structural system which is supported at more than one point and has different excitations applied at each may be expressed in the following concise matrix form:

$$\begin{bmatrix} M_a & O \\ O & M_s \end{bmatrix} \begin{bmatrix} \ddot{U}_a \\ \overline{U}_s \end{bmatrix} + \begin{bmatrix} C_{aa} & C_{aa} \\ C_{as} & C_{ss} \end{bmatrix} \begin{bmatrix} U_a \\ \overline{U}_s \end{bmatrix}$$

$$+ \begin{bmatrix} K_{aa} & K_{aa} \\ K_{as} & K_{ss} \end{bmatrix} \begin{bmatrix} U_a \\ \overline{U}_s \end{bmatrix} = \begin{bmatrix} F_a \\ F_s \end{bmatrix} \qquad (3.7\text{-}4)$$

Where:

U_a = Displacement of active (unsupported) degrees of freedom.

$\overline{U}_s$ = Specified displacements of support points.

M_a, M_s = Lumped diagonal mass matrices associated with the active degrees of freedom and the support points.

C_{aa} = Damping matrix expressing the forces developed in the active degrees of freedom.

K_{aa} = Elastic stiffness matrix expressing the forces developed in the active degrees of freedom.

C_{ss}, K_{ss} = Support forces due to the unit velocities and displacement of the supports.

C_{as}, K_{as} = Damping and stiffness matrices denoting the coupling forces developed in the active degrees of freedom by the motion of the supports (and vice versa).

F_a =   Prescribed external time-dependent forces applied on the active degrees of freedom.

F_s =   Reaction forces at the system support points.

The procedure utilized to construct the damping matrix is discussed in <Section 3.7.2.15>. The mass matrix and elastic stiffness matrix are formulated by using standard procedures.

Equation 3.7-4 can be separated into two sets of equations. The first set of equations can be written as:

$$[M_s] \{\ddot{\overline{U}}_s\} + [C_{ss}] \{\dot{\overline{U}}_s\} + [K_{ss}] \{\overline{U}_s\} + [C_{as}] \{\dot{U}_a\} + [K_{as}] \{U_a\} = \{F_s\}$$

(3.7-5a)

and the second set as:

$$[M_a] \{\ddot{U}_a\} + [C_{aa}] \{\dot{U}_a\} + [K_{aa}] \{\overline{U}_a\} + [C_{as}] \{\dot{\overline{U}}_s\} + [K_{as}] \{\overline{U}_s\} = \{F_a\}$$

(3.7-5b)

The timewise solution of Equation 3.7-5a is obtained by using the standard normal mode solution technique. After obtaining the displacement response of the active degrees of freedom, U_a, Equation 3.7-5b then can be used to solve the support point reaction forces, F_s. Modal damping is used to determine the solutions of the uncoupled form of Equation 3.7-5a. The procedure is identical to that discussed in <Section 3.7.2.15>.

When the equipment is supported at two or more locations, an upper bound envelope of all the individual response spectra is used to calculate maximum inertial responses of multiple supported items. Alternately, the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically

to all the floors, provided there is no significant shift in frequencies of the spectra peaks.

In lieu of the response spectrum approach, time histories of support motions may be used as excitation to the subsystems. Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system (i.e., RPV and intervals) would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiple support systems.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the secondary stresses due to support displacements. Further details are given in <Section 3.7.2.1.2.5>.

3.7.2.1.2.5   Differential Seismic Movement of Interconnected
                Components

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitation is discussed in the paragraphs that follow.

The relative displacements between the supporting points induce additional stresses in the equipment supported at these points. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. These displacements are used to calculate stresses.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the modal displacement is used. Therefore, the mathematical model of the

equipment is subjected to a maximum displacement at its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes (modes contributing the most to the total displacement response of the supporting point) of the structure.

The meaning of "significant" modes is further explained as follows: All modes within a frequency range of interest are included in the dynamic analysis. Generally, the number of modes considered for the analysis of any given system is dependent on the system characteristics and the amplitude/frequency content of the input forcing functions. The criterion is to choose the number of modes to cover the peak responses of the applicable loads to totally represent the actual system responses at the peak response frequency ranges. In accordance with SRP 3.7-2, Section II.1.a(5), the participation of all significant modes is assured when the inclusion of additional modes does not result in more than a 10 percent increase in responses.

The total stresses due to the relative anchor/support displacements are obtained by combining the modal results using the SRSS method. Since the maximum displacement for different modes do not occur at the same time, the SRSS method is a realistic and practical method.

When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses for piping and primary stresses for supports.

3.7.2.1.2.6   Seismic Qualification by Testing

For certain Seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results will reflect the following:

a.   Performance data of equipment which, under the specified conditions, has been subjected to dynamic loads equal to or greater

than those to be experienced under the specified seismic conditions.

b.   Test data from previously tested comparable equipment which, under similar conditions, has been subjected to dynamic loads equal to or greater than those specified.

c.   Actual testing of equipment in accordance with one of the methods described in <Section 3.9> and <Section 3.10>.

Alternate test procedures that satisfy the requirements of these criteria will be allowed, subject to review by the responsible engineer.

3.7.2.2   Natural Frequencies and Response Loads

The first few significant natural frequencies for representative Seismic Category I structures are presented in <Table 3.7-4>. The response spectra at various critical locations within the Category I structures were generated to define the seismic environment at the equipment location for the subsystem analysis.

Response loads (displacements, accelerations and mass) which were determined by seismic analysis are shown in <Table 3.7-5> for the reactor building, auxiliary building, control complex, and intermediate/fuel handling buildings for the OBE. Mode shapes are presented in <Table 3.7-6>, <Table 3.7-7>, <Table 3.7-8>, and <Table 3.7-9>.

3.7.2.3   Procedures Used for Modeling

Two basic techniques were used to model the Seismic Category I structures. In the first technique, the dynamic system was represented by a system of lumped masses located at the elevations of mass concentration, such as floor slabs. For structures such as the

containment shield building having continuous mass distribution, a
sufficient number of mass points were chosen so that the vibration mode
of interest was adequately defined. Soil is represented by springs.
This modeling procedure was used for all the Seismic Category I
structures with the exception of the diesel generator and offgas
buildings.

In the second technique applied to the diesel generator and offgas
buildings, the soil was represented as a two dimensional finite element
model using triangular elements. The degree of refinement in the
mathematical model was based upon the complexity of the actual
structures and the information required from the analysis.

<Figure 3.7-11>, <Figure 3.7-12>, <Figure 3.7-13>, and <Figure 3.7-14>
show the mathematical models used in the seismic analysis of the reactor
building, auxiliary building, intermediate/fuel handling buildings, and
control complex, respectively.

When subsystems are uncoupled from the system analysis, the energy
feedback is ignored. Thus, more conservative results are generated for
both subsystem and system responses (Reference 13). In this plant
design, only the reactor vessel is coupled with the structure in the
dynamic analysis. All other equipment is uncoupled from the structures.

The cooling and emergency service water tunnels were analyzed for
seismic loading by the finite element approach. Analysis of the tunnels
is discussed in <Section 3.8.4>.

3.7.2.4   Soil-Structure Interaction

In the soil-structure interaction analysis for buildings founded on
shale or rock-like material, lumped mass models are used in the dynamic
analysis. The techniques and the soil springs used follow (Reference 1)
and (Reference 3). These models include the dynamic compliances of the

shale as represented by the soil springs. Since these buildings are founded on competent rock, any embedment has only negligible effects on the dynamic response and is not considered in the lumped mass model. Buoyancy has negligible effect on the dynamic analysis of a stable structure and thus, can be uncoupled and considered only in the static analysis.

Caissons are used under a portion of the intermediate building in order to found the entire structure on shale. The caissons are not relied upon to resist lateral loads on the building. Caissons are included in the lumped mass model for seismic analysis; however, their stiffness in the horizontal direction is less than 2 percent of the stiffness supplied by the soil springs for the portion of the structure founded directly on the shale. For a discussion of the caissons, see <Section 3.8.5.1>. For a discussion of the caissons supporting the FHB rail bay exterior mat extension, refer to <Section 3.8.4.1.3.1>.

The soil structure interaction effects for the cooling water tunnels were analyzed by the finite element method with an energy absorbing boundary to represent the semi-infinite half space. The finite element method was used for other safety class structures, as discussed in <Section 3.7.1.4>. The nonlinear stress-strain and strain-damping characteristics of the rock were considered insofar as they are implicit in the use of a damping factor of 5 percent of critical damping. Linear elastic behavior of the rock is considered to be a reasonable approximation for calculation purposes because of the low in situ stress levels and the low magnitude of stress changes induced in the rock by the presence of the tunnel.

The structure-structure interaction is negligible since each structure is separated from the adjacent structures by a seismic separation ("rattle space") and the buildings are founded on competent shale.

3.7.2.5   Development of Floor Response Spectra

Floor response spectra have been developed using a time-history analysis
technique as described in <Section 3.7.2.1>. The floor response spectra
are used to predict the seismic environment for equipment and
substructures located at various elevations within each Seismic
Category I structure.

Using the absolute acceleration time-history response of a particular
mass point, a spectrum response curve was developed and enveloped into a
design acceleration spectrum to be utilized for the seismic analysis of
equipment located at the mass point. Horizontal and vertical response
spectra were computed using various damping values for both SSE and OBE.
The seismic response to the time-history is obtained through the use of
the DYNAL computer program. The methods by which the effects of
parameter variations are accounted for in the floor response spectra as
discussed in <Section 3.7.2.9>.

3.7.2.6   Three Components of Earthquake Motions

The effects of both horizontal and vertical components of seismic motion
are considered for all Seismic Category I structures. In order to
adequately account for the responses of systems subjected to
multi-directional excitation, the manner in which their additive effects
are combined has been considered. Therefore, the combined response is
determined statistically by the method of SRSS which accounts for the
randomness of the ground motion. The SRSS method, considering three
components of ground motion, was used for the final seismic stress
calculation for structural steel and reinforced concrete structural
design.

3.7.2.7   Combination of Modal Responses

In using the response spectrum analysis to predict seismic design loads,
the modal responses were combined in accordance with <Regulatory
Guide 1.92>.

Modal responses for NSSS components are combined as described in
<Section 3.7.3.7.1>.

3.7.2.8   Interaction of Non-Category I Structures with Seismic
             Category I Structures

The interaction of non-Category I structures with Seismic Category I
structures is dependent on the structural relative locations.

Non-Category I structures which are adjacent to Seismic Category I
structures are designed to preclude any affect on these adjacent
structures. A three-inch minimum gap separates the non-Category I
structures from the Seismic Category I structures and precludes
interaction during a seismic event. In addition, the non-Category I
structures are designed to prevent collapse against Seismic Category I
structures. The non-Category I structures are designed for the loads
calculated for Zone 3 of the Uniform Building Code (1976 Edition). As
stated in Figure 1 of this Code, Zone 3 corresponds to a Modified
Mercalli Scale of VIII or higher and exceeds the UBC recommended Zone 1
design accelerations for the site location. It is also higher than the
modified Mercalli scale of VII to VIII which forms the basis for
selection of the 0.15g ground acceleration used in the SSE ground
response spectra see <Section 2.5.2>. The load combinations, design
codes and acceptance criteria for these non-Category I structures under
the Uniform Building Code seismic loads are identical to those used for
safety class structures under SSE loads, all as described in
<Section 3.8.4>.

As shown in <Table 3.7-5>, the largest Category I structure OBE displacement is 0.42 inch. The non-Category I structure can be assumed to have a ductility ratio of 3, and the yield displacement of 0.84 inch (conservatively assuming 2 times maximum OBE displacement for all buildings) for the SSE. Consequently, the largest predicted displacement is about 2.52 in. The root sum square value of 0.82 and 2.52 is 2.65, which is less than the 3 in. seismic gap used in design.

3.7.2.9   Effects of Parameter Variations on Floor Response Spectra

The floor response spectra were generated for the two horizontal and the vertical inputs for SSE and OBE using the most probable parameters. The peaks of the floor response spectra were broadened 15 percent on each side to allow for variations in structural and soil properties.

3.7.2.10   Use of Constant Vertical Load Factor

Dynamic analysis with vertical input was used instead of constant vertical load factors.

3.7.2.11   Methods Used to Account for Torsional Effects

A structure with eccentricity between the center of mass and the center of rigidity greater than 5 percent of the dimension of the structure normal to the input direction, is considered to have pronounced torsional modes. For a structure with pronounced torsional modes, where the horizontal responses are significantly coupled, a three dimensional model is used in the analysis to calculate torsional responses. In the model, walls are simulated as single members and floors are treated as rigid diaphragms. Mass centers and centers of rigidity are calculated, and considered, in the geometry of the model. The acceleration time history is input at the support of the model and DYNAL is used to calculate the actual torsional effects. Responses due to horizontal

excitations and vertical excitation are calculated separately but the effects are additive in determining forces throughout the structure.

For structures having less than 5 percent eccentricity, torsional effects were not considered in the dynamic analysis; however, loads calculated by the dynamic analysis are applied statically at a 5 percent eccentricity.

The RPV is an axisymmetric model with no built-in eccentricity. Hence, the torsional effects on the RPV are only those associated with the reactor building model.

3.7.2.12   Comparison of Responses

3.7.2.12.1   NSSS

Either the time history method or the response spectra approach was used for the seismic analysis of NSSS components. Generally, the responses computed by both methods are comparable in magnitude with the loads being determined by the response spectrum being somewhat more conservative. As both approaches are acceptable, additional comparison of results was deemed unnecessary.

3.7.2.12.2   Balance of Plant

Response spectrum analysis was not performed for structures. The time history method is believed to generate more conservative results because of the enveloping process used in generating the artificial time history.

3.7.2.13   Methods for Seismic Analysis of Dams

There is no dam for this plant.

3.7.2.14   Determination of Seismic Category I Structure Overturning
             Moments

Overturning moments are considered for each safety class structure in
two directions corresponding to the orientation assumed for the
horizontal components of the seismic event. The horizontal seismic
components are considered separately in these calculations. The maximum
vertical earthquake component is considered by reducing the dead weight
of the structure in counteracting the maximum overturning moment due to
one horizontal input. Soil reaction is calculated by adding or
subtracting (whichever controls) the vertical earthquake component to
the dead weight and other loads on the structure. Additional
information on analysis for stability is given in <Section 3.8.5.4>.

3.7.2.15   Analysis Procedure for Damping

3.7.2.15.1   NSSS

In a linear dynamic analysis the procedure used to properly account for
damping in different elements of a coupled system model is as follows:

a.   The structural damping of the various structural elements of the
     model are first specified. Each value is referred to as the
     damping ratio $\left(\beta_j\right)$ 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 $\left(\phi\right)$ normalized such that:

$$\phi_i^T \ K\phi_l = \ \omega_i^2$$

Where:

    K =   Stiffness matrix.

    ω_i =   Circular natural frequency of mode i.

    ϕ_i^T =   Transpose ϕ_i (which is column vector of d

                         corresponding to the mode shape of mode i.

                         Matrix ϕ contains all translational and rotational

                         coordinates.

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 $\left(\beta_i\right)$ for the exponent mode:

$$\beta_i = \sum_{j=1}^{n} \left(\frac{\phi_i^T \, \beta_j \quad K_j \quad \phi_i}{\omega_i^2} \right)$$

Where:

    n =   Total number of structural elements.

    ϕ_i =   Mode shape for mode i (ϕ_i^T as transpose).

    β_j =   Percent damping associated with element j.

    K_j =   Stiffness contribution of element j.

    ω_i =   Circular natural frequency of mode i.

3.7.2.15.2   Balance of Plant

Damping factors are as shown in <Table 3.7-1> and <Table 3.7-10>.

The soil springs and viscous damping values were calculated by the
formula of Whitman and Richart (Reference 3). The soil hysteretic
damping was determined from tests as reported in <Section 3.7.1.3>. In
the case of different damping values involved in one mathematical model,
the method of weighted average damping value (Reference 1) was used.
The viscous soil damping value is as shown in <Table 3.7-1>. The
theoretical weighting function is:

$$\beta_i = \frac{\sum_j \left(\beta_j \; \frac{\Omega}{\omega_j} \; + \; D_j \right) \; E_{ij}}{\sum_j E_{ij}}$$

Where $E_{ij}, \beta_j, D_j, \Omega_i,$ and ω_j are respectively the energy, viscous damping,
hysteretic damping, modal frequency, and frequency associated with the
dashpot.

However, as a conservative approach, the weighting function used in the
finite element method is:

$$\beta_i = \frac{\sum_j \left(D_j \; \cdot \; E_{ij} \right)}{\sum_j E_{ij}}$$

In analysis using a soil spring model, the viscous damping cannot be
ignored because the model cannot take radiational damping into account.
As a conservative approach, the frequency dependent factor for the
viscous damping was not used. Hence, the weighting function used for
the soil spring method is:

$$\beta_i = \frac{\sum_j \left(\beta_j \; \text{or} \; D_j \right) \; E_{ij}}{\sum_j E_{ij}}$$

3.7.3   SEISMIC SUBSYSTEM ANALYSIS

3.7.3.1   Seismic Analysis Methods

3.7.3.1.1   Balance of Plant Scope

Seismic analyses were performed for those subsystems that could be
modeled to correctly predict the seismic response.

Individual components were modeled as multi-degree-of-freedom, lumped
mass systems with mass free interconnections and sufficient mass points
to ensure adequate representation. The resulting systems were analyzed
using the response spectrum modal analysis technique. An alternative
time history method may also be applied. The time history method, when
used, conservatively, simulates the response spectrum envelope of
interest. A stress analysis was then performed using the inertia
forces, or equivalent static loads, obtained from the dynamic analysis.
Moments, shears, accelerations, deflections, and stresses are calculated
on a mode by mode basis. The total seismic response is obtained by
combining each modal response using the square root of the sum of the
squares method. The absolute sum of the responses is considered for
closely spaced modes, as set forth in <Section 3.7.3.7>. In cases for
which some dynamic degrees of freedom do not contribute to the total
response, kinematic condensation was employed in the analysis.

3.7.3.1.2   Components and Equipment Provided by the NSSS Vendor

Seismic analysis methods for subsystems within General Electric scope of
responsibility are given in <Section 3.7.2.1.2>.

3.7.3.2   Determination of Number of Earthquake Cycles

Other than as discussed in <Section 3.7.1> for high frequency ground
motions, earthquake cycles are defined as follows.

3.7.3.2.1   Balance of Plant

In accordance with Section III of the ASME Code (Reference 4), the
effects of cyclic loadings are considered for Class 1 and Class MC
components. During the plant life one safe shutdown earthquake (SSE)
and five operating basis earthquakes (OBE) are considered with
10 maximum stress cycles per earthquake.

3.7.3.2.2   Components and Equipment Provided by the NSSS Vendor

3.7.3.2.2.1   NSSS Piping

Fifty peak OBE Cycles are postulated for fatigue evaluation.

3.7.3.2.2.2   Other NSSS Equipment and Components

To evaluate the number of cycles which exist within a given earthquake,
a typical boiling water reactor building-reactor dynamic model was
excited by three different recorded time histories: May 18, 1940,
El Centro NS component, 29.4 sec; 1952, Taft N 69° W component, 30 sec;
and March 1957, Golden Gate S 80° E component, 13.2 sec. The modal
response was truncated such that the response of three different
frequency bandwidths could be studied: 0-10, 10-20 and 20-50 Hertz.
This was done to give a good approximation to the cyclic behavior
expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the
results from several different points of the dynamic model, the cyclic
behavior given in <Table 3.7-11> was formed.

Independent of earthquake or component frequency, 99.5 percent of the
stress reversals occur below 75 percent of the maximum stress level, and
95 percent of the reversals lie below 50 percent of the maximum stress
level.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

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 3.7-11> according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, 1/2 percent (0.005) of these cycles are conservatively assumed to be at the peak load, 4.5 percent (0.045) at or above three-quarter peak. The remainder of the cycles will have negligible contribution to fatigue usage.

The safe shutdown earthquake has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40-year life of a plant. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and therefore not required by ASME Section III.

The OBE is an upset condition and therefore, must be included in fatigue evaluations according to ASME Section III. Investigation of seismic histories for many plants shows that during a 40-year life it is probable that five earthquakes with intensities one-tenth of the SSE intensity and one earthquake approximately 20 percent 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.

3.7.3.3   Procedure Used for Modeling

3.7.3.3.1   Balance of Plant

Equipment within the balance of plant scope is modeled as a series of
discrete mass points, connected by mass free members, having sufficient
mass points to ensure adequate representation of dynamic behavior.
Detailed modeling of piping systems is described in <Section 3.7.3.8>.

3.7.3.3.2   Equipment and Components Provided by the NSSS Vendor

3.7.3.3.2.1   Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of three
dimensional straight or curved pipe elements. The mass of each pipe
element is lumped at the nodes connected by weightless elastic members,
representing the physical properties of each segment. The pipe lengths
between mass points is no greater than the length which would have a
natural frequency of 33 Hertz when calculated as a simply supported
beam. In addition, mass points are located at all points on the piping
system where concentrated weight such as valves, motors, etc., are
located and also at points of significant change in the geometry of the
system. All concentrated weights on the piping system such as main
valves, relief valves, pumps, and motors are modeled as lumped masses.
The torsional effects of the valve operators and other equipment with
offset center of gravity with respect to center line of the pipe is
included in the analytical model. If the torsional effect is expected
to cause pipe stresses less than 500 psi, this effect may be neglected.

The criteria employed for decoupling the main steam and recirculation piping systems for establishing the analytical models to perform seismic analysis is given below:

a. The small branch lines (6-inch diameter and less) decoupled from the main steam and recirculation piping systems are analyzed separately, because the dynamic interaction is insignificant due to the disparity in the moment of inertia of the two lines.

b. The stiffness of all the anchors and their supporting steel is large enough to effectively decouple the piping on either side of the anchor for analytic and code jurisdictional boundary purposes. The RPV is very stiff compared to the piping system, thus during normal operating conditions the RPV is also assumed to act as an anchor. Penetration assemblies (head fittings) are also very stiff compared to the piping system and are assumed to act as an anchor. The stiffness matrix at the attachment location of the process pipe (i.e., main steam, RCIC, RHR supply, or RHR return) head fitting is sufficiently high to decouple the penetration assembly from the process pipe. General Electric analysis indicates that a satisfactory minimum stiffness for this attachment point is equal 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.

3.7.3.3.2.2   Modeling of Equipment

For dynamic analysis, Seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

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 as

significant if the corresponding natural frequencies are less than
33 Hertz and the stress calculated from these modes are greater
than 10 percent of the total stresses obtained from lower modes.
The hydrodynamic effect is addressed in <Section 3.9>.

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, the impeller in the analysis of pump shaft, etc.

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.

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.

3.7.3.4   Basis for Selection of Frequencies

3.7.3.4.1   Balance of Plant

The selection of a "rigid" frequency to preclude resonance is based on
the floor response curves. This "rigid" frequency is the one beyond
which no secondary peak is present and is related to the frequency value
(33 Hertz) at which the ground design spectrum approaches maximum ground
acceleration (other than as discussed in <Section 3.7.1> for high
frequency ground motions) and beyond which there is no significant
structural mode. Hence, the "rigid" frequency for equipment setting
inside a building can be specified as 33 Hertz.

3.7.3.4.2   Equipment and Components Provided by the NSSS Vendor

All frequencies in the range of 0.25 to 33 Hertz are considered in the
analysis and testing of structures, systems and components. These

frequencies would 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 Hertz, it is
treated as rigid and analyzed accordingly. Frequencies less than
0.25 Hertz are not considered as they represent very flexible structures
which are not encountered in this plant.

The frequency range of 0.25 to 33 Hertz covers the range of the broad
band response spectrum used in the design (other than as discussed in
<Section 3.7.1> for high frequency ground motions).

3.7.3.5   Use of Equivalent Static Load Method of Analysis

The equivalent static loads, for flexible equipment having simple frame
type structural configurations, are taken as 1.5 times the product of
the equipment masses and the peak spectral accelerations of the
applicable floor response spectra. For tray, duct and conduit supports,
a factor of 1.0 is applied to the peak acceleration of the applicable
floor response spectrum based on a conservative analysis of a typical
support.

When static coefficient analyses are performed for equipment with other
more complex configurations, justification based on the equipment
eigenvalue and frequency/amplification content of the applied loads will
be provided for the value of the static coefficient used.

3.7.3.6   Three Components of Earthquake Motion

Responses to the two horizontal and the vertical component seismic
inputs are calculated separately for the entire subsystem. The maximum
value of a particular response due to simultaneous action of three
components of earthquake motion was obtained by taking the square root
of the sum of the squares of corresponding maximum response values to

each of the three components calculated separately. This procedure is in conformance with the guidance of <Regulatory Guide 1.92>.

3.7.3.7   Combination of Modal Responses

3.7.3.7.1   Balance of Plant

Modal responses and spatial components of earthquake in seismic response analysis were combined in accordance with <Regulatory Guide 1.92>.

3.7.3.7.2   NSSS

In a response spectrum modal dynamic analysis, if the modes are not closely spaced (i.e., if the frequencies differ from each other by more than 10 percent of the lower frequency), the modal responses are combined by the square-root-of-the-sum-of-the-squares (SRSS) method as described in <Section 3.7.3.7.2.1>. If some or all of the modes are closely spaced, a double sum method, as described in <Section 3.7.3.7.2.2>, is used to evaluate the combined response. In a time-history method of dynamic analysis, the vector sum at every step is used to calculate the combined response. The use of the time-history analysis method precludes the need to consider closely spaced modes. The procedure is in accordance with <Regulatory Guide 1.92>.

3.7.3.7.2.1   Square Root of the Sum of the Squares Method

Mathematically, this SRSS method is expressed as:

$$R = \left(\sum_{i=1}^{n} \left(R_i \right)^2 \right)^{1/2}$$

where:

> R = Combined Response,

> R_i = Response in the ith mode, and

> n = Number of modes considered in the analysis

3.7.3.7.2.2   Double Sum Method

This method is defined mathematically as:

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k\, R_s \right| \left| \varepsilon_{ks} \right| \right]^{1/2}$$

where:

> R = Representative maximum value of a response of a given element to a given component of excitation.

> R_k = Peak value of the response of the element due to the kth mode.

> N = Number of significant modes considered in the modal response combination.

> R_s = Peak value of the response of the element attribute to s^{th} mode.

Also,

$$\varepsilon_{ks} = \left[1 + \left(\frac{\omega_k^1 - \delta_s^1}{\beta_k^1\, \omega_k + \beta_s^1\, \omega_s} \right)^2 \right]^{-1}$$

in which,

$$\omega_k^1 = \omega_k \left(1 - \beta_k^2 \right)^{1/2} \; ; \; \beta_k^1 = \beta_k + \frac{2}{t_d \, \omega_k}$$

where:

ω_k = Modal frequency in the kth mode.

β_k = Damping ratio in the kth mode.

t_d = Duration of earthquake.

3.7.3.8 <u>Analytical Procedures for Piping</u>

3.7.3.8.1 Balance of Plant Scope

a. Analysis of Major Piping

For all Class 1 piping, all Class 2 and 3 piping above 250°F and almost all Class 2 and 3 piping above 10 inches, the piping system geometry, cross sectional dimension and physical properties of each pipe segment and the restraint conditions are supplied as inputs to computer programs. The mass of each piping segment is lumped at the element nodes by the computer. Additional concentrated masses are specified separately for valves, actuators and other concentrated weights at the centers of gravity for the individual assembly or subassemblies to represent both bending and torsional effects of the assembly. If any valves and valve operator assemblies have natural frequencies less than 33 Hertz their flexibility will be considered in the piping analysis.

Restraint conditions of supports are specified in three translational and three rotational directions in the model, in either global or local coordinates for each support point. The

restraints may be free, rigid or elastic with a specified spring constant for each translational or rotational direction. When coupling effects between any two joint degrees of freedom are significant, a 6 by 6 stiffness matrix is used to describe an elastic foundation. Moment release at nodal points is used for pin connections or flexible joints whenever applicable.

A discrete system of equations based upon the input data is then developed. The resulting homogeneous equations are solved as an eigenvalue problem. The floor response spectrum method is used in calculating the responses of each mode including nodal displacements, end forces and moments, and support loads. These modal responses are combined by the SRSS method for all modes within the cut-off frequency. In addition, the effects of the high frequency residual modes not included are added to the SRSS response within the cut-off frequency as one term, using the acceleration at the highest frequency from the SRSS response under 33 Hertz to obtain the total response unless the effects of the residual modes have been shown to be insignificant. The definition and grouping method of combining close modes, described in <Section 3.7.3.7.1>, is applied in nodal deflection, element end forces and moments, acceleration and support loads.

The responses from the two horizontal and the vertical components of an earthquake are calculated separately as described above. These responses are then combined using the SRSS method. The resultant end moments are used in the applicable ASME Code, Section III, equations for stress evaluation.

b.   Simplified Dynamic Analyses

Safety Class 2 and 3 piping, up to 10 inches in diameter and up to 250°F, may be supported and seismically restrained in accordance with Maximum Support Spacing Criteria as described below.

For the purpose of a computer dynamic analysis, the piping is modeled as a simply supported multi-span beam. The model natural frequency and mode shape are determined for each mode being considered. Based on the floor response spectrum curve (which accounts for both seismic and operation or accident induced safety/relief valve structural dynamic responses) for the appropriate building and elevation, the acceleration for a particular natural frequency is determined. Modal accelerations at points along the beam are then given by:

$$A = \sqrt{\sum_{n=1}^{N} \left(A_a\right)_n \left(\phi\right)_n \left(P\right)_n {}^2}$$

Where:

A_a = Acceleration for floor response spectrum curve at support locations of attachment.

ϕ = Mode shape.

P = Participation factor.

n = Mode number.

Multiplication of the acceleration by the mass density of the pipe gives the equivalent static load on the beam, from which bending moments and stresses are calculated. The calculated stress is compared with the allowable stress.

The maximum support spacing is then taken as the span length which produces stresses within the allowable limits. The analysis is performed for each pipe diameter, wall thickness and weight.

At each support point, the pipe is seismically restrained in the horizontal and/or vertical directions, and, to assure that the system is supported in a manner which is similar to the model, horizontal and vertical restraints are provided for each significant concentrated load (valves, inline instruments) and for each change in direction.

Where relative structural movement exists within the same structure or where piping spans between adjacent structures, the effect of differential piping support movements are evaluated. The relative movements of pipe supports are considered separately in each of the spatial directions. The results are then combined by the SRSS method.

3.7.3.8.2   Components and Equipment Provided by the NSSS Vendor

The analytical procedures for piping analysis are described in <Section 3.7.2.1.2>. Methods to include differential pipings support movements at different support points are described in <Section 3.7.2.1.2.5>.

3.7.3.9   Multiply Supported Equipment Components with Distinct Inputs

3.7.3.9.1   Balance of Plant Scope

There is no Balance of Plant Equipment that is supported at different locations (elevations and/or floors).

3.7.3.9.2   Components and Equipment Provided by the NSSS Vendor

For components and equipment provided by the NSSS vendor, methods used to account for multiply supported equipment components with distinct inputs are given in <Section 3.7.2.1.2.4>.

3.7.3.10   Use of Constant Vertical Static Factors

3.7.3.10.1   Balance of Plant Scope

The response spectrum method is used for the vertical seismic subsystem dynamic analysis. However, for the cases where the equipment's lowest frequency in the vertical direction is more than 33 Hertz, the maximum floor acceleration is used for equipment design.

3.7.3.10.2   Components and Equipment Provided by the NSSS Vendor

Constant vertical static factors are not used by the General Electric Company.

3.7.3.11   Torsional Effect of Eccentric Masses

3.7.3.11.1   Balance of Plant

For seismic analysis of piping the torsional effect of valves and other eccentric masses is taken into account by lumping masses at the mass center of the valve or other eccentric mass.

3.7.3.11.2   Components and Equipment Provided by the NSSS Vendor

Torsional effects of eccentric masses are discussed in <Section 3.7.3.3.2>.

3.7.3.12   Buried Seismic Category I Piping Systems and Tunnels

3.7.3.12.1   Safety Class Piping

Seismic analysis of buried safety class piping is performed in three
phases, as follows:

a.   Calculation of maximum soil strain resulting from the propagation
      of seismic waves, considering a net design, wave velocity composite
      which assumes a combination of various wave types (shear wave,
      compression wave and Rayleigh wave).

b.   Determination of the extent to which the pipe deforms elastically
      as a result of soil strain, or as a result of relative ground
      movement at building/soil interfaces, considering the influence of
      friction forces between soil and pipe and treatment of soil as a
      continuous elastic support.

c.   Calculation of the stresses in the pipe which result from such
      elastic deformation and comparison with allowable stresses as
      discussed in <Section 3.9>.

(Reference 5), (Reference 7), (Reference 8), (Reference 9),
(Reference 14), and (Reference 15) were used as the basis for the
analytical determination of seismic stresses in buried safety class
piping. Sources for various parameters used in the analysis are as
follows:

a.   Maximum acceleration for SSE, <Section 2.5.2>.

b.   Fill surface elevations and subsoil conditions, <Section 2.5.4>.

c.   Properties $\left(\gamma_t, \bar{c}, \bar{Q}, \mu, \text{and } G_{max} \right)$ of Zone I and Zone II fill soils,
      <Section 2.5.4> and <Section 2.5.6>.

d. Properties $\left(\gamma_t, \bar{c} \text{ and } \bar{Q}\right)$ of Zone III fill soil are assumed, based
    upon <Section 2.5.4> and U.S. Navy Design Manual, NAVDOCK DM-7,
    "Soil Mechanics, Foundations and Earth Structures," Bureaus of
    Yards and Docks, 1962.

3.7.3.12.2   Safety Class Manholes

Newmark's formula (Reference 10) for underground piping was utilized to
analyze the safety class manholes of the underdrain system. However,
since this formula results in an upper bound solution, the results were,
in some case, considered to be too conservative. In these cases, a
dynamic analysis of the manhole and the surrounding soil was performed
utilizing a plane strain finite element model.

Plane strain finite element analyses were used to analyze two typical
manholes (one shallow gravity system manhole, and one deep pumping
system manhole) to verify the use of Newmark's formula.

The strain dependent shear moduli of the soils are taken into account.
When the differences between input shear moduli and the shear moduli
corresponding to the output strain values are within 10 percent of each
other, the results are considered acceptable.

The strain dependent part of the total damping used was compared to the
damping values corresponding to the output strain values. When the
difference are within 10 percent of each other, the results are
considered acceptable.

3.7.3.13   Interaction of Other Piping with Seismic Category I Piping

3.7.3.13.1   Balance of Plant

Nonsafety piping potentially jeopardizing items important to safety is classified as "nonsafety, Seismic Category I" and is designed and constructed such that an SSE would not cause failure of any item important to safety <Regulatory Guide 1.29>, (paragraph C.2).

A safety, Seismic Category I piping system is analyzed by including the piping extending to at least the first restraint in each of the three mutually orthogonal directions beyond the defined seismic Category I boundaries. Wherever necessary, piping segments and restraints beyond the region described above are included to ensure that both the elastic reaction and the effects of masses of the non-Seismic Category I piping on the Seismic Category I piping are adequately represented <Regulatory Guide 1.29>, (paragraph C.3).

The Quality Control requirements imposed on the nonsafety Seismic Category I pipe and supports are the following <Regulatory Guide 1.29>, (paragraph C.4):

a.   Design drawings clearly identify which nonsafety piping and supports are Seismic Category I.

b.   Design and design control for the nonsafety Seismic Category I pipe and supports are carried out in the same manner as items important to safety. This includes the performance of appropriate design reviews.

c.   Installations are inspected for compliance with design drawings using a checklist established specifically for this category pipe and supports.

3.7.3.13.2   Equipment and Components Provided by the NSSS Vendor

When other piping is attached to Seismic Category I piping, the other piping is analytically simulated in a manner that does not significantly degrade the accuracy of the analysis of the Seismic Category I piping. Furthermore, the other 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 Analysis for Reactor Internals

3.7.3.14.1   Reactor Pressure Vessel and Internals

The seismic loads on the reactor pressure vessel (RPV) and internals are based on a dynamic analysis of the entire RPV building complex with the appropriate forcing function supplied at ground level. For this analysis, the models shown in <Figure 3.7-16> and the mathematical model of the building are 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 and the effects of both bending and shear are included. Mass points are located at all points of critical interest such as anchor, 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. Further, to facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel), are selected at the same elevation. The various lengths of control rod drive housings are grouped into the two representative lengths shown.

These lengths represent the longest and shortest housings that will adequately represent the full range of frequency response of the housings.

The high fundamental natural frequencies of the CRD housings result in very small seismic loads. Furthermore, the small frequency differences between the various housings due to the length differences result in negligible differences in dynamic response. Hence, the modeling of intermediate length members becomes unnecessary. Not included in the mathematical model are light components such as jet pumps, in-core guide tubes and housings, sparger, and their supply headers. This is done to reduce the complexity of the dynamic model. If the seismic responses of these components are needed, they can be determined after the system response has been found.

The presence of fluid and other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which will serve to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in (Reference 11). The seismic model of the RPV and internals has two horizontal coordinates for each mass point considered in the analysis. The remaining translational coordinate (vertical) is excluded because the vertical responses of RPV and internals are well above the significant horizontal responses. Furthermore, all support structures, building and containment walls have a common centerline, and hence, the coupling effects are negligible. A separate vertical analysis is performed. Dynamic loads due to vertical motion are added to or subtracted from the static weight of components, whichever is more conservative. The two rotational coordinates about each node point are excluded because the moment contribution of rotary inertia from surrounding nodes is negligible. Since all deflections are

assumed to be within the elastic range, the rigidity of some components may be accounted for by equivalent linear springs.

The shroud support plate is loaded in its own plane during a seismic event and hence is extremely stiff. Therefore it may be 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 as an equivalent torsional spring.

The maximum shears and moments along the RPV, and within the internals are determined from the seismic analysis performed. These seismic loads are then combined with the operating loads and the calculated stress results are then compared to the limits given in Section III of the ASME Boiler and Pressure Vessel Code to confirm the integrity of the structure, systems, component, and equipment.

3.7.3.14.2   Damping Values

The damping values used in the seismic analysis for the reactor internals are given in <Table 3.7-12>.

3.7.3.15   Analysis Procedure for Damping

Analysis procedures for damping are provided in <Section 3.7.2.15>.

3.7.4   SEISMIC INSTRUMENTATION

3.7.4.1   Comparison with <Regulatory Guide 1.12>

The seismic instrumentation provided is in accordance with <Regulatory Guide 1.12>.

A Free-Field sensor is provided near the Microwave Tower Building. The Free-Field sensor is located >300 feet from the plant structures.

As the containment structures for both Unit 1 and Unit 2 are identical in design and founded on the same structural support material, the response of the two containments to a seismic event should be identical. Therefore, sensors are provided on the Unit 1 side, only.

3.7.4.2   Location and Description of Instrumentation

The seismic instrumentation provided is qualified in accordance with IEEE 344, 1975 (Reference 12).

3.7.4.2.1   Seismic Monitoring System Description

The Seismic Monitoring System (SMS) is comprised of six digital recorders with external, solid-state sensors. All stations are linked to the Seismic Monitor Control Center (SMCC). The SMCC controls the network of recorders and provides assessment of data gathered from the recorders. The SMCC provides local indication and relay output to actuate control room alarm and indication at the seismic monitoring system status panel in the control room.

The accelerometer sensors are each paired with a motion recorder. The accelerometers are triaxial inertial moment devices that are interrogated by the motion recorders. The motion recorders are controlled from the SMCC, but can operate autonomously if network communication is lost.

Normal power is provided from the non-safety related 120 VAC system. Battery backup is provided for the SMCC and at each motion recorder in the event of loss of normal system power.

The accelerometer recorder pair continuously monitors structure accelerators in three axes (Longitudinal, Transverse, and Vertical) and the recorder initiates the recording function when acceleration exceeds the trigger setpoint of 0.005g in any axes.

At the recorder, if the acceleration sensor signal exceeds the trigger threshold, an "event" is created and is saved on local memory, as well as transferred to the SMCC for further processing by a personal computer workstation with data analysis software. The recorder also compares the data to alarm thresholds, and sets software alarm flags if the thresholds are exceeded. All data files and alarm flags are forwarded to the SMCC to enable display in the main cabinet, system alarm indications and relay outputs, and event reports. Additionally, the SMCC enables software-triggering of all linked recorders when any one of the recorders is triggered. Due to the capability of the recorder to store data in a ring buffer, pre-event trigger data is also recorded in the event file.

Alarm setpoints are maintained in the motion recorders and monitored at the SMCC for OBE and SSE. Data analysis software is used to assess field data gathered from the motion recorders.

The SMCC reads the status of the alarm flags from all the recorders and evaluates it with a voting logic. If the voting logic of the SMCC is activated it sets the SMCC status information accordingly and activates an output relay contact for control room annunciation and status information.

In case of OBE or SSE exceedance after a common trigger, an alarm message is written to the Log-file and an alarm relay in the SMCC is activated. An automated analysis of digital earthquake records is provided by the SMCC. Manual data analysis in interactive operation is possible as well.

Data analysis software includes the following functions:

- Display of the time-history for each location, for each axis together with the trigger- and OBE/SSE alarm levels.

- Computation and display of time and amount of the peak value of the resultant acceleration on horizontal plane for each location.

- Computation and display of the response spectra for each location, for each axis with a given damping value and graphical comparison with the user specified reference spectra for OBE and SSE (location specific).

- Computation and display of the time-history of the CAV-value (Cumulative Absolute Velocity), for each axis and graphical comparison with the limit values for OBE and SSE.

3.7.4.2.2   Accelerometer Sensor Locations

Criteria for the selection of types and locations of seismic instrumentation are in accordance with <Regulatory Guide 1.12>. Sensors are located on elevations that have been modeled as mass points in the dynamic analysis of the structures. Sensors are affixed either directly to structure concrete or on rigid mounting plates that are affixed directly to the structure.

If an earthquake occurs, the recorded responses of the previously discussed seismic instrumentation are compared to calculated responses as discussed in <Section 3.7.4.4>.

For instruments using a bracket or plate as a mounting adapter, shown in <Figure 3.7-17>, <Table 3.7-13> presents the calculated and tested (in selected cases) lowest natural frequency (of three dimensions) of each mechanical system (adapter bracket or plate, plus instrument). For instruments not using a bracket or plate as a mounting adapter, the instruments are rigidly bolted to the concrete or steel surface monitored in accordance with the instrument manufacturer's instructions.

Instrument assemblies are specified and designed to be free of spurious resonances within the frequency range of the instrument.

Mountings of seismic instrumentation do not affect the measurements obtained.

The accelerometer sensors are located in the Reactor Building, Intermediate Building and Free Field as follows:

- Reactor Building foundation Elevation 574' (0D51N0232)
- Reactor Building internal structure Elevation 644' (0D51N0231)
- Reactor Building internal structure Elevation 664' (0D51N0230)
- Intermediate Building foundation Elevation 574' (0D51N0233)
- Intermediate Building structure elevation 599' (0D51N0234)
- Free Field at Microwave Tower (0D51N0240)

3.7.4.3   Control Room Operator Notification

Control room seismic monitoring system information available to the operator are as follows:

An annunciator alarm is actuated in the control room when any of the accelerometers has sensed acceleration greater than the trigger threshold or when OBE or CAV limits are exceeded in either of the horizontal directions or in the vertical direction. The annunciator is also activated if there is a diagnostic error at the SMCC.

3.7.4.4   Comparison of Measured and Predicted Responses

3.7.4.4.1   General Requirements

In the event of an earthquake, the control room operator first
determines whether or not the OBE acceleration level has been exceeded.
This is accomplished by inspection of the indications and alarms
described in <Section 3.7.4.3>. If the OBE spectrum and CAV has been
exceeded, the plant is shut down to evaluate the exceedance. However,
if the exceedance is only in the frequency region above 10 Hz, shutdown
may be deferred with NRC concurrence. Evaluation of the exceedance is
accomplished by comparing measured and predicted responses as follows:

a.   The triaxial time-history record of the event is obtained at the
      SMS instrument panel for each of the sensors. This response is
      then compared to the design response spectra for these locations.

For earthquakes where spectrum levels do not exceed the OBE and CAV, the
plant remains in operation.

3.7.4.4.2   The Earthquake of January 31, 1986 (Reference 16),
                 (Reference 17), and (Reference 18)

On January 31, 1986, a Richter Magnitude 5 earthquake occurred at Leroy,
Ohio, approximately 11 miles to the south of Perry Nuclear Power Plant.
At the time of the earthquake, all seismic instruments except D51-R130
(ENGDAHL/PAR-400) were functional. The records indicated that this

earthquake had a short duration, low velocity, small displacement, high frequency, and high acceleration. It lacked damage potential because of high frequency content. The design ground OBE spectrum was exceeded by a spike around the 20 Hz region. However, in the potentially damaging frequency region of 2-10 Hz, the earthquake recorded spectrum was significantly below the design ground OBE spectrum.

Immediately after the earthquake, plant operations personnel were dispatched into the plant to survey for any damage. No earthquake related damage was reported. Subsequently, a team of engineers and technicians were organized to perform a detailed walkdown of all plant areas. Again, no damage related to the earthquake was reported. At the request of the NRC, the plant structures and components required for cold shutdown and possibly sensitive to high frequency input were reanalyzed and compared with pre-established seismic resistant capability. CEI conducted both Perry specific detailed analyses and generic analyses as follows:

a.   Quantification of available seismic margin for safe shutdown active equipment sensitive to high frequency above the January 31, 1986, earthquake.

b.   Generic evaluation of a high-frequency, short-duration earthquake with regard to its energy content, and effects of the January 31, 1986, earthquake on passive equipment and structures.

c.   Effects of a high frequency event similar to the January 31, 1986, earthquake with longer duration and higher amplitude.

Over 170 safety-related components were studied in the seismic margins analysis of equipment originally qualified by both test and analysis. Included were pumps, motors, valves, actuators, switchgear, motor control centers, relays, switches, and instrumentation. Newly generated floor response spectra based on recorded time histories as input were

compared to the original seismic qualification requirements for the equipment. All calculated margins were very conservatively determined to be greater than 1.0.

Energy content of a short duration, high frequency earthquake and the Perry SSE design earthquake were evaluated using ductility ratios. An elasto-plastic single degree of freedom (SDOF) system was analyzed using simplified pulse type acceleration time histories to simulate earthquake induced support motions. The evaluation uses the concept of ductility demand and energy consumed by a SDOF system. The computer program ADINA was utilized to perform the ductility demand and simplified parametric studies incorporating those basic equations were utilized to calculate energy consumption.

For Perry, the study concentrated on passive equipment and structures having a natural frequency close to 20 Hz since the elastic spectra generated from the January 31, 1986, event exceeded the SSE spectra in this region. The January 31, 1986, recorded acceleration time histories were used as base input motions to the SDOF system.

The study concluded that the Perry SSE design earthquake (broad band, long duration) has higher ductility requirements and energy consumption than the January 31, 1986, earthquake (narrow band, short duration). Further, these same conclusions should also be applicable for the more generic case of a high frequency, short duration event as compared to a <Regulatory Guide 1.60> type SSE design earthquake.

For consideration of a more severe event, an equivalent of the recorded high frequency, short duration earthquake was extended to increase the local acceleration peak by 100% and double the overall time duration. The resulting ductility demand of the extended earthquake was increased by less than 10%. For effect on generated elastic response spectra, the maximum amplitude of the January 31, 1986, recorded acceleration time history was locally increased by 100% while the overall duration (S-wave

portion) was tripled. Results indicated that response spectra
amplitudes in the 20 Hz region did not change significantly, however,
spectra did increase nominally in the lower frequency region. These
small increases were well enveloped by <Regulatory Guide 1.60> SSE
design spectra. Thus, in both considerations (ductility demand and
effect on elastic spectra), it was concluded that the PNPP design has
adequate safety margin, including evaluation of an event both larger and
with longer duration than the January 31, 1986, event.

3.7.5   REFERENCES FOR SECTION 3.7

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TABLE 3.7-1

DAMPING FACTORS

Component or Structure	Percent of Critical Damping	
	Operating Basis Earthquake	Safe Shutdown Earthquake
Equipment and large diameter piping systems, pipe diameter greater than 12 in.[2]	2	3
Small diameter piping systems, diameter less than or equal to 12 in.[2]	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7
Underground concrete tunnels[1]	5	5
Soil viscous damping	10	10

NOTES:

[1] Finite element model that includes rock-lining interaction.
[2] Damping values as defined in ASME Code Case N411 may be utilized for both OBE and SSE for piping stress analyses.

TABLE 3.7-2

METHODS USED FOR SEISMIC ANALYSIS OF CATEGORY I STRUCTURES

Category I Structure	Modeling Techniques for Structures and Foundations	Damping	Method of Analysis
Reactor Building	Lumped masses and half space equivalent soil spring constant	Soil-structure interacted composite model damping	Time history analysis
Auxiliary Building	Same as above	Same as above	Same as above
Control Building	Same as above	Same as above	Same as above
Intermediate and Fuel Handling Buildings	Same as above	Same as above	Same as above
Radwaste Building	Same as above	Same as above	Same as above
Diesel Generator Building	Finite element	Soil-structure interacted composite model damping	Time history analysis
Emergency Service Water Pumphouse	Lumped masses and half space equivalent soil spring constant	Same as above	Same as above
Offgas Building	Finite element	Same as above	Same as above

TABLE 3.7-3

COMPARISON OF CALCULATED SEISMIC LOADS TO DESIGN
SEISMIC LOADS OF CATEGORY I EQUIPMENT, SSE CONDITION

	Equipment	Calculated Results		Design Seismic Load
		Natural Frequency	Seismic Loads	
1.	RCIC Pump	>33 Hz	0.35g (horiz.) 0.55g (vert.)	3.0g (horiz.) 2.0g (vert.)
2.	RCIC Turbine	16 Hz (horiz.) 18 Hz (vert.)	2.25g (horiz.) 2.25g (vert.)	2.25g (horiz.) 2.25g (vert.)
3.	SLC Tank[3]	58.8 Hz except for fluid sloshing mode which is 0.58 Hz	0.45g (horiz.) 2.00g (vert.)	1.75g (horiz.) 1.75g (vert.)
4.	Spent Fuel and Defective Fuel Racks[1]	14.5 Hz	4,698 lb/in.^{2}	26,600 lb/in.^{2}
5.	Spent Fuel and Defective Fuel Racks[2]	14.5 Hz	10,476 lb/in.^{2}	26,600 lb/in.^{2}
6.	New Fuel Racks[2]	14.5 Hz	10,476 lb/in.^{2}	26,600 lb/in.^{2}
7.	Refueling Platform	12.15 Hz	31,220 lb/in.^{2} 38,490 lb/in.^{2}	31,680 lb/in.^{2} (OBE) 40,600 lb/in.^{2} (SSE)
8.	Control Room Panels	(Seismic Adequacy Determined by Test)		
9.	Fuel Prep Machine	17.0 Hz	5,860 lb/in.^{2}	29,400 lb/in.^{2} (SSE)
10.	RHR Heat Exchanger[4]	14 Hz (horiz.) 39 Hz (vert.)	1.0g (horiz.) 0.4g (vert.)	1.0g (horiz.) 0.4g (vert.)
11.	Hydraulic Control Unit	(Seismic Adequacy Determined by Test)		

TABLE 3.7-3 (Continued)

NOTES:

[1] Beams and cruciforms.
[2] Beams only.
[3] Seismic loads are less than the design seismic loads when horizontal and vertical loads are combined.
[4] Loads specified correspond to the lowest dominate frequency. Dynamic analysis was performed using a response spectrum.

TABLE 3.7-4

SIGNIFICANT NATURAL FREQUENCIES FOR
CATEGORY I STRUCTURES

Structure	Significant Natural Frequencies (cps)			
Reactor Building	3.87	4.93	5.23	6.39
	6.96	8.33	9.67	10.68
	13.21	14.14	17.62	18.69
	19.28	20.29	21.76	21.76
	25.33	29.22	30.38	
Auxiliary Building	8.90	9.08	17.47	24.17
	25.42	39.24		
Control Building	4.75	4.91	10.84	13.35
	13.60	22.16	23.17	28.92
	31.01	31.82		
Intermediate and Fuel Handling Buildings	3.75	5.22	7.86	8.88
	9.72	11.77	13.17	14.86
	17.17	17.52	22.70	26.08
	26.49	28.41	30.73	

TABLE 3.7-5

OBE RESPONSE LOADS FOR SEISMIC CATEGORY I STRUCTURES

Elevation	Mass (kip-sec^2/ft)	Acceleration (g)			Displacement (in.)		
		NS	EW	Vertical	NS	EW	Vertical
Reactor Building (Shield Building)							
600'-6"	139.3	0.116[1]		0.124	0.040[1]		0.009
622'-6"	142.9	0.143		0.152	0.072		0.013
644'-6"	136.7	0.178		0.178	0.106		0.015
662'-7"	128.5	0.211		0.202	0.136		0.017
688'-6"	121.8	0.278		0.234	0.179		0.020
704'-0"	106.6	0.318		0.250	0.204		0.022
725'-0"	122.6	0.367		0.268	0.237		0.023
746'-0"	174.7	0.411		0.280	0.267		0.024
766'-0"	90.3	0.453		0.283	0.290		0.024
Auxiliary Building							
568'-4"	483.9	0.089	0.090	0.101	0.003	0.003	0.007
599'-0"	465.8	0.155	0.177	0.168	0.015	0.019	0.015
620'-6"	434.8	0.249	0.257	0.206	0.025	0.029	0.018
652'-0"	375.9	0.348	0.338	0.231	0.035	0.039	0.020
Control Building							
574'-10"	674.845	0.86	0.094	0.087	0.004	0.003	0.005
599'-0"	337.888	0.153	0.128	0.118	0.035	0.028	0.018
620'-6"	309.317	0.201	0.163	0.167	0.063	0.049	0.027
638'-6"	205.590	0.243	0.200	0.222	0.089	0.070	0.034
654'-6"	246.894	0.276	0.232	0.249	0.111	0.086	0.038
679'-6"	251.553	0.319	0.256	0.287	0.134	0.108	0.042
707'-2"	283.851	0.373	0.271	0.342	0.154	0.126	0.043

TABLE 3.7-5 (Continued)

Elevation	Mass (kip-sec^2/ft)	Acceleration (g)			Displacement (in.)		
		NS	EW	Vertical	NS	EW	Vertical

Intermediate/Fuel Handling Building

Elevation	Mass	NS	EW	Vertical	NS	EW	Vertical
574'-10"	493.5	0.090	0.086	0.097	0.008	0.007	0.005
599'-0"	377.9	0.141	0.134	0.122	0.032	0.025	0.010
620'-6"	291.9	0.190	0.182	0.175	0.056	0.047	0.014
639'-6"	219.1	0.241	0.211	0.221	0.077	0.090	0.019
654'-6"	258.1	0.281	0.220	0.235	0.095	0.133	0.019
665'-0"	211.2	0.302	0.239	0.250	0.108	0.164	0.026
682'-6"	388.4	0.336	0.280	0.265	0.128	0.217	0.034
707'-6"	261.2	0.398	0.424	0.270	0.153	0.293	0.034
721'-6"	15.2	0.431	0.516	0.270	0.163	0.331	0.034
753'-9"	4.1	0.509	0.728	0.258	0.189	0.417	0.021

NOTE:

[1] North-South and East-West components are identical due to symmetry.

TABLE 3.7-6

MODE SHAPES OF REACTOR BUILDING[1]

	Joint	1st Mode	2nd Mode	3rd Mode	4th Mode	5th Mode	6th Mode	7th Mode
Shield	1	1.000	-0.097	0.082	-0.021	0.0019	0.0005	0.0056
Building	2	0.920	-0.089	0.076	-0.020	0.0017	0.0004	0.0046
	3	0.817	-0.078	0.066	-0.017	0.0014	0.0003	0.0029
	4	0.705	-0.066	0.055	-0.014	0.0010	0.0002	0.0010
	5	0.618	-0.056	0.046	-0.011	0.0007	0.0001	-0.0004
	6	0.469	-0.038	0.031	-0.0063	0.0002	0.0	-0.0025
	7	0.366	-0.027	0.021	-0.0033	-0.0001	-0.0001	-0.0036
	8	0.248	-0.014	0.010	-0.0002	-0.0004	-0.0002	-0.0043
	9	0.139	-0.0023	0.0001	0.0022	-0.0007	-0.0002	-0.0045
Contain-	10	0.218	0.081	-0.120	1.000	0.017	0.002	0.011
ment	11	0.187	0.070	-0.104	0.863	0.014	0.001	0.008
	12	0.165	0.062	-0.091	0.743	0.012	0.001	0.006
	13	0.148	0.055	-0.081	0.649	0.010	0.001	0.005
	14	0.119	0.044	-0.064	0.488	0.008	0.001	0.002
	15	0.099	0.036	-0.052	0.376	0.006	0.0004	0.0003
	16	0.075	0.027	-0.037	0.246	0.003	0.0002	-0.002
	17	0.052	0.018	-0.024	0.128	0.001	-0.00004	-0.003
Drywell	18	0.205	0.202	-0.384	-0.043	0.009	0.001	0.002
	19	0.168	0.164	-0.309	-0.034	0.007	0.0004	0.0005
	20	0.140	0.135	-0.253	-0.027	0.005	0.0003	-0.0006
	21	0.102	0.091	-0.168	-0.016	0.003	0.0001	-0.002
	22	0.066	0.052	-0.093	-0.006	0.001	-0.0001	-0.003
	23	0.044	0.029	-0.048	-0.0005	0.0001	-0.0001	-0.004
Biolog-	24	0.190	0.605	0.579	-0.002	-0.284	-0.103	-0.726
ical	25	0.149	0.451	0.427	0.0002	-0.206	-0.073	-0.515
Shield	26	0.112	0.313	0.289	0.002	-0.135	-0.044	-0.310
Wall								
Pedestal	27	0.073	0.167	0.144	0.003	-0.061	-0.015	-0.110
	28	0.053	0.093	0.072	0.004	-0.033	-0.008	-0.067
	29	0.036	0.037	0.017	0.005	-0.012	-0.002	-0.028
Foun-	30	0.025	0.008	-0.009	0.004	-0.001	-0.0002	-0.004
dation	31	0.020	0.007	-0.007	0.003	-0.001	-0.0002	-0.003
Mat	46	0.015	0.006	-0.006	0.002	-0.001	-0.0001	-0.003

TABLE 3.7-6 (Continued)

	Joint	1st Mode	2nd Mode	3rd Mode	4th Mode	5th Mode	6th Mode	7th Mode
Reactor	32	0.275	1.000	0.932	-0.025	-0.449	0.004	1.000
Vessel	33	0.245	0.873	0.813	-0.021	-0.385	0.002	0.812
and	34	0.222	0.778	0.724	-0.017	-0.336	0.0003	0.667
Inter-	35	0.199	0.684	0.637	-0.014	-0.288	-0.001	0.523
nals	36	0.164	0.539	0.501	-0.009	-0.203	-0.008	0.2996
	37	0.144	0.456	0.423	-0.006	-0.159	-0.011	0.182
	38	0.124	0.374	0.345	-0.003	-0.117	-0.013	0.074
	39	0.108	0.311	0.285	-0.002	-0.089	-0.013	0.002
	40	0.230	0.955	1.000	-0.044	1.000	-0.465	0.344
	41	0.179	0.668	0.675	-0.023	0.421	-0.184	0.189
	42	0.153	0.533	0.527	-0.015	0.212	-0.117	0.121
	43	0.128	0.405	0.387	-0.007	0.027	-0.045	0.052
	44	0.108	0.311	0.285	-0.002	-0.089	-0.013	0.002
	45	0.185	0.762	0.802	-0.034	0.707	1.000	-0.145

NOTE:

[1] All mode shapes given for the reactor building are in the horizontal plane.

TABLE 3.7-7

MODE SHAPES OF AUXILIARY BUILDING

Joint	1st Mode (E-W)	2nd Mode (N-S)	3rd Mode Vertical	4th Mode (E-W)	5th Mode (N-S)	6th Mode (E-W)	7th Mode (N-S)
17	0.076	0.088	0.214	-0.243	-0.325	-0.362	-0.560
27	0.471	0.436	0.603	-0.962	-0.955	-0.691	-0.479
37	0.735	0.708	0.834	-0.545	-0.434	1.000	1.000
47	1.000	1.000	1.000	1.000	1.000	-0.313	-0.255

TABLE 3.7-8

MODE SHAPES OF CONTROL BUILDING

Joint[1]	1st Mode (N-S)	2nd Mode (E-W)	3rd Mode Vertical	4th Mode (N-S)	5th Mode (E-W)	6th Mode (N-S)	7th Mode (E-W)
	(13)	(11)	(13)	(13)	(13)	(13)	(11)
	0.021	0.024	0.103	0.089	-0.095	-0.127	-0.159
	(23)	(21)	(23)	(23)	(23)	(23)	(21)
	0.210	0.211	0.318	0.729	-0.653	-0.831	-0.901
	(33)	(31)	(33)	(33)	(33)	(33)	(31)
	0.388	0.383	0.510	1.000	-0.870	-0.512	-0.508
	(43)	(41)	(43)	(43)	(43)	(43)	(41)
	0.563	0.547	0.669	0.851	-0.737	0.473	0.472
	(53)	(51)	(53)	(53)	(53)	(53)	(51)
	0.707	0.680	0.814	0.477	-0.418	1.000	1.000
	(73)	(71)	(73)	(73)	(73)	(73)	(71)
	0.867	0.855	0.929	-0.220	0.300	0.515	0.555
	(83)	(81)	(83)	(83)	(83)	(83)	(81)
	1.000	1.000	1.000	-0.904	1.000	-0.694	-0.753

NOTE:

[1] As noted in parenthesis.

TABLE 3.7-9

MODE SHAPES OF INTERMEDIATE/FUEL
HANDLING BUILDINGS

Joint	1st Mode (E-W)	2nd Mode (N-S)	3rd Mode (N-S)	4th Mode Vertical	5th Mode (E-W)	6th Mode (E-W)	7th Mode (N-S)
8	0.011	0.029	0.019	0.007	-0.051	-0.063	-0.054
9	0.011	0.031	0.020	0.007	-0.052	-0.061	-0.053
10	0.012	0.033	0.021	0.007	-0.054	-0.058	-0.052
21	0.047	0.157	0.034	-0.013	-0.214	-0.190	-0.233
31	0.097	0.279	0.060	-0.018	-0.295	-0.251	-0.356
40	0.196	0.393	-0.091	-0.023	-0.275	-0.226	-0.475
50	0.298	0.494	-0.128	-0.022	-0.196	-0.124	-0.455
60	0.373	0.562	-0.134	-0.030	-0.117	-0.013	-0.365
70	0.504	0.673	-0.153	-0.038	0.057	0.198	-0.109
80	0.693	0.806	-0.200	-0.036	0.451	0.625	0.133
90	0.786	0.865	-0.216	-0.036	0.617	0.738	0.234
100	1.000	1.000	-0.286	-0.017	1.000	1.000	0.467
20	0.038	0.070	0.127	0.031	-0.158	-0.166	-0.110
30	0.065	0.111	0.216	0.035	-0.229	-0.243	-0.086
11	0.065	0.115	0.397	0.136	-0.229	-0.243	-0.086
12	0.304	0.292	0.884	0.780	-0.401	-0.619	1.000
13	0.440	0.358	1.000	1.000	-0.436	-0.598	0.940

TABLE 3.7-10

<u>PERCENT OF CRITICAL DAMPING</u>
<u>ROCKING OF ENTIRE BUILDING</u>

Material	Range of Shear Wave Velocity	Damping Factor
On rock	C $\geq$ 6,000 fps	2-5
On firm soil	C $\geq$ 2,000 fps	5-7
On soft soil	C < 2,000 fps	7-10

TABLE 3.7-11

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT

	Frequency Bandwidth (Hertz)		
	0 − 10	10 − 20	20 − 50
Total Number of Seismic Cycles	168	359	643
No. of Seismic Cycles (0.5% of Total) between 75% and 100% of Peak Loads	0.8	1.8	3.2
No. of Seismic Cycles (4.5% of Total) between 50% and 75% of Peak Loads	7.5	16.2	28.9

TABLE 3.7-12

CRITICAL DAMPING RATIOS FOR DIFFERENT MATERIALS (NSSS)

| Item | Percent Critical Damping[1][2] | |
	OBE Condition	SSE Condition
Welded structural assemblies (equipment and supports)	2.0	3.0
Vital piping systems - Diameter greater than 12 in.	2.0	3.0
- Diameter less than or equal to 12 in.	1.0	2.0
Equipment	2.0	2.0
Reactor pressure vessel, support skirt, shroud head, separator, and guide tubes	2.0	4.0
Fuel	6.0	6.0

NOTES:

[1] Other values may be used if they are indicated to be reliable by experiment or study.

[2] Damping values as defined in ASME Code Case N-411-1 may be utilized for building filtered response spectra piping stress analysis.

TABLE 3.7-13

SEISMIC INSTRUMENTATION

SENSING ELEMENTS[1]

Sensing Element Description	Tag No.	Location	Mounting Type	Lowest Natural Frequency of Mount[2]

TABLE 3.7-13 (Continued)

Sensing Element Description	Tag No.	Location	Mounting Type	Lowest Natural Frequency of Mount[2]
Accelerometer	0D51N0232	Reactor building foundation in the fuel handling building at Elevation 574'-10"	Bolted to bracket which is bolted to shield building wall	See Note[4]
Accelerometer	0D51N0231	Reactor building internal structure at azimuth 260 deg Elevation 644'-6"	Bolted to bracket which is bolted to internal structure wall	See Note[3]

TABLE 3.7-13 (Continued)

Sensing Element Description	Tag No.	Location	Mounting Type	Lowest Natural Frequency of Mount[2]
Accelerometer	0D51N0230	Reactor building internal structure at azimuth 275 deg Elevation 664'-7"	Bolted to concrete floor	See Note[3]
Accelerometer	0D51N0233	Intermediate building foundation at Elevation 574'-10"	Bolted to concrete floor	See Note[3]
Accelerometer	0D51N0234	Intermediate building at Elevation 599'-0"	Bolted to concrete floor	See Note[3]
Accelerometer Free Field	0D51N0240	North of Microwave Tower	Bolted to concrete on a poured foundation	See Note[3]

NOTES:

[1] All listed sensing elements are triaxial.

[2] Lowest natural frequency, in any of three dimensions, of mounting bracket.

[3] This instrument does not use a bracket as a mounting adapter. The instrument is bolted rigidly to a plate embedded in the concrete surface that it monitors and is free from spurious resonances within its frequency range.

[4] The instrument uses a bracket or plate as a mounting adapter which is rigidly attached to the structure that it monitors and is free from spurious resonances within its frequency range (Reference 19).

TABLE 3.7-14 (DELETED)

3.8   DESIGN OF CATEGORY I STRUCTURES

3.8.0   GENERAL DESCRIPTION

3.8.0.1   Reactor Building Complex

The reactor building complex consists of the interior structure
(including drywell and suppression pool), containment vessel, annulus
concrete, and shield building. These structures house and protect the
reactor and some safety class equipment. The structures are supported
by a common foundation mat at Elevation 574'-10". The interior
structure is separate from the containment vessel, annulus concrete and
shield building above the mat, however, the containment vessel and
shield building are tied together by the annulus concrete, to
Elevation 598'-4". The relationship of the structures is shown in
<Figure 3.8-1>. If a Loss-of-Coolant Accident (LOCA) occurs, these
structures function together to contain the released materials and
energy. For a complete physical description of each of these structures
refer to <Section 3.8.1.1> for the shield building, <Section 3.8.1.8>
for the annulus concrete, <Section 3.8.2.1> for the containment vessel,
and <Section 3.8.3.1> for the interior structure.

The pressure and temperature transients for the drywell and containment
vessel during the postulated Loss-of-Coolant Accidents (LOCA) are
described in detail in <Section 6.2>.

In summary, the effects of one of these LOCA (main steam line break) on
the drywell and containment vessel are:

a.   Steam line pipe in the drywell breaks.

b. Pressure and temperature build up rapidly in drywell to maximum
    calculated values as discussed in <Section 6.2.1>.

c. Increase in drywell pressure causes steam-air mixture to force down
    the water in the interspace between the weir wall and the inside of
    the drywell wall, and to blow this mixture through the vent holes
    in the drywell wall. This action causes an almost complete
    condensing of the steam into the suppression pool water.

d. As the blowdown continues, air, noncondensible gases and some steam
    escape to the containment vessel atmosphere. This causes the
    pressure and temperature in the containment vessel to increase to
    maximum calculated values as discussed in <Section 6.2.1>.

e. After the Emergency Core Cooling System (ECCS) has functioned for a
    short period and the reactor has been scrammed, the water steam
    mixture cascading from the ruptured pipe causes a condensing of the
    steam air mixture in the drywell. The condensing in turn causes an
    inward acting pressure differential of approximately 6.8 psi
    (calculated maximum on the drywell). The drywell is designed for
    -21.0 psi differential pressure; operation of the drywell vacuum
    breakers is not required to protect the drywell structure.

The annulus space between the containment vessel and shield building is
maintained at a slight negative pressure during both normal plant
operation and a LOCA. Any radioactive material which might escape from
the containment vessel will be collected in the annulus space. The
annulus exhaust gas treatment system which maintains the annulus space
at a slight negative pressure, will collect and treat radioactive
material prior to discharge. A more complete description of the
functional criteria, design bases, etc., is given in <Section 3.8.3.1>,
<Section 3.8.2.1> and <Section 3.8.1.1> for the interior structure
(drywell), containment vessel and the shield building, respectively.

Other smaller postulated steam line breaks, as described in
<Section 6.2>, give less critical pressure conditions for both the
drywell and the containment vessel. These pressure loads are also
considered in the design.

3.8.0.2   Other Safety Class Structures

The relationship of the safety class structures is shown in
<Figure 3.2-1>. The Reactor Building Complex and the other safety class
structures are separated from each other to preclude interaction between
the structures. For a detailed description of the other safety class
structures refer to <Section 3.8.4>.

3.8.0.3   Foundations and Concrete Supports for Safety Class
            Structures

The relationship, design and configuration of the foundations of safety
class structures is discussed in <Section 3.8.5>.

3.8.0.4   Expansion Bolt Installation in Safety Class Concrete
            Structures

During the operations phase, Hilti Kwik-Bolt, Hilti Kwik-Bolt II, Hilti
Kwik-Bolt 3 and Drillco Maxi-Bolt type expansion anchors will typically
be used as required to attach systems, components or structures to
existing Safety Class concrete structures. These expansion anchors will
be installed in accordance with approved procedures and instructions
such that the structural integrity of the Safety Class concrete
structures is not impaired due to the bolt installation process
<Section 3.8.3.3.8>.

3.8.1   CONCRETE CONTAINMENT

The containment structure for this plant is composed of the free standing steel containment vessel and the annulus concrete acting compositely. A shield building forms a housing for the steel containment and annulus concrete which strengthens and stiffens the containment vessel. Both the shield building and the annulus concrete are discussed in this section beginning with the shield building. The shield building has the following functions:

a.   Forms a biological shield for radiation from the reactor.

b.   Provides protection for the containment vessel from ground water contact and pressure.

c.   Provides weather and exterior missile protection for the containment vessel.

d.   Provides a relatively leak tight structure so that the annulus exhaust gas treatment system can be used to minimize the escape of radioactive particles to the environment, by maintaining the annulus air space at a slight negative pressure.

3.8.1.1   Description of the Shield Building

The shield building is a reinforced concrete structure consisting of a flat foundation mat, a cylindrical wall and a shallow dome. The general configuration of the shield building and its relation to the other structures of the reactor building complex is shown in <Figure 3.8-1>. The foundation mat, common to the shield building, annulus concrete, containment vessel, and interior structure is circular in plan with a diameter of 136 feet and a thickness of 12 feet 6 inches. The

foundation mat is founded on Chagrin shale at Elevation 562'-3",
approximately 56 feet below grade. Foundations are discussed in detail
in <Section 3.8.5>.

The shield building cylindrical wall extends from the top of the
foundation mat at Elevation 574'-10" to Elevation 749'-9" and has
outside diameter of 136 feet with a wall thickness of 3 feet 0 inches.
Shallow dome has a radius of 120 feet 0 inches, with a thickness of
2 feet 6 inches. There is no thickened ring girder, but the elevation
of the wall at the junction of the wall and dome was raised to provide a
greater section to help resist the outward thrust of the dome.

A cross section of the ring girder is shown in <Figure 3.8-2>. Details
of the typical reinforced section for the shield building wall and dome
are shown in <Figure 3.8-3>. Typical details of the cylindrical wall
foundation mat junction can be found in <Figure 3.8-82>.

Access to the inside of the containment vessel through the shield
building is provided by two personnel airlocks of approximately 9 feet
6 inches in diameter. These are described in <Section 3.8.2> and shown
in <Figure 3.8-4>.

The equipment access opening is octagonal in shape, and 20 feet across
the flats. This opening is shielded by removable reinforced concrete
beams which are provided with seals to minimize the leakage of air. See
<Figure 3.8-5> for more details.

Details of mechanical and electrical penetrations are discussed in
<Section 3.8.2> and shown in <Figure 3.8-6> and <Figure 3.8-7>. The
design of penetration sleeves is such that they allow differential
movement between the shield building and the containment vessel. A 2"
thickness of compressible material has been provided around the six
penetrations within the annulus concrete to permit differential movement
between the containment vessel and shield building. Typical details are

shown in <Figure 3.8-104>. At the shield building wall the penetration
sleeves are designed to maintain the leak tightness requirements for the
shield building.

The functional criteria for the shield building does not require vacuum
breakers. Purge valves are required as part of the annulus exhaust gas
treatment system.

The concrete will have a minimum 28 day cylinder compressive strength of
3,000 psi. The steel reinforcement is in accordance with the
requirements of ASTM A615-72, Grade 60.

Waterproofing of the portion of the shield building below grade is
described in <Section 3.8.5>.

A general reinforcing pattern of orthogonal bars arranged vertically and
circumferentially in both faces of the wall was used in the shield
building wall. A description of the reinforcement is provided below and
in <Table 3.8-12>. <Table 3.8-12> also shows design axial forces,
moments and transverse shears with governing load combinations at
critical sections of the shield building.

Due to local non-axisymmetric loadings, the reinforcement for the wall
has three distinct designs around the circumference as follows:

a.   Area subjected to soil pressures at centerline 97.5 degrees azimuth
     for Unit 1 has vertical reinforcement on the outside face of No. 18
     at 12 inches alternated with No. 11 at 12 inches and an inside face
     reinforcement of No. 11 at 12 inches alternated with No. 9 at
     12 inches. Unit 2 is similar.

b.   Steam tunnel area subjected to pressure and pipe anchor loadings at
     centerline 0 degrees azimuth has an outside face reinforcement of

No. 14 at 12 inches alternated with No. 11 at 12 inches and an
inside face reinforcement of No. 11 at 6 inches.

c. The remainder of the structure which is not subjected to any local
    loadings has an outside face reinforcement of No. 11 at 12 inches
    alternated with No. 9 at 12 inches and an inside face reinforcement
    of No. 11 at 12 inches alternated with No. 9 at 12 inches.

The horizontal reinforcement for areas a. and b., above, was No. 9 at
6 inches each face for the lower 30 feet of the wall. For the remainder
of the structure at this elevation the reinforcement was No. 9 at
6 inches outside face and No. 9 at 12 inches inside face. The vertical
reinforcement generally decreased above this region to No. 9 at
12 inches each face at the ring girder. The horizontal reinforcement
also generally decreased above this elevation until just below the ring
girder where it increased to No. 11 at 6 inches each face.

The reinforcing pattern for the dome is essentially radial and
circumferential with the center section arranged orthogonally for ease
of placing. Radial reinforcement for the dome is generally No. 9 at
12 inches decreasing to No. 8 at 12 inches toward the apex.
Circumferential reinforcement is No. 11 at 12 inches near the ring
girder decreasing to No. 8 at 12 inches toward the apex. <Figure 3.8-3>
shows the typical wall, ring girder and dome reinforcement.

3.8.1.2   Applicable Codes, Standards and Specifications to the
            Design

3.8.1.2.1   Codes

a. ACI 318-71, "Building Code Requirements for Reinforced Concrete,"
    American Concrete Institute.

Chapters Used

1	General Requirements
2	Definitions
7	Details of Reinforcement
8	Analysis and Design - General
9	Strength and Serviceability
10	Flexure and Axial Loads
11	Shear and Torsion
12	Development of Reinforcement
19[8]	Shells and Folded Plate Members

Appendix B

Chapters Not Used

3[1]	Materials
4[1]	Concrete Quality
5[1]	Mixing and Placing Concrete
6[1]	Formwork, Embedded Pipes and Construction Joints
13[2]	Slab Systems with Panels
14[3]	Walls
15[4]	Footing
16[5]	Precast Concrete
17[6]	Composite Concrete Flexural Members
18[7]	Prestressed Concrete
20[9]	Strength Evaluation of Existing Structures

Chapters Not Used

Appendix A[10]   Special Provision for Seismic Design

NOTES:

(1) ACI 301-72 will be used to define requirements in this area as modified by <Section 3.8.1.6>.

(2) Panel slab systems are not used on this structure.

(3) The cylindrical wall is more properly considered a shell structure than a conventional wall.

(4) This structure has a foundation mat not individual footings.

(5) The shield building is not a precast concrete structure.

(6) Composite flexural members are not used on the shield building.

(7) The shield building is not a prestressed concrete structure.

(8) Chapter 19 is used except that Sections 19.2.2, 19.5.1, 19.5.2, 19.5.4, and the last sentence of Section 19.5.3 are not applicable.

(9) The shield building is not an existing structure.

(10) The provisions of Appendix A apply to moment resisting frames and shear walls.

b. ACI 301-72, "Specification for Structural Concrete for Buildings," American Concrete Institute.

All chapters of ACI 301 are applicable except Chapter 15 which has requirements for prestressed concrete.

c. ACI 307-69, "Specification for the Design and Construction of Reinforced Concrete Chimneys," American Concrete Institute.

Chapters 4.6, 4.7 and 4.8 of ACI 307-69 are applicable in determining thermal stresses. All other chapters or parts of chapters are not applicable because of their specialized nature (applicable to reinforced concrete chimneys).

d. Ohio Building Code 1970 Edition, Chapter BB-33, "Reinforced and Plain Concrete, Reinforced Gypsum Concrete and Reinforced Masonry,"

Board of Building Standard Department of Industrial Relations,
State of Ohio.

Sections Used

BB-33-01 Definitions
BB-33-02 Abbreviations BB-33-03 - Accepted engineering practice
          and approved standards
BB-33-07 Reinforced concrete - general design and construction

Sections Not Used

BB-33-04[1]   Reinforced concrete materials
BB-33-05[1]   Concrete, classification and quality
BB-33-06[1]   Concrete, ready mixed
BB-33-08[2]   Reinforced concrete, supervision
BB-33-09 to
BB-33-13[3]   Plain concrete

Sections Not Used

BB-33-14[4]   Reinforced gypsum concrete, precast
BB-33-15[4]   Reinforced gypsum concrete, design and construction
BB-33-16 to
BB-33-20[5]   Reinforced masonry

NOTES:
[1] ACI 301-72 will be used in this area as modified by
    <Section 3.8.1.6>.
[2] Not applicable to this design.
[3] Plain concrete is not used on this structure.
[4] Reinforced gypsum is not used on this structure.
[5] Reinforced masonry is not used on this structure.

3.8.1.2.2   Standards

a.   ASTM Standards: Applicable ASTM standards are discussed in the
      following sections. The date of a particular standard may vary for
      different items because of the difficulty in purchasing material to
      an outdated standard. Since the latest ASTM standards reflect
      industry practice used for fabrication and erection, it was
      permitted to use an updated standard where no unacceptable loss of
      quality would result.

      1.   Concrete: <Section 3.8.1.6.1>

      2.   Reinforcing Steel: <Section 3.8.1.6.2>

      3.   Cadweld Splices: <Section 3.8.1.6.3>

b.   Applicable Regulatory Guides to Design

      Regulatory guides pertaining to seismic design classification and
      seismic design are referenced in <Section 3.2> and <Section 3.7>
      respectively.

      1.   <Regulatory Guide 1.10>, "Mechanical (Cadweld) Splices in
           Reinforcing Bars of Category I Concrete Structures." This
           guide was used with modifications specified in
           <Section 3.8.1.6.3>.

      2.   <Regulatory Guide 1.15>, "Testing of Reinforcing Bars for
           Category I Concrete Structures." This guide was used with
           modifications specified in <Section 3.8.1.6.2>.

      3.   <Regulatory Guide 1.55>, "Concrete Placement in Category I
           Structures."

4.   <Regulatory Guide 1.142> (April 1978), "Safety-Related
      Concrete Structures For Nuclear Power Plants (Other Than
      Reactor Vessels and Containments)." We generally comply with
      this guide except the 0.9 load factor for dead load was not
      used as required by Item 11. We have assessed this reduced
      load factor on several key structures including the Drywell,
      Reactor Building foundation mat, the Auxiliary Building roof
      slab, and the Fuel Handling Building roof slab. Our
      evaluations demonstrate that we meet the intent of this
      Regulatory Guide requirement.

3.8.1.2.3   Principal Plant Specifications

The principal specifications for the shield building are:

a.   Concrete supply.

b.   Fabrication and placing of reinforcing steel and embedded items.

c.   Placement of structure concrete.

d.   Supply and installation of waterproofing and waterstops.

These specifications include the applicable design requirements of the
shield building. The detailed material specifications, the detailed
quality control provisions and any special construction technique
requirements all as described in <Section 3.8.1.6>, are included in
these specifications.

3.8.1.3   Loads and Load Combinations

3.8.1.3.1   Loads Used in the Design

a.   Dead Load (symbol D)

     Dead load includes the weight of the shield building plus any
     permanent attachments including cable trays and piping less than
     18" diameter.

b.   Live Load (symbol L)

     1.   For the dome of the shield building:

          (a) 25 psf live load.

          (b) 84.5 psf snow and ice load based on the meteorological
               recommendations of <Section 2.3.1>.

     2.   For stairs and platforms - 100 psf

     3.   Any equipment or systems not permanently attached.

     4.   Piping 18-inch diameter or larger.

c.   Groundwater Loads (symbol G)

     The design bases for groundwater loading conditions are described
     in <Section 2.4.13>, <Section 2.5.4>, and <Section 3.8.5.3.4>.

d.   Lateral Soil Pressures (symbol H for static and F_{eqo} and F_{eqs} for
     dynamic soil pressures)

The static and dynamic soil pressures used in the design are given in <Section 2.5>. Static lateral soil pressures are included under the symbol H. Dynamic lateral soil pressures are included under the symbols F_{eqo} or F_{eqs} as appropriate.

e.   Normal Operating Temperature Induced Forces (symbol T_o)

Thermal effects and induced loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition calculated from:

1.   Annulus space: maximum temperature   104°F, normal operating

                    minimum temperature   50°F, shutdown

2.   Ambient temperature: maximum 7 day mean 83°F
                           minimum 7 day mean 10°F

3.   Soil temperature: average ground temperature 55°F.

f.   Wind and Tornado Loads (symbols W for design wind W_t and for tornado wind, and Z for pressure differential due to the tornado pressure drop).

The wind and tornado loads including tornado pressure drop, are described in <Section 3.3>.

g.   Seismic Loads (symbols F_{eqo} for OBE and F_{eqs} for SSE)

Seismic loads are the forces imparted to the structure by ground accelerations due to the OBE and SSE. For further details on the methods of evaluating seismic response refer to <Section 3.7>. The effect of the interaction between the containment vessel, annulus concrete and shield building was considered.

h.   Annulus Pressure Load

The annulus space is maintained at 0.40 inch water gauge in water less than atmospheric during normal operating and accident conditions. This extremely low pressure, equivalent to 0.015 psi, is not a significant load and consequently, is ignored in the design.

i.   Accident Temperature and Pressure

1.   Accident Temperature (symbol T_a)

Thermally induced loads generated by the postulated break and including T_o.

2.   Accident Pressure Loads (symbol P_a)

Pressure equivalent static load generated by the postulated break including an appropriate dynamic load factor to account for any dynamic nature of the pressure pulse. The direct pressure effects due to pool swell and other LOCA pressures transmitted from the containment vessel through the filled annulus concrete into the shield building are considered in the design of these structures.

j.   Penetration Loads (symbols R_o, R_a, Y_r, Y_j, and Y_m)

In general, the penetration design ensures that no load is transmitted to the shield building with the exception of the steam tunnel area. The two feedwater lines are anchored in the shield building wall. All other piping systems with guard pipes are anchored in the drywell wall and are provided with guard pipe guides at the shield building wall. These guides transmit only in-plane loads to the shield building wall. The equipment hatch

and personnel access airlocks are supported from the containment vessel and transmit no load to the shield building wall.

The feedwater penetration anchorages are designed to act as pipe restraints for axial, moment and torsional loads. These are treated as local loads on the shell and are applied as a result of interaction of the sleeve anchor and reinforced concrete wall. The penetration loads were broken down into the following categories:

1. Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition (symbol R_o).

2. Pipe reactions under thermal conditions generated by the postulated break and including R_o (symbol R_a).

3. 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 factor to account for the dynamic nature of the load (symbol Y_r).

4. Jet impingement static load on the structure generated by the postulated break, including an appropriate dynamic factor to account for the dynamic nature of the load (symbol Y_j). The shield building wall exposed to the steam tunnel is designed for the direct jet impingement due to the postulated rupture of high energy piping as described in <Section 3.6>.

5. Missile impact equivalent static loads generated by or during the postulated break, like pipe whipping, and including an appropriate dynamic factor to account for the dynamic nature of the load (symbol Y_m).

k.   Missile Loads (symbol M)

The missile loads for the design are the tornado missile as
described in <Section 3.5>.

l.   Explosions and Flammable Vapor Clouds (symbol P_r)

The exterior surfaces of the shield building are designed to
withstand the effects of the postulated blast loading discussed in
<Section 2.2.3>. The assumed pressure distribution for the walls
is similar to that used for the wind and tornado loadings with a
maximum positive pressure of twice the calculated peak overpressure
equal to 2.4 psi. The Shield Building dome is designed for a
positive pressure equal to the peak calculated overpressure of
1.2 psi.

m.   Safety/Relief Valve Operation (symbols P_{srv} and T_{srv})

The induced loads on the shield building are due to the pressure
(P_{srv}) and thermal (T_{srv}) transients associated with the operation of
the safety/relief valve system. The direct pressure effects are
transmitted only through the region of the annulus concrete. The
vibratory response loads are considered on the entire structure. A
more detailed discussion of the loads is presented in <Appendix 3A>
and <Appendix 3B>.

3.8.1.3.2   Load Combinations

a.   Combinations

The load combinations listed in <Table 3.8-1> for reinforced
concrete structures were investigated in determining the most
critical conditions for design. Many of the loads vary within a
possible range. For this type of load the specific value used is

that which produced the most critical conditions for the load combination being considered.

b.   Design Approach

The load combinations discussed above are in accordance with the requirements of the applicable codes: ACI 318-71, ASME Code Section III, Division 2 and "Structural Design Criteria for Category I Structures Outside Containment" (Reference 1) for reinforced concrete using the strength design method and the criteria discussed in <Section 3.8.1.4>. Structural steel members are designed in accordance with the 1969 AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings." The design approach for missile design and pipe whip restraint is described in <Section 3.5> and <Section 3.6> respectively.

3.8.1.3.3   Load Factors

a.   Load Factors Based on "Structural Design Criteria for Category I Structures Outside Containment:" (Reference 1)

The load factors presented in <Table 3.8-1> are in accordance with (Reference 1), except for the groundwater (G) load factor. The factor for the groundwater pressure is derived from ACI 318 and is the same as for dead load. This factor has been adopted because of the conservative assumption in groundwater level and the exactness with which the load can be calculated.

Equation (13) uses load factors of 1.0 for the postulated gas line, or gas storage explosion. The factors of 1.0 have been adopted because of the conservatism in the blast loading calculations, and the manner in which the pressures are applied. This is consistent with the philosophy applied to tornado and missile loadings.

b. Dynamic Load Factors

In general, dynamic load factors apply to loads where local areas
(restraints, barriers, etc.) are allowed to deform and to absorb
energy. Deformation limits are conservative as compared to
ultimate deformations.

For structural members supporting restraints, barriers, etc.,
equivalent static loads are conservatively chosen from dynamic
analysis of the local areas (i.e., maximum reactions in critical
directions) or conservatively estimated using (Reference 2) or
(Reference 3). Appropriate dynamic load factors are included in
the definition of P_a, Y_r, Y_m, Y_j, and M in agreement with
(Reference 1). The dynamic load factor is in addition to the "load
factors" explicitly included in the load combinations shown in
<Table 3.8-1>.

Design for pipe whip restraint, and jet impingement loads is
discussed in <Section 3.6>. Design for missile loads is discussed
in <Section 3.5>.

3.8.1.3.4   Load Combinations for Local Areas

The load factors and load combinations for the design of local areas are
the same as those used for the general structure and outlined in
<Table 3.8-1>. Methods of analyzing for local loads and for the effects
of discontinuities are discussed in <Section 3.8.1.4>. Loads; defined
by the symbols R_o, R_a, Y_r, Y_j, Y_m; and due to pipe reactions and jet
impingement forces are generally localized in nature. The tornado
missile load (M symbol) is a localized load but of general application
to all parts of the Shield Building above grade, and not protected by
adjacent structures.

3.8.1.3.5   Time Dependent Loads

The Shield Building is not a prestressed concrete pressure vessel therefore variation in loads due to time dependent losses such as creep and shrinkage are not significant.

3.8.1.3.6   Thermal Effects

Temperature profiles based on the temperatures given in <Section 3.8.1.3.1.e> are shown in <Figure 3.8-10>. These profiles are the basis for selection of the critical temperature induced loads to be used in load combinations presented in <Table 3.8-1>.

3.8.1.3.7   Extent of Compliance to ACI 349-76 "Code Requirements for
                 Nuclear Safety-Related Concrete Structures," American
                 Concrete Institute.

The extent of compliance to ACI 349-76 is discussed in <Section 3.8.3.3.7>.

3.8.1.4   Design and Analysis Procedures

Since the shield building is not a containment, it was not designed in accordance with Article CC-3000 of ASME Proposed Section III, Division 2, but rather to the applicable codes described in <Section 3.8.1.2>.

3.8.1.4.1   Assumptions of Boundary Conditions

a.   In analysis, the basic assumption in accordance with ACI 318-71 Section 19.2.1 is that the material is ideally elastic, homogeneous and isotropic.

b.   The concrete design is based on the assumptions given in
     Section 10.2 of ACI 318-71 including the assumption that tensile
     strength of the concrete is neglected in calculations (cracked
     section).

c.   The analysis assumed that the structure is fixed at the Shield
     Building wall to reactor building mat junction for all loads except
     for thermal. The mat stiffness was included for this load. An
     accurate representation of the ring girder (dome to wall junction)
     was included in the analytical model such that no additional
     assumptions on boundary conditions were required.

3.8.1.4.2   Treatment of Axisymmetric and Non-axisymmetric Loads

a.   Axisymmetric Loads

     The shield building was analyzed and designed for the following
     axisymmetric loads which are defined in <Section 3.8.1.3>:

     1.   Dead Load

     2.   Live Load

     3.   Groundwater Loads

     4.   Normal Operating and Accident Temperature Induced Forces

     The analytical approach used is described in <Section 3.8.1.4.3>.

b.   Non-axisymmetric Loads

     The shield building was analyzed and designed for the following
     non-axisymmetric loads and are as defined in <Section 3.8.1.3>:

     1.   Lateral Soil Pressures

2.   Wind Loads

3.   Tornado Wind Loads

4.   Seismic Loads (OBE and SSE)

5.   Local pipe reaction and pipe rupture forces including jet
      impingement forces and local compartmental pressure loads on
      the shell outside the shield building due to postulated pipe
      breaks within the auxiliary building steam tunnel compartment.

6.   Vibratory and direct pressure effects as transmitted through
      the containment vessel and/or annulus concrete due to Steam
      Relief Valve Operation.

7.   Asymmetric pool swell air bubble loads as defined in
      <Appendix 3B>.

The effective pressures for the above loads were represented in the
ELAD or ASHSD 2 computer programs using Fourier series.

Input data for the OBE and SSE analysis of the Shield Building
consisted of inertial forces obtained from the seismic analysis
results as described in <Section 3.7>.

The only pipes anchored in the shield building wall are the
feedwater lines. The wall penetration sleeve is designed as an
anchor to transmit all normal operating and pipe rupture loads into
the wall.

The vibratory loads due to steam relief valve operation were
evaluated in the same manner as the seismic loadings.

3.8.1.4.3   Analytical Techniques

a.   Cylindrical Shell, Dome and Annulus Concrete

    1.   Static Analysis

        The static analysis of the shield building was initially
        performed using the computer program ELAD. ELAD is a finite
        element program designed to determine elastic deformations,
        strains and principal values of stress within axisymmetric
        solid structures or arbitrary shape subjected to axisymmetric
        or non-axisymmetric pressure, concentrated loads and
        temperatures. All boundary conditions consistent with the
        theory of elasticity are permitted. The axisymmetric finite
        element model used in ELAD for determining the structural
        response to both the axisymmetric and non-axisymmetric
        loadings is shown in <Figure 3.8-9>.

        Since the annulus concrete was added to the shield building
        after the shield building had been constructed, an additional
        static analysis of the containment vessel, shield building and
        annulus concrete was performed using the ASHSD 2 and ANSYS
        computer programs. ASHSD 2 is a finite element program with
        the capability to determine elastic deformations and stresses
        within axisymmetric shell or solid structures of arbitrary
        shape subjected to axisymmetric or non-axisymmetric pressure
        or concentrated loads. All boundary conditions consistent
        with the theory of elasticity are permitted. The axisymmetric
        finite element model used for the ASHSD 2 analyses is shown in
        <Figure 3.8-105>. ANSYS is a general purpose finite element
        program with the capability to solve static and dynamic
        problems involving elasticity, plasticity, creep, and
        swelling, buckling problems, and steady-state and transient
        heat transfer, and fluid flow problems. The axisymmetric

finite element model used for the ANSYS analyses is shown in
<Figure 3.6-106>.

The ASHSD 2 computer program was used to analyze the shield
building and annulus concrete for the accident pressures,
groundwater pressure, steam tunnel pressure, suppression pool
water pressure, and seismic loads. The ANSYS computer program
was used to analyze the lower shield building and annulus
concrete for thermal loads.

The original ELAD analysis results for dead and soil pressure
loads were used since the shield building was already
constructed and the soil against the shield building was
already backfilled without the annulus concrete being in
place.

In the determination of shield building thermal stresses, for
all sections away from the discontinuities, the cracked
section approach as presented in ACI 307-69 was used. The
equations were modified to reflect Poisson's effect by
dividing by (1-n). For the ring girder area, reinforcement
was provided based on a completely elastic thermal stress
analysis, using the ELAD model.

2.   Dynamic Analysis (Seismic Analysis)

The shell and dome are analyzed for seismic ground motions by
means of the program DYNAL. Using a lumped mass model of the
structure, the program calculates the flexibility matrices and
the natural frequencies and mode shapes of the structure.
These properties are then used along with the time history
ground response to generate structural responses at points of
mass concentration. See <Section 3.7> for details of seismic
analysis.

Safety/relief valve discharge, pool swell, condensation
oscillation, and chugging loads were analyzed by use of the
ASHSD 2 computer program and the model shown in
<Figure 3.8-105>. These loads are described in <Appendix 3A>
and <Appendix 3B>.

The safety/relief valve discharge load used was greater than
the loads presented in <Appendix 3A> and <Appendix 3B>. The
condensation oscillation and chugging loads produced stresses
which are small and could be neglected.

The structural responses are in terms of velocity,
acceleration and deflection. The accelerations are used to
give equivalent static loads which are analyzed by the method
described in <Section 3.8.1.4.3.a.1>.

3.   Exterior Missiles:

The analytical techniques for missile loads are discussed in
<Section 3.5>.

4.   Heat Transfer Analysis:

For steady-state and transient heat transfer, two dimensional
analyses were performed using the HEATING 2 (Reference 4)
digital computer code. Transient and steady-state temperature
profiles were obtained for the conditions of startup,
shutdown, normal operation, and LOCA. These are shown in
<Figure 3.8-10>. The temperature profiles are the basis for
the selection of critical temperatures which are analyzed as a
static load using the methods described in
<Section 3.8.1.4.3.a.1>.

b. Foundation Mat

    The analysis of the foundation mat is discussed in <Section 3.8.5>.

c. Local Areas

    1. Cylindrical Wall to Mat Junction

        A fixed base was used for the ELAD model in all load
        evaluations including thermal. Reinforcement was provided for
        the forces predicted from these analyses.

        A fixed base was used for the shield building and containment
        vessel and a free base was used for the annulus concrete in
        the ASHSD 2 model for all load evaluations except thermal. In
        order to get a more accurate assessment of the thermal forces
        at the base of the shield building, the mat foundation
        stiffness and thermal effects were included in the ANSYS
        model.

    2. Ring Girder:

        The ring girder area (dome to cylindrical wall junction) was
        included in the full ELAD model of the shield building and,
        therefore, required no assumptions on boundary conditions.
        Elastic thermal stresses were used for the design of
        reinforcement in this area.

    3. Openings and Penetrations

        The only penetrations on the shield building for which there
        are anchoring forces normal to the wall are the feedwater
        penetrations. Special analyses were not performed for any

penetrations except the feedwater anchors. However,
additional reinforcing was provided around all penetrations to
satisfy strength requirements and to minimize cracking.
Details of this reinforcement are discussed in
<Section 3.8.1.4.5>.

Because the shield building wall in the area of the feedwater
line anchor is broken up into very distinct beam strips by
numerous penetrations, a STRUDL frame model as shown in
<Figure 3.8-11> was used to analyze this area for the local
feedwater line anchor forces. The boundary conditions were
varied to determine the maximum bending moments. Results from
this analysis were then superimposed on the results from the
ELAD general analysis and reinforcement provided for the
combined results.

d.   Effects on Analysis of Variations in Material Properties and
     Assumptions

     For seismic design the effect of variations in material properties
     and assumptions are discussed in <Section 3.7>. For static
     analyses the basic assumptions are given in <Section 3.8.1.4.1>.
     These are the usual assumptions made for concrete structures under
     the ACI 318-71 code. Material properties and quality control
     specified in <Section 3.8.1.6>, assure that material properties are
     within the ranges of values anticipated by ACI 318-71. For
     concrete structures designed by this code the variations in
     assumptions and material properties are allowed for in two ways:

1.   Load Factors, according to the "Commentary on Building Code
     Requirements for Reinforced Concrete (ACI 318-71)"
     Section 9.1, are used "to provide for excess load effects ...
     and simplified assumptions in structural analysis."

2. The φ factors which "provide for the possibility that small adverse variations in material strengths, workmanship and dimensions ... may combine to result in undercapacity." (Section 9.1 of "Commentary on Building Code Requirements for Reinforced Concrete") ACI 318-71. The φ factors for the analyses are listed in <Section 3.8.1.4.5> and are based on ACI 318-71 Section 9.2.

3.8.1.4.4   Expected Behavior Under Loads

a.   Non-axisymmetric, localized, transient loads

The methods of analysis and design used for these structures are expected to reproduce the behavior of the constructed system. The analytical and design methods considered axisymmetric, non-axisymmetric and local loads, transient thermal gradients and the effects of discontinuities.

3.8.1.4.5   Design Methods

a.   General:

The method of design, used for reinforced concrete sections is the strength design method as outlined in ACI 318-71.

All stresses and strains are in accordance with the member capacity requirements as set forth in the loading combinations of <Section 3.8.1.3.2> with the exception of missile loadings which were handled as discussed in <Section 3.5>.

The capacity reduction factors will be in accordance with Section 9.2 of ACI 318-71 as follows:

φ factor	Type of member load
0.9	Bending
0.9	Axial compression
0.9	Axial tension
0.85	Shear and torsion
0.70	Bearing on concrete

b. Deformations and Deflections:

No permanent overall deformation occurs during any of the factored loading combinations set forth in <Section 3.8.1.3.2>. However, under nonfactored missile loads plastic deformation of very localized areas is permitted as described in <Section 3.5>.

c. Thermal Limitations:

The design of the shield building is based on limiting the temperature of the reinforced concrete during normal plant operation to 150°F except for local areas such as pipe penetrations where 200°F maximum is permitted.

d. Vital Subcompartments:

The shield building has no vital subcompartments.

e. Methods of Reinforcing Critical Areas

As noted in <Section 3.8.1.4.3.c>, the only penetrations for which there are anchoring forces normal to the shield building wall are the feedwater penetrations. For all openings except the feedwater penetrations, the orthogonal reinforcement was terminated at the openings and the reinforcement area is replaced on each side and above and below the openings. To control local cracking and

strains, additional reinforcement was added diagonally around the penetrations in two directions on each face.

The reinforcement in the steam tunnel area of the shield building was based on the results of both the ELAD general analysis and the STRUDL frame analysis for the local feedwater anchor loads. Reinforcement was placed along the effective beam strips between penetrations. Diagonals were again added in two directions in each face to control local cracking and strains but were not considered to add to the overall strength of the structure nor its ability to generate internal forces to equilibrate the affects of the applied loads. Transverse shear reinforcement was provided based on the combined results of the two analyses. The feedwater pipe anchor forces were large enough to produce a punching shear requirement for radial shear ties. These ties are provided based on Equation (11-13) of ACI 318-71.

The reinforcement for the steam tunnel area of the shield building is shown in <Figure 3.8-8> and <Figure 3.8-12>.

f.   Methods of Design for Shear Effects

Shear reinforcement is determined by the methods of Chapter 11 ACI 318-71.

3.8.1.4.6   Computer Programs

a.   Computer programs used in the structural analysis of the shield building:

The computer programs used in the analysis of safety class structures are often broad in scope and only the applicable portions of the particular program are discussed in each section.

1. McDonnell Douglas ICES-DYNAL:

   DYNAL was developed by the Computer Science Department of McDonnell Douglas Automation Company and is presently being operated under Release 3.2 dated February 2, 1973, updated to September 10, 1973. The structural dynamic analyses available in DYNAL are based on the modal superposition method using time history analysis. A simplified set of equations is formed in terms of "normal coordinates" and then solved. These "normal coordinates" are obtained by forming the stiffness and mass matrices of the structural system and solving for the normal modes and frequencies by the HOW method. The program capabilities for analysis using shock spectrum excitation or response spectrum are not utilized. Output obtained includes structural response in terms of displacement, velocity and accelerations at selected nodal points, maximum accelerations and floor response curves.

   DYNAL is available in the public domain and has been widely used since its commercial release in 1970. The program is written in the same language as ICES-STRUDL and is currently being run on the McDonnell Douglas, St. Louis, MO, IBM 360/195, dual processor computer system under operating system OS/MVT/ASP Release 21.6.

2. ELAD:

   ELAD is a computer program designed to determine elastic deformations, stresses, strains, and principal values of stress within axisymmetric solid structures of arbitrary shape subjected to axisymmetric or non-axisymmetric pressure, concentrated loads and temperatures. All boundary conditions consistent with theory of elasticity are permitted. A linear, orthotropic, elastic stress-strain relationship is assumed

throughout the model. ELAD was developed by the Service
Bureau Corporation, Inglewood, California, under contract to
the Air Force Weapons Laboratory, Kirtland AFB, New Mexico,
and is published as Technical Report No. AFWL-TR-69-70 dated
October 1969. It is available from the U.S. Government
Document Clearing House.

This program is widely available in the public domain and is
being run on the McDonnell Douglas, St. Louis, MO, IBM 360/195
dual processor computer system under OS/MVT/ASP Release 216,
and also on the Gilbert Associates Inc., Reading, PA,
IBM 370/155 under O/S 21.7 MFT with HASP 3.1.

3.   ICES-STRUDL-II:

ICES-STRUDL-II is a widely used, well known, analytical
program developed by Massachusetts Institute of Technology and
released to the public domain in November 1968. This program
has a wide range of usage for static and dynamic analysis of
frame members and reinforced concrete structures. STRUDL
includes the capability for linear and non-linear, static and
dynamic analysis.

The program is run on Gilbert Associates, Inc., Reading, PA,
IBM 370/155 computer and also McDonnell Douglas, St. Louis,
MO, IBM 360/195 dual processor computer system.

4.   ASHSD 2:

See <Section 3.8.2.4.5.a.3> for a description of this program.

5. ANSYS:

    ANSYS is a proprietary engineering analysis computer program
    developed by Swanson Analysis Systems, Incorporated, for
    applications in the solution of large-scale engineering
    problems. The analysis capabilities of the program include
    static and dynamic; elastic, plastic, creep, and swelling;
    buckling, small and large deflections; steady-state and
    transient heat transfer and fluid flow.

b. Design Control and Verification of Computer Programs:

    One of the following control procedures is used to comply with the
    requirements of <10 CFR 50, Appendix B> and to demonstrate the
    applicability and validity of all computer programs used for the
    structural design of Safety Class structures. Either:

    1. The computer program is a recognized code in the public domain
        and has sufficient history of use to justify its applicability
        and validity, or

    2. By comparison of the results with one or more of the
        following:

        (a) Physical (test) data;

        (b) Another previously verified program, preferably in the
             public domain, having similar computational objectives or
             input;

        (c) Accepted results published in the literature which deal
             with the particular subject matter of the code, such as
             recognized journals, periodicals, reports, industry
             standards, or classical textbook solutions;

(d) Hand calculations.

Test problems used must be demonstrated to be similar to or within the range of applicability of the problems analyzed by the program to justify acceptance of the program.

c. Verification of Program Usage:

1. ICES-DYNAL:

This program has been widely used in the public domain for the past three years. For the analysis of safety class structures, the results obtained, using the particular revision and computer hardware/software combination, have been verified by comparison of results with those of similar programs and test data available. Documentation of the program is traceable to the McDonnell Douglas Automation Company, St. Louis, Missouri.

2. ELAD:

This program has been widely used in the public domain for the past three years. For the analysis of safety class structures, the results obtained, using the particular revision and computer hardware/software combination, have been verified by comparison of results with those of similar programs and test data available. Results for this program have also been cross-checked for the two computer centers to ensure similar results for each problem. Documentation of this program is traceable to the McDonnell Douglas Automation Company, St. Louis, Missouri.

3.   ICES-STRUDL-II:

     This program has been widely used in the public domain for the
     past ten years for the analysis of frame and reinforced
     concrete structures. The results of the structural analyses
     utilizing this program and computer hardware/software
     combination have been verified by test problems and comparison
     of results from other programs. Documentation of this program
     is traceable to the McDonnell Douglas Automation Company,
     St. Louis, Missouri, and the Massachusetts Institute of
     Technology, Cambridge, Massachusetts.

4.   ASHSD 2:

     See <Section 3.8.2.4.5.b.3>.

5.   ANSYS:

     See <Section 3.8.2.4.5.a.9>.

3.8.1.5   Structural Acceptance Criteria

The structural acceptance criteria for the shield building is based upon
accepted codes and standards, with modification to suit the operational
and functional criteria of these structures.

3.8.1.5.1   Stress and Strain Criteria

a.   The stress criteria will be based on strength design concept of
     ACI 318. The factored loads in the load combinations given and
     discussed in <Section 3.8.1.3> are based on either ACI 318 or
     ACI 359. The stress criteria and load combinations are modified by
     the use of the capacity reduction factor (ϕ) given in Section 9.2
     of ACI 318. Thus, stresses and strains will be within the limit

assumed by these codes. Any displacements will be elastic such that gross permanent deformations will not occur.

b.   The response of the structures to seismic loads is developed from the analytical techniques described in <Section 3.8.1.4>. The stresses due to static and dynamic loads are algebraically summed for the loading combinations set out in <Section 3.8.1.3>, to obtain the critical stress for each part. The limitation for tangential shear stress, as required by ASME Section III, Division 2, code paragraph CC 3520, is not applicable, since the concern in limiting the tangential shear stress is to limit the strain imparted to a containment liner. There are no comparable conditions in the shield building.

3.8.1.6   Material Specifications, Quality Control and Special Construction Techniques

The design of the shield building is based upon material specifications giving acceptable limitations of physical and chemical properties for the structural materials used, and upon imposing quality control provisions to ensure that the materials meet the specifications.

The organization, responsibilities and general provisions for the Quality Assurance Program are described in <Chapter 17>. The quality control provisions that have been imposed for the structures are described herein. Wherever engineer approved design documents call for deviations or exceptions from the accepted codes and standards within this section, the approved design documents shall govern.

3.8.1.6.1   Concrete Construction

3.8.1.6.1.1   Codes and Standards

The following codes and standards were used to establish requirements for concrete construction:

ACI 301-72   "Specification for Structural Concrete for Buildings."

ACI 306-66   "Recommended Practice for Cold Weather Concreting."

ACI 318-71   "Building Code Requirements for Reinforced Concrete."

ACI 347-68   "Recommended Practice for Concrete Formwork."

ACI 305-72   "Recommended Practice for Hot Weather Concreting."

ACI 211.1-70   "Recommended Practice for Selecting Proportions for Normal Weight Concrete."

ACI 304-73   "Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete."

ACI 214-65   "Recommended Practice for Evaluation of Compression Test Results of Field Concrete."

ACI 315-74   (Proposed Revision of) "Manual of Standard Practice for Detailing Reinforced Concrete Structures."

ACI 309-72   "Recommended Practices for Consolidation of Concrete."

ACI 308-71   "Proposed Recommended Practice for Curing Concrete."

3.8.1.6.1.2   Material Specifications

a.   Cement

The cement used in developing the design mixes conforms to ASTM
Specification C 150-73a for Type II Cement, modified to meet
optional requirements of C 150-73a for moderate heat of hydration,
alkali content and false set. Where the heat of hydration is not a
concern and the aggregate is not deleteriously reactive as
confirmed by Potential Reactivity Test (ASTM C289-71), Type I
cement modified to meet the optional requirements of C150-73a for
false set may be used in design mixes.

b.   Admixtures

1.   Air Entraining Admixture

Air entraining admixtures conform to the requirements of
ASTM C 260-73.

2.   Water Reducing Densifier

A water reducing densifier is added to the concrete and
conforms to ASTM C 494-71.

3.   Calcium Chloride, and admixtures containing Chlorides,
Sulphides, Nitrates, or Boron will not be used.

c.   Aggregates

The aggregate used for normal weight concrete complies with
ASTM C 33-71a, except as modified by <Section 3.8.1.6.1.5.b.2>.
The aggregate gradation for porous concrete differs from
ASTM C 33-71a, and is selected to maximize the permeability and

meet the strength requirements of porous concrete. The type and size of aggregate, slump and additives are established to minimize shrinkage and creep.

d. Water

The water for concrete is tested to show compliance with the requirements listed in this section before being used in the concrete.

1. Water quality complies with the following:

    (a) Chlorides as Cl   250 ppm maximum

    (b) Solids   1,000 ppm maximum

    (c) Sulphate ion   250 ppm maximum

    (d) pH   between 6.0 and 8.0

2. A comparison of the proposed mixing water properties is made with distilled water, using the following tests made with the proposed portland cement:

	Test	Method	Limits of Comparison
(a)	Soundness (Autoclave Expansion)	ASTM C151-71	+0.10 percent difference in gauge length
(b)	Vicat Time of Setting	ASTM C191-71	Initial Set ±10 minutes Final Set ±1 hour
(c)	Compressive Strength	ASTM C109-73	10 percent reduction of strength

3.8.1.6.1.3   Design Mixes

a. Trial design mixes for normal weight concrete are proportioned in
    accordance with Method 1 or Method 2 of Section 3.8.2 of ACI 301-72
    and tested in accordance with the following ASTM standards to
    ascertain conformance to concrete specifications.

Test	ASTM
Method of Test for Compressive Strength of Cylindrical Concrete Specimens	C 39-72
Method of Test for Slump of Portland Cement Concrete	C 143-71
Method of Making and Curing Concrete Test Specimens in the Laboratory	C 192-69
Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method	C 231-72T

b. Mix designs for heavy weight concrete are proportioned by
    determining the amount of water to be added to a premixed,
    prepackaged powder, Chemtree 1-20-26, manufactured by Chemtree
    Corporation. The amount of water added is governed by meeting the
    3,000 psi at 90 days compressive strength requirement and the
    195 pcf minimum requirement. The following tests are used to
    determine the required properties:

Test	ASTM
Method of Test for Compressive Strength of Cylindrical Concrete Specimen	C 39-72
Method of Test for Unit Weight, Yield and Air Content (Gravimetric) of Concrete	C 138-73

c. A mix design for porous concrete was developed by a statistical analysis of permeability and strength test data which demonstrated the following:

1. The average compressive strength is 1,000 psi or greater at 28 days and no strength test result fell below 800 psi.

2. No more than 10 percent of all specimens tested had a strength less than 1,000 psi.

3. The average coefficient of permeability was 3 feet per minute or greater at 28 days and no permeability test fell below a value of 2 feet per minute.

4. No more than 20 percent of all specimens tested had a coefficient of permeability less than 3 feet per minute.

The following tests were used to determine the required properties (from test placements):

Test	ASTM
Method of obtaining and testing drilled core and sawed beams of concrete	C 42-68
Method of Test for Permeability of Granular Soils, except for appropriate modifications to permit testing a porous concrete specimen instead of a granular soil	D 2434-68
Method of Test for Compressive Strength of Cylindrical Concrete Specimens	C 39-72

d. Mix designs for grout for baseplates are proportioned by determining the amount of water to be added to a premixed, prepackaged grout, Masterflow 713 or Embeco 636, manufactured by Master Builders Company, or newer products including Masterflow 713

Plus or Embeco 636 Plus, subsequent to July 2001. The amount of water added is governed by meeting the 5,000 psi cube strength at 28 days requirement. Dimensional stability requirements of no contraction and maximum 0.20 percent expansion are met at a water content which results in a fluid grout. This water content is not exceeded in the design mix. The following tests are used to determine the required properties:

Test	Standard
Method of Test for Compressive Strength of Hydraulic Cement Mortars	ASTM C 109-73 (modified for premixed grouts)
Method of Test for Flow of Grout Mixtures	CRD-C79-58
Methods of Sampling and Testing Expansive Grouts	CRD-C589-70

NOTE: Updated standards determined to be Technically Equivalent to the standards listed are also used.

e. Mix designs for nonshrink tunnel grout are proportioned by determining the amount of water to be added to a premixed, prepackaged grout, In-Pakt, manufactured by Intrusion-Prepakt Company. The amount of water added is governed by meeting the 4,000 psi cube strength at 28 days requirement. Dimensional stability requirements of no contraction and maximum 0.40 percent expansion are met at a water content which results in a fluid grout. This water content is not exceeded in the design mix. The tests used to determine the required properties are identical to those given above in subitem (d).

f. Mix designs for contact grout are proportioned by mixing water, sand and cement to a consistency suitable for field placement which meets at least 100 psi cube strength at one day. The maximum water/cement ratio is 0.70. The compressive strength is determined by ASTM C 109-73.

g.   Mix designs for grout for leveling porous concrete are proportioned
     by mixing water, sand and cement to a stiff consistency that
     develops 1,000 psi cube strength at 28 days. Stiff is defined as
     the consistency where the grout can be placed on the porous
     concrete such that minimal penetration of the grout occurs into the
     voids of the porous concrete. The compressive strength is
     determined by ASTM C 109-73.

3.8.1.6.1.4   Production Concrete

The production mixes are those mixes developed from the design mix
testing program and proportioned in accordance with the results of the
trial design mix using material qualified and accepted for the work.
During construction, minor modification in the mixes may be necessitated
by variation in aggregate gradation and moisture content. The
production concrete is mixed and transported in accordance with
ASTM C 94-73a. Placing, consolidating and curing of concrete are in
accordance with ACI 301-72, except that a coat of cement grout at
construction joints as required by Section 8.5.3 of ACI 301 is not
required. Cold weather curing of concrete is in accordance with
ACI 306. Placing, consolidating and curing of premixed, prepackaged
materials are in accordance with manufacturer's recommendations.

3.8.1.6.1.5   Concrete Quality Control

a.   General Provisions

     The services of a testing laboratory were obtained prior to
     commencing concrete work. The testing laboratory makes design
     mixes described in <Section 3.8.1.6.1.3>, using the proposed
     materials to consistencies satisfactory for the work, in order to
     determine suitable mix proportions necessary to produce concrete
     and grout conforming to the type and strength requirements
     specified. During concrete operations, inspectors at the batch

plant certify the mix proportions of each batch delivered to the site and periodically sample and test the concrete ingredients.

These inspectors ensure that a ticket is provided for each batch, documenting the time loaded, actual proportions of the mix, amount of concrete, and concrete design strength. The cleanliness of trucks, and the handling and storage or aggregate are checked by the batch plant inspectors. The water and ice additions at the batch plant are modified, if necessary, as required by measurements of the moisture content of the aggregates and gradation changes.

Inspectors at the construction site inspect reinforcing and form placement, make slump tests, make test cylinders, check air content, check placement temperature, and record weather conditions.

Requirements for placing and consolidating concrete are detailed in ACI 301.

b. Concrete Materials

1. Cement

All cement is sampled and tested at the mill for conformance to ASTM C 150-73a and certified mill test reports are submitted to the owner at that time. All test data are subject to verification by in-process tests by the testing laboratory provided by the owner. All cement is sampled and tested in accordance with ASTM C 150. The chemical and physical tests are conducted on every test sample. The cement is not released for use on the project until it is verified that it conforms to ASTM C 150 and the project concrete specifications.

2.   Aggregates

The fine aggregate conforms to ASTM C 33-71a with modification
to suit local conditions. Samples of the proposed fine
aggregate are tested to ensure compliance with the
specification. The coarse aggregate conforms to ASTM C 33.
Samples of the proposed coarse aggregate are tested to ensure
compliance with the specification.

The amount of fines passing the No. 200 sieve (known as dust
of fracture) may exceed ASTM C 33-71a requirements for coarse
aggregates, provided the total amount of fines passing the
No. 200 sieve for both fine and coarse aggregate shall not
exceed the weighted maximum amounts of fines permitted for
fine plus coarse aggregates. In no case shall the amount of
fines passing the No. 200 sieve exceed 2.3 percent.

The following tests related to coarse aggregate are not
required for the reasons given:

(a)   ASTM C 87, "Test for Effect of Organic Impurities in Fine
      Aggregate on Strength of Mortar:" Tests for effects of
      impurities should not be required unless such impurities
      are shown by ASTM C 40 to be present.

(b)   ASTM C 78, "Test for Flexural Strength of Concrete Using
      Simple Beam with Third Point Loading:" Flexural strength
      is not specified.

(c)   ASTM C 29, "Test for Unit Weight of Concrete,:" This
      test is intended to be used where slag is used in the
      concrete.

(d)   ASTM C 666, "Test for Resistance to Freezing and
      Thawing:" Historical records indicate no tendency for
      aggregate to be subject to freeze-thaw deterioration.
      Also, presence of external air will counteract the very
      small potential for any freeze-thaw action.

The type and frequency, during concrete operations, of ASTM
standard tests are:

(a)   "Organic Impurities in Sands for Concrete,"
      ASTM C 40-73-daily.

(b)   "Materials Finer Than No. 200 Sieve in Mineral Aggregates
      by Washing," ASTM C 117-69-daily.

(c)   "Sieve or Screen Analysis of Fine and Coarse Aggregates,"
      ASTM C 136-71-daily.

(d)   "Friable Particles in Aggregates," ASTM C 142-71-monthly
      and at change of source.

(e)   "Lightweight Pieces in Aggregate," ASTM C 123-69-monthly
      and at change of source.

(f)   "Total Moisture Content of Aggregate by Drying,"
      ASTM C 566-67, twice daily, and as required for control
      of production.

(g)   "Scratch Hardness of Coarse Aggregate Particles,"
      ASTM C 235-68, monthly and at change of source.

(h)   "Resistance to Abrasion of Large Size Coarse Aggregate by
      Use of the Los Angeles Machine," ASTM C 535-69, every
      6 months and at change of source.

(i) "Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine," ASTM C 131-69, every 6 months and at change of source.

(j) "Potential Reactivity of Aggregates," ASTM C 289-71, every 6 months and at change of source.

(k) "Soundness of Aggregates by Use of Sodium Sulphate or Magnesium Sulphate," ASTM C 88-73, every 6 months and at change of source.

The frequency of these tests is changed if and when the source of the aggregate changes, or if field conditions require more frequent testing, or as required by the Engineer. Both fine and coarse aggregates are tested for moisture content twice daily as a minimum or as required by climatic conditions. All aggregate test samples are taken from the batch plant stock pile.

3. Water

The water used in the production of trial mixes and production concrete is as described in <Section 3.8.1.6.1.2.d>. The water is tested for compliance with these requirements for use in the trial mixes and for the production concrete.

4. Admixtures

The air entraining admixture used conforms to and is tested in accordance with ASTM C 260-73. The supplier submits to the owner documentation to demonstrate that the admixture supplied is essentially identical in concentration, composition and performance to the admixture originally tested to meet ASTM C 260-73 prior to or with each delivery of admixture.

The concrete air content is to be taken for every 100 cubic
yards of concrete during the concrete operations.

5. Water Reducing Densifier

    The water reducing densifier admixture used conforms to and is
    tested in accordance with ASTM C 494-71, Type "A" or Type "D".
    The quantity added, the control temperatures and method of
    mixing, conform to the manufacturer's recommendations for use
    of their product. The supplier submits to the owner
    documentation to demonstrate that the admixture supplied is
    essentially identical in concentration, composition and
    performance to the admixture originally tested to meet
    ASTM C 494, Type "A" or Type "D," prior to or with each
    delivery of admixture.

c. Test Evaluations

    The evaluation of the test results for the 3,000 psi concrete is in
    accordance with Chapter 17, of ACI 301-72.

    1. Preliminary

        (a) Concrete design mixes and the associated tests are
             performed by the testing laboratory provided by the
             owner. The proportions for the concrete mixes are
             determined by Method 1 of Section 3.8 of ACI 301-72. All
             concrete materials are tested as previously described.
             Compression test methods conform to ASTM C 39-71 and
             making and curing concrete test specimens conform to
             ASTM C 192-69.

        (b) Grouts have compression strength tests made by the
             testing laboratory in advance of grouting operations, to

verify the mix formula and batch procedure. Specimens are made and tested for strength in accordance with standard test for "Compressive Strength of Hydraulic Cement Mortars," ASTM C 109-73 (modified if necessary for premixed fluid grouts) and for dimensional stability, if required, in accordance with Corps of Engineers specification, "Method of Sampling and Testing Expansive Grouts," CRD-C-589, or more current standards that provide similar results.

2.   Field Tests

During concrete operations, an inspector at the batch plant certifies the mixed proportions of each batch delivered to the job site, and samples and tests all concrete ingredients. Inspectors at the job site inspect reinforcing, embedments and form placements, make slump tests, make test cylinders, check air content, check concrete temperature, record weather conditions, and inspect concrete placing and curing.

Concrete test cylinders and slump tests are molded, cured, capped, and tested in accordance with ACI 301-72, except that samples of concrete for cylinders may be taken from a single location in the middle 70 percent of the load as an option to samples taken from several points in the load as required by ACI 301, Section 16.3.4.1.

The following tests are also performed:

(a) Sampling and testing the concrete materials as previously described.

(b) Making slump tests with the following frequencies:

(1)   When concrete has been mixed in a central mix plant
      or in transit mix trucks, a slump test is made on
      the first two trucks and thereafter for each
      100 cubic yards of concrete.

(2)   Slump tests are made on each concrete batch used for
      test cylinders.

(3)   Slump tests are made at any time the inspector has
      reason to suspect that the concrete slumps are not
      within the allowable tolerances.

(4)   The concrete air entrainment content is taken at
      100 cubic yard intervals.

(5)   The concrete unit weight is determined daily during
      production.

(6)   The batch plant scales are calibrated to the
      ASTM C 94-739 standard at maximum intervals of
      90 days and certified.

(7)   Temperature of concrete at time of placement is
      recorded at 100 cubic yard intervals.

Heavyweight concrete is sampled from each batch and tested to
verify compressive strength and density.

Porous concrete is sampled at every 100 cubic yards placed, or
fraction thereof, and is placed in a test slab. From the test
slab, samples are cored and tested for compressive strength
and permeability.

Grout for baseplates and nonshrink tunnel grout are sampled from each 2,000 pounds of dry grout material, or portion thereof, placed in one day, and laboratory-mixed and tested for compressive strength and dimensional stability.

Contact grout and grout for leveling porous concrete are sampled at 4 test cubes per each 10 cubic yards, or fraction thereof, placed in one day. Samples are tested for compressive strength.

3.8.1.6.1.6   Tolerances

The finished concrete tolerances for the shield building will be:

a.   Cylindrical Wall

   1.   Variation from plumb: not more than ±4 inches for the total structure height, taken at the vertical axis of the cylindrical wall, or more than ±1 inch in any 20 feet of wall height.

   2.   Variation from true circular section: not more than ±3 inches in radius from the vertical axis of the cylindrical wall.

   3.   Variation of wall thickness: not more than -1/4 inch or more than +1 inch.

b.   Dome

   1.   Variation from true spherical section: not more than ±3 inches in radius.

   2.   Variation in dome thickness: not more than +1 inch or more than -1/2 inch.

3.8.1.6.1.7   Special Construction Techniques

The cylindrical portion of the Shield Building was constructed using
conventional forming techniques. The dome was constructed by first
forming a structural concrete layer of 9 inch thickness, supported off
the dome of the Containment Vessel. After curing, this layer will then
support the concrete for the remaining thickness of the dome.

3.8.1.6.2   Reinforcing Steel

3.8.1.6.2.1   Codes and Standards

The following codes and standards are used to establish the approved
design documents governing reinforcing steel:

ASTM A 615-72   Standard Specification for Deformed Billet Steel Bars for
                Concrete Reinforcement

ASTM A 370-72   Standard Methods and Definitions for Mechanical Testing
                of Steel Products

NRC <Regulatory Guide 1.15>   Testing of Reinforcing Bars for Concrete
                              Structures

3.8.1.6.2.2   Materials

The reinforcing steel is deformed bar conforming to ASTM A 615,
Grade 60.

3.8.1.6.2.3   Quality Control

a.   Reinforcing Steel

   1.   General

        The specifications for reinforcing steel included the
        following quality control measures.

        (a) The certified mill test reports are provided for each
             heat of steel covering chemical composition and
             mechanical properties. Bars are branded in the deforming
             process to carry identification as to manufacturer, size,
             type, and yield strength.

        (b) In accordance with NRC <Regulatory Guide 1.15>, tests are
             performed on reinforcing steel by a Testing Laboratory to
             confirm compliance with physical requirements and
             verification of mill test results. The frequency of
             testing is at least one full diameter specimen from each
             bar size for each 50 tons or fraction thereof, of
             reinforcing bars produced from each heat. The tests are
             made in accordance with ASTM A 615-72 and ASTM A 370-72,
             with the additional requirement that only full size bars
             be tested.

        (c) In the event that a tested specimen does not show
             conformance with ASTM A 615-72, Grade 60, a single retest
             is permitted if any one of the following special
             conditions apply to that tested specimen:

             (1) As described by ASTM A 615-72, bars that develop
                  flaws during tensile or bending tests are discarded
                  and another specimen of the same size is randomly
                  selected from the same lot.

(2) If any tensile test specimen fail to develop the
     required ASTM strength and also any part of the
     fracture is outside the middle third of the gauge
     length, as indicated by scribe scratches marked on
     the specimen before testing, a retest is allowed as
     permitted by ASTM A 615-72.

(3) A single retest is permitted for any tension test
     specimen that fail within the middle third of the
     gauge length, but does not meet the minimum strength
     and elongation requirements of ASTM A 615-72,
     Grade 60, provided that all of the additional
     conditions are met:

     i.   The tensile strength of the specimen is not
          less than 88,000 psi.

     ii. The yield strength of the specimen is not less
          than 59,000 psi.

     iii. The elongation of the specimen is not less
          than:

          • 7 percent for bar No. 3, 4, 5, 6

          • 6 percent for bar No. 7, 8

          • 5 percent for bar No. 9 through 18

2. Non-complying Reinforcing Steel

    Any materials not complying with the ASTM specified strengths
    are not accepted and the material is not used for fabrication
    of concrete reinforcement.

3. Deformations

    The reinforcing bar deformations are inspected and checked to
    assure their compliance with ASTM A 615.

4. Traceability

    Traceability of the reinforcing steel, with regard to mill
    heat number is provided until fabrication. Only reinforcing
    steel from heats that are successfully tested as described in
    <Section 3.8.1.6.2.3.a> are used for fabrication.

3.8.1.6.2.4   Tolerances

The fabrication tolerances for the reinforcement are in accordance with
the requirements of ACI 318. Placing tolerances are based on the
1975 Supplement to ACI 318 except as noted below that a 1 inch tolerance
is permitted on the position of a bar in a slab or wall 26 inch or
thicker. The tolerance on minimum cover is identical to that of the
1975 Supplement. The 1 inch tolerance for 26 inch thick members is
justified since it is a smaller percentage of the total thickness than
the 1/2 inch tolerance for 10 inch thick members. The tolerances are
summarized as follows:

a. For locations of bars within the depth or thickness of a member and
    for clear concrete cover in flexural members, walls and compression
    members:

Thickness of Member (inches)	Tolerance on Location (inches)	Reduction on Cover (inches)
Less than 10	±3/8	3/8
10 to less than 26	±1/2	1/2
26 or greater	±1	1/2

with the exception that the reduction on the cover for formed
soffits (underside of interior beams, lintels, floors, or other
structural members) shall not be larger than 1/4 inch. The depth
of cover in any case shall not be less than two-thirds of the
specified cover.

b.   For longitudinal location of bends and for ends of bars:
      ±2 inches, except at ends of members where tolerances shall be
      ±1/2 inch.

c.   For spacing of bars: ±1 inch on the specified spacing, except that
      at embedments ±2 inches on the specified spacing is acceptable.

3.8.1.6.2.5   Special Construction Techniques

The reinforcement for the cylindrical wall was preassembled in partial
mats on the ground and lifted into place as a unit.

3.8.1.6.3   Cadweld Splices

3.8.1.6.3.1   General Requirements

a.   Cadweld splices will be used on bar sizes exceeding No. 11, on
      structural steel anchorages and where there is insufficient length
      for lap splices.

b.   Cadweld splices will be made with T-series sleeves and B-series
      sleeves as described in Erico Products Catalogue (Reference 5).
      These splice sleeves are capable of developing the minimum ultimate
      tensile strength of the reinforcing steel.

c.   A manufacturer's representative, experienced in Cadweld splicing
      will be present at the outset of splicing to demonstrate the
      equipment and techniques for making quality splices.

d.   A quality control program based on the 1977 ASME Code Section III, Division 2, Subsection CC-4333, will be used for inspection and testing of Cadweld splices.

e.   Sister splices can be used for testing in combination with the required number of production splices. Sister splices are removable splices made in place next to production splices and under the same conditions. The position and orientation of a sister splice is similar to the production splice that it represents.

3.8.1.6.3.2   Quality Control

a.   Qualification of Operators

Prior to the production splicing of reinforcing bars, each member of the splicing crew (or each crew if the members work as a crew) prepares two qualification splices on the largest bar size for each of the splice positions (e.g., horizontal, vertical, diagonal) to be used. The qualification splices are made using the same materials (e.g., bar, sleeve, powder) as those to be used in the structure. The qualification splices shall meet the criteria set forth in Erico Products publication (Reference 6).

b.   Fabrication Procedure

The following quality control procedures are followed to ensure acceptable splices:

1.   The splice sleeve, powder and molds are stored in a clean dry area with adequate protection from the elements to prevent absorption of moisture.

2. Each splice sleeve is visually examined immediately prior to use to ensure the absence of foreign material on the inside surface. Light or spotty rust on the inside surface of the sleeve is acceptable.

3. The molds are preheated to drive off moisture at the beginning of each shift when the molds are cold or when a new mold is used.

4. Bar ends to be spliced are power wire brushed for a distance of two inches plus 1/2 the sleeve length from end of bar to remove mill scale, rust, concrete, and other foreign material. Prior to brushing, water, grease and paint are removed by heating the bar ends with a torch.

5. A permanent line is marked on each bar for a reference point to confirm that the bar ends are properly centered in the splice sleeve.

6. Before the splice sleeve is placed into final position, the bar ends are examined to ensure that the surface is free from moisture. If moisture is present, bar ends are heated until dry.

7. Special attention is given to maintaining the alignment of sleeve and guide tube to ensure a proper fill.

8. The splice sleeve is flame dried after all materials and equipment are in position.

9. All completed splices are visually inspected at both ends of the splice sleeve and at the tap hole in the center of the splice in accordance with the requirements of Erico Products "Inspection of Cadweld Rebar Splice" (Reference 6). Splices

which do not meet this criteria are rejected and will not be used as tensile test samples.

10. Records are kept of all Cadweld splices placed. These records will enable a test splice to be chosen at random and to establish its location. The records plus test results of all Cadweld splices tested will be kept up-to-date and will be available to all inspectors. These records shall show:

    (a) The number assigned to each splice

    (b) The date that the splice was formed

    (c) The crew which carried out the splicing

    (d) The size, location and orientation of the splice

    (e) The record of visual inspection

    (f) The results of tensile tests performed

c. Testing Frequency

Separate test cycles shall be established for splices in horizontal, vertical and diagonal bars. The minimum number of splices to be tested is as follows:

1. If only production splices are used in a test cycle, the sample frequency shall be:

    (a) One of the first ten splices

    (b) One of the next 90 splices

(c) Two of the next and subsequent units of 100 splices

2.   If production and sister splices are tested, the sample
     frequency shall be:

(a) One production splice of the first ten production
     splices.

(b) One production and three sister splices for the next
     90 production splices.

(c) One splice, either a production or sister splice for the
     next and subsequent units of 33 splices. At least
     one-fourth of the total number of splices tested shall be
     production splices.

(d) Straight sister splices shall be substituted for
     production splice test samples on radius bent bars, on
     splices connecting reinforcing steel to structural steel
     and where sufficient length of protruding bar is not
     available for the removal of a production splice and
     consequential resplicing.

d.   Acceptance Criteria

1.   Sound, nonporous filler material is visible at both ends of
     the splice sleeve and at the tap hole in the center of the
     sleeve. Filler material is usually recessed 1/4 in. from the
     end of the sleeve due to the packing material, and is not
     considered a poor fill.

2.   Splices which contain slag or porous metal in the riser, tap
     hole or at the ends of the sleeve are rejected. A single

shrinkage bubble present below the riser is not detrimental
and is distinguished from general porosity as described above.

3. Evidence of filler material between the sleeve and bar for the
    full 360 degrees; however, the splice sleeves do not need to
    be exactly concentric or axially aligned with the bars.

4. The tensile strength of each individual test splice shall be
    equal to or greater than 125 percent of the minimum yield
    strength specified in ASTM A 615 for the grade of
    reinforcement being used.

5. The average tensile strength of a group of consecutive samples
    in a test cycle shall equal or exceed the guaranteed ultimate
    strength for that grade of reinforcement as specified in
    ASTM A 615. Where the number of test samples in a cycle
    exceeds 15, a running average shall be calculated based on the
    last 15 samples tested.

e. Splice Nonconformance Criteria

1. If a production or sister splice sample fails to meet the
    acceptance criteria and failure occurred in the bar, another
    sample will be chosen for testing. The cause of failure in
    the bar will be investigated. Any necessary corrective action
    affecting splice samples will be implemented prior to
    continuing the testing.

2. If any production splice used for testing fails to meet the
    acceptance criteria and failure did not occur in the bar, the
    production splice made just preceding and the production
    splice made just following the nonconforming splice will be
    tested. If any sister splice used for testing fails to meet
    the acceptance criteria and failures does not occur in the

bar, two additional sister splices will be tested. If either
of these retests fails to meet the acceptance criteria,
splicing by the crew performing the work represented by the
failed splice will be halted. Splicing will not be resumed
until the cause of the failures has been determined and
corrected.

3. If the running average tensile strength of the samples in a
test cycle fails to meet the acceptance criteria, splicing
will be halted. Splicing will not be resumed until the cause
of the nonconforming strengths has been determined and
corrected.

3.8.1.6.3.3   Tolerances

The fabrication tolerances for Cadweld splices will be as stated in
<Section 3.8.1.6.3.2.b>.

3.8.1.6.3.4   Special Construction Techniques

There are no special construction techniques required to fabricate
Cadweld splices.

3.8.1.7   Testing and Inservice Inspection Requirements

There are no planned systematic testing or inservice surveillance
programs for the shield building.

3.8.1.8   Annulus Concrete

3.8.1.8.1   Description of Annulus Concrete

The annulus concrete extends from the top of the foundation mat at
Elevation 574'-10" to Elevation 598'-4" and has a radial thickness of

4 feet 10-1/2 inches. This annulus concrete provides stiffness to the steel containment vessel to reduce the dynamic response of the steel containment vessel due to the postulated SRVD loading phenomena. This function is required to qualify equipment and piping attached to the containment vessel. The containment vessel is also structurally strengthened by the annulus concrete. <Appendix 3A> and <Appendix 3B> contain a detailed discussion of the SRVD loading which precipitated the addition of the annulus concrete.

Details of mechanical and electrical penetrations are discussed in <Section 3.8.2> and shown in <Figure 3.8-6> and <Figure 3.8-7>. The design of penetration sleeves in the containment vessel and shield building are such that they allow differential movement between the shield building, annulus concrete and the containment vessel. At least a 2 inch thickness of compressible material has been provided around the six penetrations within the annulus concrete to permit differential movement between the containment vessel and shield building in the annulus concrete. Typical details are shown in <Figure 3.8-110>.

In addition to providing compressible material around the embedded penetrations, a one inch thickness of compressible material is provided on the top surface of the bottom four ring stiffeners of the containment vessel in order to reduce thermal stresses caused by the vertical growth of the containment vessel.

Three inches of compressible material are also provided between the containment vessel and the annulus concrete below the lowest Stiffener No. 1 and above the Stiffener No. 4 to reduce thermal compressive stresses. The concrete will have a minimum 28 day cylinder compressive strength of 3,000 psi. The steel reinforcement is in accordance with the requirements of ASTM A615-72, Grade 60.

A general reinforcing pattern of orthogonal bars arranged vertically and circumferentially in both faces of the wall was used in the annulus

concrete. The vertical reinforcement is No. 18 at 15 inches each face.
This reinforcement is not embedded in the mat but rather has a cadweld
at each end of the bar to provide a mechanical anchorage which will
develop the bars more quickly. The horizontal reinforcement is No. 18
at 12 inches for each face except for approximately the top eight feet
at the outside face where No. 18 at 6 inches is provided.

Transverse (radial) shear reinforcement is provided by No. 7 tie bars
spaced circumferentially at each vertical bar in the bottom and every
other bar in the top section. The vertical distribution has four ties
below ring Stiffener No. 1, between ring Stiffeners No. 1 and No. 2, and
between ring Stiffeners No. 2 and No. 3. There are three ties between
ring Stiffeners No. 3 and No. 4 and above ring Stiffener No. 4. The
general reinforcement pattern is shown in <Figure 3.8-110>. Extra steel
is provided in the area of the attachment plate stiffeners and around
penetrations.

3.8.1.8.2   Applicable Codes, Standards and Specifications to the
                 Design

3.8.1.8.2.1   Codes

a.   ASME - Boiler and Pressure Vessel Code - Section III, Division 2,
     Subsection CC, 1980 Edition with Summer 1981 Addenda and Code
     Case N-258 - "Design of Interaction Zones for Concrete Containments
     Section III, Division 2," March 1980, with proposed Revision 1.

b.   ACI 318-71 "Building Code Requirements for Reinforced Concrete"
     American Concrete Institute, Chapters 1, 2 and 7.

3.8.1.8.2.2   Standards

a.   ASTM Standards: Applicable ASTM standards are cited in the ASME
     Code. The date of a particular standard may vary for different

items because of the difficulty in purchasing material to an
outdated standard. Since the latest ASTM standards reflect
industry practice used for fabrication and erection, it was
permitted to use an updated standard where no unacceptable loss of
quality would result.

b.   Applicable Regulatory Guides to Design

      Regulatory guides pertaining to seismic design classification and
      seismic design are referenced in <Section 3.2> and <Section 3.7>,
      respectively.

      1.   <Regulatory Guide 1.10>, "Mechanical (Cadweld) Splices in
            Reinforcing Bars of Category I Concrete Structures." This
            guide was used with any modifications specified in
            <Section 3.8.1.8.6.3>.

      2.   <Regulatory Guide 1.15>, "Testing of Reinforcing Bars for
            Category I Concrete Structures." This guide was used with any
            modifications specified in <Section 3.8.1.8.6.2>.

      3.   <Regulatory Guide 1.55>, "Concrete Placement in Category I
            Structures."

3.8.1.8.2.3   Principal Plant Specifications

The principal specifications for the annulus concrete are:

a.   Concrete supply.

b.   Fabrication and placing of reinforcing steel and embedded items.

c.   Placement of structure concrete.

d.   Design.

These specifications include the applicable design requirements of the
annulus concrete. The detailed material specifications, the detailed
quality control provisions and any special construction technique
requirements as described in <Section 3.8.1.8.6>, are included in these
specifications.

3.8.1.8.3   Loads and Load Combinations

3.8.1.8.3.1   Loads Used in the Design

The annulus concrete design considered all the loads from the shield
building <Section 3.8.1.3.1> and from the containment vessel
<Section 3.8.2.3.1>.

3.8.1.8.3.2   Load Combinations

a.   Combinations

     The load combinations listed in <Table 3.8-13> were investigated in
     determining the most critical conditions for design. Many of the
     loads vary within a range; therefore, the specific value used is
     that which produced the most critical conditions for the load
     combination being considered.

b.   Design Approach

     The load combinations discussed above are in accordance with the
     requirements of the ASME Code Section III, Division 2, for
     reinforced concrete and the criteria discussed in
     <Section 3.8.1.8.4>.

3.8.1.8.3.3   Load Factors

The load factors presented in <Table 3.8-13> are in accordance with the
ASME Code. In general, dynamic load factors apply only to loads in
local areas where dynamic analysis is impracticable.

3.8.1.8.3.4   Load Combinations for Local Areas

The load factors and load combinations for the design of local areas are
the same as those used for the general structure and are outlined in
<Table 3.8-13>. Methods of analyzing for local loads and for the
effects of discontinuities are discussed in <Section 3.8.1.8.4>. Loads;
defined by the symbols R_o, R_a, Y_r, Y_j, Y_m; and due to pipe reactions and
jet impingement forces are generally localized in nature.

3.8.1.8.3.5   Time Dependent Loads

The annulus concrete is not a prestressed concrete pressure vessel,
therefore variation in loads due to time dependent losses such as creep
and shrinkage are not significant.

3.8.1.8.3.6   Thermal Effects

Temperature profiles based on the temperatures given in
<Section 3.8.1.3.1.e> and <Section 3.8.1.3.1.i> are the basis for
selection of the critical temperature induced loads to be used in load
combinations presented in <Table 3.8-13>.

3.8.1.8.4   Design and Analysis Procedures

Since the annulus concrete is defined by ASME Code Case N-258 to be part
of the containment, it was designed in accordance with Article CC-3000
of ASME Section III, Division 2 (Reference 20). The annulus concrete is

required to limit stresses in the vessel and to reduce the vibratory response of the vessel due to the postulated SRV loading phenomena.

In the analysis, the basic assumption of the material being ideally elastic, homogeneous and isotropic is made. Tensile strength of the concrete is considered in the analysis to calculate the reduced shear capacity and to predict cracking, but not for bending strength.

The annulus concrete is analyzed using two computer programs - ASHSD2 and ANSYS. The ASHSD2 program is used to analyze the containment vessel, annulus concrete, the shield building for static loads, suppression pool dynamic loads, and seismic loads. The finite element model, which is used for these analyses, is shown in <Figure 3.8-105>. Since the ASHSD2 program does not have thermal load capability, a second finite element model is required to analyze the response to thermal loads. The ANSYS thermal analysis model is shown in <Figure 3.8-106>. See <Section 3.8.1.4.6> for a description of the computer programs.

3.8.1.8.4.1   Assumptions of Boundary Conditions

a.   Containment Vessel - Annulus Concrete Interface

     The interface between the containment vessel and the annulus
     concrete is represented in the ASHSD2 finite element model with
     common nodes for the axisymmetric solid elements and the
     axisymmetric shell elements. This representation is selected for
     the mechanical loads because these loads do not produce a tendency
     for significant slip at the interface, compared to the thermal
     loads. Some of these loads also are non-axisymmetric or dynamic
     and ASHSD2 does allow these types of loads. Because ASHSD2 did not
     have thermal load capability, an ANSYS model was developed for the
     thermal loads (Reference 20).

b.   Basemat Foundation – Annulus Concrete Interface

The basemat had been placed without considering the annulus filled
with concrete; therefore, there is no mechanical connection
(dowels) between the basemat and the annulus concrete. The
original ASHSD2 analysis for mechanical loads conservatively
modeled this condition with the base of the annulus concrete being
independent of the basemat with no restraint against either upward
or downward vertical movement. However, the shield building and
vessel were fixed at the basemat. This model required the vessel
and shield building to carry all the transverse shear forces. The
results of this analysis indicated that the shield building was
overstressed. The next logical step was to more realistically
model this interface area; therefore, the basemat stiffness was
added to the model removing the fixed conditions of the vessel and
shield building. The results of this analysis indicated that the
shield building was marginally within allowables for the shear
forces. Although the shear stresses were within allowables, the
decision was made to mechanically protect the shield building. To
achieve this, the basemat was prepared for the new concrete by
cutting a shear key to resist some of the radial shear being
transferred through the annulus concrete.

The analysis for the thermal loads with ANSYS incorporated a "gap"
element to create the effect of a compression with no tension
capability boundary between the basemat and annulus concrete. The
"gap" element accurately models the actual interface.

c.   Shield Building – Annulus Concrete Interface

The shield building – annulus concrete interface was modeled as a
monolithic section, i.e., no slip is assumed to occur along the
interface. An analysis was performed to demonstrate that the
stresses in the Containment Vessel are not significantly influenced

by this boundary condition. For the purpose of the analysis, the
vessel stresses produced by the long term LOCA load combination
were compared for the case of including the 3 ft shield building as
a monolithic part of the 5 ft annulus concrete and for the case
where the shield building is removed from the model. This analysis
confirmed that the boundary condition is conservative with respect
to the containment vessel and annulus concrete.

3.8.1.8.4.2   Treatment of Axisymmetric and Non-axisymmetric Loads

a.   Axisymmetric Loads

The annulus concrete was analyzed and designed for the following
axisymmetric loads which are defined in <Section 3.8.1.3> and
<Section 3.8.2.3>.

1.   Dead Load

2.   Live Load

3.   Groundwater Loads

4.   Normal Operating and Accident Temperature Induced Forces

The analytical approach used is described in <Section 3.8.1.8.4.3>.

b.   Non-axisymmetric Loads

The annulus concrete was analyzed and designed for the following
non-axisymmetric loads and are as defined in <Section 3.8.1.3> and
<Section 3.8.2.3>:

1.   Lateral Soil Pressures

2. Wind Loads

3. Tornado Wind Loads

4. Seismic Loads (OBE and SSE)

5. Local pipe reaction and pipe rupture forces including jet impingement forces and local compartmental pressure loads on the shell outside the shield building due to postulated pipe breaks within the auxiliary building steam tunnel compartment.

6. Vibratory and direct pressure effects as transmitted through the containment vessel and/or shield building due to Steam Relief Valve Operation.

7. Axisymmetric pool swell air bubble loads as defined in <Appendix 3B>.

3.8.1.8.4.3   Analytical Techniques

a. Static Analysis

A static analysis of the annulus concrete using a model including the containment vessel, shield building and annulus concrete was performed using the ASHSD2 and ANSYS computer programs. ASHSD2 is a finite element program with the capability to determine elastic deformations and stresses within axisymmetric shell or solid structures of arbitrary shape subjected to axisymmetric or non-axisymmetric pressure or concentrated loads. All boundary conditions consistent with the theory of elasticity are permitted. The axisymmetric finite element model used for the ASHSD2 analyses is shown in <Figure 3.8-105>. ANSYS is a general purpose finite element program with the capability to solve static and dynamic problems involving elasticity, plasticity, creep, and swelling,

buckling problems, and steady-state and transient heat transfer and fluid flow problems. The axisymmetric finite element model used for the ANSYS analyses is shown in <Figure 3.6-106>.

The ASHSD2 computer program was used to analyze the annulus concrete for the accident pressures, groundwater pressure, steam tunnel pressure, suppression pool water pressure, and seismic loads. The ANSYS computer program was used to analyze the annulus concrete for thermal loads.

b.   Dynamic Analysis (Seismic Analysis)

Safety/relief valve discharge, pool swell, condensation oscillation, and chugging loads were analyzed by use of the ASHSD2 computer program and the model shown in <Figure 3.8-105>. These loads are described in <Appendix 3A> and <Appendix 3B>. See <Section 3.7> for details of the seismic analysis.

The safety/relief valve discharge load used was greater than the loads presented in <Appendix 3A> and <Appendix 3B>. The condensation oscillation and chugging loads produced stresses which are small and could be neglected.

The structural responses are in terms of velocity, acceleration, and deflection. The accelerations are used to give equivalent static loads which are analyzed by the method described in <Section 3.8.1.4.3.a.1>.

c.   Effects on Analysis of Variations in Material Properties and Assumptions

For seismic design the effect of variations in material properties and assumptions are discussed in <Section 3.7>. For static analyses the basic assumptions are given in <Section 3.8.1.8.4.1>.

These are the usual assumptions made for concrete structures under
the ASME Code. Material properties and quality control specified
in <Section 3.8.1.6>, assure that material properties are within
the ranges of values anticipated by the ASME Code.

3.8.1.8.4.4   Expected Behavior Under Loads

a.   Non-axisymmetric, localized, transient loads

The methods of analysis and design used for these structures are
expected to reproduce the behavior of the constructed system. The
analytical and design methods considered axisymmetric,
non-axisymmetric and local loads, transient thermal gradients, and
the effects of discontinuities.

3.8.1.8.4.5   Design Methods

a.   General:

The method of design, used for reinforced concrete sections is the
classical linear elastic design method with load factors as defined
in the ASME Code.

All stresses and strains are in accordance with the member capacity
requirements as set forth in the loading combinations of
<Section 3.8.1.8.3.2>.

b.   Deformations and Deflections:

No permanent overall deformation occurs during any of the factored
loading combinations set forth in <Section 3.8.1.3.2>.

c.   Thermal Limitations:

     The design of the annulus concrete is based on limiting the
     temperature of the reinforced concrete during normal plant
     operation to 150°F except for local areas such as pipe penetrations
     where 200°F maximum is permitted.

d.   Vital Subcompartments:

     The annulus concrete has no vital subcompartments.

e.   Methods of Reinforcing Critical Areas

     No penetrations are anchored in the annulus concrete.

f.   Methods of Design for Shear Effects

     1.   Transverse (Radial) Shear Reinforcement

          The horizontal ties (shear reinforcement) were designed to
          carry the transverse shear force in excess of what the
          concrete can carry in accordance with the criteria of the ASME
          Code, Section III, Division 2, Subsection CC 3421.4.1. The
          ties are No. 7 bars spaced circumferentially at each vertical
          bar in the bottom and every other bar in the top section. The
          vertical distribution of shear ties is as follows:

          Below horizontal Stiffener No. 1 -   4 tie elevations
          Between horizontal Stiffeners No. 1 &
            No. 2 -   4 tie elevations
          Between horizontal Stiffeners No. 2 &
            No. 3 -   4 tie elevations

Between horizontal Stiffeners No. 3 &

 No. 4 -   3 tie elevations

Above horizontal Stiffener No. 4 -   3 tie elevations

Extra ties are used around penetrations and attachment plate
stiffeners.

2.   Tangential Shear Reinforcement

Based on SRP 3.8.1, inclined reinforcement is required if the
tangential shear stress is greater than 40 psi for
abnormal/severe environmental loads and 60 psi for
abnormal/extreme environmental loads. These limits are very
conservative when compared with the ASME Code.

For the minimum reinforcement provided in the annulus
concrete, CC3421.5.1(a) of the ASME Code allows 107 psi before
inclined reinforcement would be required. However, the
maximum calculated tangential shear stress is 83 psi, which
occurs for the abnormal/extreme environmental condition;
therefore, inclined reinforcement is not required by the Code.
The SRP 3.8.1 requirements would result in inclined
reinforcement consisting of No. 5 bars at a 12 inch center to
center spacing. This amount of reinforcement seems rather
inconsequential relative to the No. 18 bars provided in the
vertical and horizontal directions. The stresses in the
orthogonal reinforcement and the strains in the concrete are
not significantly reduced by the addition of the
No. 5 inclined bars.

The design of the annulus concrete for tangential shear is
based on the shear allowable of the ASME Code rather than the
reduced allowables presented in SRP 3.8.1 for four reasons.
First, the magnitude of the tangential shear stresses are not

as severe as those for a typical concrete containment
subjected to the same seismic input. Second, the results of
recent research indicates that the tangential shear allowables
of the ASME Code are conservatively low considering the
magnitude of the stresses in the orthogonal reinforcement in
the annulus concrete. More importantly, calculations show
that concrete and steel strains are small which is the intent
of the SRP 3.8.1. The final reason is that calculations have
shown that the tangential shear forces can be transferred at
the basemat from the annulus concrete to the containment
vessel and shield building. Both of these structures can
safely carry this additional load.

3.8.1.8.4.6   Computer Programs

Computer programs used in the structural analysis of the annulus
concrete are discussed in <Section 3.8.1.4.6>.

3.8.1.8.5   Structural Acceptance Criteria

The structural acceptance criteria for the annulus concrete is based
upon accepted codes and standards, with modification to suit the
operational and functional criteria of these structures.

3.8.1.8.5.1   Stress and Strain Criteria

a.   The stress criteria is based on values presented in the ASME Code.
     The factored loads in the load combinations given and discussed in
     <Section 3.8.1.8.3> are based on ASME Section III, Division 2 and
     SRP 3.8.1. Displacements will be elastic such that gross permanent
     deformations will not occur.

b.   The response of the structures to seismic loads is developed from
     the analytical techniques described in <Section 3.8.1.8.4>. The

stresses due to static and dynamic loads are algebraically summed for the loading combinations set out in <Section 3.8.1.8.3>, to obtain the critical stress for each part.

3.8.1.8.6   Material Specifications, Quality Control and Special
                 Construction Techniques

The design of the annulus concrete is based upon material specifications giving acceptable limitations of physical and chemical properties for the structural materials used, and upon imposing quality control provisions to ensure that the materials meet the specifications.

The organization, responsibilities and general provisions for the Quality Assurance Program are described in <Chapter 17>. The quality control provisions that have been imposed for the structures are described herein. Wherever engineer approved design documents call for deviations or exceptions from the accepted codes and standards within this section, the approved design documents shall govern.

3.8.1.8.6.1   Concrete Construction

3.8.1.8.6.1.1   Codes and Standards

The following codes and standards were used to establish requirements for concrete construction:

ACI 347-78   "Recommended Practice for Concrete Formwork."

ACI 211.1-77   "Recommended Practice for Selecting Proportions for
               Normal Weight Concrete."

ACI 304-73   "Recommended Practice for Measuring, Mixing,
               Transporting, and Placing Concrete."

ACI 214-77   "Recommended Practice for Evaluation for Compression Test
                  Results of Field Concrete."

ACI 309-72   "Recommended Practices for Consolidation of Concrete."

3.8.1.8.6.1.2   Material Specifications

a.   Cement

     Cement used in the production of concrete shall conform to
     ASTM C 150, Type II, for moderate heat of hydration. The cement
     shall conform to the following optional specification limits
     defined in ASTM C 150:

     1.   58 percent maximum sum of tricalcium aluminate and tricalcium
          silicate.

     2.   0.6 percent maximum alkalies in the event the aggregates are
          found to be deleteriously reactive.

     3.   False set shall satisfy the requirements of Table 2A of
          ASTM C 150.

     4.   These tests shall be performed in accordance with the test
          methods listed in the index of the 1973 Annual Book of ASTM
          Standards, unless otherwise specified herein.

b.   Admixtures

     1.   Air Entraining Admixture

          Air entraining admixtures conform to the requirements of
          ASTM C 260-73.

2.   Water Reducing Densifier

     Water reducing densifiers, if added to the concrete, conform
     to ASTM C 494-71.

3.   Calcium Chloride, and admixtures containing Chlorides,
     Sulphides, Nitrates, or Boron will not be used.

c.   Aggregates

     The aggregate used for normal weight concrete complies with
     ASTM C 33-71a, except as modified by <Section 3.8.1.6.1.5.b.2>.
     The type and size of aggregate, slump and additives are established
     to minimize shrinkage and creep.

d.   Water

     The water for concrete is tested to show compliance with the
     requirements listed in this section before being used in the
     concrete.

1.   Water quality complies with the following:

     (a) Chlorides as Cl   250 ppm maximum

     (b) Solids   2,000 ppm maximum

     (c) Sulphate ion   250 ppm maximum

     (d) pH   between 6.0 and 8.0

2.   A comparison of the proposed mixing water properties is made
     with distilled water, using the following tests made with the
     proposed portland cement:

	Test	Method	Limits of Comparison
(a)	Soundness (Autoclave Expansion)	ASTM C151-71	±0.10 percent difference in gauge length
(b)	Vicat Time of Setting	ASTM C191-71	Initial Set ±10 minutes Final Set ±1 hour
(c)	Compressive Strength	ASTM C109-73	10 percent reduction of strength

3.8.1.8.6.1.3   Design Mixes

a.   Trial design mixes for normal weight concrete are proportioned in
     accordance with Method 1 or Method 2 of Section 3.8.2 of ACI 301-72
     and tested in accordance with the following ASTM standards to
     ascertain conformance to concrete specifications.

Test	ASTM
Method of Test for Compressive Strength of Cylindrical Concrete Specimens	C 39-72
Method of Test for Slump of Portland Cement Concrete	C 143-71
Method of Making and Curing Concrete Test Specimens in the Laboratory	C 192-69
Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method	C 231-72T

3.8.1.8.6.1.4   Production Concrete

The production mixes are those mixes developed from the design mix
testing program and proportioned in accordance with the results of the
trial design mix using material qualified and accepted for the work.

During construction, minor modification in the mixes may be necessitated by variation in aggregate gradation and moisture content. The production concrete is mixed and transported in accordance with ASTM C 94-73a.

3.8.1.8.6.1.5   Concrete Quality Control

a.   General Provisions

The services of a testing laboratory were obtained prior to commencing concrete work. The testing laboratory makes design mixes described in <Section 3.8.1.8.6.1.3>, using the proposed materials to consistencies satisfactory for the work, in order to determine suitable mix proportions necessary to produce concrete conforming to the type and strength requirements specified. During concrete operations, inspectors at the batch plant certify the mix proportions of each batch delivered to the site and periodically sample and test the concrete ingredients. These inspectors ensure that a ticket is provided for each batch, documenting the time loaded, actual proportions of the mix, amount of concrete, and concrete design strength. The cleanliness of trucks, and the handling and storage or aggregate are checked by the batch plant inspectors. The water additions at the batch plant are modified, if necessary, as required by measurements of the moisture content of the aggregates and gradation changes.

Inspectors at the construction site inspect reinforcing and form placement, make slump tests, make test cylinders, check air content, check placement temperature, and record weather conditions.

b. Concrete Materials

1. Cement

All cement is sampled and tested at the mill for conformance
to ASTM C 150-73a and certified mill test reports are
submitted to the owner at that time. All test data are
subject to verification by in-process tests by the testing
laboratory provided by the owner. All cement is sampled and
tested in accordance with ASTM C 150-73a. The chemical and
physical tests are conducted on every test sample. The cement
is not released for use on the project until it is verified
that it conforms to ASTM C 150-73a and the project concrete
specifications.

2. Aggregates

The fine aggregate conforms to ASTM C 33-71a with modification
to suit local conditions. Samples of the proposed fine
aggregate are tested to ensure compliance with the
specification. The coarse aggregate conforms to
ASTM C 33-71a. Samples of the proposed coarse aggregate are
tested to ensure compliance with the specification.

The following tests related to coarse aggregate are not
required for the reasons given:

(a) ASTM C 87, "Test for Effect of Organic Impurities in Fine
     Aggregate on Strength of Mortar:" Tests for effects of
     impurities should not be required unless such impurities
     are shown by ASTM C 40 to be present.

(b)   ASTM C 78, "Test for Flexural Strength of Concrete Using Simple Beam with Third Point Loading:" Flexural strength is not specified.

(c)   ASTM C 29, "Test for Unit Weight of Concrete,:" This test is intended to be used where slag is used in the concrete.

(d)   ASTM C 666, "Test for Resistance to Freezing and Thawing:" Historical records indicate no tendency for aggregate to be subject to freeze-thaw deterioration. Also, presence of external air will counteract the very small potential for any freeze-thaw action.

The type and frequency, during concrete operations, of ASTM standard tests are:

(a)   "Organic Impurities in Sands for Concrete," ASTM C 40-73-daily.

(b)   "Materials Finer Than No. 200 Sieve in Mineral Aggregates by Washing," ASTM C 117-69-daily.

(c)   "Sieve or Screen Analysis of Fine and Coarse Aggregates," ASTM C 136-71-daily.

(d)   "Friable Particles in Aggregates," ASTM C 142-71-monthly and at change of source.

(e)   "Lightweight Pieces in Aggregate," ASTM C 123-69-monthly and at change of source.

(f)   "Total Moisture Content of Aggregate by Drying," ASTM C 566-67, twice daily, and as required for control of production.

(g)   "Scratch Hardness of Coarse Aggregate Particles,"
ASTM C 235-68, monthly and at change of source.

(h)   "Resistance to Abrasion of Large Size Coarse Aggregate by
Use of the Los Angeles Machine," ASTM C 535-69, every
6 months and at change of source.

(i)   "Resistance to Abrasion of Small Size Coarse Aggregate by
Use of the Los Angeles Machine," ASTM C 131-69, every
6 months and at change of source.

(j)   "Potential Reactivity of Aggregates," ASTM C 289-71,
every 6 months and at change of source.

(k)   "Soundness of Aggregates by Use of Sodium Sulphate or
Magnesium Sulphate," ASTM C 88-73, every 6 months and at
change of source.

(l)   "Standard Method of Test for Specific Gravity and
Absorption of Coarse Aggregate," ASTM C 127-73, monthly
during production and at change of source.

(m)   "Standard Method of Test for Specific Gravity and
Absorption of Fine Aggregate," ASTM C 128-68, monthly
during production and at change of source.

(n)   "Standard Tests for Water - Soluble Chlorides Present as
Admixes in Graded Aggregate Road Mixer," ASTM D 1411-69,
every 6 months and at change of source.

The frequency of these tests is changed if and when the source
of the aggregate changes, or if field conditions require more
frequent testing, or as required by the Engineer. Both fine
and coarse aggregates are tested for moisture content twice

daily as a minimum, or as required by climatic conditions. All aggregate test samples are taken from the batch plant stock pile.

3. Water

    The water used in the production of trial mixes and production concrete is as described in <Section 3.8.1.8.6.1.2.d>. The water is tested for compliance with these requirements for use in the trial mixes, and for the production concrete.

4. Admixtures

    The air entraining admixture used conforms to and is tested in accordance with ASTM C 260-73. The supplier submits to the owner documentation to demonstrate that the admixture supplied is essentially identical in concentration, composition and performance to the admixture originally tested to meet ASTM C 260-73 prior to or with each delivery of admixture. The concrete air content is to be taken for every 100 cubic yards of concrete during the concrete operations.

5. Water Reducing Densifier

    Any water reducing densifier admixture used conforms to and is tested in accordance with ASTM C 494-71, Type "A" or Type "D." The quantity added, the control temperatures and method of mixing, conform to the manufacturer's recommendations for use of their product. The supplier submits to the owner documentation to demonstrate that the admixture supplied is essentially identical in concentration, composition and performance to the admixture originally tested to meet ASTM C 494, Type "A" or Type "D" prior to or with each delivery of admixture.

c.   Test Evaluations

     The evaluation of the test results for the 3,000 psi concrete is in
     accordance with Chapter 17 of ACI 301-72.

     1.   Preliminary

          (a) Concrete design mixes and the associated tests are
               performed by the testing laboratory provided by the
               owner. The proportions for the concrete mixes are
               determined by ACI 301-72. All concrete materials are
               tested as previously described. Compression test methods
               conform to ASTM C 39-72 and making and curing concrete
               test specimens conform to ASTM C 192-69.

     2.   Field Tests

          During concrete operations, an inspector at the batch plant
          certifies the mixed proportions of each batch delivered to the
          job site, and samples and tests all concrete ingredients.
          Inspectors at the job site inspect reinforcing, embedments and
          form placements, make slump tests, make test cylinders, check
          air content, check concrete temperature, record weather
          conditions, and inspect concrete placing and curing. Concrete
          test cylinders and slump tests are molded, cured, capped, and
          tested in accordance with ACI 301-72 except that samples of
          concrete for cylinders may be taken from single location in
          the middle 70 percent of the load as an option to samples
          taken from several points in the load as required by ACI 301,
          Section 16.3.4.1.

          The following tests are also performed:

          (a) Sampling and testing the concrete materials as previously
               described.

(b) Slump tests will be made at the following minimum
        frequencies:

        (1) A slump test will be made on the first two trucks
             and thereafter for each 50 cubic yards placed of
             each class of concrete.

        (2) Slump tests will be made on each concrete batch used
             for test cylinders.

        (3) Slump tests will be made at any time the constructor
             has reason to suspect that the concrete slumps are
             not within the allowable tolerances.

        (4) Slump tests will be made on each concrete batch to
             which water from the truck water system was added to
             the batch.

3.8.1.8.6.1.6   Tolerances

The only tolerance is on the top elevation of the concrete (±1/2 inch).

3.8.1.8.6.2   Reinforcing Steel

3.8.1.8.6.2.1   Codes and Standards

The following codes and standards are used to establish the approved
design documents governing reinforcing steel:

ASTM A 615-72 Standard Specification for Deformed Billet Steel Bars for
               Concrete Reinforcement

ASTM A 370-72 Standard Methods and Definitions for Mechanical Testing
               of Steel products

NRC <Regulatory Guide 1.15>   Testing of Reinforcing Bars for Concrete
Structures

3.8.1.8.6.2.2   Materials

The reinforcing steel is deformed bar conforming to ASTM A 615,
Grade 60.

3.8.1.8.6.2.3   Quality Control

See the discussion for reinforcing steel in <Section 3.8.1.6.2.3>.

3.8.1.8.6.2.4   Tolerances

The fabrication tolerances for the reinforcement are in accordance with
the requirements of ACI 318. Placing tolerances are based on the ASME
Code, Section III, Division 2, Article CC 4342. The tolerances are
summarized as follows:

a.   For locations of bars within the depth or thickness of a member:

Thickness of Member (inches)	Tolerance on Location (inches)	Reduction on Cover (inches)
Less than 10	±3/8	3/8
10 to less than 26	±1/2	1/2
26 to greater	±1	1/2

b.   For longitudinal location of bends and for ends of bars:
      ±2 inches.

c.   For spacing of bars: ±1 inch on the specified spacing, except that
      at embedments ±2 inches on the specified spacing is acceptable.

3.8.1.8.6.3   Cadweld Splices

3.8.1.8.6.3.1   General Requirements

a. Cadweld splices will be used on bar sizes exceeding No. 11, on
    structural steel anchorages and where there is insufficient length
    for lap splices.

b. Cadweld splices will be made with T-series and B-series sleeves as
    described in Erico Products Catalogue (Reference 5). These splice
    sleeves are capable of developing the minimum ultimate tensile
    strength of the reinforcing steel.

c. A manufacturer's representative, experienced in Cadweld splicing
    will be present at the outset of splicing to demonstrate the
    equipment and techniques for making quality splices.

d. A quality control program based on the 1977 ASME Code Section III,
    Division 2, Subsection CC-4333, will be used for inspection and
    testing of Cadweld splices.

e. Sister splices can be used for testing in combination with the
    required number of production splices. Sister splices are
    removable splices made in place next to production splices and
    under the same conditions. The position and orientation of a
    sister splice is similar to the production splice that it
    represents.

3.8.1.8.6.3.2   Quality Control

a. Qualification of Operators

    Prior to the production splicing of reinforcing bars, each member
    of the splicing crew (or each crew if the members work as a crew)

prepares two qualification splices on the largest bar size for each of the splice positions (e.g., horizontal, vertical, diagonal) to be used. The qualification splices are made using the same materials (e.g., bar, sleeve, powder) as those to be used in the structure. The qualification splices shall meet the criteria set forth in Erico Productions publication (Reference 6).

b.   Fabrication Procedure

The following quality control procedures are followed to ensure acceptable splices:

1.   The splice sleeve, powder and molds are stored in a clean dry area with adequate protection from the elements to prevent absorption of moisture.

2.   Each splice sleeve is visually examined immediately prior to use to ensure the absence of foreign material on the inside surface. Light or spotty rust on the inside surface of the sleeve is acceptable.

3.   The molds are preheated to drive off moisture at the beginning of each shift when the molds are cold or when a new mold is used.

4.   Bar ends to be spliced are power wire brushed for a distance of two inches plus 1/2 the sleeve length from end of bar to remove mill scale, rust, concrete, and other foreign material. Prior to brushing, water, grease and paint are removed by heating the bar ends with a torch.

5.   A permanent line is marked on each bar for a reference point to confirm that the bar ends are properly centered in the splice sleeve.

6. Before the splice sleeve is placed into final position, the bar ends are examined to ensure that the surface is free from moisture. If moisture is present, bar ends are heated until dry.

7. Special attention is given to maintaining the alignment of sleeve and guide tube to ensure a proper fill.

8. The splice sleeve is flame dried after all materials and equipment are in position.

9. All completed splices are visually inspected at both ends of the splice sleeve and at the tap hole in the center of the splice in accordance with the requirements of Erico Products "Inspection of Cadweld Rebar Splice" (Reference 6). Splices which do not meet this criteria are rejected and will not be used as tensile test samples.

10. Records are kept of all Cadweld splices placed. These records will enable a test splice to be chosen at random and to establish its location. The records plus test results of all Cadweld splices tested will be kept up-to-date and will be available to all inspectors. These records shall show:

    (a) The number assigned to each splice

    (b) The date that the splice was formed

    (c) The crew which carried out the splicing

    (d) The size, location and orientation of the splice

    (e) The record of visual inspection

(f) The results of tensile tests performed

c. Testing Frequency

Separate test cycles shall be established for splices in horizontal, vertical and diagonal bars. The minimum number of splices to be tested is as follows:

1. If only production splices are used in a test cycle, the sample frequency shall be:

(a) One of the first ten splices

(b) One of the next 90 splices

(c) Two of the next and subsequent units of 100 splices

2. If production and sister splices are tested, the sample frequency shall be:

(a) One production splice of the first ten production splices.

(b) One production and three sister splices, for the next 90 production splices.

(c) One splice, either a production or sister splice for the next and subsequent units of 33 splices. At least one-fourth of the total number of splices tested shall be production splices.

(d) Straight sister splices shall be substituted for production splice test samples on radius bent bars, on splices connecting reinforcing steel to structural steel

and where sufficient length of protruding bar is not
available for the removal of a production splice and
consequential resplicing.

d.   Acceptance Criteria

1.   Sound, nonporous filler material is visible at both ends of
the splice sleeve and at the tap hole in the center of the
sleeve. Filler material is usually recessed 1/4 inch from the
end of the sleeve due to the packing material, and is not
considered a poor fill.

2.   Splices which contain slag or porous metal in the riser, tap
hole or at the ends of the sleeve are rejected. A single
shrinkage bubble present below the riser is not detrimental
and is distinguished from general porosity as described above.

3.   Evidence of filler material between the sleeve and bar for the
full 360 degrees; however, the splice sleeves do not need to
be exactly concentric or axially aligned with the bars.

4.   The tensile strength of each individual test splice shall be
equal to or greater than 125 percent of the minimum yield
strength specified in ASTM A 615 for the grade of
reinforcement being used.

5.   The average tensile strength of a group of consecutive samples
in a test cycle shall equal or exceed the guaranteed ultimate
strength for that grade of reinforcement as specified in
ASTM A 615. Where the number of test samples in a cycle
exceeds 15, a running average shall be calculated based on the
last 15 samples tested.

e.   Splice Nonconformance Criteria

1.   If a production or sister splice sample fails to meet the
     acceptance criteria and failure occurred in the bar, another
     sample will be chosen for testing. The cause of failure in
     the bar will be investigated. Any necessary corrective action
     affecting splice samples will be implemented prior to
     continuing the testing.

2.   If any production splice used for testing fails to meet the
     acceptance criteria and failure did not occur in the bar, the
     production splice made just preceding and the production
     splice made just following the nonconforming splice will be
     tested. If any sister splice used for testing fails to meet
     the acceptance criteria and failures do not occur in the bar,
     two additional sister splices will be tested. If either of
     these retests fails to meet the acceptance criteria, splicing
     by the crew performing the work represented by the failed
     splice will be halted. Splicing will not be resumed until the
     cause of the failures has been determined and corrected.

3.   If the running average tensile strength of the samples in a
     test cycle fails to meet the acceptance criteria, splicing
     will be halted. Splicing will not be resumed until the cause
     of the nonconforming strengths has been determined and
     corrected.

3.8.1.8.6.3.3   Tolerances

The fabrication tolerances for Cadweld splices are as stated in
<Section 3.8.1.8.6.3.2.b>.

3.8.1.8.6.4   Lowest Service Metal Temperature

The lowest service metal temperature shall be 35°F and the lowest
temperature during construction is -15°F.

3.8.1.8.7   Testing and Inservice Inspection Requirements

There are no planned systematic testing or inservice surveillance
programs for the annulus concrete except for the containment structural
integrity test described in <Section 3.8.2.7.1>.

3.8.2   STEEL CONTAINMENT

3.8.2.1   Description of the Containment

3.8.2.1.1   General

The containment vessel is a pressure retaining structure composed of a
free standing steel cylinder with an ellipsoidal dome, secured to a
steel lined reinforced concrete foundation mat. The mat is the common
foundation for the three major structures of the reactor building
complex. The free standing portion of the containment vessel is
supported by and anchored into the foundation mat, and is designed,
fabricated and erected in accordance with the requirements of ASME Code
Section III for Class MC components. Neither the shield building
(except in the area of the filled annulus) nor the interior structure
(except for grating supports) contacts the free standing section of the
containment vessel. Sufficient clearance is provided to ensure that
contact does not occur during any of the postulated load combinations.
<Section 3.8.0> gives a description of the reactor building complex
structures and <Figure 3.8-1> shows their physical relationship.

The containment vessel is designed to contain radioactive material which
might be released from the nuclear steam supply system following a

loss-of-coolant accident. The steel containment vessel ensures a high degree of leak tightness during normal operating and accident conditions.

The containment vessel is a safety class structure, as defined in <Section 3.2>, with an internal free air volume of approximately 1.2×10^6 cubic feet. It is designed for a maximum internal pressure of 15 psig with a coincident temperature of 185°F at accident conditions, and a maximum external pressure differential of 0.8 psi due to accidental operation of the spray headers. The maximum design leakage rate for the containment vessel is 0.2 percent by weight of contained air in 24 hours at 7.80 psig internal pressure (Pa). The design of the containment vessel considers dead load, live load, construction loads, temperature gradients, and the effects of penetrations for both the accident and normal (including seismic) operating conditions.

The basic dimensions of the containment vessel are:

a.   Cylinder inside diameter - 120 feet.

b.   Cylinder height - 152 feet - 2 inches.

c.   Ellipsoidal dome ratio - 2:1.

The containment vessel cylinder has six external stiffening rings at various elevations. Details of the containment vessel cylinder, dome and stiffeners are shown in <Figure 3.8-1>.

Two personnel access airlocks and one equipment hatch are provided. Details of these are shown in <Figure 3.8-4> and <Figure 3.8-5>. The reinforced concrete foundation mat which provides support for the interior structure, containment vessel and shield building is discussed in <Section 3.8.5>.

The lower 18 feet 6 inches of the containment vessel forms the outside of the suppression pool; the inside of the suppression pool is formed by the drywell weir wall. The latter is part of the interior structure and is discussed in <Section 3.8.3>. Corrosion of the lower 23 feet 6 inches of the containment vessel and exposed steel mat liner is minimized by the use of ASME SA 516, Grade 70, plate clad with stainless steel. The location and the relationship of the suppression pool to the containment vessel is shown in <Figure 3.8-1>.

At Elevation 721'-0" a 125 ton capacity polar crane is supported off the containment vessel. The crane support girder is shown in <Figure 3.8-14>.

The unstiffened ellipsoidal dome is designed to support a nine inch layer of wet concrete. This nine inch layer was then used to form the working surface for pouring the remaining 21 inch thickness of the shield building dome.

The addition of the concrete between the shield building and containment vessel to Elevation 598'-4" increases the strength and stiffness of the containment vessel.

3.8.2.1.2   Plate Thickness

Final design plate thickness of the containment vessel cylinder and dome is 1-1/2 inches. This is increased to three-inch thick plate around the penetrations in the suppression pool region to provide local reinforcement for containment vessel pressure loads as well as penetration loads. Base liner plates are 3/8 inch thick carbon steel where they are covered by concrete, and half inch thick stainless clad where they are exposed to the suppression pool water.

3.8.2.1.3   Test Channels and Test Assemblies

Steel test channels are provided along welds of the steel liner of the foundation mat so that local leak testing of welds can be performed. The channels are segmented by area. One plug fitting is provided for each area and extends through any covering material, including concrete.

3.8.2.1.4   Personnel Access Air Locks

Two personnel access airlocks with an outside diameter of 9 feet 7 inches are provided, one at Elevation 599'-9", the other at Elevation 689'-6".

The personnel access airlocks are welded steel assemblies with double doors, each equipped with double gaskets and designed to provide the capability of leak rate testing the airlock between doors and the cavity between door seals at a pressure of Pa (7.80 psig). Since personnel airlock door seals must be tested frequently, connections for leak rate testing including the potential for an automatic leak rate monitor (ALRM) are provided to the seal test points on the inner door and outer door of each airlock. The seal test connection system is shown schematically in <Figure 3.8-100>.

The airlock doors are designed as pressure seating doors. Since both doors swing towards the reactor, any positive pressure buildup in the containment will tend to close the doors against their door jambs. When pressure testing between door seals, the doors are prevented from opening by means of the two latch cylinders mounted on each door. When pressurizing the entire airlock for a structural integrity or leak rate

test, test clamps must be installed on the airlock side of the reactor end door. Test pressures will determine the number of test clamps used on the reactor end door per airlock manufacture procedures. Test pressures above 8.58 psig require 4 clamps. These test clamps consisting of steel bars and clevis pins, engage clevises on the door and the door jamb to mechanically secure the door. Additional details of the locks are:

a. The doors are interlocked to prevent both being opened simultaneously. Interlocks are connected so that one door must be completely closed and sealed before the opposite door can be opened.

b. Remote open/closed indication lights are provided in the control room to indicate the position of any personnel access air lock door. A special remote alarm is provided whenever both doors in an airlock are not locked closed.

c. Locks have an interior lighting system which is capable of operating from the emergency power supply.

d. Locks have an emergency communication system.

e. Provisions are made to permit bypassing the door interlocking system to allow doors to be left open when the plant is shut down.

f. The floor system is designed so that it can be easily removed.

g. The locks are designed, fabricated, tested, and inspected in accordance with the ASME Code, Section III, Class MC.

h. Lock hinges are capable of independent three-dimensional adjustment to assist proper seating.

i. Equalizing valves are provided.

j. Provisions are made for inservice leak testing of the door bulkhead
    penetration gaskets at 7.80 psig (Pa).

3.8.2.1.5   Equipment Access Hatch

An equipment access hatch with a clear inside diameter of 20 feet is
provided at Elevation 620'-6" to allow passage of large equipment and
components into the containment vessel. The flanged joint between the
hatch and cover is designed to accommodate double seals. Periodic leak
testing of the hatch is accomplished by pressurizing the space between
the seals.

3.8.2.1.6   Penetrations

Typical mechanical and electrical penetrations are shown in
<Figure 3.8-6> and <Figure 3.8-7>. Penetrations are of the double
barrier type and are designed to provide an annular air space that can
be pressurized to the containment vessel design pressure (15.0 psig),
for leak testing during plant operation and to 17.25 psig for proof
testing.

a. Mechanical Penetrations

    1. For hot pipe lines, thermal expansion is considered, and an
        expansion joint is provided between the pipe and sleeve at the
        second barrier to accommodate the calculated axial and lateral
        pipe motions.

    2. Thermal insulation is provided where it is required.

    3. Process pipes within penetrations are of greater schedule than
        the same pipes elsewhere in the system.

4. Bellows, expansion joints, gaskets, canopies, protectors, or
    other flexible members are designed for a minimum of
    500 cycles of movement associated with each penetration.

5. Penetration sleeves are installed in their respective plate
    sections in the shop.

6. Each penetration is equipped with a test connection located
    outside of the containment vessel to enable leak testing of
    the annulus space.

7. Mechanical penetration primary barriers are designed,
    fabricated, tested, and inspected in accordance with ASME
    Code, Section III, Class MC, Class 1, or Class 2.

8. The fuel transfer penetration connects the upper fuel pool
    above the drywell to the fuel handling building. This
    penetration has a guard pipe and is inclined at an angle of
    57° to the horizontal. The penetration, in connecting these
    two structures, passes through the containment vessel and
    shield building. The design of the penetration allows for the
    postulated deflections and movements of these two structures.
    A modification to provide for leak rate testing of the
    transfer tube bellows has been installed. The detail of the
    fuel transfer penetration is shown in <Figure 3.8-15>.

b. Electrical Penetrations

1. These penetrations have cartridge assemblies which are
    installed in the penetration sleeves in the field.

2. The penetration sleeves are shop welded to their plate
    sections.

3.   The penetration sleeves, cartridges and flanges are designed,
        fabricated, tested, and inspected in accordance with the ASME
        Code, Section III, Class MC.

4.   The penetrations are designed to comply with IEEE Standard 317
        (Reference 7).

5.   The complete penetration assembly is designed to be capable of
        withstanding a structural integrity test pressure of
        17.25 psig in the interspace between cartridges, sleeves and
        flanges.

6.   Penetrations are equipped with a test connection located
        outside the containment vessel.

7.   The penetration assembly design is qualified by testing one
        assembly of each type in the accident environment as required
        by IEEE Standard 317 (Reference 7).

8.   Thermocouples are installed in some assemblies to monitor heat
        generated by conductors contained within the assembly.

3.8.2.1.7   Penetration Appurtenances

Penetration sleeves are shop attached to a reinforcing plate which is
then field installed. Where several penetrations are grouped rather
close together, the reinforcing plate is large enough to accommodate all
of the penetrations and these "ganged" penetrations are field installed
as a single unit.

3.8.2.1.8   Containment Vessel Vacuum Relief System (Vacuum Breakers)

See <Section 6.2.1.1.4.2> for system description, system design bases
and effects on containment vessel design.

3.8.2.2   Applicable Codes, Standards and Specifications

3.8.2.2.1   Codes

The applicable code for the containment vessel is the ASME Code,
Section III, Subsection NE.

3.8.2.2.1.1   Code Classification

The steel containment vessel is classified Class MC in accordance with
Sub-Article NA-2130, Section III of the ASME Code.

3.8.2.2.1.2   Code Compliance

a.   Containment vessel

     The steel cylindrical shell and dome of the steel containment
     vessel, including all penetrations and attachments within the
     boundaries defined in <Section 3.8.2.2.1.3>, are designed,
     fabricated and erected in accordance with Subsection NE, Class MC
     components, including the requirements for quality assurance of
     Article NA-4000 and inspection requirements of Article NA-5000 of
     Section III of the ASME Code and <Regulatory Guide 1.57>, except as
     modified by <Section 3.8.2.5>.

b.   Code stamp

     The steel containment vessel is not ASME Code stamped. However,
     all other requirements of the Code applicable to Class MC
     containment vessels are met.

3.8.2.2.1.3   Code Boundaries

The boundaries for the steel containment vessel consist of those defined
in Paragraphs NE-3254 and NE-1132 of the ASME Code Section III, and the
additional boundaries listed below.

a.   The steel containment vessel shell and dome including the portion
     of the shell embedded in the concrete mat foundation, but not
     including the associated anchorage steel.

b.   The attachment weld of the bottom liner plate to the steel
     containment vessel shell.

c.   The leak tight base liner plate is outside of the boundaries for
     the steel containment vessel.

d.   The attachment welds of the crane girder, piping supports, walkway
     or platform supports, and other attachments to the shell of the
     steel containment vessel.

3.8.2.2.1.4   Base Liner

The portion of the base liner exposed to suppression pool water is
designed to the requirements of Subsection CC-3000, Section III, of the
ASME Code, Division 2, except as provided in CC-1120 which provides that
the liner meets Subsection NE allowables of Division 1 for loads not
carried by the concrete. The base liner inside the Drywell is protected
from the environment by the 8'-8" thick reinforced concrete support mat.
This section of the liner provides a leak-tight barrier with the
concrete designed to carry all imposed loads. The strain criteria
adopted for this section of the liner is in accordance with the ASME
Code, Section III, Division 2, Subsection CC-3700.

3.8.2.2.1.5   Structural Steel Attachments

Structural steel attachments beyond the boundaries established for the
steel containment vessel and designed and constructed in accordance with
the "Specification for the Design, Fabrication and Erection of
Structural Steel for Buildings," AISC 1969 including Supplements No. 1
and 2, except:

a.   As modified by "Structural Design Criteria for Category I
      Structures Outside Containment" (Reference 1).

b.   Qualification of welding procedures and welders/operators per ASME
      Code Section IX may be substituted for AWS D1.1 Section 5
      qualifications with the exception that ASME IX Welding Procedure
      Specifications for partial and complete penetration groove welds
      shall conform with the AWS D1.1 requirements for prequalified
      welded joints, specifically joint groove type/shape (i.e., groove
      angle and type, root opening and face, and omission of backing
      material). Use of non-prequalified AWS D1.1 joints shall require
      procedure qualification.

3.8.2.2.2   Standards

Applicable standards are given in <Section 3.8.2.6.1.1>. <Regulatory
Guide 1.57> is used for the design of the containment vessel, with
modification given in <Section 3.8.2.5>.

3.8.2.2.3   Specifications

Applicable plant principal specifications for the containment vessel
scope of work include final design, fabrication, erection, and
structural integrity testing. The design requirements included all
loads and load combinations listed in <Section 3.8.2.3> and the
structural acceptance criteria given in <Section 3.8.2.5>. Fabrication

and erection requirements are in accordance with the requirements of
ASME Code Section III, Subsection NE, except for the modifications
described in <Section 3.8.2.7>. Quality Control provisions, tolerances,
etc. are described in <Section 3.8.2.7>.

3.8.2.3   Loads and Load Combinations

3.8.2.3.1   Loads Used in the Design

Loads used in the design are in accordance with the ASME Code
Section III, Subsection NE-3110, and are summarized below.

a.   Dead Load (symbol D)

     Dead load includes the weight of the containment vessel, calculated
     at 490 pounds per cubic foot, plus any permanent attachments such
     as spray headers, polar crane, equipment hatch, personnel access
     locks, penetrations, etc.

b.   Construction Loads

     1.   Nine inches of wet concrete plus formwork for the entire
          shield building dome, applied as point loads to the
          containment vessel dome.

     2.   Fifty pounds per square foot over any part of the containment
          vessel dome, not concurrent with Item 1, above.

     3.   Loads due to a 65 mph wind 30 feet above grade, as given by
          ASCE Paper No. 3269 "Wind Forces on Structures."

     4.   Snow load of 30 psf, not concurrent with Item 1 or 3, above.

5.   Wet concrete due to annulus concrete placement.

c.   Live Loads (symbol L)

Live load includes any equipment loads not included in item (a) above, also:

1.   Temporary loads during construction.

2.   A uniform load of 100 psf or a concentrated load of 1,000 pounds applied to the passage area of the personnel access airlocks.

3.   A load of 100 psf on platforms and walkways.

4.   Reactions from spray system activation.

5.   Polar crane live load plus impact allowances.

6.   Weight of suppression pool water.

7.   A uniform load of 500 psf or two 20,000 pound wheel loads at five inch tread, applied to the passage area of the equipment hatch.

d.   Wind and Tornado Loads (symbol W and W_t)

Because the containment vessel is totally enclosed by the shield building the effects of wind and tornado loadings are insignificant following completion of the shield building construction. Wind load is allowed for during the construction stage prior to completion of the shield building.

e.   Seismic Loads (symbol F_{eqo} and F_{eqs})

The seismic loads investigated for the containment vessel are based
on the postulated OBE and SSE conditions set forth in
<Section 3.7>.

f.   Operational, Design and Test Pressure Loads (symbol Z)

The operational, design and test pressures are:

1.   Design external   0.8 psig

2.   Operating external P_{atm} - 0.015 psia

3.   Operating internal   2.0 psia +14.7 psia

4.   Structural integrity test   17.25 psig

5.   Blowdown of a main steam relief valve line increases the
     suppression pool pressure above normal hydrostatic pressures.
     The pressure varies with time and distance from the relief
     line as described in <Appendix 3A> and <Appendix 3B>. Any or
     all of the 19 relief lines may be blown down in any sequence.

6.   Tornado depressurization - external   -1.3 psig
     Effective negative external pressure based on known vent paths
     relative to -3.0 psig outside the shield buildling.

g. Normal Operating Temperature Loads (symbol T_o)

1. Normal operating air (simultaneous) temperatures are:

	Annular Space	Inside Containment
(a) Minimum operating, °F	50	60

		Annular Space	Inside Containment

(b) Normal operating, °F   90   95

(c) Maximum operating, °F   104   105

2.   The suppression pool is limited to the following temperatures by plant operating procedures during normal operation:

(a)   Minimum pool temperature of 60°F.

(b)   Maximum pool temperature of 140°F with safety/relief valve operation.

(c)   Maximum normal operating pool temperature of 150°F, continuous safety/relief valve blowdown without bubble pressure loads.

h.   Loss-of-Coolant Accident Temperature (T_a) and Pressure (P_a)

The design allows for postulated accident conditions of:

1.   Internal pressure, psig   7.80

2.   Temperature, °F   160.5 inside the containment vessel

                                    180 outside the containment vessel in annular space

3. Suppression Pool Loads

    Suppression pool dynamic loads on the containment vessel are
    described in <Appendix 3A> and <Appendix 3B>. Hydrostatic
    loads are described below in item j.

i. Interior Missiles (symbol Y_m)

    There are no interior missile loads on the containment vessel.

j. Hydrostatic Loads (symbol G_{sp}, G')

    1. The design allows for the water retained in the suppression
        pool. This water is present for the life of the plant with
        occasional draining of the pool for inspection of the
        submerged areas. Normal pool depth is 18 feet 6 inches with a
        variation of plus 0 feet minus 6 inches. (G_{sp})

    2. For the recovery of the reactor fuel and core in a
        postaccident condition, the containment vessel is designed to
        withstand internal water rising to Elevation 641'-0" a height
        of 66 feet 2 inches. (G')

k. Penetration Loads (symbol R_o, R_a, Y_r, and Y_j)

    The loads imposed by personnel access locks, equipment hatch,
    mechanical and electrical penetrations for normal operation (R_o)
    are considered together with the dead load. Loads applied as a
    result of rupture of any one pipe or by interaction of penetrations
    (R_a and Y_r) are considered with accident conditions applicable to
    the type and size of the break. The only jet impingement loads (Y)
    due to pipe rupture on the containment vessel result from breaks in
    the CRD piping and are less than 3,000 pounds spread over 3 square
    inches.

l. Safety/Relief Valve Operation (symbols P_{srv} and T_{srv}) loads on the containment vessel due to safety/relief valve operation are as described in <Appendix 3A> and <Appendix 3B>. In addition, cyclic loadings considered for the containment vessel fatigue analysis are:

1. Startup and shutdown of plant:

Case/Temp (°F)		Quencher(s)	Discharge Events
(a)	Single valve, single Actuation/(60-90)	LLSP See Note[2]	34
(b)	Single valve, subsequent Actuation[1], [3]	LLSP See Note[2]	1,512
(c)	Two valves, first Actuation/(60-90)	LLSP, Adjacent See Note[2]	1
(d)	19 valves/(60-90)	All	253
(e)	8 ADS/(120)	V2,4,7 V9,11,13 V16,18	1

NOTES:

[1] For case (b), the temperatures for subsequent Actuations 1 to 14 are as follows:

Subsequent Actuation: 1 2 3 4 5 6 7-10 11-14 Local pool temperature (°F): 100 110 120 125 127 128 129 130

[2] LLSP is the low-low set point valve. Adjacent is the adjacent valve that actuates next in sequence (has the next lowest set point).

[3] All 1,512 subsequent actuations of the low-low set point valve will occur as a 15 actuation series. The initial actuation is at a pool temperature of 90°F and is followed by 14 subsequent actuations with temperatures as shown in Note[1] above.

2.   Normal operating pressure fluctuations:

    (a) ±1 inch of water gauge - 1,000 cycles

    (b) 0 to 2 psig - 10 cycles

3.   Safety/relief valve discharge:

    At the location of the low-low set point valve the containment
    vessel and base liner are exposed to pressure fluctuations
    induced by safety/relief valve discharge; therefore the design
    at this location includes 1,800 discharge events with an
    average of nine cycles per event. At other points on the
    containment, the number of discharge events used in design is
    given in <Section 3.8.2.3.1.1.1> with nine stress cycles per
    event.

3.8.2.3.2   Load Combinations

The design considers the load combinations given in <Table 3.8-2> in
determining the strength requirements (SR) for the various sections of
the containment vessels except that <Table 3.8-13> is used for all ASME
Code Section III, Division 2, analyses. Many of the applied loads vary
within a range, and for this type of load, the one used is that value
producing the most critical combination. The absolute sum method was
used to combine all loads for the steel containment vessel. Allowable
stress criteria applied to each of these combinations discussed in
<Section 3.8.2.5>.

3.8.2.3.3   Load Factors

No load factors are used on any of the actual applied loads for
Section III, Division 1, analyses. Instead, the stress criteria used
vary as given by ASME Code Section III, Subsection NE-3200, and

elaborated further in <Section 3.8.2.5>. Load factors are used with
Section III, Division 2, analyses. The quencher foundation and
containment vessel embedment are evaluated for pullout using
Section III, Division 2, stress criteria.

3.8.2.3.4   Load Combinations for Local Areas

Local effects of penetrations, locks and hatches are considered in the
general analysis by the use of the symbols R_o, R_a and Y_r. The loads used
for the design of individual components are the same as the general load
combinations.

3.8.2.3.5   Thermal Effects

The temperature induced stresses, are treated as secondary stresses and
are investigated as discussed in ASME Code, Subsection NE-3213.13.

3.8.2.3.6   Containment Capability Analysis

As part of the overall evaluation of the Perry Nuclear Power Plant
Hydrogen Control System <Section 6.2.9>, containment vessel pressure
capability has been evaluated (Reference 21) and (Reference 22). The
cases considered and a summary of analytical results for these
evaluations are as follows.

ASME Service level[3]	Loadings Considered	Results/Capability
C	45 psig internal pressure and dead load	All stresses within service level C allowables
C	See Note[1]	50.2 psig

ASME Service level[3]	Loadings Considered	Results/Capability
D	See Note[2]	57.4 psig[4]

NOTES:

[1] (Maximum internal containment pressure and dead load) in order to reach service level C allowables.

[2] (Maximum internal containment pressure and dead load) in order to reach service level D allowables.

[3] ASME Section III, Division 1, Subsection NE.

[4] Considered a more realistic evaluation of containment vessel pressure capability, considering the nature and probability of the assumed hydrogen generation event.

Per Table 17 of (Reference 22), the maximum calculated containment internal pressure due to hydrogen generation/burning is 21.2 psig which is significantly below the more realistic service level D structural capacity of 57.4 psig.

3.8.2.4   Design and Analysis Procedures

3.8.2.4.1   General

Final analysis and design of the containment vessel were performed by the vessel contractor, Newport News Industrial Corporation (NNIC) of Newport News, Virginia. The analysis and design were based on the ASME Code, Section III, Subsection NE, for Class MC vessels, using a combination of hand calculations, empirical formulas and finite element computer programs to determine the appropriate plate thickness and stress levels within the vessel. The computer models used in the analysis are shown in <Figure 3.8-16>, <Figure 3.8-17>, <Figure 3.8-107>, <Figure 3.8-108>, and <Figure 3.8-109>.

3.8.2.4.2   General Assumptions and Boundary Conditions

It is assumed that the ASTM A 516, Grade 70, plate used for the
Containment Vessel is linear elastic, homogeneous and isotropic with a
uniform yield strength equal to the specified minimum. The base of the
structure is assumed to be fixed.

3.8.2.4.3   Analytical Techniques

a.   Static Analysis

     The static analysis uses two computer programs and three models.

     1.   Vessel dead load (axisymmetric) done on Ghosh-Wilson.

     2.   All other axisymmetric static loads for dome and vessel are
          done on STRAP.

     3.   All asymmetric static vessel loads done on Ghosh-Wilson.

     To obtain the general shell stresses for axisymmetric loads,
     including internal and external pressures, hydrostatic pressures,
     thermal, construction loads, and snow loads, the STRAP computer
     program with the models shown in <Figure 3.8-16> is used. Stresses
     from the asymmetric static loadings, hydrostatic and inertia
     effects due to pool slosh, wind loads and polar crane travel, are
     obtained using the Ghosh-Wilson program with the model shown in
     <Figure 3.8-17>. An NNIC in-house Fourier coefficient generator
     program calculates a Fourier representation of the input loadings
     and these are input into Ghosh-Wilson to calculate the stresses.
     Thermally induced stresses due to the filled annulus are calculated
     using the ANSYS computer program, using the model shown in
     <Figure 3.8-107>, and the local effect on the stiffeners was

investigated using the STRAP computer program, using the model
shown in <Figure 3.8-108>.

b.   Dynamic Analysis

The containment vessel is subjected to three dynamic loading
conditions; seismic events, safety/relief valve discharge (SRVD)
and loss-of-coolant accident (LOCA). Analytical techniques for
each of these conditions are:

1.   Seismic Analysis

The earthquake analysis is performed using the response
spectrum modal analysis technique described in <Regulatory
Guide 1.92>. The Ghosh-Wilson Program allows the floor
response curves to be input to the model shown in
<Figure 3.8-17>; then, using the technique above, the response
to each earthquake component is obtained. Thus, the total
response of the containment vessel is obtained by taking the
square root of the sum of the squares of the responses from
the component earthquakes. An additional analysis was
performed for the filled annulus using the Ghosh-Wilson
Program and the model shown in <Figure 3.8-109>. This
analysis was performed to verify that the filled annulus does
not increase the seismic response and seismic stresses.

2.   SRVD Analysis

The SRVD event is characterized by two types of loadings; a
transient asymmetric pressure exerted on the vessel by the
suppression pool water and vibration of the vessel through the
base mat. Four cases are considered, a single valve acting,
two adjacent valves acting, 8 (ADS), and all 19 valves acting.
<Appendix 3A> and <Appendix 3B> present a detailed discussion

of the SRVD loadings. The Ghosh-Wilson program is used to
calculate the maximum stresses due to the pressure loadings
and the vibratory motion. A Fourier series representation of
the asymmetric loading is input to the computer model for a
typical loading case. Effects of the water in the suppression
pool are included in the analysis by applying appropriate
values to the Lame's Constants and modeling the fluid with
standard solid finite elements. The fluid elements are
rigidly coupled to the vessel elements in the radial direction
but are allowed to move in the axial direction. This is
accomplished by using shear links between the appropriate
nodal points.

3.   LOCA Analysis

Loading conditions for a LOCA event vary with time after the
accident. Long term conditions are a gradual buildup of
pressure and temperature to the design values of 15 psi and
185°F. These loads are applied as static cases. Short term
loads due to a LOCA consist of hydrostatic loads, transient
loads due to wetwell pressurization, high mass flux steam
condensation, containment pressurization, and pool swell air
bubble pressurization. A detailed discussion of LOCA
conditions is given in <Section 6.2.1> and <Appendix 3A> and
<Appendix 3B>. The Ghosh-Wilson program is used to analyze
for the transient loads above. The pool swell air bubble load
is represented by a Fourier series and is applied as an
asymmetric load over 180° of the vessel.

c.   Heat Transfer Analysis

The heat transfer analyses for the containment vessel, annulus
concrete and foundation mat are performed using many different
models of key regions to consider the different coefficients of

thermal conductivity and varying boundary conditions existing under both normal operating and accident conditions. The analyses utilize an iterative technique to establish the initial, steady-state conditions through the model. Starting at the normal operating condition inside containment at steady-state with t = 0, the transient temperature response of the suppression pool, containment air and annulus air, are input into the model at varying time steps. Other boundary conditions, including the outside of the shield building and bottom side of the reactor building foundation mat, are held constant. Several containment vessel film coefficients are used in these models to bound the analyses.

This analysis was used in determining stresses in the doubler plate and shell during long term LOCA. For short term LOCA effects in the doubler plate and lower shell, a steady-state analysis was performed using the AXSOL computer program. Stresses in the upper containment were determined on the STRAP computer program using a steady-state analysis. The induced stresses are combined with the other loads in <Section 3.8.2.3>.

d.   Fatigue Analysis

A fatigue analysis of the vessel was performed in accordance with Subsection NE-3222.4 of the ASME Code, Section III. It was found that requirement No. 2 of NE-3222.4(d) was not met. Therefore, a more detailed analysis was performed in accordance with Section NE-3222.4(e). Peak stresses are in accordance with Section NE-3228.3.

e.   Analysis of Local Areas

1.   Polar Crane Girder

A 43° model of the girder with ribs and offset T beams and
rail is modeled on the STRAP computer program. The model
starts at the line of symmetry between the wheels, and
symmetry boundary conditions are applied. Loads are applied
independently to four locations. The loads are applied either
over a stiffener or between stiffeners, and at either the
outside edge or center of the girder. Both the vertical and
horizontal components are considered.

2.   Vessel Stiffeners

The stiffeners are modeled into the Ghosh-Wilson finite
element model. The stiffeners are also modeled into the STRAP
model for the lower penetrations and for some upper
penetrations, where applicable.

3.   Penetrations

Penetrations are analyzed using the STRAP finite element
program. Each penetration is examined to determine if it
requires analysis or if it could be represented by a similar
penetration that has been analyzed. The models are extended
sufficiently to account for localized stresses in the shell.

4.   Vessel Embedment

The embedment is analyzed to Subsection NE of the ASME Code,
Section III, by the finite element program SAP4 and the model
shown in <Figure 3.8-18>. Pull out of embedments is analyzed
in accordance with Division II of Section III. The 1-1/2 x 2

bar attached to the embedment foot is evaluated using the
rules of Section NF.

5.   Base Liner

The suppression pool liner is analyzed by considering a
portion of it to act as a beam-column. The type of loading,
relatively equal spacing between the circumferential tee
section supports, small spacing between the tee section
supports compared to their radii, and the fact that the liner
plate segments are supported along only two edges, allows the
idealization which considers portions of the liner plate to
act as a beam.

6.   Quencher Foundation

The analysis of the quencher foundation is performed in two
independent parts. The base plate, including the quencher
support pipe and stiffeners, and I-beam foundation are
analyzed separately. A finite element analysis is performed
to determine the stresses in the base plate, support pipe,
stiffeners, and weld between the quencher support pipe and the
base plate.

The I-beam foundation is analyzed by considering the structure
to act as a rigid body with rotation of the structure
resulting from vertical strains in the webs of the I-beam
supports. The assumption of rigid body motion is based upon
the relatively large structural members being used for the
relatively small foundation and upon the fact that the modulus
of elasticity for concrete is approximately 10 percent of the
value for steel.

7.   Equipment Hatch

      The equipment hatch assembly is designed to Subsection NE of
      the ASME Code, Section III, Class MC components, using the
      SUPERB finite element program. Due to geometry and load
      symmetry only a quarter model of the collar and barrel is
      used. The model is shown in <Figure 3.8-20>.

8.   Personnel Access Airlocks

      The upper and lower airlocks are designed to Subsection NE of
      the ASME Code, Section III, Class MC components. The barrel
      and collar are analyzed using the SUPERB finite element
      program. Due to the asymmetric application of some loadings a
      full 360 degree model is used as shown in <Figure 3.8-21>.
      The bulkhead and bulkhead door are analyzed using the STARDYNE
      computer program for static and dynamic loadings. Also, a
      frequency analysis is performed using STARDYNE to determine
      the natural frequency of the bulkhead and bulkhead door. The
      lowest natural frequency is 48 cps which is above the seismic
      cut off frequency of 33 cps, but below the zero period
      acceleration for the various dynamic loads of SRVD and LOCA.
      The design response accelerations for the load combinations
      with both seismic, SRVD and LOCA are obtained by adding the
      peak seismic response to 1.5 times the peak response for the
      SRVD or LOCA loads (e.g., net response = peak (seismic) +1.5 x
      peak (SRVD or LOCA)). The 1.5 factor is used to account for
      multimode participation. However, for conservatism, the
      maximum floor response accelerations are used to determine the
      seismic stresses. Hand calculations are used to determine the
      stress levels in other components of the airlocks such as the
      floor plate, test clamps, door arms, hinges, the pool swell
      wedge deflector, etc.

f. Buckling Analysis

1. Cylinder Buckling

The buckling investigation of the containment vessel cylinder consists of two approaches. First the shell and stiffeners are verified to be in compliance with all the requirements of Subsection NE-3133 of the ASME Code, Section III, and second a detailed buckling analysis is performed using equations from "Structural Analysis of Shells" (Reference 8). For general instability of the vessel, i.e., considering the stiffeners to buckle with the shell, the factor of safety against buckling is found to be approximately 240. The most likely mode of buckling is found to be that of buckling of the plate between the stiffeners. Considering the sections between the stiffeners as simply supported cylinders subjected to pressure and axial loadings, and using the equations from ASME Code, Section III, Division 1, Subsection NE, the lowest factor of safety against buckling is 1.24. The interaction for a shell subjected to both axial compression and external pressure is given in "Structural Analysis of Shells," (Reference 8) and shows a buckling factor of safety of 2.2.

As requested by NRC, <NUREG/CR-0793> was reviewed and compared with the method used here. The most critical loading combination in the evaluation for buckling is dead load plus seismic plus negative pressure. <NUREG/CR-0793> offered an alternative method of performing buckling analysis. It also stated that the Subsection NE-3133 of ASME III, Section NE, was conservative for uniform state of stress. Since the maximum compressive stress was assumed to be uniformly distributed over the shell, the result is conservative enough to satisfy design purpose.

2.   Dome Buckling

Buckling of the dome under internal pressure loading is
investigated using the equations from "Stability of Elastic
Systems" (Reference 9). The equations in Reference 9
calculate the local winkler type buckling stress in the
circumferential direction of an ellipsoidal head under
internal pressure. The factor of safety against buckling for
this type of loading is 4.25. Buckling of the dome due to
external loadings is investigated using the MARC computer
program. To account for construction tolerances and local
imperfections two computer models are used. The first model
shown in <Figure 3.8-22>, represents a "flat" dome
approximately a 6:1 ellipse, starting at the minimum tolerance
of -6 inches in the center and assuming a drop of 1-1/2 inch
in the sixth course of 90 inches, the fifth course of
170 inches assumes a circular arc of radius of 863.5 inches,
with the remainder an ellipse to the maximum radial tolerance
of +3 inches. The second model, also shown in <Figure 3.8-22>
accounts for circumferential imperfections. The shape of this
model was developed using the deflections from the stress
analysis for the point construction loads. These deflections
are normalized to the maximum tolerance of 6 inches, then
subtracted from the "perfect" dome to produce a "wavy" shape.
The flat dome model with the point construction loads produces
the largest deflections, therefore this is the critical case.
Since the MARC program does not follow large deflections with
no increase in load, as would occur in a snap-through buckling
condition, the dome is loaded to a selected load and then
incrementally unloaded. The Belleville washer model is used
to test the validity of MARC in this application. As shown in
<Figure 3.8-23>, the MARC load-deflection curve closely
follows the test and formula curves in the pre-buckled and
post-buckled regions and also the unloading curve, therefore,

it is valid for this application. Results of the detailed
analysis of the dome show that no large increase in deflection
occurs up to and above twice the applied construction loads
and the dome returns to its original shape as the load is
removed. This indicates that the dome gradually deflects
rather than buckles under external loading.

Results of an investigation of buckling in the knuckle region
due to application of the horizontal seismic loads show that
the vertical loads will control and buckling similar to that
described above will occur at the center of the dome before
buckling occurs at the spring line.

3.8.2.4.4   Design Methods

a.   General

The method used for the preliminary design is to establish the
minimum plate thickness required by ASME Code Subsections NE-3133
and NE-3324 for external and internal pressure loads. The model
thus established is then checked by analysis <Section 3.8.2.4.3>
for the stresses, strains and displacements which are checked
against the allowables of <Section 3.8.2.5>.

b.   Vital Subcompartments

The Containment Vessel does not contain any vital subcompartments.

c.   Reinforcing of Openings

Cutouts for locks, hatch and penetrations are reinforced in
accordance with the requirements of ASME Code Section III,
Subsection NE-3330.

3.8.2.4.5   Computer Programs

a.   Following is a brief description of the computer programs used in
     the analysis of the containment vessel, its appurtenances and
     attachments:

1.   STRAP by NNSD

     The Structural Analysis Program (STRAP) is a computer program
     developed by NNSD which employs the finite element method to
     perform static and dynamic analyses of large and complex
     structures. The static analysis portion of STRAP provides the
     capability to analyze linear, three dimensional structures to
     which arbitrary mechanical and thermal loads and displacement
     boundary conditions may be applied. Structural elements
     incorporated into the program include beam members, triangles
     and quadrilaterals which have both membrane and plate bending
     stiffnesses. Loads may be applied in the form of concentrated
     joint forces and moments, temperature changes, normal
     pressure, and concentrated, uniform or linear beam member
     loads. Program output includes joint displacements, beam
     forces and moments, beam stresses, triangle stresses, and
     quadrilateral stresses. STRAP has dynamic analysis routines
     for linear elastic structures undergoing small displacement
     response. Either free or forced vibrational response may be
     obtained for both conservative and non-conservative systems.
     In particular, a STRAP user may select any desired combination
     of the following dynamic analyses:

     (a) Natural frequency and mode shape analysis

     (b) Shock analysis

     (c) Steady-state forced response analysis

(d)   Transient forced response analysis

The natural frequency and mode shape analysis routine produces natural frequencies and mode shapes which characterize the free vibration of a structure. Other dynamics routines may be used to investigate forced structural vibration. Each forced response analysis routine (shock, steady-state and transient) produces free vibration information as well as forced system displacement response. In addition, the forced response routines calculate forces, moments and stresses experienced by the structure as it vibrates.

2.   SAP IV by Bathe, Wilson & Patterson

The Structural Analysis Program (SAP IV) is a computer program for static and dynamic analysis of linear structural systems using the finite element concept. SAP IV has a large element library which includes:

(a)   Three dimensional truss element

(b)   Three dimensional beam element

(c)   Plane stress, plane strain and axisymmetric elements

(d)   Three dimensional solid element

(e)   Thick shell element

(f)   Thin plate and shell element

(g)   Boundary element

(h)   Pipe element.

The input and output capabilities vary from element to element. SAP IV has an eigenvalue/eigenvector routine for recovering natural frequencies. Its dynamic analyses capabilities include:

(a) Dynamic response analysis for arbitrary time dependent loads using mode superposition,

(b) Dynamic response analysis for arbitrary time dependent loads using direct integration

(c) Response spectrum analysis.

3.   GHOSH-WILSON PROGRAM - ASHSD 2

A finite element method is presented for the dynamic analysis of complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The three dimensional axisymmetric continuum is represented either as axisymmetric thin shell or as a solid of revolution or as a combination of both. The axisymmetric shell is discretized as a series of frustrums of cones and the solid of revolution as triangular or quadrilateral "toroids" connected at their nodal point circles.

Hamilton's variational principle is used to derive the equations of motion for this discrete structure. This leads to a mass matrix, stiffness matrix and load vectors which are all consistent with the assumed displacement field. But to minimize computer storage and execution time a diagonal mass matrix has been assumed in writing the computer program. These equations of motion are solved numerically through the time domain either by direct integration or by mode

superposition. In both cases the numerical scheme adopted is the step-by-integration procedure. For an earthquake analysis, the response spectrum technique may be used to obtain approximate values of the maximum response quantities if detailed time history of the response is not desired. A different approach of solving the set of integral equations of motion instead of the differential equations is indicated.

This method of analysis is applied to various practical cases like nuclear reactor containment pressure vessel, response of tubes to moving pressure, etc. The above formulation is also applied to investigate structure-foundation interaction effects. The results of this investigation has led to a better understanding of the interaction phenomenon. Some recommendations are made for the design of nuclear reactor pressure vessel under seismic forces which takes into account the interaction effects of the foundation.

4.   AXSOL (Finite Element Stress Analysis of Axisymmetric Solids with Orthotropic, Temperature-Dependent Material Properties) by Edward L. Wilson and Robert M. Jones

The finite element method is applied to the determination of displacements and stresses in axisymmetric solids with orthotropic, temperature dependent properties. The continuous body is replaced by a system of ring elements with triangular or quadrilateral cross sections. Because of the choice in element shape and material properties, the procedure can be applied to bodies composed of many different materials of complex geometry. This analysis is used as a basis for the development of a general computer program which is capable of thermal stress analysis of axisymmetric solids. Mesh generating and temperature interpolating options facilitate program usage.

5.   GNATS by NNSD Computer Technical (Julian F. Cox and
     Douglas Carneal)

     GNATS is a computer code based on the finite-difference
     method. It has been developed to evaluate temperature
     distributions and heat flows in complex structures. This
     program has the capability to evaluate multidimensional heat
     flow with simultaneous conduction, convection and radiation
     heat transfer in the transient or steady-state condition. The
     computer code is tailored for use in design-oriented thermal
     analyses. Changes in structural configuration, materials,
     fluids, boundary conditions, etc., are easily implemented and
     compatibility between the structure and the enclosed thermal
     system or environment is readily achieved.

6.   MARC-CDC

     MARC-CDC is a general purpose finite element program designed
     for the linear and nonlinear analysis of structures in the
     static and dynamic regime. Its extensive element library
     makes it useful in elastic analysis and its broad coverage of
     the structural mechanics area makes it invaluable as a
     nonlinear analysis tool.

     The program may be used with a node-tying facility. This
     facility permits the tying together of different elements and
     the imposition of displacement constraints.

     The elastic-plastic and large displacement analysis is
     effected in a series of piecewise linear increments. Optional
     facilities enable the lowest eigenvalue for buckling to be
     obtained after each applied increment of load. This
     eigenvalue furnishes the factor which must be used to scale
     the next increment of load to cause collapse.

The dynamic analysis may be carried out by either the modal or the direct integration procedure. The dynamic analysis can be carried out with any of the nonlinear features in the program that make physical sense. Nonlinear dynamic analysis may be performed by use of the direct integration procedure in MARC.

7.   MRI/STARDYNE

The MRI/STARDYNE Analysis System consists of a series of compatible digital computer programs designed to analyze linear elastic structural models using the finite element method. The system encompasses the full range of static and dynamic analyses.

The STARDYNE system can be used to evaluate a wide variety of static and dynamic problems. The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal applied loads and/or prescribed displacements. Using the nodal mode technique, dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady-state harmonic, random, and shock spectra excitation types. Dynamic response results can be presented as structural deformations and/or internal member loads/stresses.

Data input and output formats, both numerical and graphical, have been prepared to enable the user to obtain a solution in the most logical and straight forward manner possible while keeping the required data input as simple and minimal as practical. The programmed mathematical operations in the matrix decomposition, the eigenvalue-eigenvector extraction and the error analysis, contain state-of-the-art innovations in the field of numerical analysis.

8. SUPERB

    The SUPERB computer program is based on the finite element
    method of structural analysis. It takes full advantage of
    advanced isoparametric elements. These elements are
    formulated by highly complex relationships (shape functions)
    and as a result can assume highly complex geometrics and
    represent high order displacement variations. The number of
    elements required to describe the geometry of a particular
    structure is thus drastically reduced when compared to the
    number of simple or "conventional" elements. Numerical
    accuracy is also enhanced. SUPERB contains the fundamental
    elements for axisymmetric, plane stress, plain strain, flat
    plate, curved shell, and solid representations. Linear,
    parabolic and cubic displacement functions are available along
    with suitable combinations of the same. Isoparametric
    elements are obtained by direct application of a three
    dimensional elasticity equation. SUPERB has node generation,
    element generation, pressure generation, force generation,
    temperature generation, and boundary condition generation
    capability which in turn reduces input preparation time.

9. ANSYS-CDC

    ANSYS-CDC is a large scale finite element program for
    structural, heat transfer and fluid-flow analysis. ANSYS
    performs linear and nonlinear elastic analysis of structures
    subjected to a wide variety of static and dynamic loading
    conditions. The program can consider the effects of
    plasticity, creep, swelling, and large deformations. The
    extensive element library includes the capability to model
    interaction effects such as gaps, friction, cables, and hooks.
    ANSYS is maintained by its writers to provide state-of-the-art

finite element analysis methods in a modular package which
allows new additions without changes to the previous features.

b. Verification of Programs

1. STRAP

This computer program employs accepted and proven displacement
functions and element stiffness matrices which are available
in the public domain. All matrix manipulation and solution
routines are also available in the public domain. Analysis
results from this program have been favorable compared with
the results from NASTRAN and STRUDL (both are public domain
programs). When one program is upgraded, a set of test
problems is analyzed to verify the program's accuracy prior to
its release for use.

2. SAP4

This is a public domain program and has been widely used since
its release.

3. Ghosh-Wilson (ASHSD2)

This is a public domain program.

4. AXSOL

This is a public domain finite element program (Aerospace
Report No. TR-0158 (53816-22)-1 and Air Force Report
No. BSD-TR-67-228). Test problems which are provided in the
above mentioned reports were analyzed to verify the programs
accuracy prior to its release for use.

5.   GNATS

This program is based on the finite difference method, which is a well established solution technique. Equations for developing the convection, conduction and radiation coefficients are available in the public domain. Test problems are used to verify the program's accuracy prior to its release for use.

6.   MARC

This is a public domain program and is a widely accepted finite element program. It's application to the dome buckling problem was verified by running the Belleville washer problem, as described in <Section 3.8.2.4.3.f.2>.

7.   SUPERB

This is a public domain program and has been generally used for the analysis of highly complex structures.

8.   STARDYNE

This is a public domain program widely used in the finite element analysis of structures.

9.   ANSYS-CDC

This is a public domain program and is a widely accepted finite element computer code. The program was written by Swanson Analysis Systems, Incorporated and is maintained on the CDC Cybernet System.

3.8.2.5   <u>Structural Acceptance Criteria</u>

Structural acceptance criteria for stresses, strains and deformations for the final design of the steel containment are in accordance with the ASME Code, Section III, Subsection NE for Class MC components and <Regulatory Guide 1.57> except that for concurrent loadings that result from flooding of the containment for core recovery following an accident and the vibratory motion of the OBE, stress limitations shall be:

a.   General membrane (P_m)

     1.   Tension   1.5 S_m

     2.   Compression   1.5 S_m

b.   Local membrane (P_L)

     Tension and compression   1.8 S_m or 1.5 S_y, whichever is greater

c.   Local membrane plus bending (P_L + P_b)

     Tension and compression   1.8 S_m or 1.5 S_y, whichever is greater

d.   For the flooded plus OBE condition, the meridional compressive allowable stress shall be either Subitem 1 or Subitem 2 below. However, the critical buckling stress may not exceed the proportional limit, which is found using a 0.1% offset. Vertical stiffeners may be utilized to limit the shell meridional compressive stress to the applicable allowable stresses.

1. For an unstiffened vessel, the meridional compressive allowable shall be the larger of either:

    (a) The meridional compressive allowable of NE-3133.6 of ASME Code Section III, Summer 1972 Addenda, or

    (b) $S_{LA} = 0.0625E \ (T/R) + 0.5 \ (\Delta\sigma_{cr})$; where $\Delta\sigma_{cr}$ is the term which accounts for the stiffening effect of internal pressure and is found in accordance with "The Stability of Thin-walled Unstiffened Circular Cylinders Under Axial Compression Including the Effects of Internal Pressure" by Harris, Suer, Skene, and Benjamin which appeared in Journal of the Aeronautical Sciences - August 1957.

    $$\Delta\sigma_{cr} = \frac{Et}{R} \ Y,$$

    $$Y = 0.01983 + 0.7886 \ x - 1.5272 \ x^2 + 1.5208 \ x^3 - 0.73323 \ x^4 + 0.13398 \ x^5,$$

    $$x = \frac{P}{E} \left(\frac{R}{t} \right)^2$$

2. For a stiffened vessel, the meridional compressive allowable stress shall be either:

    (a) $S_{LA} = 1.0 \ 0.5E \ 1.3(T/L)^2 + 0.125(T/R)$, or

    (b) $S_{LA} = 0.0625 \ E(T/R) + 0.5 \ (\Delta\sigma_{cr})$

3. In order to be able to consider the stiffening effect of internal pressure, $\Delta\sigma_{cr}$, stiffeners must be spaced more than $6\sqrt{Rt}$ apart.

The validity of the design is also demonstrated by the structural integrity test described in <Section 3.8.2.7.1>. <Table 3.8-11> provides a comparison of the containment vessel stresses for the governing load combinations at key locations with the appropriate ASME Section III, Division I, allowables.

3.8.2.6   Materials Quality Control and Special Construction Techniques

The free standing portion of the containment vessel is designed, fabricated and erected in accordance with the applicable requirements of ASME Code Section III for Class MC vessels. Material specifications and quality control provisions are in accordance with the requirements of the ASME Code and supplementary requirements, both summarized in this section. Organization, responsibilities and general provisions for the Quality Assurance Program are described in <Chapter 17>. Quality control provisions that are imposed for the structures are described herein.

3.8.2.6.1   Steel for Containment Vessel, Penetrations, Locks, and Hatch

3.8.2.6.1.1   Codes and Standards

The following codes and standards are used to establish the approved design documents governing the containment vessel and penetrations and related work. The date of a particular standard may vary for different items because of the difficulty in purchasing material to an outdated standard. Since the latest standards reflect industry practice used for fabrication and erection, it was permitted to use an updated standard where no unacceptable loss of quality would result.

a. ASME Boiler and Pressure Vessel Codes, 1974, including 1974 Summer addenda, except that the Class 2 pneumatic system shall use the 1977 ASME Code with Winter 1978 Addenda.

    1. Section II   Material Specifications Part A - Ferrous

    2. Section II   Material Specifications Part C - Welding, Rods, Electrodes, and Filler Materials

    3. Section III   Rules for Design Construction of Nuclear Power Plant Components, Class MC and NF (except that the spray header support brackets may meet the material design, fabrication, installation, and examination requirements of Subsection NF of Section III, including Winter 1977 addenda). The 1976 Winter Addenda, paragraph NE-4431 may be used for edge distance requirements when welding to attachment plates on the vessel.

    4. Section V   Non-Destructive Examination (except that testing of inaccessible welds is to Summer 1976 Addenda)

    5. Section VIII Pressure Vessels, Division 1

    6. Section IX   Welding Qualifications

b. American Welding Society (AWS) D1.1-85, "Structural Welding Code" with the following exceptions:

    1. AWS D1.1-85 is applicable to building structures and miscellaneous supports for HVAC duct, cable tray, conduit, etc., other than piping/I&C supports, designed and installed after October 31, 1986.

2.   AWS D1.1-72, including Revision 1, 1973, and Revision 2, 1974, is applicable to building structures designed and installed <u>before</u> October 31, 1986.

3.   AWS D1.1-75 is applicable to miscellaneous supports for HVAC duct, cable tray, conduit, etc., other than piping/I&C supports, designed and installed <u>before</u> October 31, 1986.

4.   Qualification of welding procedures and welders/operators per ASME Code Section IX may be substituted for AWS D1.1 Section 5 qualifications with the exception that ASME IX Welding Procedure Specifications for partial and complete penetration groove welds shall conform with the AWS D1.1 requirements for prequalified welded joints, specifically joint groove type/shape (i.e., groove angle and type, root opening and face and omission of backing material). Use of non-prequalified AWS D1.1 joints shall require procedure qualification.

c.   ASTM A 108-69 - Specification for Cold-Finished Carbon Steel Bars and Shafting.

d.   "Recommended Practice for Non-destructive Testing Personnel Qualification and Certification," SNT-TC-1A.

e.   <Regulatory Guide 1.31>, "Control of Stainless Steel Welding," Rev. 1, June 1973.

f.   <Regulatory Guide 1.57>, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," June 1973.

3.8.2.6.1.2   Materials

a.   Steel Plate for Containment Vessel

1.   Above Elevation 597'-10", the containment vessel plate material is SA-516, Grade 70, in accordance with the ASME Code, Section II, with supplementary requirements S5 of SA-20 for impact testing in accordance with NE-2300 of ASME Code, Section III.

2.   The lower 18 feet 6 inches of the containment vessel, are exposed to suppression pool water. The material for the lower 13 feet 6 inches is the same as that described in a.1. above, with a nominal 10 percent thickness of stainless steel cladding conforming to SA-240, Type 304, of the ASME Code, Section II. The thickness of the cladding is determined by the thickness of the nominal shell plate at the section considered. The material for the next 9 feet 6 inches is the same as that described in a.1., with a nominal 10 percent thickness of stainless steel cladding conforming to SA-240, Type 304 L, of the ASME Code, Section II. Stainless steel surface finish is hot rolled, annealed and pickled as specified in SA-480 of the ASME Code, Section II. All stainless clad plate meets the requirements of SA-264 of the ASME Code, Section II. All cladding is ultrasonically examined in accordance with SA-578 of the ASME Code, Section II.

3.   The lowest service metal temperature anticipated is 50°F. Plate material which is loaded during service in the through-thickness (short transverse) direction is in accordance with the additional supplementary requirements S1, S3 and S4.1 and S8 of SA-20. Areas of plate which are loaded in the through-thickness direction are:

(a) Polar crane girder

(b) Containment vessel spray system and air duct system
     brackets

(c) Containment vessel air handling and filter unit brackets

(d) Equipment access hatch cover monorail support brackets

(e) Large penetrations

b. Steel Plate Material for the Base Liner

The material for the base liner exposed to the suppression pool
water is the same as that in <Section 3.8.2.6.1.2.a.2>, above, for
the lower 13 feet 6 inches of the containment vessel. The material
for the base liner not exposed to the suppression pool water is the
same as that in <Section 3.8.2.6.1.2.a.1>, above.

c. Penetration Sleeves

Material for penetration sleeves in nominal sizes up to and
including 20 inches in diameter inclusive conforms to SA-333,
Grade 6, in accordance with ASME Code, Section II. Material for
penetration sleeves larger than 20 inches in diameter conforms to
the requirements of SA-155, Grade KCF70, Class 1. All sleeves are
impact tested in accordance with the requirements of ASTM E 23-72.
The test temperature is the same as that used for containment
vessel plate. Penetration sleeves submersed in the suppression
pool are fabricated from clad material conforming to the
requirements of <Section 3.8.2.6.1.2.a>, above, such that no carbon
steel is exposed to pool water.

d. Penetration Fittings

The material for penetration fittings conforms to the requirements
of SA-333, Grade 6, of ASME Code, Section II or SA-115,
Grade KCF70, Class 1.

e. Personnel Locks and Equipment Hatch

Material for the personnel access air locks, and the equipment
access hatch:

2. Cold finished bars for shafts which penetrate the air lock
    door bulkheads, meeting ASME-ASTM SA-479, Type 304.

3. Plate meeting ASME-ASTM SA-516, Grade 70, and
    Subsection NB-2300 of ASME Code, Section III, or EQUAL. The
    lowest service metal temperature (LSMT) is 50°F both within
    containment and outside the shield building except that for
    the equipment hatch flanges the LSMT is 55°F.

f. Containment Vessel Attachments Including the Polar Crane Girder and
Ring Stiffeners

Attachments made directly to the containment vessel conform to
NE-4430 of ASME Code, Section III and:

1. Plate material, conforms to SA-516, Grade 70, of the ASME
    Code, Section II, with the supplementary requirement for
    impact testing in accordance with NE-2300 of the ASME Code,
    Section III.

2. Rolled structural sections or plate not welded directly to the containment vessel conform to SA-36 of the ASME Code, Section II, and meet the requirements of NE-4431 of the ASME Code, Section III.

g. Welding Materials

All welding filler metals and fluxes used in the fabrication of the containment vessel and its components conform to the requirements of NE-2400 of Section III of the ASME Code.

1. Carbon Steel

   (a) Covered carbon steel electrodes conform to ASME SFA-5.1 E7015, E7016 or E7018, tested in accordance with NE-2431.2 "Standard Test Requirements."

   (b) Submerged arc electrode wire and flux, tungsten inert gas rods and gas metal arc electrode wire for carbon steel welding conform with all the requirements of SFA-5.17 or SFA-5.23, and SFA-5.18 or SFA-5.20, respectively, tested in accordance with NE-2431.1.

2. Stainless Steel

   (a) Stainless steel filler metals meet the ferrite requirements of <Regulatory Guide 1.31>. All stainless steel filler metals are tested in accordance with NE-2432 "Chemical Analysis of Filler Metal or Weld Deposits."

   (b) Covered stainless steel electrodes conform with the requirements of ASME SFA-5.4, classification E308-15, E308-16, E308L-15, E308L-16, E309-15, E309-16, E309L-5, or E309L-16. Classification E309 electrodes contain less

than .08 percent carbon; classification E309L electrodes contain no more than .04 percent carbon.

(c) Stainless steel filler metal for gas tungsten arc welding or for gas metal arc welding conform with ASME SFA-5.9, classifications ER308, ER308L, ER309, ER309L, and the high silicon version of the classification including ER308 Si, ER308L Si, ER309 Si, and ER309L Si. Classification ER309 contains less than .08 percent carbon. Type ER309L filler metal meets the requirements for Type ER309 of ASME SFA-5.9, except that the carbon content does not exceed 0.03 percent.

(d) Stainless steel electrodes for flux cored arc welding conform with all the requirements of SFA-5.22, classifications E308T-X, E308LT-X, E308LT-3, E309T-X, E309LT-X, or E309LT-3.

3.8.2.6.1.3 Quality Control

a. Material Certification and Traceability

1. All materials in the containment vessel and penetrations have certified mill test reports and certification of specifications. The weld materials have certified materials test reports.

2. By use of the markings or "as-built" drawings each part of the containment vessel pressure boundaries is traceable to the actual mill test reports.

b.   Welding Requirements

The qualification of welders and weld procedures is in accordance
with the requirements of ASME Section IX, and either ASME
Section IX or AWS D1.1 for non-ASME jurisdictional boundary welds.

The welds are designed and fabricated to ASME Section III,
Class MC, and qualified under paragraph NE-4300. The repair of
welds is in accordance with paragraph NE-4400 of ASME Section III.
Repairs made after September 9, 1996 comply with the requirements
of ASME Section XI.

c.   Weld Non-Destructive Examination

1.   Non-destructive examination of welds is in accordance with
      NE-5000 of the ASME Code, Section III.

2.   Non-ASME Code embedment welds are examined to Section NE-5000
      of the ASME Code.

3.   Full penetration attachment welds are examined by ultrasonic
      or radiographic methods. Where ultrasonic or radiographic
      testing can not be used, the root layer, every 3/8 inch of
      weld metal and the final surface are inspected by the magnetic
      particle method.

4.   Fillet welds are examined by the magnetic particle or liquid
      penetrant methods, whichever is applicable.

5.   Longitudinal butt welds in penetration sleeves are examined in
      accordance with Section NE-5211 of the ASME Code, Section III.

6.   Magnetic particle acceptance standards are in accordance with
      NE-5340 of the ASME Code, Section III.

7.   Liquid penetrant acceptance standards are in accordance with
      NE-5350 of the ASME Code, Section III.

8.   Ultrasonic acceptance standards are in accordance with NE-5330
      of the ASME Code, Section III.

9.   Radiographic acceptance standards are in accordance with
      NE-5320 of the ASME Code, Section III.

d.   Weld Leak Testing

1.   During the structural integrity test described in
      <Section 3.8.2.7.1>, an inspection is made of all accessible
      welds as required by ASME Section III, Division 1,
      Subsection NE 6315. This inspection will use an ultrasonic
      leak locator along with pressure readings to identify, locate
      and mark any leaks. Leaks detected in the above tests are
      repaired in accordance with NE-4400 of ASME Code, Section III.

2.   All welds on the base liner are covered by steel test
      channels, angles or plate located on the inside face of the
      base liner. The test members are segmented and interconnected
      so that each of the areas can be tested at one location.
      After those welds provided with pressure test systems are
      examined, the pressure test system is installed and the welds
      are leak tested. Where pressure test systems are installed,
      the system attachment weld is tested for leak tightness by
      pressurization to at least 25 psig in accordance with T1020 of
      ASME Section V. If the indicated loss of test pressure for a
      2 hour period exceeds 0.4 psig after compensation for
      temperature and humidity changes, attachment welds are
      soap-bubble tested in accordance with T1030 of ASME Code,
      Section V. Areas of weld that are rejected are repaired and
      reexamined. If a pressure loss still occurs after the

pressure test system welds are found to be acceptable, the
pressure test system is removed and the vessel weld is
reexamined, repaired as required and retested. All
requirements for examination of inaccessible welds on the
containment vessel as provided by Section NE-5211.2 of ASME
Code, Section III, are followed.

3.8.2.6.1.4   Tolerances

Tolerances for the containment vessel comply with the requirements of
ASME Code, Section III, Subsection NE, with the exception that the
diametrical tolerances of 1 percent noted in NE-4221.1 of ASME Code
Section III are reduced to 1/2 of 1 percent.

3.8.2.6.1.5   Special Construction Techniques

No special construction techniques are used in erecting the containment
vessel.

3.8.2.6.1.6   Corrosion Protection

Corrosion protection for the carbon steel plate in the suppression pool
area (to three feet above maximum water level) is provided by clad
stainless material applied to the cylindrical wall and foundation mat
surfaces. <Section 3.8.2.6.1.2.a.2> discusses the materials used.

3.8.2.7   Testing and Inservice Inspection Requirements

3.8.2.7.1   Structural Integrity Test

In addition to the local structural integrity testing of components
during fabrication and erection, the containment vessel and the annulus
concrete are structural integrity tested. The test is conducted in
accordance with ASME Code Case N-258 with proposed Revision 1 which

references in Section III, Division 1, Subsection NE 6000, and Division 2, Subsection CC 6000, for pneumatic tests. The Code year for this test is to be the 1980 Edition with the Summer 1981 Addenda. This test is conducted at a pressure of 115 percent of the design pressure (1.15 x 15 = 17.25 psig).

3.8.2.7.2   Integrated Leak Rate Testing

The leakage of the containment vessel is checked prior to startup and is periodically checked during the life of the plant. Details of these tests are given in <Section 6.2.6>.

3.8.2.7.3   In-Plant Safety/Relief Valve Testing

This section is not applicable to Perry. The conditions at Perry are similar in regards to SRV discharge performance with the conditions in plants previously tested. This eliminates the need for any new tests at Perry. (Letter from the NRC dated March 1, 1983.)

3.8.3   INTERNAL CONCRETE AND STEEL STRUCTURES

The following are the principal structures inside containment:

a.   Reactor pressure vessel pedestal and biological shield wall.

b.   Weir wall and weir wall support mat.

c.   Drywell walls, top slab, head, and fuel transfer and storage pools.

d.   Structural steel frames and floors.

<Figure 3.8-1> provides the general arrangement of these structures.

3.8.3.1   Description of Internal Concrete and Steel Structures

3.8.3.1.1   Reactor Pressure Vessel Pedestal and Biological Shield
                 Wall

The reactor pressure vessel pedestal and biological shield wall are
concrete filled double steel walled cylinders as shown in
<Figure 3.8-24>.

The inner and outer plates of the biological shield wall are 1 inch
thick SA 516, Grade 70, steel plate. The annular space between the
2 cylinders is stiffened by 24 - 1 inch thick vertical webs and by
miscellaneous horizontal plates. The remainder of the annular volume is
filled by concrete with a density of at least 140 pcf. Flow diverters,
as shown in <Figure 3.8-25>, are fitted to the recirculation suction
line penetrations so that the flow from a reactor pressure vessel safe
end nozzle break is directed into the drywell volume with minimal
pressurization of the annulus between the biological shield wall and the
reactor pressure vessel.

A structural steel frame is cantilevered from the biological shield wall
to provide support for the recirculation pump and motor. Pipe whip
restraint and support structures, of both the structural steel and box
girder type, are also cantilevered from the biological shield wall. The
radial beams for the platform at Elevation 599'-9" are supported from
the biological shield wall and at the weir wall.

The reactor pressure vessel pedestal is cylindrical in shape with an
inside diameter of 19 feet 7 inches with an overall thickness of 6 feet
0 inches and is 28 feet 10 inches high. It is founded on the reactor
building complex mat at Elevation 574'-10". Anchor details are shown in
<Figure 3.8-26>. Construction of the pedestal is similar to that of the
biological shield wall with 1 inch thick, stiffened, SA 516, Grade 70,
steel plate but the annulus is filled with normal weight 3,000 psi

concrete. The biological shield wall is supported on the pedestal at Elevation 603'-8". The outside plates of the biological shield wall and the pedestal are continuous. A stiffened steel ring beam in the pedestal distributes load to both inner and outer skin. The pedestal is haunched at the top to provide bearing area for the reactor vessel sole plate, and the haunch details are as shown in <Figure 3.8-26>. The reactor vessel skirt is 20 feet 11 inches outside diameter with a 6 inch thick flange plate bolted to the pedestal by means of 2 rows of 60 bolts each. Rectangular openings in the pedestal are provided for Control Rod Drive (CRD) mechanism operation, maintenance, removal, and replacement.

3.8.3.1.2   Weir Wall and Weir Wall Support Mat

The weir wall and integral weir wall support mat are 68 feet 8 inches in outer diameter. The weir wall is 24 feet 2 inches high and 1 foot 6 inches thick while the weir wall support mat is 8 feet 8 inches thick. These structures are constructed from 3,000 psi concrete reinforced as shown in <Figure 3.8-27>, with ASTM A 615, Grade 60, rebar. The outer face of the weir wall is lined with Type 304 stainless steel for the lower 13 feet 6 inches and with Type 304-L stainless steel for the upper 10 feet 5 inches.

3.8.3.1.3   Drywell Walls, Top Slab, Head, and Fuel Transfer and
               Storage Pools

The drywell wall is generally a right, vertical cylinder. It is 83 feet 0 inches outside diameter, 85 feet 9 inches high and 5 feet 0 inches thick. The drywell wall is subdivided into two regions which have different construction and design methods.

The lower 26 feet 2 inches of the drywell is the vent region. The main suppression pool area in the containment vessel is connected to the drywell by 120 vents. The vent sleeves are 28 inch outside diameter, 1/4 inch thick, stainless steel tubes located in 3 rows of 40 vents.

Vents are constructed from ASTM A 240, Grade 304, stainless steel. The vent structure is a steel and concrete composite construction which consists of two concentric cylinders fabricated from 1 inch thick ASTM A 516, Grade 70, steel with a ten percent Type 304 stainless steel cladding. The annulus between the cylinders is stiffened vertically by radial steel plates and is filled with 5,000 psi concrete. The steel plates are designed to carry all membrane tensile forces in this region. At Elevation 574-10", the vent region is anchored to the containment base mat by means of vertical tension ties in the form of anchor bars made of ASTM A 537 CL.2 steel for the transfer of any uplift forces to the base mat. The anchor plate and stiffeners at the bottom of the anchor bars are sized for the capacity of the anchor bars. There are a total of 144 anchor bars on each face of the drywell. Anchor bars are 1-1/2" by 8" and 1-1/2" by 4" on the outside and inside faces of the drywell wall, respectively. The maximum forces in the anchors occur under the abnormal/extreme environmental load combinations as shown on in <Table 3.8-3> and result in a maximum tensile stress in the anchor bars of 36.2 Ksi (outside face) and 47.9 Ksi (inside face), as compared to an allowable of 54 Ksi. Anchor bars are embedded 6'0" into the reactor building mat and develop the required anchor capacity.

The entire drywell cylindrical wall is recessed into the reactor building foundation and provides a continuous shear key for resisting radial/transverse shear forces by direct bearing of the drywell wall base on the foundation mat. Radially oriented structural tees have been provided to resist the tangential/in-plane shear forces by bearing on the foundation mat concrete. A total of 81 of these structural tees have been provided for the drywell structure and are welded to both faceplates of the drywell vent structure. Anchor details are as shown in <Figure 3.8-28>.

The upper drywell wall region is designed as a reinforced concrete cylinder connected to the lower vent region by cadwelding all vertical and diagonal rebars to the ring girder at Elevation 600'-10", as shown

in <Figure 3.8-28>. The main reinforcement consists of No. 18 vertical
rebars spaced radially at a centric angle of 2° 30', No. 18 hoop rebars
spaced vertically 12 inches center to center, and diagonal No. 14 rebars
inclined approximately 45° to the vertical rebars and running in both
directions. These rebars are provided on both faces of the cylinder and
are spaced at 24 inches center to center. On the outside face of the
cylinder, additional No. 11 rebars are placed midway between the No. 18
vertical rebars and extended to Elevation 612'-1". The upper drywell
wall extends up to Elevation 660'-7" where it is integrally connected to
the 4 feet 0 inch thick drywell top slab. A typical section of the
drywell reinforcement is shown in <Figure 3.8-29>.

Above the vent region, the inside face of the drywell is formed with a
1/4 inch thick steel plate of ASTM A 285, Grade A, material which is
stiffened vertically by 2 inch x 3 inch x 1/4 inch angles spaced at
approximately 1 foot 3 inches and horizontally by stiffener rings of
3 inch x 4 inch x 1/4 inch angles spaced at about 5 feet 0 inches center
to center. The stiffener angles are ASTM A36 material. This steel
liner is very conservatively not considered in the design as
contributing to structural strength or leak tightness of the drywell.
For further discussion on drywell liner leak tightness, see USAR
<Section 3.8.3.3.8> and <Section 6.2.1.1.5.4>.

The drywell top slab is a flat, horizontal, circular, reinforced
concrete slab. It contains a central circular opening of 31 feet
11-1/2 inch diameter which is closed by the drywell head. The
configuration and seal details of the 14 feet 9-1/4 inch deep steel
ellipsoidal drywell head are shown in <Figure 3.8-30>. The top slab is
stiffened by two longitudinal reinforced concrete walls which are part
of the upper pool wall system. <Figure 3.8-29> shows the typical
reinforcement in a section of the drywell wall and top slab. A plan
view is provided as <Figure 3.8-87>.

The personnel access air lock has an outside diameter of 9 feet 8 inches and is located at the 599'-9" elevation floor to provide access to the drywell. For large pieces of equipment, an 11 foot 0 inch square clear opening, bolted, double gasket sealed, equipment access hatch is provided at Elevation 599'-9". For details of the lock and hatch, see <Figure 3.8-31> and <Figure 3.8-32>. To facilitate the movement of equipment from the containment vessel equipment hatch to inside the drywell, two 30 ton monorail systems are provided to service the hatch; one inside the drywell and one inside the containment (outside of the drywell). The personnel air lock and the equipment hatch are integrally connected by full penetration welds to steel frames designed to act as end anchorage for all of the drywell wall reinforcement in the vicinity of the lock and hatch. Additional reinforcement is provided to take account of stress concentrations around these large openings as shown in <Figure 3.8-88> and <Figure 3.8-89>.

Penetrations through the drywell wall for piping and electrical systems are of the single barrier leak tight type, as shown in <Figure 3.8-6> and <Figure 3.8-7>. The main steam lines are anchored at the drywell and are provided with guard pipes through to the isolation valves outside containment, as shown in <Figure 3.8-6>. The main steam line penetration anchor sleeve detail is shown in <Figure 3.8-104>.

A number of major reinforced concrete compartments are attached to the exterior of the drywell wall. The fuel storage and transfer pools are rectangular shaped compartmented structures, constructed of reinforced concrete lined with stainless steel, approximately 44 feet wide by 102 feet long. This structure is supported on the drywell and the drywell top slab.

3.8.3.1.4   Structural Steel Floors and Framing

The structural steel used for floors and framing of the reactor building complex conforms to the requirements of ASTM A 36.

The structural steel framing system used between the containment vessel and outside of the drywell wall consists of 17 support columns located on a circle at approximately 57 feet 6 inches from centerline of the RPV, radial beams which connect the columns to the drywell wall and chord beams which tie the columns and radial beams together. There is no structural tie between the containment vessel and the structural steel floors or columns. These floors are used for the support of equipment, cable trays, pipe runs, walkways, stairs, platforms, etc.

Pipe whip restraint structures are generally independent of the floor steel and are discussed in <Section 3.6>.

All floors and frames have been designed to withstand normal operating loads, including dead loads, live loads, equipment loads, pipe, cable tray and duct loads, and temperature loads as well as seismic and SRVD loads. Floors, frames and columns are also designed to withstand loads due to the dynamic effects of the pool swell phenomena and other accident loads as identified in <Section 3.8.3.3>.

Structural steel floors and frames at various elevations are shown in <Figure 3.8-33>, <Figure 3.8-34>, <Figure 3.8-35>, <Figure 3.8-36>, through <Figure 3.8-37> and described below:

a.   Floors Between the Drywell and Containment - Outside Drywell

     1.   Floor at Elevation 599'-9"

          Located in the annular space between the drywell and containment vessel, the floor is only partially covered with grating and provides access to the drywell personnel access airlock. No equipment and only a small amount of piping is supported off this floor.

2.   Floor at Elevation 620'-6"

     Located in the annular space between the drywell and
     containment vessel this floor provides access to the equipment
     access hatch. This floor is covered by grating and provides
     support for equipment including the CRD hydraulic control
     units and various instrument and control panels.

3.   Floor at Elevation 633'-0"

     This is not a general floor area, but two maintenance
     platforms: one between Azimuth 54° and Azimuth 121° 30' and
     the other between Azimuth 234° and Azimuth 306°. These
     platforms are provided to allow access to the hydraulic
     control unit valves.

4.   Floor at Elevation 642'-0"

     Located in the annular space between the drywell and
     containment vessel, this floor is covered with 3/4 inch
     checkered floor plate between Azimuth 251° and Azimuth 306°
     and covered with galvanized grating elsewhere. This floor
     supports the ATWS pumps and boron solution liquid storage tank
     as well as HVAC equipment.

5.   Floor and Structural Steel Framing at Elevation 664'-7"

     Located in the annular space between the drywell and
     containment vessel, this floor is covered with galvanized
     grating where floor area is required. Structural steel
     framing located approximately between Azimuth 216° and
     Azimuth 306° is used to support cable trays, ducts and pipes.

6. Operating Floor at Elevation 689'-6"

    Located in the annular space between the drywell and
    containment vessel, this floor is covered with 3/4 inch
    checkered floor plate between Azimuth 251° and Azimuth 289°,
    and is covered with galvanized grating elsewhere. This floor
    is used to support equipment and the drywell head during
    shutdown.

b. Floors Between the Biological Shield Wall and Drywell Wall - Inside
    Drywell

    1. The floor at Elevation 599'-9" is covered with galvanized
        grating and is used for access.

    2. The floor at Elevation 630'-1" is covered with galvanized
        grating. A portion of this floor is part of the pipe
        restraint structure. This floor is also used to support
        equipment and for access. The radial beams of this floor are
        supported from the biological shield wall and have sliding
        joints on the drywell wall.

    3. Other small platforms and frames in this area are used mainly
        to support light equipment and for access.

3.8.3.2   Applicable Codes, Standards and Specifications to the
            Design

Structural analysis, and design for the interior concrete and steel
structures conform to the codes, standards and specifications listed
below unless noted otherwise. Where contradictions occur among codes,
the more stringent were applied.

3.8.3.2.1   Codes

a.   ACI 301-72 "Specification for Structural Concrete for Buildings,"
      American Concrete Institute.

      All chapters of ACI 301 are applicable except Chapter 15 which
      deals with requirements for prestressed concrete.

b.   ACI 318-71 "Building Code Requirements for Reinforced Concrete,"
      American Concrete Institute.

      <u>Chapters Used</u>

      1   General Requirements
      2   Definitions
      7   Details of Reinforcement
      8   Analysis and Design - General
      9[(2)]   Strength and Serviceability
      10   Flexure and Axial Loads
      11   Shear and Torsion
      12   Development of Reinforcement

      <u>Chapters Used</u>

      14   Walls
      17   Composite Concrete Flexual Members
      19   Shells and Folded Plate Structures
      Appendix A
      Appendix B

      <u>Chapters Not Used</u>

      3[(1)]   Materials
      4[(1)]   Concrete Quality

5[1]   Mixing and Placing Concrete

6[1]   Formwork, Embedded Pipes and Construction Joints

13[3]   Slab Systems with Panels

15[4]   Footings

16[5]   Precast Concrete

18[6]   Prestressed Concrete

20[7]   Strength Evaluation of Existing Structures

NOTES:

[1] CI 301-72 was used to define requirements in this area as modified by <Section 3.8.3.6>.

[2] Both ACI 318 and ASME Section III, Division 2, were used in determining load combinations <Section 3.8.3.3.3>.

[3] Slab system with panels are not used on the interior structures.

[4] A foundation mat is used rather than footings.

[5] Precast concrete is not used in the interior structures.

[6] Prestressed concrete is not used in the interior structures.

[7] The interior structures contain no existing structures.

c. ASME Section III, Division 2, "Proposed Standard Code for Concrete Reactor Vessels and Containments."

d. Ohio Building Code 1970 Edition, Chapter BB-33, "Reinforced and Plain Concrete, Reinforced Gypsum Concrete and Reinforced Masonry," Board of Building Standards Department of Institutional Relations, State of Ohio.

Sections Used

BB-33-01 Definitions

BB-33-02 Abbreviations

BB-33-03 Accepted engineering practice and approved standards

Sections Used (Continued)

BB-33-07 Reinforced concrete - general design and construction

Sections Not Used

BB-33-04[1]   Reinforced concrete materials

BB-33-05[1]   Concrete, classification and quality

BB-33-06[1]   Concrete, ready mixed

BB-33-08[2]   Reinforced concrete, supervision

BB-33-09 to

BB-33-13[3]   Plain concrete

BB-33-14[4]   Reinforced gypsum concrete precast

BB-33-15[4]   Reinforced gypsum concrete design and construction

Sections Not Used

BB-33-16 to

BB-33-20[5]   Reinforced masonry

NOTES:

[1] ACI 301-72 is used in these areas as modified by
    <Section 3.8.3.6>.

[2] Not applicable to this design.

[3] Plain concrete is not used in this structure.

[4] Reinforced gypsum is not used in this structure.

[5] Reinforced masonry is not used in this structure.

e.   CRSI "Manual of Standard Practice"

f.   ASME-1974 Boiler and Pressure Vessel Code (ASME Code) including the
     Summer 1975 Addenda except as follows:

1. Material may be purchased in accordance with the issue of the code in effect at the time the material order is placed.

2. Personnel access airlocks shall include the Summer 1976 Addenda.

3. Code Case N-192 is acceptable for flexible pressure hose on the personnel access airlocks when designed in conjunction with the 1977 Edition with the Winter 1978 Addenda.

4. Pneumatic valves and associated systems shall be designed and purchased in accordance with the 1977 Edition with the Winter 1978 Addenda.

g. American Welding Society (AWS) D1.1-85, "Structural Welding Code" with the following exceptions:

1. AWS D1.1-85 is applicable to building structures and miscellaneous supports for HVAC duct, cable tray, conduit, etc., other than piping/I&C supports, designed and installed <u>after</u> October 31, 1986.

2. AWS D1.1-72, including Revision 1, 1973, and Revision 2, 1974, is applicable to building structures designed and installed <u>before</u> October 31, 1986.

3. AWS D1.1-75 is applicable to miscellaneous supports for HVAC duct, cable tray, conduit, etc., other than piping/I&C supports, designed and installed <u>before</u> October 31, 1986.

4. Qualification of welding procedures and welders/operators per ASME Code Section IX may be substituted for AWS D1.1 Section 5 qualifications with the exception that ASME IX Welding Procedure Specifications for partial and complete penetration

groove welds shall conform with the AWS D1.1 requirements for prequalified welded joints, specifically joint groove type/shape (i.e., groove angle and type, root opening and face and omission of backing material). Use of non-prequalified AWS D1.1 joints shall require procedure qualification.

h.   AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

All parts of the AISC Code are applicable except as amended by "Structural Design Criteria for Category I Structures Outside Containment" (Reference 1).

3.8.3.2.2   Standards

a.   ASTM Standards: Applicable ASTM standards are discussed in the following Sections: The date of a particular standard may vary for different items because of the difficulty in purchasing material to an outdated standard. Since the latest ASTM standards reflect industry practice used for fabrication and erection, it was permitted to use an updated standard where no unacceptable loss of quality would result.

1.   Concrete: <Section 3.8.3.6.1>

2.   Reinforcing Steel: <Section 3.8.1.6.2>

3.   Cadweld Splices: <Section 3.8.1.6.3>

4.   Steel Plate: <Section 3.8.2.7.1> and <Section 3.8.1.6.4>

5.   Structural Steel: <Section 3.8.3.6.5>

6.   Stainless Steel: <Section 3.8.3.6.6>

3.8.3.2.3   Applicable Regulatory Guides to Design

Regulatory guides pertaining to seismic design classification and
seismic design are referenced in <Section 3.2> and <Section 3.7>,
respectively.

1. <Regulatory Guide 1.10>, "Mechanical (Cadweld) Splices in
    Reinforcing Bars of Seismic Category I Concrete Structures." This
    guide was used with modifications specified in <Section 3.8.1.6.3>.

2. <Regulatory Guide 1.15>, "Testing of Reinforcing Bars for
    Category I Concrete Structures." This guide was used with
    modifications as set forth in <Section 3.8.1.6.2>.

3. <Regulatory Guide 1.31>, "Control of Stainless Steel Welding."

4. <Regulatory Guide 1.55>, "Concrete Placement in Category I
    Structures."

5. <Regulatory Guide 1.57>, "Design Limits and Loading Combinations
    for Metal Primary Reactor Containment System Components."

6. <Regulatory Guide 1.69>, "Concrete Radiation Shields for Nuclear
    Power Plants."

7. <Regulatory Guide 1.71>, "Welder Qualification for Areas of Limited
    Accessibility."

8. <Regulatory Guide 1.142> (April 1978), "Safety-Related Concrete
    Structures For Nuclear Power Plants (Other Than Reactor Vessels and
    Containments)." Also see <Section 3.8.1.2.2.b.4>.

3.8.3.2.4   Material and Construction Specifications

Specifications prepared by the engineer cover the areas related to the design and construction of the interior structures. The specifications emphasize important points of industry standards for these structures and reduce the options that would otherwise be permitted. Unless noted otherwise, these specifications do not deviate from the applicable industry standards. The following areas are addressed:

a.   Fabrication and erection of structural steel.

b.   Fabrication of embedded steel.

c.   Fabrication and erection of steel plate structures inside
      containment.

d.   Fabrication and erection of stainless steel liners for fuel pools.

e.   Concrete supply.

f.   Placement of structural concrete.

g.   Fabrication and placing of reinforcing steel and embedded items.

h.   Supply and installation of waterproofing and waterstops.

3.8.3.3   Loads and Load Combinations

3.8.3.3.1   Loads Used in the Design

a.   Dead Load (symbol D)

      The dead load considered in the design is the weight of all
      concrete (150 lb/cu ft for normal concrete, 225 lb/cu ft for

heavyweight concrete) and/or structural steel (490 lbs/cu ft) plus any permanent attachments including known equipment loads, cable trays and pipe runs of less than 18 inches in diameter.

b. Live Load (symbol L)

The live load considered in the design is the weight of any equipment or system not permanently attached; pipe runs 18 inches in diameter or larger; plus an allowance for transient loads during construction, operation or maintenance.

A uniform allowance of 100 lb/sq ft is made for stairs and walkways.

c. Seismic Loads (symbols F_{eqo} for OBE and F_{eqs} for SSE)

Seismic loads are the forces imparted on the structure by ground accelerations during a postulated OBE or SSE. For further details see <Section 3.7>.

d. Normal Operating Temperature Induced Loads (symbol T_o)

Temperature induced forces during normal operation are evaluated for the various sections of the interior structure based on the most adverse temperature gradients predicted by normal plant operation. Typical transients are shown in <Figure 3.8-38>, <Figure 3.8-40> and <Figure 3.8-41>.

	Outside Drywell	Inside drywell
minimum	60°F	60°F
normal	95°F	135°F
maximum	105°F	150°F (local)
		145°F (bulk air)

e. Accident Pressure (symbol P_a) and Accident Temperature (symbol T_a) Induced Loads

The interior structure components are designed for the worst postulated accident pressure and accident temperature conditions. These conditions vary with different components and are dependent on the postulated accident. A discussion of the accidents investigated is found in <Section 6.2> and <Appendix 3A> and <Appendix 3B>. Bubble pressure, bulk swell and froth swell loads, drag pressure, and other pool swell related loads are considered as abnormal accident pressure loads. The pool swell induced loads and associated accident pressures are combined and take into consideration time dependent behavior. In summary, the design conditions for the interior structures are discussed below. Both transient and steady-state temperature profiles were considered in the design. The thermal effects and heat transfer analysis are discussed in more detail in <Section 3.8.3.3.6> and <Section 3.8.3.4.2.a.4>, respectively.

1. Main Steam Line or Recirculation Line Break

    (a) The design condition of an outward acting maximum pressure of 30 psig exists at approximately 1.8 seconds after the major break. There is no thermal response to the temperature differential predicted for this short time.

    (b) An inward acting pressure of 21 psig at five minutes after the accident.

    (c) A 15 psig containment pressure and 19.3 psig drywell pressure at 3 hours after the break occurs, with minimum ECCS operating.

(d) Platforms in the annulus between the drywell and
     containment are designed to withstand the upward flow
     effects of noncondensible air and a mixture of air, steam
     and water due to pool swell. The pool dynamic loads for
     the interior structures are discussed in <Appendix 3A>
     and <Appendix 3B>.

2. Small Line Break Inside The Drywell

The design pressure due to the pressure build-up inside the
drywell due to a small line break is 3.0 psig and corresponds
to the pressure at which the steam-air-mixture commences
bubbling through the suppression pool.

3. Small Line Break Inside The Containment Vessel (Outside
    Drywell)

Vacuum breakers are provided to limit the maximum differential
pressure to less than 0.8 psi and so prevent overtopping of
the drywell weir wall. This is not a governing design
criterion because overtopping of the drywell weir wall does
not constitute a safety concern.

The design pressure for high energy pipe breaks within each
subcompartment is discussed in <Section 6.2.1.2>.

f. Penetration Loads (symbols R_o, R_a, Y_r, Y_j, and Y_m)

Penetration loads are defined in <Section 3.8.1.3.1.j> and are in
accordance with the definitions of "Structural Design Criteria for
Category I Structures Outside Containment" (Reference 1).

These are treated as local loads on the individual structures and
are developed as the result of interaction of penetrations and the

reinforced concrete wall. For the drywell wall, all penetrations,
except the feedwater line which is a guide only, are designed as
pipe restraints for axial, moment and torsional loads.

The interior structures are designed to take the full impact of jet
forces, without failure, to the criteria of <Section 3.6>. Where
vital components cannot withstand the direct impact of a jet,
barriers are provided in accordance with <Section 3.6> criteria.

g.   Water Pressure (symbol G)

The design bases for groundwater loading conditions are described
in <Section 2.4.13>, <Section 2.5.4>, and <Section 3.8.5.3.5>.
Water pressure includes the weight of water and associated
hydrostatic pressure in the upper fuel pool and suppression pool.

For postaccident core recovery, the containment is flooded to a
depth of 66 feet, and there will be no differential pressure on the
interior structures due to this water.

h.   Missile Loads (symbol M)

The sources of postulated interior missiles and the necessary
protective barriers are identified and discussed in <Section 3.5>.
The missile loads predicted from this analysis are considered in
the design of each structure.

i.   Preoperational Proof Pressure (symbol P_t) and Leak Rate Tests

A preoperational proof pressure test of the drywell will be
performed at a pressure equal to the design pressure and at ambient
temperature conditions. Following this test is a high pressure
leak rate test at design pressure and ambient temperature, followed
by a low pressure leak test at approximately 2.5 psig. For a full

discussion on the criteria and procedures for these tests, see
<Section 6.2.6.5>.

j.   Safety/Relief Valve Operation (symbols P_{srv} and T_{srv})

The induced loads on the interior structural components are due to
the pressure and thermal transients associated with the operation
of the safety/relief valve system. The loads are discussed in
<Appendix 3A> and <Appendix 3B>.

k.   Normal Operating-induced Pressure, or Differential Pressure Due to
Tornado Pressure Drop (symbol Z)

Apply as applicable, based on location of the structure.

3.8.3.3.2   Loading Combinations

a.   Loading combinations for interior structure reinforced concrete
components including drywell wall and top slab, fuel pool walls,
drywell weir wall, and support mat are detailed in <Table 3.8-3>.

b.   Loading combinations for the biological shield wall, reactor vessel
pedestal and for the structural steel framing are given in
<Table 3.8-4>.

c.   The absolute sum method was used to combine all loads for internal
concrete and steel structures, except as described in
<Section 3.8.3.4.2.h>.

d.   Design Approach

For a discussion of the design approach see <Section 3.8.1.3.2.b>.

3.8.3.3.3   Load Factors for Reinforced Concrete Design

a.   Load Factors are discussed in <Section 3.8.1.3.3.a>.

     Dynamic load factors are discussed in <Section 3.8.1.3.3.b>.

3.8.3.3.4   Load Combinations for Local Areas

The load factors used for the design of local areas are the same as
those used for the general structure and outlined in <Section 3.8.3.3.2>
and <Section 3.8.3.3.3>.

3.8.3.3.5   Time Dependent Effects

Prestressing is not included in the design for any part of the interior
structure and therefore the effects of time dependent variations in
loads are not of a significant nature.

3.8.3.3.6   Thermal Effects

Temperature profiles have been calculated for the various pertinent
operational conditions of the plant using the method discussed in
<Section 3.8.1.4.3.a.4> and are shown in <Figure 3.8-38>,
<Figure 3.8-39>, <Figure 3.8-40>, and <Figure 3.8-41> for typical
sections.

3.8.3.3.7   Extent of Compliance to ACI 349-76 "Code Requirements for
                 Nuclear Safety-Related Concrete Structures," American
                 Concrete Institute

ACI 318-71 which was used for concrete structures on the PNPP and
ACI 349-76 are quite similar. For this reason, PNPP is, in general, in
compliance with the requirements of ACI 349-76. Specific sections where
ACI 349 differs from ACI 318 and PNPP compliance to these items are
summarized below:

Paragraph

6.3.3   Limits concrete temperatures to 150°/200°F for normal
                 operating conditions and 350°/650°F for accident

conditions. PNPP is in compliance.
<Section 3.8.1.4.5.c> and <Section 3.8.3.4.4.c>.

6.4   Requires engineer approval of all construction joints not shown on the drawings. PNPP requires engineer approval of all construction joints not shown on the engineers' drawings in the concrete placement specification. PNPP is in compliance.

7.3.2   ACI 301 reinforcement placement tolerances are adopted by ACI 349. Rebar placement tolerances for PNPP are as described in <Section 3.8.1.6.2.4> and <Section 3.8.3.6.2>. Where deviations from ACI 301 exist, justification is provided in these sections.

7.5.5   Specifies stagger requirements of 36 inch minimum for mechanical splices which do not meet specified strain requirements. A 3'-0" stagger was used for adjacent cadweld splices in the drywell structure. A 2'-0" stagger was specified for adjacent cadwelds for the No. 11 horizontal reinforcement in the weir walls. At certain construction openings where adjacent bars were cadwelded no stagger was provided. The use of mechanical splices was minimal except for the drywell structure and weir wall as discussed above.

The concern of the code was that at unstaggered mechanical splices, a crack could develop which would permit additional member rotations with a reduction in member capacity. Tests in beams have demonstrated this. The examples cited above are not subject to this concern. The cylindrical shape of the weir would preclude such a condition by geometry considerations alone. The construction openings in walls are relatively small and

generally subjected to membrane loads rather than the flexural loadings of a beam.

7.5.6 Requires tests acceptable to the NRC on welded splices and positive connections. Such tests have been provided on PNPP as described in <Section 3.8.1.6.3>. PNPP is in compliance.

7.6.4 Requires mechanical splices in biaxial tension regions. In structures such as the drywell cylindrical wall and top slab where large areas of biaxial tension exist, mechanical cadweld splices have been used. PNPP is in compliance.

7.13 Specifies minimum shrinkage and temperature reinforcement ratios of reinforcement area to gross concrete area. Reinforcement ratios provided for PNPP exceed the minimum requirements of ACI 349. PNPP is in compliance.

Chapter 8 Deletes reference to the "Alternate Design Method." The alternate design method was not used on PNPP. PNPP is in compliance.

9.3 Specifies that the required strength U shall at least equal the strengths required by the specified load combinations.

The load combinations as specified in <Section 3.8.1.3.2>, <Section 3.8.3.3.2>, <Section 3.8.4.3.2>, and <Section 3.8.5.3> meet or exceed the requirements of ACI 349. Only the load factor of 1.7 specified for R_o exceeds the 1.3 factor used for the same term on PNPP. This is not a controlling case, however, because on PNPP thermal has been combined with R_o and

because of the higher factor used on the steam relief valve reactions. We have assessed this increased load factor on several key structures including the shield building (feed water penetration anchors) and the drywell (main steam penetration anchors). Our evaluations demonstrate that we meet the intent of ACI 349 as modified by <Regulatory Guide 1.142>.

9.5   Minimum thicknesses of beams and one-way slabs required by ACI 349 are somewhat greater than ACI 318 requirements. Because most of the slab thicknesses used on PNPP are determined by shielding requirements, slab thicknesses generally exceed the requirements for specified spans and support conditions. Prestressed concrete was not used on the PNPP. PNPP is in compliance.

3.8.3.3.8   Bolt Installations Through Drywell Liner Plate

In cases where attachments need to be made directly to the inside drywell wall/ceiling in areas of the 1/4" thick liner plate, concrete expansion type bolts are typically used when cast-in-place embedments are not available. The expansion bolts are installed through the liner plate and are anchored directly into the concrete of the drywell structure. The great majority of expansion bolts fall into three main types, with three main subcategories respectively, as follows. Other bolt size/load combinations are used but in a very limited manner quantity wise.

Original Hilti Kwik-Bolt Installations:

a.   5/8″ diameter bolts embedded approximately 4″

Minimum spacing between bolts approximately 3″. Maximum loading
per bolt is 2.83 Kips (tension) and 3.13 Kips (shear), with most
bolts loaded significantly less than these values.

b.   3/4″ diameter bolts embedded approximately 4″

Minimum spacing between bolts is approximately 3″. Maximum loading
per bolt is 3.38 Kips (tension) and 4.40 Kips (shear), with most
bolts loaded significantly less than these values.

c.   3/4″ diameter bolts embedded approximately 7″

Minimum spacing between bolts is approximately 4″. Maximum loading
per bolt is 5.58 Kips (tension) and 4.93 Kips (shear), with most
bolts loaded significantly less than these values.

New Hilti Kwik-Bolt II Installations

a.   5/8″ diameter bolts embedded approximately 4″

Minimum spacing between bolts approximately 6″. Maximum loading
per bolt is 3.51 Kips (tension) and 2.63 Kips (shear), with most
bolts loaded significantly less than these values.

b.   3/4″ diameter bolts embedded approximately 4″

Minimum spacing between bolts approximately 6″. Maximum loading
per bolt is 3.88 Kips (tension) and 3.25 Kips (shear), with most
bolts loaded significantly less than these values.

c.   3/4″ diameter bolts embedded approximately 7″

Minimum spacing between bolts approximately 6″. Maximum loading per bolt is 5.49 Kips (tension) and 3.25 Kips (shear), with most bolts loaded significantly less than these values.

New Hilti Kwik-Bolt 3 Installations

a.   5/8″ diameter bolts embedded approximately 4″

Minimum spacing between bolts is approximately 10 3/8″ to obtain maximum allowable loading capacity. Maximum loading per bolt is 3.82 Kips (tension) and 3.66 Kips (shear), with most bolts loaded significantly less than these values.

b.   3/4″ diameter bolts embedded approximately 4 3/4″

Minimum spacing between bolts is approximately 12 1/4″ to obtain maximum allowable loading capacity. Maximum loading per bolt is 5.36 Kips (tension) and 5.49 Kips (shear), with most bolts loaded significantly less than these values.

c.   3/4″ diameter bolts embedded approximately 6 1/2″

Minimum spacing between bolts approximately 13 5/8″ to obtain maximum allowable loading capacity. Maximum loading per bolt is 6.64 Kips (tension) and 5.49 Kips (shear), with most bolts loaded significantly less than these values.

Each bolt is typically installed through an oversized hole in the 1/4″ liner plate to allow for liner thermal expansion without imposing loads onto the bolts. Each bolt has sealer tape installed in the annular space between the bolt shank and liner oversized hole in order to maintain a degree of leak tightness. Since the concrete cover is

nominally 5-1/2" to the inside face vertical steel reinforcement, the majority of bolts (embedded from 4" to 7" as noted above) will not extend beyond the first layer of reinforcement. All expansion bolts through the drywell liner are installed per specific approved procedures/instructions, including liner plate weld repair requirements for unused or abandoned drilled bolts.

The use of expansion bolts in this type application has been evaluated for affects on drywell structural and leak-tight integrity. More specifically, some of the areas evaluated are as follows:

a.   Affect on leak tightness and structural integrity of the 1/4" liner plate.

b.   Potential for cracking of the concrete drywell structure; with and without expansion bolt installations; under all normal, transient and accident conditions.

c.   Effects of expansion bolt installation on drywell bypass leakage.

The results of the evaluation are documented in (Reference 23), <Reference 24) and <Reference 25). In summary, it was concluded that expansion bolts installed in this manner will not impair the integrity

of the drywell structure. Some of the specific bases for this conclusion are as follows:

- The steel liner is not required to be leak-tight. The drywell structure itself, without assistance from the liner, is designed to meet the bypass leakage requirements of <Section 6.2.1.1.5>.

- Structural integrity of the liner is not affected since bolt loads are transferred directly to the concrete.

- Bolts used in this manner will not contribute to initiation or propagation of concrete cracks due to the small loads and shallow embedments typically used with these bolts. Additionally, both the normal operating and accident temperature transients through the drywell structure tend to keep the inside of the drywell in compression, thus minimizing potential for crack initiation and/or propagation.

- Through-thickness cracking of drywell concrete is not anticipated to occur under any normal, transient or accident conditions and therefore no significant bypass leakage flowpaths through the drywell structure will occur.

3.8.3.4   Design and Analysis Procedures

3.8.3.4.1   Design Assumptions and Boundary Conditions

a.   Drywell Wall, Top Slab and Fuel Pool Walls

Several finite element models are used to analyze the drywell structure. The first model is a three dimensional finite element model used for the overall analysis of the drywell. It encompasses the drywell wall, drywell top slab, refueling pools, and major

subcompartments. Because of symmetry of loadings and structure, only half of the drywell is modeled as shown in <Figure 3.8-42>. The vent region is idealized by quadrilateral sandwich plate elements which consist of a concrete core lined on both sides by equal thickness steel face plates. Both the concrete core and the face plates are assumed to be linearly elastic, homogeneous, isotropic materials. Vent holes are not modeled, but their effects are fully accounted for by the method of reduced elasticities as discussed in <Section 3.8.3.4.2>. The reinforced concrete section of the drywell wall, the top slab, the refueling pool, and the subcompartments are idealized by quadrilateral or triangular plate elements. The concrete is assumed as a linearly elastic, homogeneous, isotropic material. The analyses used by the computer is based on the program NASTRAN. The boundary conditions along the lines of symmetry at the edges of the model are satisfied with typical conditions of symmetry such as lateral translation and edge rotation being restrained. The bottom (base mat interface) of the model is assumed to be fixed.

The second model is an axisymmetric finite element model which encompasses the base mat, the weir wall and weir wall support mat and the lower portions of shield building, drywell wall and reactor pedestal as shown by <Figure 3.8-43>. This model is used to determine thermally induced forces and moments at the base of the drywell. Concrete sections are modeled by axisymmetric solid elements. The material is assumed to be linearly elastic, homogenous and isotropic. Nonlinear thermal gradients are input by specifying nodal point temperatures. Face plates of the pedestal and the drywell vent region are not modeled but temperature effects of face plates are calculated by hand and applied to the model in the form of traction forces. The advantage of this analysis is that the interaction of various structural components are taken into account and more realistic forces and moments are obtained at the intersection of the drywell wall and the base mat. Cracking of

the concrete is conservatively neglected. Concrete cracking due to
thermal loads, however, is considered in the design of drywell
wall, top slab and drywell vent region as discussed in
<Section 3.8.3.4.2>. The computer program ELAD is employed in this
analysis. Results obtained are used for the design of the drywell
to base mat anchor system.

To analyze the areas in the proximity of the personnel lock and the
equipment hatch, two independent finite element models are made as
shown in <Figure 3.8-44> and <Figure 3.8-45>. The embedded steel
frames of the lock and the hatch are designed for and act as end
anchorages for the wall reinforcement interrupted by the lock and
hatch openings. Therefore, the steel frames are analyzed as an
integral part of the drywell wall. The STARDYNE finite element
computer program is used in this analysis. Concrete is represented
by triangular plate elements and the steel frame by beam elements
in these models. The displacements calculated by the overall
analysis are used as input to the program at the edge of the model
in the form of enforced boundary displacements. The concrete and
steel are assumed to be linearly elastic, homogeneous and
isotropic.

Because of the large number of penetrations enclosed in the steam
tunnel area of the drywell wall, a more detailed analysis of this
area was required. Since the wall in this area is broken up into
very distinct "beam strips," a STRUDL frame model as shown in
<Figure 3.8-46> was used to analyze this area for the local
penetration anchor forces. The boundary conditions were varied to
determine the maximum bending moments. Results from this analysis
were then superimposed on the results of the NASTRAN general
analysis and reinforcement provided for the combined results.

Stress concentration effects are investigated by detailed finite
element models of the drywell vents and steam relief valve piping

penetrations as shown in <Figure 3.8-47> and <Figure 3.8-48>.
These local analyses included the steel face plates only; the
concrete core is assumed to be totally ineffective in resisting
tensile forces. The boundary conditions are obtained from the
results of the overall analysis and are applied along the edges of
the model as boundary forces. The inside and outside face plates
are analyzed separately. The computer program used for the local
analyses is NASTRAN and the materials are assumed as linearly
elastic, homogeneous and isotropic.

When evaluating the fuel pool walls, fuel transfer area and other
subcompartments for local loadings they are conservatively assumed
to be fixed at the drywell for all loadings except thermal loads
and drywell pressure loads. Forces and moments were taken directly
from the overall computer analysis for thermal loading and pressure
loading to take into account the interaction between the drywell
and its subcompartments.

b.   Reactor Pressure Vessel Pedestal and Biological Shield Wall

The principal design assumptions and boundary conditions for these
structures are that they can be analyzed independently and that
interfaces are rigid. The concrete fill in these structures is
considered capable of supporting its own dead weight, except the
biological shield wall concrete dead weight is conservatively
assumed to load the reactor pressure vessel pedestal steel, and is
ineffective other than as a load inducing component during dynamic
events.

c.   Weir Wall and Weir Wall Support Mat

The weir wall support mat is restrained from uplift by a flange
welded to the reactor pressure vessel pedestal at its inner
circumference and by anchor bolts along its outer periphery, as
shown in <Figure 3.8-26>.

These pinned connections represents the only boundary conditions between the weir mat and the adjacent structures. The weir wall is considered fixed and modeled integrally with the weir mat.

d.   Structural Steel Frames and Floors

Generally, the analytical model for the static analysis of each platform consisted of a STRUDL frame model of the beams in a platform plus the columns to the floors above and below the platform. The appropriate member end conditions were input. Loads and displacements at the columns were then used for design of the columns. A dynamic DYNAL frame model, shown in <Figure 3.8-13> and representative of the platforms over the pool area, is used to determine the response of the platforms to the pool dynamic loads resulting from a main steam or recirculation line break. This analysis is discussed further in <Appendix 3A>.

3.8.3.4.2   Analytical Techniques

a.   Drywell Wall, Top Slab and Fuel Pools

1.   Static Analysis

The structural analysis of the drywell is performed by using the finite element program NASTRAN. NASTRAN is a large capacity program specifically designed for the analysis of complex structures. Based on the displacement method, the program assembles the individual element stiffnesses into a global structure stiffness matrix through appropriate matrix transformations and combinations. The system is then solved for the generalized coordinates by matrix decomposition. For this analysis the vent region will be treated as a perforated shell. A method of analysis for this type of shell is presented by A. Ghali (Reference 10) which models the shell as

an assemblage of plane triangular or quadrilateral plate
elements. A typical segment with one opening is analyzed
using the finite-element technique to obtain the reduced
elasticity matrices relating stress to strain components of
the perforated plate. The reduced elasticities are then used
for elements at the vent region in a global analysis of the
drywell structure using flat plate finite-elements.

The analysis procedure for the drywell is presented below:

(a) Determine M_o and P_o, the moment and force, due to loads
     other than thermal loads such as dead load, live load,
     seismic and pressure loads, by the model discussed in
     <Section 3.8.3.4.1.a>. The analyses are based on the
     assumption that the structure remain linearly elastic and
     uncracked.

(b) Determine mean temperature $\overline{Y}$ by the equation

$$\overline{Y} = \frac{1}{A} \int T(Z)\, dZ.$$

     Where $T(Z)$ is the applied thermal gradient through the
     thickness Z and A is the area of cracked concrete section
     per unit width.

     Since the extent of cracking is unknown at the beginning,
     an iterative procedure must be used to find the final
     cracked section. Starting by assuming a mean
     temperature, usually the average temperature of the
     section, calculate the unit cracked sectional area A and
     the associated mean temperature $\overline{Y}$. Based on this new
     mean temperature $\overline{Y}$, another area A and temperature $\overline{Y}$
     may be found. The process is repeated until the starting
     temperature equals the calculated mean temperature. It

was found that three to four iterations will generally yield the desired accuracy.

(c)   Based on the cracked section determined in step (b), find the thermal gradient moment M_g' by the equation

$$M_g' = -\frac{E\alpha}{1-v} \int T(Z)ZdZ$$

Where E = Young's modulus, α = coefficient, thermal expansion, and v = poisson's ratio. Thus, the thermal gradient moment M_g' is based on a fully restrained cracked section.

(d)   Using the same finite element model in step (a), but revising the element sectional properties to those of the cracked section, determine the force and moment P_{ta} and M_{ta} due to temperature difference between the mean temperature obtained in step (b) and the base, or stress free, temperature.

(e)   The final design moment and force for the section is the sum of all the forces and moments obtained in steps a, c and d. Thus,

Design moment = M_o + M_g' + M_{ta}

Design force = P_o + P_{ta}

2.   Dynamic Analysis

The seismic analysis of the reactor building uses a lumped mass beam and soil spring model as discussed in <Section 3.7>. The inertia forces due to the lumped masses are redistributed as equivalent static loads on the various static computer

models. SRVD dynamic effects are evaluated using a shell of
revolution model using soil finite elements for the
Ghosh-Wilson ASHSD program which provides equivalent static
internal shears, forces and moments directly. This analysis
is discussed further in <Appendix 3A>.

3.   Interior Missiles

Missiles within the reactor building are relatively small as
compared with the mass of surrounding reinforced concrete
structures. The high velocity and small impact area will
cause the missiles to penetrate into the protective structures
until their energies are dissipated. The analytical procedure
used in design of missile protection shielding is described in
<Section 3.5>. It has not been found unnecessary to use any
missile shields for the drywell because the massive concrete
walls will withstand all postulated missiles without any
detriment to structural performance or the bypass leakage
requirements of <Section 6.2.1>.

4.   Heat Transfer Analysis

The temperature gradients throughout the structure are
established using the method described in
<Section 3.8.1.4.3.a.4>. The postulated transient and
steady-state temperature conditions are investigated for
combinations of temperature to determine the critical thermal
gradients at any point in the structure.

5.   Crack Analysis of Drywell

Although the crack analysis described in <Section 6.2.1>
indicates that the drywell will not crack sufficiently to
cause leakage problems, as additional assurance a 1/4 inch

stiffened steel plate is used to form the interior surface of the drywell. In addition, a study of potential cracking of the reinforced concrete drywell was performed as discussed in <Section 6.2.1>. Also, a preoperational proof pressure test and leak rate tests will be performed to the pressure and criteria discussed in <Section 3.8.3.7> and <Section 6.2.6.5> and technical specifications. For further discussion on liner leak tightness and drywell structure concrete cracking potential, see <Section 3.8.3.3.8> and <Section 6.2.1.1.5>.

b.   Reactor Pressure Vessel Pedestal and Biological Shield Wall

1.   Static Analysis

The principal analytical tool employed in the analysis of the biological shield wall and RPV pedestal is the finite element computer program STARDYNE. The steel of these structures is modeled as six degree of freedom plate finite elements as shown in <Figure 3.8-49> and <Figure 3.8-50>. The concrete fill in these structures is considered capable of supporting its own dead weight, except that the biological shield wall concrete dead weight is conservatively assumed to load the RPV pedestal steel, and is ineffective other than as a load inducing component under seismic events. The RPV pedestal is considered completely restrained at Elevation 574'-10" and provides a rigid interface for all attached structures as does the biological shield wall. Internal stiffeners and/or increased plate thicknesses are provided as required to achieve the rigid interface.

2. Dynamic Analysis:

    These structures are analyzed in the same manner as the
    drywell wall.

3. Heat Transfer Analysis

    The analysis for temperature effects is discussed in
    <Section 3.8.3.4.2.a.4>. The heat generation resulting from
    radiation was investigated for the biological shield wall
    only, as this is the most affected region with the effects on
    objects remote from the wall orders of magnitude smaller. The
    method of obtaining the radiation generated heat is as
    follows:

    The heat production rate at any point in a medium due to a
    neutron or gamma-ray field existing at a point can generally
    be related to the following major quantities: the gamma-ray
    flux density energy spectrum and the kinetic energy of the
    resulting charged particle emission due to neutron capture.
    The ANISIN (Reference 11) Computer Code is used to obtain the
    above spectra distributions through the shield.

    Using this result, the volumetric rate of heat generation as a
    function of position is then obtained:

    $$H(r, t) = \int_E \mu_d (r, E)\ E\ \phi (r, t, E)\ dE$$

    where

    $H(r,t)$ = volumetric heat generation at position (r) in
                 Mev/cm^3-sec

    $\mu d(r,E)$ = linear energy deposition coefficient, cm^{-1}

    $\phi(r,t,E)$ = flux spectra, Mev/cm^2-sec

Once the volumetric heat generation rate (H(z,t)) within the shield is known, the problem can then be solved using the standard equations of heat transfer with boundary effects.

4.   Pipe Whip Restraint Support Structure Analysis

The structural steel support structures provided to carry the pipe whip restraints is analyzed using the computer programs STRUDL and STARDYNE. The frame forces imposed on the biological shield wall are added to the other load cases as given in <Section 3.8.3.3.2.b> and applied to the model described in <Section 3.8.3.4.2.b.1>.

c.   Weir Wall and Weir Wall Support Mat

Both the ELAD and NASTRAN finite element programs are used in the analysis of the weir wall and weir wall mat. ELAD is used for mechanical loads, and NASTRAN is used to evaluate the effects of temperature. For the ELAD analysis, only the weir wall and weir wall mat are included in the model <Figure 3.8-85> since the boundary conditions at the RPV pedestal and at the 1/4 inch outer-liner plate could be realistically represented as pinned. For the thermal analysis, the interaction between the various structures was considered by using a model, as shown in <Figure 3.8-86>, which included the reactor building foundation mat and portions of the RPV pedestal, drywell wall and shield building in addition to the weir wall and weir mat.

One of the major loadings on the weir wall is the jet force due to a recirculation line break. The effective pressures from this load were represented in the ELAD computer program using Fourier series. The membrane, flexural and shear forces predicted from this analysis were included in the design of the weir wall.

Because the steam relief valve discharge loads only react through the drywell vents, the major part of this load is resisted primarily by the weir wall mat by direct bearing into the RPV pedestal. Only that portion of the weir wall directly across from the top vent is impacted by the SRV pressure loads and, as a result, this is not a critical load for the design of the weir wall.

The thermal analyses using the NASTRAN model showed significant moments and bending deformations in the weir wall particularly for the small break accident (SBA) case. For the weir mat, a cracked section approach as presented in ACI-307-69 was used. The equations were modified to reflect Poisson's effort by dividing by (1-m). For the weir wall, thermal stresses were a function of the outward growth of the mat as well as a gradient across the wall. The NASTRAN model was run using the actual thermal gradients in the mat and the average temperature in the weir wall. The elastic discontinuity moment at the wall to mat junction was then combined with the cracked moment due to the wall gradient calculated using the approach described above. The wall was then designed for this moment, the concurrent membrane forces and the moments and membrane forces due to concurrent mechanical loads.

d.   Structural Steel Frames and Floors

The static analysis of the structural steel frames and floors makes extensive use of the STRUDL computer program. The columns are considered fixed at the base mat, and all other joints are considered pinned. Dynamic effects are included through the use of floor response curves for the seismic and SRVD conditions, and by use of a separate dynamic model using the program DYNAL to determine the response due to pool swell. The mass ratio effect of the structural steel to concrete mass was taken into account for the seismic design of the steel platforms.

e.   Local Areas

      The computer programs used in the analyses of the various interior
      structures are capable of analyzing the effects of corners and
      general discontinuities. The vent region, as well as other major
      openings and penetrations, are analyzed using graded fine mesh
      finite element models. Boundary conditions for each of these large
      scale models are obtained from the general analysis discussed in
      <Section 3.8.3.4.2.a.1>. Local reinforcement or stiffening is
      provided around the openings or penetrations for the calculated
      stress concentrations.

f.   Analysis of the Equipment Hatch, Personnel Door and Drywell Head

      Analysis of the drywell equipment hatch, personnel door and drywell
      head is in accordance with the requirements of the ASME Code,
      Section III, Division 1, Class MC, for the metallic components and
      proposed ASME Code Section III, Division 2, for concrete anchorage
      details. <Figure 3.8-30>, <Figure 3.8-31> and <Figure 3.8-32> give
      typical details of these appurtenances and the anchorage details.
      The STARDYNE computer program is used for the static analysis, and
      dynamic effects are evaluated by the use of floor response curves.

g.   Variations in Material Properties and Assumptions

      For a general discussion on the effects of variations in material
      properties and assumptions, see <Section 3.8.1.4.3.d>.

h.   Analysis of Drywell Personnel Airlock Shield Door Structural
      Support System

      The shield door supports and the 620'-6" platform have been
      reanalyzed to demonstrate adequacy for required design basis
      loading combinations. The reanalysis was performed using a time

history analysis as described in <Section 3.7.3.1.1>. The loads
were combined in accordance with GESSAR Section 3BA.8.4. The
resulting stresses were found to be less than the acceptance
criteria of <Section 3.8.3.5.2>. The modified drywell personnel
airlock shield door structural support system is adequate for the
design basis loads in either the open or closed position.

3.8.3.4.3   Expected Behavior Under Load

The methods of analysis and design used for these structures predict the
behavior of the as constructed system. The analytical techniques allow
for discontinuities, changes in section and materials such that these
effects can be allowed for in the structural design. No impairment of
structural integrity or measurable permanent deformation is expected for
these structures, with the exception of some pipe whip restraint
structures which will yield under the postulated pipe break loads.

3.8.3.4.4   Design Methods

a.   Concrete Structures

     All internal concrete structures are designed for the strength
     requirements of ACI 318. In addition, the drywell diagonal
     reinforcement was designed to the criteria for tangential shear
     reinforcement of the ASME Code, Section III, Division 2, and also
     meets the restrictions of the NRC Standard Review Plan 3.8.1,
     Section II.5.

     The design approach followed the tangential shear requirements of
     ASME Code, Section III, Division 2. Supplemental calculations
     demonstrate that sufficient diagonal reinforcement was provided
     such that for the abnormal/severe and abnormal/extreme
     environmental load combinations as listed in <Table 3.8-3>, the

tangential shear stress carried by the concrete will not exceed 40 and 60 psi, respectively.

b.   Methods of Reinforcing Critical Areas

Rebar requirements for areas around the equipment hatch, the personnel lock and at the drywell steam tunnel area were determined based on separate analyses using more detailed computer models as discussed in <Section 3.8.3.4.1>.

At the personnel lock and equipment hatch, reinforcement interrupted by these openings was cadwelded to the embedded frame provided for this purpose. At the smaller openings, the main reinforcement was either bent to miss the penetration or terminated at the penetration. Where reinforcement is terminated at a penetration, the reinforcement area is replaced each side and above and below the openings. This supplemental reinforcement has an embedment length at each end equal to at least 1.25 times the basic rebar development length. Rebar terminated at a penetration is anchored by a standard hook, U-bar or a mechanical anchor. To control local cracking and strains, additional rebar were added diagonally around the penetrations in two directions on each face. Reinforcement for the drywell steam tunnel area is shown in <Figure 3.8-51> and <Figure 3.8-52>.

c.   Thermal Limitations

For normal plant operation, the concrete temperatures are limited to 150°F except for local areas such as hot penetrations where 200°F is permitted. For the worst postulated accident conditions, the concrete temperature will not exceed 285°F.

3.8.3.4.5   Computer Programs

a.   Computer Programs used in the Structural Analysis of the Interior
      Structures

    1.   MRI/STARDYNE

          The MRI/STARDYNE analysis system consists of a series of
          compatible computer programs designed to analyze linear
          elastic structural models. The system can be used to evaluate
          a wide variety of static and dynamic problems with only the
          static version used for the Perry Nuclear Power Plant project.
          The static capability includes the computation of structural
          deformations and member loads and stresses caused by any
          arbitrary set of thermal conditions, nodal applied loads
          and/or prescribed displacements. MRI/STARDYNE was developed
          by the Mechanics Research Institute for Control Data
          Corporation in 1971. It is a large capacity finite-element
          program which is designed for the analysis of truss, frame and
          plate structures. Based on the direct stiffness method, the
          program assembles the individual element stiffnesses into a
          global structure stiffness matrix, using appropriate matrix
          transformation and combination techniques; this system of
          equations is then solved for the generalized coordinates by
          Cholesk decomposition. A complete discussion of the theory
          used in STARDYNE and the theoretical and analytical
          verification is provided in Control Data Corporation STARDYNE
          Theoretical Manual (Publication No. 86616300).

          This program has been widely used for analysis of complex
          structures since its release for commercial use in 1971. The
          program is run on the Control Data Corporation CYBERNET system
          in New York on a CDC 6600 computer.

2.   NASTRAN

      NASTRAN is a general purpose finite element program for the
      analysis of complex problems. The program 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. The grid point definition forms the basic
      framework for the structural model. Static loads are applied
      to the model by concentrated loads at grid points, pressure
      loads on surfaces or indirectly by means of the mass and
      thermal expansion properties of structural elements. A
      detailed discussion on the theory and development of NASTRAN
      is presented as part of the three volume set of manuals.
      NASTRAN was developed by Computer Services Corporation with
      MacNeal Schwendler, Martin Baltimore and Bell Aerosystems Co.
      as sub-contractors for the National Aeronautics and Space
      Administration in 1970.

      The program has been widely used for the analysis of complex
      structures and components since its release to the public
      domain. For the Perry Nuclear Power Plant the program is run
      on the McDonnell Douglas, St. Louis, MO., IBM 360/195 dual
      processor computer system and on the Control Data Corporation
      CYBERNET System in New York on a CDC 6600 Computer.

3.   ELAD

      For program description and computer hardware/software see
      <Section 3.8.1.4.6.a.2>.

4.   ICES-DYNAL

For program description and computer hardware/software see
<Section 3.8.1.4.6.a.1>.

5.   ICES-STRUDL-II

For program description and computer hardware/software see
<Section 3.8.1.4.6.a.3>.

b.   Design Control and Verification of Computer Programs:

The methods used for the design control and verification of the
computer programs used in the analysis of the interior structure is
as discussed in <Section 3.8.1.4.6.b>.

c.   Verification of Program Usage:

1.   MRI/STARDYNE

This program has been widely used in the public domain for the
past six years. The results obtained for the structural
analyses using this program and computer hardware/software
combination have been verified by comparison of results with
those of similar programs, hand calculations and test
problems. Documentation of this program is traceable to
Control Data Corporation, Minneapolis, Minnesota.

2.   NASTRAN

This program has been used by many sections of industry over
the past eight years for many different types of analytical
problems. The results of the structural analyses utilizing
this program and computer hardware/software combination have

been verified by test results on sample problems.
Documentation of the program is traceable to the McDonnell
Douglas Automation Company, St. Louis, Missouri, and to the
National Aeronautics and Space Administration.

3.   ELAD

For program verification refer to <Section 3.8.1.4.6.c.2>.

4.   ICES-DYNAL

For program verification refer to <Section 3.8.1.4.6.c.1>.

5.   ICES-STRUDL-II

For program verification refer to <Section 3.8.1.4.6.c.3>.

3.8.3.5   Structural Acceptance Criteria

The structural acceptance criteria for the internal structures is based
on accepted codes and standards, with modification to suit the
operational and functional criteria of these structures. For the
drywell, the criteria is supplemented by a once only preoperational
proof pressure test as discussed in <Section 3.8.3.7>.

3.8.3.5.1   Concrete Structures

The structural acceptance criteria for concrete components are based
upon the strength design method of ACI 318 using the load combinations
detailed in <Section 3.8.3.3>. These criteria are modified by the
capacity reduction factor ϕ of Section 9.2 of ACI 318 to ensure that the
structural behavior will be elastic for all postulated loads and load
combinations.

3.8.3.5.2   Steel Structures

Structural steel acceptance criteria are generally based on the
recommendations of the AISC for the loads and load combinations set out
in <Section 3.8.3.3>. The criteria do not include the AISC
recommendations for plastic design, nor do they permit any increase in
allowable stress for the operational basis earthquake loads so that
elastic behavior is anticipated.

The criteria for pipe whip restraints permit plastic deformation of the
restraint but limit the base structures to elastic stresses. The
controlled deformation imparted to the restraint enables the energy in
the ruptured pipe to be dissipated while controlling the pipe
displacement.

The Von-Mises stresses for the biological shield wall, RPV pedestal and
the drywell vent structure are limited to .9 Fy for accident conditions,
and to .6 Fy for normal operating conditions, except for regions of
local discontinuity where stresses approaching the ASME criterion of
3 S_m are permitted. Thermal stresses on structural elements, except
plate structures, are permitted to be neglected when they are shown to
be secondary and self limiting in nature and the material is shown to be
ductile. For load combinations with accident thermal stresses included,
stresses in plate structures are limited to the ASME limit of 3 S_m.

The drywell head, airlocks and hatches use ASME Code, Section III,
Division 1, Subsection NE, Class MC, criteria.

3.8.3.6   Materials, Quality Control and Special Construction
            Techniques

This design is based on material specifications giving acceptable
limitations of physical and chemical properties for the structural

materials used, and on imposing quality control provisions to ensure that the materials meet the specifications.

The organization, responsibilities and general provisions for the Quality Assurance Program are described in <Chapter 17>. The quality control provisions that have been imposed for the structural materials are described in this section. Wherever engineer approved design documents call for deviations or exceptions from the accepted codes and standards within this section, the approved design documents shall govern.

3.8.3.6.1   Concrete Construction

The materials and quality control for the concrete construction of the interior structures are as described in <Section 3.8.1.6.1.1> except that the specified compression strength requirements for the various structures are as listed below:

Structure	Specified Strength $\left(f'_C\right)$ in psi
Weir wall and mat	3,000 at 28 days
RPV pedestal structural fill concrete	3,000 at 28 days
Biological shield wall chemtree concrete	3,000 at 90 days
Drywell wall, stop slab and attached compartments	5,000 at 90 days

3.8.3.6.2   Reinforcing Steel

The material specification, quality control and construction techniques for reinforcing steel for the interior structures are as described in <Section 3.8.1.6.2>, except that the tolerances for placing of

reinforcement as described in <Section 3.8.1.6.2.4> are as modified
below:

a. For locations of bars within the depth or thickness of a member and
    for clear concrete cover in flexural members, walls and compression
    members, tolerances follow <Section 3.8.1.6.2.4.a.> except as
    follows:

Member Description and Thickness	Specified Covers Inches	Tolerance on Location Inches	Reduction on Cover Inches
Drywell Cylindrical	4 O.F.	-1, +2	1
Wall - 5 feet 0 inches thick	5-1/2 I.F.	-1, +2	1

b. For the longitudinal location of bends and for ends of bars, the
    tolerances are as specified in <Section 3.8.1.6.2.4.b> except that
    at penetrations, hooks, U-bars, or ends of bars may be placed
    within ±6 inches of the specified location. Additional
    reinforcement has been placed around openings as described in
    <Section 3.8.3.4.4.b>.

c. The spacing of reinforcement for the interior structures including
    the drywell top slab and subcompartment slabs and walls is in
    accordance with <Section 3.8.1.6.2.4.c>. For the drywell
    cylindrical wall only, the following tolerances were permitted:

    1. All reinforcement was required to be within 6 inches of its
        theoretical location but with the addition restriction below.

    2. The horizontal continuous reinforcement which is spaced
        nominally at 12 inches center-to-center may be spaced at not
        less than 9 inches nor more than 15 inches center-to-center.

3. The dowels for subcompartment walls were permitted to be moved up to 3 inches in the plane of the wall to eliminate interferences.

4. The dowels for subcompartment slabs were permitted to be moved up to 3 inches horizontally in the plane of the slab to eliminate interferences.

3.8.3.6.3   Cadweld Splices

The material specification, quality control and construction techniques for cadweld splices for the internal structures are as described in <Section 3.8.1.6.3>.

3.8.3.6.4   Steel Plate

The material specification, quality control and construction techniques for steel plate for the internal structures conform to the requirements described in <Section 3.8.2.6.1> except that AWS welding requirements may be substituted for ASME welding requirements.

3.8.3.6.5   Steel

3.8.3.6.5.1   Structural Shapes and Plates

Structural shapes and plates are made from steel conforming to ASTM A 36.

3.8.3.6.5.2   Bolts and Hardened Washers

High strength bolts and hardened washers conform to ASTM A 325.

All bolts other than high-strength bolts conform to ASTM A 307.

3.8.3.6.5.3   Welding Electrodes

Welding electrodes are in accordance with the provisions of the American
Welding Society Specification, "Structural Welding Code," AWS D1.1 and
are suitable for the type of steel to be welded. Generally, electrodes
for manual shielded metal arc welding process are E7015, E7016 or E7018.
Electrodes for all other welding processes are low hydrogen type and
give a deposit of 70,000 psi minimum tensile strength, with impact
properties equal to those of the manual shielded metal arc electrodes
specified above.

3.8.3.6.5.4   Material Control

Manufacturer's certificates are obtained for all lots, batches or heats
of material received to ensure compliance with the specified properties.

3.8.3.6.6   Stainless Steel

a.   Plate

     The liner plate for fuel pools is 1/4 inch thick conforming to the
     requirements of ASTM A 240, Type 304. The plate is hot-rolled,
     annealed, surface cleaned, and polished in accordance with
     ASTM A 480 to a surface equivalent to a No. 4 finish.

b.   Structural Shapes, Pipe and Tubing and Bolt Materials

     1.   All stainless steel structural shapes conform to ASTM A 276,
          Type 304, hot-rolled, annealed, blast cleaned, and pickled.

     2.   All stainless steel pipe conform to ASTM A 312, Grade TP 304.

     3.   Stainless steel equipment attaching bolts or studs conform to
          ASTM A 193, Class 2, Grade B8 or B8C.

c. Welding Electrodes for Stainless Steel

1. Manual welding electrodes for welding stainless steel to
    stainless steel conform to AWS A5.4, Type E308.

2. For manual or machine mechanical inert gas (MIG) or tungsten
    inert gas (TIG) welding of stainless steel to stainless steel,
    the welding rods or wire conform to AWS A5.9, Type ER308.

3. Manual welding electrodes for welding carbon steel to
    stainless steel conform to AWS A5.4, Type E309.

4. For manual or machine TIG or MIG welding of carbon steel to
    stainless steel, the welding rods or wire conform to AWS A5.9,
    Type ER309.

d. Embedded Studs for Stainless Steel

The stud material conforms to ASTM A 270, Type 304. The studs are
welded by automatic stud welders in accordance with RDT
Standard RDT-F6-6T.

3.8.3.7   Testing and Inservice Surveillance Requirements

A once only preoperational proof pressure test will be performed at the
drywell design pressure (P_t). This structural integrity test will be
integrated with a high pressure leak rate test at the same pressure.
Subsequently, a preoperational low pressure leak rate test will be
performed at a pressure as defined in the technical specifications.
These low pressure leak rate tests will be periodically performed as an
inservice inspection of the drywell. Test intervals, test pressures,
test duration, and acceptance criteria are as specified in
<Section 6.2.6.5> and technical specifications.

3.8.4   OTHER SEISMIC CATEGORY I STRUCTURES

The following safety class (Category I) structures are considered in
this section:

a.   Auxiliary Buildings (2)

b.   Fuel Handling Building (1)

c.   Intermediate Building (1)

d.   Control Complex (1)

e.   Radwaste Building (1)

f.   Diesel Generator Building (1)

g.   Offgas Buildings (1)

h.   Emergency Service Water Pumphouse (1)

i.   Offshore Intake Structures (2)

j.   Offshore Discharge Structure (2)

k.   Cooling Water Tunnels (2)

l.   Underdrain System manholes (26)

m.   Condensate Storage Tank Foundations and Dike Walls (1)

n.   Service Water Valve Pit (1)

o.   Electrical Manholes (4)

Refer to <Section 3.2> for the safety classification of each structure.

3.8.4.1   Description of the Structures

3.8.4.1.1   General

Certain safety class (Category I) structures, namely the auxiliary buildings, fuel handling building, intermediate building, control complex, radwaste building, diesel generator building, and offgas buildings, are seismically separated from the reactor building and each other by a 3 inch rattle space above the foundations. The relative motions of adjacent safety class (Category I) structures at critical elevations are determined to establish the required seismic separation. Foundations are considered in <Section 3.8.5>. See <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>, <Figure 1.2-11>, <Figure 1.2-12>, and <Figure 1.2-13> for the general arrangement of plant structures.

The structures are constructed from reinforced concrete and/or structural steel using the following materials:

a.   Concrete having a 28-day cylinder strength of 3,000 and 4,000 psi.

b.   Structural steel meeting ASTM A 36.

c.   Reinforcing steel meeting ASTM A 615-72, Grade 60.

Other materials and concrete strengths are identified in <Section 3.8.4> as related to functional description and design.

3.8.4.1.2   Auxiliary Building

This building is a reinforced concrete structure approximately 97 feet
high by 102 feet wide by 192 feet long with the top of mat at
Elevation 568'-4". It houses safety class systems and components,
including pumps and motors for the residual heat removal system (RHRS),
the high pressure core spray (HPCS), low pressure core spray (LPCS), and
the reactor core isolation cooling (RCIC) system. See <Figure 1.2-3>,
<Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>,
<Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>, and <Figure 1.2-11> for
general configuration and relationship of the auxiliary building to
other structures.

The concrete floors are supported by interior columns, interior walls
and by exterior walls. Exterior walls and the roof are a minimum of
2 feet thick for protection of safety class equipment from exterior
missiles.

The auxiliary building has a tunnel to house the main steam and
feedwater lines which run from the reactor building complex to the steam
tunnel. The auxiliary building tunnel is approximately 36 feet wide and
28 feet high.

The function of this tunnel is to withstand the pressures and
temperatures that could be produced by a postulated break in a feedwater
or main steam line. Seals are provided around the tunnel at its
junction with the steam tunnel on one end and a similar tunnel coming
from the shield building on the other end. The seals do not form a
structural connection to these other tunnels and, therefore, the
auxiliary building remains structurally separate from other buildings
above the foundation.

3.8.4.1.3   Fuel Handling Building

3.8.4.1.3.1   General

The fuel handling building is a three story building, located between
the Unit 1 and Unit 2 reactor buildings. The entire exterior of the
building is reinforced concrete including the foundation, the walls and
the roof slab. The roof slab is supported by structural steel framing.
Reinforced concrete is also used for the interior walls and slabs. The
function of the fuel handling building is to store new fuel and to
receive and store spent fuel from the reactor. The approximate plan
dimensions are 144 feet x 108 feet with the top of the foundation mat at
Elevation 574'-10".

The fuel handling building is provided with four pits for fuel handling
and storage:

a.   Cask storage pool and decontamination pool.

b.   Spent fuel pool.

c.   Fuel transfer pool.

d.   Fuel preparation pool.

The pools are interconnected by means of gates, to allow the underwater
passage of fuel assemblies from one pool to another.

Each pool is a stainless steel lined and reinforced concrete structure
<Figure 1.2-5>. The wall liners consist of 1/4 inch stainless steel
plates welded together and anchored to the concrete by 3" x 2" x 1/4"
angle stiffeners placed vertically at 15 inch center-to-center spacing.
The floor liner consists of 1/2 inch stainless steel plates welded
together and anchored to floor embedments placed horizontally at 4 foot

center-to-center spacing. These angles and floor embedments effectively anchor the liners to the concrete structure. Where attachments to the pool walls or floors are required, embedded plates are provided to transfer the attachment loads directly into the concrete. The liners are cut out around and welded to each embedment plate. Attachments are made to the embedded plates and not the liners. A system of leak chases, divided into zones, are provided behind the liner plate welds to channel the postulated leakage to a central collection point for leak identification and evaluation. The pool concrete structure with the liner is designed to Seismic Category I requirements to prevent damage to the stored fuel. Storage and handling of spent fuel is accomplished as described in <Section 9.1.2.2.2>. The spent fuel is stored in the fuel handling pools in densified racks described in <Section 9.1.2>. These racks are free standing (no pool wall attachments) and are designed to withstand the postulated loads including a drop accident, as described in <Section 9.1.2.1.2> and <Section 9.1.2.3.4>. Embedded plates anchored to the pool floor slab support the racks by transferring all rack loads into the concrete slab.

An overhead gantry crane is provided to handle and maneuver fuel assemblies between the three pits. A bridge crane traveling at right angles to the gantry crane will handle casked fuel to and from railroad cars. The bridge crane cannot traverse the spent fuel and fuel transfer pools due to the layout of the cranes. The unlikely possibility of a fuel cask being dropped on stored fuel is thus precluded.

The fuel handling building is designed to withstand tornado missiles as described in <Section 3.5.1>.

Spent Fuel Dry Storage (SFDS) operations are discussed in <Section 9.1.4.2.10.14>.

In order for the loaded HI-STORM to move outside of the FHB on rails, a reinforced concrete structural mat with railroad rails was installed and designed to match the 620'-6" FHB floor elevation. The function of this

exterior mat extension is to provide a solid level surface to allow the transfer of the HI-STORM into and out of the FHB rail bay. The rails inside the FHB have been extended onto the exterior mat extension for rail continuity. The mat extension is of sufficient size to allow for movement of the loaded HI-STORM to the Independent Spent Fuel Storage Installation (ISFSI).

The layout and general arrangement of the fuel handling building are shown in <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, and <Figure 1.2-10>.

3.8.4.1.3.2   Functional Criteria

The functions of the fuel handling building are:

a.   Store new and spent fuel assemblies.

b.   House safety class equipment required to maintain the fuel
     assemblies.

c.   Protect the fuel assemblies and safety class equipment from the
     effects of the environment.

d.   Provide access to the reactor building complex through the
     equipment access hatch.

e.   House additional safety class equipment required to secure systems
     in the reactor building complex.

3.8.4.1.4   Intermediate Building

The intermediate building is a five story reinforced concrete structure
located between the fuel handling building and the control complex in
the east-west direction and the reactor building complexes in the
north-south direction.

The intermediate building and fuel handling building are interconnected;
a 3 inch rattle space isolates them from all other structures.

The exterior walls are sized to insure protection from exterior
missiles. The top of the foundation mat is at Elevation 574'-10". The
layout of the intermediate building is shown in <Figure 1.2-3>,
<Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>,
<Figure 1.2-8>, <Figure 1.2-9>, and <Figure 1.2-10>.

The function of the intermediate building is to house safety class systems and equipment. Some of these systems service the fuel handling building, with other systems servicing the reactor building complex. The intermediate building also provides personnel access to the annulus space.

3.8.4.1.5   Control Complex

This building is a steel framed structure with exterior reinforced concrete walls of approximate dimensions 132 feet 6 inches high, 141 feet wide and 142 feet long, with the top of the foundation mat at Elevation 574'-10".

The control complex houses plant personnel and electrical controls to monitor and control normal plant functions and safety class systems.

The outside walls and roof of the control complex are 2 foot thick reinforced concrete for protection from exterior missiles and shielding of the control complex. The interior structure consists of steel columns, girders and beams with the floor deck of reinforced concrete. The building has no structural connection with other structures. For layout and configuration details of the control complex, see <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, and <Figure 1.2-10>.

3.8.4.1.6   Radwaste Building

The safety class portion of this building is a reinforced concrete structure of approximate dimensions 70 feet high, 101 feet wide and 217 feet long, with the top of the foundation mat at Elevation 574'-10". The nonsafety portion of the radwaste building is a steel framed, steel cladded structure approximately 76 feet long, 19 feet wide and 40 feet high above grade. The steel frame attaches to the west wall of the safety class portion of the building.

The radwaste building houses equipment used in the storage and processing of liquid and solid radioactive wastes.

The exterior walls and roof of the safety class portion consist of 2 to 3 foot thick reinforced concrete to provide shielding for the environment against radiation from the radioactive wastes. Interior shield walls are provided around tanks where necessary to shield personnel during normal operation of the plant. Tanks are mounted on the foundation of the building and on the floors at Elevations 602'-0" and 623'-6". The concrete floor slabs are supported by exterior and interior walls and by interior columns. The building has no structural attachment to other structures above the top of foundation. The layout and configuration of the radwaste building are shown in <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, and <Figure 1.2-10>.

3.8.4.1.7   Diesel Generator Building

This structure is a reinforced concrete building approximately 165 feet long, 78 feet wide and 26 feet high, with top of the foundation mat at Elevation 620'-6". The structural system consists of walls and roof sized for protection against missiles. Removable concrete missile shields are provided in front of each man access door and labyrinth shields for each man access door. The design of these shields is in accordance with the methods discussed in <Section 3.5>. The safety class equipment is mounted on the foundation slab at Elevation 620'-6". A reinforced concrete air intake structure, 165 feet long, 32 feet wide and 20 feet high, is mounted to the roof of the building. The walls and top of this housing are also designed for missile protection. In addition, concrete barriers are provided at the exhaust louvers to prevent external missiles from entering the diesel generator building. For additional details, see <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, and <Figure 1.2-10>.

The diesel generator building houses generators, day tanks and other equipment necessary to supply standby electric power to operate safety systems in the event of a power failure of the plant generating equipment and offsite power. This standby electrical system is described in <Section 8.3>.

3.8.4.1.8   Offgas Building

This building is a reinforced concrete structure 99 feet 9 inches long, 52 feet wide and 80 feet high, that houses equipment used in the filtering and absorption of radioactive noncondensible gases from the main and auxiliary condensers. See <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, and <Figure 1.2-10> for general configuration and relation to other structures.

The building is a four story structural system that is completely enclosed by a 2 foot thick roof slab and exterior walls of 2 feet minimum thickness, which are designed to resist the exterior missiles as defined in <Section 3.5>. The top of the concrete mat foundation is at Elevation 584'-0".

3.8.4.1.9   Emergency Service Water Pumphouse

The emergency service water pumphouse is a reinforced concrete structure, rectangular in plan, located between the nonsafety class service water pumphouse and the nonsafety class discharge tunnel entrance structure. The five foot thick foundation mat is placed on one foot of porous concrete which is founded on chagrin shale at approximately Elevation 531'-0". The structure has a reinforced concrete floor at Elevation 586'-6", which supports the pumps and screens. Above this floor an overhead crane is supported on steel girder rails which bear on reinforced concrete wall pilasters. Above that is a 2 foot thick reinforced concrete roof supported by steel

girders spanning the width of the building. Penetrating through the foundation mat are the 10 foot diameter riser shafts from the emergency and alternate emergency intake tunnels, which supply the pumphouse with lake water. Below the floor at Elevation 586'-6", the pumphouse structure is divided into compartments that are separated by means of reinforced concrete walls. During normal operation, lake water will be supplied to the emergency service pumps through the intake tunnel. Should the intake tunnel become obstructed, an alternate source of lake water is available through the discharge tunnel. Under normal operating conditions, the discharge tunnel is isolated from the emergency service water pumphouse by means of motor operated sluice gates. During an emergency, these gates will be opened to supply the necessary lake cooling water to the emergency service pumps. Regardless of the operating mode (normal or emergency), lake water passes through the traveling screens that have been installed ahead of the emergency service pumps. The water is then pumped from the compartments through two pipes to the auxiliary building. For details of the structure refer to <Figure 3.8-64>.

3.8.4.1.10   Offshore Intake Structures

Each offshore intake is a submerged steel plate multiport structure, circular in plan, resting on a working mat founded in shale bedrock at a minimum depth of four feet below the bottom of Lake Erie. Each intake is composed of an independent 36 foot diameter circular intake structure or intake head that is connected to a six foot diameter vertical downshaft which conveys the cooling water to the underground ten foot diameter intake tunnel. Each of the two intake heads is provided with a horizontal circular velocity cap and vertical inflow ports around the periphery.

Each intake structure is surrounded by 10 vertical reinforced concrete ice protection caissons which are placed in a 70 foot diameter circle around the intake. The caissons are six feet in diameter with the

exception of two caissons around the No. 1 intake structure which are seven feet in diameter due to the poorer foundation conditions of weathered and/or fractured shale at their locations. All caissons are founded in chagrin shale a minimum of 12 feet below lake bottom. The embedment depth of each caisson varies between 12 and 20.8 feet depending on the depth of weathered shale at each location.

Inflow of cooling water into the intake structures will be through the vertical ports around the perimeter of the circular structures. Provision for backfitting trash racks is provided around the periphery of each structure so that intake water may be filtered to remove large pieces of flotsam if the need arises.

The intake structures are safety class since they supply water to both the emergency service water pumphouse and the service water pumphouse. In the unlikely event of a complete blockage of both intake structures, an alternate supply of water can be obtained from the offshore discharge structure (nozzle). For details of the functional criteria, back-up systems, hydraulic details, etc. refer to <Section 2.4> and <Section 9.2>. The intake structures and riser shafts are shown in <Figure 3.8-65>, <Figure 3.8-66>, <Figure 3.8-67>, <Figure 3.8-68>, and <Figure 3.8-69>.

3.8.4.1.11   Offshore Discharge Structure

The offshore discharge structure consists of a submerged 3 foot diameter steel diffuser nozzle encased in a 17 foot diameter protective reinforced concrete ice protection caisson founded in chagrin shale a minimum of 7 feet below lake bottom. A six foot diameter riser shaft connects the 10 foot diameter underground discharge tunnel to the nozzle. The structure and riser shaft are shown in <Figure 3.8-65>, <Figure 3.8-66>, <Figure 3.8-67>, and <Figure 3.8-69>.

3.8.4.1.12   Cooling Water Tunnels

a.   General Features

     Service and cooling water for the power plant will be obtained from
     Lake Erie through the offshore intake structures. It will be
     carried to the power plant through an intake tunnel in the
     underlying bedrock, and returned to the lake through a similar
     discharge tunnel. Short tunnels of the same diameter will be used
     near the shore facilities to tap the cooling water tunnels for the
     plant's emergency service water. The tunnels are shown on
     <Figure 3.8-65>, <Figure 3.8-66>, and <Figure 3.8-70>. An
     explanation of the tunnel function is given in <Section 2.4.11> and
     <Section 9.2>.

     The cooling water intake and discharge tunnels and the emergency
     service water tunnels are lined with a permanent lining of
     cast-in-place concrete to a finished inside diameter of 10 feet.

b.   Excavation and Temporary Support

     Portions of the intake and discharge tunnels and emergency service
     water tunnels were excavated by conventional drill and blast
     tunneling techniques, while most of their length was excavated by
     tunneling machines. Tunnel cross-sections are shown in
     <Figure 3.8-71> and <Figure 3.8-72>.

     As the tunnel was excavated, a temporary support system consisting
     of rock bolts, steel straps and wire mesh were installed to
     maintain the safety and integrity of the excavation. The
     beneficial effects of the temporary supports were neglected in
     design and analysis of the permanent concrete lining. The
     temporary supports were, therefore, not designed as safety class
     structures, and are not discussed in the following sections.

c.   Permanent Lining

Upon completion of excavation, the tunnels were lined with
4,000 psi cast-in-place concrete to a circular cross section. This
concrete permanent lining serves several functions:

1.   Provides smooth walls to improve hydraulic flow
     characteristics.

2.   Prevents deterioration of the excavated surfaces due to
     unrestrained loosening and the effects of water.

3.   Supports the static ground loads which develop.

4.   Improves the tunnel stability under earthquake loading.

The minimum concrete thickness is defined by the "A" line shown in
<Figure 3.8-69>, <Figure 3.8-71>, and <Figure 3.8-72>. The
permanent concrete lining is reinforced where required to satisfy
design criteria explained in <Section 3.8.4.4>.

3.8.4.1.13   Underdrain System Manholes

The 13 underdrain manholes and the 13 gravity discharge manholes are all
part of the pressure relief underdrain system described in
<Section 2.4.13>. The manholes are reinforced concrete vertical shafts
having rectangular cross sections with 18 inch thick walls. All of the
manholes are penetrated by the steel gravity drain pipe at approximate
invert Elevation 585'-0". The underdrain manholes are also penetrated
by a 12 inch ID porous concrete pipe at Elevation 568'-6" or lower.
Access to each manhole is provided by a removable steel access cover
approximately 4 feet by 4 feet in plan. See <Figure 3.8-73> for
additional typical details.

3.8.4.1.14   Condensate Storage Tank Foundations and Dike Walls

The condensate storage tank is located within a reinforced concrete dike
wall and is adjacent to the turbine building. Refer to <Section 9.2.6>
for the system description. For layout and configuration details of
this structure see <Figure 3.8-74>.

The tank is supported by a ring girder which in turn rests on a 3 feet
6 inch thick common foundation mat. Sand fill is placed under the tank
within the ring girder. The top of foundation mat is at
Elevation 618'-6".

The inside radius of the ring girder is 24 feet 6 inches whereas that of
the dike wall is 33'-0". The height of the 2 foot thick dike wall is
25 feet 6 inches above the foundation mat.

A continuous waterstop has been placed between the dike wall and
foundation mat to prevent any water leakage caused by a rupture of the
tank.

3.8.4.1.15   Service Water Valve Pit

The service water valve pit is an underground reinforced concrete
structure approximately 18 feet high by 15 feet wide by 18 feet long
with the top of the mat at Elevation 602'-0". It houses safety class
valves for the service water system. See <Figure 3.8-75> for general
configuration and relation of service water valve pit to other
structures.

The concrete mat, walls and roof are two-feet thick. A seven-foot
square concrete hatch is provided for maintenance of the sump.

The function of this structure is to protect the service water valves
from a seismic event.

3.8.4.1.16   Electrical Manholes

The electrical manholes are reinforced concrete boxes used as
distribution points for cable routing. The manholes are all less than
18 feet deep, and approximately 15 feet by 15 feet in plan or smaller.
The walls are 18 inches thick for four of the five safety class manholes
and 12 inches thick for one relatively small manhole (7 feet by 7 feet
in plan). All of these manholes have 30 inch thick roofs for missile
protection. Access to the manholes is provided through 30 inch deep
removable concrete beams. See <Figure 3.8-76> for additional details.

3.8.4.2   Applicable Codes, Standards and Specifications

3.8.4.2.1   Codes

a.   Concrete Structures, Excluding Cooling Water Tunnels and Offshore
     Structures

     1.   ACI 301-72, "Specification for Structural Concrete for
          Buildings," American Concrete Institute.

          All chapters of ACI 301 are applicable except Chapter 15 which
          has requirements for prestressed concrete.

     2.   ACI 318-71, "Building Code Requirements for Reinforced
          Concrete," American Concrete Institute.

          Chapters Used

          1   General Requirements
          2   Definitions
          7   Details of Reinforcement
          8   Analysis and Design
          9   Strength and Serviceability

Chapters Used (Continued)

10   Flexure and Axial Load

11   Shear and Torsion

12   Development of Reinforcement

13   Slab Systems with Panels

14   Walls

15   Footings

Appendix B – Notation

Chapters Not Used

3[1]   Materials

4[1]   Concrete Quality

5[1]   Mixing and Placing Concrete

6[1]   Formwork, Embedded Pipes and Construction Joints

16[2]   Precast Concrete

17[3]   Composite Concrete Flexural Members

18[4]   Prestressed Concrete

19[5]   Shells and Folded Plate Structures

20[6]   Strength Evaluation of Existing Structures

Appendix A

NOTES:

[1] ACI 301-72 will be used to define requirements in this area
    as modified by <Section 3.8.4.6>.

[2] These are not precast concrete structures.

[3] Composite flexural members are not used on these
    structures.

[4] These are not prestressed concrete structures.

[5] These are not shell or folded plate structures.

[6] These are not existing structures.

3. Ohio Building Code 1970 edition, Chapter BB-33, "Reinforced and Plain Concrete, Reinforced Gypsum Concrete and Reinforced Masonry," Board of Building Standards Department of Industrial Relations, State of Ohio.

Sections Used

BB-33-01 Definitions
BB-33-02 Abbreviations
BB-33-03 Accepted engineering practice and approved standards
BB-33-07 Reinforced concrete general design and construction

Sections Not Used

BB-33-04[1]   Reinforced concrete materials
BB-33-05[1]   Concrete, classification and quality
BB-33-06[1]   Concrete, ready mixed
BB-33-08[2]   Reinforced concrete, supervision
BB-33-09 to
BB-33-13[3]   Plain concrete
BB-33-14[4]   Reinforced gypsum concrete, precast
BB-33-15[4]   Reinforced gypsum concrete, design and construction
BB-33-16 to
BB-33-20[5]   Reinforced masonry

NOTES:

[1] ACI 301-72 will be used in this area as modified by <Section 3.8.4.6>.
[2] Not applicable to these designs.
[3] Plain concrete is not used on these structures.
[4] Reinforced gypsum is not used on these structures.
[5] Reinforced masonry is not used on these structures.

b. Steel Structures

   1. AISC "Specification for the Design, Fabrication and Erection
       of Structural Steel for Buildings," February 12, 1969.

       All parts of the AISC Code are applicable except as amended by
       "Structural Design Criteria for Category I Structures Outside
       Containment" (Reference 1).

c. Cooling Water Tunnels and Offshore Structures

   1. Concrete Structures

       ACI-301-72 Chapters Used

       3   Proportioning
       7   Production of Concrete
       9   Repair of Surface Defects
       12   Curing and Protection
       16   Testing
       17   Evaluation and Acceptance of Concrete

       ACI-318-71 Chapters Used

       1   General Requirements
       2   Definitions
       3   Materials
       6   Formwork, Embedded Pipes and Construction Joints
       7   Details of Reinforcement
       8   Analysis and Design - General Consideration
       9   Strength and Serviceability Requirements

<u>ACI-322-72 Chapters Used</u>

1   General Requirements

2   Definitions

7   Design and Analysis

2.   Steel Structures

AISC "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969; including Supplements 1, 2 and 3.

3.8.4.2.2   Standards

a.   ASTM Standards: Applicable ASTM standards are discussed in <Section 3.8.4.6> for the following:

1.   Concrete.

2.   Reinforcing Steel.

3.   Cadweld Splices.

4.   Structural Steel.

5.   Stainless Steel.

6.   Cooling Water Tunnels and Offshore Structures.

The date of a particular standard may vary for different items because of the difficulty in purchasing material to an outdated standard. Since the latest ASTM standards reflect industry practice used for fabrication and erection, it was permitted to use an updated standard where no unacceptable loss of quality would result.

b.   Applicable Regulatory Guides

1.   <Regulatory Guide 1.10>, "Mechanical (Cadweld) Splices in
     Reinforcing Bars of Category I Concrete Structures." This
     guide was used with modifications specified in
     <Section 3.8.1.6>.

2.   <Regulatory Guide 1.15>, "Testing of Reinforcing Bars for
     Category I Concrete Structures." This standard was used with
     the modifications specified in <Section 3.8.1.2>.

3.   <Regulatory Guide 1.31>, "Control of Stainless Steel Welding."

4.   <Regulatory Guide 1.55>, "Concrete Placement in Category I
     Structures."

5.   <Regulatory Guide 1.69>, "Concrete Radiation Shields for
     Nuclear Power Plants."

6.   <Regulatory Guide 1.142>, (April 1978), "Safety-Related
     Concrete Structures For Nuclear Power Plants (Other Than
     Reactor Vessels and Containments)." Also see
     <Section 3.8.1.2.2.b.4>.

3.8.4.2.3   Principal Plant Specifications

The principal specifications prepared by the engineer for the safety
class structures are:

a.   Fabrication and erection of structural steel.

b.   Fabrication of embedded steel.

c.   Fabrication and erection of stainless steel liners for fuel pools.

d.   Concrete supply.

e.   Placement of structural concrete.

f.   Fabrication and placing of reinforcing steel and embedded items.

g.   Supply and installation of waterproofing and waterstop.

h.   General construction of tunnels, intake and discharge structures.

These specifications include the applicable requirements described in <Section 3.8.4.2>, <Section 3.8.4.3>, <Section 3.8.4.4>, and <Section 3.8.4.5>. The detailed material specifications, the detailed quality control provisions and any special construction technique requirements, all as described in <Section 3.8.4.6>, are included in these specifications.

3.8.4.3   Loads and Load Combinations

3.8.4.3.1   Loads Used in the Design

a.   Dead Load (symbol D)

Dead load includes the weight of the structural components plus any permanent attachments including cable trays and piping less than 18 inches in diameter.

b.   Live Load (symbol L)

     Live loads are as follows:

     1.   For roofs:

          (a) 25 psf live load.

(b) 84.5 psf snow and ice load based on the meteorological recommendations of <Section 2.3.1>.

2. For stairs and platforms 100 psf.

3. For floors:

(a) 100 psf for routine personnel and maintenance loads.

(b) More than 100 psf where required for equipment removal or maintenance.

4. Piping 18 inches in diameter or larger.

c. Ground Water Loads (symbol G)

The design bases for ground water loading conditions are described in <Section 2.4.13>, <Section 2.5.4>, and <Section 3.8.5.3.5>.

d. Soil Pressures (symbol H for static and F_{eqo} and F_{eqs} for dynamic soil pressures)

The static and dynamic soil pressures used in the design are given in <Section 2.5>. Static lateral soil pressures are included under the symbol H. Dynamic lateral soil pressures are included under the symbols F_{eqo} and F_{eqs} as appropriate.

e. Normal Operating Temperature Induced Forces (symbol T_o)

Temperature induced forces will be evaluated for the various sections of the safety class structures based on the most adverse

temperature gradients predicted for normal plant operation. The
exterior temperatures to be used are:

1.   Ambient temperature – minimum 7 day mean 10°F.

                            maximum 7 day mean 83°F.

2.   Soil temperature - average ground temperature 55°F.

f.   Wind and Tornado Loads (symbol W for design wind and W_t for tornado
     wind, and z for pressure differential due to tornado pressure drop)

     The wind and tornado loads, including tornado pressure drop, are
     described in <Section 3.3>.

g.   Seismic Loads (symbol F_{eqo} for OBE and F_{eqs} for SSE)

     Seismic loads are the forces imparted to the structure by ground
     accelerations due to the OBE and SSE. For further details on the
     determination of seismic response refer to <Section 3.7>.

h.   Accident Temperature and Pressure Loads (symbols T_a and P_a)

     Some compartments within the auxiliary building, for example, the
     steam tunnel, RHR heat exchanger rooms and RCIC pump rooms are
     subjected to possible high energy line breaks. Each compartment
     will be designed for the worst postulated accident condition; for a
     discussion on the types of accidents investigated see
     <Section 6.2>.

i.   Penetration, Piping and Equipment Missile Loads (symbols R_o, R_a, Y_r,
     Y_j, and Y_m)

     Penetration, piping and equipment missile loads are as defined in
     <Section 3.8.1.3.1.j> and are in accordance with the definitions of

"Structural Design Criteria for Category I Structures Outside Containment" (Reference 1).

j.   Missile Loads (symbol M)

The postulated missile loads for the design are defined in <Section 3.5.1>.

k.   Explosions and Flammable Vapor Clouds (symbol P_r)

The exterior surfaces of all safety class structures are designed to withstand the effects of the postulated blast loading discussed in <Section 2.2.3>. The design static positive pressures is 2.4 psi on all exterior walls not protected by adjacent structures and 1.2 psi static positive inward acting pressure on the roof.

l.   Cooling Water Tunnel Loads

Loads used in designing the permanent tunnel lining are:

1.   Dead Load Weight (symbol D)

Dead load weight of the lining.

2.   Operating Basis Earthquake (OBE) (symbol F_{eqo})

The permanent lining was designed for loads resulting from an OBE.

3.   Safe Shutdown Earthquake (SSE) (symbol F_{eqs})

The permanent lining was designed for loads resulting from an SSE.

4.   Fluid Pressures (symbol G)

     The permanent lining was designed for differential fluid
     pressures across the lining. The limiting condition for this
     loading was the case of no internal pressure and an external
     pressure equal to the tunnel depth below either ground water
     level or lake level.

5.   Grout Pressure (symbol G')

     The permanent lining was designed for loads induced by
     grouting outside of the permanent lining during construction.
     See <Section 3.8.4.4.2> for further discussion of grouting.

6.   Ground (Rock) Loads (symbol H')

     The permanent lining was designed for a vertical ground load
     equivalent to the weight of a 30 foot height of rock over the
     tunnel crown for the cooling water tunnels and the service
     water tunnels. Based upon the Terzaghi Criteria
     (Reference 12) for rock loads on tunnel supports, this is a
     conservative assumption of rock loads, corresponding to the
     maximum loading which can be expected in material such as the
     chagrin shale at the tunnel locations.

     Existing horizontal compressive pressures in the rock may
     cause horizontal movement of the tunnel walls. Nearly all of
     this movement is expected to occur shortly after excavation,
     before the installation of the permanent lining. Therefore,
     active horizontal rock pressures were not a controlling
     consideration in the analysis of the permanent tunnel lining.

7. Thermal Effects Under Normal Operation (symbol T_o)

The lining was designed for thermal effects induced in the permanent lining by seasonal variations in the intake water temperature. Prior to operation, the permanent lining will be at the ambient rock temperature of approximately 55°F. For design purposes under normal operating conditions, it was assumed that water temperatures in the intake tunnel may vary from 32°F to 81°F with seasonal variations in lake temperature. Water temperatures in the discharge tunnel under normal operating conditions could be up to 21.3°F higher than water temperatures in the intake tunnel.

8. Thermal Effects Due to Safe Shutdown Earthquake (symbol T')

The permanent lining was designed for thermal effects induced by rapid changes in water temperature in the discharge tunnel, if the alternate service water intake tunnel <Figure 3.8-70> is activated. In such a case, the direction of flow in the discharge tunnel would be reversed and the permanent lining could be subjected to a rapid 21.3°F drop in water temperature.

m. Offshore Structure Loads

1. Ice Loads

Basically, two approaches were used to compute the horizontal ice pressures expected at the submerged offshore intake and discharge structures. The first considers a static condition in which the ice is loosely piled and approximates a granular soil. The second considers a dynamic condition, treating the impinging ice as a floating ice island which collides with the submerged structure.

Besides the vertical forces produced by waves <Section 3.8.4.3.1.n>, the intake structure could be subjected to vertical loads due to granular ice accumulation on top of the submerged structure and the hydrostatic loads resulting from a partial blockage of the intake ports. For design purposes it was considered that the hydrostatic pressure acting on the top of the intake heads corresponds to the difference between the maximum registered lake water level and the minimum allowable water level in the emergency service water pumphouse.

(a) Static Loads

The magnitude of the ice and hydrostatic loads on the submerged offshore structures depends on the nature of the ice (Reference 13). If the ice is granular, the static pressure against each of the structures is:

(1) Horizontal active ice pressure   0.85 psi

(122 lbs/ft^2)

(2) Horizontal passive ice pressure   2.83 psi

(407 lbs/ft^2)

(3) Maximum vertical pressure   9.49 psi

(1,367 lbs/ft^2)

(b) Dynamic Ice Loads

If the submerged ridges of a moving large ice floe crush against any of the offshore structures, the dynamic horizontal pressures developed depend on the size of the structure and the impact velocity (Reference 14). The

effective horizontal impact pressure of an ice floe
traveling at a velocity of 3 fps is:

(1) Discharge structure   111.5 psi

                                    (16,056 lbs/ft^2)

(2) Intake structure ice
     protection caissons   185.1 psi

                                    (26,654 lbs/ft^2)

2.   Dynamic Wave Action Loads for Intake and Discharge Structures

Wave action generated during wind storms causes acceleration
of subsurface as well as surface water particles. With both
structures founded on shale bedrock beneath the lake bottom,
scour below the structures due to the movement of water
particles will not occur.

The wave induced vertical differential pressures on a velocity
cap of an intake structure were calculated according to the
theory of Durgin and Shiau (Reference 14). The diameter of
the intake cap used in the calculation was 35 feet. The
intake cap is located approximately seven feet from the lake
bottom as shown in <Figure 3.8-67>. The pressure
distributions were calculated for each 1/8th cycle of wave
position over the structure for water depths between 23 and
34 feet (probable maximum setup, <Section 2.4.5.2>) and
between 4.3 and 8 seconds of wave period. The critical
maximum pressure was provided by the shallow water and found
to be 300 lbs/ft^2 for a 4.2 second wave period. The maximum
total inertial force on the structure due to the horizontal
component of the wave motion is 3.61 psi being exerted over
the projected area (Reference 14).

(a) Wave Loads Summary

The magnitude of wave induced vertical and horizontal
pressures acting on the submerged offshore structures
are:

(1) Vertical upward pressure   +2.08 psi

$(+300 \text{ lbs/ft}^2)$

(2) Vertical downward pressure   -2.08 psi

(-300 lbs/ft^2)

(3) Horizontal inertial pressure   3.61 psi

(520 lbs/ft^2)

3.   Design Loads Summary (Symbol E)

The offshore intake and discharge steel structures are
designed to withstand wave loads and static pressures due to
granular ice accumulations which will act directly upon them.
Protective works consisting of 10 vertical, reinforced
concrete caissons are built around the steel intake structures
to protect them from impact by, and the horizontal dynamic
forces resulting from, large moving ice floes. A 17 foot
diameter protective reinforced concrete ice protection caisson
encases the 3 foot diameter steel diffuser nozzle and provides
similar protection to the discharge structure from the ice
floes (Reference 13) <Section 2.4.7>.

(a) Horizontal Loads

The horizontal pressure used in design of the offshore
intake structures is 3.6 psi. The horizontal pressure
used in the design of the vertical concrete caissons that

form the ice protection for the intake structure is
185.1 psi. The horizontal pressure used in the design of
the concrete caisson around the discharge structure is
111.5 psi.

(b) Vertical Loads

The vertical pressure used in the design of the offshore
structures is 9.49 psi acting vertically downward and
2.08 psi acting upward. These two loads were not applied
simultaneously when designing the structures.

3.8.4.3.2   Loading Combinations

The following combinations of loads were investigated in determining the
most critical conditions for design. Many of the loads vary within a
possible range. For this type of load the specific value used was that
which produced the most critical conditions when combined with other
loads.

3.8.4.3.2.1   Land Based Structures

The land based structures are:

a.   Auxiliary Buildings (2).

b.   Fuel Handling Building (1).

c.   Intermediate Building (1).

d.   Control Complex (1).

e.   Radwaste Building (1).

f.   Underdrain System Manholes (26).

g.   Diesel Generator Building (1).

h.   Offgas Buildings (1).

i.   Emergency Service Water Pumphouse (1).

j.   Condensate Storage Tank Foundations and Dike Walls (2).

k.   Service Water Valve Pit (1).

l.   Electrical Manholes (4).

The loading combinations for these land based structures were divided
into loading combinations for steel structures and loading combinations
for concrete structures and are listed in <Table 3.8-5> and
<Table 3.8-6>.

3.8.4.3.2.2   Tunnels and Offshore Structures

These structures are:

a.   Offshore Intake Structures (2).

b.   Offshore Discharge Structures (1).

c.   Cooling Water Tunnels (2).

The load combinations for these structures are listed in <Table 3.8-7>.

3.8.4.3.3   Extent of Compliance to ACI 349-76 "Code Requirements for
             Nuclear Safety-Related Concrete Structures," American
             Concrete Institute

The extent of compliance to ACI 349-76 is discussed in
<Section 3.8.3.3.7>.

3.8.4.4   Design and Analysis Procedures

3.8.4.4.1   Safety Class Structures Excluding the Cooling Water
             Tunnels

a.   General

     1.   Assumptions and Boundary Conditions for Static Analyses

          (a) Concrete Structures

               The following assumptions were used as part of the
               analytical approach:

               (1) Free body models of parts of the structure are
                    isolated. Support conditions at the boundaries of
                    models are conservatively assumed to be those giving
                    the highest moments and shears in the structure.

               (2) Floors and roofs act as stiff diaphragms to transmit
                    horizontal forces into shear walls.

               (3) Each structure settles uniformly.

               (4) In accordance with ACI 318-71, Section 19.2.1, the
                    material is ideally elastic, homogeneous and
                    isotropic.

(b) Steel Structures

   The following assumptions were used as part of the
   analytical approach:

   (1) Material is ideally elastic, isotropic and
       homogeneous.

   (2) Framed connections are generally assumed to be
       pinned. Where moment resisting connections are
       used, moment continuity is taken into account in the
       analytical techniques. Support conditions at
       boundaries of models are assumed to be fixed or
       pinned as can be justified by the available
       restraint at the boundary.

   (3) Each structure settles uniformly.

2. Assumptions and Boundary Conditions for Dynamic Analyses

   Assumptions and boundary conditions are discussed in
   <Section 3.7>.

3. Analytical Techniques

   (a) Static Analyses

   The final static analyses were performed utilizing the
   following techniques:

Component	Applicable Analytical Technique	
	Concrete	Steel
Single span beams and slabs	Statics See Note[1]	Statics See Note[1]

	Applicable Analytical Technique	
Component	Concrete	Steel
Multispan slabs, one and two way	ACI 318 See Note[2] or moment distribution	not applicable
Multispan beams	Moment distribution See Note[3] or STRUDL See Note[4]	Moment distribution See Note[3] or STRUDL See Note[4]
Frames	Moment distribution See Note[3] or STRUDL See Note[4]	Moment distribution See Note[3] or STRUDL See Note[4]
Walls - single span normal loads	Bureau of Reclamation See Note[5]	not applicable
Walls - multispan normal loads	Moment distribution See Note[3]	not applicable

NOTES:

[1] For determinate structures, moments and shears can be calculated directly by principals of static equilibrium.

[2] ACI 318, Chapter 8, has formulas for calculating end moments for continuous or framed in beams and slabs, having uniform loading.

[3] Moment Distribution: A well known iterative technique for solving continuous beams and frames having one degree of indeterminacy.

[4] ICES STRUDL: A widely used computer program for solving structural models of discrete elements, plates, shells, or combinations of these.

NOTES: (Continued)

[5] "Moments and Reactions for Rectangular Plates:"
(Reference 15) A handbook published by U.S. Dept. of
Interior, Bureau of Reclamation giving coefficients
for calculating edge moments and reactions in plates
and slabs.

(b) Dynamic Analyses

Analytical techniques for OBE and SSE are described in
<Section 3.7.2>.

(1) Seismic loads from OBE and SSE are reduced to
equivalent static loads as described in
<Section 3.7.2>.

(2) Loads from postulated pipe rupture are taken to be
the maximum reaction obtained from the dynamic pipe
rupture analysis. Refer to <Section 3.6> for
description of analysis.

(c) Missile Analyses

Analyses and design for missiles is described in
<Section 3.5>.

4. Design Methods

(a) General

The method of design used for the reinforced concrete
structures is the strength design method as described in

ACI 318-71 with the exception of design for missile
protection which is described in <Section 3.5>.

The method of design for steel is the AISC Specification
for Structural Steel with the following exceptions:

(1) For some load combinations the allowable stresses
     have been increased above those specified in AISC
     Sections 1.5.1, 1.5.2, 1.5.3, 1.5.4, and 1.5.5 as
     specified in <Table 3.8-5>.

(2) For load combinations on local areas that include
     missile loads or pipe whip, the design methods are
     discussed in <Section 3.5> and <Section 3.6>,
     respectively.

(b) Deformations and Deflections

No permanent deformations of any concrete structure are
allowed for the load combinations of <Section 3.8.4.3.2>
with the exception of missile loads when some local
plastic deformation is allowed as described in
<Section 3.5>.

For structural steel members the only permanent
deformation allowed is in the pipe restraint collars
which are designed into the plastic range as discussed in
<Section 3.6>.

b. Auxiliary Building

The concrete floor slabs and roof were analyzed using beam or plate
theory and were designed as one or two-way slabs, or a combination
of both, depending on the proportions of the slab. The steam

tunnel slab, which is comprised of two separate slabs at different elevations but connected to one common concrete beam, was analyzed using moment distribution, while the beam was analyzed using MASS 01, a finite difference method solver. The roof slab over the steam tunnel was analyzed using moment distribution, and the circular roof beam at the reactor end of the steam tunnel was analyzed considering finite elements using ICES-STRUDL.

Distribution of the seismic forces on the walls was accomplished by hand calculations considering the building geometry and the mass stiffness of building elements. Lateral loads were carried to the foundation by a system of slabs and shear walls. The shear walls were designed for shear and flexural requirements for both in-plane and out-of-plane loads. Vertical slab loads were transmitted to the columns and walls.

c.   Fuel Handling Building

The design and analysis methods used for this structure are discussed for the following:

1.   Floor Slabs

The floor slabs are 2 or 3 feet thick as dictated by design or shielding requirements. Analysis of the various slab elements was done by the computer program MASS 01, a finite difference method solver.

2.   Fuel Pool Walls

The fuel pool walls are integrally connected to the slabs at Elevations 620'-6" and 599'-0" and the foundation mat at Elevation 574'-10". Walls vary in thickness from 4 to 7 feet. The analysis includes all loads as prescribed in

<Section 3.8.4.3> as well as hydrodynamic loads due to the
pool water excitation during a seismic event. Determination
of these loads utilized a procedure outlined in "Earthquake
Engineering for Nuclear Reactor Facilities," John A. Blume &
Assoc (Reference 16). The computer program MASS 01 was used
to determine the design forces.

3.   Walls Above Grade Elevation 620'-0"

   (a) Exterior Walls

        Exterior walls on the north, east and south sides of the
        fuel handling building are typically 3 feet thick
        stiffened vertically, for their full 60 feet height by
        3 foot by 6 foot pilasters on the exterior wall-face.
        The pilasters are analyzed and designed primarily as
        flexural elements lending no appreciable membrane shear
        stiffness. The walls are considered the mechanism for
        shear transfer to the mat and to resist vertical loads.

   (b) Interior Walls

        The interior walls above grade are considered to act as
        shear walls for the distribution of seismic loads to the
        overall structure only where they are continuous from the
        roof to the floor slab at Elevation 620'-6"; however, the
        analysis does consider the shear load in each wall for
        the design of that wall.

4.   Fuel Handling Area Crane

   The bridge for the fuel handling area crane spans 66 feet
   10 inches between the east exterior wall and the interior wall
   on column line IB-8 as shown in <Figure 1.2-8>. Continuous

reinforced concrete corbels support the crane rail at
approximately Elevation 660'-6". The corbels are designed for
a seismic event occurring when the crane is loaded at its
rated capacity.

5.   Roof Slab and Supporting Steel

The two foot roof slab is designed as a continuous beam
spanning across the steel roof girders. Girders are assumed
simply supported. Intermediate spanner beams are only
designed to support the weight of the steel decking and wet
concrete.

6.   Seismic Considerations

The fuel handling and intermediate buildings are tied together
by a common mat and at the common wall at column line IB-7 as
shown in <Figure 1.2-13>. This is achieved by continuous
reinforcement from intermediate building slabs at
Elevations 599'-0" and 620'-6" and the roof at
Elevation 682'-6" into the counterpart slabs and fuel pool
walls in the fuel handling building. Seismic load
distribution was accomplished by hand calculations according
to the geometry and stiffness of supporting walls. Slabs were
assumed to act as rigid diaphragms. Lateral and vertical
loads were transferred by the walls to the foundation slabs.
See <Section 3.8.5.1> for a discussion of the foundation
configuration and the design approach on supporting caissons.

d.   Intermediate Building

Concrete floor slabs and the roof were designed as one or two-way
slabs using geometry aspect ratio as a criterion. The floors and
roof were analyzed using plate theory or the computer program

MASS 01. Where plate theory was used, parametric studies were performed to obtain upper bound solutions. The boundary conditions included in the model were based upon stiffness considerations.

Lateral slab loads were carried in diaphragm action to the walls and vertical slab loads were transmitted to columns and walls.

The shear walls were analyzed for shear and flexural requirements including the effects of penetrations.

e.   Control Complex

The structural steel frame consisting of floor beams and columns is designed to carry the vertical floor loads only. The floor beams are designed as simple spans and the columns are designed for axial loads. The steel decking spanning beams are provided as permanent formwork for concrete slabs.

The concrete structure consisting of floor slabs and external walls completely encloses the steel frame structure, the concrete walls being independent of steel columns. The concrete structure is designed to take all lateral loads on the building.

All floor slabs except those at Elevations 638'-6" and 654'-6" are designed as simply supported, spanning between the steel beams. The slabs at Elevations 638'-6" and 654'-6" supporting heavy loads due to cable tray and duct hangers are designed to act together (composite) with the steel beams.

The walls typically span one way between floor slabs, supporting soil and water pressure below grade in addition to wind and seismic forces. The exterior walls and roof also act as missile barriers. Seismic load distribution was accomplished by hand calculations according to the geometry and stiffness of supporting walls. Slabs

are assumed to act as rigid diaphragms. Lateral and vertical loads
are transferred by the walls to the foundation.

For missile design capability, the method described in
<Section 3.5.3> was used.

f.   Radwaste Building

The concrete floor slabs and roof were analyzed and designed as
spanning either one way or two ways. The walls were designed for
lateral and vertical loads. The seismic load distribution was
accomplished by hand calculations considering the geometry and
stiffness of the walls. Floors and roof slab were considered as
rigid diaphragms in transferring lateral loads. Lateral loads were
carried down to the foundation by shear walls. The shear walls
were designed for shear and flexural requirements for both in-plane
and out-of-plane loads. Vertical loads were taken by internal
walls, columns and external walls.

The roof slab and exterior walls were designed for missile
protection, using the methods as described in <Section 3.5.3>.

Loads from the nonsafety portion are considered in the design of
the safety class portion of the radwaste building. The nonsafety
portion attaches to the safety class portion at the transverse
concrete walls so that the loads are directly supported by these
walls acting in shear.

g.   Diesel Generator Building

The concrete floor and roof slabs were analyzed using beam theory
and were designed as one-way slabs.

Distribution of the seismic forces on the walls was accomplished by hand calculations considering the geometry and stiffness of the walls. Floors and roof slab were considered as rigid diaphragms in transferring lateral loads. Lateral loads were carried to the foundation by a system of slabs and shear walls. The shear walls were designed for shear and flexural requirements for both in-plane and out-of-plane loads. Vertical slab loads were transmitted to the walls.

The roof slab and exterior walls were designed to prevent missile penetration in accordance with <Section 3.5.3>.

h.   Offgas Building

All loads acting on the offgas building are transmitted by the exterior walls, interior walls and columns to the foundation mat which bears on structural fill.

i.   Emergency Service Water Pumphouse

The concrete floor and roof slabs, as well as the concrete walls, were analyzed by breaking them down into beams and slabs. Then beam and plate theory was applied for solving the particular elements.

Seismic analysis was performed using the ICES-DYNAL program as applied to a lumped-mass cantilever model. Distribution of the seismic forces on the walls was accomplished by hand calculations. Lateral loads were assumed to be carried to the foundation by the roof and floor slabs, and by qualified vertical shear walls.

In addition, the slabs and walls above ground level were designed to withstand local seismic accelerations; those below ground level and submerged in water were designed to resist the forces of the

sloshing earthquake effect; finally, those below ground level
retaining the backfill were designed to resist the dynamic
(earthquake) soil pressure.

The shell of the structure (exterior above ground walls and the
roof) was also analyzed for tornado and missile loads. Further,
the exterior walls below grade level were checked for the extreme
environmental loads of the water table to grade.

j.   Underdrain System Manholes

To determine seismic forces the underdrain manholes were analyzed
using the computer program LUSH which is described in
<Section 3.8.4.4.3>. The computer model included the manhole and
surrounding soil. The strain dependent properties of soil were
taken into account. Ground accelerations were input at the base of
the model. The resulting accelerations of the manhole and the
dynamic soil pressures were used in the design of the manholes in
addition to static soil pressures.

k.   Electrical Manholes

The electrical manholes were designed to resist static plus dynamic
(seismic) soil pressures on the walls. The roofs were designed to
resist tornado missiles. In particular the removable 30 inch deep
beams were designed to resist a maximum equivalent static load
determined from consideration of the input of tornado missiles on
the roof of the manholes. See <Section 3.5.3> for a discussion of
missile barrier design using equivalent static loads.

l.   Condensate Storage Tank Foundations and Dike Walls

The concrete ring girder and foundation mat were designed using
beam theory. The soil pressures were calculated assuming the

circular foundation mat acts as a rigid plate. The concrete dike wall was analyzed using Kalnins KSHEL1 computer program as discussed in <Section 3.8.4.4.3>.

The dike wall was designed for missile penetration using the methods described in <Section 3.5.3.3>. Lateral loads acting on the dike wall were transmitted to the foundation mat.

m.   Service Water Valve Pit

The service water valve pit was analyzed for lateral earth pressure loads, hydrostatic loads, HS-20-44 truck loads, and seismic loads.

The concrete mat and roof were analyzed using beam theory and were designed as one-way slabs. The soil pressures were calculated assuming the foundation mat acts as a rigid plate. The walls were designed for one-way action using beam strips. Lateral loads acting on the walls were transmitted to the foundation mat.

The hatch cover portion of the roof was designed for missile penetration using the methods described in <Section 3.5.3.3>.

n.   Intake and Discharge Structures

These fabricated steel structures were analyzed using elastic beam and slab theory for the component elements. The analyses considered the static and dynamic forces produced by the fluid movements around and through the intake and discharge structures during operation, horizontal and vertical wave and seismically induced forces, as well as the structural dead weight loads. The discharge structure was designed to sustain the fully effective vertical and horizontal impact pressures from ice flows, whereas the intake structures, which are each encircled by ice protection caissons, were designed to sustain a lower horizontal ice impact

pressure. In addition, the discharge structure was designed to
sustain the thermal gradient caused by an emergency flow reversal.

3.8.4.4.2   Cooling Water Tunnels

a.   Rock - Structure Interaction

The bedrock through which the tunnel was driven is a relatively
uniform clay shale of 3,500 psi average unconfined compression
strength. Jointing and bedding is generally tight. Extensive
zones of intense fracturing or alteration have not been encountered
during subsurface investigations, although a low angle, small
displacement, thrust fault was encountered during excavation of the
tunnels <Section 2.5> and <Appendix 2D>. Laboratory tests indicate
the rock does not have a high swell potential. The vertical in
situ ground pressure at tunnel level where the tunnels are under
the greatest depth of cover is approximately 184 psi, computed as
follows:

1.   35 feet of lacustrine deposits
     and upper till   at 130 pcf = 4,550 psf

2.   25 feet of lower till   at 140 pcf = 3,550 psf

3.   105 feet of chagrin shale   at 160 pcf = 18,400 psf
                                                      26,500 psf
                                                  =   184 psi

This vertical in situ ground pressure is very low with respect to
the average unconfined compression strength of the shale.

Because of the existence of in situ compressive stresses in the
rock mass around the tunnel and the relatively low level of
stresses induced in the rock by the presence of the tunnel, the

anisotropic strength of the rock and the inability of the rock to resist tensile stresses are not significant, thus a non-linear analysis was not attempted.

Tunnels are stable without support in such rock if it were not for progressive failure mechanisms such as loosening of the rock along pre-existing joints and bedding planes, or surface deterioration and raveling due to water action, air slaking and stress relief, which allow rock from the arch to fall under the influence of gravity. The temporary support restrained these progressive failure mechanisms during tunnel construction. The permanent concrete lining will prevent their subsequent development.

After the permanent concrete lining was constructed, cement grout was injected under pressure through grout holes in the tunnel crown into the boundary between the concrete lining and the surrounding rock mass. This grout injection serves to:

1. Fill any voids which remained between the concrete lining and the surrounding ground.

2. Penetrate any cracks or voids in the rock which may have pre-existed or which may have developed as a result of tunnel excavation.

3. Prestress the permanent concrete lining and transfer to the concrete lining any rock loads which may have been supported by the temporary supports.

As a result of these construction procedures, the permanent tunnel lining was analyzed as an elastic structure completely enclosed in and bonded to the surrounding elastic rock mass.

b.   Analytical Techniques

1.   Thermal Stresses

Stresses induced by the most critical temperature gradients
were superimposed upon the other load combinations to
determine net stresses in the lining.

2.   Static Loads

(a)   Intake, Discharge and Emergency Service Water Tunnels

Behavior of the concrete lining in response to static
dead load D, water pressures G, grout pressures G', and
ground loads H', was analyzed with a finite element model
using the GENSAP code. GENSAP analyzes one, two or
three-dimensional structural systems using the finite
element approach. The code handles most common
structural elements, and both static and dynamic
loadings. Static structural problems are solved by a
matrix displacement method of linear structural analysis.

The finite element model for the static analysis is shown
in <Figure 3.8-77>, with the boundary conditions shown.
The overhead boundary is the ground surface; lateral and
bottom boundaries were chosen sufficiently far from the
tunnel to be unaffected by expected tunnel response.
From elastic theory, static effects decay with distance
from the tunnel in proportion to $(a/r)^2$ where a = tunnel
radius and r = radius to the point in question. For the
static analyses, the minimum boundary distance from the
center of the 5 foot radius tunnel is 50 feet, i.e.,
$(a/r)^2 = 1/100$. Thus, at the model boundaries, static

effects have decayed to one percent of their magnitude at the tunnel boundary.

The tunnel liner was modeled with two layers of finite plane strain elements, permitting computation of bending stress in the lining. The surrounding shale was also modeled by plane strain elements. The plane strain finite element is an isoparametric element that can develop variable strain distribution within the element, and is capable of handling both normal and shear strain.

(b) Shafts and Elbows

The behavior of the shafts and elbows were extrapolated from the results of the finite element analysis of the intake, discharge and emergency service water tunnels.

3. Seismic Loads

(a) Intake, Discharge and Emergency Service Water Tunnels

Behavior of the concrete lining in response to seismic loads F_{eqo} and F_{eqs} was analyzed with a finite element model. Both time marching and modal analysis were provided. For the time marching method, the dynamic version of the GENSAP code computes the dynamic response of the linearly elastic system to time dependent ground acceleration in the horizontal and vertical direction, input on the left hand side of the model. The dynamic response is obtained by integrating the equations of motion directly with respect to time. The modal extraction technique used for the modal analysis is a form of the Rayleigh-Ritz method which involves a

transformation from the nodal coordinates of the finite
element formulation to a smaller number of generalized
coordinates. The transformation matrix is obtained from
the displacements of the model due to a prescribed set of
force patterns.

The finite element model for the dynamic time marching
analysis is shown in <Figure 3.8-78> and is the same as
for the static case, except the lateral and bottom
boundary conditions are changed. Ground motions are input
on the left hand side. Both the right hand side and
bottom boundaries were extended further from the tunnel.
Both right hand and bottom boundaries are energy
absorbing boundaries which allow the stress waves to
travel out of the model and thus prevent the formation of
fictitious boundary reflections. A symmetrical model
similar to the static model in <Figure 3.8-77> was used
for the modal analysis except that the bottom and
vertical boundaries were located 75 feet from the
centerline of the tunnel.

In the analysis, response spectra for various points in
the model were calculated for the time marching analysis
and compared with the input spectrum. The model does not
display any significant tendency to modify the input
motion, hence, it is concluded to be satisfactory. The
boundaries of the modal analysis model were selected
using results of the time marching solution to make sure
that boundary interference was insignificant.

(b) Shafts and Elbows

The dynamic response of the shafts and elbows were
analyzed using equivalent static load procedures

extrapolated from the results of the finite element analysis of the intake, discharge and emergency service water tunnels.

c.   Expected Behavior Under Loads

In accordance with the concepts of flexible tunnel linings and soil-structure interaction, as presented by Peck, Hendron and Mohraz (Reference 17), final design and analysis indicates that the concrete permanent lining behaves as a relatively flexible lining in a stiff surrounding mass, in response to external pressures and ground loads. This means external pressures, loads and resulting reactions, which tend to develop, are carried by the lining primarily in uniform circumferential compression (tangential thrust), with small bending moments and eccentricity of loads. Design and analysis indicate that reinforcing steel is not necessary in the permanent lining, except at discontinuities such as tunnel junctions.

The resultant stresses computed are within code allowable limits for structural plain concrete, including ultimate load factors which equaled or exceeded those required by the standard review plan. Based on experience gained from the behavior of similar tunnels, the Perry tunnels are considered adequate (Reference 18).

d.   Design Methods

1.   General

The permanent concrete lining is designed according to strength design concepts set forth in ACI 318-71, as modified herein, and as described in detail in (Reference 18).

2.   Unreinforced Sections

Unreinforced concrete linings are designed using ACI 322-72 as a supplemental to ACI 318-71.

For combinations of moment and thrust in which the full thickness of the lining falls within the stress block defined by ACI 318-71, Section 10.2.7, strength of the unreinforced concrete permanent lining is calculated by:

$$P_u = \phi \ (0.85f'_c) \ A_g$$

The capacity reduction factor ϕ is taken as 0.75. This value of ϕ is justified by the redistribution of stress in the surrounding ground, reduction of load on the tunnel lining and redistribution of stress within the lining in response to deformations of the lining.

The ACI Committee Report, Commentary on ACI 318-71, Section 9.1.1, Page 29, explains the basis for the normal capacity reduction factors. Paragraph 5 of Section 9.1.1 of the commentary provides for modification of normal capacity reduction factors for structures and loadings " . . . materially different from those encountered in the normal design situations."

3.   Reinforced Sections

The permanent concrete lining at tunnel junctions was designed as a reinforced concrete member subjected to combined flexure and axial load, in accordance with ACI 318-71, Section 10.

3.8.4.4.3   Computer Programs

a.   Computer Programs Used in the Structural Analyses of Safety Class
     Structures other than Containment

     1.   ICES-STRUDL-II

          For program description and computer hardware/software see
          <Section 3.8.3>.

     2.   GENSAP

          For program description and computer hardware/software see
          <Section 3.8.4.4.2.b.2.(a)>.

     3.   MASS-01

          MASS-01, "Mat and Slab Solver," is a user oriented computer
          program which solves mat, slab and wall bending problems. The
          program utilizes finite difference theory to form a
          symmetrical matrix in order to solve the uncoupled
          Lagrange-Germain-Huber variable thickness form of the
          biharmonic partial differential equation for steady-state
          transverse slab loading. The slab is divided into a discrete
          number of node points so that the resulting network forms the
          basis of the matrix.

          The boundary conditions which MASS-01 solves are not limited
          to the classical boundaries: clamp, simple and free. A slope
          deflection approach to the boundary conditions also enables
          the user to obtain solutions to elastic boundaries. Thus, the
          user can specify, for both interior and exterior boundaries, a
          geometry consisting of walls and columns perpendicular to the
          slab and composite or non-composite beam framing. The effect

of interaction, both bending and torsion, is also included so that the computer model effectively represents the structure.

4.   KALNINS Static Analysis of Axisymmetric Shells

KALNINS uses a multisegment method of direct numerical integration of boundary value problems and was developed by Aturs Kalnins and published in the Journal of Applied Mechanics, Vol. 31, September 1964, pp. 467-476, and in the Journal of the Acoustical Society of America, Vol. 36, July 1964, pp. 1355-1365. The program calculates elastic deflections and stresses in a thin-walled, axisymmetric shell when subjected to any arbitrary surface, edge and/or ring loads. The solution is based on the linear theory of elasticity and takes into consideration bending as well as membrane action of the shell in response to applied load. Results are in terms of resultant forces and couples with stresses calculated by assuming a linear distribution through the thickness.

This program has been widely used for thin shell analysis since its release to the public domain in 1968. The program is being run on Gilbert Associates Inc., Reading, PA, IBM 370/158 computer under IBM operating system MVS 3.7 with JES 2 and also on the Control Data Corp. CDC computers of the Rockville, MD, computer center.

5.   ICES-DYNAL

See <Section 3.8.1.4.6.a.1>.

6. LUSH - Response Analysis of Soil Structure System

LUSH is a finite element program for earthquake analysis of plane structures. In an approximate manner, it takes into account strong non-linear effects which occur in soil masses when subjected to strong earthquake motions. It uses a combination of the equivalent linear method described by Seed & Idriss (1969) and the method of complex response with complex moduli.

This program has been widely used in the public domain. The program is being run on the United Computing Systems, Inc., CDC 175 computer system.

b. Design Control and Verification of Computer Programs

The methods used for the design control and verification of the computer programs used in the analysis of safety class structures are discussed in <Section 3.8.1.4.6.b>.

c. Verification of Program Usage

1. ICES-STRUDL-II

For program verification refer to <Section 3.8.3>.

2. GENSAP

GENSAP is an expanded version of the SAP computer program. SAP is a well recognized program in the public domain. The SAP program was developed under the direction of Professor E. L. Wilson at the University of California, Berkeley, California, and is available to the public through

the National Information Service - Earthquake Engineering,
729 Davis Hall, Berkeley, California. This program is written
completely in FORTRAN IV language and is operable on a
CDC 6600 computer. The date of release of the SAP program to
the public domain was September 1970. The dynamic analysis by
normal mode in GENSAP is based entirely on SAP, except for the
modifications needed to change the output capabilities and
formats. The GENSAP analysis was performed on a Univac 1108
computer, by Agbabian Associates, Los Angeles, California.
GENSAP has been used extensively at Agbabian Associates for
all types and sizes of problems since March 1971. GENSAP has
also been used by other organizations such as the U.S. Army
Corps of Engineers, Huntsville, Alabama, and Systems
Professional in Los Angeles, California. Test problems have
been run to verify that the GENSAP computer program when run
on the Univac computer gives the same results obtained for the
same problems when run on the CDC 6600 computer with the SAP
program by the University of California, or as obtained from
classical analytical solutions. Documentation of GENSAP is
traceable to Agbabian Associates, Los Angeles, California.

3.   MASS-01

MASS-01 has been verified by comparison to classical solutions
for plates and shells. The documentation of the program is
available at Gilbert Associates, Inc., Reading, PA.

4.   KALNINS

This program has been widely used in the public domain for the
past five years. For the analysis of safety class structures,
the results obtained, using the particular revision and
computer hardware/software combination, have been verified by

comparison of test results with previously obtained solutions
and test problems supplied with the program manual.
Documentation for the program is traceable to the author,
Dr. Arturs Kalnins, Lehigh University, Pennsylvania.

5.   ICES-DYNAL

See <Section 3.8.1.4.6.c.1>.

6.   LUSH

This program has been widely used in the public domain. For
the analysis of Safety Class structures, the results obtained,
using the particular version and computer hardware/software
combination, have been verified by comparison of results to
published test problems. Documentation of this program is
traceable to the University of California - Berkeley.

3.8.4.5   Structural Acceptance Criteria

a.   Stress and Strain Criteria for Concrete Structures Excluding the
     Cooling Water Tunnels

     The stress criteria will be based on the strength design concept of
     ACI 318-71. The factored loads in the load combination are
     discussed in <Section 3.8.4.3>. The stress criteria and load
     combinations are modified by the use of the capacity reduction
     factor (ϕ) given in Section 9.2 of ACI 318-71. Thus stresses and
     strains will be within the elastic limit for the postulated loads
     and load combinations. Any displacements will be elastic, such
     that gross permanent deformations will not occur.

b. Structural Steel Stress Criteria

1. The stress criteria will be based on the recommendations of
    the AISC specification for the loads and load combinations set
    out in <Section 3.8.4.3>. Thus the elastic design approach
    will not result in the basic structure experiencing gross
    deformations except for local loads, <Section 3.5> and
    <Section 3.6>.

2. The stress criteria for pipe whip restraints will allow gross
    deformation of the restraint to occur. The deformation
    imparted to the restraint enables the energy in the ruptured
    pipe to be dissipated and controlled. The deformation of the
    restraint will not cause deformation of the anchoring
    structure. Thus the deformation is limited to the restraint.
    The stress criteria, analytical techniques and design methods
    for the pipe whip restraints is given in <Section 3.6>. For
    deformation criteria for missiles refer to <Section 3.5>.

c. Cooling Water Tunnels

1. Permanent Lining

    (a) Factor of Safety

         The nominal factor of safety of the permanent lining
         under the postulated loadings is determined by the load
         factors and capacity reduction factors used in design.

    (b) Deformation

         Deformations of the lining will be the combined elastic
         and creep deformations experienced under working loads.

(c) Longitudinal Behavior

Well distributed fine circumferential tensile cracking in
the concrete permanent lining due to concrete shrinkage,
temperature changes and longitudinal dynamic forces in a
seismic event, will not impair the tunnel functions, and
will be accepted. Longitudinal reinforcing is not
provided to restrain these effects. Concrete shrinkage
is minimized by:

(1) Curing of the concrete as specified in
     <Section 3.8.4.6.2.b.6.(h)>.

(2) The high humidity and moderate temperature of the
     normal tunnel environment during construction.

(3) The continuously saturated condition of the lining
     during normal plant operation.

Well distributed cracking, if cracking does develop, will
be ensured by the mechanical interlock which exists
between the cast-in-place concrete lining and the
irregular rock surfaces of the tunnel excavation. This
mechanical interlock will prevent longitudinal strains
from accumulating over long distances and developing
large cracks.

(d) Tunnel Faulting Considerations

The very limited extent of the fault (gauge thickness
0.1 ft; <Section 2.5> and <Appendix 2D> in comparison to
the lining thickness mitigates its ability to influence the
tunnel linings structural capacity in the circumferential

direction. From the standpoint of seismic design, the
non-seismic fault is of no significance other than the fact
that it represents a narrow zone in which the elastic
modulus of the material has been slightly altered
(Reference 18).

3.8.4.6   Materials, Quality Control and Special Construction
              Techniques

3.8.4.6.1   Summary of Material Specifications and Quality Control

The material specifications and quality control provisions are
summarized in <Table 3.8-8> giving cross references to applicable
sections of the USAR. Cadweld splices are discussed in
<Section 3.8.1.6>.

3.8.4.6.2   Cooling Water Tunnels and Offshore Structures

a.   Standards and Specifications

     1.   ACI (American Concrete Institute)

          ACI 301-72, "Specifications for Structural Concrete for
          Buildings."

          ACI 304-73, "Recommended Practice for Measuring, Mixing,
          Transporting, and Placing Concrete."

          ACI 305-72, "Recommended Practice for Hot Weather Concreting."

          ACI 306-66, "Recommended Practice for Cold Weather
          Concreting."

          ACI 308-71, "Recommended Practice for Curing Concrete."

ACI 309-72, "Recommended Practice for Consolidation of Concrete."

ACI 318-71, "Building Code Requirements for Reinforced Concrete."

ACI 347-68, "Recommended Practice for Concrete Formwork."

ACI Title No. 59-57, "Durability of Concrete in Service."

ACI Title No. 68-33, "Placing Concrete by Pumping Methods."

2.   ASTM Standards (American Society for Testing and Materials)

ASTM A 36-74, "Structural Steel, Spec. for."

ASTM A 325-74, "High Strength Bolts for Structural Steel Joints Including Suitable Nuts and Plain Hardened Washers, Spec. for."

ASTM A 441-75, "High Strength Low-Alloy Structural Manganese Vanadium Steel, Spec. for."

ASTM A 578-71b, "Straight-Beam Ultrasonic Examination of Plain and Clad Steel Plates for Special Applications, Spec. for."

ASTM A 615-74, "Deformed and Plain Billet-Steel Bars for Concrete Reinforcement, Spec. for."

ASTM C 109-73, "Standard Method of Test for Compressive Strength of Hydraulic Cement Mortars."

ASTM C 309-74, "Liquid Membrane-Forming Compounds for Curing Concrete, Spec. for."

ASTM E 109-63, "Dry Power Magnetic Particle Inspection."

3. AWS (American Welding Society)

    AWS D1.1-75, "Structural Welding Code," including revisions
    AWS D1.1-Rev. 1-76 and AWS D1.1-Rev. 2-77.

4. Federal and Military Specifications

    (a) Military Specification MIL-P-23236, "Paint Coating
         Systems, Steel Ship Tank, Fuel, and Salt Water Ballast,"
         December 17, 1965.

    (b) CRD C 588-76, Corps of Engineers' Standard "Method of
         Sampling and Testing Expansive Grouts."

    (c) CRD C 79-58, Corps of Engineers' Standard "Method of Test
         for Flow of Grout Mixtures."

5. American Society of Mechanical Engineers (ASME) Boiler and
    Pressure Vessel Code, Section II (Parts A and C), III
    (Division 1), V, and IX; 1977 Edition with addenda for Winter
    of 1978.

b. Material

1. Cooling Water Tunnels and Offshore Structures

    Concrete and reinforcing steel used in the permanent concrete
    tunnel lining and the offshore structures conformed to the
    materials and quality control provisions of
    <Section 3.8.1.6.1.1>, <Section 3.8.1.6.1.2>,
    <Section 3.8.1.6.1.3>, <Section 3.8.1.6.1.4>, and
    <Section 3.8.1.6.1.5> (Concrete Construction - Codes and

Standards through Concrete Quality Control) and of
<Section 3.8.1.6.2> (Reinforcing Steel) except as modified
below:

(a) Minimum concrete compressive strength of 4,000 psi at
     28 days.

(b) Minimum of 6.5 bags of cement per cubic yard of concrete.

(c) Maximum aggregate size of 1-1/2 inches, except for the
     ice protection caissons which is 3/4 inch.

(d) Consistency

     (1) The maximum slump for concrete at the point of
          placement, prior to being compacted by approved
          mechanical vibrators, was five inches.

     (2) A tolerance of up to one inch above the indicated
          maximum was allowed for individual batches provided
          the average for all batches or the most recent ten
          batches tested, whichever is fewer, did not exceed
          the maximum limit.

2.   Offshore Structures

(a) Materials

     (1) Steel conforms to ASTM A 441 (discharge structures)
          and ASME SA-299 (intake structures).

     (2) Steel plates were inspected by ultrasonic inspection
          methods conforming to ASTM A 578 or ASME SA-578, and

visual examination conforming with the ASME Code, Section V, Article 9.

(3) Steel plates and structural steel were certified by the manufacturer prior to use.

(b) Welding

(1) Welding was performed with either the shielded metal arc, submerged arc or flux cored arc welding process.

(2) All welding procedures and qualification of welders and welding operators conformed to the requirements of the American Welding Society (AWS) D1.1 or Section IX of the ASME Code.

c. Quality Control

1. Cooling Water Tunnels

See <Section 3.8.1.6.1.5> and <Section 3.8.1.6.2.3>.

2. Offshore Structures

(a) Testing

(1) Welds were radiographed. Radiographing was performed in accordance with the requirements of AWS D1.1 or the ASME Code (Section V), and as herein specified. Radiographing included the furnishing of equipment, materials and labor for the radiographing of 25 percent of the total number of linear feet of full penetration groove welds.

(2) Welds showing defects were removed and repaired as specified. Additional radiography was performed, the frequency and extent of which depended on the number of defective welds, as specified.

(3) Magnetic particle inspection was required for inspection of fillet welds. Magnetic particle examination was in accordance with Section 6.7 of AWS D1.1 or Section V of the ASME Code. This examination was performed for the root pass and final surface only.

d. Special Construction Techniques and Requirements - Cooling Water Tunnels

Tunnel construction requirements are outlined as follows:

1. Excavation and Temporary Support

(a) Excavation Methods

Tunnel excavation was primarily performed by tunneling machines, although conventional drill and blast techniques were also employed.

(b) Tunneling Machines

Tunneling machines were of a type which afford adequate protection against loss of ground and permit placement of ground support as required by ground conditions.

Methods used to advance the tunneling machines insured their correct alignment at all times, without blinding or imposing excessive loads on the surrounding ground.

(c) Blasting

Blasting was permitted only after precautions were taken for the protection of all persons, work and property. Blasting within 50 feet of adjacent structures or concrete, except shotcrete, was permitted only after submission by the contractor of a plan showing the relative positions of the structures or concrete, the area to be blasted and the blasting technique to be employed. All adjacent structures and concrete were protected by limiting the size of blasts, by covering blasts with blasting mats or by other means to prevent damage from shock waves or fly rock.

Ground vibrations were seismographically monitored by measuring the peak particle velocity of the ground in the vicinity of blasting. The peak particle velocity at any structure in the vicinity of blasting operations did not exceed two inches per second. Peak particle velocities not exceeding three inches per second were permitted in the vicinity of cast-in-place concrete placed as part of the tunnel construction contract.

(d) Lines of Excavation

Minimum lines of excavation were shown on the drawings and designated as the "A" lines. No unexcavated material, timbering or portions of steel supports except as noted on the drawings were permitted to project within the "A" lines.

(e) Temporary Support

The minimum temporary support system required to support the tunnel excavation consisted of rock bolts supplemented with steel strapping and wire mesh as described in <Section 3.8.4.1.12.b>. Alternative or additional temporary support systems which were permitted to be used for varying ground conditions were shotcrete and steel supports. These methods were included in the approved construction specifications although they were not required to be used in the work. See <Figure 3.8-71> and <Figure 3.8-72> for details.

(f) Ventilation

Geological investigations indicated the presence of gas in the rock formation through which the tunnels were driven.

The detailed construction specifications included gas detection and control requirements such as ventilation of dangerous concentrations of gas, testing for gas concentrations, shutting down the work if dangerous gas levels were encountered, and establishing safe working conditions for resumption of the work.

2. Rock Bolts

(a) General

The rock bolt layout shown on <Figure 3.8-71> was the minimum required temporary support system. Modifications as required to suit ground conditions included variations

in diameter, pattern, spacing, length of rock bolts,
anchorage, and bolt assemblies. The beneficial effects
of the temporary supports were neglected in the design
and analysis of the permanent concrete tunnel lining.

3.   Tunnel Lining

Construction of the cast-in-place concrete permanent tunnel
lining met the following requirements:

(a) Tolerances

(1) Tolerances for cast-in-place concrete tunnel lining
conformed to the following:

Departure from or return to
established alignment not
greater than . . . . 1 inch per 100 feet

Maximum departure from
established alignment . . . . 3 inches

Maximum departure from
established grade . . . . . . 1 inch

Variation in thickness at
any point . . . . -0

Variation from inside
dimensions . . . . 1/2 of 1 percent

Variation from plumb for shafts:

in 10 feet . . . . 1/2 inch

in 40 feet or more . . . 2 inches

(2) Tolerances for placing of reinforcing steel bars
conformed to the following:

Variation from 2 inch cover
specified

Inside bars . . . -1/4 inch, +(1-1/4) inch

                    Outside bars . . . -0, +1/2 inch

              Variation from 3 inch
              cover specified . . . . ±1/2 inch

              Variation from indicated
              spacing . . . . ±1 inch

    (b) Formwork

         (1) General

              Design, construction, inspection, and stripping of
              forms, and material used in forms were in accordance
              with ACI 347-68, except as modified by the approved
              construction specifications.

         (2) Forms For Concrete Tunnel Lining

              Forms for the tunnel lining consisted of steel and
              were constructed in such lengths that each concrete
              placement could be completed without intervening
              cold joints.

    (c) Preparation for Placing

         (1) General

              Water in the tunnel was controlled so that at no
              time during placement or hardening of the concrete
              would water wash, mix with or seep into the
              unhardened concrete.

(2) Surface Preparation

All surfaces against which concrete was placed was free from standing and running water, mud, debris, oil, frost, ice, and loose material.

(3) Joints in Concrete

Construction joints were clean and damp, but not wet, when covered with fresh concrete. Cleaning consisted of the removal of all laitance, loose or defective concrete, coatings, sand, curing compound, and other foreign material. Construction joints were washed thoroughly with air-water jets or wet sand-blasted or both, as approved by the resident tunnel engineer. Excess water was removed before the new concrete was placed.

(d) Transporting Concrete

The methods and equipment used for transporting or conveying concrete from the point of delivery and the time that elapsed during transportation or conveyance was such as would not cause appreciable segregation of coarse aggregate in the concrete as it was incorporated into the work.

(e) Placing

(1) General

Placing and consolidation of concrete was performed in accordance with ACI 301-72, ACI 304-73 and ACI 309-72, except as modified by the approved

construction specifications. In the event of
conflict, ACI 301-72 governed. Concrete pumping
work was in accordance with ACI 301-72, Chapter 9 of
ACI 304-73 and ACI Title No. 68-33.

(2) Concrete Tunnel Lining

Concrete tunnel lining was formed and placed for the
full circular tunnel section in one continuous
operation.

Concrete in the full circular section was placed by
pumping or other approved methods. The equipment
used in placing the concrete and the method of its
operation was such as would permit introduction of
the concrete into the forms without high velocity
discharge and resultant segregation.

(3) Consolidation

All concrete was consolidated by vibration to the
maximum practicable density, so that it was free
from pockets of coarse aggregate and entrapped air,
and so that it closed tightly against the subgrade
and all surfaces of forms and embedded materials.

Consolidation of concrete in the tunnel lining was
performed by immersion type vibrators and by form
vibrators. Form vibrators were rigidly attached to
the forms and operated at speeds of at least
8,000 cpm when vibrating concrete. Form vibrators
were operated at successive locations not more than
5 feet apart, closely behind the advancing slope of
concrete in the sidewalls and shoulders of the arch.

The location of the form vibrators at the crown,
position of the end of the discharge line, operation
of vibrators, discharge of concrete, and withdrawal
of the discharge line was coordinated so as to
obtain maximum filling of the crown with concrete
and so as to avoid settlement and flow of concrete
from the filled crown due to improperly positioned
and timed vibration.

(f) Curing and Protection

Curing of concrete was in accordance with ACI 301-72 and
ACI 308-71, except as modified by the approved
construction specifications. A curing compound meeting
the requirements of ASTM C 309-72, Types 1 and 2, was
used. Curing compounds were not applied to surfaces
where bond was required for additional concrete.

Freshly deposited concrete was protected from excessively
hot or cold temperatures in accordance with ACI 305-72
and ACI 306-66.

(g) Deficient Concrete

Deficient concrete was removed and repaired in accordance
with ACI Title No. 59-57, Chapter 7, unless modified by
the approved construction specifications.

4. Grouting

(a) General

Specifications included consolidation grouting to control
water flows and to consolidate ground; and contact

grouting to fill voids between the concrete lining and
the excavated surfaces.

(b) Consolidation Grouting

Consolidation grouting is a technique used to expedite
construction, and is not necessary for the stability of
the completed tunnel. The approved construction
specifications permitted the use of consolidation
grouting, although the conditions encountered in the work
did not require its use.

(c) Contact Grouting

Detailed specifications for contact grouting were
included in the approved construction documents. General
features of contact grouting were as follows:

(1) Materials

Grout consisted of a mixture of cement and water or
cement, sand and water. Materials for grout were as
specified in <Section 3.8.1.6.1.2.a>,
<Section 3.8.1.6.1.2.c> and <Section 3.8.1.6.1.2.d>,
for concrete.

(2) Equipment

Grout holes and vent holes were drilled with either
rotary or percussion drilling equipment. The
equipment for mixing and injecting grout was of
adequate size capable of satisfactorily agitating
the grout and forcing it into the grout holes in a
continuous flow at approved pressures.

(3) Grout and Vent Holes

Holes through the tunnel lining for the purpose of
injecting grout or air release were at the locations
and to the depths shown on <Figure 3.8-71> and
<Figure 3.8-72>. Holes were formed by embedding
pipe in the concrete during placing of the lining or
by drilling through the concrete lining. Pipe and
fittings conformed to ASTM A 120, Schedule 40,
Black.

(4) Grout Placement

Contact grouting was not performed behind concrete
lining less than 14 days old. Grouting was
continued until a clear grout stream emerged from
the vent. Pressure at the hole was as approved, but
did not exceed 50 psi. The grouting of a hole was
considered complete when the hole refused grout at
90 percent of the highest pressure directed for that
hole. After the grouting of a hole was completed,
the pressure was maintained by means of a stop cock
or other suitable device until the grout had set
sufficiently to be retained in the hole.

e. Special Construction Techniques and Requirements - Offshore
Structures

Placement of Steel Lining in Offshore Downshafts and Risers

1. Steel linings were placed in accordance with the approved
construction documents.

2. Two inch diameter threaded rods were screwed through the
    24 grout plug openings after the placement of each liner such
    that, (1) the ends of the rods extended a minimum of
    4-1/2 inches beyond the outside of the liner shell, and (2) a
    nut tack welded to the end of the rod was drawn tight against
    a lead washer and the inside of the steel liner. The material
    for the nuts and rods conformed to ASTM A 36-74 steel. After
    the installation, the area of the nuts and washers was
    completely covered with a protective one-coat application of
    coal tar epoxy meeting Military Specification MIL-P-23236,
    Type 1, Class 2. After installation of the rods, an approved
    plant-mix grout was placed to completely fill the void between
    the steel liner and the rock excavation limits. The grout
    attained a minimum compressive strength of 4,000 psi at
    28 days when tested in accordance with ASTM C 109-73.

3. The transition area at the bottom of the steel linings was
    filled with nonshrinking, pre-mix grout. The nonshrink grout
    was mixed and placed in accordance with the manufacturer's
    recommendations and the contractor's approved construction
    procedures. The nonshrink grout had a minimum ultimate
    compressive strength of 4,000 psi at 28 days when tested in
    accordance with ASTM C 109-73. Dimensional stability was
    tested in accordance with CRD C 588-76.

3.8.4.7   Testing and Inservice Inspection Requirements

There are no planned systematic testing or inservice surveillance
programs for the safety class structures discussed in this section.

At present there are no trash racks at the offshore intake structure
ports. The intake heads were designed in such a way that trash racks
can be installed if required. Need for trash racks will be based on an
inservice surveillance program of monitoring the intake heads and their
ports.

3.8.5   FOUNDATIONS AND CONCRETE SUPPORTS

3.8.5.1   Physical Description

Major safety class structures are listed in <Table 3.8-9>.

The design of the foundations for these structures considers the
stiffness of the mat. Loads from walls and columns are assumed to be
transferred to the mat by direct bearing. Where moment carrying
connections occur, the design incorporates the applied moments at the
junction with vertical structural elements.

The reactor building complex mat does not provide the immediate
foundation for the reactor or reactor coolant pumps. <Section 3.8.3>
gives details of the reactor foundation (reactor vessel pedestal).
Adjacent safety class structures are separated by a sufficient space to
accommodate seismic movements. Groundwater is prevented from entering
the "rattle space" by a continuous waterproofing membrane which extends
under the foundation mats. In addition, the joints between the safety
class foundation mats contain waterstops. Should the waterproofing
membrane and waterstops fail for any reason, the walls of adjacent
safety class structures are designed for water elevation at 590'-0",
corresponding to the design basis groundwater elevation described in
<Section 3.8.5.3.4>.

Where possible, shear transfer from the bearing material to the
reinforced concrete foundation is by friction; otherwise shear transfer
from the bearing material to the reinforced concrete foundation is by a
combination of friction and passive soil pressure against shear keys.
In either case, the waterproofing membrane and porous concrete have
sufficient shear strength and coefficient of friction to develop the
required frictional resistance. The coefficient of friction used in the
design is 0.5. This value is the minimum for the waterproofing
membrane, as required by the project specifications, and is also lower

than the expected coefficient of friction for any of the bearing
material: shale, concrete, porous concrete, waterproofing membrane, and
interfacing surfaces. Expected coefficients are in the range of 0.6 to
0.8 for waterproofing membrane on concrete. The other media are
expected to have coefficients of friction of above 0.6 as recommended by
the soils consultant, Woodward-Clyde Consultants. The general
arrangement of shear keys required to help transfer shear from the
bearing material to the concrete foundations is shown in
<Figure 3.8-90>, <Figure 3.8-91>, <Figure 3.8-92>, <Figure 3.8-93>,
<Figure 3.8-94>, and <Figure 3.8-95>. Details of the waterproofing
membrane are illustrated in <Figure 3.8-79>.

Caissons are used as vertical support for a portion of the fuel handling
building <Figure 3.8-80>. The piers extend from the bottom of the mat
through the underlying backfill material and the porous concrete blanket
and penetrate into the shale. The vertical bearing pressure under the
caissons is less than the allowable bearing pressure of the shale, as
established by <Section 2.5>. The caissons are not considered as
providing resistance to the lateral movement of the building.
Resistance to lateral forces on the building is provided by the friction
of the portion of the foundation mat that rests on shale. For the
design of caissons, deflections are obtained at the top of the caissons
from the seismic analysis <Section 3.7.1>. These deflections are used
to calculate moments and shears by means of a beam on elastic foundation
model. In this model, conservative values of shears and moments are
calculated using parametric studies in which both the stiffness of the
soil and the end conditions are varied within possible bounds. Soil
stiffness values are obtained from the soil consultant, Woodward-Clyde
Consultants.

The configuration, layout, identification of bearing strata, immediate
bearing material, etc., for the foundations of the safety class
structures listed above, are shown in <Figure 3.8-81>.

3.8.5.2   Applicable Codes, Standards and Specifications

3.8.5.2.1   Codes

a.   ACI 301-72, "Specifications for Structural Concrete for Buildings,"
     American Concrete Institute.

     All chapters of ACI 301 are applicable except Chapter 15 which has
     requirements for prestressed concrete.

b.   ACI 318-71, "Building Code Requirements for Reinforced Concrete,"
     American Concrete Institute.

     Chapters Used

     1   General Requirements
     2   Definitions
     7   Details of Reinforcement
     8   Analysis and Design
     9   Strength and Serviceability
     10   Flexure and Axial Load
     11   Shear and Torsion
     12   Development of Reinforcement
     13   Footings
     Appendix B - Notation

     Chapters Not Used

     $3^{(1)}$ Materials
     $4^{(1)}$ Concrete Quality
     $5^{(1)}$ Mixing and Placing Concrete
     $6^{(1)}$ Formwork, Embedded Pipes, and Construction Joints
     $13^{(2)}$ Slab Systems with Panels
     $14^{(2)}$ Walls

Chapters Not Used (Continued)

16[2]   Precast Concrete

17[2]   Composite Concrete Flexural Members

18[2]   Prestressed Concrete

19[2]   Shells and Folded Plate Structures

20[2]   Strength Evaluation of Existing Structures

             Appendix A

NOTES:

[1] ACI 301-72 will be used to define requirements in this area as
    modified by <Section 3.8.5.6>.

[2] These sections are not applicable to foundation structures.

c.   Ohio Building Code, 1970 Edition, Chapter BB-33 "Reinforced and
     Plain Concrete, Reinforced Gypsum Concrete and Reinforced
     Masonry," Board of Building Standards Department of Industrial
     Relations, State of Ohio.

Sections Used

BB-33-01 Definitions

BB-33-02 Abbreviations

BB-33-03 Accepted engineering practice and approved standards

BB-33-07 Reinforced concrete general design and construction

Sections Not Used

BB-33-04[1]   Reinforced concrete materials

BB-33-05[1]   Concrete, classification and quality

BB-33-06[1]   Concrete, ready mixed

BB-33-08[2]   Reinforced concrete, supervision

BB-33-09 to

Sections Not Used (Continued)

BB-33-13[3]   Plain concrete

BB-33-14[4]   Reinforced gypsum concrete, precast

BB-33-15[4]   Reinforced gypsum concrete, design and construction

BB-33-16 to
BB-33-20[5]   Reinforced masonry

NOTES:

[1] ACI 301-72 will be used in this area as modified by
    <Section 3.8.4.6>.

[2] Not applicable to these designs.

[3] Plain concrete is not used on these structures except as a
    foundation.

[4] Reinforced gypsum backfill material is not used on these
    structures.

[5] Reinforced masonry is not used on these structures.

d.   ASME-ACI, "Proposed Standard Code for Concrete Reactor Vessels and
     Containment" - Proposed Section III, Division 2, of ASME Boiler
     and Pressure Vessel Code - Issued May 1974, as Committee Draft.

3.8.5.2.2   Standards

a.   ASTM Standards: The more significant ASTM standards are listed in
     <Table 3.8-8>.

b.   Applicable Regulatory Guides

     Regulatory guides pertaining to seismic design classification and
     seismic design are referenced in <Section 3.2> and <Section 3.7>,
     respectively.

1.   <Regulatory Guide 1.10>, "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures." This standard was used with modifications specified in <Section 3.8.1.6.3>.

2.   <Regulatory Guide 1.15>, "Testing of Reinforcing Bars for Category I Concrete Structures." This standard was used with the modifications specified in <Section 3.8.1.6.2>.

3.   <Regulatory Guide 1.55>, "Concrete Placement in Category I Structures."

4.   <Regulatory Guide 1.142>, (April 1978), "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)" <Section 3.8.1.2.2.b.4>.

3.8.5.2.3   Principal Plant Specifications

The principal specifications prepared by the engineer for the foundation structures are:

a.   Concrete supply.

b.   Placement of structural concrete.

c.   Fabrication and placing or reinforcing steel.

d.   Plant excavation and backfill.

e.   Fabrication of embedded steel.

f.   Supply and installation of waterproofing and waterstops.

The detailed material specifications, the detailed quality control provisions and any special specifications, the detailed quality control provisions and any special construction technique requirements, all as described in <Section 3.8.5.6>, are included in these specifications. The specification for excavation and backfill includes the requirements and quality control provisions described in <Section 2.5.4>.

3.8.5.3   Loads and Loading Combinations

3.8.5.3.1   General

The loads and loading combinations which are applied to foundations are identified in <Section 3.8.1.3>, <Section 3.8.2.3>, <Section 3.8.3.3>, and <Section 3.8.4.3>. The design approach and load factors for the concrete foundations are also identified in these same sections for the load combinations given.

3.8.5.3.2   Soil Bearing

The soil bearing values that are developed under foundations from the loading combinations specified in <Section 3.8.5.3.1> are within the recommended limits given in <Section 2.5.4>.

3.8.5.3.3   Settlement

As indicated in <Section 2.5.4>, settlement or heave of the foundations is not expected to exceed 1/2", and angular distortion due to settlement is not expected to exceed 1 in 1,500. All safety class structures are structurally separate so that differential settlement of one structure does not impose loads on another structure. Since the predicted long term settlements of safety class structures are very small, they are not considered in the analysis of these structures.

3.8.5.3.4   Groundwater Elevation

As described in <Section 2.4.13.5>, a Safety Class 3, Seismic Category I underdrain system functions to assure that the groundwater never exceeds Elevation 590'-0" for all safety class structures except the emergency service water pumphouse. During normal plant operation, the groundwater elevation is maintained below 568'-6" by pumps. The design basis for all safety class structures except the emergency service water pumphouse is, therefore, groundwater at Elevation 590'-0"; the water pressure corresponding to this elevation is considered for all loading combinations.

The emergency service water pumphouse includes a porous concrete blanket beneath the foundation base mat, extending vertically up along the exterior building walls. The bottom of the base mat is at Elevation 532'-0". The water elevation inside the pumphouse ranges from a minimum elevation of 562.09' to an approximate maximum of 583.61'. Porous concrete backfill is placed adjacent to the exterior walls to approximately Elevation 558'-0" (top of shale). Class A fill is provided along the walls from Elevation 558'-0" to grade (Elevation 616'-6"). This structure is designed for groundwater at Elevation 618'-0" for all loading combinations.

In addition, a local load corresponding to groundwater elevation at 618'-0" is applied for the static design of exterior walls and mats for all safety class structures except for the emergency service water pumphouse which has been designed for water to Elevation 618'-0" for all loading combinations. This load is included to provide assurance of structural adequacy for a postulated massive liquid spill and is included only under the normal operation load combinations.

3.8.5.3.5   Extent of Compliance to ACI 349-76, "Code Requirements for Nuclear Safety-Related Concrete Structures," American Concrete Institute.

The extent of compliance to ACI 349-76 is discussed in
<Section 3.8.3.3.7>.

3.8.5.4   Design and Analysis Procedures

3.8.5.4.1   Analytical Techniques

As indicated in <Section 3.8.5.1>, most of the safety class structures
have constant thickness mats. Loads are transmitted to the mats from
walls, columns, water, and internal equipment in accordance with the
principles of statics and stress analysis. As an example, the shield
building is treated as a thin-walled cantilever beam. Accelerations
obtained from the dynamic analysis are converted into static axial load,
shear and bending moment. Shield building stresses obtained from these
equivalent static loads are then applied to the reactor building mat as
loads. The loads are transmitted through the mat, thereby developing
internal mat bending, axial load and shear, to the bearing material
which is represented by an elastic half-space, Winkler springs, or as
having a conservative linear reaction capability. Reinforcing steel is
sized on the basis of the internal axial loads, shears and moments
developed in the mat. Stress criteria are based on the strength design
concept of ACI 318-71. Since no single loading combination governed
throughout the mat section, results of the various load combinations
were enveloped to establish maximum shear and moment requirements for
the mat. A plot showing the various design moment envelopes versus
reinforced section moment capacity is provided in <Figure 3.8-102>.
Plots are provided for both top and bottom reinforcing in the radial and
circumferential directions. <Figure 3.8-103> shows a similar plot for
shear capacity versus an envelope of design shear forces. A typical
plan and section of the reinforcement for the reactor building
foundation mat are shown in <Figure 3.8-82> and <Figure 3.8-83>.

a.   Reactor Building Mat

    1.   Analysis

        The static analyses are performed using the finite-element
        computer program ELAD. ELAD is based on a linearly-elastic
        continuum finite element solution for axisymmetric solids
        subjected to axisymmetric or non-symmetric static,
        quasi-static and temperature induced loads. The analytical
        model for the reactor building complex foundation mat is shown
        in <Figure 3.8-84>.

    2.   Results of Analysis

        Analyses for the static and dynamic loads are performed
        separately. The results are summed algebraically for the
        specified loading combinations to obtain the critical case for
        each part. Parametric studies have been carried out in the
        preliminary design stage using a circular plate on an elastic
        half space as the model to provide sensitivity studies for
        variations in the assumed elastic modulus. The range assumed
        in the studies is from one-half (1/2) to one and
        one-half (1-1/2) times the recommended value of elastic
        modulus. Maximum calculated internal shear force in the mat
        varies by less than 10 percent and maximum internal moments by
        plus 30 percent and minus 10 percent from the base number
        calculated using the recommended modulus. Studies have also
        been performed for variation in the assumed shear modulus in
        the seismic analysis. These studies indicate little change
        (10 percent) in calculated bearing pressures for variations in
        assumed modulus from the minimum measured field value to three
        times this value. The studies into variations in bearing
        material properties do not consider localized spatial
        variation within the material in contact with the foundation

mats. This has been done because of the relatively uniform
properties of the shale material, as indicated by adjacent
bore hole test results. Any variations which may occur have
an insignificant effect on the internal design moments and
shears for the foundation mats.

The maximum static bearing pressure calculated is 15 ksf for
the case of a main steam line break with no seismic load
considered. The maximum dynamic bearing pressure calculated
for the Safe Shutdown Earthquake is 30 ksf.

b.   Other Safety Class Structures

Structures having foundation mats listed in <Section 3.8.5.1> are
analyzed as follows:

1.   Static Analysis

For static conditions the method of analysis used is:

(a)   As presented by Dunham (Reference 19) for mats.

(b)   SLAM, which utilizes the finite difference technique, for
those areas where large variations in applied load occur.

(c)   STARDYNE for complex areas requiring a more rigorous
analysis.

2.   Dynamic Analysis

Lumped mass models of structures including foundations are
made, using either simplified vertical cantilevered beam or
finite element models. For the simplified cantilevered
models, equivalent static forces are obtained at the assumed

mass locations which are analyzed as described in <Section 3.8.5.4.1.b.1>. For finite element models, forces are obtained at element boundaries directly from computer program output. Seismic analysis methods are described in <Section 3.7>.

c.   Porous Concrete

As explained in <Section 3.8.5.5>, the calculated stresses in porous concrete due to loads from structures are well within acceptable values for concrete as permitted by ACI 318-71. It is expected that cracking occurs in a plain concrete system as extensive as the underdrain system from any of the following causes:

1.   Shrinkage - Due to low water/cement ratio and low cement factor, shrinkage of the porous concrete is expected to be minimal. However, even minimal shrinkage could produce some random cracking considering the extent of the "blanket" of porous concrete.

2.   Temperature - During construction of the porous concrete system, thermal gradients through the material could produce random vertical cracking throughout the entire system.

3.   Differential Loads - Along building lines at sectional changes there are differential vertical loads which could produce vertical cracking.

The effect of this random cracking is negligible. Except for the perimeter of the system, the porous concrete is confined between safety class reinforced concrete structures or Class A fill and component rock. The transfer of normal and shearing stresses by

the porous concrete between the reinforced concrete and rock is not affected by vertical cracking.

d.   Stability Calculations

<Table 3.8-10> gives the calculated factors of safety for flotation and overturning for Safety Class structures with water at Elevation 590'-0". <Section 3.8.5.3.4> presents a discussion of water table levels and loading conditions. In <Table 3.8-10>, the factor of safety against flotation is determined for normal operation only (i.e., no accident or seismic event). The factor of safety against overturning is the ratio of the restoring moment to the overturning moment, with both moments taken about the toe of the foundation mat. In addition to the stability calculations, the design of foundations includes the calculation of vertical bearing pressures under the foundations. These pressures are in accordance with the strength of the bearing materials as established in <Section 2.5.4>, which are the soil bearing design values specified by the soil consultant, Woodward-Clyde Consultants. Factors of safety for sliding are not shown in <Table 3.8-10> as these are considered in the design of foundations. Factors of safety for sliding are at least 1.5 for load combinations which include design wind or OBE. For load combinations which include tornado and SSE the factor of safety against sliding is 1.1. <Section 3.8.5.1> gives more details on resistance to lateral forces.

3.8.5.4.2   Design Procedures

Design methods for the reinforced concrete foundations are described in <Section 3.8.4.4.1>.

3.8.5.4.3   Computer Programs

a.   Computer Programs Used in the Structural Analyses of Foundations

1.   ELAD

      <Section 3.8.1.4.6.a.2> presents a program description and a
      list of computer hardware/software.

2.   MRI/STARDYNE

      <Section 3.8.3.4.5.a.1> presents a program description and a
      list of computer hardware/software.

3.   SLAM

      SLAM analyzes slabs and mats by the finite difference method.
      The user selects the correct equation generating operator from
      a library routine of verified operators. The program writes a
      linear equation for each node relating the known loads and
      boundary conditions to the governing biharmonic partial
      differential equation in terms of the unknown nodal
      deflections. Simultaneous solution of these linear equations
      yields a set of nodal deflections which are used to compute
      shears and moments. Walls, beams, openings, and various
      boundary conditions can be treated by selecting the proper
      operator from the library. Mat foundations are treated as a
      Winkler model by adding a spring term to the nodal equations.
      This is a program developed by Gilbert Associates, Inc.,
      Reading, PA, for the IBM 370/155 computer under IBM operating
      system O/S 21.7 MFT with HASP 3.1.

b.   Design Control and Verification of Computer Programs

      The methods for the design, control and verification of the
      computer programs used in the foundation analyses are discussed in
      <Section 3.8.1.4.6.b>.

c.   Verification of Program Usage

    1.   ELAD

       Program verification is given in <Section 3.8.1.4.6.c.2>.

    2.   MRI/STARDYNE

       Program verification is given in <Section 3.8.3.4.5.c.1>.

    3.   SLAM

       SLAM has been verified by comparison of its answers with classical solutions and step by step manual duplication of its calculations. The initial test problem was compared to a NASTRAN <Section 3.8.3.4.5.a.2> solution of the same problem. Documentation of SLAM is available at Gilbert Associates, Inc., Reading, PA.

3.8.5.5   <u>Structural Acceptance Criteria</u>

The structural acceptance criteria for the reinforced concrete foundations is the same as for the reinforced concrete structures which they support. For the reactor building complex mat see <Section 3.8.1.5> and for all other mats see <Section 3.8.4.5>.

The stresses and strains expected for porous concrete underlying the foundation mats are less than those permitted by applicable sections of ACI 318-71. For example, expected stresses at the toe of the reactor building mat for the SSE combined with normal loads and with groundwater at Elevation 590'-0" are:

a.   Normal vertical compressive stress ~210 psi.

b.   Horizontal shearing stress ~80 psi.

c.   Corresponding principal compression stress ~240 psi.

d.   Principal tension stress ~30 psi.

In the calculation of principal stress, no credit is taken for lateral
confinement.

The maximum normal vertical compressive stress of 210 psi is well within
the ultimate bearing value of $0.85 \, \phi f'_c$ (595 psi for f'_c = 1,000 psi and
ϕ = 0.70), as specified by ACI 318-71 Section 10.14.1.

The maximum horizontal shearing stress in the porous concrete (80 psi)
is considered acceptable since a coefficient of friction of 0.5 results
in a shearing stress capacity of 105 psi for a normal stress of 210 psi.

Finally, concern for diagonal cracking is eliminated by comparing the
30 psi principal tension to a conservative capacity of $3 \sqrt[\phi]{f'_c}$ (81 psi
for ϕ = 0.85 and f'_c = 1,000 psi).

Assurance that inplace strength is achieved as specified (i.e., f'_c =
1,000 psi) is discussed in <Section 3.8.5.6.b>.

3.8.5.6   Material Specifications, Quality Control and Special
             Construction Techniques

The material specifications, quality control provisions and special
construction technique for concrete construction, reinforcing steel and
cadweld splices are discussed in <Section 3.8.1.6>. Other items are as
discussed below:

a.   Excavation and Placement of Structural Fills

     <Section 2.5.4> presents a description of the material
     specifications quality control provisions acceptance criteria, and
     any special construction techniques for the bearing material and
     structural load bearing fill.

b.   Porous Concrete

     Porous concrete is used in the underdrain system which underlies
     safety class structures in the main plant area, as described in
     <Section 2.4.3.5>. Structurally, the porous concrete serves the
     same function as a concrete "mud mat" commonly used as a medium
     between supporting rock and reinforced concrete foundations. The
     porous concrete layer is classified Safety Class 3, Seismic
     Category I. The material specifications, quality control
     provisions and special requirements such as strength and
     permeability are discussed in <Section 3.8.1.6.1>.

     Specific placing requirements detailed in the project
     specifications include:

     1.   The free fall distance for the placement of porous concrete
          shall not exceed 3 feet to preclude segregation.

     2.   Experience, based on a literature search, indicates that
          bleeding of porous concrete is of little concern. Since some
          of the porous concrete is placed using vertical forms, form
          pressures are of some concern. Therefore, it is considered
          that a limit on rate of pour was a superior criterion to a
          pour lift height criterion. This rate of pour is set at five
          feet per hour, as required in project specifications for
          pouring walls using normal weight concrete and is used in lieu
          of the lift height criteria previously considered.

3. In any pour, porous concrete shall be raked level to the top
    of the forms.

4. Concrete vibrators shall not be used to consolidate porous
    concrete.

5. The porous concrete shall be worked by rodding only to the
    extent necessary to ensure complete filling of the forms.

6. Porous concrete shall be cured by covering with burlap or
    other fibrous material and maintaining the material in a moist
    condition for at least 14 days.

7. The uniform size of the aggregate and the low allowable free
    fall distance precludes segregation of the concrete.

8. All formed surfaces at which accumulations of mortar could
    result (which could form a hydraulic discontinuity) are
    inspected. This inspection provides assurance that the forms
    have been removed. Where such surface accumulations are
    noted, it is required that they be removed by mechanical means
    without damaging sound material.

9. The temperature of porous concrete at the time of placement
    shall be not less than 50°F nor greater than 80°F.

3.8.5.7   Testing and Inservice Inspection Requirements

There are no planned systematic testing or inservice surveillance
programs for the foundations discussed in this section.

3.8.6   REFERENCES FOR SECTION 3.8

1.   "Structural Design Criteria for Category I Structures Outside Containment," Structural Engineering Branch, Directorate of Licensing, Attachment A, dated September 24, 1973, AEC request for additional information to Dockets Nos. 50-440/441.

2.   R. A. Williamson and R. R. Alvy, "Impact Effect of Fragments Striking Structural Elements," Holmes & Narver, Inc., Anaheim, California, 1973.

3.   John M. Biggs, "Introduction to Structural Dynamics," McGraw-Hill, 1964.

4.   R. D. Brunell, "HEATING 2," Atomic International, Computer Technology Group, AI-64 Memo-177.

5.   "Cadweld Rebar Splicing," Erico Products Catalogue No. RB20M-173, Cleveland, Ohio.

6.   "Inspection of the Cadweld Rebar Splice," Erico Products publication RB-5M768.

7.   IEEE 317, "Standards for Electrical Penetration Assemblies in Containment Structures for Nuclear Generating Station," The Institute of Electrical and Electronic Engineers, Inc., 1971.

8.   E. H. Baker, L. Kovalevsky and F. L. Rish, "Structural Analysis of Shells," McGraw-Hill, 1972.

9.   Vol'mir, A. S., "Stability of Elastic Systems."

10. A. Gahli, "Finite Element Analysis of Perforated Shells," Proceedings of IASS Symposium "Shell Structures and Climatic Influences," University of Calgary, July 3-6, 1972.

11. M. A. Boling and W. A. Rhoads, "ANISIN/DRFII," Atomics International, Computer Technology Group, AI-66 Memo-171.

12. Proctor and White, "Rock Tunneling with Steel Supports" published by Commercial Shearing and Stamping Co., 1946.

13. Wahanik, R. J., "Submerged Offshore Structures for Perry Nuclear Power Plant," GAI Report No. 1891, March 4, 1976.

14. Alden Research Laboratories, "Hydraulic Model Studies of Offshore Intake Structure for Perry Nuclear Power Plant," December 1974.

15. W. T. Moody, "Moments and Reactions for Rectangular Plates," Engineering Monograph No. 27, a Water Resources Technical Publication, Bureau of Reclamation, U.S. Department of the Interior, U.S. Government Printing Office, Washington, 1966.

16. Blume, J. A., Sharpe, R. L., Garrison, K., "Earthquake Engineering for Nuclear Reactor Facilities," John A. Blume and Associates, Engineers, 1971.

17. "Proceedings, First Rapid Excavation and Tunneling Conference," ASCE and AIME, Chicago, Illinois, 1972, K. S. Lane, L. A. Garfield, Editors.

18. STRAAM Engineers, Inc., "Design of Concrete Final Lining for Cooling and Emergency Service Water Tunnels, Perry Nuclear Power Plant, Units 1 and 2," November 1979.

19. C. W. Dunham, <u>Foundations of Structures</u>, McGraw Hill, New York, 1950.

20. GAI Report, "Containment Annulus Concrete Design, Construction and Testing for the Perry Nuclear Power Plant," Rev. 2, Transmitted to USNRC, March 1983.

21. PY-CEI/NRR-0131L dated February 11, 1985, from M. R. Edelman to B. J. Youngblood, "SER Confirmatory Issue (3), Containment Ultimate Capacity Analysis."

22. The Cleveland Electric Illuminating Co., "Preliminary Evaluation of the Perry Nuclear Power Plant Hydrogen Control System," Docket Nos. 50-440/50-441, Revised March 21, 1985.

23. PY-CEI/NRR-0138L dated September 19, 1984, from M. R. Edelman to B. J. Youngblood, "Response to NRC Question Nos. 220.32 through 220.36 (480-54) - Containment Drywell Wall Structural and Bypass Leakage Integrity."

24. Letter of January 14, 1985, from B. J. Youngblood to M. R. Edelman, "Staff Evaluation Findings Pertaining to the Construction Appraisal Team Concern Re Containment Drywell Wall Structural and Leak Tightness Integrity in the Perry Nuclear Power Plant (Units 1 and 2)."

25. Memorandum "E"-SO-13330, November 27, 1991 from S. Tomaszewski/ C. Angstadt to K. Donovan.

26. Calculation G58-S-SC-002, "Evaluation of the Fuel Handling Building Floor and Spent Fuel Pit for Cask Loads".

27. Calculation G58-S-SC-003, "Design of the Fuel Handling Building Exterior Mat Extension".

TABLE 3.8-1

CONCRETE SHIELD BUILDING LOAD COMBINATIONS

Load Combination	Stress Criteria[12]	D	L[1]	G	G′	H	W	W_t[2]	F_{eqo}	F_{eqs}	Z	T_o	P_{srv}	T_{srv}[3]	P_a	T_a	M	R_o	R_a	Y_r	Y_j	Y_m	P_t	P_r	Remarks
Construction		1.4	1.7	1.4		1.7	1.7					1.3													See Note[4]
Normal Operation		1.4	1.7	1.4		1.7						1.3	1.7	1.3				1.3							See Note[5]
Normal Operation with Severe Environmental		1.4	1.7	1.4		1.7						1.3	1.7	1.3				1.3							See Note[5]
		1.2		1.4		1.3	1.7							1.4											See Note[5]
		1.4	1.7	1.4		1.7		1.9				1.3	1.7	1.3				1.3							See Note[5]
		1.2		1.4		1.3		1.9						1.4											See Note[5]
Normal Operation with Extreme Environmental		1.0	1.0	1.0		1.0	1.0				1.0	1.0	1.0	1.0				1.0							See Note[5]
		1.0	1.0	1.0		1.0				1.0		1.0	1.0	1.0				1.0							See Note[5]
Normal Operation with Abnormal Loads		1.0	1.0	1.0		1.0							1.25	1.0	1.5	1.0			1.0						See Note[6][9]
		1.0	1.0	1.0		1.0					1.0								1.0					1.0	Explosions
Normal Operation with Severe Environmental and Abnormal Loads		1.0	1.0	1.0		1.0			1.25				1.0	1.0	1.25	1.0			1.0	1.0	1.0	1.0			See Note[6][7] See Note[8][9]

Load Combination	Stress Criteria $^{(12)}$									LOAD SYMBOLS $^{(13)}$															
		D	$L^{(1)}$	G	G'	H	W	$W_t^{(2)}$	F_{eqo}	F_{eqs}	Z	T_o	P_{srv}	$T_{srv}^{(3)}$	P_a	T_a	M	R_o	R_a	Y_r	Y_j	Y_m	P_t	P_r	Remarks
Normal Operation with		1.0	1.0	1.0		1.0			1.0				1.0	1.0	1.0			1.0		1.0	1.0	1.0	1.0			See Note $^{(6)(7)}$ See Note $^{(8)(9)}$
Extreme Environmental and		1.0	1.0	1.0		1.0	1.0			1.0	1.0	1.0	1.0	1.0			1.0								See Note $^{(5)(10)}$ For overall structure
Abnormal Loads		1.0	1.0				1.0				1.0						1.0								See Note $^{(10)(11)}$ For local areas

NOTES:

$^{(1)}$ Both cases of L having full value or being completely absent are considered.

$^{(2)}$ Applies to shield building and reactor building foundation mat only.

$^{(3)}$ The maximum or minimum suppression pool temperature producing the worst thermally induced stresses for the case considered.

$^{(4)}$ Wind load appropriate to the stage of construction. After completion of construction the shield building is the only structural element exposed to wind or tornado.

$^{(5)}$ Operation of safety relief valves as described in <Appendix 3A> and <Appendix 3B>.

$^{(6)}$ Operation of a single relief valve first pop combined with LOCA pool swell pressure loadings.

$^{(7)}$ The maximum values of Y_r, Y_j and Y_m will be used simultaneously unless time histories are used to justify otherwise.

$^{(8)}$ Local stresses due to the concentrated loads Y_r, Y_j and/or Y_m may exceed the design stress criteria provided there is no loss of function of the structural element considered.

$^{(9)}$ For a small line or intermediate size break the design includes the operation of a single or multiple safety relief valves as described in <Appendix 3A> and <Appendix 3B> with the appropriate design pressure (P_a) and temperature (T_a).

$^{(10)}$ The load combination shown is for tornado generated missiles on the external surfaces of the shield building.

$^{(11)}$ Strength design by ACI 318 is not used for local areas. Analysis and design is as described in <Section 3.5.3>.

$^{(12)}$ For stress criteria, see <Section 3.8.1.5>.

$^{(13)}$ For definition of load symbols, see <Section 3.8.1.3>.

TABLE 3.8-2

CONTAINMENT VESSEL LOAD COMBINATIONS

LOAD SYMBOLS[10]

Load Combination	Stress Criteria[9]	D	L[1]	G_{sp}	G'	W[2]	F_{eqo}	F_{eqs}	Z	T_o	P_{srv}	T_{srv}	P_a[3]	T_a	R_o	R_a	Y_r	Y_j	P_t	Remarks
Construction		1.0	1.0	1.0		1.0					1.0									
Testing		1.0	1.0								1.0					1.0			1.0	See Note[4]
		1.0	1.0	1.0							1.0	1.0	1.0			1.0				See Note[5]
Normal Operation		1.0	1.0	1.0					1.0	1.0	1.0		1.0			1.0				See Note[6]
Normal Operation Severe Environmental		1.0	1.0	1.0			1.0		1.0	1.0	1.0		1.0			1.0				See Note[6]
Normal Operation with Extreme Environmental		1.0	1.0	1.0				1.0	1.0	1.0	1.0		1.0			1.0				See Note[6]
		1.0	1.0	1.0					1.0	1.0	1.0		1.0			1.0				See Note[11]
Normal Operation with Abnormal Loads		1.0	1.0	1.0									1.0	1.0	1.0	1.0	1.0	1.0	1.0	See Note[7][8]
Normal Operation with Severe Environmental and Abnormal Loads		1.0	1.0	1.0			1.0						1.0	1.0	1.0	1.0	1.0	1.0	1.0	See Note[7][8]
Normal Operation with Extreme Environmental and Abnormal Loads		1.0	1.0	1.0				1.0					1.0	1.0	1.0	1.0	1.0	1.0	1.0	
		1.0	1.0	1.0			1.0						1.0	1.0	1.0	1.0	1.0	1.0	1.0	See Note[11]
Postaccident Core Recovery		1.0	1.0				1.0				1.0					1.0				
		1.0	1.0		1.0		1.0				1.0					1.0				

TABLE 3.8-2 (Continued)

NOTES:

(1) Both cases of L having full value or being completely absent are considered.
(2) Considered during construction only. Shielding Building carries all wind loads during normal operation.
(3) The maximum or minimum suppression pool temperature producing the worst thermally induced stresses for the case considered.
(4) Structural integrity test of the containment vessel.
(5) Testing safety relief valves.
(6) Operation of safety relief valves as described in <Appendix 3A> or <Appendix 3B>.
(7) Operation of a single relief valve first pop combined with LOCA pool swell pressure loadings.
(8) For a small line or intermediate size break, the design includes the operation of a single or multiple safety relief valve as described in <Appendix 3A> and <Appendix 3B> with the appropriate design pressure (P_a) and temperature (T_a).
(9) For stress criteria, see <Section 3.8.2.5>.
(10) For definition of load symbols, see <Section 3.8.2.3>.
(11) Symbol Z in these two load combinations is the tornado depressurization load. For other load combinations containing symbol Z, the pressure loads are described in <Section 3.8.2.3f>.

TABLE 3.8-3

REACTOR BUILDING INTERNAL CONCRETE STRUCTURE LOAD COMBINATIONS

Load Combination	Stress [11] Criteria	D	L [1]	G	G'	F_{eqo}	F_{eqs}	Z	T_o	P_{srv}	T_{srv} [2]	P_a	T_a	M	R_o	R_a	Y_r	Y_j	Y_m	P_t	Remarks
Construction		1.4	1.7	1.4					1.3												
Testing		1.4	1.7	1.4					1.3	1.7	1.3										See Note [3]
		1.4	1.7	1.4					1.3											1.4	See Note [4]
Normal Operation		1.4	1.7	1.4				1.4	1.3	1.7	1.3				1.3						See Note [5]
Normal Operation with Severe Environmental		1.4	1.7	1.4	1.9			1.4	1.3	1.7	1.3				1.3						See Note [5]
		1.2		1.4	1.9			1.2	1.4												See Note [5]
Normal Operation with Extreme Environmental		1.0	1.0	1.0			1.0	1.0	1.0	1.0	1.0				1.0						See Note [5]
Normal Operation with Abnormal Loads		1.0	1.0	1.0						1.25	1.0	1.5	1.0			1.0					See Note [6][9]
		1.0	1.0	1.0								1.0		1.0		1.0					Interior Missile See Note [10]
Normal Operation with Severe Environmental and Abnormal Loads		1.0	1.0	1.0		1.25				1.0	1.0	1.25	1.0			1.0	1.0	1.0	1.0		See Note [6][7] See Note [8]
Normal Operation With Extreme Environmental and Abnormal Loads		1.0	1.0	1.0			1.0			1.0	1.0	1.0	1.0			1.0	1.0	1.0	1.0		See Note [6][7] See Note [8]
Postaccident		1.4	1.7		1.4	1.4			1.3							1.4					
Core Recovery		1.2	1.4		1.2	1.4			1.0							1.2					

TABLE 3.8-3 (Continued)

NOTES:

(1) Both cases of L having full value or being completely absent are considered.

(2) The maximum or minimum suppression pool temperature producing the worst thermally induced stresses for the case considered.

(3) Testing of safety relief valves.

(4) Structural integrity test of the drywell with water in the upper pool and suppression pool, and with vent plugs in suppression pool.

(5) Operation of safety relief valves as described in <Appendix 3A> and <Appendix 3B>.

(6) Operation of a single relief valve first pop combined with LOCA pool swell pressure loadings.

(7) The maximum values of Y_r, Y_j and Y_m will be used simultaneously unless time histories are used to justify otherwise.

(8) Local stresses due to the concentrated loads Y_r, Y_j and/or Y_m may exceed the design stress criteria provided there is no loss of function of the structural element considered.

(9) For a small line or intermediate size break, the design includes the operation of a single or multiple safety relief valves as described in <Appendix 3A> and <Appendix 3B> with the appropriate design pressure (P_a) and temperature (T_a).

(10) Strength design by ACI 318 is not used for local areas. Analysis and design is described in <Section 3.5.3>.

(11) For stress criteria, see <Section 3.8.3.5>.

(12) For definition of load symbols, see <Section 3.8.3.3>.

TABLE 3.8-4

LOAD COMBINATIONS FOR STRUCTURAL STEEL PLATE AND FRAME STRUCTURES
INSIDE THE REACTOR BUILDING

Load Combination	Stress Criteria	Load Symbols [14]																					Remarks
		D	L [1]	G	G^1	F_{eqo}	F_{eqs}	Z	T_o	P_{srv}	T_{srv} [2]	P_{rhr}	T_{rhr} [2]	P_a	T_a	M	R_o	R_a	Y_r	Y_j	Y_m	P_t	
Construction	1.0S	1.0	1.0	1.0					1.0														
Testing	1.0S	1.0	1.0						1.0								1.0					1.0	See Note [3]
Normal Operation	1.5S	1.0	1.0	1.0				1.0	1.0	1.0	1.0						1.0						See Note [5]
	1.0S	1.0	1.0	1.0				1.0		1.0													See Note [5]
	1.5S	1.0	1.0	1.0				1.0	1.0			1.0	1.0				1.0						See Note [13]
	1.0S	1.0	1.0	1.0				1.0				1.0											See Note [13]
Normal Operation with Severe Environmental	1.5S	1.0	1.0	1.0		1.0		1.0	1.0	1.0	1.0						1.0						See Note [5]
	1.0S	1.0	1.0	1.0		1.0		1.0	1.0	1.0													See Note [5]
Normal Operation with Abnormal Loads	1.6S	1.0	1.0	1.0				1.0		1.0	1.0	1.0	1.0					1.0					See Note [6] [9] See Note [12]
	1.6S	1.0	1.0	1.0				1.0				1.0		1.0				1.0					See Note [10] Interior Missiles (for local areas)
Postaccident Recovery	1.5S	1.0	1.0		1.0			1.0									1.0						
	1.6S	1.0	1.0		1.0	1.0		1.0									1.0						
Normal Operation with Extreme Environmental	1.6S	1.0	1.0	1.0			1.0	1.0	1.0	1.0	1.0						1.0						See Note [5]
	1.6S	1.0	1.0	1.0			1.0	1.0	1.0			1.0	1.0				1.0						See Note [13]

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TABLE 3.8-4 (Continued)

Load Combination	Stress Criteria	D	L[1]	G	G[1]	Feqo	Feqs	Z	To	Psrv	Tsrv[2]	Prhr	Trhr[2]	Pa	Ta	M	Ro	Ra	Yr	Yj	Ym	Pt	Remarks
Normal Operation with Severe Environmental and Abnormal Loads	1.6S	1.0	1.0	1.0		1.0		1.0		1.0	1.0	1.0	1.0					1.0	1.0	1.0	1.0		See Note [6][7] See Note [8][9] See Note [11] See Note [12]
Normal Operation with Extreme Environmental and Abnormal Loads	1.7S	1.0	1.0	1.0			1.0	1.0		1.0	1.0	1.0	1.0					1.0	1.0	1.0	1.0		See Note [6][7] See Note [8][9] See Note [11] See Note [12]

Load Symbols [14]

NOTES:

[1] Both cases of L having full value or being completely absent are considered.

[2] The maximum or minimum suppression pool temperature producing the worst thermally induced stresses for the case considered.

[3] Structural integrity test of the drywell with no water in the upper fuel pool or suppression pool.

[4] (Deleted)

[5] Operation of safety relief valves as described in <Appendix 3A> and <Appendix 3B>.

[6] Operation of a single relief valve first pop combined with LOCA pool swell pressure loadings.

[7] The maximum values of Y_r, Y_j and Y_m will be used simultaneously unless time histories are used to justify otherwise.

[8] Local stresses due to the concentrated loads Y_r, Y_j and/or Y_m may exceed the design stress criteria provided there is no loss of function of the structural element considered.

[9] For a small line or intermediate size break, the design includes the operation of a single or multiple safety relief valves as described in <Appendix 3A> and <Appendix 3B> with the appropriate design pressure (P_a) and temperature (T_a).

[10] Strength design by ACI 318 is not used for local areas. Analysis and design is as described in <Section 3.5.3>.

[11] In computing the required section strength, S, the plastic section modulus of steel shapes may be used.

[12] Thermal stresses on structural elements, except plate structures, are permitted to be neglected when they are shown to be secondary and self-limiting in nature and when the material is ductile. Stresses in plate structures are limited to the ASME limit of $3S_m$.

[13] Operation of the RHR heat exchanger safety relief valves.

[14] For definition of load symbols, see <Section 3.8.3.3>.

TABLE 3.8-5

LOAD COMBINATIONS FOR STEEL STRUCTURES OUTSIDE CONTAINMENT

Load Combination	Stress[6] Criteria	D	L[1]	W	W_t	F_{eqo}	F_{eqs}	Z	T_o	P_a	T_a	M	R_o	R_a	Y_r	Y_j	Y_m	P_r	Remarks
Normal Operation	1.0 S	1.0	1.0																
	1.5 S	1.0	1.0						1.0				1.0						
Normal Operation with Severe Environmental	1.0 S	1.0	1.0	1.0															
	1.5 S	1.0	1.0	1.0					1.0				1.0						
	1.0 S	1.0	1.0			1.0													
	1.5 S	1.0	1.0			1.0			1.0				1.0						
Normal Operation with Extreme Environmental	1.6 S	1.0	1.0		1.0			1.0	1.0				1.0						
	1.6 S	1.0	1.0				1.0		1.0				1.0						
Normal Operation with Abnormal Loads	1.6 S	1.0	1.0							1.0	1.0			1.0					See Note[4]
	1.6 S	1.0	1.0							1.0		1.0		1.0					Interior Missile (for local areas)[3]
	1.6 S	1.0	1.0						1.0				1.0					1.0	Gas Pipeline or Gas Storage Explosion
Normal Operation with Severe Environmental & Abnormal Loads	1.6 S	1.0	1.0			1.0				1.0	1.0			1.0	1.0	1.0	1.0		See Note[2][4]
Normal Operation with Extreme Environmental & Abnormal Loads	1.6 S	1.0	1.0				1.0			1.0	1.0			1.0	1.0	1.0	1.0		See Note[2][4]
	1.6 S	1.0	1.0		1.0							1.0							Tornado Missile (for local areas) See Note[3]

NOTES:

[1] Both cases of L having full value or being completely absent are considered.
[2] The maximum values of Y_r, Y_j and Y_m will be used simultaneously unless time histories are used to justify otherwise.
[3] Strength design by ACI 318 is not used for local areas. Analysis and design is as described in <Section 3.5.3>.
[4] Thermal stresses are neglected when they are shown to be secondary and self-limiting in nature and when the material is ductile.
[5] For definition of load symbols, see <Section 3.8.4.3>.

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TABLE 3.8-5 (Continued)

NOTES: (Continued)

(6)   For Normal Operations without thermal loads, includes dead load, live load, wind, OBE and SRVA all under service conditions the
      33 1/3% increase for wind and earthquake is not permitted.

For Extreme and Abnormal Loads, includes seismic, tornado, accident, pipe rupture and missiles, but excludes pipe whip restraints: members
in bending, tension or compression use 1.6 times the applicable AISC allowable stress for all combinations except SSE combined with
accident; members in bending, tension or compression use 1.7 times the applicable AISC allowable stress for SSE plus accident combination;
members in shear use 1.5 times the applicable AISC allowable stresses; members in bearing use 1.33 times the allowable stresses of AISC
Sections 1.5.1.5 and 1.5.2.

TABLE 3.8-6

LOAD COMBINATIONS FOR CONCRETE STRUCTURES OUTSIDE CONTAINMENT

Load Symbol[4]

Load Combination	D	L[1]	G	H	W	W_t	F_{eqo}	F_{eqs}	Z	T_o	P_a	T_a	M	R_o	R_a	Y_r	Y_j	Y_m	P_r	Remarks
Normal Operation	1.4	1.7	1.4	1.7						1.3				1.3						
Normal Operation with Severe Environmental	1.4	1.7	1.4	1.7	1.7					1.3				1.3						
	1.2		1.4	1.3	1.7															
	1.4	1.7	1.4	1.7			1.9			1.3				1.3						
	1.2		1.4	1.3			1.9													
Normal Operation with Extreme Environmental	1.0	1.0	1.0	1.0		1.0			1.0	1.0				1.0						
	1.0	1.0	1.0	1.0				1.0		1.0				1.0						
Normal Operation with Abnormal Loads	1.0	1.0	1.0	1.0							1.5	1.0			1.0					
	1.0	1.0									1.0		1.0		1.0					
	1.0	1.0	1.0	1.0							1.0				1.0				1.0	Interior Missile Explosions See Note[3]
Normal Operation with Severe Environmental & Abnormal Loads	1.0	1.0	1.0	1.0			1.25				1.25	1.0			1.0	1.0	1.0	1.0		See Note[2]
Normal Operation with Extreme Environmental & Abnormal Loads	1.0	1.0	1.0	1.0				1.0			1.0	1.0			1.0	1.0	1.0	1.0		See Note[2]
	1.0	1.0				1.0			1.0				1.0							Tornado Missile (for local areas) See Note[3]

NOTES:

[1] Both cases of L having full value or being completely absent are considered.

[2] The maximum values of Y_r, Y_j and Y_m will be used simultaneously unless time histories are used to justify otherwise.

[3] Strength design by ACI 318 is not used for local areas. Analysis and design is as described in <Section 3.5.3>.

[4] For definition of load symbols, see <Section 3.8.4.3>.

TABLE 3.8-7

LOAD COMBINATIONS FOR TUNNELS AND OFFSHORE STRUCTURES

Load Combinations	Stress Criteria	Load Symbols[4]										Remarks
		D	L	G	G'	H'	E	F_{eqo}	F_{eqs}	T_o	T'	
Cooling Water Tunnels												See Note[2]
Normal Operation		1.4		1.4	1.4	1.7				1.3		
Normal Operation plus Operating Basis Earthquake (OBE)		1.4		1.4	1.4	1.7		1.9		1.3		
Safe Shutdown Earthquake (SSE)		1.0		1.0	1.0	1.0			1.0		1.0	
Offshore Intake and Discharge Structures												See Note[2]
Normal Operation	1.0 S	1.0	1.0									
	1.0 S	1.0	1.0							1.0		
Normal Operation plus Extreme Environmental	1.0 S	1.0	1.0				1.0			1.0		See Note[1]
	1.5 S	1.0	1.0				1.0			1.0		See Note[1]
Operating Basis Earthquake (OBE)	1.0 S	1.0	1.0					1.0				
	1.5 S	1.0	1.0				1.0	1.0		1.0		See Note[1]
Safe Shutdown Earthquake (SSE)	1.6 S	1.0	1.0				1.0		1.0		1.0	See Note[1]

Load Symbols[4]

Load Combinations	Stress Criteria	D	L	G	G'	H'	E	F_{eqo}	F_{eqs}	T_o	T'	Remarks
Ice Protection Structures												See Note[2]
Normal Operation plus Extreme Environmental							1.0					See Note[1][3]

NOTES:

[1] E = Extreme Environmental Effects of Wave or Ice Loadings.
[2] Design loads are described in <Section 3.8.4.3.1>.
[3] A load factor of 1.0 was used for ice loading (E) on the ice protection structures due to the high degree of conservatism inherent in the assumptions made in developing the ice loads. See <Section 3.8.4.3.1.m>. An earthquake loading was not considered to occur simultaneously with the ice loading impact upon the protection structures since these are considered to be two unrelated, extreme events.
[4] For definition of load symbols, see <Section 3.8.4.3>.

TABLE 3.8-8

SUMMARY OF MATERIAL SPECIFICATIONS AND QUALITY CONTROL[1]

Safety Class Structure	Conc. Strength	Reinforcement	Structural Steel	Stainless Steel	Special Construction Techniqs.	Tolerances
Auxiliary Building	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	ASTM A 36 (3.8.3.6)	–	None	ACI 301 & AISC (3.8.3.6)
Fuel Handling Building	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	ASTM A 36 (3.8.3.6)	ASTM A 240 (3.8.3.6)	None	ACI 301 & AISC (3.8.3.6)
Control Complex	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	ASTM A 36 (3.8.3.6)	None	None	ACI 301 & AISC (3.8.3.6)
Radwaste Building	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	ACI 301 (3.8.3.6)
Intermediate Building	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	ACI 301 (3.8.3.6)
Offgas Building	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	ACI 301 (3.8.3.6)
Steam Tunnels	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	ACI 301 (3.8.3.6)
Emergency Service Water Pumphouse	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	ACI 301 (3.8.3.6)
Diesel Generator Building	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	ACI 301 (3.8.3.6)
Underdrain Manholes	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	
Electrical Manholes	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	

TABLE 3.8-8 (Continued)

Safety Class Structure	Conc. Strength	Reinforcement	Structural Steel	Stainless Steel	Special Construction Techniqs.	Tolerances
Condensate Storage Pit	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	
Condensate Storage Tank Foundations and Dike Walls	3,000 (3.8.1.6)	ASTM A 615-60 (3.8.1.6)	None	None	None	
Cooling Water Tunnels	4,000 (3.8.4.6)	ASTM A 615-60 (3.8.1.6)	None	None	(3.8.4.6)	(3.8.4.6)
Offshore Structures	4,000 (3.8.4.6)	ASTM A 615-60 (3.8.1.6)	ASTM A 441 or ASME SA-299 (3.8.4.6)	None	(3.8.4.6)	ACI 301 (3.8.3.6)

NOTE:

[1] USAR Sections noted in parantheses ( ) describe the applicable requirements.

TABLE 3.8-9

MAJOR SAFETY CLASS STRUCTURES

Structure	Type of Foundation	Length (feet)	Breadth (feet)	Thickness (feet)	Bearing Material
Reactor Buildings (Two)	Mat	136' diameter		12.5	Chagrin shale See Note[5]
Auxiliary Buildings (Two)	Mat	192	102	6.0	Chagrin shale See Note[5]
Intermediate Building (One)	Mat and Caissons	223	243	9.5	Chagrin shale See Note[5]
Control Complex (One)	Mat	148	142	6.0	Chagrin shale See Note[4][5]
Radwaste Building (One)	Mat	217	97	4.0	Chagrin shale See Note[5]
Underdrain Manholes	See Note[3]	See Note[3]	See Note[3]	See Note[3]	See Note[3]
Diesel Generator Building (One)	Mat	170	79	4.5	Load bearing fill See Note[2][5]
Offgas Buildings (One)	Mat	100	50	4.0	Load bearing fill See Note[2]
Emergency Service Water Pump House (One)	Mat	95	55	5.0	Chagrin shale See Note[5]

TABLE 3.8-9 (Continued)

Structure	Type of Foundation	Length (feet)	Breadth (feet)	Thickness (feet)	Bearing Material
Offshore Intake Structures (Two)	See Note[1]	–	–	–	–
Offshore Discharge Structure (One)	See Note[1]	–	–	–	–
Cooling Water Tunnels (Two)	See Note[1]	–	–	–	–

NOTES:

[1] The foundation of each structure is part of the structure and cannot be readily separated. <Section 3.8.4> gives a description of these structures.

[2] The load bearing fill is supported by lower till.

[3] Physical description is given in <Section 2.4.13.5>.

[4] Fill concrete, f'_c = 1,500 psi is used under this foundation as a rock-like medium between the porous concrete and the bottom of the mat foundation.

[5] A 12" layer of porous concrete, described in <Section 2.4.13.5> and <Section 3.8.5.3.4>, forms a supporting medium between the bottom of the foundation mat and the bearing material.

TABLE 3.8-10

FACTORS OF SAFETY AGAINST FLOTATION, OVERTURNING AND SLIDING
(Water at Elevation 590'-0" w/Underdrain System)

Structure	Flotation	Overturning		Sliding	
		OBE	SSE	OBE	SSE
Auxiliary Building (Mat Bottom at Elevation 564'-4")	2.5	2.8	1.7		See Note[1]
Control Building (Mat Bottom at Elevation 568'-10")	3.1	2.1	1.2		See Note[1]
Radwaste Building (Mat Bottom at Elevation 569'-10")	3.7	2.2	1.4		See Note[1]
Reactor Building Complex (Mat Bottom at Elevation 562'-3")	5.8	3.7	2.0	2.05	1.11
Intermediate Building (Mat Bottom at Elevation 568'-10")	3.9	4.8	2.5		See Note[1]
Offgas Building (Mat Bottom at Elevation 580'-0")	6.6	3.5	1.8		See Note[1]

NOTE:

[1] Shear keys are provided to insure minimum factors of safety (1.5 for OBE and 1.1 of SSE) are met.

TABLE 3.8-11

COMPARISON OF CONTAINMENT VESSEL STRESSES FOR THE
GOVERNING LOAD COMBINATIONS AT KEY LOCATIONS WITH ASME SECTION III DIVISION ALLOWABLES

Location	Governing Load Combination	STRESS INTENSITY WITH ALLOWABLE PSI					
Reference <Figure 3.8-17> "Containment Vessel Finite Element Model" and <Figure 3.8-18> "Containment Vessel Embedment Model"		Primary Membrane (Pm)	Allowable	Local Primary Membrane Plus Primary Bending (Pl + Pb)	Allow-able	Primary Membrane Plus Primary Bending Plus Secondary (Pl + Pb + Q)	Allow-able
Junction of Cylinder and Foundation Mat-Section above Nodal Points 438, 440, 442 & 444 <Figure 3.8-18> at Elev. 574'-10" in Doubler Plate	$S = D + L + G + F_{eqs} + P_{srv} + T_{srv} + P_a + T_a + R_a + Y_r$	14,450	23,160	19,118	45,000	42,278	57,900
Junction of Cylinder and Foundation Mat-Section above Nodal Points 447, 448, 450 452 in <Figure 3.8-18> at Elev. 574'-10" in Containment Shell	$S = D + L + G + F_{eqs} + P_{srv} + T_{srv} + P_a + T_a + R_a + Y_r$	3,844	23,160	26,434	45,000	44,085	57,900
Below stiffener No. 1 at Elev. 578'-0" as shown in <Figure 3.8-17>, Element 6	$S = D + L + G + F_{eqo} + P_{srv} + T_{srv} + P_a + T_a + R_a + Y_r$	11,516	19,300	–	–	48,370	57,900
Stiffner No. 1 at Elev. 581'-2" as shown in <Figure 3.8-17>, Element No. 9	$S = D + L + G + F_{eqo} + P_{srv} + T_{srv} + P_a + T_a + R_a + Y_r$	5,074	19,300	–	–	49,832	57,900
Junction of Cylinder and Polar Crane Girder as shown in <Figure 3.8-17>, Elements 57 & 62	$S = D + L + G + F_{eqo} + P_{srv} + T_{srv} + P_a + T_a + R_a + Y_r$	8,437	19,300	9,147	28,950	9,147	57.900
	$S = D + L + T_o + R_o + G' + F_{eqo}$	12,915	28,950	41,988	57,900	41,988	57,900
Junction of Cylinder and Dome @ Elev. 727'-0" as shown in <Figure 3.8-17>, Element 65	$S = D + L + G + F_{eqo} + P_{srv} + T_{srv} + P_a + T_a + R_a + Y_r$	12,199	19,300	14,083	28,950	14,083	57,900

TABLE 3.8-12

DEISGN AXIAL FORCES, MOMENTS AND TRANSVERSE SHEARS AND REINFORCEMENT PROVIDED
FOR THE GOVERNING LOAD COMBINATIONS AT CRITICAL SECTIONS OF THE SHIELD BUILDING

Location	Governing Load Combination	Forces (Kips Per Foot and Kip-Feet Per Foot) (-M is Tension O.F.)					Reinforcement Provided		
Reference <Figure 3.8-9> "Analytical Model of Shield Building"		Vertical Membrane $N\phi$	Vertical Bending Moment $M\phi$	Hori-zontal Membrane $N\theta$	Hori-zontal Bending Moment $M\theta$	Trans-Verse Shear V	Vertical	Hori-zontal	Shear
Junction of Cylinder and Ring Girder-Section above Nodal Points 445-451	$U=1.4D+1.7L+ 1.4G+1.9F_{eqo}+ 1.3T_o+1.7H+ 1.7P_{srv}+1.3T_{srv}+ 1.3R_o$	-55.1 $K/_f$	-123.6 $K-f/_f$	183.3 $K/_f$	-21.4 $K-f/_f$	36.2 $K/_f$	#9 @ 12" E.F.	#11 @ 6" E.F.	#5 @ 24" Horiz. and 15" Vert.
Junction of Mat and Cylinder Section above Nodal Points 1-7. Soil Pressure Area Abat. C_L 97.5° Azimuth as described in <Section 3.8.1.1.a> Steam Tunnel Area Abt C_L 0° Azimuth as described in <Section 3.8.1.1.b> General Area as described in <Section 3.8.1.1.c>	$U=1.2D+1.4G+ 1.3H+1.9F_{eqo}+ 1.4 P_{srv}$ or $U=1.0D+1.0L+ 1.0G+1.0H+ 1.0F_{eqs}+ 1.0P_{srv}+ 1.0T_{srv}+1.0P_a+ 1.0T_a+1.0R_a+1.0 (Y_r+Y_j+Y_m)$	159.4 $K/_f$ 146.1 $K/_f$	-296.6 $K-f/_f$ -293.2 $K-f/_f$	32.2 $K/_f$ 118.9 $K/_f$	-53.7 $K-f/_f$ 66.9 $K-f/_f$	82.6 $K/_f$ 79.7 $K/_f$	#18 @ 12" Alt W/#11 @ 12" O.F. #11 @12" Alt. W/#9 @ 12" I.F.	#9 @ 6" E.F.	
	$U=1.0D+1.0L+ 1.0G+1.0F_{eqs}+ 1.0P_{srv}+1.0T_{srv}+ 1.0P_a+1.0T_a+ 1.0R_a+1.0 (Y_r+Y_j+ Y_m)$	163.8 $K/_f$	74.3 $K-f/_f$	31.0 $K/_f$	11.9 $K-f/_f$	41.6 $K/_f$	#14 @ 12" Alt. W/#11 @ 12" O.F. and #11 @ 6" I.F.	#9 @ 6" E.F.	#6 @ 24" Horiz. and 15" Vert.
	$U=1.0D+1.0L+ 1.0G+1.0F_{eqs}+ 1.0P_{srv}+1.0T_{srv}+ 1.0P_a+1.0T_a+ 1.0R_a+1.0 (Y_r+Y_j+ Y_m)$	109.8 $K/_f$	221.8 $K-f/_f$	10.9 $K/_f$	40.1 $K-f/_f$	60.4 $K/_f$	#11 @ 12" Alt. W/#9 @12" E.F.	#9 @ 6" O.F. and #9 @ 12" I.F.	#6 @ 24" Horiz. and 15" Vert.

TABLE 3.8-13

ASME SECTION III DIVISION 2 LOAD COMBINATIONS

Load Combination	Stress[10] Criteria	Load Symbols[11]																		Remarks
		D	L[1]	G	G′	W[2]	F_{eqo}	F_{eqs}	T_o	P_{srv}	T_{srv}[3]	P_a[9]	T_a	R_o	R_a	P_t[4]	T_t	G_{sf}	W_t	
Service Load Conditions																				
Testing		1.0	1.0	1.0											1.0	1.0		1.0	1.0	See Note[5]
		1.0	1.0	1.0					1.0	1.0	1.0				1.0				1.0	See Note[5]
Construction		1.0	1.0	1.0		1.0			1.0											
Normal Operating		1.0	1.0	1.0					1.0	1.0	1.0				1.0				1.0	See Note[6]
Severe		1.0	1.0	1.0			1.0		1.0	1.0	1.0				1.0				1.0	See Note[6]
Environmental		1.0	1.0	1.0		1.0			1.0	1.0	1.0				1.0				1.0	See Note[6]
Factored Load Conditions																				
Severe Environmental		1.0	1.3	1.0			1.5		1.0	1.3	1.3				1.0				1.0	See Note[7],[8]
		1.0	1.3	1.0		1.5			1.0	1.3	1.3				1.0				1.0	See Note[7],[8]
Extreme Environmental		1.0	1.0	1.0				1.0	1.0	1.0	1.0				1.0				1.0	See Note[7],[8]
		1.0	1.0	1.0					1.0	1.0	1.0				1.0			1.0	1.0	See Note[7],[8]
Abnormal		1.0	1.0	1.0						1.25	1.0	1.5	1.0		1.0				1.0	See Note[7],[8]
		1.0	1.0	1.0						1.0	1.0	1.0	1.0			1.25			1.0	See Note[7],[8]
Abnormal/Severe Environmental		1.0	1.0	1.0			1.25			1.0	1.0	1.25	1.0		1.0				1.0	See Note[7],[8]
		1.0	1.0	1.0		1.25				1.0	1.0	1.25	1.0		1.0				1.0	See Note[7],[8]
		1.0	1.0	1.0	1.0		1.0		1.0								1.0			
		1.0	1.0	1.0	1.0	1.0			1.0								1.0			
Abnormal/Extreme Environmental		1.0	1.0	1.0				1.0		1.0	1.0	1.0	1.0		1.0				1.0	See Note[7],[8]

NOTES:

(1) Both cases of L having full value or being completely absent are considered.

(2) The shielding building transmits wind loads to the annulus concrete.

(3) The maximum or minimum suppression pool temperature producing the worst thermally induced stresses for the case considered.

(4) Structural integrity test of the containment vessel.

(5) Testing safety/relief valves.

(6) Operation of safety/relief valves as described in <Appendix 3A> and <Appendix 3B>.

(7) Operation of a single relief valve first pop combined with LOCA pool swell pressure loadings.

(8) For a small line or intermediate size break, the design includes the operation of a single or multiple safety/relief valve as described in <Appendix 3A> and <Appendix 3B> with the appropriate design pressure (P_a) and temperature (T_a).

(9) P_a represents pool swell pressure, condensation oscillation pressure, chugging pressure, and the design accident pressure.

(10) For stress criteria, see <Section 3.8.1.8.5>.

(11) For definition of load symbols, see <Sections 3.8.1.8.3, 3.8.1.3, and 3.8.2.3>.

3.9   MECHANICAL SYSTEMS AND COMPONENTS

3.9.1   SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1   Design Transients

A summary of design transients used in the design and/or fatigue
analysis of a typical plant are listed in <Table 3.9-1>. The number of
cycles or events associated with each transient are included.
Transients or combinations of transients are classified with respect to
the plant and system operating condition categories identified as
"Normal," "Upset," "Emergency," "Faulted" (service levels A, B, C, D,
respectively), or "Testing" in the ASME Code, Section III, Division I,
NA-2140, as applicable.

The ASME Code Class 1 components not supplied by the NSSS vendor are
comprised of piping, valves, containment penetrations, and pipe
supports. These components have been specified and designed in
accordance with the system design transients listed in <Table 3.9-2>.

The design transients identified in <Table 3.9-1> and <Table 3.9-2> are
not design limits. Design limits are established by the design code for
the structure system or component. In some special cases, the design
limit may be maintained at a lower value than that established by the
governing design code. For example, to avoid high energy pipe breaks,
design limits may be reduced to avoid postulation of pipe ruptures.

A component cyclic or transient monitoring program is used to monitor
plant operations and ensure the reactor vessel is maintained within
design limits.

3.9.1.2   <u>Computer Programs Used in Analyses</u>

<Section 3.9.1.2.1>, <Section 3.9.1.2.2>, <Section 3.9.1.2.3>, <Section 3.9.1.2.4>, and <Section 3.9.1.2.5> discuss computer programs used in the analysis of major NSSS safety-related components. (Computer programs were not used in the analysis of all components, thus, not all components are listed.) These NSSS computer programs can be divided into two categories, GE programs and vendor programs.

<Section 3.9.1.2.6>, <Section 3.9.1.2.7>, <Section 3.9.1.2.8>, and
<Section 3.9.1.2.9> discuss computer programs used in the analysis of
balance-of-plant components.

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.

a.	SEISMO2	l.	PDA
b.	STRESINT	m.	PISYS
c.	MASS	n.	LI0N401
d.	DYSEA	o.	RINEX
e.	FAP-71	p.	IMPULSE-1
f.	ANSYS	q.	BILRD
g.	CRDSS01	r.	WBHFN
h.	ANSI 7	s.	KK-8
i.	RVFOR	t.	GEAPL01
j.	TSFOR	u.	SPECA04
k.	LUGST	v.	ASHSDO2

GE Vendor Programs

The verification of the following two groups of vendor programs is
assured by contractual requirements between GE and the vendors. Per the
requirements, the quality assurance of these proprietary programs used
in the design of N-stamped equipment is in full compliance with
<10 CFR 50, Appendix B>.

Byron-Jackson Programs

a.   Recirculation Pump Programs   e.   MULTISPAN

b.   RTRMEC   f.   2DFMAP

c.   FMAP   g.   CRISP

d.   FLTFLG

CB & I Programs

a.   7-11 GENOZZ   m.   1037-DUNHAM'S

b.   9-48 NAPALM   n.   1335

c.   1027   o.   1606 & 1657-HAP

d.   846   p.   1635

e.   781-KALNINS   q.   953

f.   979-ASFAST   r.   1666

g.   766-TEMAPR   s.   1684

h.   767-PRINCESS   t.   E1702A

i.   928-TGRV   u.   955-MESHPLOT

j.   962-E0962A   v.   1028

k.   984   w.   1038

l.   992-GASP

Other Vendor Programs

a.   pc-Crack

b.   ANSYS

3.9.1.2.1   Reactor Pressure Vessel and Internals

The computer programs used in the preparation of the stress report for
the reactor pressure vessel and internals stress report are identified
and their use summarized in the following paragraphs.

a. Reactor Pressure Vessel

    The following computer programs are used to assure the structural
    and functional integrity of the reactor pressure vessel.

    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 to it. If the input configuration will
        not comply with the specifications, the program will redesign
        it to yield an acceptable result.

    2. CB&I Program 9-48 "NAPALM"

        The basis for the program NAPALM, Nozzle Analysis Program--All
        Loads Mechanical, is to analyze nozzles for mechanical loads
        and find the maximum stress intensity and location at
        specified locations from the point of load application. At
        each location the maximum stress intensity is calculated for
        both the inside and outside surfaces of the nozzle. The
        program gives the maximum stress intensity for both the inside
        and outside surfaces of the nozzle as well as its angular
        location around the circumference of the nozzle from the
        reference location. The principal stresses are also printed.
        The stresses resulting from each component of loading
        (bending, axial, shear, and torsion) are printed, as well as
        the loadings which caused these stresses.

3.   CB&I Program 1027

      This program is a computerized version of the analysis method
      contained in the Welding Research Council Bulletin No. 107,
      December 1965 (Reference 1).

      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.

4.   CB&I Program 846

      This program computes the required thickness of a
      hemispherical head with a large number of circular parallel
      penetrations by means of the area replacement method in
      accordance with the ASME Code, Section III.

      In cases where the penetration has a counterbore, the
      thickness is determined so that the counterbore does not
      penetrate the outside surface of the head.

5.   CB&I Program 781 – "KALNINS"

      This program is a thin elastic shell program for shells of
      revolution. This program was developed by Dr. A. Kalnins of
      Lehigh University. Extensive revisions and improvements have

been made by Dr. J. Endicott to yield the CB&I version of this program.

The basic method of analysis was published by Professor Kalnins in the Journal of Applied Mechanics, Volume 31, September 1964, Pages 467 through 476 (Reference 2).

The KALNINS thin shell program (Program 781) is used to establish the shell influence coefficient and to perform detailed 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.

(c) Thermal load.

6. CB&I Program 979 - "ASFAST"

ASFAST program (Program 979) performs the stress analysis of axisymmetric, bolted closure flanges between head and cylindrical shell.

7. CB&I Program 766 - "TEMAPR"

This program will reduce any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resultant equivalent gradient will have the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plate thickness and actual temperature distribution. The output contains the

average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV language.

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 Code.

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 is 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.

10. CB&I Program 962 - "E0962A"

Program E0962A is one of a group of programs (E0928A, E0962A, E1606A, 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.

11. CB&I Program 984

Program 984 is used to calculate the stress intensity of the stress differences, on a component level, between two

different stress conditions. The calculation of the stress intensity of stress component differences (the range of stress intensity) is required by Section III of ASME Code.

12. CB&I Program 992 – "GASP"

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 and is written in FORTRAN IV language (Reference 3). The structures may have arbitrary geometry and linear or nonlinear 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 a 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.

13. CB&I Program 1037 - "DUNHAM'S"

DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacements of 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.

14. CB&I Program 1335

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).

15. CB&I Programs 1606 and 1657 - "HAP"

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 axisymmetric body subjected to transient disturbances. Programs 1606 and 1657 are identical except that 1606 has a larger storage area allocated and can be used to solve larger problems. The model for Program 1606 is compatible with CB&I stress Programs 992 and 1037.

16. CB&I Program 1635

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 that can be
       incorporated into a final stress report.

(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.

17.   CB&I Program 953

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.

(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. Since they have the same number, there is no
       possibility of confusion if transferring temperature data
       from one program to the other.

18.   CB&I Program 1666

This program is primarily written to calculate the temperature
differences at selected critical sections of the nuclear
reactor vessel components at different time points of thermal
transients during its life of operation and list them in a
tabular form. Since there is no calculation unique to nuclear
components, this program can be used with any model that sees
thermal transients over a period of time. This program helps
ascertain the time points in thermal transients when the
thermal stresses may be critical.

19. CB&I Program 1684

This program is written to better expedite the fatigue analysis of nuclear reactor components as required by the ASME Code, Section III. Specifically, this program is an expansion of an earlier program, 984. The features of this program allow the user to easily perform the complete secondary stress and fatigue evaluation, including partial fatigue usage calculation of a component, in one run. An additional option allows the user to document the input stress values in a format suitable for a stress analysis report. The program is written to allow for a minimum amount of data handling by the user after the initial deck is established.

20. CB&I Program "E1702A"

This program evaluates the stress intensity factor K sub (I) due to pressure, temperature and mechanical load stresses for a number of different stress conditions (times) and at a number of different locations (elements). It then calculates the maximum RT sub (NDT) the actual material can have, based on a 1/4T flaw size and compares it with the ordered RT sub (NDT). If the ordered RT sub (NDT) is larger than the maximum RT sub (NDT), the maximum allowable flow size is calculated. The rules of ASME Code, Appendix G are used except that WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," can be used to calculate K sub (I) due to pressure in a nozzle-to-shell junction.

For a more thorough description of the fracture problem, see WRC Bulletin No. 175.

21. CB&I Program 955 "MESHPLOT"

   This program plots input data used for finite element
   analysis. The program plots the finite element mesh in one of
   three ways: without labels, with node labels or with element
   labels. The output consists of a listing and a plot. The
   listing gives all node points with their coordinates, and all
   elements with their node points. The plot is a finite element
   model with the requested labels.

22. 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.

23. 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.

24. pc-CRACK

   pc-CRACK was used to determine the stresses and support the
   design of the weld overlay for the feedwater nozzle to
   safe-end weld 1B13-N4C-KB. pc-CRACK is a personal computer
   based program for performing fracture mechanics analysis on a
   variety of structural components and materials. The program
   has the capability to perform rapid linear elastic fracture
   mechanics, elastic-plastic fracture mechanics, and limit load
   analysis. The program also provides a graphical

representation of the analyzed area. This program is an
Appendix B software program by Structural Integrity
Associates, Inc. This program has been previously used for
BOP piping as described in <Section 3.9.1.2.6.cc>.

25. ANSYS is a general-purpose finite element computer program.
     Applications include reactor components inclusive of nozzle
     stress determination as well as determination of the increases
     in stud and vessel stresses due to reduced pass tensioning and
     tension sequencing of the reactor vessel closure flange. For
     a more thorough software description see
     <Section 4.1.4.1.4.1>, "Program Description".

b. Reactor Internals

1. Fuel Support Loads Program/SEISMO2, SEISMO3V

    SEISMO2 and SEISMO3V computes the vertical fuel support loads
    using the component element method in dynamics. The
    methodology is based on the publication "The Component Element
    Method in Dynamics," by S. Levy and J. P. D. Wilkinson, McGraw
    Hill Co., New York, 1976.

2. Stress Intensities Program/STRESINT

    The time-share program STRESINT is used to calculate the
    principal stresses and intensities of a stress tensor.
    STRESINT computes the principal stresses by obtaining roots of
    an eigenvalue cubic equation. Stress intensities are found by
    taking the absolute values of the differences between the
    principal stresses.

3.   Other Programs

         The following programs are also used in the analysis of core

         support structures and other safety-related reactor

         internals: MASS, DYSEA, FAP-71, and ANSYS. These programs

         are described in detail in <Section 4.1>.

c.   Control Rod Drive

   The program CRDSS01 is used to obtain performance data for various
   inputs to the control rod drive stress analysis for both code and
   non-code parts. Experimental data on such items as pressure drops,
   friction factors and effects of misalignment are used in setting up
   and perfecting this code. Internal drive pressures and
   temperatures used in the stress analysis are also determined during
   testing of the prototype control rod drives.

3.9.1.2.2   Piping

The computer programs used in the analysis of NSSS piping systems are
identified and their use summarized in the following paragraphs.

a.   Component Analysis/ANSI 7

   The ANSI 7 computer program determines stress and cumulative 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.

b.   Dynamic Forcing Functions

   1.   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 on the pipe
      wall. This program computes the transient fluid mechanic
      parameters and the resultant pipe forces using the method of
      characteristics.

2.   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.   Integral Attachment/LUGST

      The computer program "LUGST" evaluates the stresses in the
      pipe wall that are produced by loads applied to the integral
      attachments.

4.   Piping Dynamic Analysis Program/PDA

      Pipe whip analyses were performed using the PDA computer
      program, which determines 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
      configurations, which involve a straight, uniform pipe fixed
      at one end and subjected to a time dependent thrust force at
      the other. A typical restraint used to reduce the resultant
      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. On the basis of 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.

5.   Piping Analysis Program/PISYS

PISYS is a computer code 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 is possible. PISYS is based on classical linear
elasticity theory 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 a report to the Commission, "PISYS Analysis of
NRC Problems," NEDO-24210, August 1979.

6.   Piping Analysis Temperature Gradient Program: LION401

LION401 is a digital computer program which is used to solve
the steady-state or transient temperature distribution in any
three-dimensional configuration. The heat source may be
externally conducted or internally generated.

In addition to the solving of heat conduction in structural elements, LION401 may also be used in such cases as forced convection, free convection or radiation where the output will yield temperatures and heat fluxes for points representing the surface of the structure.

The program solves the transient heat conduction equations for a three-dimensional field using a first forward difference method.

The main purpose of the program at the present time is to calculate the temperature gradient for ASME Section III NB-3600 Class I piping analysis.

7.   Response Spectrum Method Program: RINEX

RINEX is an Engineering Computer Program (ECP) with the capability to interpolate and extrapolate curves of Amplified Response Spectrum (ARS) used in the response spectrum method of dynamic analysis. RINEX has, as options, the capability to: (1) generate additional curves of ARS with different oscillator damping from two or more existing curves and (2) incorporate nonconstant modal damping into the dynamic analysis of seismic and non-seismic loads in the nuclear power plants. The nonconstant modal damping analysis can calculate spectral acceleration at the discrete eigenvalues of a dynamic system using either the strain energy weighted modal damping or the ASME-PVRC damping curve. An additional damping analysis includes an option to create a continuous response spectrum curve corresponding to the ASME-PVRC damping curve. The advantage of such a continuous curve is that it defines the input that is directly applicable to any dynamic system.

3.9.1.2.3   Pumps and Motors

a.   Recirculation Pump

Computer programs using finite element methods are used in the analysis of the recirculation pump casing for the various loading conditions imposed by thermal and mechanical loads during plant operating and postulated conditions.

In general, the finite element techniques are used to solve heat transfer transient analysis problems, and also to perform stress analysis for various thermal and mechanical loadings by using the same finite element model representing the pump body. The output of these programs is in the form of temperature profiles, deflections and stresses at the modal points of the finite element idealization of the pump structure.

The pump vendors used either the accepted industry or public domain computer codes, or their own developed proprietary program. Vendors have to comply with the quality assurance and verification requirements of <10 CFR 50, Appendix B> as part of the contract conditions imposed by the General Electric Company. In addition, accuracy of computer program methods is demonstrated by comparison with selected sample problems which lend themselves to alternate computation methods to provide closed-form analytical solutions.

b.   ECCS Pumps and Motors

RTRMEC is a computer program which calculates and displays results of mechanical analysis of a motor rotor assembly when acted upon by external forces at any point along the shaft (rotating parts only). The shaft deflection analysis includes magnetic and centrifugal forces. The calculation for the seismic conditions 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. 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.

The following computer programs were used to analyze rotating and stationary components in the ECCS pumps:

1. FMAP is a computer program which solves for the natural frequencies and associated mode shapes of a two-dimensional frame. The frame is defined as a system of uniform, weightless members whose ends, or joints, are rigidly attached. All weights are lumped at the joints. Each joint has three degrees of freedom: two translations in the plane of the frame and a rotation about the axis normal to the plane. The frame is in the X-Y plane, and all motion of the frame is in this plane.

2. The FLTFLG computer program determines stresses in bolted flanged connections where the flanges are flat faced and bolted together directly or separated by a metal spacer such that there is metal-to-metal contact beyond the bolt circle.

    Calculation procedures follow rules set forth in Appendix II, Part B, ASME Boiler and Pressure Vessel Code Section VIII, Division 1, 1971, Winter 1973 Addendum.

3. MULTISPAN is a computer program which performs the bending analysis of variable cross-section continuous beams up to ten spans. The analysis yields reactions, internal forces, displacements, and maximum shear and bending stresses.

4. 2DFMAP is a computer program which solves for the natural frequencies and the associated mode shapes of a rigidly jointed, two-dimensional lumped-mass frame. The solution is based on small-deflection theory assuming linear stiffnesses for the frame. Stiffness matrix alterations can be used to add complex structural elements which cannot be represented by members. Gaussian elimination is available to reduce the size of the stiffness matrix if relatively small weights are associated with any degree of freedom. The frequencies and mode shapes are computed using the Householder-Strum method and inverse iteration.

5. The CRISP computer program determines the fundamental and harmonic modes of lateral vibration of rotating elements of arbitrary flexural rigidity. The computational method is based on a transfer matrix representation of the rotor shaft which includes the effect of multiple supports with dissimilar elasticity and damping in the bearings, and with dissimilar elasticity and mass of the bearing supports. In addition to calculating the natural frequencies, the program provides lateral deflections for the determination of rotor stresses, running clearances and severity of vibration at the different resonant frequencies. Vibration amplitudes of the bearing supports are also provided for determining support resonant frequencies and for obtaining an optimum design through modifications of the bearings and their supports.

3.9.1.2.4   RHR Heat Exchangers

Following are the computer programs used in dynamic and static analysis to determine structural and functional integrity of the RHR heat exchanger.

a. 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.

b. Natural Frequency Program/WBHFN

    WBHFN calculates the natural frequency of the RHR Heat Exchanger
    considering the stiffness of the supporting steel structure. In
    this program the heat exchanger is modeled as a flexible beam
    supported on two springs, the two springs being the stiffness
    characteristics of the upper keyway supports and the lower support
    tiedown brackets. Three different methods of calculating the
    natural frequency of the RHR heat exchanger could be used. All
    three methods were used to calculate the natural frequency of a
    typical system with all results being within 7% of each other.

c. Stress Analysis of Supports/KK-8

    KK-8 performs a full stress analysis of the upper and lower
    supports of the RHR heat exchanger. This program computes the
    membrane stresses on the shell of the heat exchanger by the use of
    the Bijlaard analysis, per Welding Research Council Bulletin
    No. 107, as well as the net section stresses (shear, tensile,
    bearing) in the lower support plate and the upper keyway brackets.
    It also computes the stresses in the welds of the supports as well
    as the stresses in the welds attaching the supports to the shell.

3.9.1.2.5   Annulus Pressurization Analysis

a.   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 GEAPL01.
     GEAPL01 converts pressure time histories into time varying loads
     and forcing functions for DYSEA. The overall time histories for
     resultant forces and moments at specified points of resolution can
     also be obtained from GEAPL01.

b.   Acceleration Response Spectrum Program/SPECA04

     SPECA04 generates acceleration response spectra 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 analyses.

c.   Axisymmetrical Structure Analysis Program/ASHSD02

     ASHSD02 performs dynamic analysis of axisymmetric structures
     subjected to any arbitrary static or dynamic loading or base
     acceleration. The finite element methodology is used to generate
     vibration mode shapes and frequencies, time history of nodal
     displacement and acceleration, and element stress for each Fourier
     term.

3.9.1.2.6   Balance of Plant Computer Programs

The computer programs used in the analyses of BOP components are
summarized in the following paragraphs.

All computer programs used for the design and qualification of BOP
safety-related components, whether used by GAI or by vendors supplying
the components, have been verified in accordance with the requirements
of <10 CFR 50, Appendix B>.

a.   PIPDYN II (M041)

      PIPDYN II is a FORTRAN-language computer program developed by the
      Franklin Institute Research Laboratories for the analysis of
      general three-dimensional beam structures, but more specifically
      for the static and dynamic analyses of piping systems. The
      essential part of the program is based on general structure theory
      with the aid of the finite element method. The program permits the
      user to describe the physical properties of a structure, constraint
      conditions and loading information in very general terms. The
      program provides alternative stress computations in accordance with
      requirements of either the subarticles "NB-3600, Piping Design" and
      "NC-3600, Piping Design" of the ASME Code, Section III, up to and
      including the Winter 1972 Addenda, or ANSI B31.1b-1973 Power Piping
      Code. The computer output results are updated to meet the
      requirements of the 1974 ASME Code, Section III, Division 1, up to
      and including the Winter 1975 Addenda. PIPDYN II is used in static
      and dynamic analysis of piping systems except where plasticity,
      nonlinearity or large displacement considerations are required.

      PIPDYN II is a recognized program and has sufficient history of use
      to justify its applicability and validity. The theory,
      assumptions, computational techniques, and results have been
      examined and found to be within acceptable standards. Independent

comparisons of results have been made to classical problems, recognized benchmark calculations and piping systems analyzed by using other programs, with acceptable results.

b.   TPIPE

TPIPE is a special purpose computer program capable of performing static and dynamic linear elastic analyses of power related piping systems of virtually unlimited size. The dynamic analysis option includes (1) frequency extraction, (2) response spectrum, (3) time history modal superposition, and (4) time history direct integration methods.

In addition to these basic analysis capabilities, the program can process plot requests of the undeformed or deformed structural geometry, postprocess pipe member end forces through ASME Section III, Class 1, 2 or 3 stress evaluation equations, and provide support load and data for pipe attachment evaluation. TPIPE can also perform thermal transient heat transfer analysis to provide the linear thermal gradient, nonlinear thermal gradient and gross discontinuity expansion difference required for a complete Class 1 stress evaluation.

Multiple class (1-2-3) piping systems can be analyzed together but individually evaluated by TPIPE within a single computer run or over a series of computer runs.

The major computational algorithms which solve the linear equilibrium equations and calculate the dynamic structural frequencies and mode shapes were taken from the efficient general purpose structural analysis program SAP IV, "A Structural Analysis Program for Static and Dynamic Response of Linear Systems."

The program was verified by comparison to benchmark problems.

c. Alternate Method of Piping Analysis Program (M085)

This program computes support spacing, support loads and pipe deflections for the combined effect of dead weight and seismic inertia load based on classical frequency domain analysis of beam vibration, and is used to develop a simplified seismic support spacing criteria. This method treats piping between supports as a single span beam and determines the basic spans for straight pipe of various diameters to meet stress and deflection criteria. Span reduction factors are developed to determine permissible lengths for those spans containing a concentrated weight, branch connection or change in direction.

The seismic loading on a pipe is specified by response spectrum inputs. Floor acceleration response spectra must be specified for 3 mutually orthogonal directions for the OBE and for the SSE. The pipe stress is allocated for dead weight and seismic load combinations from internal pressure, stress intensification factor and the allowable stress multipliers of the ASME Code. The calculation meets ASME B&PV Code Section III, Subsection NC requirements for dead weight, internal pressure, and seismic loading or other dynamic loads specified by floor response spectra.

The basic pipe spans are computed from the stress allocated for dead weight and dynamic loads for the OBE and the SSE response spectra. The shorter of these two spans is used for computing support loads and maximum deflection. This deflection is then checked against the limit imposed by the users.

The program was verified by the following two methods:
(1) comparison of results with hand calculations and (2) comparison of results obtained from the verified stress analysis program, TPIPE.

d. Flange Qualification Program

    This program was developed by Gilbert Associates, Inc. to calculate
    and evaluate stresses in the flange and bolts in accordance with
    the ASME Code, Section III, Appendix XI, Winter 1975 Addenda,
    Section III, Paragraphs NB-3647, NC-3647, ND-3647, and ASME Code
    Case 1677. The program has been verified by solving a series of
    test cases and comparing these results with hand calculations.

e. ANSYS

    ANSYS is a general purpose finite element computer program for the
    solution of diversified analysis problems. Analysis capabilities
    include static and dynamic; plastic, creep and swelling; small and
    large deflections; and steady-state and transient heat transfer.
    The program is used in both the linear and nonlinear analysis of
    special two and three dimensional components subjected to loads
    such as: shear, axial, bending, torsion, pressure, and
    temperature. Output consists of nodal displacements, stresses,
    temperatures, etc., for use in evaluating the component against
    required allowables.

    The program has been developed and verified by Swanson Analysis
    Systems Inc., Elizabeth, Pennsylvania.

f. STRUDL/DYNAL

    The Structural Design Language, STRUDL, is a recognized, widely
    used computer program for structural design developed by the
    Massachusetts Institute of Technology and released to public domain
    in 1968. This program is used in analysis of piping and equipment
    supports, and for thermal analysis of selected piping systems.
    Documentation for the verification of this computer program is on
    record at Gilbert Associates, Inc.

g. GTSTRUDL

GTSTRUDL is used in structural analysis and design to perform
linear small displacement static and dynamic analysis of structures
composed of any combination of members and finite elements with the
same or variable number of degrees-of-freedom per joint.

GTSTRUDL design procedures include steel design and code checking
for member elements in accordance with the 1969 and 1978 AISC
(American Institute of Steel Construction) Specifications for
general steel structures, the 1971 ASCE (American Society of Civil
Engineers) Manual No. 52, "Guide for Design of Steel Transmission
Towers" for steel transmission tower design, and the 1980
API (American Petroleum Institute) "Recommended Practice for
Planning, Designing and Constructing Fixed Offshore Platforms", for
the design of steel tubular members.

The program verification included a comparison against hand
calculations of sample problems pertaining to the AISC Code and the
ASME Code, Section III, Subsection NF.

h. WERCO

WERCO is a program written by AAA Technology and Specialties Co.,
Inc. to perform the stress calculations as presented in the Welding
Research Council (WRC) Bulletin No. 107, entitled "Local Stresses
in Spherical and Cylindrical Shells Due to External Loadings." The
program contains a series of non-dimensional curves that are used
to obtain the stresses at four locations around the attachment and
on the interior and exterior surfaces of the shell at those
locations. The program was verified by comparing the results of
sample program hand calculations with those of the WERCO program.

i.   PIPELUG (M083)

PIPELUG is a program developed by Gilbert Associates, Inc. for computing pipe stress at a shear lug attachment for Class 1, Class 2 and Class 3 piping. The program is based on the Welding Research Council Bulletin No. 198, entitled "Secondary Stress Indices for Integral Structural Attachments to Straight Pipe" and on ASME Code Case 1745. The program was verified by published results as well as comparison with sample calculations.

j.   GCWHAM (M090)

GCWHAM is a computer program, which was modified by Gilbert Associates, Inc. from WHAM-6 (Argonne Computer Code Abstract 278), and is applicable to calculation of pressure, velocity, and force transients in fluid-filled piping networks. GCWHAM can be applied to multi-loop, complex piping networks consisting of a number of dead ends, elbows, orifices, multiple branch tees, changes of flow passage cross sections, check valves, pumps, pressurizers or tanks, and exit valves. The program also considers hydraulic losses. Transients can be initiated by fast closure of one or more exit valves, by a prescribed change in tank pressure, or by pump power failure. The program was verified by the following two methods: (1) comparison of results with hand calculations, and (2) comparison of results obtained from the verified program, WAVENET.

k.   ENVSPEC (S125)

The ENVSPEC program is a Fortran language computer program developed by Gilbert Associates, Inc. It is used to generate an enveloping spectrum for a given set of response spectra.

Verification was accomplished by executing S125 test runs and comparing the resulting enveloped spectra with hand calculations and plots.

l.   GCMØ95

GCMØ95 is a Fortran language computer program developed by Gilbert Associates, Inc. It is used to evaluate ASME Code Class 1 equations and perform Class 1 fatigue analysis.

Verification was accomplished by comparing GCMØ95 calculated solutions with results from the certified program TPIPE.

m.   NB3200 (M107)

M107 is a Fortran language computer program developed by Gilbert Associates. It is used to perform detailed ASME Code Section NB-3200 stress evaluations of specialized piping components utilizing finite element analyses.

Verification was accomplished by executing M107 test runs and comparing the results against hand calculations.

n.   PRYTEN (S121)

PRYTEN is a Gilbert Associates, Inc. program developed for computing the bolt pullout force for 2 and 4 bolt base plates. The program performs a flexible plate analysis which determines prying loads on the base plate bolts in addition to direct tension loads. The program was verified by comparing results against hand calculations.

o.   FORTPRY (CS090)

FORTPRY is a Gilbert Associates, Inc. program developed for
computing the bolt pullout force and the base plate stress for 2,
4, 6, and 8 bolt base plates. The program performs a flexible
plate analysis which determines direct tension forces and prying
forces on the base plate bolts. The program was verified by
comparing results against hand calculations.

p.   RELAP5

RELAP5 is an advanced one dimensional thermal hydraulic analysis
code developed for the U.S. Nuclear Regulatory Commission by EG&G
Idaho, Inc. The program predicts time dependent thermal-fluid
conditions based on a nonhomogeneous, nonequilibrium hydrodynamic
model. The program generates forcing functions in piping segments
for such postulated events as safety/relief valve discharge.
Verification of the program was accomplished by (1) comparison of
results against published literature, and (2) comparison with
actual test results published in the Electric Power Research
Institute's Safety and Relief Valve Test Report.

q.   LOADCOMB (M093)

The M093 program was developed by Gilbert Associates, Inc. This
program calculates the combined pipe support design load for
various plant conditions in accordance with the support design load
combinations provided in <Table 3.9-21>. The program has the
option of developing stress intensification factors for pipe
support anchor plates under concentrated, uniform axisymmetric and
non-axisymmetric bending loads.

Verification was accomplished by comparing M093 results against
hand calculations.

r. REPORT (M109)

The M109 program was developed by Gilbert Associates, Inc. The program extracts selected TPIPE program results and tabulates in a format to be used in design and analysis. M109 lists equations for each load condition, support loads and displacements for each load condition, combined support loads for each load combination, and screens results to provide maximum design loads. The program will also extract necessary load and stress data from the TPIPE program to be used for integral support attachment design, flange qualification, equipment nozzle qualification, and branch line qualification.

Verification was accomplished by comparing the results of M109 test runs against hand calculations.

s. WAVENET

The WAVENET computer program was developed by R. T. Bradshaw, Inc. for the solution of wave transmission (i.e., fluid vibration) problems in complex networks of pipes or ducts. The program is applicable to a variety of applications including: pulsation in gas or liquid transmission pipeline systems, transient pulses induced by rapid changes in local flow conditions (e.g., rupture and valve closure), and excitation of piping by elastic (acoustic) waves induced by turbulent flows.

In addition to detailed information on local pressures and velocities throughout the system, the program develops detailed information on pressure-induced and momentum-induced structural forces at junctions, bends, valves, and orifices.

Verification was accomplished by comparing the results of sample problems with hand calculations.

t. Simplified Dynamic Analysis (M051)

The M051 program was developed by Gilbert Associates, Inc. This
program treats piping spans between rigid supports and/or seismic
restraints as independently supported beams. The exact solution of
the inertial response of the first five modes of a single span,
simply supported pipe is computed using floor response spectra as
the forcing function. One or two mutually perpendicular response
spectra can be used.

In general, the program can be used to compute the first piping
natural period and the natural frequency of the piping model as
well as the ratio between the first natural period and the
allowable natural period (percentage of the building predominant
peak), the inertial seismic reaction on a support, the deadweight
reaction on vertical supports, maximum seismic inertial stresses,
and ratio with the allowable seismic stress. In addition, the
program will compute the maximum mid-span deflection versus the
allowable deflection and determine reduced piping spans to account
for concentrated weights.

Results from the M051 program are used to develop a simplified
seismic support spacing criterion.

Program verification was accomplished by comparing M051 results
against hand calculations.

u. CRD Piping Analysis Program/IMPULSE-I

IMPULSE-I is an ECHO Energy Consultants, Inc. proprietary computer
code which evaluates fluid transients in piping networks. IMPULSE
includes extensively verified capabilities for simulation of BWR
control rod drive (CRD) systems.

IMPULSE-I employs the method-of-characteristics technique with constant time intervals coupled with a large set of mechanical equipment models. The program can completely describe the transient response of a wide variety of mechanical equipment models and boundary conditions such as pumps, valves, surge tanks, hydraulic control unit (HCU), and CRD piston systems.

IMPULSE-I is written in the Fortran V language and is compiled and available on the CYBER 175 and VAX-11/750. Application of the program is limited to piping networks initially containing water or gases with mach number less than 0.6. The wide variety of mechanical equipment components modeled within the program (i.e., valves, pumps, hydraulic control units) allows application of the program to any general piping system.

IMPULSE-I was developed, documented, and verified according to the criteria and guidelines of ANSI N413-1974 and the ECHO QA Program. The verification uses a combination of published experimental data and had calculations to test its capability. Twenty test runs provide the basic comparisons of IMPULSE-I with a wide series of experimental data.

v. EDASP

Equipment Dynamic Analysis Software Package is a microcomputer-based software package developed by Stevenson and Associates. This software package is used to develop dynamic response analytical models of instrument racks based on in situ testing. The in situ testing and response analysis procedures are performed according to recommendations set forth in IEEE 344-1975 and <NUREG/CR-3875>. The dynamic excitation is defined by the enveloped floor response spectra for the appropriate building elevation.

The responses at various points on the instrument rack are recorded during the test and the EDASP software package is used to develop an analytical model of the rack. EDASP is then used to develop new response curves for the racks and instruments when the racks are physically altered by changing instruments and their locations on the rack.

EDASP was verified by Stevenson and Associates. The basic method of verification was to run several models and compare the results with those from an established program.

w.   PENG LOCALS

PENG LOCALS is a program written by Peng Engineering to perform the stress calculations as presented in the Welding Research Council (WRC) Bulletin No. 107, entitled "Local Stresses in Spherical and Cylindrical Shells due to External Loadings." The program contains a series of non-dimensional curves that are used to obtain the stresses at four locations around the attachment on the interior and exterior surfaces of the shell at those locations. The program was verified by comparing the results of sample problems with those of the WERCO Program.

x.   TORQUE

Site Engineering Department (SED) developed the program "TORQUE" to be used on an IBM-PC (min) or equivalent. "TORQUE" determines the required "TORQUE" to develop a specified bolt "preload", with three different coefficients of friction, which are either default values or user input. "TORQUE" also determines the bolt load in a bolt when a certain "TORQUE" value is applied. Once again, this is accomplished for three different coefficients of friction. The program was verified by comparing results against hand calculations.

y.   PIPESTRESS

PIPESTRESS is a computer program developed by SST Systems, Inc.
which performs linear elastic analysis of three dimensional piping
systems. PIPESTRESS has static and dynamic analysis capabilities
including detailed uniform and multi-level response spectrum
analysis, time history and fatigue calculations and multiple load
cases and load combinations. In addition to these capabilities,
PIPESTRESS can post-process pipe member end forces through ASME
Section III, Class 1, 2 or 3 stress evaluation equations, and
provide support load data for pipe attachment evaluation.
PIPESTRESS is the main processor of PS+CAEPIPE program.

This program was verified by comparison to benchmark problems.

z.   PS+CAEPIPE

PS+CAEPIPE is a group of interrelated computer programs developed
by SST Systems, Inc. for performing linear elastic analysis of
piping systems subject to a variety of loading conditions. This
program has the same main processor as PIPESTRESS (described in
Item "y" above) and, therefore, has the same analysis capabilities.
Software modules for PS+CAEPIPE are pre-processor, batch input
file, PIPESTRESS (main processor), report generation and post
processor.

The program was verified by comparison to benchmark problems.

aa.   CONVER

PC Computer Program "CONVER" converts floor response spectra curves
in a digitized format (compatible with existing T-PIPE format) to a
digitized format for input into piping analysis program PS+CAEPIPE.

CONVER can also incorporate the Response Spectra Damping values of (ASME III) Code Case N-411-1. CONVER has been verified by hand calculations. (Note: EDASP is utilized to generate 2% and 5% "Damped" Response Spectra curves for input into CONVER's N-411-1 option).

bb. EXTRSP(S153)

Program EXTRSP was developed by Gilbert Associates, Inc. and was verified in accordance with <10 CFR 50, Appendix B> and <10 CFR 21>. Program EXTRSP accesses a specified set of digitized response spectra for use in enveloping response spectrum records. The resulting data will be used for input into analysis programs TPIPE and PIPESTRESS as discussed in <Section 3.9.1.2.6.b> and <Section 3.9.1.2.6.y>.

cc. PC-CRACK

Program PC-CRACK was developed by Structural Integrity, Inc. and was verified in accordance with <10 CFR 50, Appendix B>, ANSI/IEEE ANS-7-4.3.2-1982, and Structural Integrity Associates, Inc. Quality Assurance Manual. PC-CRACK analyzes flaws using fracture mechanics methods on a wide variety of structural components and materials.

dd. AutoPIPE

AutoPIPE is a PC based computer program developed by Engineering Design Automation which performs linear elastic analysis of three dimensional piping systems subject to a variety of loading conditions. AutoPIPE has static and dynamic analysis capabilities including modal, response spectrum, harmonic, and force spectrum analyses.

The program was verified by comparison to benchmark problems.

ee. POWER-ACM

Program POWER-ACM, was developed by SenEnTec and was verified in
accordance with the requirements of <10 CFR 50, Appendix B>.
Evidence of the verification of input, output and methodology is
documented in SenEnTec's permanent verification and validation
documentation. The SenEnTec Acoustic Chugging Methodology (SACM)
is the basis for the program. POWER-ACM evaluates the solution for
the acoustic wave equation in circular (annular and cylindrical)
suppression pool geometry. It was specifically designed to be
capable of performing all calculations necessary to determine the
time-dependent pressure field in either cylindrical, annular, or
segmented annular geometry due to unstable steam condensation.

ff. HEATING5

HEATING5 was developed by W. D. Turner, D. C. Elrod and
I. I. Siman-Tov of the Computer Science Division, Union Carbide
Nuclear Division for DOE. HEATING5 was verified in accordance with
<10 CFR 50, Appendix B> and <10 CFR 21>. The Code is designed to
solve steady-state and/or transient heat conduction problems in
one-, two-, or three-dimensional Cartesian or cylindrical
coordinates or one-dimensional spherical coordinates.

3.9.1.2.7   Valves

The Seismic Category I valves that utilize computer analysis are
designed and fabricated in accordance with ASME Section III.

3.9.1.2.8   Class 1 Containment Penetrations

The computer programs used in the analysis of the ASME Code Class 1
penetrations are identified and their use summarized in the following
paragraphs. These penetrations were originally designed and analyzed by

Nutech Corporation. The final analysis was modified by Gilbert
Associates to account for changes to design loads.

a.   Primary Membrane and Primary Bending Stress Program (PRS3)

Computer program PRS3 is a program to calculate primary membrane
and primary bending stresses due to pressure and mechanical loads.
The program was written by the Nutech Corporation and provides
calculations and checks against allowables established by the ASME
Boiler and Pressure Vessel Code for primary stress (P_m-P_L, and P_L-P_b
stresses). The primary stress intensities are calculated in
accordance with Paragraph NB-3215 of the ASME Code and compared
with input values of S_m or 1.5 S_m. Values of primary stress
intensity greater than S_m or 1.5 S_m are flagged by PRS3.

b.   Heat Transfer Analysis Program (HEATER)

The basis for HEATER is a program developed by E. L. Wilson
(Reference 4) to perform transient temperature analysis of plane
and axisymmetric solids. The program uses a finite element
definition of the solid, and thermal properties are defined for
each element.

In its original form, the program provided temperature-time
histories of the solid due to a step change in a step boundary
temperature. Nutech has modified the program to calculate element
temperatures due to an arbitrary fluid temperature change as a
function of time. Heat transfer from the fluid across the solid
boundary is by convection using an input value of convective heat
transfer coefficient. The time instant producing the most severe
gradient through the element for each thermal transient is used for
stress analysis.

c.   Stress Analysis Program (COSTAR)

The computer program COSTAR, developed and verified by Nutech, is a
finite element program for the analysis of axisymmetric solids of
revolution subject to axial, shear, pressure, moment, and torsional
loadings. The program calculates stresses, displacements and
principal stresses due to any arbitrary loads expanded into Fourier
components, including thermal and nodal loads. The program uses
isotropic materials. A post-processor program to COSTAR performs
stress and fatigue calculations required by the ASME Code.

d.   Piping Stress Analysis Program (PISTAR)

The computer program PISTAR, developed and verified by Nutech, is a
finite element program for the analysis of piping systems subject
to various loadings. The program calculates stresses, reactions
and deflections for the piping system for any given loading, and
also performs stress evaluation of the piping system required by
the ASME Code.

3.9.1.2.9   Class 2 Containment Penetrations (M097)

The M097 program takes penetration loads and unit load-stresses obtained
from ANSYS (3.9.1.2.6e) finite element analysis of penetration
assemblies and then computes various categories of stresses in
accordance with ASME Section III, NE-3200. The computed results are
compared with the code allowable stress limits for all stress categories
and all service conditions. The program verification was accomplished
by comparing program results with hand calculations.

3.9.1.3   Experimental Stress Analysis

The following sections list those NSSS components upon which
experimental stress analysis was used in conjunction with analytical
methods.

3.9.1.3.1   Testing of Piping Components

The following components have been tested to verify their design
adequacy:

a.   Piping snubbers

b.   Pipe whip restraints

Descriptions of the snubbers and whip restraint tests are contained in
<Section 3.9.3.4> and <Section 3.6>, respectively.

3.9.1.3.2   Orificed Fuel Support, Vertical and Horizontal Load Tests

The BWR 6 Orificed Fuel Support (OFS) is designed under the provisions
of the ASME Code, Section III, Subsection NG, as a core support
structure and, therefore, complies with NA-3352.1. In order to meet
this requirement, an analysis was performed using the finite element
method. However, the complexity of the OFS design, as well as the
nonlinear behavior of the OFS during analysis, preempted the use of
experimental stress analysis. Accordingly, a series of horizontal and
vertical load tests were performed in order to conform to the
requirements of the Code and the design specification. The results of
these tests indicate that the resultant stresses due to static and
dynamic loading of the OFS are below the stress limit allowables,
including a 0.65 quality factor. The allowable stress limits were
obtained 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.

3.9.1.3.3   Control Rod Drive

Experimental data was used in refining the SCRAM 1 Code. The output of
SCRAM 1 was used in the dynamic analysis of both code and noncode parts.
Pressures used in the analysis were also determined during actual
testing of prototype control rod drives.

3.9.1.4   <u>Evaluation of Emergency and Faulted Conditions</u>

The following test results show functional capability of BOP piping
systems is assured at ASME Service Level C stress limits.

a.   ASME Paper 78-PVP-83 – "Evaluation of the Functional Capability of
      ASME Section III, Class 1, 2 and 3 Piping Components."

b.   ASME 74-NE-1 – "Plastic Deformation of Piping Due to Pipe - Whip
      Loading."

c.   <NUREG/CR-0261> – "Evaluation of the Plastic Characteristics of
      Piping Products in Relation to ASME Code Criteria."

This work reviewed a significant amount of test data and found that none
could be found in which large enough displacements were applied to
produce significant reductions in flow area, e.g., 50% reduction of flow
area. A local reduction in straight pipe area of 50% does not mean that
the functional capability of the piping system has been reduced by 50%.
Such a local flow restriction might give only a 1% loss in flow
capacity, for a given pressure drop, or 2% higher pressure drop for a
given flow rate.

The applicant has demonstrated the functional capability of all piping
components in essential ASME Class 1, 2 and 3 piping systems designed
using level C service stress limits. This included use of the criteria
in the Interim Technical Position "Functional Capability of Passive

Piping Components," Mechanical Engineering Branch, Division of Systems Safety. Where additional criteria are required to evaluate functional capability, applicable criteria in NEDO-21985, dated September 1978, were used.

Seismic Category I equipment was evaluated for the faulted loading conditions. In all cases, calculated stresses were within the allowable limits. The following paragraphs in <Section 3.9.1.4> 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 <Section 3.9.2.5>, <Section 3.9.3>, <Section 3.9.4>, <Section 3.9.5>, and <Table 3.9-3>.

Deformations under faulted conditions have been evaluated in critical areas and no cases are identified where design limits, such as clearance limits, are exceeded.

For reactor internals, analyses have been performed to ensure that the damaged non-seismically qualified internals do not impair the integrity and the intended functions of the seismically qualified internals.

All break exclusion region piping was evaluated as Seismic Category I for the safe shutdown earthquake.

3.9.1.4.1   Control Rod Drive System Components

a.   Control Rod Drives

     The major control rod drive components that are analyzed for
     faulted conditions are: ring flange, main flange and indicator
     tube. The maximum stresses for these components, for various plant
     operating conditions, is given in <Table 3.9-3u>.

b.   Hydraulic Control Unit

A discussion of the seismic considerations for the hydraulic control unit (HCU) is found in <Section 3.9.2.2.3.d>.

3.9.1.4.2   Reactor Internal Components

a.   Control Rod Guide Tube

The maximum calculated stress on the control rod guide tube occurs in the base during the faulted condition and is 21,763 psi. The "faulted" limit is the lesser of 2.4 S_m or 0.7 S_u at the design temperature per ASME Code, Section III, Table I-1.2 and F 1322-1 where S_u = 57,500 psi and S_m = 16,000 psi @ 575°F. The maximum stress summary for various plant operating conditions is given in <Table 3.9-3aa>.

b.   Incore Housing

The maximum calculated stress on the incore housing occurs at the outer surface of the vessel penetration during the faulted condition and is 21,026 psi. The "faulted" limit is the lesser of 2.4 S_m or 0.7 S_u at the design temperature per ASME Code, Section III, F1323.1 (b) where S_u = 80,000 psi and S_m = 23,300 psi at 575°F. The method of analysis and the maximum stress summary for various plant operating conditions is given in <Table 3.9-3ab>.

c.   Jet Pump

The elastic analysis for the jet pump faulted conditions shows that the maximum stress occurs at the elbow/thermal sleeve interface and is 55,258 psi. The maximum allowable for this condition per ASME Code, Section III, is 3.6 S_m or 60,840 psi. The method of analysis

and the maximum stress summary for various plant operating conditions is given in <Table 3.9-3w>.

d.   LPCI Coupling

The location of the highest primary stress ($P_m + P_b$) is at the strut to shroud attachment weld. The smallest margin at the weld is from the faulted condition of NL + ΔP_A + JR + AP/F + SSE, resulting from a large line break plus SSE. The calculated stress is 34,807 psi. The allowable stress is 35,658 psi, or 2.4 x 1.5 x 0.7 S_m. A weld quality factor of 0.7 is included as required by ASME Code, Section III, Table NG-33521-1 <Table 3.9-3y>.

e.   Orificed Fuel Support

Orificed fuel support is analyzed for the faulted condition. The method of analysis, testing and calculated stresses are given in <Section 3.9.1.3.2>, and summarized in <Table 3.9-3ac>.

f.   Control Rod Drive (CRD) Housing

The SSE is classified as a faulted condition; however, in the CRD housing analysis the SSE event has been treated as an emergency condition. The maximum stress on the CRD housing during the faulted condition is 14,727 psi. The maximum design stress limit for this event is 1.2 S_m = 20,000 psi, and the ultimate strength of the material is 57,500 psi. <Table 3.9-3v> shows the allowable and calculated stress values for the highly stressed areas of the control rod drive housing.

3.9.1.4.3   Reactor Pressure Vessel Assembly

The reactor pressure vessel assembly includes the reactor pressure
vessel, support skirt and shroud support.

For faulted conditions, the reactor vessel was evaluated by elastic
analysis. For the shroud support, an elastic analysis was performed,
and for compressive load, buckling was evaluated. The method of
analysis and maximum stress summary for various plant operating
conditions is given in <Table 3.9-3a>.

3.9.1.4.4   Core Support Structures

The core support structures are evaluated for the faulted condition
loads, and the basis for determining the faulted loads for seismic and
other dynamic events is given in <Section 3.7> and <Section 3.9.5>,
respectively. The method of analysis and the maximum stress summary for
various plant operating conditions is given in <Table 3.9.3a>,
<Table 3.9.3b>, <Table 3.9.3v>, <Table 3.9.3aa>, <Table 3.9.3ab>, and
<Table 3.9.3ac>.

3.9.1.4.5   Main Steam Isolation, Recirculation Gate and
                 Safety/Relief Valves

<Table 3.9-3g>, <Table 3.9-3h> and <Table 3.9-3j> provides a maximum
stress summary and the methods of analyses for the consideration of the
faulted condition of the safety/relief, main steam isolation and
recirculation gate valves, respectively.

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 is used to
evaluate components not defined in the ASME Code. The ASME Code

allowable stresses are applied to determine acceptability under applicable loading conditions, including faulted condition.

3.9.1.4.6   Recirculation System Flow Control Valve

The recirculation system flow control valve is analyzed for faulted conditions by the elastic analysis methods from the ASME Code, Section III. The method of analysis and the maximum stress summary for various plant operating conditions is given in <Table 3.9-3f>.

3.9.1.4.7   Main Steam and Recirculation Piping

For main steam and recirculation system piping, elastic analysis methods are used in 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 pertaining to the consideration of the faulted condition on the main steam and recirculation piping is in <Table 3.9-3d> and <Table 3.9-3e>.

3.9.1.4.8   Nuclear Steam Supply System Pumps, RHR Heat Exchanger and
                 RCIC Turbine

The recirculation, ECCS, RCIC, and SLC pumps; 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 are within the elastic limits. The analytical methods, maximum stress limits and allowable stresses are given under the respective equipment table in the 3.9-3 series.

3.9.1.4.9   Control Rod Drive Housing Supports

Examples of the calculated stresses and the allowable stress limits for
the faulted condition for the control rod drive housing supports are
shown in <Table 3.9-3ac>.

3.9.1.4.10   Fuel Storage and Refueling Equipment

Refueling and servicing equipment is classified as essential components
per the requirements of <10 CFR 50, Appendix A>. This equipment and
other equipment whose failure would degrade an essential component is
classified in <Section 9.1> as Seismic Category I. These components are
subjected to an elastic dynamic finite element analysis to generate
loadings. This analysis utilizes appropriate floor response spectra and
combines loads at frequencies up to 33 Hertz for seismic and up to a
higher cut-off frequency for the other hydrodynamic loads <Appendix 3A>
in three directions. Imposed stresses are generated and combined for
normal, upset and faulted conditions. Stresses are compared, depending
on the specific safety class of the equipment, to industrial codes,
ASME, ANSI or industrial standards, or AISC allowables.

Examples of the calculated stresses, and the allowable stress limits for
the faulted conditions for the fuel preparation machine, refueling
platform and inclined fuel transfer tube are given in <Table 3.9-3s>.

3.9.1.4.11   Fuel Channels

Structural analyses of the GE BWR fuel assembly, including channel
design bases, analytical methods and evaluation results, including those
applicable to the faulted conditions, are contained in <Reference 5> and
<Reference 6>. The fuel assembly design acceleration envelope,
including SSE and DBA, is given in <Table 3.9-3s>.

3.9.1.4.12   Balance of Plant Piping, Valves and Equipment

Elastic-plastic methods of analysis have not been used to evaluate the design of safety-related code or noncode balance-of-plant components. An exception to this may be the evaluation of hypothetical pipe rupture events requiring the evaluation of inelastic stresses and deformations as described in <Section 3.6.2>.

Excluding some potential pipe rupture events, no balance-of-plant valves or equipment have been analyzed to the faulted component stress limits given in the ASME Code, Section III. When the faulted plant condition has been analyzed, the ASME Section III allowable stress was either the upset or emergency limit. Determination of the allowable stress level was based on whether the component was active or nonactive, in accordance with <Regulatory Guide 1.48>, and whether the component needs to function during or after the incident.

For balance-of-plant piping, the faulted service limits (D) as given in ASME Section III are used to evaluate piping under postulated faulted plant conditions. <NUREG-0800>, July 1, 1981, is also used to determine the allowable stress levels. Functional check calculations are performed for piping required to function whenever Emergency or Faulted limits are used.

The piping and supports for the four main steam lines running through the auxiliary and turbine buildings are analyzed and designed for the OBE and SSE. The piping and supports for the two feedwater lines in the auxiliary building are analyzed and designed for the OBE and SSE. Some of the piping inside the turbine building (that nearest the piping in the auxiliary building) is analyzed and designed for the OBE and SSE. (A significant portion of this piping is not ASME.) The supports nearest the class break are designed for the OBE and SSE to ASME limits. The remaining supports further away from the class break are designed for the OBE and SSE but not necessarily to ASME limits.

Further discussion of plant condition versus component allowable stress is provided in <Section 3.9.2>.

3.9.1.4.13   Containment Penetrations

All ASME Code, Section III, Class 1 penetration components were evaluated by elastic analysis methods for faulted loading conditions. The faulted loading condition stress levels were limited to the ASME Code, Section III, Appendix F limits.

3.9.2   DYNAMIC TESTING AND ANALYSIS

3.9.2.1   Piping Tests and Analysis

Piping vibration, thermal expansion and dynamic effects testing were conducted during preoperational (pre-fuel load) or startup (post-fuel load) testing programs. The purpose of these tests was to ensure that piping vibrations were within acceptable limits and that the piping systems expand thermally in a manner consistent with the design intent. As used in the following subsections, the nuclear steam supply system (NSSS) piping consists primarily of main steam and recirculation piping. Other piping and instrumentation is considered Balance of Plant (BOP) piping.

3.9.2.1.1   NSSS Piping Vibration

3.9.2.1.1.1   Preoperational Vibration Testing of Recirculation
                Piping

The preoperational vibration test phase verified that operating vibrations in the recirculation piping were within acceptable limits. This phase of the test used visual observation to supplement remote measurements. If, during steady-state operation, visual observation indicated significant vibration, measurements were made with a handheld

vibrograph. Visual observations and manual and remote measurements were
made during the following steady-state conditions:

a.   Recirculation system minimum flow.

b.   Recirculation system at 50 percent of rated flow.

c.   Recirculation system at 75 percent of rated flow.

d.   Recirculation system at 100 percent of rated flow.

3.9.2.1.1.2   Preoperational Vibration Testing of Small Attached
                   Piping

During visual observation of each of the above test conditions (a)
through (d), special attention was given to small attached piping and
instrument connections to ensure they were not in resonance with the
recirculation pump motors or flow induced vibrations. If the operating
vibration acceptance criteria were not met, corrective action, such as
modification of supports, was undertaken. Small bore pipe and
instrumentation lines tested included but were not limited to the
following:

a.   Reactor pressure vessel level indicator instrumentation lines (used
     for monitoring both steam and water levels).

b.   Main steam instrumentation lines for monitoring main steam flow
     (used to actuate main steam isolation valves during high steam
     flow).

c.   Reactor core isolation cooling (RCIC) instrumentation lines on the
     RCIC steam line outside containment (used to monitor high steam
     flow and actuate isolation).

d.   Control rod drive lines inside containment (not normally pressurized but required for scram).

3.9.2.1.1.3   Startup Vibration Testing of Main Steam, Recirculation and RCIC Piping

This phase of the program verified that the main steam and recirculation piping vibration were within acceptable limits. Pipe vibration measurements were made on the recirculation lines, main steamlines and RCIC turbine inlet line. Because of limited access due to high radiation levels, no visual observation was performed during this phase of the test. Remote vibration measurements were made during the following steady-state conditions:

a.   Main steam flow at 20-30 percent of rated flow.

b.   Main steam flow at 45-55 percent of rated flow.

c.   Main steam flow at 70-80 percent of rated flow.

d.   Main steam flow at 98-100 percent of rated flow.

3.9.2.1.1.4   Operating Transient Loads on Main Steam and Recirculation Piping

The operating transient test phase verified that pipe stresses were within code limits. The amplitude of displacements and number of cycles per transient of the main steam and recirculation piping were measured and the displacements compared with acceptance criteria. The displacements were correlated with stresses to verify that the pipe stresses remained within ASME Code limits. Remote vibration measurements were taken during the following transients:

a.   Recirculation pump starts.

b.   Recirculation pump trip at maximum rated flow.

c.   Turbine stop valve or turbine control valve closure at rated power.

d.   Manual discharge of each SRV at normal operating pressure
      (≥920 psig) and at planned transient tests that result in SRV
      discharge.

3.9.2.1.2   NSSS Thermal Expansion Testing

For main steam and recirculation piping, preoperational and startup
thermal expansion testing programs were performed through the use of
visual observations of pipe support spring and snubber settings by
qualified individuals and remote potentiometer sensors to verify normal
thermal movement occurred in piping systems. This program confirmed the
following:

a.   The piping system during system heatup and cooldown is free to
      expand, contract and move without unplanned obstruction or
      restraint in the x, y and z directions.

b.   The piping system is working in a manner consistent with the
      assumptions of the NSSS stress analysis.

c.   There is adequate agreement between calculated and measured values
      of displacement.

d.   There is consistency and repeatability in thermal displacements
      during heatup and cooldown of the NSSS systems.

Limits of thermal expansion displacements were established prior to
start of testing; actual measured displacements were compared to
determine acceptability of the actual motion. If the measured
displacement did not vary from the acceptance values by more than the

specified tolerance, the piping system responded in a manner consistent with predictions and was therefore accepted. Two levels of acceptance limits were established to check the systems as explained in <Section 3.9.2.1.4>.

3.9.2.1.3   Dynamic Effects Testing of NSSS Piping

To add assurance that snubbers would adequately perform their intended function during plant operation, a dynamic testing program was performed as part of startup testing. The main purposes of this program were to ensure:

a.   Vibration levels on piping from the various dynamic loadings during transient and steady-state conditions are below the predetermined acceptable limits.

b.   Long term fatigue failure of piping will not occur due to underestimating the dynamic effects caused by cyclic loading during plant transient operations.

This dynamic testing accounted for the hydrodynamic and applicable operating transient loads. The maximum stresses developed in the piping by these loads were used as a basis for establishing criteria which will assure proper functioning of the snubbers.

The criteria for vibration displacements were based on an assumed linear relationship between displacements, snubber loads and magnitude of applied loads. Thus, the magnitude of limits of displacements, snubber loads and nozzle loads are all proportional. Maximum displacements (Level 1 limits) were established to prevent the maximum stress in the piping systems from exceeding the normal and upset primary stress limits and/or the maximum snubber load from exceeding the maximum load to which the snubber was tested.

Based on the above criteria, Level 1 displacement limits were established for all instrumented points in the piping system. These limits were compared with the field measured piping displacements as explained in the following section.

Related verifications of snubber integrity included:

a.   Sample production snubbers of each size (e.g., 10 kips, 20 kips and 50 kips) were qualified and tested for design and faulted loadings prior to shipment to field. Snubbers were tested prior to installation to allow free piping movements at low velocity.

b.   During plant startup, the snubbers were checked for proper settings and for any evidence of oil leak.

3.9.2.1.4   Test Evaluation and Acceptance Criteria for NSSS Piping

The piping response to test conditions was considered acceptable based on a review by the organization responsible for the stress report and a determination that the piping responded in a manner consistent with the predictions of the stress report, and/or that piping stresses were within code limits (ASME Code, Section III, NB-3600). Acceptable deflection and vibration limits were determined after the completion of piping systems stress analysis and provided in the startup test specifications. To insure test data integrity and test safety, criteria were established to facilitate assessment of the test while it was in progress. These criteria, designated Level 1 and 2, are described in the following paragraphs.

a.   Level 1 Criteria

     Level 1 established the maximum limits for the level of pipe motion which, if exceeded, made a test hold or termination mandatory.

If the Level 1 limit was exceeded, the plant was placed in a
satisfactory hold condition, and the responsible piping design
engineer was advised. Following resolution, applicable tests were
repeated to verify that the requirements of the Level 1 limits were
satisfied.

b.   Level 2 Criteria

Level 2 specified the level of pipe motion which, if exceeded,
required that the responsible piping design engineer be advised.
If the Level 2 limit was 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
was reached by all appropriate parties, including the responsible
piping design engineer. Depending upon the nature of such
resolution, the applicable tests were repeated.

For startup vibration testing of NSSS piping, the following
acceptance criteria were generally applied:

1.   For steady-state vibration, the piping peak stress (zero to
     peak) due to vibration only (neglecting pressure) did not
     exceed 1/2 of the endurance limit as defined by the ASME Code
     at 10^6 cycles.

2.   For operating transient vibration the piping bending stress
     (zero to peak) due to operating transient only did not exceed
     1.2 S_m or pipe support loads did not exceed the Service
     Level D ratings for Level 1 criteria. The 1.2 S_m limit
     insures that the total primary stress, including pressure and
     dead weight, will not exceed 1.8 S_m, the new Code Service
     Level B limit. Level 2 criteria are based on pipe stresses
     and support loads not to exceed design basis predictions.

Design basis criteria require that operating transient stresses and loads are not to exceed any of the Service Level B limits including primary stress limits, fatigue usage factor limits, and allowable loads on snubbers.

3.9.2.1.5   Corrective Actions for NSSS Piping Tests

During the course of the tests, the remote measurements were regularly checked to determine compliance with Level 1 criteria. If trends indicated that Level 1 criteria may be violated, the measurements were monitored at more frequent intervals. The test was suspended or terminated as soon as Level 1 criteria were violated. As soon as possible after the test hold or termination, the following corrective actions was taken:

a.   Installation Inspection

      A walkdown of the piping and suspension was made to identify any obstruction or improperly operating suspension components. Snubbers and spring hangers were not to be at their travel limits. If vibration exceeded criteria, the source of the excitation was identified to determine whether it was related to equipment failure. Action was taken to correct any discrepancies before repeating the test.

b.   Instrument Inspection

      Instrument installation and calibration was checked and any discrepancies corrected. Additional instrumentation was added, if necessary.

c.   Repeat Test

If actions (a) and (b) above identified discrepancies that could
account for failure to meet Level 1 criteria, the test was
repeated.

d.   Resolution of Findings

If the Level 1 criteria was violated on the repeat test, or no
relevant discrepancies were identified in a and b, the organization
responsible for the stress report reviewed the test results and
criteria to determine whether the test could be safely continued.

If the test measurements indicated failure to meet Level 2 criteria, the
following corrective actions were taken after completion of the test:

a.   Installation Inspection

A walkdown of the piping and suspension was made to identify any
obstruction or improperly operating suspension components. If
vibration exceeded limits, the source of the vibration was
identified. Action, such as suspension adjustment, was taken to
correct any discrepancies.

b.   Instrument Inspection

The instrument installation and calibration was checked and any
discrepancies corrected.

c.   Repeat Test

If (a) and (b) above identified a malfunction or discrepancy that
could account for failure to meet Level 2 criteria and appropriate
corrective action had been taken, the test might not be repeated.

d. Documentation of Discrepancies

If the test was not repeated, discrepancies found were documented
in the test evaluation report and correlated with the test
condition. The test was not considered complete until test results
were reconciled with the acceptance criteria.

3.9.2.1.6   Measurement Locations for NSSS Piping

Remote vibration measurements were made in the three orthogonal
directions near the first downstream safety/relief valve on each steam
line, and in the three orthogonal directions on the piping between the
recirculation pump discharge and the first downstream valve. During
preoperational testing prior to fuel load, visual inspection of the
piping was made, and any visible vibration measured with a handheld
instrument.

For each of the selected remote measurement locations, Level 1 and 2
vibration and strain limits were prescribed in the startup test
specification. Level 2 limits were based on the results of the stress
report adjusted for operating mode and instrument accuracy; Level 1
limits were based on maximum allowable code stress limits.

3.9.2.1.7   Vibration Testing of Balance of Plant (BOP)
             Safety-Related and Seismically Qualified Piping

Safety-related piping systems designated as Class 1, 2 or 3 are designed
in accordance with the ASME Code, Section III. Each system is designed
to withstand dynamic loadings from operational transient conditions that
will be encountered during expected service as required by
Subsections NB-3622, NC-3622 and ND-3622 of the ASME Code. A
preoperational vibration and dynamic effects test of safety-related
piping was performed during startup functional testing to determine that

piping vibrations were acceptable. Vibration measurements were obtained for loading conditions such as pump trips and valve closures.

In addition to safety-related piping, certain nonsafety-related piping designed in accordance with ANSI B31.1 (Reference 7) was also vibration tested. This nonsafety-related piping includes piping which is seismically supported to preclude gross failure and subsequent damage to essential components.

3.9.2.1.8   Thermal Expansion Testing of Balance of Plant Piping

Portions of the Balance of Plant piping which were accessible were visually inspected under the startup test program by qualified individuals for proper pipe support spring and snubber settings, and piping thermal clearances. Portions of inaccessible piping were remotely monitored with potentiometer signals.

Limits to thermal expansion were established prior to start of testing and actual measured displacements were compared to determine acceptability of the actual motion. If the measured displacement did not vary from the acceptance values by more than the specified tolerance, the piping system responded in a manner consistent with predictions and was therefore accepted. Two acceptance limits were established to check the systems as explained in <Section 3.9.2.1.11>.

3.9.2.1.9   Plant Conditions for Which BOP Piping was Tested

Piping was tested as the plant systems operated in various flow modes. Typical plant conditions included but were not limited to the following:

a.   Main steam line piping strain and vibration during MSIV isolation.

b.   Main steam line vibration due to turbine stop valve or turbine
     control valve closure.

c.   Feedwater system water hammer vibration during feedwater pump trip.

d.   Miscellaneous safety class piping system vibration during partial and full load system operation.

3.9.2.1.10   Measurement Locations for Balance of Plant Piping

Measurement locations for each system to be tested were described in the detailed system test programs.

3.9.2.1.11   Startup Test Evaluation and Acceptance Criteria for Balance of Plant Piping

The response of piping to test conditions was considered acceptable based on a review by the organization responsible for the stress report and a determination that the piping responded in a manner consistent with the predictions of the stress report and/or that piping stresses were within code limits. To ensure test data integrity and test safety, criteria were established to facilitate assessment of the test while testing was in progress. These criteria, designated as Level 1 and Level 2, follow:

a.   Level 1 Criteria

     Level 1 criteria established bounds on movement that, if exceeded, make a test hold or termination mandatory. The limits on movement are based on maximum allowable code stress limits.

b.   Level 2 Criteria

     Conformance with Level 2 criteria demonstrated that the piping responded in a manner consistent with stress report predictions. Failure to satisfy Level 2 criteria did not mean that the piping response was unsatisfactory; it meant that the system did not

respond in accordance with theoretical predictions and further
analyses, based upon test results, were necessary. Level 2
criteria separated test results that were consistent with
predictions and required no analytical review from test results
that had to be evaluated.

For steady-state vibration, piping peak stresses (zero to peak) due
to vibration only (neglecting pressure) did not exceed 1/2 of the
endurance limit as defined by the ASME Code at 10^6 cycles.

3.9.2.1.12   Corrective Actions for BOP Piping Tests

During the course of the tests, remote measurements were regularly
checked to determine compliance with Level 1 criteria. If trends
indicated that Level 1 criteria may be violated, the measurements were
checked at more frequent intervals. The test was subjected to a hold or
terminated as soon as Level 1 criteria were violated. As soon as
possible after establishment of a test hold or termination of the test,
the following corrective actions were taken:

a.   Installation Inspection

     A walkdown of the piping and suspension was performed to identify
     any obstruction or improperly operating suspension components. If
     vibration exceeded the criteria, the source of the excitation was
     identified to determine whether it was related to equipment
     failure. Action was taken to correct any discrepancies prior to
     repeating the test.

b.   Instrument Inspection

     Instrument installation and calibration was checked and
     discrepancies were corrected. Additional instrumentation was added
     if necessary.

c.   Repeat Test

If items (a) and (b) above identified discrepancies that could
account for failure to satisfy Level 1 criteria, the test was
repeated.

d.   Resolution of Findings

If Level 1 criteria were violated during the repeat test or no
relevant discrepancies were identified, the organization
responsible for the stress report reviewed the test results and
criteria to determine whether testing could be continued safely.

If test measurements indicated failure to satisfy Level 2 criteria, the
following corrective actions were taken after completion of the test:

a.   Installation Inspection

A walkdown of the piping and suspension was performed to identify
any obstruction or improperly operating suspension components. If
vibration exceeded limits, the source of the vibration was
identified and action, such as suspension adjustment, was taken to
correct the discrepancies.

b.   Instrumentation Inspection

The instrumentation installation and calibration were checked and
discrepancies corrected.

c.   Repeat Test

If items a and b above identified a malfunction or discrepancy that
could account for failure to comply with Level 2 criteria and

appropriate corrective action had been taken, the test may not have
been repeated.

d.   Documentation of Discrepancies

      If the test was not repeated, the discrepancies discovered as a
      result of items (a) and (b) were documented in the test evaluation
      report and correlated with the test condition. The test was not
      considered complete until test results were reconciled with the
      acceptance criteria.

3.9.2.2   Seismic and Hydrodynamic Load Qualification of
              Safety-Related Mechanical Equipment

This section describes the criteria for dynamic (vibrational) load
qualification of safety-related mechanical equipment and the
qualification tests and/or analyses applicable to this plant on a
component-by-component basis. In some cases, a module or assembly
consisting of mechanical and electrical equipment is qualified as a
unit, e.g., ECCS pumps. These modules are generally discussed in this
section rather than in <Section 3.10>. Electrical supporting equipment,
such as control consoles, cabinets and panels, are discussed in
<Section 3.10.3.2>.

3.9.2.2.1   Qualification Method

The ability of equipment to perform its safety-related function during
and after the application of dynamic (vibrational) loads is demonstrated
by tests and/or analyses. Selection of qualification method is
discussed in <Section 3.10.1.2>.

3.9.2.2.2   Analytical Qualification

Analyses have been performed for all supports of mechanical equipment to assure their structural capability to withstand seismic excitation. The input motion to the equipment was characterized by floor response spectra. The ASME Code-allowable stress levels in Subsection NF, Component Support Structures, have been satisfied for the load combinations described in <Table 3.9-3> and inservice loadings defined in <Section 3.9.3.1.2>. A comparison of calculated stresses versus code allowable stresses for component supports is presented in <Table 3.9-3>, <Table 3.9-9>, <Table 3.9-10>, <Table 3.9-12>, <Table 3.9-13>, <Table 3.9-14>, <Table 3.9-24>, <Table 3.9-25>, <Table 3.9-26>, <Table 3.9-27>, <Table 3.9-28>, <Table 3.9-29>, and <Table 3.9-36>.

3.9.2.2.3   Seismic and Hydrodynamic Load Qualification of Specific
             Mechanical Components

The following sections discuss the testing and analytical qualification of safety-related mechanical equipment. Specific input parameters used for this analysis are shown in <Table 3.9-4>. Specific analytical results are presented here and in <Table 3.9-3>.

For the following equipment, Hydrodynamic loads have no impact on this equipment because the Auxiliary Building and the Containment Building are not on the same basemat:

•   ECCS Pump and Motor Assembly

•   RCIC Pump Assembly

•   RCIC Turbine Assembly

•   Standby Liquid Control (SLC) Pump & Motor Assembly

•   Residual Heat Removal (RHR) Heat Exchangers

a.   Jet Pumps

A dynamic analysis of the jet pumps has been performed and the
stresses resulting from the analysis are below the code allowables.

b.   Control Rod Drive (CRD) and CRD Housing

The dynamic qualification of the CRD housing (with enclosed CRD) is
done analytically; analytical results established the structural
integrity of these components. Preliminary dynamic tests are
conducted to verify the operability of the control rod drive during
a dynamic event. A simulated test, imposing a static bow in the
fuel channels, has been performed and the CRD functioned
satisfactorily.

c.   Core Support (Fuel Support and CR 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 code allowable.

d.   Hydraulic Control Unit (HCU)

The qualification of the Hydraulic Control Unit (HCU) was
determined by test. The Perry plant-unique loads obtained from
analysis were assessed at the HCU support interface locations
(floor, middle and upper attachments). The required response
spectra (RRS), representing the worst load combination for the
upset and faulted cases, were compared with the test response
spectra (TRS).

Based on the enveloping of the RRS by the TRS, the qualification of
the HCU has been concluded to be adequate for Perry application.

e. Fuel Assembly (Including Channels)

GE BWR fuel channel design bases, analytical methods and evaluation
results, including seismic and hydrodynamic load considerations,
are contained in (Reference 5) and (Reference 6). The resultant
combined acceleration profiles including fuel lift for all
normal/upset and faulted events are less than the respective design
basis acceleration profiles.

f. Recirculation Pump and Motor Assembly

The recirculation pump, including its appurtenances and supports,
individually and as an assembly, is designed to withstand seismic
forces including hydrodynamic loads:

1. The flooded pump motor assembly is analyzed as a free body
    supported by constant support hangers from the brackets on the
    motor mounting member, with hydraulic snubbers attached to
    brackets on the pump case and the top of the motor frame.

2. Primary stresses due to horizontal and vertical seismic forces
    including hydrodynamic load are considered. Horizontal and
    vertical seismic forces including hydrodynamic load are
    applied to mass centers and equilibrium reactions are
    determined for motor and pump brackets.

3. 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 motor and flange bolting, and pump case.

4. The maximum combined tensile stress in the cover bolting was
    calculated, including tensile stress from design pressure.

5. The brackets on the pump case were designed to withstand loads resulting from the building seismic response.

6. Analyses have been completed which demonstrate that the natural frequency of the assembled pump and motor structure is greater than 33 Hertz.

g. ECCS Pump and Motor Assembly

A prototype ECCS pump motor has been seismically qualified via a combination of static analysis and dynamic testing. The complete motor assembly is being qualified via dynamic testing, in accordance with IEEE 344-1975 and IEEE 323-1974. The qualification test program included demonstration of startup and shutdown capabilities, as well as no-load operability during seismic loading conditions.

For static analysis, the seismic forces on each component or assembly are obtained by concentrating its mass at the center of mass of the component or assembly, and multiplying by the seismic acceleration (earthquake coefficient). The magnitude of the earthquake coefficients are 0.7g vertical and 3.0g horizontal, except below mounting flange, which is 1.0g horizontal.

The qualification of the pump motor assemblies as units while operating under SSE conditions was provided in the form of a static earthquake-acceleration analysis. Since the natural frequency is above 33 Hertz, the units were considered to be supported as designed and maximum specified vertical and horizontal accelerations were applied simultaneously and constantly, in the worst case combination. The results of the analysis indicate the pump is capable of sustaining the above loadings without overstressing the pump components.

h.   RCIC Pump Assembly

The RCIC pump construction is a barrel type on a large
cross-sectioned pedestal. Qualification by analysis was performed.
The seismic design analysis is based on 1.5g horizontal and
1.5g vertical accelerations. Results are obtained by using
acceleration forces acting simultaneously in three directions; one
vertical and two horizontal, and calculated using the square root
of the sum of the squares method. The pump mass, support system
and accessory piping have been shown, by analysis, to have a
natural frequency greater than 33 Hertz.

The RCIC pump assembly has been analytically qualified by static
analysis for seismic loading as well as the design operating loads
of pressure, temperature and external piping loads. The results of
this analysis confirm that the stresses are substantially less than
90 percent of allowable.

i.   RCIC Turbine Assembly

The RCIC turbine has been qualified via a combination of static
analysis and dynamic testing in accordance with the guidelines of
IEEE-344-1975. The turbine assembly consists of rigid masses,
wherein static analysis has been utilized, interconnected with
control levers and electronic control systems, necessitating final
qualification via dynamic testing. Static loading analysis has
been employed to verify the structural integrity of the turbine
assembly and the adequacy of bolting under operating and seismic
loading conditions. The qualification test program on the complete
turbine assembly included demonstration of startup and shutdown
capabilities, as well as no-load operability during seismic loading
conditions <Figure 3.9-1> and <Figure 3.9-2>.

j.   Standby Liquid Control (SLC) Pump and Motor Assembly

The SLC positive displacement pump and motor, mounted on a common
baseplate, have been qualified by static analysis.

The seismic design analysis is based on 1.75g horizontal and
1.75g vertical accelerations. Results are obtained by using
acceleration forces acting simultaneously in three directions, one
vertical and the other two horizontal, and calculated using the
square root of the sum of the squares method. The pump/motor/base
assembly has been shown by static analysis to have a natural
frequency greater than 33 Hertz. The SLC pump and motor assembly
has been analytically qualified by static analysis for seismic
loading as well as the design operating loads of pressure,
temperature and external piping loads. The results of this
analysis confirm that the stresses are substantially less than
90 percent of allowable.

k.   Residual Heat Removal (RHR) Heat Exchangers

A dynamic analysis has been performed to verify that the RHR heat
exchanger will withstand seismic loadings <Figure 3.9-3> and
<Figure 3.9-4>. Seismic testing is an impractical method to verify
the seismic adequacy of passive equipment when predictable seismic
loads can be determined by analysis.

l.   Standby Liquid Control Storage Tank

The standby liquid control storage tank is a cylindrical tank,
9 feet in diameter and 12 feet high, bolted to a steel pedestal.
The standby liquid control storage tank has been qualified for
seismic and hydrodynamic loads by analysis for:

1.   Stresses in the tank bearing plate.

2.   Bolt stresses.

3.   Sloshing loads imposed at natural frequency of
      sloshing = 0.58 Hertz.

4.   Minimum wall thickness.

5.   Buckling.

The results of the analysis <Table 3.9-3m> confirm that stresses at
all investigated locations are less than their corresponding
allowables.

m.   Main Steam Isolation Valves (MSIV)

      The main steam isolation valves are qualified for seismic and
      hydrodynamic loads by analysis and test.

      The fundamental requirement of the MSIV following an SSE or other
      faulted hydrodynamic loading is to close and remain closed after
      the event. This was demonstrated by the tests and analyses
      outlined in <Section 3.9.3.2.3.a.1>.

n.   Standby Liquid Control Valve (Explosive Valve)

      The standby liquid control valve complies with IEEE 344-1975 for
      seismic and hydrodynamic loads. The qualification test
      demonstrated the absence of natural frequencies in the frequency
      range of the input response spectra and the ability to remain
      operable after the application of horizontal and vertical dynamic
      loading in excess of the required response spectra (RRS).

o.   Main Steam Safety/Relief Valves

     Due to the complexity of the structure and the performance
     requirements of the valve, the total assembly of the safety/relief
     valve (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.
     Tests and analyses demonstrated satisfactory operation of the
     valves during and after the test. Code analysis results are
     summarized in <Table 3.9-3g>.

p.   Fuel Pool Cooling and Cleanup System Pump and Motor Assembly

     The analysis performed on the fuel pool cooling and cleanup system
     pump/motor assembly shows that the fundamental frequency of the
     combination of pump and motor is greater than 33 Hertz. Therefore,
     static analysis was used.

     This analysis shows that the pump will continue to operate when
     subjected to a combination of SSE plus normal operating loads.
     Also, the motor will continue to operate under the effect of
     seismic loads.

     No inelastic analysis is required for ASME Code, Section III,
     Subsection ND, components.

     The design loading combination for the pump is as follows:

     Normal Plus Upset

     This combination includes the simultaneous effect of normal
     operating loads, design pressure, temperature, nozzle loads, dead
     weight loads, seismic (OBE loads), and torsional loads due to
     rotation of the component assembly.

Faulted Loads

Same as normal plus upset, with the addition of SSE loads.

Analysis was performed on the fuel pool cooling and cleanup system pump/motor assembly as follows:

1. Analysis in lieu of testing was performed to ensure that pump running clearances are met under seismic conditions. Analysis included deflection of pump shaft and pump pedestal.

2. Natural frequency was calculated to verify that: the equipment is rigid, natural frequency is greater than 33 Hertz and resonance is avoided with operating frequency.

3. Component support includes pedestal and pedestal bolts.

4. Loads combined were seismic, pipe loads, temperature, and equipment dead weight.

5. Stress limits were in accordance with the ASME Code, Section III, Subsection ND and Subsection NF, Paragraph NF 1211.b.

6. Stress values are presented in <Table 3.9-29>.

q. Safety-Related Pumps

The testing and analytical qualification of active pumps are presented in <Section 3.9.3.2>.

r.   Safety-Related Valves

The testing and analytical qualification of active valves are presented in <Section 3.9.3.2>.

s.   Safety-Related Cranes

The reactor building polar crane, fuel handling crane and emergency service water pump house crane have been designed to withstand the SSE without dropping a full rated load. However, these cranes may be considered as not operable after the event. These cranes are also designed to withstand an OBE without dropping a full rated load and to retain structural and mechanical integrity. The vendor has demonstrated the ability of the equipment to satisfy these requirements by dynamic modal analysis. The reactor building polar crane is qualified to the additional building accelerations induced by hydrodynamic forces discussed in <Appendix 3A>.

The crane analysis used a lumped mass model, Harnischfeger's MRI/STARDYNE computer program and hand calculations to develop the bridge and trolley natural frequency and moments, reaction loads and component stresses. Acceleration values used in the seismic analysis by Harnischfeger were supplied by Gilbert Associates, Inc., in the form of response curves which provide acceleration as a function of frequency at the component location.

Component stresses in the cranes were evaluated, taking into consideration the load combinations and stress limits identified in <Table 3.9-6>.

t. Emergency Service Water Pumphouse Traveling Screen

The emergency service water pumphouse traveling screen has been
seismically qualified by means of stress and deflection analysis of
the screen, the gear/motor drive and the electrical panel.

Major screen components and the frame have been analyzed for stress
using the "General Frame Analysis" computer program, written by
ECOM Associates. Results of the computer analysis indicate that
the stress in the structural elements is within satisfactory limits
for the OBE and SSE loading conditions.

Gear drives for the screen have been statically analyzed to assure
that unit component loads are acceptable during and after the
seismic event. Inertial effects of the motor and motor coupling
were included in the analysis.

A seismic modal analysis of the electrical control panel was
performed to evaluate the control panel box, panel box supports and
attachment of devices to the control panel. Frequencies of
vibration as low as 9.7 Hertz were observed in the torsional mode
of vibration. All stresses calculated due to the combined effects
of service loads and earthquake loads were found to be acceptable.
Due to the complex nature of the flexibility of the structure which
included the torsional stiffness of the pedestal, as well as the
stiffness of the back of the panel box, the flexibility arrays were
computed by hand. Seismic modal analyses were then performed for
the vertical and two horizontal directions following a frequency
analysis.

u. Suction Line Strainer for Core Cooling Systems

The suction line strainer was qualified by analysis to the building
vibrations caused by seismic and hydrodynamic forces. The

hydrodynamic forces are described in <Appendix 3A>. The
appropriate dynamic load factors were applied in conjunction with
the bounding hydrodynamic loading identified within each load
combination.

Results of the analysis of the suction line strainer for core
cooling systems inclusive of RCIC, indicate that the strainer will
maintain its structural and pressure-retaining integrity for
anticipated loading under normal, emergency, upset, and faulted
plant conditions.

v.   Emergency Closed Cooling Heat Exchanger and Fuel Pool Heat
     Exchanger

The fundamental frequency of the fuel pool heat exchangers and
supports is 35 Hertz, and the fundamental frequency of the
emergency closed cooling heat exchanger and supports is 57 Hertz.
Since the fundamental frequency is above 33 Hertz, a static
analysis was performed.

Results of the static analysis of the equipment by means of the
methods described below indicate component stresses are within the
allowable values stipulated by the ASME Code, Section III. The
combination of loadings for the equipment and the corresponding
allowable stress levels are presented in <Section 3.9.3>. Input
accelerations for the determination of seismic loads by Struthers
Wells Corporation were derived from floor response spectra for the
control complex and fuel handling buildings as appropriate.

The method of static analysis for each heat exchanger is
essentially that developed by P. B. Bijlaard for local loadings on
cylinders. This method has been organized for practical
application in Welding Research Council Bulletin No. 107, extended
by Struthers Wells Corporation to include the effect of pressure

and the presence of a reinforcing pad, and automated in the
Struthers Wells Corporation computer program NOZM2. A complete
documentation and verification package concerning the NOZM2 program
is provided by Struthers Wells Corporation Report 1-75-620-3.

The geometry analyzed corresponds to the fully corroded condition
of the equipment.

Loadings and resultant stresses in the heat exchanger saddles and
lateral supports were analyzed based upon the Zick Method as
described in "Process Equipment Design," by Brownell and Young.
This method has been extended by Struthers Wells Corporation to
include stacked vessels and inertia loadings in the axial,
transverse and vertical directions, as well as dead weight
loadings. This extended method is coded into the Struthers Wells
Corporation proprietary computer program, TUSUP, modified to
account for sliding supports and lateral bracing.

w.   High Density Spent Fuel Storage Racks

     The spent fuel storage rack structures discussed in <Section 9.1>
     have been dynamically and statically analyzed to withstand the
     forces and moments caused by OBE and SSE seismic events.

     The design and analyses have accounted for the following:

     1.   Dead and live loads defined as the wet weight of the structure
          plus the fuel load.

     2.   Seismic loads including impact due to seismic.

     3.   Thermal loads due to differential expansion of materials.

     4.   Impact loads due to accident condition fuel drop.

5. The virtual horizontal mass effects of all internal trapped water and external water.

6. Added forces due to fuel bundles impacting through a 0.375 inch gap during horizontal seismic excitation.

7. Time history time increment variations of ±15%.

8. Structural damping (in air) of 7% for bolted structures for SSE and 4% for OBE.

9. Simultaneous horizontal and vertical seismic inputs.

10. Temperature effects from an excursion of 70°F to 212°F.

The time history conforms to equipment response spectra as developed in the building analysis at the spent pool floor elevation.

The time studies have been analyzed via the computer code ANSYS to determine the effects of rack lift-up and fuel rattling. The modules have been represented by a simplified dynamic model. However, the mass and flexibility correspond to the detailed finite element model developed for the static-dynamic analysis.

Maximum inertial and restraint loads as determined in the dynamic (ANSYS) model have been used as input to the static (SAP IV) model to determine member stresses under seismic loading.

The seismic analysis acceptance criteria are based on the following
load combinations and stress limits:

    <u>Load Combination</u>   <u>Stress Limit</u>

    D + L + T + E F_s

    $D + L + T^1 + E^1$ $1.6\ F_s$

where:

    F_s = Allowable working stress:

    (a) For aluminum: based on the Aluminum Construction Manual
        Section 1, Specifications for Aluminum.

    (b) For stainless steel: based on ASME III, Appendix XVII.

    D = Dead load of racks including the support framing.

    L = Live load due to the weight of fuel assemblies, varying
        from zero to full load, and loadings corresponding to
        varying placement of the fuel assemblies in the rack so
        that the most critical loads are obtained.

    T = Thermal loads for water temperature equal to 150°F. The
        minimum water temperature is 40°F.

    E = Loads and resulting forces and moments generated by the
        Operating Basis Earthquake (OBE) resulting from ground
        surface horizontal acceleration and vertical ground
        surface acceleration acting simultaneously.

E^1 = Loads and resulting forces and moments generated by the Design Basis Earthquake (DBE) resulting from ground surface horizontal acceleration and vertical ground surface acceleration acting simultaneously.

T^1 = Thermal loads for loss of coolant condition corresponding to pool surface temperature equal to 212°F.

For a detailed analysis of the spent fuel storage rack structures, refer to the following PAR Systems Corp. reports:

(a) DC-3156-1 Design and Fabrication Criteria.

(b) DD-3156-1 Seismic Model, Description, Formulation, and Assumptions.

(c) DR-3156-3 4-Cavity Module Seismic Analysis.

(d) DR-3156-4 Seismic Time History Rack Analysis.

(e) DR-3156-5 Static Seismic Rack Analysis.

x.   Safety-Related Tanks

Safety-related tanks are qualified to seismic vibration loads. The safety-related tanks housed inside the reactor building will also be qualified to the building vibrations induced by the hydrodynamic loadings described in <Appendix 3A>.

The following is a brief description of the approach followed to analyze safety-related tank designs. A detailed discussion of techniques and procedures used in the analysis is presented in the Structural Dynamics Research Corporation, "Final Report on ASME III

Analysis of Nuclear Safety-Related Shop Fabricated Tanks," Proposal Number 6017-11 (Reference 8).

The general approach used in evaluating the stresses for each tank and support configuration consists of a combination of finite element computer models and hand calculations. The ANSYS finite element program was employed in both static and dynamic analyses to determine the stresses in the tank vessel walls and support structures. Forces and moments calculated in the finite element models were used to evaluate such component items as beam stiffness, ring stiffeners, bolted flanges, and anchor bolts. Local shell stresses in the vicinity of external pipe nozzle attachments were calculated by using Welding Research Council Bulletin No. 107.

To compute stresses in underground tanks due to seismic disturbances, the effects of soil deformations influencing behavior of the tank were considered.

Each tank and supporting structure were evaluated under the following types of loading:

1.   Normal, static loadings due to items such as internal pressure, hydrostatic pressure, dead weight, and soil weight.

2.   Seismic loads due to SSE and OBE as defined in the specification floor response curves.

3.   Design piping loads applied to the external nozzle connections for the purpose of determining only local shell stresses.

The individual stress components determined from the above loadings were combined in accordance with the load combinations and stress limits defined in <Section 3.9.3>.

A summary of the important results from the seismic and stress analyses of each tank design is presented by <Figure 3.9-5>, <Figure 3.9-6>, <Figure 3.9-7>, <Figure 3.9-8>, <Figure 3.9-9>, <Figure 3.9-10>, <Figure 3.9-11>, <Figure 3.9-12>, <Figure 3.9-13>, <Figure 3.9-14>, and <Figure 3.9-15>. Each figure shows a schematic representation of the tank and support structure plus a table of components.

The stress levels in each item tabulated in Figures 3.9-5 through 3.9-15 were investigated under the loading conditions. Only results pertaining to the final acceptable tank and support design and any necessary design modifications have been included in each figure. Detailed results, such as maximum stresses predicted for each item, are presented for each tank system in (Reference 8).

In the stress analysis summary, the status of stress levels is either satisfactory or unsatisfactory, according to the allowable stress criteria presented in <Section 3.9.3>. Detailed stress criteria for each tank can be found in (Reference 8).

The original design refers to the configuration of the tank and support structure that was analyzed and evaluated before any necessary design modifications were made. The tables of component items in Figures 3.9-5 through 3.9-15 represent the original designs. Any overstressed components are indicated by item number. The plant condition under which overstressing of the component occurred is also listed. Necessary modifications to the overstressed items that reduced the stress levels sufficiently to satisfy the specified allowable stress criteria are also tabulated. Finally, the status of the stress levels in the tank and support structure, including design modifications, is indicated. In all cases the modifications have been made and the final status was determined to be satisfactory.

y.   Safety-Related Instrumentation

Methods and procedures used for testing and analysis to seismically
qualify instrumentation are described in <Section 3.10>.

z.   Safety-Related Ventilation Equipment

The ventilation equipment listed in <Table 3.9-7> has been
seismically tested or analyzed. Items 1 through 7 below describe
the test and/or analytical methods and results applicable to each
manufacturer.

1.   Westinghouse Sturdevant Fans

Nine fan sizes were analyzed: 3060, 3054, 3022, 3049, 3037,
409M, 8037, and 3030 (two). These analyses satisfied
requirements for Westinghouse fans listed in <Table 3.9-7>.
The analyses are discussed in Westinghouse Report
No. PHY-2110, December 8, 1978, supplemented March 12, 1979.

Critical portions of the fan assembly were analyzed to verify
that natural frequencies are 33 Hertz or higher. Seismic
stresses were obtained using appropriate floor response
spectra for SSE and OBE. Horizontal accelerations were
determined by the square root of the sum of the squares
method. Fan housing bracing and deflection were reviewed with
formulas derived from testing of a simulated housing model. A
computer program was used to determine fan resonant speed,
shaft stress and pedestal loading.

Inlet vane controls were tested on a 33 inch fan. Since the
test loading greatly exceeded that of the supplied fans, the
vane controls are suitable for seismic loading.

Stresses obtained under SSE conditions were compared to the yield point of the material and the factor of safety is listed in the columns labeled "FS1" in <Table 3.9-8>. OBE stresses were compared to allowable working stress and the factor of safety is listed in the columns labeled "FS2" in <Table 3.9-8>. The factors of safety in <Table 3.9-8> are linearly adjusted to reflect the increased weight of the replacement motors.

Original motors for the Westinghouse fans were furnished by Reliance Electric Co. Reliance Electric Co. Report No. 78-1-36 describes analysis of a 100 horsepower motor typical of the type furnished with these fans. The analysis is based upon computer program No. 706 which is documented in proprietary Reliance Electric Co. Report RSP-3-0074-01-751-713, November 1975. The procedure used is the dynamic-rigid analysis specified by IEEE Standard 344. Results are tabulated in <Table 3.9-9> where calculated stress data is compared to safety limit stress data. The comparison indicates that motor function will not be adversely affected by the postulated seismic event. Replacement motors are also seismically qualified in accordance with <IEEE Standard 344>. The results in <Table 3.9-9> are not reflective of the replacement motors.

2.   CVI Corp. Filter Units

Four types of filter plenums and components were analyzed to demonstrate the ability of the equipment to satisfy structural requirements under applicable seismic and operating loads.

The analysis included separate qualification of the housing
structure, enclosed components and support structure. Static
and dynamic calculations were performed. The dynamic analysis
included calculation of stresses resulting from shock loading
in the x, y and z directions. Adequacy of the design is
demonstrated by the stresses in the plenum structures and
components which are summarized in <Table 3.9-10>.

3. Filter Plenums Supplied by American Air Filter

Plenums for the fuel handling building supply, motor control center/switchgear and miscellaneous electrical system, and control room supply were analyzed. Analyses are discussed in American Air Filter Reports NESE-294, December 21, 1977, and NESE-309, March 15, 1978.

Analytical procedures were used to demonstrate the ability of the equipment to perform required functions under applicable seismic and operating loads. The analysis included separate qualification of the housing structure, enclosed components and support structure.

The plenum model described structural members, joints and housing wall surfaces. A structural analysis computer program, STRUDL/DYNAL, was used to determine natural frequencies, modes of vibration less than 40 Hertz and to perform a dynamic response analysis.

Loadings of members due to static loads were summed to the response analysis loads. These combined loads were applied to the flexibility matrix developed from the modal analysis stiffness matrix. Joint displacements were calculated for loads applied to the flexibility matrix. From the joint displacements, member end forces and moments are obtained.

Results of the motor control center/switchgear and miscellaneous electrical area plenum and control room supply plenum were code checked in accordance with the criteria of Subsection NF of the ASME Code, Section III.

Analytical results were interpreted and judged in the
following areas:

(a) Mode shapes and frequencies.

(b) Member and elemental stresses.

(c) Joint displacement.

(d) Reactions.

For finite element stresses, a yield strength of 36,000 psi
and limiting stresses of 21,600 psi for OBE loads, and
32,400 psi for SSE loads were used. Analytical results from
the STRUDL computer program were checked and no finite element
stress approached American Institute of Steel
Construction (AISC) allowables under any loading condition.

Results of the analyses for anchoring are presented in
<Table 3.9-11>.

4.   Carrier Air Handling Units

Three air handling units, sizes 39ED15, 39ED18, and 39BA050
were analyzed. The analysis is discussed in John Henry
Associates Reports JHA-76-73, JHA-76-73A and JHA-83-226.

A computer model of each unit was developed for use with the
COC computer code STARDYNE. The air handling units were
divided into beam and plate elements to describe the mass and
stiffness distribution of the system and these elements were
terminated at nodal points.

The fundamental natural frequency of the air handling units was determined to be above 33 Hertz. Therefore, a static analysis was performed. Additionally, coil support structures were analyzed in accordance with the requirements of Subsection NF of the ASME Code, Section III.

Adequacy of the design is indicated by comparison of calculated beam, plate and anchor bolt stresses with allowable stresses for the air handling unit structure. Similar calculations and comparisons were performed for the support structure. The comparisons, presented by <Table 3.9-12>, indicate that equipment function will not be adversely affected by the postulated seismic event.

Air handling unit motors were analyzed in accordance with a Reliance Electric Company computer program, No. 706, documented in proprietary Reliance Electric Company Report RSP-3-0074-01-751-713, November 1975. The procedures used in the dynamic-rigid analysis are specified by IEEE Standard 344. Analyses are summarized in Reliance Electric Company Reports 77-A-34 and 77-A-36. Results are tabulated in <Table 3.9-13> where calculated stress data are compared to safe limit stress data. This comparison indicates that motor function will not be adversely affected by the postulated seismic event.

5.   Water Chillers

Seismic analysis of a typical hermetic centrifugal liquid chiller was performed by NUS Corporation. The analysis is described in NUS Corporation Report No. 1985, March 15, 1977.

Three water chillers are provided. Since these units are structurally identical and are subjected to identical

loadings, a single analysis and finite element model were used to analyze the chillers for dead weight, nozzle and seismic loadings. Frequencies, mode shapes, seismic loads, and stresses were determined using the STARDYNE computer program and applicable response spectra.

The DBE and OBE earthquake conditions were analyzed simultaneously. Since the chiller fundamental frequency is less than 33 Hertz, a dynamic analysis was performed. Modes were combined by the square root of the sum of the squares method.

A dead weight plus nozzle load analysis was performed using the finite element model and the STATIC option of the STARDYNE computer program.

Total stress on each component was determined by summing the effects of seismic, nozzle and dead weight loads. Where applicable, general primary membrane stresses due to pressure were included. A local shell analysis of all major pipe-to-shell attachments was performed using the Welding Research Council Bulletin No. 107 and the Johns and Orange pressure discontinuity analysis. Support components were analyzed in accordance with Subsection NF of the ASME Code, Section III.

Deflections at each nodal point in the STARDYNE model were calculated using an absolute sum of seismic, nozzle and dead weight deflection results.

Adequacy of the design is demonstrated by the comparison of maximum stresses to Code allowable in <Table 3.9-14>.

6.   Water Chiller Control Panel

A carrier Model 19FA control panel is mounted on each water chiller. A panel typical of the Model 19FA panels was seismically tested by Acton Environmental Testing Corporation under general supervision of NUS Corporation. This testing is described in NUS Corporation Report TR-77-17A, March 21, 1977.

The typical panel was attached to a test fixture in a manner simulating the inservice mounting arrangement. The test fixture was then mounted on the moving table of the Acton Environmental Testing Corporation's seismic vibration facility. Eleven accelerometers were mounted at various points on the panel and at the interface between the panel and test fixture.

Biaxial low level sine sweep resonance surveys were performed. The biaxial sine beat tests were performed in each of four directions. Instruments, sensors and relays were subjected to simulated operating conditions during these tests.

Test response spectra of acceleration versus frequency were developed from the results of the above tests. The test response spectra exceeded the required response spectra over the entire frequency range for all test directions. Based on test results, including proper function of the panel, switches, sensors, and relays during and after testing, it was concluded that the water chiller control panels and components are seismically qualified.

7.   Buffalo Forge Co. Fans

     Two Adjustax fans (43B7 and 60D5) were analyzed. The analysis
     is described in McMahon Engineering Co. Report
     No. 76J-2935-36.

     A static analysis of the fans was performed by evaluating the
     stresses caused by the seismic inputs for each of the three
     principal axes. These stresses were then combined by the
     square root of the sum of the squares method to produce the
     combined seismic stress.

     Maximum operating stresses plus the stresses due to dead
     weight of equipment were calculated and added directly to the
     combined seismic stresses.

     Motors for these fans were qualified by means of analysis in
     accordance with IEEE Standard 344.

     Margins of safety were determined for the fans and motors by
     comparing total stresses. This comparison, presented by
     <Table 3.9-15>, indicates that the capability of the equipment
     to function will not be adversely affected by the postulated
     seismic event.

3.9.2.3   Dynamic Response Analysis of Reactor Internals Under
          Operational Flow Transients and Steady-State Conditions

The major components within the reactor vessel were subjected to
extensive testing coupled with dynamic system analyses to properly
evaluate the resultant 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 analysis of the response signals measured for reactor
internals of many similar designs are performed to obtain the parameters
which determine the amplitudes and modal contributions in the vibration
responses. These studies provide useful predictive information for
extrapolating the results from tests of components with similar designs
to components of different designs. This vibration prediction method is
appropriate where standard hydrodynamic theory cannot be applied due to
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 vibration modes and frequencies. The analysis models
     used for Seismic Category I structures are similar to those
     outlined in <Section 3.7.2>, Seismic System Analysis.

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 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.

e.   Predicted vibration amplitudes for components of the prototype
      plant <Section 3.9.2.4> 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 was earlier obtained from the
      dynamic analysis in Paragraph a.)

The dynamic modal analysis also forms the basis for interpretation of
the preoperational and initial startup test results <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 acceptance criteria associated with the
preoperational and initial startup tests are contained in
<Section 14.2.12.2.30>.

3.9.2.3.1   Jet Pumps, Core Support, Steam Separators, LPCI Coupling

The magnitude of the jet reaction loads applied to the reactor internal
structures caused by acceleration and deceleration of the flow under
normal and upset conditions are negligible compared to the differential
pressure loads, and generally need not be considered. Jet reaction
loads that require consideration are those associated with the jet pump
assembly and riser, and within the steam separator itself. The upward
jet reaction loads on the separator assembly are canceled by the
downward jet impingement loads at the upper surface of the shroud head
dome.

Vibratory loads are continuously applied during normal operation and the
stresses are limited to ±10,000 psi to prevent fatigue failure. This
±10,000 psi value allows the acceptance of vibratory loadings in reactor
internal components without the need for additional detailed fatigue
analysis. For components with alternating stress levels in excess of

±10,000 psi, additional analysis is performed to ensure that the fatigue usage factor does not exceed 1.0 as described in <Section 3.9.5.3.5> and <Table 3.9-35>. Prediction of vibration amplitudes, mode shapes and frequencies for normal reactor operations are based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation.

In order to evaluate the dynamic response of the jet pumps, two locations were chosen for monitoring on jet pumps in the prototype plant. These locations are the riser brace and the diffuser of the jet pump. The reasons for selecting these positions were sensitivity and accessibility. Knowing the strain response at these gauge locations, the stresses at other locations can be predicted as well as the mode of vibration, response frequency and displacement. These values are compared to analytical criteria and, thus, their acceptability is evaluated.

The load due to crossflow from the jet pumps to the peripheral control rod guide tubes is 620 pounds on the bottom one-eighth of the guide tube length, 345 pounds on the next higher one-eighth of the guide tube length and 130 pounds on the next one-quarter length of the guide tube.

The stresses produced due to vibratory loads are 375.5 psi and are considered negligible.

The dynamic loads due to flow induced vibration from the feedwater jet impingement would have no significant effect on the steam separator assembly.

Analysis has shown that the impingement feedwater jet velocity is 12 ft/sec, far below the critical velocity of 118 ft/sec. Also, analysis has shown that the excitation frequency of the steam separator skirt is 5.1 Hertz, and the natural frequency of the skirt is 50 Hertz.

The load due to flow-induced vibration will have no effect on the LPCI coupling since the calculated natural frequency of the coupling is over 50 Hertz.

The calculated stresses due to the hydrodynamic forces during normal operating conditions are small and considered negligible when compared to the design allowable stresses. Locations for which calculations were made include the weld joints, elbows and rings.

3.9.2.4   Preoperational Flow-Induced Vibration Testing of Reactor Internals

Vibration measurement and inspection programs were conducted during preoperational and initial startup testing of first-of-a-kind reactor internals configurations (Grand Gulf 1 is the designated prototype 251 size BWR/6 plant. Perry 1 is the designated prototype 238 size BWR/6 plant) in accordance with guidelines of <Regulatory Guide 1.20> for prototype reactor internals. These programs were conducted in the prototype plants in three phases, described as follows:

a.   Preoperational Tests Prior to Fuel Loading

Steady-state test conditions included balanced (two-pump) recirculation system operation and unbalanced (single-pump) operation over the full range of flow rates up to rated flow. Transient flow conditions included single and two-pump trips from rated flow. The specified test duration was 35 hours of balanced operation, plus 14 hours of single-pump operation of each recirculation loop, for a total of 63 hours. This subjected major components to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements were obtained during this test and a close visual inspection of internals was conducted before and after the test.

b.   Precritical Testing with Fuel

This vibration measurement series was conducted with the reactor
assembly complete but prior to reactor criticality. Flow
conditions included balanced, unbalanced and transient conditions
like the first test series. The purpose of this series was to
verify the anticipated effect of the fuel on the vibration response
of internals. Previous vibration measurements in BWRs
(Reference 9) have shown that the fuel adds damping and reduces
vibration amplitudes of major internal structures. Thus, the first
test series (without fuel) was a conservative evaluation of the
vibration levels of these structures.

c.   Initial Startup Testing

Vibration measurements were made during reactor startup at
conditions up to 100 percent power and maximum flow. Balanced,
unbalanced and transient conditions of recirculation system
operation were evaluated. The primary purpose of this test series
was to verify the anticipated effect of two-phase flow on the
vibration response of internals. Previous vibration measurements
in BWRs (Reference 9) have shown that the effect of the two-phase
flow is to broaden the frequency response spectrum and diminish the
maximum response amplitude of the shroud and core support
structures.

Vibration sensor types included strain gauges, displacement sensors
(linear variable transformers) and accelerometers. Accelerometers
were provided with double integration signal conditioning to give a
displacement output. Sensor locations will include the following:

1.   Top of shroud head, lateral acceleration (displacement).

2.   Top of shroud, lateral displacement.

3. Jet pump riser braces, bending and extension strains.

4. Jet pump diffuser, bending strain.

5. Control rod guide tube, bending strain (preoperational only).

6. Incore housings, bending strain.

7. Core spray sparger piping, bending strain.

In all prototype plant vibration measurements, only the dynamic component of strain or displacement was recorded. Data was recorded on magnetic tape, and provision was made for selective online analysis to verify the overall quality and level of the data. Interpretation of the data required identification of the dominant vibration modes of each component by the test engineer, using frequency, phase and amplitude information from the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes was then made on the basis of the analytically obtained normal mode which best approximates the observed mode.

Visual inspections were conducted prior to and following preoperational testing for vibration, wear or loose parts. At the completion of preoperational testing, the reactor vessel head and the shroud head were removed, the vessel was drained and major components were inspected on a selected basis. The inspections covered the shroud, shroud head and core support structures, the jet pumps, and the peripheral control rod drive and incore guide tubes. Access was provided to the reactor lower plenum for these inspections.

d.   Compliance With <Regulatory Guide 1.20>

PNPP is committed to comply with <Regulatory Guide 1.20>.

<Regulatory Guide 1.20> describes a comprehensive vibration
assessment program for reactor internals during preoperational and
initial startup testing. The vibration assessment program meets
the requirements of Criterion 1, "Quality Standards and Records,"
of <10 CFR 50, Appendix A> and <10 CFR 50.34>, "Contents of
Applications: Technical Information".

Vibration testing of reactor internals is performed on all GE-BWR
plants. Perry, being the first BWR/6 238 plant, is considered a
prototype and has been instrumented and subjected to preoperational
and startup flow testing to demonstrate that flow-induced
vibrations, similar to those expected during operation, will not
cause damage.

General Electric is committed to confirm satisfactory vibration
performance of internals in these plants through preoperational
flow testing followed by inspection for evidence of excessive
vibration. Extensive vibration measurements in prototype plants
together with satisfactory operating experience in all BWR/4 plants
have established the adequacy of BWR/6 reactor internal designs.

3.9.2.5   Dynamic System Analysis of the Reactor Internals Under
             Faulted Conditions

In order to assure that no significant dynamic amplification of load
occurs as a result of the oscillatory nature of the blowdown forces
<Figure 3.9-16> and <Figure 3.9-17>, a comparison was made of the
periods of the applied forces and the natural periods of the core
support structures acted upon by the applied forces. These periods were
determined from a 12 node vertical dynamic model of the BWR6-238 RPV and

internals. Only motion in the vertical direction was considered; hence, each structural member (between two mass points) only had an axial load. Masses of the RPV, core support structures and water inside the RPV were considered.

Typical curves of the variation of pressures during a steam line break are shown in <Figure 3.9-16> and <Figure 3.9-17>. The accident analysis method is described in <Section 3.9.5.2>.

The time-varying pressures were applied to the dynamic model of the reactor internals. 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 dynamic components of forces from these loads combine with the force components from other dynamic loads (including seismic and hydrodynamic), all acting in the same direction, by the square root of the sum of the squares (SRSS) method. The resultant force was combined with other steady-state and static loads on an absolute sum basis to determine the design load in a given direction.

A summary of the results of the reactor internals dynamic analysis is given in <Table 3.9-3b>.

3.9.2.6   Correlations of Reactor Internals Vibration Tests with the Analytical Results

BWR 4 and 5 reactors now in service provide the basis for analytical prediction of vibrational behavior during the component design stage. GE Licensing Topical Report NEDE-24057-P, presented to the NRC for Susquehanna, contains results of such tests and measurements. However, the BWR 4 and 5 operational experience has not been used in lieu of vibration measurement for the Perry reactor internals. Perry's component design adequacy for flow-induced vibration is confirmed through actual in-reactor measurements. Additionally, Grand Gulf, with

similar flow characteristics and internals design, will provide
additional operating experience.

Prior to initiation of the instrumented vibration measurement program
for Perry, extensive dynamic analyses of the reactor and internals were
performed. The results of these analyses were used to generate
allowable vibration levels during the vibration test. The vibration
data obtained during the test are analyzed in detail to obtain vibration
amplitudes, natural frequencies and mode shapes; comparison is then made
to those parameters obtained from the dynamic model for seismic and LOCA
analyses.

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>.

This section also lists the major ASME Code 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
Code Class 1 equipment are addressed in <Section 3.9.1.1>. Seismic
related loads are discussed in <Section 3.7>. The suppression
pool-related hydrodynamic loads are described in detail in <Appendix 3A>
and in the Dynamic Forcing Function Information Report (Reference 10).

<Table 3.9-3> is the major part of this Section; it presents the
consideration of loads, analytical methods (by reference or example) and

also the calculated stress or other design values of the most critical
areas in the design of each component. These values are also compared
to applicable Code allowables.

<Table 3.9-3> represents the generic loading combinations required to be
considered for the design and analysis of a typical BWR standard plant
and are therefore applicable to all ASME Code Class 1, 2 and 3
components, component supports and core support structures. Specific
loading combinations considered for equipment are derived from
<Table 3.9-3> and are contained in the design specification and/or
stress report for the respective equipment.

3.9.3.1.1   Loading Combinations, Design Transients and Stress Limits
               for NSSS Components and Supports

a.   Plant Conditions

     All events that the Plant might credibly experience during a
     reactor year are evaluated to establish a design basis for plant
     equipment. These events are divided into four plant conditions.

     The plant conditions described in the following paragraphs are
     based on event probability, i.e., frequency of occurrence, and
     correlated design conditions defined in the ASME Code, Section III.

     1.   Normal Condition

          Normal conditions are any conditions in the course of system
          startup, operation in the design power range, normal hot
          standby (with condenser available), and system shutdown other
          than upset, emergency, faulted, or testing. For essential
          systems, normal system operation also includes prevailing
          conditions while operating during an emergency or faulted
          plant condition.

2. Upset Condition

    Upset conditions are any deviations from normal conditions
    anticipated to occur often enough that design should include a
    capability to withstand the conditions without operational
    impairment. The upset conditions include those transients
    which result from any single operator error or control
    malfunction, transients caused by a fault in a system
    component requiring its isolation from the system, and
    transients due to loss of load or power, or an operating basis
    earthquake. Hot standby with the main condenser isolated is
    an upset condition.

3. Emergency Condition

    Emergency conditions are those deviations from normal
    conditions which require shutdown for correction of the
    conditions or repair of damage in the reactor coolant pressure
    boundary. 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; and improper assembly of the
    core during refueling.

4. Faulted Condition

    Faulted conditions are those conditions associated with
    extremely low probability, and 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 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 a small break
    accident or large break accident dynamic motion associated
    with a safe shutdown earthquake and hydrodynamic loads plus a
    loss of offsite power, or the safe shutdown earthquake.

b. Correlation of Plant Conditions with Event Probability

    The probability of plant conditions occurring is listed below.
    This correlation can be used to identify the appropriate plant
    condition for any hypothesized event or sequence of events.

Plant Conditions	Event Encounter Probability Per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	$1.0 > p > 10^{-2}$
Emergency (low probability)	$10^{-2} > p > 10^{-4}$
Faulted (extremely low probability)	$10^{-4} > p > 10^{-6}$

c.   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 deterioration in ability to accomplish 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 safety functions as required by any subsequent design condition event.

d.   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 the Seismic Category I fluid system components. GE-supplied NSSS analyses, designs and/or equipment utilized in this facility are in compliance with the intent of <Regulatory Guide 1.48> through incorporation of the alternate approach cited in <Table 3.9-16>.

See the Regulatory Guide assessment matrix in <Section 1.8> for scope of conformance.

e.   Reactor Pressure Vessel Assembly

The reactor pressure vessel assembly consists of the reactor pressure vessel, vessel support skirt and shroud support.

The reactor pressure vessel, vessel support skirt and shroud
support are constructed in accordance with Section III of the ASME
Code. The shroud support consists of the shroud support plate, the
shroud support cylinder and its legs. The reactor pressure vessel
assembly components are classified as ASME Code, Subsection NB.
Complete stress reports on these components have been prepared in
accordance with the ASME Code requirements. The stress analysis is
performed on the reactor pressure vessel, vessel support skirt and
shroud support for various plant operating conditions (including
faulted conditions) by using elastic methods except as noted in
<Section 3.9.1.4.3>.

<Table 3.9-3a> provides a summary of the stress criteria, loading
conditions, calculated stresses, and allowable stresses.

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>.

f.   Main Steam Piping

The main steam piping extending from the reactor pressure vessel to
the outboard main steam isolation valve is designed in accordance
with the ASME Code, Section III, Subsection NB-3600. The loading
conditions, stress criteria, calculated stresses and allowable
stresses are summarized in <Table 3.9-3d>.

The rules contained in Appendix F of the ASME Code, Section III are
used in evaluating faulted loading conditions, independent of all
other design and operating conditions. Stresses calculated on an
elastic basis are evaluated in accordance with F-1360.

g. Recirculation System Piping

    The recirculation system piping bounded by the reactor pressure
    vessel nozzles is designed in accordance with the ASME Code,
    Section III, Subsection NB-3600. The loading conditions, stress
    criteria, calculated stresses, and allowable stresses are
    summarized in <Table 3.9-3e>. The rules contained in Appendix F of
    ASME Code, Section III are used in evaluating faulted loading
    conditions, independently of all other design and operating
    conditions. Stresses calculated on an elastic basis are evaluated
    in accordance with F-1360.

h. Recirculation System Valves

    The recirculation system flow control and suction and discharge
    gate valves are designed in accordance with the ASME Code,
    Section III, Class I, Subsection NB, Paragraph 3500. These valves
    are not required to operate under the safe shutdown earthquake.
    Loading conditions and other stress analysis information are
    presented in <Table 3.9-3f> for flow control valves and
    <Table 3.9-3j> for gate valves.

i. Recirculation Pump

    Recirculation pumps are designed in accordance with the ASME Code,
    Section III. These pumps are not required to operate during the
    safe shutdown earthquake. The loading combinations and other
    stress analysis information are presented in <Table 3.9-3i>.

j.   Standby Liquid Control (SLC) Tank

      The standby liquid control tank is designed in accordance with the
      ASME Code, Section III. The loading conditions, stress criteria,
      calculated stress, and allowables stresses are summarized in
      <Table 3.9-3m>.

k.   Residual Heat Removal Heat Exchangers

      The heat exchangers are designed in accordance with the ASME Code,
      Section III. The stress analysis methods, calculated stresses and
      allowable limits for the RHR heat exchangers are shown in
      <Table 3.9-3o>.

l.   RCIC Turbine

      Although not under the jurisdiction of the ASME Code, the RCIC
      turbine has been designed and fabricated following the basic
      guidelines of the ASME Code, Section III, for Class 2 components.

      <Table 3.9-3q> contains a summary for the RCIC turbine components
      of loading conditions, stress criteria, calculated stresses, and
      allowable stresses.

m.   RCIC Pump

      The RCIC pump has been designed and fabricated to the requirements
      for an ASME Code, Section III, Subsection NC component.

      The loading conditions, stress criteria, calculated stresses, and
      allowable stresses for the RCIC pump components are summarized in
      <Table 3.9-3r>.

n.   ECCS Pumps

The RHR, LPCS and HPCS pumps are designed and fabricated in accordance with the requirements of the ASME Code, Section III. The loading conditions, stress criteria, calculated stresses, and the allowable stresses for ECCS pumps are summarized in <Table 3.9-3n>.

o.   Standby Liquid Control Pump

The standby liquid control pump has been designed and fabricated following the requirements for an ASME Code, Section III, Subsection NC component.

The loading conditions, stress criteria, calculated stresses, and allowable stresses for the standby liquid control pump components are summarized in <Table 3.9-3L>.

p.   Main Steam Isolation and Safety/Relief Valves

Loading conditions, calculated stresses and allowable limits are shown for the safety/relief and main steam isolation valves in <Table 3.9-3g> and <Table 3.9-3h>, respectively.

The main steam isolation and safety/relief valves are designed in accordance with the requirements of the ASME Code, Section III, Subsection NB-3500.

q.   Safety/Relief Valve Discharge Piping

The relief valve discharge piping extending from the relief valve discharge flange to the first downstream anchor is designed in accordance with the ASME Code, Section III, Subsection ND-3600. The loading conditions and stress criteria, calculated stresses and

allowable stresses are summarized in <Table 3.9-3k>. The rules
contained in the ASME Code Case 1606-1 will be used in evaluating
faulted loading conditions, independently of all other design and
operating conditions.

3.9.3.1.2   Load Combinations, Design Transients and Stress Limits
                 for Balance of Plant Components and Supports

Balance of plant systems and components are identified in accordance
with ASME Code Class and Safety Class as discussed in <Section 3.2>.
Design limits and load combinations for Seismic Category I fluid system
components are provided in the Section 3.9 tables. ASME Code Class 1, 2
and 3 systems and components are designed to operate under the following
plant conditions:

a.   ASME Code Class 1, 2 and 3 systems and components are designed to
      operate under anticipated environmental conditions such as
      pressure, temperature, irradiation, etc., that may occur during
      normal plant operations and transients, including startup, power
      generation, relief valve operation, and shutdown.

b.   Components of essential systems, required to function during and/or
      after any of the abnormal events identified in <Section 3.9.1> are
      designed to function under environmental conditions that would
      occur during and after such events. <Section 3.11> describes
      environmental design conditions associated with such abnormal
      events.

The plant conditions postulated to occur during the life of the plant
are identified in <Section 3.9.3.1.1.a>.

Loads considered in component design are those effects derived from plant and system conditions of operation, natural phenomena and site related hazards. These loads include, but are not limited to, effects resulting from:

a.   Internal or external pressure.

b.   Deadweight of component and contents.

c.   Superimposed loads, such as the effect of other components, insulation and piping.

d.   Vibrations and seismic inertial loads and seismic anchor displacements for both OBE and SSE. For safety-related components in the reactor building, those building vibrations induced by the hydrodynamic loads described in <Appendix 3A> are also considered.

e.   Thermal expansion and thermal transient loads.

f.   Dynamic forces associated with safety/relief valve operation.

g.   Dynamic loads resulting from steam/water hammer.

h.   Loads resulting from postulated pipe rupture.

Loads producing primary stresses are grouped according to the nature and type of loading as follows:

a.   Sustained loads.

b.   Occasional loads.

c.   OBE inertial loads.

d.   SSE inertial loads.

e.   Design basis pipe break (DBPB) loads.

f.   Main steam pipe break (MSPB) loads.

g.   LOCA loads.

Loading combinations are further classified as service loadings. Each service loading is within the appropriate ASME Code allowable stress limits. Service loadings for a component are defined as follows:

a.   Level A Service Loading:

      Level A service loadings are those load combinations which include all sustained loads resulting from normal plant/system operation. Normal system operation includes prevailing conditions of essential systems while operating during an emergency or faulted plant condition.

b.   Level B Service Loading:

      Level B service loading is the combination of loads resulting from sustained loads, plant/system operating transients comprising upset plant/system conditions, and OBE loadings.

c.   Level C Service Loading:

      Level C service loading is the combination of loads resulting from sustained loads, postulated plant/system transients comprising emergency or faulted plant/system conditions, and their effects.

d.   Level D Service Loading:

     Level D service loading is the combination of loads resulting from
     sustained loads, postulated plant/system transients comprising
     certain faulted plant/system conditions and their effects, and SSE
     loadings.

To satisfy the service loading and stress limit requirements, component
design requirements stipulate the following, which is beyond the scope
of the ASME Code:

a.   Pumps and valves of systems classified as active will satisfy the
     requirements of the pump and valve operability assurance program as
     described in <Section 3.9.3.2>.

b.   Piping, tanks and appurtenances which interconnect active pumps and
     valves are evaluated to assure that functional capability is not
     impaired whenever service limits C or D are used.

Corresponding service loadings and stress limits applicable to the
design of ASME Code Class components are presented as follows:

     <Table 3.9-18> -   Class 1, 2 and 3 valves
     <Table 3.9-19> -   Class 2 and 3 pumps
     <Table 3.9-20> -   Class 2 and 3 vessels
     <Table 3.9-21> -   Class 1, 2 and 3 piping
     <Table 3.9-21a> -   Class 1, 2 and 3 component supports

Design criteria for supports are presented in <Section 3.9.3.4>.

3.9.3.2   Pump and Valve Operability Assurance

The GE supplied active pumps and valves are listed in <Table 3.9-22>.

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 includes active pumps and valves in fluid systems such as
the emergency core cooling systems and main steam system.

Safety-related valves and pumps are qualified by prototype testing and
analysis to satisfy stress and deformation criteria at the critical
location. Operability is assured by meeting the requirements of the
following program.

3.9.3.2.1   ECCS Pumps

All active pumps are qualified for operability by first being subjected
to rigid tests both prior to installation and after installation in the
plant. The in-shop tests include (1) hydrostatic tests of
pressure-retaining parts to 125 percent of design pressure, (2) seal
leakage tests, and (3) performance tests, while the pump is operated
with flow, to determine total developed head, minimum and maximum head
and net positive suction head (NPSH) requirements. Also monitored
during these operating tests are bearing temperatures (except water
cooled bearings) and vibration levels. Both are shown to be below
specified limits. After the pump is installed in the plant, it
undergoes operability tests, functional tests, and 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 a faulted condition by assuring that (1) the pump

will not be damaged during the faulted event, and (2) the pump will
continue operating despite the faulted loads.

3.9.3.2.1.1   Consideration of Loading, Stress and Acceleration
                   Conditions in the Analysis

In order to avoid damage during the faulted plant condition, the
stresses caused by the combination of normal operating loads, nozzle
loads, SSE, hydrodynamic loads, and dynamic system loads are limited to
the material elastic limit, as indicated in <Section 3.9.3.1> and
<Table 3.9-3>. A three-dimensional finite element model of the
pump/motor and its support was developed and dynamically analyzed using
the response spectrum analysis method. The 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 checked
to evaluate operability. The average membrane stress for the faulted
condition loads is maintained at less than 1.2S, and the maximum stress
in local fibers (local membrane stress plus primary bending stress) is
limited to 1.8S. The maximum faulted event nozzle loads are also
considered in an analysis of the pump and its supports to assure that a
system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with
the restrictive stress limits of <Table 3.9-3> as allowable, will assure
that critical parts of the pump will not be damaged during the faulted
condition and, therefore, the reliability of the pump for post-faulted
operation will not be impaired.

3.9.3.2.1.2   Pump/motor Operation During and Following SSE Loading

The rotor is designed to not contact the stator. If it were to contact,
the following applies: active pump motors are designed to rotate at a

constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the dynamic event, will prevent the rotor from becoming seized. In actuality, the seismic and hydrodynamic 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 loads and will operate at design speed despite the SSE loads.

The functional capability of the active pumps after a faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. For active pumps, the faulted condition is greater than the normal condition only due to seismic (SSE) and hydrodynamic loads on the equipment itself. The SSE event is 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 post-faulted condition operating loads will be no worse than the normal plant operating limits. This is assured by requiring that the imposed nozzle (steady-state) loads for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted capability of the pumps to function under these applied loads is proven during normal operating plant conditions for active pumps.

3.9.3.2.1.3   Environmental Qualification of ECCS Motors

Qualification of the ECCS pump Class 1E motors complies with IEEE 323-1974. The qualification of all motor sizes is based on completion of a type test, followed by review and comparison of design and material details and seismic (including hydrodynamic) load analysis of production units (ranging from 600 to 3,500 bhp) vs. the motor type tested. Manufacturing, inspection and routine tests by the motor manufacturer on production units were performed on the test motor.

The type test has been performed on a 1,250 hp vertical motor in accordance with IEEE 323-1974, first simulating normal operation during design life, then a number of vibratory tests including seismic tests, and then the abnormal environmental conditions possible during and after a loss-of-coolant accident (LOCA). The type test was as follows:

a. Thermal aging of the motor's electrical insulation system (which is a part of the stator only) was based on extrapolation in accordance with the temperature life characteristic curve from IEEE Standard 275-1966. The amount of aging equaled the total estimated operating days at maximum insulation surface temperature.

b. Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

c. The normal operationally induced current vibration effect on the insulation system has been simulated by 1.5 g's horizontal vibration acceleration at current frequency for one hour duration.

d. The dynamic loads deflection analysis on the rotor shaft, performed to insure adequate rotation clearance, has been verified by static loading and deflection of the rotor in the type-test motor.

e. Dynamic loads aging and testing has been 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 safe shutdown earthquake, including hydrodynamic loads with motor starts and operating combinations as may possibly occur during a plant life.

f. An environmental test simulating a LOCA condition of 100 days duration has been performed with the test motor fully loaded, simulating pump operation. The test consisted of startup and

six hours operation at 212°F ambient temperature and 100 percent steam environment. Another startup and operation of the test motor after one hour stand-still in the same environment was followed by sufficient operation at high humidity and temperature, based on extrapolation in accordance with the temperature-life characteristic curve from IEEE 275-1966, for the insulation type used on the ECCS motors.

3.9.3.2.2   SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact and rigid, with natural frequencies well above 33 Hertz. With this fact verified, each equipment assembly has been seismically qualified via static analysis only. This static qualification verifies operability under seismic conditions, and assures structural loading stresses within ASME Code limitations.

3.9.3.2.3   NSSS Valves

3.9.3.2.3.1   ASME Code Class 1 Active Valves

The Class 1 active valves are the main steam isolation valves, safety/relief valves, standby liquid control valves, and HPCS valves. Each of these valves is designed to perform its mechanical motion in conjunction with a design basis accident including hydrodynamic loads. Qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.3.1.1   Main Steam Isolation Valve (MSIV)

The MSIVs are evaluated for operability during seismic and hydrodynamic load events by both analysis and test.

a.   First, the valve body is designed in accordance with the ASME Code, Section III, Subsection NB <Table 3.9-3h> which limits deformation within the elastic limit of the material by limiting pressure and pipe reaction input loads (including seismic and hydrodynamic loads). This assures that only small deformations are allowed in the operating area of the valve body, hence, no interference with valve operability.

b.   The entire topworks assembly was dynamically qualified by bidirectional random frequency shake test. The loadings include SRV aging, OBE and SSE motions, and chugging motions. The SRV aging lasted 15 minutes for each pair of vertical and one of the two major horizontal axes. The motion simulation involved 5 intervals of 30 seconds each for the two bidirectional combinations. The SSE simulation involved 1 interval of 30 seconds for the two bidirectional combinations. The chugging motion involved 15 minutes of bidirectional loadings for each pair of major orthogonal axes. The testing covered seismic and hydrodynamic loads. The test response spectra exceeded the required response spectra by 10%. During each test interval the MSIV topworks was cycled from full open to full close to demonstrate operability. After the complete dynamic test program the MSIV topworks was again cycled to assure operability.

Pipe anchors and restraints are located and designed to limit the dynamic response and amplified accelerations to within design limits for the MSIVs. The mathematical modeling of the assembly accounts for the natural frequencies of the assembly as determined by the analysis and confirmed by a generic test.

c.   Main steam isolation valve operability following a downstream line break was demonstrated by the "state line test" described in report APED-5750 (March 1969). The test specimen was a 20 inch valve of a

design representative of the MSIVs. Operability during seismic and
hydrodynamic accelerations is addressed in <Section 3.9.2.2.1>.

Environmental qualification of sensitive electrical/pneumatic
equipment to meet performance requirements is defined in
<Section 3.11>.

3.9.3.2.3.1.2   Main Steam Safety/Relief Valve (SRV)

The SRV is qualified by test for operability during a seismic event
(Reference 26). Structural integrity of the configuration during a
dynamic event is demonstrated by both code analysis and test.

a.   The 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.   A production SRV's operability was demonstrated by a dynamic
     qualification (shake table) test with applied moments and "g" loads
     greater than or equal to the required design limit loads and
     conditions.

     A mathematical model of this valve is included in the main steam
     line system analysis in order to assure that the equipment design
     limits are not exceeded.

c.   The SRV is qualified generically to both IEEE 323-74 and
     IEEE 344-1975. The environmental testing of the
     electrical/pneumatic equipment demonstrated operability after
     radiation aging (30 x 10^6 rads), thermal aging (392°F for 24 hours)
     and mechanical aging (1,000 cycles with a specified load). The SRV

seismic testing demonstrated operability during the seismic and
hydrodynamic events. The SRV is seismically qualified to response
spectra induced at the valve inlet flange of 9 g's for the
horizontal principal axis and 6 g's for the vertical principal
axis, with a concurrent static moment load of 800,000 in.-lb on the
inlet flange and 600,000 in.-lb on the outlet flange.

3.9.3.2.3.1.3   Standby Liquid Control Valve (Explosive Valve)

a.   The standby liquid control valve has been qualified generically to
     both IEEE 323-1974 and IEEE 344-1975.

b.   The seismic and hydrodynamic load testing demonstrated operability
     after completion of the dynamic event which imposes a horizontal
     loading equivalent to 6.5 g's and a vertical loading equivalent to
     4.5 g's.

3.9.3.2.3.1.4   HPCS Valve

The HPCS Class 1 active valve is a motor operated gate valve. This
valve is seismically qualified by stress analysis, and operability has
been assured by static deflection analysis to demonstrate the
operability during and after a simulated design basis load application.
The actuator is qualified in accordance with IEEE 382-1972 to levels
exceeding design loadings.

3.9.3.2.3.2   ASME Code Class 2 and Class 3 Active Valves

The six HPCS Class 2 active valves are motor operated valves.

These valves are generically qualified by testing valves that are
typical of the valves supplied by GE. These tests ensure operability
during and after simulated design basis loads. The actuators are

qualified to IEEE 382-1980, to "g" levels that exceed the Required Response Spectra (RRS).

The four control rod drive (CRD) scram discharge volume (SDV) vent and drain Class 2 valves are air-operated valves. The operability and structural integrity of these valves are demonstrated by type test in accordance with IEEE 344-1975.

There are no Class 3 active valves supplied in NSSS scope.

3.9.3.2.4   Active Pumps and Valves Not supplied by General Electric

3.9.3.2.4.1   Pumps

The safety-related pumps are tabulated in <Table 3.9-23>.

All active pumps were qualified for operability by first being subjected to rigid tests prior to installation in the plant and after installation in the plant. The in-shop tests included (1) ASME Code hydrostatic tests of pressure retaining parts at a test pressure adjusted for the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, (2) seal leakage tests and (3) performance tests, while the pump is operating with flow, to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor parameters. Also monitored during these operating tests were bearing temperatures (except water cooled bearings) and vibration levels. Both were below specified limits. After the pump was installed in the plant, operability tests, functional tests and the required periodic inservice tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the design specification requires the pump manufacturer to comply with the design load combinations and allowable stress limits as discussed in <Section 3.9.3.1> and to provide assurance

of operability verification and certification for the pump-motor assembly.

The safety-related active pumps were analyzed for operability during an SSE condition to assure that (1) the pump will not be damaged during the seismic event, and (2) the pump will continue operating despite the SSE loads.

The pump manufacturer was required to perform a test or dynamic analysis to determine the seismic load from the applicable floor response spectra. Manufacturers of active pumps were also required to demonstrate by test or analysis that faulted condition nozzle loads will not impair the operability of the pump during or following the faulted condition. Components of the pump, having a natural frequency above 33 Hertz, are considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas outside the reactor building. A static shaft deflection analysis of the rotor was performed with the conservative SSE accelerations acting simultaneously. The deflections determined from the static shaft analysis were compared to the allowable impeller clearances.

The nature of seismic disturbances dictates that the maximum contact (if it occurs) be of short duration. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE and dynamic system loads are limited to the material elastic limit. The average membrane stress (s_m) for the faulted condition loads are maintained at 1.0S (s_y = yield stress) and the maximum stress in local fibers (s_m + bending stress = s_b) will be limited to 1.5S. The maximum seismic nozzle loads were also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

Performance of these analyses, with the conservative loads stated and with the restrictive stress limits as allowables, assured that critical parts of the pump will not be damaged during the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event.

If the natural frequency was found to be below 33 Hertz, an analysis was performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations were determined using the same conservatisms contained in the horizontal and vertical accelerations used for "rigid" structures. The static analysis was performed using the adjusted accelerations; the stress limits stated above were satisfied.

The second criterion necessary to assure operability is that the pump will function throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, will prevent the rotor from becoming seized. In actuality, the seismic 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 SSE and will operate at design speed despite the SSE loads.

In completing the seismic qualification procedures, the pump motor and all electrical components vital to the operation of the pump will be independently qualified for operation during the maximum seismic event in accordance with IEEE Standard 344-1975.

If the testing rather than the analysis option was chosen, sine beat testing for electrical equipment was performed by satisfying one or more of the following requirements to demonstrate that multifrequency

response is negligible, or the sine beat input is of sufficient magnitude to conservatively account for this effect.

a.   The equipment response is basically due to one mode.

b.   The sine beat response spectra envelopes the floor response spectra in the region of significant response.

c.   The floor response spectra consists of one dominant mode and has a peak at this frequency.

The degree of crosscoupling in the equipment, in general, determined if a single or multiaxis test was required. Multiaxis testing was required if there is considerable crosscoupling. If coupling is very light, then single axis testing was justified. Or, if the degree of coupling could be determined, then single axis testing was used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

From the previous arguments, the safety-related pump/motor assemblies were demonstrated to remain undamaged and operable under SSE loadings, and therefore perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

The functional capability of Class 2 and Class 3 active pumps (there are no Class 1 active pumps) after a faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. For Class 2 and Class 3, the faulted condition is greater than the normal condition only due to seismic SSE loads on the equipment itself and the increase in nozzle loads due to the SSE on the connecting pipe. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse than the normal plant operating limits. This is assured by requiring

that the imposed nozzle loads (steady-state loads) for normal conditions and postulated conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted capability of the pumps to function under these applied loads is proven during normal operating plant conditions for active pumps.

3.9.3.2.4.1.1   Emergency Service Water (ESW) Pumps Analysis Summary

A seismic stress and deflection analysis of the larger ESW pumps and the small ESW pump was performed.

The natural frequencies of the pumps were determined by developing a lumped mass model of the pump. The computer code ICES-STRUDL was used to perform the frequency analysis.

After the frequencies were determined, the lateral seismic loads were obtained from the response spectra curve furnished and noted in the design specification. The maximum seismic loads within plus or minus 10 percent of the frequencies were used for the seismic analysis. The vertical frequency was rigid; therefore, a conservative static g-load was applied in the vertical direction. The responses due to the vertical and both horizontal earthquake components were combined as the square root of the sum of the squares (SRSS). A dynamic model and static analysis of the pump and motor assemblies were made using the same model as used for the frequency analysis. Computer output from ICES-STRUDL included all forces, moments and deflections at all joints for each member of the model.

The normal operating loads were combined with the seismic loads for the vertical and lateral directions. This combination was made for the faulted loading condition and held to normal allowables. The resultant deflections were compared to operating clearances or other operating criteria.

A summary of the deflections and stresses is given in <Table 3.9-24> and <Table 3.9-25>.

Additional details of the analysis and the certification statements assuring operability can be found in McDonald Engineering Analysis Report Numbers ME-453 (Reference 11a) and supplemental Enertech Design Report Number MA21099 (Reference 11b) for the ESW "C" pump. Additional details of the analysis and the certification statements assuring operability can be found in McDonald Engineering Analysis Report Number ME-454 (Reference 12a) and supplemental Enertech Design Report Number MA21101 (Reference 12b) for the ESW "A" and "B" pumps.

3.9.3.2.4.1.2   Emergency Service Water Screen Wash Pumps Analysis
                     Summary

A seismic stress and deflection analysis of the ESW screen pumps was performed. The natural frequencies of the pumps were determined by developing a lumped mass model of the pump. The computer code ICES-STRUDL was used to perform the frequency analysis.

Once the frequencies were determined, the lateral seismic loads were obtained from the response spectra curve furnished and noted in the design specification. The maximum seismic loads within plus or minus 10 percent of the frequencies were used for the seismic analysis. The vertical frequency was rigid; therefore, a conservative static g-load was applied in the vertical direction. The responses due to the vertical and both horizontal earthquake components were combined as the square root of the sum of the squares (SRSS). Computer output from ICES-STRUDL included all forces, moments and deflections at all joints for each member of the model.

The normal operating loads were combined with the seismic loads for the vertical and lateral directions. This combination was made for the faulted loading condition and held to normal allowables. The resulting deflections were compared to operating clearances or other operating criteria.

A summary of the deflections and stresses is given in <Table 3.9-26>.

Additional details of the analysis and the certification statement assuring operability can be found in McDonald Engineering Analysis Report Number ME-452 (Reference 13).

3.9.3.2.4.1.3   Water Leg Pumps Analysis Summary

A seismic stress and deflection analysis of the RHR, LPCS, HPCS, and RCIC water leg pumps was performed. The original RHR, HPCS and RCIC motors have been replaced and have been qualified by manufacturer testing in accordance with IEEE 344-1975 and IEEE 323-1974.

The original pump and motor combination were modeled as lumped masses on their associated flexible support to determine whether it is in the rigid or flexible category. The analyses (and testing for replacement RHR, HPCS and RCIC motors) indicated that the pumps, (original and replacement) motors and shafts each have a natural frequency greater than 33 Hertz, and are therefore treated as rigid and analyzed accordingly <Section 3.7.3.4>. The results of these analyses indicated that static analysis is suitable for this equipment.

The pump-motor structural bases were analyzed considering all external loads imposed on them. Replacement RHR, HPCS, and RCIC motors were seismically qualified by testing, which included performance of 5 OBE and 2 SSE tests and function testing following the seismic tests, which resulted in no degradation to the motors. Replacement RHR, HPCS, and RCIC motors demonstrated capability to run on demand during hydrodynamic and seismic testing and maintained structural and electrical integrity throughout the testing, with no seismic anomalies identified. Piping loads were applied to the nozzles and transmitted through the structure. Seismic loads were obtained using the response spectra curve furnished in the design specification. Suction and discharge nozzle stresses were calculated in accordance with the ASME Code.

Resultant actual and allowable stresses are listed in <Table 3.9-27> with updated actual stress values due to the change in weight of the replacement RHR, HPCS, and RCIC motors.

The pumps were analyzed for operability in a seismic event by first determining the shaft critical frequency to assure that it is well displaced from the running frequency or critical seismic frequency. Additionally, the maximum shaft deflection and pump case distortion due to operating, external and seismic loads, as well as the maximum misalignment that might reduce the running clearances and result in pump seizure, were determined.

The analysis in Report Number 1A015/22 (Reference 14) by Van Gulik and Associates, Inc., assures operability by determining that at no time during operation will any contact occur.

3.9.3.2.4.1.4   Diesel Generator Fuel Oil Transfer Pumps Analysis Summary

A seismic stress and deflection analysis of the standby diesel generator and HPCS diesel generator fuel oil transfer pumps was performed.

A dynamic model was developed and a computer frequency analysis performed to obtain natural frequencies of the assembly, as required by the specification.

The lowest natural frequency of the pump system, including bed plate, is 42 Hertz.

The nozzle loads and seismic loads were imposed on the computer model of the assembly and the resultant stresses and deflections were calculated. The stresses were then compared to allowables given in the specifications. The deflections were compared to operating clearances or other limiting criteria.

The nozzles were analyzed for the maximum nozzle loads. The equivalent pressure caused by nozzle bending moments and axial loads was calculated in accordance with the ASME Code including ASME Code Case 1677 and imposed on the flanges.

This pump casing is of complex geometry and has been well verified for normal operation by service experience and hydrostatic tests. The seismic nozzle loads impose negligible stress in the casing except at nozzle penetrations and the frame adapter flange, which have been analyzed.

Actual and allowable stresses are given in <Table 3.9-36>.

Since the faulted actual stresses, deflections and loads are less than code allowable, only the faulted case calculations are given. Results of the analysis show that the pumps are structurally adequate to withstand the specified loads and will perform their intended function during upset, emergency and faulted conditions. Further details of the analysis can be found in McDonald Engineering Analysis, Inc. Report Number ME-532 (Reference 15).

3.9.3.2.4.1.5   Control Complex Chilled Water Pumps and Emergency Closed Cooling Pumps Analysis Summary

A seismic stress and deflection analysis of the control complex chilled water pumps and emergency closed cooling pumps was performed.

The fundamental frequency of the pumps/motors was demonstrated to be above the rigid frequency by analysis. Accordingly, a static analysis of stress and deflection as required by the design specification was performed.

Results of the stress analysis are shown in <Table 3.9-28>. Calculated deflections were less than allowable, indicating no possible interference would impair operability during or after an event.

Details of the analysis and the certification statement assuring operability can be found in Ingersol-Rand Company's Report Number EAS-TR-7807 IMR (Reference 16).

3.9.3.2.4.1.6   Fuel Pool Circulating Pumps Analysis Summary

A seismic stress and deflection analysis of the fuel pool circulating pumps was performed.

A dynamic model of the pump/motor/baseplate system was developed and a computer frequency analysis made. The lowest frequency of the system determined by the analysis was above 35 Hertz. Therefore, the system can be treated statically. Using the response spectra curve and load and stress combinations noted in the design specification, the nozzle loads, seismic loads and normal loads are imposed upon a computer model and a stress and deflection analysis of the entire assembly made.

The resultant stresses were compared to the allowables given in the specification and governing codes. The deflections were compared to operating clearances or other limiting criteria.

The nozzle discontinuity stresses were calculated by the method described in the ASME Code, Paragraph ND-3652, where the pump casing/discharge nozzle intersection is treated as an equivalent tee in a conservative manner. The suction nozzle was treated as a curved elbow.

The discharge and suction flanges were treated by the method described in the ASME Code, Section ND-3647, which is for normal loads and external forces and moments caused by weight and temperature gradients. No known accepted method exists to treat the flanges for seismic forces and moments. Thus, in this analysis they were treated the same as deadweight and thermal loads, which is believed to be conservative.

A summary of the deflections and stresses is given in <Table 3.9-29>.

Additional details of the analysis and the certification statement assuring operability can be found in McDonald Engineering Analysis Co., Inc., Report Number ME-512 (Reference 17).

3.9.3.2.4.1.7   Drywell Purge Compressors Test Summary

The assurance of operability for the drywell purge compressors is
confirmed by prototype test. A drywell purge compressor identical in
design and manufacture to the units supplied for the Perry Nuclear Power
Plant was tested by Turbonetics, Inc. The test conditions for the
prototype unit were more severe than the expected worst case conditions
to be experienced by the units installed in the Perry Nuclear Power
Plant and defined by the seismic response spectra curves included in the
design specification data sheets. The prototype test program was
carried out for a drywell purge compressor to be supplied for the Grand
Gulf Nuclear Station. Results and details of the test program are
documented in Turbonetics, Inc., Test Reports TB1-77-TR-1
(Reference 18), -3 (Reference 19), -4 (Reference 20), and
-5 (Reference 21). The test results indicate that the equipment will
perform the intended safety function during and after a LOCA or seismic
event.

3.9.3.2.4.2   Valves

The non-NSSS safety-related active valves are identified in design basis
documents/quality assurance records and databases. These sources
provide information relevant to valve safety classification, valve
electrical, mechanical and seismic functions, USAR Figure, valve size
and type, valve operator type, active function and qualification
summary.

Safety-related active valves are designed to perform their mechanical
motion during an accident condition. The operability assurance program
ensures that these valves will operate both during and after a seismic
event.

The safety-related active valves are subjected to a series of stringent
tests prior to service and during the plant life. Prior to

installation, the tests performed include: Shell hydrostatic test to
ASME Section III requirements, backseat and main seat leakage tests,
disc hydrostatic tests, functional tests to verify that the valve will
open and close within the specified time limits when subjected to the

design differential pressure, and operability qualification of valve actuators for the environmental conditions over the installed life.

Cold hydro qualification tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operation are performed onsite to verify and assure the functional capability of the valves. These tests and appropriate maintenance assure operability of the valves for the design life of the plant. If not qualified by test, a stress analysis of the extended structure of active valves was performed for static equivalent seismic SSE loads applied to the center of gravity of the extended structure. The maximum stress limits allowed in these analyses were those recommended by the ASME Code for the particular ASME class of valve analyzed.

Included in the design specification data sheets are seismic acceleration levels, allowable stresses and load combinations as discussed in <Section 3.9.3.1>. The design specifications required a certification of operability assurance.

Assurance of operability verification and certification for the active valves under all plant loading combinations defined in <Regulatory Guide 1.48> was provided by the valve manufacturer. Assurance of operability was demonstrated by either test or analysis. The method of test to verify operability considered the structural interaction of the entire assembly (valve and actuator). Testing was in accordance with one of the following procedures (A, B, C, or D):

Test Procedure A:

a.   The valve was installed in a test fixture and then mounted to a biaxial seismic vibration facility with the horizontal input motion in the direction of flow.

b.   The valve was pressurized to the design pressure before and during the test.

c.   Accelerometers were mounted on the valves and test table to adequately monitor input on the horizontal and vertical axes. Output of all accelerometers was displayed on oscillographic recorders.

d.   The valve was operated from the fully open to the fully closed position to demonstrate operability before the seismic test.

e.   A resonance search, from 1 to 33 Hertz at 0.3 g peak vertical and horizontal acceleration sequentially, at a sweep rate of 1 octave/minute was examined for evidence of resonance of the valve assembly (minimum of 3 to 1 amplification over input level).

f.   A biaxial sine beat vibration test equivalent to 10-cycle sine beats at 3.0 g input acceleration, at 1, 5, 10, 15, 20, 25, 30, and 33 Hertz, and at any resonant frequency, was performed. Reduced acceleration levels were acceptable at low frequencies (1, 5 and 10 Hertz) providing the levels used were equal to the maximum shaker table capability. The valve was cycled from the fully open to the fully closed position during this test. Sufficient delay between beats was allowed to preclude significant superposition of motion.

g.   Following completion of the uniaxial resonance searches and biaxial sine beat vibration test, the test item was rotated 90° on the test table and the resonance searches and sine beat vibration tests described above were repeated.

h.   The valve subjected to operability testing was subsequently tested to demonstrate proper operation, by being cycled to the fully open

and fully closed positions without exceeding the normal operating
load of the valve actuator.

Test Procedure B:

A prototype valve, either full or reduced scale, was bench tested to
demonstrate valve opening, or closing, or both under conditions
equivalent to the combined plant conditions which the valve is expected
to withstand when the "active" function is required.

Test Procedure C:

A prototype valve, either full or reduced scale, was bench tested under
test conditions which simulate separately each of the plant loadings
which the valve is expected to withstand in combination during valve
opening, or closing, or both. This test program was supplemented by
analyses to demonstrate that the individual test loadings were
sufficiently higher than the plant loadings to provide adequate margins
for assurance of operability under combined loading conditions. In
addition, the analyses demonstrated that the strains in critical
component parts of the valve under individual test loadings were
greater, by a substantial margin, than those which the valve may
experience under the combined plant loading conditions.

Test Procedure D:

For rigid valve assemblies (fundamental frequency greater than
33 Hertz), a static side load test was acceptable for verification and
certification of operability of active valves under all plant loading
combinations as defined in <Regulatory Guide 1.48>.

For rigid valve assemblies a static deflection analysis was performed by
applying static equivalent seismic loads to the center of gravity of the
extended structure. The calculated deflection was compared to an

allowable deflection, based on manufacturing tolerances, and shown to be small enough to not affect the active function of the valve.

Before manufacturing, the manufacturer had to describe the methods and procedures to verify operability of the valve under all design loading combinations, including a description of any mathematical models and test procedures. The final report, including final analyses and test results that demonstrated operability under all loading conditions, was provided prior to valve shipment.

The above referenced reports are maintained on file.

3.9.3.2.4.2.1   Valve Motor Operators

Motor operators and other electrical appurtenances necessary for valve operation were qualified as operable during SSE in accordance with the seismic qualification standard IEEE 344, prior to installation on the valve. The accelerations to which valves and valve operators were qualified have been factored into pipe systems design. The piping design will maintain motor operator accelerations below these levels.

3.9.3.2.4.2.2   Qualification of Check Valves

Operability of the check valves has been assured.

Because of the simplicity of the check valve design, an analysis was considered satisfactory to assure operability. The purpose of analysis was to demonstrate the seismic capability of various sizes of check valves, in accordance with specific seismic requirements as set forth by valve procurement specifications.

These requirements were as follows:

a. Each valve assembly shall withstand the inertial load caused by an operating basis earthquake (OBE) and safe shutdown earthquake (SSE) without causing failure of any part or change in integrity of any pressure-retaining part.

b. Each valve shall be designed to withstand vibratory motion in any direction due to a simultaneous seismic acceleration of 3.0g in two orthogonal horizontal and one vertical direction.

c. All seismic loads may be assumed to act at the center of gravity of the equipment.

d. The seismic analysis shall determine that the fundamental frequency of the valve is greater than 33 Hertz.

The certification of operability for check valves is provided in the vendor seismic/stress analysis reports for each valve procurement specification.

3.9.3.2.4.2.3   Non-NSSS Safety Class Safety and Relief Valves

The certification of operability and further details are provided in the vendor seismic qualification and assurance of operability reports.

3.9.3.2.4.2.4   Qualification of Butterfly Valves

Operability of active butterfly valves has been assured. The certification of operability and further details are provided in the vendor seismic qualification test and analysis reports.

3.9.3.2.4.2.5   Qualification of Gate and Globe Valves

Operability of active gate and globe valves has been assured. The certification of operability and further details are provided with the vendor seismic qualification reports for each valve equipment specification.

3.9.3.3   Design and Installation of Pressure Relief Devices

3.9.3.3.1   Main Steam Safety/Relief Valves and Discharge Piping Up to First Anchor

Main steam safety/relief valve (SRV) lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period from opening of the safety/relief valve until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the SRV cause the discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a stepwise 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 code flow rating increased by a factor to account for the method of establishing the rating. Simultaneous discharge of all valves was assumed in the analysis, to evaluate maximum stress in the piping. Reaction loads on the pipe were determined at each location corresponding to the position of an elbow. These loads were composed of pressure times area, momentum change and fluid friction terms.

<Figure 3.9-18> shows a pipe section load transient typical of that produced by relief valve discharge.

The method of analysis applied to determine piping system response is time history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resultant 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 ASME code stress limits corresponding to event classification (normal, upset, emergency, and faulted), are applied to the steam and discharge pipe.

3.9.3.3.2   Design and Installation Details for Mounting of Pressure Relief Devices

Design criteria used for safety and relief valves were in accordance with the requirements of Paragraphs NC and ND 3677 of Section III of the ASME Code and <Regulatory Guide 1.67>. Installation was classified as either "open relief system" or "closed relief system."

a.   Open Relief System

The steady-state load due to the steam reaction force from the opening of a relief valve included the sum of the momentum and pressure effects as follows:

$$F = WV_e / g_c + \left(P_e\right)A$$

where:

F = Reaction force (lb)

W = Mass flow rate (lbm/sec)

g_c = 32.2 (lbm-ft/lbf-sec^2)

V_e = Exit velocity (ft/sec)

P_e = Exit static pressure (lbf/in.^{2})

A = Exit flow area (in.^{2})

The forcing function was derived from the characteristics of the steam at the time the valve opens, and the characteristics of the valve and associated piping. To ensure consideration of the effects of the rapidly applied load on the valve nozzle and the header pipe, a dynamic load factor (DLF) was used. The DLF was determined analytically by considering valve opening time and valve/system dynamic characteristics. The results of stress analysis were used to ensure that the stresses at the valve installation were within the allowable limits of the ASME Code, Section III. Consideration was given to the nozzle/header junction and to the header pipe which was sufficiently reinforced, when necessary, to preclude the possibility of an overstress condition in the assembly.

Where more than one valve was installed on a common header, each valve was first considered to open independently, using the DLF for the valve and its support. Then the system was analyzed with all valves opening in the most severe combination.

ASME Code Class 2 and 3 piping components were evaluated by using equation (9) of Subsection NC/ND-3652.2 of the ASME Code, Section III. The moment term in this equation includes the moments resulting from the discharge thrust forces and earthquake loadings.

b.   SRV Discharge Piping from First Anchor to Suppression Pool

The closed discharge systems were analyzed for dynamic effects of the fluid discharge by using the time related forces from a thermal hydraulic response analysis as an input to a pipe stress time history analysis. The stresses resulting from these loads were

then combined as required in Subsections NC and ND 3652 of the ASME Code, Section III.

3.9.3.4   Component Supports

3.9.3.4.1   Piping Supports

Piping supports are designed in accordance with Subsection NF of the ASME Code, Section III. Supports are either designed by load rating per Paragraph NF-3260 or to the stress limits for linear supports per Paragraph NF-3230 and plate and shell per Paragraph NF-3220. To avoid buckling in the component supports, the allowable loads are limited to two thirds of the critical buckling loads per Appendices F and XVII of the ASME Code. The critical buckling loads for ASME Class I component supports in the NSSS scope, subjected to faulted loads which are more severe than normal, upset and emergency loads, are determined by the vendor using the methods discussed in Appendix F of the ASME Code. In general, the load combinations correspond to those used to design the supported pipe. Design loading combinations of BOP piping and component supports are presented in <Table 3.9-21a>. Design transient cyclic data are not applicable to piping supports because no fatigue evaluation is necessary to meet the code requirements.

The design criteria and dynamic testing requirements for component supports are given below:

a.   Component Supports

     All component supports are designed, fabricated and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed. All component supports are designed in accordance with the rules of Subsection NF

of the ASME Code and Code Case N-413, as applicable. Repair
welding of structural steel rolled shapes and plates may be in
accordance with Code Case N-393.

b.   Spring Hangers and Constant Support Hangers

The design load on hangers is the load caused by dead weight. The
hangers are calibrated to ensure that they support the design load
at both their hot and cold load settings. Hangers provide down
travel and up travel in excess of the specified thermal movement.

c.   Snubbers

1.   Required Load Capacity and Snubber Location

Class 1, 2 and 3 piping systems, including valves and the
suspension system between anchor points, are mathematically
modeled for structural analysis. In the mathematical model
for Class 1 Systems, the snubbers are modeled as springs with
a given spring stiffness depending on the snubber size. In
Class 2 and 3 Systems, snubbers are modeled as rigid supports
in seismic analysis. The analysis determines the forces and
moments acting on each piping component and the forces acting
on snubbers due to all dynamic loading conditions defined in
the piping design specification. The design load on snubbers
includes loads caused by seismic forces including hydrodynamic
forces (operating basis earthquake and safe shutdown
earthquake), system anchor movements, reaction forces caused
by relief valve discharge, and turbine stop valve closure.

Snubber placement and loading direction are first decided by
estimation so that stresses in the piping system will have
acceptable values. Snubber placement and loadings are refined
by performing computer analyses on the piping system as

described above. For Class 2 and 3 small bore safety-related piping systems less than 250°F, snubber placement and loadings have been determined by application of standard cold spacing design criteria which were developed to be consistent with applicable ASME Code and <Regulatory Guide 1.48> and <Regulatory Guide 1.84>.

The suspension design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

2.   Inspection, Testing, Repair, and/or Replacement of Snubbers

The suspension design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the handling, testing, storage, installation, maintenance, adjustment, and repair of the snubber. Each snubber also has an installation location drawing, which identifies pipe and structure location, hot and cold settings, and additional installation data.

The suspension design specification requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

Snubber examination and testing to be conducted in accordance with the requirements of the Operational Requirements Manual 6.4.1.

3.   Snubber Design and Testing

To assure that the required structural and mechanical
performance characteristics and product quality are achieved,
the following requirements for design and testing are imposed
by the design specification:

(a) The snubbers are designed in accordance with ASME III,
     Subsection NF. These design requirements include stress
     analysis of the snubber component parts under normal,
     upset, emergency, and faulted loads. Calculated stresses
     are then compared against allowable stresses of the
     material as given in ASME III, to ensure that they are
     below the allowable limit.

     NSSS bolting used in these supports meets criteria of
     NF-3280 for Service Levels A and B and NF-3230 for
     Service Levels C and D. NF-3280 is applicable to bolting
     for Service Levels A and B. NF-3230 is applicable to
     linear supports; it refers to Appendix 17 which is
     applicable to bolting for Service Levels C and D.

(b) The snubbers were tested to ensure that they can perform
     as required during the operating basis earthquake (OBE),
     the safe shutdown earthquake (SSE), hydrodynamic events,
     and under anticipated operational transient loads or
     other plant mechanical loads.

Two snubbers of each size and each model in safety-related applications at PNPP 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 conditions in the following manner:

    i. The snubbers were subjected to a force that varied approximately as a sine wave.

    ii. The frequency (Hz) of the input force was in increments of 5 Hz within the range of 3 to 33 Hz.

    iii. The test was conducted with the snubber at room temperature and at 200°F.

    iv. The peak load in both tension and compression was equal to or higher than the rated load of the snubbers.

    v. The duration of the test at each frequency was 10 seconds 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:

    i. The snubbers were subject to forces that varied approximately as a sine wave.

    ii. The test was conducted with the snubbers at room temperature.

iii. The peak load in both tension and compression was equal to 1.5 times the rated load of the snubbers.

iv. The duration of the test was 10 seconds.

Snubbers were qualified for service by General Electric by testing for bleed rate, lockup rate, drag or friction force, and response to dynamic loading. The dynamic loading test was accomplished by subjecting the snubber to a sinusoidal force that is equal to the rated load of the snubber. The force was applied at frequencies at 5 Hz increments within the range of 3 Hz to 33 Hz. The dynamic load tests were conducted with the snubber at both room temperature and at 200°F.

The snubbers were modeled as linear elastic springs in the dynamic analysis of the piping system. The vast majority of all dynamic loadings occur with frequencies ranging from 3 Hz to 33 Hz. By using the results of the dynamic testing, spring constants were calculated. These constants increase with higher frequencies. The average spring constant, including all lost motions (dead band, etc.) of the snubber, was then used by General Electric in the analytical model of attached piping.

d. Struts

The design load on struts includes loads caused by dead weight, thermal expansion, operating basis earthquake (OBE) and safe shutdown earthquake (SSE), hydrodynamic loads, system anchor

displacements, reaction forces caused by relief valve discharge, and turbine stop valve closure.

Struts are designed in accordance with the ASME Code, Paragraph NF-3000 to be capable of carrying the design load for all operating conditions.

e. Bolting

The following equipment involves component support bolting.

(a) RWCU Pump

This non-safety pump has no active safety function. The suction and discharge piping from the pump ultimately connects to safety related piping. The pump flange allowable nozzle reactions are developed with the guidance of the ASME Code, Section III, subsection NC/ND 3658, July 1983. The load combination for the flange qualification includes the deadweight and safe shutdown earthquake reactions imposed by the piping. The pump anchorage acceptance criteria is ACI-349-85 Appendix B.

(b) RCIC and SLC Pumps

The pump-to-base plate bolting is designed as follows:

1. Normal plus Upset

    a) Primary membrane: 1.0S

b)   Primary membrane plus bending: 1.5S, where S is the allowable stress limit per the ASME Code Section III, Appendix I, Table I-7.3

2.   Emergency or Faulted

Stresses shall be less than 1.2 times the allowable limit
values for Normal plus Upset above.

3.9.3.4.2   Reactor Pressure Vessel (RPV) Support Skirt

The RPV support skirt is analyzed to meet the requirements of the ASME
Code, Section III, Subsection NF. The loading conditions, stress
criteria, calculated stresses, and allowable stresses in the critical
support areas are summarized in <Table 3.9-3a>. The stress level
margins prove the adequacy of the RPV support skirt.

3.9.3.4.3   NSSS Floor Mounted Equipment (Pumps, Heat Exchanger and
              RCIC Turbine)

The high pressure core spray, low pressure core spray, residual heat
removal, reactor core isolation cooling, standby liquid control, reactor
water cleanup, and RCIC systems are all analyzed to verify the adequacy
of their support structures under various plant operating conditions.
In all cases, the stress loads 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
<Table 3.9-3> under the respective equipment table.

3.9.3.4.4   Supports for ASME Code Class 1, 2 and 3 Active Components

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 as follows:

a. Simulate actual pump mounting conditions and inservice static and dynamic loadings on the pump.

b. Monitor pump operability during testing.

c. Verify normal operation of the pump during and after the test. Any deflection or deformation of the pump supports which precludes pump operability is not accepted.

d. Inspect supports for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Seismic qualification of component supports by analysis has generally been accomplished by checking stresses at all support elements and parts such as baseplate holddown bolts, pump support pads, pump pedestal, and foundation against allowable limits specified in the ASME Code, Subsection NF.

a. For normal and upset plant conditions, the deflections and deformations of the supports are verified to be within elastic limits and not in excess of design values based on design verification tests to ensure operability of the pumps.

b. For emergency and faulted plant conditions, the deformations do not exceed the values which ensure operability of the pumps.

3.9.3.4.5   Fuel Pool Cooling and Cleanup System Pump

The pump pedestal and pedestal bolts have been analyzed as discussed in <Section 3.9.3.1>. Loads from seismic, connecting pipes, temperature, and dead weight were considered. The stress limits of the ASME Code,

Section III, Subsection NF were satisfied. The analysis included deflection of the pedestal.

3.9.3.4.6   Balance of Plant Equipment Component Supports

The loadings as specified in the design specifications were taken into account in designing component supports for ASME Code constructed items. These loadings include, but were not limited to, the following:

a.   Weight of the component and normal contents under operating and test conditions

b.   Weight of the component support

c.   Superimposed loads and reactions induced by the adjacent system components

d.   Dynamic loads, including loads caused by earthquake vibration and hydrodynamically induced building vibrations

e.   Restrained thermal expansion

f.   Anchor and support movement effects

Design loading combinations of BOP ASME Code Class 1, 2 and 3 component supports are presented in <Table 3.9-21a>.

3.9.4   CONTROL ROD DRIVE SYSTEM

This plant is equipped with a hydraulic control rod drive which includes the control rod drive mechanism (CRDM), the hydraulic control unit (HCU), the condensate supply system, and the scram discharge volume, extending to the coupling interface with the control rods.

3.9.4.1   Descriptive Information on CRDS

Descriptive information on the control rod drives, as well as the entire
control and drive system, is contained in <Section 4.6>.

3.9.4.2   Applicable CRDS Design Specifications

The Control Rod Drive System (CRDS) is designed to meet the functional
design criteria outlined in <Section 4.6> and consists of the following:

a.   Locking piston control rod drive

b.   Hydraulic control unit

c.   Hydraulic power supply (pumps)

d.   Interconnecting piping

e.   Flow, pressure and isolation valves

f.   Instrumentation and electrical controls

Quality group classification is not applicable to the CRD.

Those components of the CRD forming part of the primary pressure
boundary are designed according to the ASME Code, Section III.

Components of the CRD hydraulic system are designed according to the
codes and standards governing the individual quality groups outlined in
<Table 3.2-1>.

Pertinent aspects of the design and qualification of the CRD components
are discussed in the following locations: transients in

<Section 3.9.1.1>, faulted conditions in <Section 3.9.1.4>, seismic testing in <Section 3.9.2.2>.

3.9.4.3   Design Loads, Stress Limits and Allowable Deformations

The ASME Code components of the CRD system have been evaluated analytically and the design loading conditions, stress criteria, calculated stresses, and allowable stresses are summarized in <Table 3.9-3>. 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 has been compared with the allowables and is not a limiting factor in the analysis of the CRD components.

3.9.4.4   CRD Performance Assurance Program

The CRD test program consists of the following:

a.   Development tests

b.   Factory quality control tests

c.   Five year maintenance life tests

     Four control rod drives are normally picked at random from the
     production stock each year and subjected to various tests under
     simulated reactor conditions and 1/6 of the cycles specified in
     <Section 3.9.1.1>.

     Upon completion of the test program, control rod drives must meet
     or surpass the minimum specified performance requirements.

d.   1.5X design life tests

     When a significant design change is made to the components of the
     drive, the drive is subjected to a series of tests equivalent to
     1.5 times the life test cycles specified in <Section 3.9.1.1>.

e.   Operational tests

f.   Acceptance tests

g.   Surveillance tests

All of the above tests except c. and d. are discussed in <Section 4.6>.

3.9.5   REACTOR PRESSURE VESSEL INTERNALS

This subsection identifies and discusses the structural and functional
integrity of the major reactor pressure vessel internals.

3.9.5.1   Design Arrangements

The core support structures and reactor vessel internals (exclusive of
fuel, control rods and incore nuclear instrumentation) are identified
below:

a.   Core Support Structures

     1.   Shroud

     2.   Shroud support cylinder, plate and legs (part of the RPV core
          plate) and core plate hardware

3. Grid (only that portion below the bottom weld in the cylindrical portion is core support structure. The grid is a part of the top guide assembly.)

4. Top guide (hardware, studs, nuts, and pins between top guide and shroud)

5. Orificed fuel supports (except for orifices which do not support or restrain the core)

6. CRD housing (only that portion above the first weld that is above the housing to pressure vessel weld)

7. Control rod guide tubes

b. Reactor Internals

1. Jet pump assemblies, braces and instrumentation

2. Feedwater spargers[1]

3. Vessel head spray nozzle

4. Differential pressure and liquid control lines

5. In-core flux monitor guide tube[1]

6. Initial startup neutron sources[1]

7. Surveillance sample holders[1]

8. Core spray lines and spargers

9.   In-core instrument housings[1]

10.   LPCI Coupling

NOTE:
[1] Nonsafety class component.

A general assembly drawing of the important reactor components is shown in <Figure 3.9-19>.

The floodable inner volume of the reactor pressure vessel can be seen in <Figure 3.9-20>. 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 tubes, is such that one end is unrestricted and thus free to expand. The LPCI couplings incorporate vertically oriented slip fit joints to allow free thermal expansion.

3.9.5.1.1   Core Support Structures

The core support structures 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 3.9-20> shows the reactor vessel internal flow paths.

a.   Shroud

      The shroud support, shroud and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from

the downcomer annulus, thus providing a floodable region following
a recirculation line break. The volume enclosed by this assembly
is characterized by three regions. The upper portion surrounds the
core discharge plenum, which is bounded by the shroud head on top
and the top guide's grid plate below. The central portion of the
shroud surrounds the active fuel and forms the longest section of
the assembly. This section is bounded at the top by the grid plate
and at the bottom by the core plate. The lower portion,
surrounding part of the lower plenum, is welded to the reactor
pressure vessel shroud support.

b.   Shroud Support

The shroud support is designed to support the shroud and to support
and locate the jet pumps. The shroud support provides an annular
baffle between the reactor pressure vessel and the shroud. The jet
pump discharge diffusers penetrate the shroud support to introduce
coolant to the inlet plenum below the core.

c.   Shroud Head and Steam Separator Assembly

This component is not a core support structure or safety class
component. It is discussed here to describe the coolant flow paths
in the reactor pressure vessel. The shroud head and steam
separator assembly is bolted to the top of the top guide to form
the top of the core discharge plenum. This plenum provides a
mixing chamber for the steam-water mixture before it enters the
steam separators. Individual stainless steel axial flow steam
separators are attached to the top of standpipes that are welded
into the shroud head. The steam separators have no moving parts.
In each separator, the steam-water mixture rising through the
standpipe passes vanes that impart a spin to establish a vortex

separating the water from the steam. The separated water flows
from the lower portion of the steam separator into the downcomer
annulus.

d.   Core Plate

The core plate consists of a circular stainless steel plate with
bored holes stiffened with a rim and beam structure. The plate
provides lateral support and guidance for the control rod guide
tubes, in-core flux monitor guide tubes, peripheral fuel supports,
and startup neutron sources. The last two items are also supported
vertically by the core support plate.

The entire assembly is bolted to a support ledge on the lower
portions of the shroud.

e.   Top Guide

The top guide consists of a circular grid plate with square
openings welded to the bottom of the top guide cylinder. Each
opening provides lateral support and guidance for four fuel
assemblies, or in the case of peripheral fuel, less than four fuel
assemblies. Notches are provided in the bottom of the
intersections to anchor the in-core flux monitors and startup
neutron sources. The top guide is bolted to the shroud. The core
spray spargers are installed in the upper portion of the top guide
cylinder.

f.   Fuel Support

The fuel supports, shown in <Figure 3.9-21>, are of two basic
types: 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 will support 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
will support 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 plate. The control rods pass
through slots in the center of the four-lobed orificed fuel
support. A control rod and the four adjacent fuel assemblies
represent a core cell <Section 4.4>.

g.   Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from
the top of the control rod drive housings up through holes in the
core plate. Each tube is designed as the guide for a control rod
and as the vertical support for a four-lobed orificed fuel support
piece and the four fuel assemblies surrounding the control rod.
The bottom of the guide tube is supported by the control rod drive
housing, which in turn transmits the weight of the guide tube, fuel
support and fuel assemblies to the reactor vessel bottom head. A
thermal sleeve is inserted into the control rod drive housing from
below and is rotated to lock the control rod guide tube in place.
A key is inserted into a locking slot in the bottom of the control
rod drive housing to hold the thermal sleeve in position.

3.9.5.1.2   Reactor Internals

a.   Jet Pump Assemblies

The jet pump assemblies are not core support structures but are
discussed here to describe coolant flow paths in the vessel. The

jet pump assemblies are located in two semi-circular 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 (Reference 23) and (Reference 24). Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser <Figure 3.9-22>. 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 to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. 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 hold-down clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

b.   Steam Dryers

The steam dryer assembly is not a core support structure or safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe, below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly, which may occur under accident conditions, is restricted by steam dryer hold-down brackets attached to the reactor vessel top head.

c.   Feedwater Spargers

These components are not core support structures or safety class components. They are discussed here to describe flow paths in the vessel. 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.

d.   Core Spray Lines and Liquid Control Line

These components are not core support structures. They are discussed here to describe safety class features inside the reactor pressure vessel. The core spray lines are the means for directing flow to the core spray nozzles which distribute coolant during accident conditions. The core spray line associated with the high pressure core spray system also serves as the liquid control line for providing a path for liquid control solution injection.

Two core spray lines enter the reactor vessel through the two core spray nozzles. The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the top guide cylinder immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the top guide cylinder. The two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the top guide and vessel. The other core spray line is identical except that it enters the opposite side of the vessel, and the spargers are at a slightly different elevation inside the top guide cylinder. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers <Section 5.3.3.1>, <Section 6.3> and <Figure 3.9-19>. Use of the HPCS spray line for liquid control solution injection facilitates good mixing and dispersion.

e.   Vessel Head Spray Nozzle

This component is not a core support structure. It is included here to describe a safety class feature in the reactor pressure vessel. 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, helps 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 nozzle.

f.   Differential Pressure Sensing Lines

These components are not core support structures or safety class components. The differential pressure lines enter the vessel through two bottom head penetrations and sense the differential pressure across the core support plate. One line terminates near the lower shroud with a perforated length below the core support plate to sense the pressure in that region.

The other line terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

g.   In-Core Flux Monitor Guide Tubes

This component is not a core support structure or safety class component. They 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 in the lower plenum to the top of the core support plate. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring and intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes. A latticework of clamps, tie bars and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

h.   Surveillance Sample Holders

This component is not a core support structure or a safety class
component. The surveillance sample holders are welded baskets
containing impact and tensile specimen capsules. 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.

i.   Low Pressure Coolant Injection (LPCI) Lines

This component is not a core support structure but is discussed
here to describe the coolant flow paths in the reactor vessel.
Three LPCI lines penetrate the core shroud through separate LPCI
nozzles. Coolant is discharged inside the core shroud.

3.9.5.2   Loading Conditions

3.9.5.2.1   Events to be Evaluated

Examination of the spectrum of conditions for which the design basis
must be satisfied by core support structures and engineered safety
feature components reveals the following significant faulted events:

a.   Recirculation Line Break: a break in a recirculation line between
     the reactor vessel and the recirculation pump suction.

b.   Steam Line Break: a break in one main steam line between the
     reactor vessel and the flow restrictor. The accident results in
     significant pressure differentials across some of the structures
     within the reactor.

c.   Earthquake: subjects the core support structures and reactor
      internals to significant forces as a result of ground motion.

d.   Safety/Relief Valve Discharge

Analysis of other conditions existing during normal operation, abnormal
operating transients and accidents shows that the loads affecting the
core support structures and other engineered safety feature reactor
internals are less severe than those caused by the above postulated
events. The faulted conditions for the reactor pressure vessel
internals including core support structures are discussed in
<Section 3.9.1.4>. Loading conditions, stress criteria, calculated
stresses, and allowable stresses are summarized in <Table 3.9-3a>,
<Table 3.9-3b>, <Table 3.9-3v>, <Table 3.9-3aa>, <Table 3.9-3ab>, and
<Table 3.9-3ac>.

The core support structures are designed in accordance with ASME Code,
Section III, Subsection NG.

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 3.9-23>
shows the nine reactor nodes. The computer code used is the General
Electric Short Term Thermal-Hydraulic Model described in (Reference 25).
This model has been approved for use in ECCS conformance evaluations
under <10 CFR 50, Appendix K>. In order to adequately describe the
blowdown pressure effects 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 25). These additional features are discussed below:

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 modelled 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.

c.   The enthalpies in the guide tubes and the bypass are calculated separately, since the fuel assembly pressure drop 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 Breaks

a.   Accident Definition

     Both a recirculation line break (the largest liquid break) and an inside steam line break (the largest steam break) were considered in determining the design basis accident for the engineered safety feature reactor internals. The recirculation line break is the same as the design basis loss-of-coolant accident described in <Section 6.3>. 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. Therefore, the steam

line break is the design basis accident for internal pressure differentials.

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 than any other line break evaluated.

b.   Effects of Initial Reactor Power and Core Flow

For analysis, the maximum internal pressure loads are divided into two parts: steady-state and transient pressure differentials. For a given plant, the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. On the other hand, the core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and consequently the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the reactor pressure vessel and thus the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that the calculated pressure differences bound those which could be expected if a steam line break should occur, an analysis has been 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 was 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). <Table 3.9-31> summarizes resulting maximum pressure differentials based on a GE12 full core configuration. [Analysis contained in Global Nuclear Fuels, DRF J11-03754-00, reviewed the application of GE14 and GNF2 fuel to Perry and concluded GE12 (with debris filters) fuel analysis bounds GE14 and GNF2 (with debris filters) fuel.]

Tabulated conditions maximize those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, the reactor will generally operate at or near full power, and second, 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>, <Section 3.8> and <Section 3.9.2.5>. Dynamic analysis is performed by coupling the mathematical model of the reactor vessel and internals, with the building model, to determine the time histories of reactor vessel and internals accelerations, forces and moments. This is done using the modal superposition method. Acceleration response spectra are also generated for subsystem analyses of selected components.

3.9.5.3   Design Bases

3.9.5.3.1   Safety Design Bases

The reactor core support structures and internals shall meet the
following safety design bases:

a.   They shall be arranged to provide a floodable volume in which the
      core can be adequately cooled in the event of a breach in the
      nuclear system process barrier external to the reactor vessel.

b.   Deformation shall be limited to assure that the control rods and
      core standby cooling systems can perform their safety functions.

c.   Mechanical design of applicable structures shall assure that safety
      design bases a and b (above) are satisfied so that the safe
      shutdown of the plant and removal of decay heat are not impaired.

3.9.5.3.2   Power Generation Design Bases

The reactor core support structures and internals shall be designed to
the following power generation design bases:

a.   They shall provide the proper coolant distribution during all
      anticipated normal operating conditions up to full power operation
      of the core without fuel damage.

b.   They shall be arranged to facilitate refueling operations.

c.   They shall be designed to facilitate inspection.

3.9.5.3.3   Design Loading Categories

The basis for determining faulted loads on the reactor internals is
shown for seismic and hydrodynamic loads in <Section 3.7>, <Section 3.8>
and <Section 3.9.2.5> and for pipe rupture loads in <Section 3.9.5.2.3>
and <Section 3.9.5.3.4>. The loading conditions for the shroud support,
core support structures, CRD and housing, jet pumps, LPCI coupling, CR
guide tube in-core housing, and orificed fuel supports are given in
<Table 3.9-3> under the respective equipment table. The core support
structures which are fabricated as part of the reactor pressure vessel
assembly are discussed in <Section 3.9.3.1.1.e>.

Core support structure and safety class internals stress limits are
consistent with ASME Code, Section III, "Categorization of Loading
Conditions" (NA-2140) and associated stress limits contained in Addenda
dated through Summer 1976. Levels A, B, C, and D service limits defined
in the Winter 1976 Addenda which replace normal, upset, emergency, and
faulted condition limits are not reflected in design documents for core
support structures and other safety class internals for this reactor.
However, for these components levels A, B, C, and D service limits are
judged to be equivalent to the normal, upset, emergency, and faulted
loading conditions limits. Therefore, for clarity, both sets of
nomenclature are retained herein.

Stress intensity and other design limits are discussed in
<Section 3.9.5.3.5>.

The design requirements for equipment classified as "other internals,"
e.g., steam dryer and shroud head, were specified by the designer with
appropriate consideration of the intended service of the equipment and
expected plant and environmental conditions under which it will operate.
Where possible, design requirements are based on applicable industry
codes and standards. If these are not available, the designer relies on
accepted industry or engineering practices.

3.9.5.3.4   Response of Internals Due to Inside Steam Line Break
                 Accident

The maximum pressure loads acting on the reactor internal components
result from an inside steam line break, and on some components the loads
are maximum core flow <Table 3.9-31>, (Case 2). This has been
substantiated by the analytical comparison of liquid versus steam breaks
and by the investigation of the effects of core power and core flow.

It has also been pointed out that it is possible but not probable that
the reactor would be operating at the rather abnormal condition of
minimum power and maximum core flow. More realistically, the reactor
would be at or near a full power condition, and thus the maximum
pressure loads acting on the internal components would be as listed
under Case 1 in <Table 3.9-31>.

3.9.5.3.5   Stress, Deformation and Fatigue Limits for Engineered
                 Safety Feature Reactor Internals (Except Core Support
                 Structure)

The stress deformation and fatigue criteria listed in <Table 3.9-32>,
<Table 3.9-33> and <Table 3.9-34>, are used, or design criteria are
based on applicable codes and standards for similar equipment, by
manufacturers' standards, or by empirical methods based on field
experience and testing. For the quantity SF_{min} (minimum safety factor)
appearing in those tables, the following values were used:

Service Level	Design Condition	SF_{min}
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Faulted	1.125

The design fatigue usage factor is limited to 1.0 for service levels A and B (normal and upset) design conditions in accordance with the ASME Code.

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined for adequate clearances during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in <Section 3.9.2.5>.

3.9.5.3.6   Stress and Fatigue Limits for Core Support Structures

The stress, fatigue and other limits for the core support structures are in accordance with ASME Code, Section III, Subsection NG, and are summarized in <Table 3.9-3b>.

3.9.5.3.7   Special Requirements for Jet Pump Holddown Beams

Regarding jet pump holddown beam integrity, the following augmented design and inspection criteria are applied.

a.   The jet pump preload will be reduced from 30,000 lbs to 25,000 lbs.

b.   The jet pump beams will be inspected for crack initiation using ultrasonic techniques.

c.   Group 2 Jet Pump Holddown Beams will be inspected after the first twelve (12) years of operation, with subsequent inspections not to exceed every ten (10) years with Hydrogen Water Chemistry (HWC) or every six (6) years with Normal Water Chemistry (NWC).

d.   Group 3 Jet Pump Holddown Beams will be inspected after the first twenty (20) years of operation, with subsequent inspections not to exceed every twelve (12) years with Hydrogen Water Chemistry (HWC)

or every eight (8) years with Normal Water Chemistry (NWC).
Group 3 Jet Pump Holddown Beams were evaluated for installation
under Engineering Change Package (ECP) 07-0022.

e.   The inspection plan is based on BWR Vessel and Internals
      Project (BWRVIP) documentation (Reference 28).

f.   Any cracked beam found during inspections will be replaced. It is
      expected that any replacements would be beams with a reduced
      tendency for cracking.

3.9.6   INSERVICE TESTING OF PUMPS AND VALVES

Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and
valves will be performed in accordance with the ASME Code and applicable
Addenda as required by <10 CFR 50.55a(f)>, except where specific written
relief has been granted by the NRC pursuant to <10 CFR 50.55a(f)(6)(i)>.
For additional discussion of the inservice inspection program, see
<Section 5.2.4> and <Section 6.6>. The parameters to be measured or
observed are discussed and defined in a separate inservice inspection
program.

3.9.6.1   Inservice Testing of Pumps

Pumps will be tested according to the requirements of the ASME Code for
Operation and Maintenance of Nuclear Power Plants (ASME OM Code).

3.9.6.2   Inservice Testing of Valves

Valves are tested according to the requirements of the ASME Code for
Operation and Maintenance of Nuclear Power Plants (ASME OM Code).

3.9.7   REFERENCES FOR SECTION 3.9

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3.	Wilson, E. L., "A Digital Computer Program for the Finite Element Analysis of Solids with Nonlinear Material Properties," Aerojet

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5.   "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P, September 1976.

6.   "BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-P, November 1976.

7.   American National Standards Institute, "Power Piping Code," ANSI B31.1.

8.   Structural Dynamics Research Corporation, Proposal No. 6017-11.

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16. Ingersoll-Rand Company, Report Number EAS-TR-7807-IMR Rev. 1, Structural Integrity and Operability Analysis of 8 x 14 SD Pump, July 12, 1978.

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20. Turbonetics, Inc., Report Number TBI-77TR-4, Post Seismic Inspection Data Base for Drywell Purge Compressor SC-6 Prototype, July 14, 1978.

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27. General Electric Company Report NEDC-32907P (Class III), "Safety Analysis Report for Perry 5% Thermal Power Uprate," September 1999.

28. BWR Vessel and Internals Project (BWRVIP), "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)."

TABLE 3.9-1

PLANT EVENTS

a. (Pressure/Temperature/Flow Transients)   No. of
    Normal, Upset and Testing Conditions   Cycles

    1.   Bolt up[1]   123

    2.   Design Hydrostatic Test   40

         a.   Leak Checks at 400 psig Prior to Power
              Operation, 3 Cycles/Startup

    3.   Startup (100°F/hr Heatup Rate)[2]   120

    4.   Daily Reduction to 75% Power[1]   10,000

    5.   Weekly Reduction 50% Power[1]   2,000

    6.   Control Rod Pattern Change[1]   400

    7.   Loss of Feedwater Heaters (80 Cycles Total)   80

    8.   OBE at Rated Operating Conditions[4]   10/50

    9.   Scram:

         a.   Turbine Generator Trip, Feedwater On,
              Isolation Valves Stay Open   40

         b.   Other Scrams   140

         c.   Loss of Feedwater Pumps, Isolation
              Valves Closed   20 |

         d.   Turbine Bypass, Single Safety or Relief
              Valve Blowdown   8

    10. Reduction to 0% Power, Hot Standby, Shutdown
         (100°F/hr Cooldown Rate)[2]   116 |

    11. Unbolt[1]   123

TABLE 3.9-1 (Continued)

b. (Pressure/Temperature/Flow Transients)   No. of
         Emergency Conditions   Cycles

    1. Scram:

        a. Reactor Overpressure with Delayed
            Scram, Feedwater Stays on,
            Isolation Valves Stay Open $1^{(3)}$

        b. Automatic Blowdown (ADS) $1^{(3)}$

    2. Improper Start of Cold Recirculation Loop $1^{(3)}$

    3. Sudden Start of Pump in Cold Recirculation Loop $1^{(3)}$

    4. Hot Standby with Reactor Drain Shut Off
        Followed by Pump Restart $1^{(3)}$

c. Faulted Condition

    1. Pipe Rupture and Blowdown $1^{(3)}$

    2. Safe Shutdown Earthquake during Refueling $1^{(3)}$

In addition to the above temperature/pressure/flow transients the
following dynamic load transients have been considered in the design
and/or fatigue evaluation:

d.	Dynamic/Transient Load	Category	Cycles/Events
1.	Operating Basis Earth-quake (OBE)[5]	Upset	10 cycles[4]
2.	Safe Shutdown Earth-quake (SSE)[6]	Faulted	1 cycle
3.	Turbine Stop Valve Closure (TSV)	Upset	690 cycles
4.	Safety/Relief Valve Actuation (Acoustic wave)[7]	Upset	5,460 cycles

TABLE 3.9-1 (Continued)

Dynamic/Transient Load	Category	Cycles/Events
5. Safety/Relief Valve Actuation	Upset	(a) Piping Analysis
		(i) All Valve Actuation[8] 660 full range cycles 880 half range cycles
		(ii) Single Valve Actuation 4,800 full range cycles 6,400 half range cycles
		(b) RPV & Internals Analysis 12,600 cycles
6. Loss-of-Coolant Accident (LOCA):		1 event
Small break LOCA	Emergency/ Faulted	
Intermediate break LOCA	Faulted	
Large break LOCA	Faulted	

NOTES:

[1] Applies to reactor pressure vessel only.

[2] Bulk average vessel coolant temperature change in any 1-hour period.

[3] The annual encounter probability of the one cycle event is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

[4] Fifty (50) peak OBE cycles for NSSS piping. Ten (10) peak OBE cycles for other NSSS equipment and components.

[5] One 50% SSE event includes 10 maximum load cycles.

[6] One stress reversal cycle of maximum seismic amplitude.

[7] Applicable to main steam piping system only (1,820 actuations with 3 acoustic cycles each).

[8] In total, 1,820 actuations are considered, out of which 220 are valve actuations and 1,600 are single valve actuations. Each actuation is accompanied by 7 cycles due to air bubble oscillation. For piping design, the first 3 of 7 cycles are considered for full pressure range; the remaining 4 cycles are considered half range due to the decaying amplitude.

TABLE 3.9-2

DESIGN TRANSIENTS CONSIDERED IN THE DESIGN OF
ASME CODE CLASS 1 COMPONENTS NOT SUPPLIED BY THE NSSS VENDOR

System	System Transient Curve
Feedwater	\<Figure 3.9-24 (1)\>
	\<Figure 3.9-24 (2)\>
Main Steam Drains	\<Figure 3.9-25 (1)\>
	\<Figure 3.9-25 (2)\>
Reactor Core Isolation Cooling	\<Figure 3.9-26 (1)\>
	\<Figure 3.9-26 (2)\>
	\<Figure 3.9-26 (3)\>
	\<Figure 3.9-26 (4)\>
Residual Heat Removal	\<Figure 3.9-27 (1)\>
	\<Figure 3.9-27 (2)\>
	\<Figure 3.9-27 (3)\>
	\<Figure 3.9-27 (4)\>
High Pressure Core Spray	\<Figure 3.9-28 (1)\>
	\<Figure 3.9-28 (2)\>
Low Pressure Core Spray	\<Figure 3.9-29 (1)\>
	\<Figure 3.9-29 (2)\>
Reactor Water Clean-up	\<Figure 3.9-30 (1)\>
	\<Figure 3.9-30 (2)\>
	\<Figure 3.9-30 (3)\>
	\<Figure 3.9-30 (4)\>

TABLE 3.9-3 INDEX

<u>LOADING COMBINATIONS, STRESS LIMITS AND ALLOWABLE STRESSES</u>

3.9-3   Design Loading Combinations for ASME Code Class 1, 2 and 3
          Components

a.   Reactor Pressure Vessel and Shroud Support Assembly

b.   Reactor Vessel Internals and Associated Equipment

c.   (Deleted)

d.   ASME Code Class 1 Main Steam Piping and Pipe-Mounted
      Equipment-Highest Stress Summary

e.   ASME Code Class 1 Recirculation Piping and Pipe-Mounted
      Equipment-Highest Stress Summary

f.   Recirculation Flow Control Valve

g.   Main Steam Safety/Relief Valve

h.   Main Steam Isolation Valve

i.   Recirculation Pump

j.   Reactor Recirculation System Gate Valve

k.   ASME Code Class 3 Safety/Relief Valve Discharge Piping-Highest
      Stress Summary

L.   Standby Liquid Control Pump

m.   Standby Liquid Control Tank

n.   ECCS Pumps

o.   Residual Heat Removal Heat Exchanger

p.   (Deleted)

q.   RCIC Turbine

r.   RCIC Pump

s.   Reactor Refueling and Servicing Equipment (including fuel assembly)

t.   (Deleted)

TABLE 3.9-3 (Continued)

u.   Control Rod Drive

v.   Control Rod Drive Housing

w.   Jet Pumps

x.   (Deleted)

y.   Highest Stressed Region on the LPCI Coupling (Attachment Ring)

z.   (Deleted)

aa. Control Rod Guide Tube

ab. In-core Housing

ac. Reactor Vessel Support Equipment

ad. Hydraulic Control Unit

ae. (Deleted)

af. HPCS System ASME Code Class 1 Valve

TABLE 3.9-3 INTRODUCTION

This table lists the major NSSS safety-related mechanical components in the plant, on a component by component basis. For each component the loading conditions, stress criteria, calculated stresses, and the allowable stresses are also summarized. The calculated stresses are supplied for information only, and represent actual stresses calculated at a point in time. These stress values are not updated for revisions to the analyses or stress reports. The calculated stresses will, however, continue to meet the ASME code allowables as revisions to the analyses or stress reports are made.

The following <Table 3.9-3> provides the design loading combinations and acceptance criteria for all NSSS ASME Code Class equipment reported in <Table 3.9-3a>, <Table 3.9-3b>, <Table 3.9-3c>, <Table 3.9-3d>, <Table 3.9-3e>, <Table 3.9-3f>, <Table 3.9-3g>, <Table 3.9-3h>, <Table 3.9-3i>, <Table 3.9-3j>, <Table 3.9-3k>, <Table 3.9-3l>, <Table 3.9-3m>, <Table 3.9-3n>, <Table 3.9-3o>, <Table 3.9-3q>, <Table 3.9-3r>, <Table 3.9-3s>, <Table 3.9-3t>, <Table 3.9-3u>, <Table 3.9-3v>, <Table 3.9-3w>, <Table 3.9-3x>, <Table 3.9-3y>, <Table 3.9-3z>, <Table 3.9-3aa>, <Table 3.9-3ab>, <Table 3.9-3ac>, <Table 3.9-3ad>, <Table 3.9-3ae>, and <Table 3.9-3af>.

Various parts of the Table are referenced in Section 3.9. The formats in various parts of the Table are not consistent because of differences in analytical method and depth of detail necessary to demonstrate the safety aspects of various components.

TABLE 3.9-3

DESIGN LOADING COMBINATIONS FOR ASME
CODE CLASS 1, 2 AND 3 COMPONENTS

Load Case[1]	N	SRV_x[4]	SRV_{ADS}	OBE	SSE	SBA/IBA[3]	DBA	ASME Code Service Limit
1	X	X						B
2	X	X		X				B[5]
3	X	X			X			D[2]
4	X		X			X(SBA only)		C[2]
5	X		X	X		X		D[2]
6	X		X		X	X		D[2]
7	X				X		X	D[2]
8	X							A
9	X			X				B

NOTES:

[1] See legend for definition of terms.
[2] 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 Interim Technical Position "Functional Capability of Passive Piping Components" - by MEB.
[3] SBA or IBA whichever is greater, except for Load Case 4.
[4] SRV_{ALL} or SRV, - whichever is controlling will be used.
[5] For load Case 2, all ASME code service level B requirements are to be met, excluding fatigue evaluation.

TABLE 3.9-3 (Continued)

<u>LOAD DEFINITIONS LEGEND</u>

N -   Normal load consists of pressure, dead weight and thermal
                loads.

OBE -   Operational basis earthquake loads.

SSE -   Loads due to vibratory motion from safe shutdown
                earthquake loads.

SRV_1 -   Safety/Relief valve discharge induced loads from one
                valve's subsequent actuation.

SRV_{ALL} -   The loads induced by actuation of all safety/relief
                valves which activate within milliseconds of each other
                (e.g., turbine trip operational transient).

SRV_{ADS} -   The loads induced by the actuation of safety/relief
                valves associated with the automatic depressurization
                system which actuate within milliseconds of each other
                during the postulated small or intermediate-size pipe
                rupture.

DBA -   Design basis accident is the sudden break of the main
                steam or recirculation lines (largest postulated breaks).
                DBA related loads include main vent clearing and pool
                swell, chugging, condensation oscillation, and annulus
                pressurization.

SBA -   Small break accident.

IBA -   Intermediate break accident.

TABLE 3.9-3a

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY[1]

The reactor vessel is designed and analyzed to comply with ASME Code, Section III (NB-3200). The results of stress and fatigue usage analysis are given in detail in the stress report. They are within the code allowables, as demonstrated by the following tabulation of the high stress areas.

The shroud support is designed and analyzed to comply with the ASME Code Section III, Subsection NG. Stress and fatigue analysis results are completed by GE and all results are within the Code limits.

Vessel Support Skirt Attachment to RPV[2]

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B & PV Code, Sec. III, Subsection NB Stress Limit for SA 533 GRB CL1.				
For Design Mechanical Condition:	Normal and Upset[3] Loads:	Primary Membrane Plus Bending	40,050	39,180
$S_{limit} = 1.5\ S_m$	1. Normal Loads 2. OBE 3. Safety/Relief Valve Discharge Caused Loads			
For Emergency Condition:	Emergency Loads:	Primary Membrane Plus Bending	64,300	See Note[4]
$S_{limit} = 1.5\ S_y$	1. Normal Loads 2. OBE 3. Safety/Relief Valve Discharge Caused Loads			

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TABLE 3.9-3a (Continued)

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
For Faulted Condition:	Faulted Loads:	Primary Membrane Plus Bending	See Note[4]	51,890
$S_{limit} = 1.5\ S_y$	1. Normal Loads 2. SSE 3. DBA			

Shroud Support Legs[5]

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B & PV Code, Sec. III, Subsec. NB for SB-166 at 550°F.	Normal & Upset Loads:	Primary Membrane Plus Bending	31,450	17,800
$S_m = 23,300$ psi, $S_u = 73,968$ psi	1. Dead Weight 2. Combined pressure and jet loads			
$S_{limit} = 1.5\ (0.9)\ \text{x}\ S_m$	3. Operating basis earthquake			
Emergency Condition	Emergency Loads:	Primary Membrane Plus Bending	47,170	17,800
$S_{limit} = 2.25\ (0.9)\ S_m$	1. Dead Weight 2. Safe shutdown earthquake 3. Combined pressure and jet loads			

TABLE 3.9-3a (Continued)

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Faulted Condition S_{limit} = 1.5 (0.9) (0.7) S_u	Faulted Loads: 1. Dead Weight 2. Safe shutdown earthquake 3. Combined Pressure and jet loads	Primary Membrane Plus Bending	69,900	44,400

RPV Nozzles - Feedwater

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Based on ASME B & PV Code, Section III, Subsection NB, for type SA 508 at 575°F. S_m = 17,700 psi S_y = 24,200 psi Condition: S_{limit} = 1.5 S_m	Normal and Upset Loads: 1. Weight of Structure 2. Pressure 3. Operating Basis Earthquake 4. SRV 5. Hydraulic	Primary Membrane Plus Bending	26,550	21,420
For Emergency Condition: S_{limit} = 1.5 S_y	Emergency Loads: 1. Weight Structure 2. Pressure 3. Operating Basis earthquake 4. SRV 5. Hydraulic	Primary Membrane Plus Bending	36,300	27,570

TABLE 3.9-3a (Continued)

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
For Faulted Condition:	Faulted Loads:			
	1. Weight of structure	Primary	36,300	27,570
	2. Pressure	Membrane		
S_{limit} = 1.5 S_y	3. Safe Shutdown Earthquake	Plus Bending		
	4. Annulus Pressurization			
	5. Hydraulic			

Cummulative Usage Factor = 0.928

CRD Housing Attachment to RPV

Based on ASME B&PV Code
Section III, Subsection NB,
for SB-167

S_m = 16,067 psi at 575°F

S_y = 24,480 psi at 528°F

For Normal and Upset Condition:	Normal and Upset Loads:	Primary	24,100	15,500
	1. Normal Loads	Membrane		
	2. OBE	Plus		
S_{limit} = 1.5 S_m	3. Safety/Relief Valve Discharge Caused Loads	Bending		

TABLE 3.9-3a (Continued)

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
For Emergency Condition:	Emergency Loads:	Primary	36,720	See Note[6]
	1. Normal Loads	Membrane		
	2. OBE	Plus		
$S_{limit} = 1.5\ S_y$	3. Safety/Relief Valve Discharge Caused Loads	Bending		

CRD Housing Attachment to RPV

For Faulted Condition:	Faulted Loads:	Primary	See Note[6]	20,600
	1. Normal Loads	Membrane		
	2. SSE	Plus		
$S_{limit}^{(6)} = 1.5\ S_y$	3. DBA	Bending		

Cumulative Usage Factor = 0.594 at Housing to Vessel Weld.

NOTES:

[1] The vessel, support skirt and shroud support, including legs, cylinder and plate, are furnished as a completed assembly by the vessel manufacturer.

[2] The vessel support skirt has been evaluated for buckling.

[3] Fatigue usage factor = 0.573 at inside surface of skirt at vessel connection, approximately 1″ below the tangent attachment.

[4] Faulted category loads were evaluated with emergency allowable stresses, hence emergency condition is not evaluated.

[5] The shroud support legs have been evaluated for buckling.

[6] Faulted category loads were evaluated with emergency allowable stresses, hence emergency condition is not evaluated.

TABLE 3.9-3b

REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Types	Allowable Stress (psi)	Calculated Stress (psi)
TOP GUIDE/GRID				
Primary Stress Limit:				
ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, for A240 at 550°F, S_m = 14,300 psi				
For Normal and Upset Condition S_{limit} = 1.5 S S_{limit} = 1.5 S_m	I. Normal and Upset Loads: 1. Normal Loads 2. OBE 3. Safety/Relief Valve Discharge Caused Loads	Primary Membrane Plus Bending	21,450	≤21,156
For Emergency Condition: S_{limit} = 2.25 S_m	II. Emergency Loads: 1. Normal Loads 2. SRV_{ADS}	Primary Membrane Plus Bending	32,175	≤21,156

TABLE 3.9-3b (Continued)

Criteria	Loading	Primary Stress Types	Allowable Stress (psi)	Calculated Stress (psi)
For Faulted Condition: $S_{limit} = 3.6\ S_m$	III. Faulted Loads: 1. Normal Loads 2. SSE 3. SRV_{ADS}	Primary Membrane Plus Bending	51,480	≤50,403

Fatigue usage factor = 0.886 at grid to top guide cylinder continuity.

Vent and Head Spray Nozzle:

S_m @ 550°F = 18,100 psi S_u = 70,000 psi For Normal and Upset Condition: $S_{limit} = 3.0\ S_m$	Normal and Upset Loads: 1. Normal Loads 2. OBE Bending plus 3. SRV Discharge Caused Loads	Primary Membrane plus Secondary Membrane	54,300	47,400
For Emergency Condition: $S_{limit} = 2.25\ S_m$	Emergency Loads: 1. Normal Loads 2. OBE Membrane plus 3. SRV Discharge Caused Loads	Primary Bending	40,725	26,220
For Faulted Condition: $S_{limit} = 0.7\ S_u$	Faulted Loads: 1. Normal Loads 2. SSE Membrane plus 3. DBA Bending 4. SRV Discharge Caused Loads	Primary	49,000	40,800

TABLE 3.9-3b (Continued)

Criteria	Loading	Primary Stress Types	Allowable Stress (psi)	Calculated Stress (psi)
Core Spray Lines and Spargers:				
S_m @ 550°F = 16,950 psi For Normal and Upset Condition:	Normal and Upset Loads:[1]			
	1. Normal Loads	Primary	50,850	16,772
S_{limit} = 3.0 S_m	2. SSE	Membrane plus		
	3. SRV Discharge Caused Loads	Bending and Secondary Membrane		
For Emergency Condition	Emergency Loads:[1]			
	1. Normal Loads	Primary	38,137	16,770
S_{limit} = 2.25 S_m	2. SRV Discharge Caused Loads	Membrane and Bending		
	3. SSE			
For Faulted Condition:	Faulted Loads:			
	1. Normal Loads	Primary	50,850	49,130
S_{limit} = 3.0 S_m	2. DBA	Membrane and		
	3. SSE	Bending		

TABLE 3.9-3b (Continued)

Core Plate (Highest stressed Location): Cover Plate ligaments.

Criteria	Loading	Limiting Stress Types	Allowable Stress (psi)	Calculated Stress (psi)
Based on ASME B&PV Code, Section III, Subsection NG for A240-304L Material				
S_m @ 550°F = 14,300 psi For Normal and Upset condition: S_{limit} = 1.5 S_m	Normal and Upset Loads: 1. Normal Loads 2. SRV Discharge Caused Loads 3. OBE	Primary Membrane plus Bending	21,450	21,051
For Emergency Condition: S_{limit} = 2.25 S_m	Emergency Loads: 1. Normal Loads 2. SRV$_{ADS}$	Primary Membrane plus Bending	32,175	24,293
For Faulted Condition: S_{limit} = 3.6 S_m	Faulted Loads: 1. Normal Loads 2. SSE Membrane plus 3. SRV$_{ADS}$	Primary Bending	51,480	45,278

NOTE:

[1] Normal and upset conditions fatigue usage factor = 0.903 vs 1.0 allowable at the ring to plate weld.

<TABLE 3.9-3c>

DELETED

TABLE 3.9-3d

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE-MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading [1]	Identification [2] of Locations of Highest Stress Points
ASME B&PV Code Section III, NB-3600						
Design Condition:					1. Normal Loads	Joint 029-LUG guide near inboard MSIV, Line A
Eq. 9 ≤1.5 S_m	Primary	16,207 psi	28,725 psi	0.56	2. OBE	
Service Levels A & B (Normal & Upset) Condition:					1. Normal Loads 2. OBE 3. Operating	Joint 002 First elbow
Eq. 12 ≤3.0 S_m	Secondary	43,465 psi	53,100 psi	0.82	Transients	Line A
Service Levels A & B (Normal & Upset) Condition: Eq. 13 ≤3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	35,808 psi	54,600 psi	0.66	1. Normal Loads 2. OBE 3. Operating Transients	Joint 062 SRV Sweepolet Line C

TABLE 3.9-3d (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification[2] of Locations of Highest Stress Points
Service Levels A & B (Normal and Upset) Conditions: Cumulative Usage Factor	N.A.	0.06	1.0	0.06	1. Normal Loads 2. OBE 3. Operating Transients	Joint 281 RCIC Sweepolet Connection Line A
Service Level B (Upset) Condition: Eq. 9 $\leq 1.8\ S_m$ & $1.5\ S_y$	Primary	19,100 psi	34,470 psi	0.55	1. Normal Loads 2. OBE 3. Operating Tranients	Joint 024 - lug guide near inboard MSIV, Line D
Service Level C (Emergency) Condition Eq. 9 $\leq 2.25\ S_m$ & $1.8\ S_y$	Primary	20,250 psi	40,950 psi	0.49	1. Normal Loads 2. SBA 3. Operating Transients	Joint 155 - SRV Sweepolet Line C
Service Level D (Faulted) Condition: Eq. 9 $\leq 3.0\ S_m$	Primary	31,593 psi	57,450 psi	0.55	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	Joint 024 - lug guide near inboard MSIV, Line D

TABLE 3.9-3d (Continued)

Component/ Load Type	Highest Calculated Load or Moment	Allowable Load or Moment	Ratio Calculated/ Allowable	Loading[1]	Identification of Equipment with Highest Loads
Service Levels A&B Loads	20,396 lbs	30,000 lbs	0.680	1. Normal Loads 2. OBE 3. Operating Transients	Mainsteam line C-Snubber S101C
Service Levels C&D Loads	41,916 lbs	45,000 lbs	0.931	1. Normal Loads 2. DBA	Mainsteam line C-Snubber S101C
Bonnet/moment	1.593×10^6 in.-lbs	2.288×10^6 in.-lbs	0.696	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	Mainsteam line D-MSIV
Flange Moment	745.180 in.-lbs	2.411×10^6 in.-lbs	0.3090	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	Mainsteam line D-SRV (Inlet)
Flange Moment	187,755 in.-lbs	1.158×10^6 in.-lbs	0.1621	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	Mainsteam line C-SVR (Outlet)

TABLE 3.9-3d (Continued)

Component/ Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated Allowable	Loading[1]	Identification of Equipment with Highest Loads
Horizontal Acceleration	11.84g	13.0g	0.911	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	Mainsteam line A-MSIV
Vertical Acceleration	4.52g	8.0g	0.565	1. Normal Loads 2. DBA	Mainsteam line D-SRV

NOTES:

[1] Appropriate loading combinations of <Table 3.9-3> were considered and the calculated stresses are reported for the governing loading combinations.

[2] Refer to <Figure 3.6-65> for the identification of node point numbers.

TABLE 3.9-3e

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE-MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Acutal/ Allowable	Loading[1]	Identification[2] of Locations of Highest Stress Points
ASME B&PV Code Section III, NB-3600						
Design Condition:		12,150	25,013	0.49	1. Normal Loads 2. OBE	Hanger Lug Node 025
Eq. 9 ≤1.5 S_m	Primary					
Service Levels A & B (Normal & Upset) Condition:		21,863	50,025	0.44	1. Normal Loads 2. OBE 3. Operating Transients	Suction Elbow Node 008
Eq. 12 ≤3.0 S_m	Secondary					
Service Levels A & B (Normal & Upset) Condition: Eq. 13 ≤3.0 S_m	Secondary (Except Thermal expansion)	32,444	50,025	0.65	1. Normal Loads 2. SSE 3. Operating Transients	Header Sweepolet Node 216

TABLE 3.9-3e (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification[2] of Locations of Highest Stress Points
Service Levels A & B (Normal and Upset) Condition: Cummulative Usage Factor	N.A.	0.12	1.0	0.12	1. Normal Loads 2. OBE 3. Operating Transients	Header Sweepolet Node 216
Service Level B (Upset) Condition: Eq. 9 ≤1.8 S_m & 1.5 S_y		22,875	28,598	0.80	1. Normal Loads 2. OBE 3. Operating Transients	Cross Node 160
Service Level C (Emergency) Condition: Eq. 9 ≤2.25 S_m & 1.8 S_y	Primary	19,649	34,317	0.57	1. Normal Loads 2. Infrequent Operating Transients	Cross Node 160
Service Level D (Faulted) Condition: Eq. 9 ≤3.0 S_m	Primary	28,888	38,130	0.76	1. Normal Loads 2. SSE 3. DBA	Cross Node 160

TABLE 3.9-3e (Continued)

Component/ Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading [1]	Identification of Equipment with Highest Loads
Service Levels A & B Loads	87,643 lbs	88,000 lbs	0.996	1. Normal Loads 2. OBE 3. Operating Transients	Loop B Strut B302
Service Levels C & D	112,477 lbs	121,488 lbs	0.926	1. Normal Loads 2. SSE 3. DBA	Loop B Strut B302
Discharge Valve Moments	998,166 in-lbs	1,532,360 in-lbs	0.65	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	
Suction Valve Moments	994,602 in-lbs	2,617,140 in-lbs	0.38	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	

NOTES:

[1] Appropriate loading combinations of <Table 3.9-3> were considered and the calculated stresses are for the governing loading combinations.

[2] Refer to <Figure 3.6-66> for the identification of node point numbers.

TABLE 3.9-3f

RECIRCULATION FLOW CONTROL VALVE 24" SIZE (FISHER) ASME CODE SECTION III 1971 EDITION, WITH W73 ADDENDA[1]

Para. No.	Component/Stress/Loading	Design Procedure	Allowable Limit	Calculated or Actual Value	Ratio Calc/ Allowed
1.0	Body, Housing, Bonnet & Covers				
1.1	Loads - Design Pressure	System Requirement	NA	1,675 psi	NA
	Design Temperature	System Requirement		575°F	
1.2	Body Pressure Rating	ASME Sec. III NB-3545.1-2	NA	985 psi	NA
1.3	Body Min. Wall Thickness	ASME Sec. III NB-3541	$t_m = 2.614$ in.	$t_m = 2.710$ in.	1.036
1.4	Max. Primary Body Membrane Stress	ASME Sec. III NB-3545.1	$P_m \leq S_m$ (575°) $= 19,600$ psi	$P_m = 10,265$ psi	0.523
1.5	Max. Primary & Sec. Body Stress	ASME Sec. III NB-3545.2	$P_m \leq 3\ S_m$ $\leq 58,800$ psi	$S_m = 22,000$ psi	0.374
1.6	Housing Min. Wall Thickness	ASME Sec. III NB-3541	$t_m = 2.549$ in.	$t_m = 2.710$ in.	1.063
1.7	Max. Primary Housing Membrane Stress	ASME Sec. III NB-3545.1	$P_m \leq S_m$ (575°) $\leq 19,600$ psi	$P_m = 8,400$ psi	0.428
1.8	Max. Primary & Sec. Housing Stress	ASME Sec. III NB-3545.2	$S_m \leq 3\ S_m$ (575°) $\leq 58,800$ psi	$S_m = 23,100$ psi	0.392

TABLE 3.9-3f (Continued)

Para. No.	Component/Stress/Loading	Design Procedure	Allowable Limit	Calculated or Actual Value	Ratio Calc/ Allowed
1.9	Cyclic Requirements	ASME Sec. III NB-3545.3	$N_a \geq 2,000$ cyc	$N_a = 10^6$ cyc	NA
1.10	Fatigue Analysis Usage Factor	ASME Sec. III NB-3550	$I_t \leq 1.0$	$I_t = .0004$	NA
1.11	Body to Housing Flange Max. Stress	ASME Sec. III NB-3647.1	$S_m = 29,400$ psi (1.5 x 19,600)	$S_m = 27,440$ psi	0.933
1.12	Body to Housing Studs - Area	ASME Sec. III NB-3647.1	$A_6 \geq 29.04$ in.^{2}	$A_6 = 31.68$ in.^{2}	1.090
	Body to Housing Primary Stress		$S_m = 27,000$ psi	$S_6 = 24,750$ psi	0.916
	Body to Housing Maximum Stress		$3\ S_m = 81,000$psi	$S_6 = 66,500$ psi	0.820
1.13	Top Housing Cover - Thickness	ASME Sec. III NB-3646 & ASME Sec. VIII UG-34	$t_m \geq 3.23$ in.	$t_m = 4.63$ in.	1.433
1.14	Top Housing Cover Studs - Area	ASME Sec. III NB-3647.1	$A_6 \geq 31.75$ in.^{2}	$A_6 = 33.60$ in.^{2}	1.058
	Top Housing Primary Stud Stress		$S_m = 27,000$psi	$S_6 = 25,900$ psi	0.959
	Top Housing Maximum Stud Stress		$3\ S_m = 81,000$psi	$S_6 = 45,100$ psi	0.556

TABLE 3.9-3f (Continued)

Para. No.	Component/Stress/Loading	Design Procedure	Allowable Limit	Calculated or Actual Value	Ratio Calc/ Allowed
1.15	Bottom Cover - Thickness	ASME Sec. III NB-3646 & Sec. VIII UG-34	$t_m \geq 1.85$ in.	$t_m = 3.56$ in.	1.924
1.16	Bottom Cover Primary Stud Stress	ASME Sec. III NB-3647.1	$S_m = 27,000$ psi	$S_6 = 25,000$ pie	0.925
	Bottom Cover Maximum Stud Stress		$3\ S_m = 81,000$ psi	$S_6 = 55,380$ psi	0.683
	Bottom Cover Studs - Area		$A_6 \geq 11.09$ in.^{2}	$A_6 = 13.44$ in.^{2}	1.211
1.17	Bonnet Cartridge - Thickness	ASME Sec. III NB-3646 & Sec. VIII UG-34	$T_m \geq 1.82$ in.	$t_m = 3.125$ in.	1.717
1.18	Bonnet Cart. Studs - Area	ASME Sec. III NB-3647.1	$A_6 \geq 12.07$ in.^{2}	$A_6 = 18.48$ in.^{2}	1.531
	Bonnet Cart. Primary Stud Stress		$S_m = 27,000$ psi	$S_6 = 17,630$ psi	0.652
	Bonnet Cart. Maximum Stud Stress		$3\ S_m = 81,000$ psi	$S_6 = 51,680$ psi	0.638

NOTE:

[1] The recirculation flow control valves are passive components and therefore not required to operate in emergency or faulted conditions. The valves have been designed for 6 g's vertical and 9 g's horizontal, which exceeds any load condition from <Table 3.9-3a>. The valves will maintain pressure integrity during and after events impossing these accelerations. The valve internals (ballshaft, linkage, bearings, etc.) have also been designed for faulted, large pipe break conditions. Qualification method is by analysis only.

TABLE 3.9-3g

MAIN STEAM SAFETY/RELIEF VALVES
ASME CODE, SECTION III, JULY 1974, INCLUDING ADDENDA THROUGH SUMMER 1976

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
1. Body inlet and outlet flange stresses	$S_H = \dfrac{fMo}{Lg_1 2_B} + \dfrac{PB}{4g_o} < 1.5\,S_m$		1.5 S_m = 26,310 psi (inlet) and = 28,350 psi (outlet)	Inlet: $\overline{S_H}$ =1.15 S_m =0.77 (allowable) S_R =0.23 S_m =0.169 (allowable)

Note, Topics 1 and 2:
Design Pressures:
P_d =1,375 psig (inlet)

(Uses same notation as codes)

S_T =0.98 S_m =0.66 (allowable)

P_b =625 psig (outlet)

$S_R = \dfrac{\left(4t_e \, / \, 3 + 1\right) Mo < 1.5 S_m}{Lt^2 \, B}$

Outlet:

S_H =1.21 S_m =0.81 (allowable)

These are the max anticipated pressures under all operating conditions. Analyses include applied moments of:
M=800,000 in.-lb (inlet) and
M=300,000 in.-lb (outlet)

$S_T = \dfrac{YMo}{t2_B} - \dfrac{Zs}{R} < 1.5\,S_m$

S_R =0.79 S_m =0.53 (allowable)

S_T =0.49 S_m =0.33 (allowable)

The analyses also include consideration of seismic, operational, and flow reaction forces. Since these safety/relief valves are pipe-mounted equipment, refer to the piping analysis for verification that the moments are not exceeded.

where:

S_H = Longitudinal "Hub" wall stress, psi
S_R = Radial "flange" (body Base, Inlet) Stress, psi
S_T = Trangential "Flange" Stress, psi

Body Material: ASME SA 352 LCB

Inlet: S_m @ 585°F=17,540 psi

Outlet: S_m @ 500°F=18,900 psi

TABLE 3.9-3g (Continued)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
2. Inlet and outlet stud area requirements	Total cross-sectional area shall exceed the greater of: $Am_1 = \dfrac{Wm_1}{Sb}$ or $Am_2 = \dfrac{Wm_2}{Sb}$ where: Am_1=total required bolt (stud) area for operating condition. Am_2=total required bolt (stud) area for gasket seating.	(Uses same notation as codes) Bolting Material: ASME SA 193 GR B7	Inlet: $\overline{Am_1}$ (>Am_2)=12.45 in.^{2} Outlet: $\overline{Am_1}$ (>Am_2)=4.65 in.^{2}	Inlet: $\overline{Ab}$ (actual area)=1.52 Am (required min) Outlet: $\overline{Am}$ (actual area)=1.84 Am (required min)
start here 3. Nozzle Wall thickness	1. Minimum Wall Thickness Criterion: t_{min} < t_A where: t_{min}=Minimum calculated thickness requirement, including corrosion allowance. t_A =Actual nozzle wall thickness. (NOTE: This t_{min} is t_m per notation of the codes).	Cestion near nozzle base: t_{m_e} < t_{m_e} (actual) Nozzle mid-section: t_{m_c} < t_{m_c} (actual) Thin section near valve seat: t_{m_b} < t_{m_b} (actual) Thinnest section at nozzle Tip-just below valve seat: t_{m_a} < t_{m_a} (actual) Nozzle Material: ASME SA 350 LF2	t_{m_e} = 0.84 in. t_{m_c} = 0.81 in. t_{m_b} = 0.79 in. t_{m_a} = 0.206 Actual thickness reeater than t_m at the section under consideration.	t_{m_e} (actual) = 1.58 t_{m_e} t_{m_c} (actual) = 1.54 t_{m_c} t_{m_b} (actual) = 1.012 t_{m_b} t_{m_a} (actual) = 1.68 t_{m_a}

TABLE 3.9-3g (Continued)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
3. Nozzle Wall thickness (Continued): (Refer to <Section 3.9.1.1.9> for thermal transients information.)	1. Cyclic Rating: Thermal:			
	$It=\Sigma\dfrac{Nri}{Ni}$	$It=\Sigma\dfrac{Nri}{Ni}$ (i=1,2,3,4&5)	It (max)≤1.0	It=0.00138 [=0.00138xIt(max)]
	Fatigue			
	Na≥2,000 cycles, as based on Sa, where Sa is defined as the larger of	Na≥2,000 cycles, as based on Sa, where Sa=Sp_1 (>Sp_2)	Na≥2,000 cycles	Na(based on Sa=Sp2) =400,000 cycles: criterion satisfied
	$SP_1 = (2/3)Q_P + \dfrac{{}^Peb}{2} + Q_{T_3}$ $+1.3Q_{T_1}$ or $SP_2 = 0.4Q_P + \dfrac{K}{2}\left(P_{eb} + 2Q_{T_3}\right)$	(Uses same notation)		
	where: Sp_1=Fatigue stress intensity at inside surface of crotch, psi Sp_2=Fatigue stress intensity at inside surface of crotch, psi			

TABLE 3.9-3g (Continued)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
4. Bonnet Flange Strength	Flange treated as a loose type flange without hub:	(Uses same notation as codes)	1.5 S_m (for max S_H, S_R, and S_T)	S_R =1.35 S_m=0.9 (allowable) S_T =0.53 S_m=0.35 (allowable) (max S_T @ back face of flange)

$$S_R = \pm \frac{6M_P}{t^2 \left(3.14\,C - nD\right)}$$

=28,350 psi

$$S_T = \pm \frac{5.46\,M_P}{Bt_2}\,0.318\,\frac{C - B}{C + B} +$$

$$\frac{sh_C}{C + A} + r_B - \frac{E\theta_A t}{B}$$

Bonnet Material: ASME SA

352LCB

where:
S_R =Radial "Flange" Stress, psi
S_T =Tangential "Flange" stress, psi

S_m at 500°F=18,900 psi

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
5. Bonnet Bolting area requirements	Total cross-sectional area shall exceed the greater of:		$Am_1 (>Am_2)$=7.399 in.^{2}	A_b (actual area)=1.34 Am (required min)

$$Am_1 = \frac{WM_1}{Sb}, \text{ or}$$

$$Am_1 = \frac{Wm_1}{Sb}$$ where Am (required

minimum) is the greater of Am1 and

$$Am_2 = \frac{Wm_2}{Sa}$$

$$Am_2 = \frac{Wm_1}{Sb}$$ Am$_2$; and ab (actual

bolt area) must exceed Am

TABLE 3.9-3g (Continued)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
5. Bonnet Bolting (Continued)	where: Am_1=Total required bold (stud) area for operating condition. Am_2=Total required bolt (stud) area for gasket seating.	Body to Bonnet Bolting Material: ASME SA 193 Gr B7		
6. Disc	The disc stress is calculated based on treating the disc as a flat circular plate, edges supported, uniform load over area with radius r_o; reference Bach's Formulas, Machinery's Handbook, 15th Ed., Page 414. From the reference, $$t = 1.2 \sqrt{\dfrac{Wr\left(1 - \dfrac{2r_o}{3R}\right)}{S}}$$ W is based on p=1375 psi under the disc.	W=27,430 lbs r_o=0.785 inch R=1.48 inches Disc Material: ASME SA 351 CF3A	t(minimum allowable)=1.067 inches	Actual t_{min}=1.068 inches =1.0009 (required minimum)

TABLE 3.9-3g (Continued)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
6. Disc (Continued)		Temperature: 585°F S_m(585°F)=18,235 psi Allowable stress is 1.5 S_m. This is the value "S" in the above formula. (1.5 S_m=27,353 psi)		
7. Seismic Capability	Stress analysis uses F sub (vertical) = (mass of valve) x (4.5g), and F sub (horizontal) = (mass of valve) x (6.5g), with 800,000 in.-lb and 300,000 in.-lb applied at the inlet and outlet, respectively. The actual capability was verified by testing (with the moments concurrently applied), and exceeds these values.			

TABLE 3.9-3h

<u>MAIN STEAM ISOLATION VALVE</u>
<u>DESIGN OF PRESSURE RETAINING PARTS - ASME B & PV CODE SECTION III 1974</u>

Item No.	Component/Load Type/ Stress Type	Design Procedure	Allowable Value	Ratio Design/Calculated Value	(Calculated) (Allowable)
1.0	Body and Bonnet				
1.1	Loads:				
	Design Pressure	GE System Specification	1,375 psi	1,375 psi	N/A
	Design Temperature	GE System Specification	586°F	586°F	N/A
	Pipe Reaction Loads	ASME III, NB-3545-2(b)(1)	N/A	N/A	N/A
1.2	Pressure Rating	Table NB-3542.1-2	P_R = 575 lbs	P_R = 575 lbs	N/A
1.3	Minimum Wall	Paragraph NB-3542	t_M ≥1.751 in.	t_M = 1.751 in.	N/A
1.4	Primary Membrane Stress	Paragraph NB-3545.1 (500°F)	P_m = 19,400 psi	P_m = 10,714 psi	0.55
1.5	Secondary Stress Due to Pipe Reaction	Paragraph NB-3545.2(b)(1)	P_{ed}, P_{ed}, and P_{et}	P = 5,277 psi	0.18
			≤1.55m (500°F)	P_{ed} = 10,065	0.35
			1.5 S_m = 29,100	P_{et} = 9,790 psi	0.34
1.6	Primary Plus Secondary Stress Due to Internal Pressure	Paragraph NB-3545.2(a)(1)	N/A	Q_b = 26,024	N/A

TABLE 3.9-3h (Continued)

Item No.	Component/Load Type/ Stress Type	Design Procedure	Allowable Value	Design/Calculated Value	Ratio (Calculated) (Allowable)
1.7	Thermal Secondary	Paragraph NB-3545.2(c)	N/A	Q_{T_2} = 601 psi	N/A
1.8	Range of Primary Plus Secondary Stress at Crotch Region	Paragraph NB-3545.2	S_n = 58,200 psi	S_n = 32,502 psi	0.56
1.9	Body Shape Rule -Radius at Crotch	Paragraph NB-3544 Paragraph NB-3544.1(a)	$r_2 \geq 0.581$ in.	r_2 = 1.125 in.	
	-Corner Radius	Paragraph NB-3544.1(b)	$r_4 \leq 1.125$ in.	r_n = 1.0 in.	
	-Longitudinal Curvature	Paragraph NB-3544.6	>0.0567 1/in.	= 0.123 1/in.	
	-No Flat Walls	Paragraph NB-3544.7	N/A	N/A	
	-Minimum Wall at Weld Ends	Paragraph NB-3544.8	<1.492 in.	= 1.519 in.	
1.10	Cyclic Requirement for Fatigue Analysis	Paragraph NB-3545.3	$N_a \geq 2,000$ cycles	N_a = 39,000 cycles	0.05
1.11	Cummulative Usage Factor Requirements for Fatigue Analysis	Paragraph NB-3550	$I_t \leq 1.0$	I_t = 0.0114	0.01

TABLE 3.9-3h (Continued)

Item No.	Component/Load Type/ Stress Type	Design Procedue	Allowable Value	Design/Calculated Value	Ratio (Calculated) (Allowable)
2.0	Body Flange/Bonnet				
2.1	Loads: (1) Design Pressure (2) Design Temperature (3) External Moments Due to Dynamic Loads That Include SSE Accelerations. (4) Spring Force				
2.2	Body Flange Stresses	Paragraph NB-3647.1		$P_{FD}^{(1)}$ = 1,748 psig	
	Longitudinal Hub Stress		$S_H \leq 1.5\ S_m$ = 26,700 psi	S_H = 21,400 psi	0.80
	Radial Flange Stress		$S_R \leq 1.5\ S_m$ = 26,700 psi	S_R = 14,373 psi	0.54
	Tangential Flange		$S_T \leq 1.5\ S_m$ = 26,700 psi	S_T = 7,194 psi	0.27

TABLE 3.9-3h (Continued)

Item No.	Component/Load Type/ Stress Type	Design Procedue	Allowable Value	Design/Calculated Value	Ratio (Calculated) (Allowable)
2.3	Bonnet Thickness	Paragraph NB-3646 Corrosion Allowance = 0.12 IN	$t_n \geq 5.7135$ in.	$t_n = 5.938$ in.	
2.4	Bonnet Reinforcement	Paragraph NB-3646(e)	Area ≥ 7.86 in.^{2}	Area = 5.938 in.^{2}	
3.0	Bonnet to Body Bolting Loads: (1) Design Pressure (2) Design Temperature (3) External Moments Due to Dynamic Loads Which Include See Accelerations (4) Actuator Operational Loads (5) Spring Force	Appendix XI	$A_b \geq 7.86$ in.^{2}	$A_b = 8.107$ in.^{2}	1.48
4.0	Valve Poppet				
4.1	Loads (1) Inline Pressure Loads				

TABLE 3.9-3h (Continued)

Item No.	Component/Load Type/ Stress Type	Applicable Valves	Design Procedure	(Allowable Value)	Design/Calculated Value	Ratio (Calculated) (Allowable)
4.2	Maximum Stress	B21-F022 B21-F028	Roark's Formulae for Stress and Strain 3rd Edition, Cases 13 and 14.	$S_t \leq 17,800$ psi	$S_t = 8,603$ psi	.048
5.0	Valve Stem					
5.1	Loads: (1) Axial Loads					
5.2	Under-Cut Thread Stress	B21-F022 B21-F028	Industry Standards	$S_t \leq 92,800$ psi	$S_t = 18,521$ psi	0.20
5.3	Thread Shear Stress	B21-F022 B21-F028	Values Indicated are for Valves in Closed Position	$S_s \leq 26,280$ psi	$S_s = 4,929$ psi	0.19
5.4	Buckling Force (5) Spring Force	B21-F022 B21-F028		$F \leq 71,959$ lbs	$F = 46,951$ lbs	0.65

NOTE:

[1] P_{FD} = flange design pressure as defined in paragraph NB-3647.1.

TABLE 3.9-3i

RECIRCULATION PUMP CASE SUMMARY OF LOAD CLASSIFICATION HIGH STRESS LOCATIONS AND LIMIT CRITERIA

Loading Condition ASME Sect. III	LOAD COMBINATION		Criteria (ASME Sect. III) NB-3220	Location	Highest Calc. Stress/ Usage Fact.	Allowable Stress	Ratio (Act.) (All.)
	Pressure	Mechanical Loads					
Design (NB-3112)	Design Pressure =1,650 psi	1. OBE (conservatively include) 2. Pump Thrust 3. Deadweight 4. Nozzle Loads 5. Gasket Seating Load 6. Lug Loads	Fig. NB-3221-1 $P_m \leq 1.0\ S_m$ $P_L + P_b \leq 1.5\ S_m$	Upper Flange	26,717 psi	28,800 psi	0.78
Design (NB-3441.3)	Design Pressure =1,650 psig		NB 3,444.3	Casing	Minimum Wall thickness = 2.91 ins.	N/A	N/A
Normal (NB-3113.1) and Upset (NB-3113.2)	Operating Pressure =1,313 psig	1. Deadweight 2. Nozzle Loads 3. Thermal Transient 4. OBE for Upset Only 5. Gasket Seating 6. Lug Loads	Fig. NB-3222-1 $P_L + P_b + P_e + Q \leq 3.0\ S_m$ $P_e \leq 3.05\ m$ Elastic-Plastic Analysis NB-3228.3(1)	Uppercase Flange	31,851[1] psi Exemption from cyclic analysis is satisfied per NB-3228	57,675 psi	1.0
Emergency (NB-3113.3)	Most Severe Emergency Pressure (1,630 psig)	1. Deadweight 2. Nozzle Loads 3. Pump Thrust 4. Gasket Seating Load 5. OBE (conservatively included) 6. Lug Loads	Fig. NB-3224-1 $P_m \leq (1.2 S_m\ or\ S_y)$ $P_L \leq (1.8 S_m\ or\ S_y)$ $P + P_L \leq (1.8\ S_M$ or $1.5 S_y)$	Upper Flange	Loading is less severe than design loading and allowable stresses are higher for emergency condition the stresses are acceptable.		
Faulted (NCA-2144)	Most Severe Faulted Pressure (1,313 psig)	1. Deadweight 2. Nozzle Loads 3. SSE 4. Pump Thrust 5. Gasket Seating Load 6. Lug Loads	Table F-1322.2-1 $P_m \leq 1.5\ S_m$ or M_m = Lesser of 2.4 S_m or 0.7 S_u	Case Bolting Motor Stand Plate/cover Plate Junction	46,463 psi 14,786 psi	52,200 psi 44,600 psi	0.795 0.351

NOTE:

[1] Primary plus secondary stress intensity range excluding thermal bending.

TABLE 3.9-3j

REACTOR RECIRCULATION SYSTEM GATE VALVES - SUCTION

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
1.0	Body and Bonnet				
1.1	Loads:				
	Design Pressure	System Requirement	1,250 psi	1,250 psi	N/A
	Design Temperature	System Requirement	575°F	575°F	N/A
1.2	Pressure Rating 735 psi	ASME Section III[1], Figure NB-3545.1-2	P_r =735 psi	P_r =735 psi	N/A
1.3	Minimum Wall Thickness, Inches	ASME Section III[1], Paragraph NB-3542	t_{min}=1.78 inches	t_{min}=1.783 inches	N/A
1.4	Primary Membrane Stress, psi	ASME Section III[1], Paragraph NB-3545.1	$P_m \leq S_m(500°F)$ = 19,600 psi	P_m =11,398 psi	0.58
1.5	Secondary Stress Due to Pipe Reaction	ASME Section III[1], Paragraph NB-3545.2 (b)(i)	P_e = Greatest Value of P_{ed}, P_{eb}, and P_{et} $\leq 1.5\ S_m(500°F)$ (1.5)(19,600) = 29,400 psi	P_{ed} = 6,408 psi P_{ed} = 14,483 psi P_{et} = 14,483 psi P_e = P_{eb} = 14,483 psi	0.49

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
1.6	Primary Plus Secondary Stress Due to Internal Pressure	ASME Section III[1], Paragraph NB-3545.2 (a)(1)	See Paragraph 1.8	$Q_P = 20,163$ psi	N/A
1.7	Thermal Secondary Stress	ASME Section III[1], Paragraph NB-3545.2(c)	See Paragraph	$Q_T = 2,835$ psi	N/A
1.8	Range or Primary Plus Secondary Stress at Crotch Region	ASME Section III[1], Paragraph NB-3545.2	$S_n \leq 3S_m (500°F)$ $= 58,800$ psi	$S_n = Q_P + P_e + 2Q$ $= 27,940$	0.48
1.9	Cycle Requirements for Fatigue Analysis	ASME Section III[1], Paragraph NB-3545.3	$N_a \geq 2,000$ cycles	$N_a = 1 \times 10^6$ cycles	N/A
1.10	Usage Factor Requirements for Fatigue Analysis	ASME Section III[1], Paragraph NB-3550	$I_t \leq 1.0$	$I_t = 0.0012$	0.0012

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.0	Body to Bonnet Bolting				
2.1	Loads: 1) Design Pressure and 2) Temp., 3) Gasket Loads, 4) Stem Operational Load, 5) Seismic Load (Safe Shutdown Earthquake)		N/A	N/A	N/A
2.2	Bolt Area	ASME Section III[1], Paragraph NB-3647.1	$A_b \geq 28.84$ in.^{2}	$A_b = 34.03$ in.^{2}	1.18
			$S_b = 28,675$ psi	$S_b = 24,254$ psi	0.85
2.3	Body Flange Stresses	ASME Section III[1], Paragraph NB-3647.1	-	-	-

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.3.1	Operating Condition		$S_h \leq 1.5\ S_m(575°F)$ $=28,838$ psi	$S_h =11,735$ psi	0.41
			$S_r \leq 1.5\ S_m(575°F)$ $=28,838$ psi	$S_r =7,991$ psi	0.28
			$S_t \leq 1.5\ S_m(575°F)$ $=28,838$ psi	$S_t =3,365$ psi	0.12
2.3.2	Gasket Seating Condition	AMSE Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(100°F)$ $=30.000$ psi	$S_h =16,872$ psi	0.56
			$S_r \leq 1.5\ S_m(100°F)$ $=30,000$ psi	$S_r =12,717$ psi	0.42
			$S_t \leq 1.5\ S_m(100°F)$ $=30,000$ psi	$S_t =5,325$ psi	0.18
2.4	Bonnet Flange Stresses	ASME Section III[1], Paragraph NB-3647.1	N/A	N/A	N/A

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.4.1	Operating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(575°F)$ $= 28,838$ psi	$S_h = 17,343$ psi	0.60
			$S_r \leq 1.5\ S_m(575°F)$ $= 28,838$ psi	$S_r = 5,047$ psi	0.18
			$S_t \leq 1.5\ S_m(575°F)$ $= 28,838$ psi	$S_t = 6,169$ psi	0.21
2.4.2	Gasket Seating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(100°F)$ $= 30,000$ psi	$S_h = 22,901$ psi	0.76
			$S_r \leq 1.5\ S_m(100°F)$ $= 30,000$ psi	$S_r = 7,041$ psi	0.23
			$S_t \leq 1.5\ S_m(100°F)$ $= 30,000$ psi	$S_t = 8,333$ psi	0.28

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
3.0	Stresses in Stem				
3.1	Loads: 1) Operator Thrust and 2) Torque				
3.2	Stem Thrust Stress	Calculate Stress Due to Operator Thrust in Critical Cross Section	$S_t \leq S_m$ = 42,275 psi	S_t =3,440 psi	0.08
3.3	Stem Torque Stress	Calculate Shear Stress Due to Operator Torque in Critical Cross Section	$S_s \leq 0.6\ S_m$ = 25,365 psi	S_s =1,995 psi	0.08
3.4	Buckling on Stem	Calculate Slenderness Ratio if Greater Than 30, Calculate Allowable Load from Rankine's Formula Using Safety Factor of 4	Max. Allowable Load = 62,715 lb	Slenderness Ratio = 59.59 Actual Load on Stem =12,476 lbs Therefore, No Buckling	0.20

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
4.0	Disc Analysis				
4.1	Loads: Maximum Differential Pressure[2]				
4.2	Maximum Stress in the Disc	ASME Section III[1], Paragraph NB-3215 and ASME Section III[1], Paragraph NB-3221.3	$S_{max} \leq 1.5 S_m (575°F)$ $= 27,488$ psi	MAX STRESS $=14,960$ psi	0.54
5.0	Yoke and Yoke Connections				
5.1	Loads: Stem Operational Loads	Calculate Stresses in the Yoke and Yoke Connections to Acceptable Structural Analysis Methods.	-	-	-
5.2	Tensile Stress in Yoke Leg Bolts	-	$S_{max} \leq S_m (100°F)$ $= 35,000$ psi	$S_{max} = 8,396$ psi	0.24
5.3	Bending Stress of Yoke Legs	-	$S_b \leq 1.5 \ S_m (185°F)$ $= 33,165$ psi	$S_b = 13,050$ psi	0.39

TABLE 3.9-3j (Continued)

REACTOR RECIRCULATION SYSTEM GATE VALVES - DISCHARGE

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
1.0	Body and Bonnet				
1.1	Loads:				
	Design Pressure	System Requirement	1,650 psi		
	Design Temperature	System Requirement	575°F		
	Pipe Reaction	Not Specified			
	Thermal Effects	Not Specified			
1.2	Pressure Rating, 970 psi	ASME Section III[1], Figure NB 3545.1-2	P_r = 970 psi	P_r =970 psi	N/A
1.3	Minimum Wall Thickness, Inches	ASME Section III[1], Paragraph NB-3542	t(nominal) = 2.432 inches	t_m =2.432 Min. inches	Tact = 2.813
1.4	Primary Membrane Stress, PSI	ASME Section III[1], Paragraph NB-3545.1	$P_m \leq S_m$(500°F) = 17,250 psi	P_m =7,730 psi	0.45

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
1.5	Secondary Stress Due to Pipe Reaction	ASME Section III[1], Paragraph NB-3545.2	P_e = Greatest	P_{ed} =2,811 psi	N/A
			Value of P_{ed}	P_{eb} =5,453 psi	N/A
			P_{eb} and P_{et}	P_{et} =5,453 psi	N/A
			≤1.5 S_m(500°F)	P_e = P_{et}=5,453 psi	0.21
			1.5 (17,250)		
			= 25,875 psi		
1.6	Primary Plus Secondary Stress Due to Internal Pressure	ASME Section III[1], Paragraph NB-3545.2 (a) (1)	S_n ≤3 S_m(500°F) = 3(17,250)	Q_p =18,760 psi	0.36
			= 51,750 psi		
1.7	Thermal Secondary Stress	ASME Section III[1], Paragraph NB-3545.2 (c)	S_n ≤3 S_m(500°F) = 51,750 psi	Q_t =5,361 psi	0.10
1.8	Sum of Primary Plus Secondary Stress	AMSE Section III[1], Paragraph NB-3545.2	S_n ≤3 S_m(500°F)	S_n =Q_p + P_e + 2Q_{T2}	
			= 51,750 psi	=25,730 psi	0.50

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
1.9	Fatigue Requirements	ASME Section III[1], Paragraph NB-3545.3	$N_a \geq 2{,}000$ cycles	$N_a = 1 \times 10^6$ cycles	
1.10	Cyclic Rating	ASME Section III[1], Paragraph NB-3550	$I_t \leq 1.0$	$I_t = 0.00067$	0.00067
2.0	Body to Bonnet Bolting				
2.1	Loads: Design Pressure, and Temperature, Gasket Loads, Stem Operational Load, Seismic Load (Design Basis Earthquake)	ASME Section III[1], Paragraph NB-3647.1	-	-	-
2.2	Bolt Area	ASME Section III[1], Paragraph NB-3647.1	$A_b \geq 44.58$ in.^{2}	$A_b = 47.5$ in.^{2}	1.07
			$S_b \leq 25{,}000$ psi	$S_b = 23{,}460$ psi	0.94
2.3	Body Flange Stresses	ASME Section III[1], Paragraph NB-3647.1	-	-	-

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.3.1	Operating Conditions	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(575°F)$ = 25,875 psi	S_h =12,926 psi	0.50
			$S_r \leq 1.5\ S_m(575°F)$ = 25,875 psi	S_r =18,303 psi	0.71
			$S_t \leq 1.5\ S_m(575°F)$ = 25,875 psi	S_t =4,445 psi	0.17
2.3.2	Gasket Seating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(100°F)$ = 30,000 psi	S_h =14,587 psi	0.49
			$S_r \leq 1.5\ S_m(100°F)$ = 30,000 psi	S_r =21,437 psi	0.71
			$S_t \leq 1.5\ S_m(100°F)$ = 30,000 psi	S_t =5,206 psi	0.17

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.4	Bonnet Flange Stresses				
2.4.1	Operating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(575°F)$ $= 25,875$ psi	$S_h = 14,493$ psi	0.56
			$S_r \leq 1.5\ S_m(575°F)$ $= 25,875$ psi	$S_r = 16,349$ psi	0.63
			$S_t \leq 1.5\ S_m(575°F)$ $= 25,875$ psi	$S_t = 4,638$ psi	0.18
2.4.2	Gasket Seating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(100°F)$ $= 30,000$ psi	$S_h = 16,007$ psi	0.53
			$S_r \leq 1.5\ S_m(100°F)$ $= 30,000$ psi	$S_r = 18,546$ psi	0.62
			$S_t \leq 1.5\ S_m(100°F)$ $= 30,000$ psi	$S_t = 5,261$ psi	0.18

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
3.0	Stresses in Stem	N/A	N/A	N/A	N/A
3.1	Loads: Operator Thrust and Torque				
3.2	Stem Thrust Stress	Calculate Stress Due to Operator Thrust in Critical Cross Section	$S_t < S_m$ $S_m = 42,275$ psi	$S_t = 8,818$ psi	0.21
3.3	Stem Torque Stress	Calculate Shear Stress Due to Operator Torque in Critical Cross Section	$S_s \leq 0.6\ S_m$ $= 25,365$ psi	$S_s = 1,544$ psi	0.06
3.4	Buckling on Stem	Calculate Slenderness Ratio. If Greater Than 30, Calculate Allowable Load From Rankine's Formula Using Safety Factor of 4.	Max Allowable Load = 350,000 LBS	Slenderness Ratio = 96.75 Actual Load on Stem = 36,683 LBS Therefore, No Buckling.	0.10
4.0	Disc Analysis				
4.1	Loads: Maximum Differential Pressure[3]	-	-	-	-

TABLE 3.9-3j (Continued)

Paragraph No.	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
4.2	Maximum Stress in the Disc	ASME Section III[1], Paragraph NB-3215 AND ASME Section III[1], Paragraph NB-3221.3	$S_{max} \leq 1.5S_m(575°F)$ = 29,400 psi	Max Stress =27,004 psi	0.92
5.0	Yoke and Yoke Connections				
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} 0.654 (575°F) = 51,660 psi	S_{max} =28,443 psi	0.55
5.3	Combined Axial and Bending Stress of Yoke Legs	-	$S_b \leq 1.5S_m(100°F)$ = 32,400 psi	S_b =31,429 psi	0.97

NOTES:

[1] ASME Section III, 1974 Edition.
[2] Valve Differential Pressure is 50 psig.
[3] Valve Differential Pressure is 450 psig.

TABLE 3.9-3k

ASME CODE CLASS 3 SAFETY/RELIEF VALVE DISCHARGE PIPING SYSTEM - HIGHEST STRESS SUMMARY

Main Steam Line

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification[2] of Locations of Highest Stress Points
Based on ASME B&PV Code, Section III, ND-3600 1983 Edition through W84 Addenda, for SA-106 GR.B, S = 15000 PSI @ 470°F						
Design Condition: Eq 8 $\leq$1.5 S_h	Sustained Loads	7,129	22,500	0.317	1. Normal Loads 2. OBE	Joint 040 Line C
Service Levels A & B (Normal & Upset) Condition: Eq 9 $\leq$1.8 S_h	Occasional Loads	16,690	27,000	0.618	1. Normal Loads 2. OBE 3. Operating Transients	Joint 040 Line A
Eq 10 $\leq$Sa Eq 11 $\leq$Sa + S_h	Thermal Expansion	17,065 21,671	22,500 37,500	0.758 0.578	1. Normal Loads 2. OBE 3. Operating Transients	Joint 040 Line C Joint 040 Line C

TABLE 3.9-3k (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification[2] of Locations of Highest Stress Points
Service Level C (Emergency) Condition Eq. 9 ≤2.25 S_h	Primary Loads	16,471	33,750	0.488	1. Normal Loads 2. SBA 3. Operating Transients	Joint 040 Line A
Service Level D Condition: ASME Code Case 160 B-1 Eq. 9 ≤3.0 S_h	Primary Loads	17,323	45,000	0.385	1. Normal Loads 2. SSE 3. IBA 4. Operating Transients	Joint 040 Line A

NOTES:

[1] Appropriate loading combinations of <Table 3.9-3> were considered and calculation stresses are reported for the governing load combinations.

[2] Refer to <Figure 3.6-65b> and <Figure 3.6-65d> for the identification of node point numbers.

TABLE 3.9-3L

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, SA182 - F304 S_y = 30,000 psi				
2) Discharge valve stop and cylinder S_y = 30,000 psi head extension, SA 479-304				
3) Discharge valve cover, cylinder S_y = 30,000 psi head & stuffing box flange plate, SA 240-304				
4) Stuffing box gland, SA 564-630 S_y = 115,000 psi				
5) Studs, SA 193B7 S_y = 105,000 psi				
6) Dowel pins[2], alignment, S_A = 23,400 psi SAE 4140				
7) Studs, cylinder tie, SA 193-B7, S_A = 25,000 psi				
8) Pump holddown bolts, SAE GR.8, T_A = 12,000 psi O_A = 15,000 psi				
9) Power frame, foot area, cast iron S_A = 15,000 psi				
10) Motor holddown bolts, SAE Gr.1 T_A = 12,000 psi O_A = 15,000 psi				
11) Motor frame, foot area, cast iron S_A = 7,500 psi				
Normal & Upset Condition Loads:				
	1. Fluid cylinder	General Membrane	17,800	
1. Design pressure	2. Discharge valve stop	General Membrane	17,800	
2. Design temperature	3. Cylinder head extension	General Membrane	17,800	See Note[3]

TABLE 3.9-3L (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
3. Operating Basis Earthquake	4. Discharge valve cover	General Membrane	17,800	
4. Nozzle Loads[1]	5. Cylinder head	General Membrane	17,800	
	6. Stuffing box flange plate	General Membrane	17,800	
	7. Stuffing box gland	General Membrane	35,000	
	8. Cylinder head studs	Tensile	25,000	
	9. Stuffing box studs	Tensile	25,000	
Emergency or Faulted Condition[4]	1. Fluid cylinder	General Membrane	21,360	4,450
1. Design pressure	2. Discharge valve stop	General Membrane	21,360	13,600
2. Design temperature	3. Cylinder head extension	General Membrane	21,360	13,600
3. Weight of structure	4. Discharge valve cover	General Membrane	21,360	8,150
4. Thermal expansion	5. Cylinder head	General Membrane	21,360	8,150
5. Safe shutdown earthquake	6. Stuffing box flange plate	General Membrane	21,360	10,390
	7. Stuffing box gland	General Membrane	42,000	11,420
	8. Cylinder head studs	Tensile	25,000	18,820
	9. Stuffing box studs	Tensile	25,000	24,750
	10. Dowel pins[2]	Shear only[2]	23,400	17,550
	11. Studs, cylinder tie	Tensile[2]	25,000	18,760
	12. Pump holddown bolts	Shear	12,000	7,560
	13. Pump holddown bolts	Tensile	15,000	9,950
	14. Power frame-foot area	Shear	15,000	1,850
	15. Power frame-foot area	Tensile	15,000	11,390
	16. Motor holddown bolts	Shear	12,000	3,470
	17. Motor holddown bolts	Tensile	15,000	5,660
	18. Motor frame-foot area	Shear	7,500	2,550
	19. Motor frame-foot area	Tensile	7,500	5,100

TABLE 3.9-3L (Continued)

Nozzle Loads

Criteria	Loadings	Component	Allowable Nozzle Load	Actual Nozzle Load
The maximum forces	Normal/Upset	Pump C41C001A & B	Suction	
and moments due to	Normal		F_o = 770 lbs	Use of maximum
piping load combinations	OBE		M_o = 490 ft-lbs	values for F_i and
shall not exceed the	SRV discharge caused			M_i in interaction
allowable limits.	loads.		Discharge	equation[5] produces
			F_o = 370 lbs	results which are
			M_o = 110 ft-lbs	less than or equal
				to 1.0 for all
	Emergency/Faulted		Suction	nozzles under
	Normal		F_o = 920 lbs	specified
	SSE		M_o = 590 ft-lbs	loadings.
	SRV discharge caused			
	loads.			
	SBA/IBA/DBA		Discharge	
			F_o = 440 lbs	
			M_o = 130 ft-lbs	

NOTES:

[1] Nozzle load produce shear load only.
[2] Dowel pins take all shear loads.
[3] 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 are not evaluated.
[4] 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 3.9-3L (Continued)

NOTES: (Continued)

[5] Load Definition: Allowable combination of forces and moments is as follows:

$$\left[F_i / F_o\right] + \left[M_i / M_o\right] \leq 1.0$$

Where: F_i = The largest absolute value of any of the three actual external orthogonal forces (F_x, F_y or F_z) that may be imposed by the interface pipe (lbs).

M_i = The largest absolute value of any of the three actual external orthogonal moments (M_x, M_y or M_z) that may be imposed by the interface pipe (ft-lbs)

NOTE: The largest absolute values of F_i and M_i are combined simultaneously for specific condition.

F_o = Allowable value of F_i when all moments are zero (lbs).

M_o = Allowable value of M_i when all forces are zero (ft-lbs).

TABLE 3.9-3m

STANDBY LIQUID CONTROL TANK[1]

Component/Loading	Method of Analysis	Allowable Stress or Minimum Thickness Required or Load	Calculated Stress or Actual Thickness or Load
1. Shell Thickness			
Loads: Normal & Upset Design Pressure and Temperature	Brownell & Young "Process Equipment Design"		
Thickness	$t = \dfrac{PR}{SE - 0.6\,P}$		
Stress	Stress limits per ASME Section III	18,300 psi	1,203 psi
2. Anchor Bolts	ASME Section III	25,000 psi	4,277 psi
3. Nozzle Loads			

Criteria	Loadings	Component	Allowable Nozzle Load	Nozzle Load
The maximum forces and moments due to piping bad combinations shall not exceed the allowable limits	Normal/Upset Normal OBE SRV discharge caused loads.	Tank C41-A001	4" φ Discharge $F_o = 1{,}100$ lbs $M_o - 928$ ft-lbs	Use of maximum values for F_i and M_i in interaction equation[2] produces results which are less than or equal to 1.0 for all nozzles under specified loadings.

TABLE 3.9-3m (Continued)

Criteria	Loadings	Component	Allowable Nozzle Load	Nozzle Load
	Emergency/Faulted Normal SSE SRV discharge caused loads. SBA/IBA/DBA		4" ϕ Discharge F_o = 1,320 lbs M_o = 1,140 lbs	

NOTES:

[1] Design assessment shows stress resulting from additional loads being lower than originally calculated loads.

[2] Load Definition: Allowable combination of Forces and Moments is as follows:

$$[F_i/F_o] + [M_i/M_o] \leq 1.0$$

Where: F_i = The largest absolute value of any of the three that may be imposed by the interface pipe (lbs).

M_i = The largest absolute value of any of the three actual external orthogonal moments (M_x, M_y or M_z) that may be imposed by the interface pipe (ft-lbs).

NOTE: The largest absolute values of F_i and M_i are combined simultaneously for a specific condition.

F_o = Allowable value of F_i when all moments are zero (lbs).

M_o = Allowable value of M_i when all forces are zero (ft-lbs).

TABLE 3.9-3n

ECCS PUMPS

RESIDUAL HEAT REMOVAL PUMP[1]

Location	Loading Condition	Criteria	Calculated Stress (PSI) or Actual Thickness (in.)	Allowable Stress (PSI) or Min. Thickness (in.)
Discharge Head Shell	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Loads	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-27	11,460	21,000
Discharge Head Cover	Design Pressure	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-34, UG-39 & UG-40	2.55 in.	1.99 in.
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Load	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-37	(Suction) 28,066 (Discharge) 20,438	31,500
Discharge Pipe	FAULTED CONDITION Design Pressure Nozzle Loads	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-27	15,124	18,000
Discharge Head Bolting	FAULTED CONDITION	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, APP. II	37,465	50,000
Motor Bolting	FAULTED CONDITION Seismic Load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, APP. II	20,753	50,000

TABLE 3.9-3n (Continued)

LOW PRESSURE CORE SPRAY PUMP[1]

Location	Loading Condition	Criteria	Calculated Stress (PSI) or Actual Thickness (in.)	Allowable Stress (PSI) or Min. Thickness (in.)
Discharge Head Shell	FAULTED CONDITION Design Pressure Nozzle Loads SSE	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-27	10,615	21,000
Discharge Head Cover	Design Pressure	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-34, UG-39 & UG-40	2.29 in.	1.71 in.
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Nozzle Loads SSE	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-37	(Suction) 24,999 (Discharge) 22,050	31,500
Discharge Pipe	FAULTED CONDITION Design Pressure Nozzle Loads	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	17,920	18,000
Discharge Head Bolting	FAULTED CONDITION Design Pressure Nozzle Loads SSE	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	34,965	50,000
Motor Bolting	FAULTED CONDITION SSE	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	14,462	50,000

TABLE 3.9-3n (Continued)

HIGH PRESSURE CORE SPRAY PUMP[1]

Location	Loading Condition	Criteria	Calculated Stress (PSI) or Actual Thickness (in.)	Allowable Stress (PSI) or Min. Thickness (in.)
Discharge Head Shell	FAULTED CONDITION Design Pressure Nozzle Loads SSE	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	8,426	21,000
Discharge Head	Design Pressure	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-34, UG-39 & UG-40	2.67 in.	2.04 in.
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Nozzle Loads SSE	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-37	(Suction) 16,310 (Discharge) 18,160	31,500 31,500
Discharge Pipe	FAULTED CONDITION Design Pressure Nozzle Loads	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	10,175	21,500
Discharge Head Bolting	FAULTED CONDITION Design Pressure Nozzle Loads SSE	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	37,420	50,000
Motor Bolting	FAULTED CONDITION SSE	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	22,432	50,000

NOTE:

[1] Operability demonstrated by analysis.

TABLE 3.9-3o

(RHR) RESIDUAL HEAT REMOVAL HEAT EXCHANGER

Loading/Component	Criteria/Location	Allowable Stress or Min. Thickness Reqd.	Calculated Stress or Thickness
1. Closure Bolting Loads: Normal	Bolting Loads and Stresses calculated per "Rules for Bolted Flange Connections," ASME Section III, App. XI		
Design Pressure and Temperature Design Gasket Load			
Loads: Faulted	a. Shell to Tube Sheet Bolts	25,000 psi	24,950 psi
	b. Channel Cover Bolts	25,000 psi	24,390 psi
Design Pressure and Temperature, Dead Weight, Nozzle Loads, SSE, SRV, LOCA			
2. Wall Thickness	Shell Side ASME Section III, Class 2 and TEMA, Class C		
	Tube Side ASME Section III, Class 3 and TEMA, Class C		
Loads: Normal			
Design Presssure and Temperature	a. Shell	0.896 in.	1.0 in.
	b. Shell Cover	0.885 in.	0.885 in. min.
	c. Channel Ring	0.924 in.	1.0 in.

TABLE 3.9-3o (Continued)

Loading/Component	Criteria/Location	Allowable Stress or Min. Thickness Reqd.	Calculated Stress or Thickness
Stress Limit per ASME Section III	d. Tubes	0.050 in.	0.054 in. min.
	e. Channel Cover	8.11 in.	8.25 in
	f. Tube Sheet	7.08 in.	7.125 in.

3. Nozzle Loads

Criteria	Loadings	Allowable Nozzle Component	Actual Nozzle Load	Load[1]
The maximum forces and moments due to piping load combinations shall not exceed the allowable limits	Normal/Upset Normal OBE	E12B001A,B,C,D	See Note[2]	See Note[2]
Primary stress shall be the smaller of 0.70 S_u or 2.40 S per ASME Section III allowable.	Emergency/Faulted Normal SSE		See Note[2]	See Note[2]

NOTES:

[1] 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° azimuth.

[2] The calculated nozzle loads exceeded the analytical expression given in [3] below. However, a stress analysis was performed and the calculated nozzle loads still meet the allowable stresses of the heat exchanger.

TABLE 3.9-3o (Continued)

NOTES: (Continued)

[3] The following expression relates the allowable combination of forces and moments:

$$\left[\frac{F_i}{F_o}\right] \quad + \quad \left[\frac{M_i}{M_o}\right] \quad \leq 1$$

Where:

F_i = The largest of any of the three actual external orthogonal forces (F_x, F_y and F_z)

M_i = The largest of any of the three actual external orthogonal moments (M_x, M_y and M_z)

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 3.9-3p

DELETED

TABLE 3.9-3p (Continued)

DELETED

TABLE 3.9-3p (Continued)

DELETED

TABLE 3.9-3p (Continued)

DELETED

TABLE 3.9-3q

RCIC TURBINE

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
The highest stressed sections of the various components of the RCIC Turbine assembly are identified. Allowable stresses are based on ASME B&PV Code, Section III, for:				

Pressure Boundary Casting SA216-NCB: S = 14,000 psi

Pressure Boundary Bolting, SA193-B7   S = 25,000 psi

Alignment Dowel Pins:AISI 4037, Rc28-35
S_a = 61,000 psi
S_y = 106,000 psi

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Normal and Upset Condition Loads:	Casting: 1) Stop valve	General Membrane	14,000	
1. Normal Loads	2) Governor valve	General Membrane	14,000	
2. OBE	3) Turbine inlet	Local Bending	21,000	
	4) Turbine case	Local Bending	21,000	See Note[1]
	Pressure Containing Bolts:	Tensile	25,000	
	Structure Alignment Pins:	Shear	61,100	
Emergency or Faulted Condition[2]:	1. Stop valve	General Membrane	16,800	9,800
1. Normal Loads	2. Governor valve	General Membrane	16,800	13,200
2. SSE	3. Turbine inlet	Local Bending	25,200	15,300
	4. Turbine case	Local Bending	25,200	18,000
	Pressure Containing Bolts	Tensile	25,000	20,100
	Structure Alignment Pins	Shear	61,100	46,880

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TABLE 3.9-3q (Continued)

Nozzle Loads

Criteria	Loadings	Compound	Allowable Nozzle Load	Actual Nozzle Load
The maximum resultant force and resultant moment due to piping load combinations shall not exceed the allowable limit	Normal Normal	Turbine E51C002	Inlet F_o = 875 lbs M_o = 2,620 ft-lbs Exhaust F_o = 2,325 lbs M_o = 7,000 ft-lbs	Use of maximum resultant values for F_r and M_r in interaction equation[3] produces results which are less than or equal to 1.0 for all nozzles under specified loadings.
	Upset Normal OBE		Inlet F_o = 575 lbs M_o = 6,000 ft-lbs Exhaust F_o = 2,325 lbs M_o = 7,000 ft-lbs	
	Emergency/Faulted Normal SSE		Inlet F_o = 575 lbs M_o = 6,000 ft-lbs Exhaust F_o = 2,790 lbs M_o = 8,400 ft-lbs	

TABLE 3.9-3q (Continued)

NOTES:

[1] Calculated stresses for the emergency or faulted condition are lower than the allowable stresses for the normal plus upset condition; therefore, the normal and upset condition is not evaluated.

[2] Operability: Analysis indicates that shaft deflection with faulted loads is 0.006 inch, which is fully acceptable, and maximum bearing load with faulted condition is 80% of allowable. Furthermore, as indicated in <Section 3.9.2.2.2.9>, the turbine assembly has been seismically qualified via dynamic testing, enveloping the response spectra defined on <Figure 3.9-1> and <Figure 3.9-2>. This qualification included demonstration of startup and shutdown capabilities, as well as no load operability during seismic loading conditions.

[3] Load Definition: Allowable combination of forces and moments is as follows:

$$\left[F_r / F_o\right] + \left[M_r / M_o\right] \leq 1.0$$

Where: F_r = The resultant of the three absolute external orthogonal forces (F_x, F_y, F_z) that may be imposed by the interface pipe (lbs).

M_r = The resultant of the three absolute external orthogonal moments (M_x, M_y, M_z) that may be imposed by the interface pipe (ft-lbs).

NOTE: The absolute resultant values of F_r and M_r are combined simultaneously for a specific condition.

F_o =   Allowable value of F_r when all moments are zero (lbs).

M_o =   Allowable value of M_r when all forces are zero (ft-lbs).

TABLE 3.9-3r

RCIC 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 connection, SA105 GR. II S_y = 36,000 psi

4. Aligning pin, SA105 GR. II S_y = 36,000 psi

5. Closure bolting, SA193-87 S_y = 105,000 psi

6. Pump hold-down bolting, SA325 S_y = 77,000 psi

7. Taper pins, SA108 GR B1112, S_y = 75,000 psi

Normal & Upset Condition Loads:

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
1. Normal Loads	1. Forged barrel	General membrane	17,500	
2. OBE	2. End cover (Suction)	General membrane	17,500	
	3. End cover (Discharge)	General membrane	17,500	See Note[1]
	4. Nozzle reinforcement	General membrane	17,500	
	5. Alignment pin	Shear	18,000	
	6. Closure bolting	Tensile	25,000	
	7. Taper pine	Shear	15,000	
	8. Pump hold-down bolts	Tensile	40,000	

TABLE 3.9-3r (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
EMERGENCY OR FAULTED CONDITION LOADS[2]				
1. Normal Loads	1. Forged barrel	General membrane	21,000	13,300
2. SSE	2. End cover (suction)	General membrane	21,000	10,760
	3. End cover (discharge)	General membrane	21,000	15,900
	4. Nozzle reinforcement at barrel	General membrane	21,000	13,180
	5. Alignment pin	Shear	18,000	9,370
	6. Closure bolting	Tensile	25,000	21,000
	7. Taper pins (baring housing)		18,000	2,280
	8. Pump hold-down bolts	Tension	48,000	6,920

Nozzle Loads

Criteria	Loadings	Compound	Allowable Nozzle Load	Actual Nozzle Load
The maximum forces and moments due to piping load combinatins shall not exceed the allowable limits.	Normal/Upset Normal OBE SRV	Pump E51C001	Suction F_o = 1,940 lbs M_o = 2,460 ft-lbs Discharge F_o = 3,715 lbs M_o = 2,950 ft-lbs	use of maximum values for F_i and M_i in interaction equation[3] produces results which are less than or equal to 1.0 for all nozzles under
	Emergency/Faulted Normal SSE SRV SBA/IBA/DBA		Suction F_o = 2,325 lbs M_o = 2,950 ft-lbs Discharge F_o = 4,450 lbs M_o = 5,200 ft-lbs	specified loadings

TABLE 3.9-3r (Continued)

NOTES:

(1) Calculated stresses for emergency or faulted condition are less than the allowable for normal plus upset condition; therefore, the normal and upset condition are not evaluated.

(2) Operability: Analyses for emergency or faulted condition show that the maximum shaft deflection is 0.002 in. with 0.006 in. allowable, shaft stresses are 3080 psi with 25000 psi allowable, and bearing loads of drive end 98 lb with 7670 lb allowable and thrust end 765 lb with 17600 lbs allowable.

(3) Load Definition: Allowable combination of forces and moments is as follows:

$$\left[F_i / F_o\right] + \left[M_i / M_o\right] \leq 1.0$$

Where: $F_i =$ The largest absolute value of any of the three actual external orthogonal forces (F_x, F_y or F_z) that may be imposed by the interface pipe (lbs).

$M_i =$ The largest absolute value of any of the three actual external orthogonal moments (M_x, M_y, M_z) that may be imposed by the interface pipe (ft-lbs).

NOTE: The largest absolute values of F_i and M_i are combined simultaneously for a specific condition.

$F_o =$ Allowable value of F_i when all moments are zero (lbs).

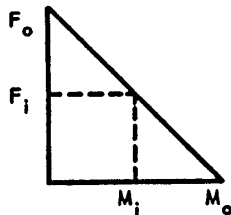$M_o =$ Allowable value of M_i when all forces are zero (ft-lbs).

TABLE 3.9-3s

REACTOR REFUELING AND SERVICING EQUIPMENT

Refueling Platform[1]

Acceptance Criteria	Complete	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
The allowable axial plus bending loads' stresses are based on AISC Part 5 Section 1.5 for ASTM A 36 Structural Steel F_u = 58,000 psi F_y = 36,000 psi				
For Normal Condition: $S_{limit} = 0.66F_y$	For Normal Condition: 1. Normal loads	Axial load plus bending	23,760	18,162
For Upset Condition: $S_{limit} = 0.9F_y$	For Upset Condition: 1. Normal loads 2. OBE	Axial load plus bending	32,400	30,194
For Faulted Condition: $S_{limit} = 0.7\ F_u$	For Faulted Condition: 1. Normal loads 2. SSE	Axial load	40,600	35,860

TABLE 3.9-3s (Continued)

Fuel Preparation Machine[1]

Acceptance Criteria	Component	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
The allowable axial load's stress is based on ASME Code Section III, ASTM A 167 for Type 302 Annealed S.S.				
F_y = 30,000 psi				
F_u = 75,000 psi				
S_m = 17,800 psi				
For normal condition: $S_{limit} = S_m$	For normal condition: 1. Normal loads	Axial load	17,800	16,030
For upset condition: $S_{limit} = 0.8 \ F_y$	For upset condition: 1. Normal loads 2. OBE	Axial load	24,000	16,148[2]
For faulted condition: $S_{limit} = 1.2 \ F_y$	For faulted condition: 1. Normal loads 2. SSE	Axial load	36,000	16,132[2]

TABLE 3.9-3s (Continued)

Inclined Fuel Transfer Tube[3]

Acceptance Criteria	Loading	Limiting Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
The allowable axial plus bending loads are based on ASME Section III, Class 2, SA182 for Type 304 stainless steel $F_u = 75,000$ psi $F_y = 30,000$ psi				
For Normal Condition: $S_{limit} = 1.5 \times 0.6 \times F_y$	For Normal Condition: 1. Normal Loads	Axial load plus bending	27,450	7,903
For Emergency Condition: $S_{limit} = 1.5 \times 0.9 \times F_y$	For Emergency Condition: 1. Normal Loads 2. OBE 3. Safety/Relief Valve Discharge Caused Loads	Axial load plus bending	41,175	27,289
For Faulted Condition: $S_{limit} = 1.5 \times 1.2 \times F_y$	For Faulted Condition: 1. Normal Loads 2. SSE 3. Safety/Relief Valve Discharge Caused Loads	Axial load plus bending	53,850	36,908

TABLE 3.9-3s (Continued)

Fuel Assembly (Including Channel)[4]

Acceptance Criteria	Loading	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis Acceleration
Acceleration Envelope	Horizontal Direction:	Horizontal Acceleration	1.9 G	See Note[5]
	1. Peak pressure			
	2. SSE			
	3. DBA			
	Vertical Direction:	Vertical Acceleration	3.5 G	See Note[5]
	1. Peak pressure			
	2. SSE			
	3. DBA			

NOTES:

[1] New loads have no impact on this component.
[2] The faulted condition load is smaller than upset because of the damping difference. Upset uses 2%, faulted uses 4%.
[3] Operability assurance is demonstrated by analysis.
[4] The calculated maximum fuel assembly gap opening for the most limiting load combination is 0.038 inch.
[5] Acceleration Bases and Evaluation Bases are contained in NEDE-21175-3-P.

<TABLE 3.9-3t>

DELETED

TABLE 3.9-3u

CONTROL ROD DRIVE

Main Flange:[2]

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for Type F304 Stainless Steel @ 575°F S_m = 16,700 psi				
For Normal and Upset Condition: S_{allow} = 1.5 x S_m	For Normal & Upset Condition: 1. Normal Loads[1] 2. OBE 3. DBA	General Membrane plus Bending	25,000	5,813
For Emergency Condition: S_{allow} = 1.8 x S_m	For Emergency Condition: 1. Normal Loads[1] 2. DBA	General Membrane plus Bending	30,000	4,300
For Faulted Condition: S_{allow} = 3.6 x S_m	For Faulted Condition: 1. Normal Loads[1] 2. SSE 3. DBA	General Membrane plus Bending	60,000	7,294

TABLE 3.9-3u (Continued)

Ring Flange:[2]

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for Type 304 Stainless Steel @ 250°F S_m = 20,000 psi				
For Normal and Upset Condition: S_{allow} = 1.5 x S_m	For Normal & Upset Condition: 1. Normal Loads[1] 2. Scram with OBE and no buffer	General Membrane + Bending	30,000	10,961
For Emergency Condition: S_{allow} = 1.8 x S_m	For Emergency Condition: 1. Normal Loads[1] 2. Scram with accumulator at over-pressure	General Membrane + Bending	36,000	1,838
For Faulted Condition: S_{allow} = 3.6 x S_m	For Faulted Condition: 1. Normal Loads[1] 2. Scram with SSE 3. Scram with stuck rod	General Membrane + Bending	72,000	4,041

TABLE 3.9-3u (Continued)

Indicator Tubes:

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for Type 316 Stainless Steel @ 250°F S_m = 20,000 psi				
For Normal and Upset Condition:[3] S_{allow} = 1.5 x S_m	For Normal & Upset Condition: 1. Normal Loads[1] 2. CRD Shim/Jog	General Membrane plus Bending	30,000	15,939
For Emergency Condition:[3] S_{allow} = 1.8 x S_m	For Emergency Condition: 1. Normal Loads[1] 2. Failure of pressure regulating system 3. Scram with accumulator at over-pressure	General Membrane plus Bending	36,000	23,826
For Faulted Condition: S_{allow} = 3.6 x S_m	For Faulted Condition: 1. Normal Loads[1] 2. Scram with SSE 3. Chugging and LOCA 4. SRV Discharge Caused Loads	General Membrane plus Bending	72,000	16,100

TABLE 3.9-3u (Continued)

NOTES:

[1] New loads do not significantly contribute to the flange stresses.
[2] Normal loads include pressure, temperature, weight, and mechanical loads.
[3] For Normal, Upset and Emergency conditions, the calculated stresses due to the hydrodynamic loads are less than the originally calculated stresses.

TABLE 3.9-3v

CONTROL ROD DRIVE HOUSING

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, for Class I vessels, for Type 304 stainless steel.				
For Normal and Upset Condition: $S_{limit} = 1.0 \, S_m =$ 16,660 psi @ 575°F	Normal and Upset Condition: 1. Design pressure 2. Stuck rod scram loads 3. Operational basis earthquake, with housing lateral support installed	Maximum membrane stress intensity occurs at the tube to the tube weld near the center of the housing for normal, upset and emergency conditions.	16,660	13,677
For Faulted Conditions[1]: $S_{limit} = 1.2 \, S_m$	Faulted Condition Loads: 1. Design pressure 2. Stuck rod scram loads 3. Safe shutdown earth-quake, with housing lateral support installed		20,000	14,727

NOTES:

[1] Analyzed to emergency conditions limits.
[2] The new loads calculated stresses are lower than the original design calculations.

TABLE 3.9-3w

JET PUMPS

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Primary Membrane Plus Bending Stress based on ASME B&PV code Section III 304SS				
For Service Levels A & B (Normal and Upset) condition: For Type 304S.S. @ 550°F $S_m = 16,900$ psi $S_{limit} = 3.0\ S_m$	Service Level A & B (Normal & Upset) 1. Normal loads 2. OBE 3. SRV Discharge Caused Loads	Primary Membrane plus Secondary Membrane and Thermal Bending	50,700	15,350
For Service Level C (Emergency) Condition: For Type 304S.S @ 550°F $S_m = 16,900$ psi $S_{limit} = 1.8\ S_m$	Service Level C (Emergency) Loads 1. Normal Loads 2. OBE 3. SRV Discharge Caused Loads	Primary Membrane plus Bending	30,420	10,220
For Service Level D (Faulted) Condition: For Type 304 S.S. @ 550°F $S_m = 16,900$ psi $S_{limit} = 3.6\ S_m$	Service Level D (Faulted) Loads 1. Normal Loads 2. SSE 3. Annulus Pressurization 4. SRV Discharge Caused Loads	Primary Membrane plus Bending	60,840	52,120

<TABLE 3.9-3x>

DELETED

TABLE 3.9-3y

HIGHEST STRESSED REGION ON THE LPCI COUPLING (ATTACHMENT RING)

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Primary Membrane Plus Bending stress based on ASME B&PV code Section III for Type 316 L, NG - 3,000 Sm @ 528°F = 14,150 psi				
For Service Levels A & B (Normal & Upset) Condition: $S_{limit} = 1.5 \times .7S_m$	Normal and Upset Loads: 1. Normal Loads 2. OBE 3. Safety/Relief Valve Discharge Caused Loads	Primary Membrane + Bending	14,858	14,033
For Service Level C (Emergency) Condition: $S_{limit} = 2.25 \times .7S_m$	Emergency Loads 1. Normal loads 2. SRV (ADS)	Primary membrane + Bending	22,290	19,254
For Service Level D (Faulted) Condition: $S_{limit} = 2.4(1.5)(.7)\ S_m$	Faulted Loads 1. Normal loads 2. SSE 3. DBA	Primary Membrane + Bending	35,658	34,807

<TABLE 3.9-3z>

DELETED

TABLE 3.9-3aa

CONTROL ROD GUIDE TUBE[1][2]

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
CONTROL ROD GUIDE TUBE				
Primary Stress Limit				
The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section III, Class CS for Type 304 stainless steel material				
For Service Levels A & B (Normal and Upset) Condition: 1.5 S_m = 1.5 x 16,000 = 24,000 psi	Service Levels A and B (Normal and Upset Condition) Applied Loads 1. External Pressure 2. Vertical Seismic and weight 3. Horizontal Seismic 4. Lateral Flow impingement 5. Vibration	Applying Vertical Seismic plus dead weight, the maximum stress under Service Levels A & B (normal and upset) conditions occurs at the guide tube base.	24,000	16,340

TABLE 3.9-3aa (Continued)

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
For Service Level D (Faulted) Condition: S_{limit} = 2.4 S_m = 2.4 x 16,000 = 38,400 psi	Service Level D (Faulted) Condition Applied Loads 1. External Pressure 2. Vertical Seismic and Weight 3. Horizontal Seismic 4. Lateral Flow Impingement 5. Vibration	Applying Vertical Seismic plus dead weight, the maximum stress under Service Level D (faulted) loading conditions occurs at the guide tube base.	38,400	21,763

NOTES:

[1] New loads do not significantly affect flange stresses.
[2] Calculated stresses for Emergency condition are lower than Normal and Upset.

TABLE 3.9-3ab

INCORE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Primary Stress Limit - The allowable primary membrane stress is based on ASME Boiler and Pressure Vessel Code, Section III for Class I vessels for type Inconel 600 austenitic high nickel alloy steel				

For Service Levels A & B (Normal and Upset) Condition: $S_{limit} = 1.0S_m$
 	
    = 23,300 psi
      at 575°F
	Service Level A & B (Normal and Upset) Condition Loads
	1. Normal Loads
	2. OBE
	3. Safety/Relief Valve Discharge Caused Loads
		Maximum membrane Stress intensity: occurs at the outer surface of the vessel penetration	23,300	18,055

Service Level D (Faulted) Condition Stress limit is the lesser of   0.7 S_u
    =0.7 x 80,000
    =56,000
or   2.4 S_m
    =2.4 x 23,000
    =55,920
	Service Level D (Faulted) Condition Loads
	1. Normal Loads
	2. SSE
	3. DBA
		General Membrane + Bending	55,920	21,026

TABLE 3.9-3ac

REACTOR VESSEL SUPPORT EQUIPMENT

Orificed Fuel Supports[2]

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Based on ASME Code Section III Subsection NG.				
S_m @ 575°F = 15,580 psi				
For Normal and Upset Condition:	Normal and Upset Loads			
S_{limit} = 1.0 S_m	1. Normal Loads 2. OBE	Primary Membrane[1]	15,580	12,413[1]
For Emergency Condition:	Emergency Loads See Note[3]			
For Faulted Condition: S = 2.4 S_m	Faulted Loads: 1. Normal Loads 2. SSE	Primary Membrane[1]	35,440	23,505[1]

TABLE 3.9-3ac (Continued)

CRD Housing Supports - Grid Structure

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Based on AISC Specification for the design, fabrication and erection of structural steel for buildings. F_y @ 150°F = 46,000 psi				
For Normal and Upset Condition:	Normal and Upset Loads:			
$f_a = 0.60\ F_y^{(4)}$ (Tension)	Negligible[5]		N/A	N/A
$f_b = 0.66\ F_y^{(4)}$ (Bending)				
$f_v = 0.40\ F_y^{(4)}$ (Shear)				
For Emergency Condition:	Emergency Loads:			
	Negligible[5]			
For Faulted Condition: $f_{a(limit)} = 1.5 \times 0.60 \times F_y^{(4)}$ (Tension)	Faulted Loads: 1. Weight of structure	Bending	41,500	f_b = 40,500
$f_{b(limit)} = 1.5 \times 0.60 \times F_y^{(4)}$ (Bending)	2. Impact force from failure of CRD	Shear	27,500	f_v = 11,600
$f_{v(limit)} = 1.5 \times 0.40 \times F_y^{(4)}$ (Shear).				

TABLE 3.9-3ac (Continued)

CRD Housing Supports - Beams

Criteria	Loading	Primary Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Based on AISC Specification for the design, fabrication and erection of structural steel for buildings.				
F_y @ 50°F = 36,000 psi				
For Normal and Upset Condition: $F_a = 0.60F_y^{(4)}$ (Tension)	Normal and Upset Loads:			
$F_b = 0.60F_y^{(4)}$ (Bending)	Negligible[5]			
$F_v = 0.40F_y^{(4)}$ (Shear)				
For Emergency Condition:	Emergency Loads:			
	Negligible[5]			
For Faulted Condition: $f_{a(limit)}$ = 1.5 x 0.60 x $F_y^{(4)}$ (Tension)	Faulted Loads: 1. Weight of structure 2. Impact force from	Top Chord	33,000	F_a = 14,000
$f_{b(limit)}$ = 1.5 x 0.60 x $F_y^{(4)}$ (Bending)	failing of CRD Housing	Top Chord	33,000	f_b = 20,000
$f_{v(limit)}$ = 1.50 x 0.40 x $F_y^{(4)}$ (Shear)		Bottom Chord	33,000	f_a = 1,700
			33,000	f_b = 21,000

TABLE 3.9-3ac (Continued)

NOTES:

(1) ASME Section 3 NG-3228.4 defines the use of test to show adequacy. NG3228.4 does not require the separation of primary and secondary stresses. However, because the summation of stresses determined for test for upset and faulted conditions are less than allowable, the 12,413 psi for upset and 23,505 psi for faulted are reported in the code stress report.

(2) New Load adequacy was assumed by the above testing and experience.

(3) Calculated stresses for emergency condition are lower than Normal and Upset.

(4) F_y = Material yield strength.

(5) All other loads are very small compared to impact force.

TABLE 3.9-3ad

<u>HYDRAULIC CONTROL UNIT</u>

This Component has been qualified by test.

<TABLE 3.9-3ae>

DELETED

TABLE 3.9-3af

HIGH PRESSURE CORE SPRAY SYSTEM ASME CODE CLASS 1 VALVE

COMPONENT/LOAD/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE	RATIO CALCULATED/ ALLOWABLE
1.0 Body and Bonnet[2]				
1.1 Loads: Design Pressure Design Temperature Pipe Reaction Thermal Effects	System Requirement System Requirement Not Specified Not Specified	1,575 psi 575°F	1,575 psi 575°F	N/A N/A
1.2 Pressure Rating, 655 psi	ASME Section III[1], Figure NB 3545.1-2	P_r = 655 psi	P_r =655 psi	N/A
1.3 Minimum Wall Thickness, inches	ASME Section III[1], Paragraph NB-3542	t(nominal) = 1.089 inches	t_m =1.089 min inches	N/A
1.4 Primary Membrane Stress, psi	ASME Section III[1], Paragraph NB-3545-1	$P_m < S_m$(500°F) = 19,400 psi	P_m =7,629 psi	0.39
1.5 Secondary Stress Due to Pipe Reaction	ASME Section III[1], Paragraph NB-3545.2.(b)(1) (S = 30,000 psi)	P_e=Greatest value of P_{ed}, P_{eb} and P_{et} ≤1.5 S_m(500°F) 1.5(19,400) = 29,100 psi	P_{ed} =5,942 psi P_{db} =10,352 psi P_{et} =10,352 psi P_e=P_{et} =10,352 psi	0.20 0.36 0.36 0.36
1.6 Primary Plus Secondary Stress Due to Internal Pressure	ASME Section III[1], Paragraph NB-3545.2(a)(1)	N/A	Q_p = 21,902 psi	N/A

TABLE 3.9-3af (Continued)

COMPONENT/LOAD/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE	RATIO CALCULATED/ ALLOWABLE
1.7 Thermal Secondary Stress	ASME Section III[1], Paragraph NB-3545.2(c)	N/A	QT =740 psi	N/A
1.8 Sum of Primary Plus Secondary Stress	ASME Section III[1], Paragraph NB-3545.2	Sn ≤3.0*Sm(500°) = 3(19,400) = 58,200 psi	Sn =Op+Pe+2QT =28,124 psi	0.48
1.9 Fatigue Requirements	ASME Section III[1], Paragraph NB-3545.3	Na ≥2,000 cycles	Na =7x10^4 cycles	N/A
1.10 Cyclic Rating	ASME Section III[1], Paragraph NB-3550	IT ≤1.0	It =0.71	0.71
2.0 Body to Bonnet Bolting				
2.1 Loads: Design Pressure and Temperature, Gasket Loads, Stem Operational Load, Seismic Load (Design Basis Earthquake)	ASME Section III[1], Paragraph NB-3647.1			
2.2 Bolt Area	ASME Section III[1], Paragraph NB-3647.1	Ab ≥18.83 in.^{2} Sb ≤29,500 psi	Ab =18.89 in.^{2} Sb =29,500 psi	N/A N/A

TABLE 3.9-3af (Continued)

COMPONENT/LOAD/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE	RATIO CALCULATED/ ALLOWABLE
2.3 Body Flange Stresses	ASME Section III[1], Paragraph NB-3647.1			
2.3.1 Operating Conditions		$S_m \leq 1.5\ S_m(575)$ = 29,100 psi	S_m =22,680 psi	0.78
		$S_r \leq 1.5\ S_m(575)$ = 29,100 psi	S_r =15,940 psi	0.55
		$S_t \leq 1.5\ S_m(575)$ = 29,100 psi	S_t =4,950 psi	
2.3.2 Gasket Seating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_m \leq 1.5\ S_m$ (100°F) = 34,950 psi	S_m =20,030 psi	0.57
		$SR \leq 1.5\ S_m$ (100°F) = 34,950 psi	SR =13,560 psi	0.39
		$S_T \leq 1.5\ S_m$ (100°F) = 34,950 psi	S_T =4,220 psi	0.12

TABLE 3.9-3af (Continued)

COMPONENT/LOAD/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE	RATIO CALCULATED/ ALLOWABLE
2.4 Bonnet Flange Stresses				
2.4.1 Operating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_m \leq 1.5\ S_m$ (500°F) = 29,100 psi	S_m =22,300 psi	0.77
		$S_R \leq 1.5\ S_m$ (500°F) = 29,100 psi	S_R =17,580 psi	0.60
		$S_T \leq 1.5\ S_m$ (500°F) = 29,100 psi	S_T =850 psi	0.03
2.4.2 Gasket Seating Condition	ASME Section III[1], Paragraph NB-3647.1	$S_m \leq 1.5\ S_m$ (500°F) = 29,100 psi	S_m =19,430 psi	0.67
		$S_R \leq 1.5\ S_m$ (500°F) = 29,100 psi	S_R =14,960 psi	0.51
		$S_T \leq 1.5\ S_m$ (500°F) = 29,100 psi	S_T =720 psi	0.02

TABLE 3.9-3af (Continued)

COMPONENT/LOAD/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE	RATIO CALCULATED/ ALLOWABLE
3.0 Stresses in Stem				
3.1 Loads: Operator Thrust and Torque				
3.2 Stem Thrust Stress	Calculate Stress Due to Operator Thrust in Critical Cross Section	$S_T < S_m$ = 22,000 psi	S_T =10,700 psi	0.49
3.3 Stem Torque Stress	Calculate Shear Stress due to Operator Torque in Critical Cross Section	$S_S \leq 0.6\ S_m$ = 13,000	S_S =10,010 psi	0.76
3.4 Buckling on Stem	Calculate Slenderness Ratio. If greater than 30, calculate allowable load from Rankine's Formula using safety factor of 4.	Max allowable load = 69,000 lbs	Slenderness ratio = 28 Actual load on Stem = 33,600 lbs Therefore, no buckling.	0.49
4.0 Disc Analysis				
4.1 Loads: Maximum Differential Pressure (2)	-	-	-	-

TABLE 3.9-3af (Continued)

COMPONENT/LOAD/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE	RATIO CALCULATED/ ALLOWABLE
4.2 Maximum Stress in the Disc	ASME Section III[1], Paragraph NB-3215 and ASME Section III[1], Paragraph NB-3227.3	$S_{max} \leq 0.6 S_m$ at (500°F) = 11,680 psi	Max Stress = 3,160 psi	0.27
5.0 Yoke and Yoke Connections			N/A	N/A
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		$S_{max} \leq S_m$ (100°F) = 35,000 psi	$S_{max} = 17,936$ psi	0.51
5.3 Bending Stress of Yoke	-	$S_b \leq 1.5 \, S_b$ (100°F) = 34,950 psi	$S_b = 15,450$ psi	0.44

NOTES:

[1] ASME Section III Edition
[2] Valve Differential Pressure is 1,575 psig

TABLE 3.9-4

SEISMIC AND HYDRODYNAMIC QUALIFICATION SUMMARY OF MAJOR SAFETY-RELATED MECHANICAL EQUIPMENT

COMPONENT	QUALIFICATION METHOD[1]	INPUT MOTION[2]	QUALIFICATION SUMMARY[3]
Jet Pump		See Note[4]	3.9-3w
Control Rod Drive		See Note[4]	3.9-3u
CRD Housing		See Note[4]	3.9-3v
CR Guide Tube		See Note[4]	3.9-3aa
Orificed Fuel Support	T	See Note[4]	3.9-3ac
Main Steam Isolation Valve	T/A		3.9-3h
Main Steam Safety/ Relief Valve	T/A	6.5g Horizontal 4.5g Vertical	3.9-3g
Recirculation Pump and Motor	A	4.5g Horizontal 3g Vertical	3.9-3i
Hydraulic Control Unit	T	<Section 3.9.2.2.3.d>	3.9-3ad
SLC Pump and Motor	A	1.75g Horizontal	3.9-3L
SLC Tank	A	1.75g Horizontal 1.75g Vertical	3.9-3m

TABLE 3.9-4 (Continued)

COMPONENT	QUALIFICATION METHOD[1]	INPUT MOTION[2]	QUALIFICATION SUMMARY[3]
RHR Pump and Motor	T/A	0.7g Vertical 3g Horizontal	3.9-3n
RHR Heat Exchanger	A	\<Figure 3.9-3\> \<Figure 3.9-4\>	3.9-3o
LPCS Pump and Motor	T/A	0.7g Vertical 3.g Horizontal	3.9-3n
RCIC Pump Assembly	A	1.5g Horizontal 1.5g Vertical	3.9-3r
RCIC Turbine	T/A	\<Figure 3.9-1\> \<Figure 3.9-2\>	3.9-3q
Fuel Assembly (including channel)		\<Table 3.9-3s\>[4]	3.9-3s

NOTES:

[1] Qualification Method: T = Testing
                          A = Analysis
                          T/A= Testing/Analysis
[2] Input Motion: Response spectrum is column entry is a figure number
                   Static g-level is column is numeric.
[3] Qualification Summary: Refer to applicable table number.
[4] Multi - response spectrum input to equipment support points are based on Rector Pressure Vessel and reactor building dynamic model analysis.

<TABLE 3.9-5>

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TABLE 3.9-6

<u>SAFETY-RELATED CRANE</u>
<u>LOAD COMBINATIONS AND STRESS LIMITS</u>

Load Combination	Type of Stress	Allowable Stress for Structural Support
Normal Plus OBE	All	AISC Code[1], Sections 1.5 through 1.10, except that the 1/3 increase permitted by Section 1.5.6 for the wind and seismic effects was not used.
Faulted Plus SSE	Bending, Tension, Compression	Smaller of: 1.5 x AISC Code[1] or 0.9 F_y
	Shear	Smaller of: 1.5 x AISC Code[1] or 0.5 F_y
	Bearing	1.33 x AISC Code[1], Section 1.5.1.5.1, Section 1.5.1.5.2 and Section 1.5.2 only

<u>NOTE</u>:

[1] American Institute of Steel Construction, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," 1969. The AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1978, is used for the A325/A490 bolt evaluation for the Fuel Handling Building Crane and the Reactor Building Polar Crane.

TABLE 3.9-7

SAFETY-RELATED VENTILATION EQUIPMENT

| | Safety | |
Vendor	Class	Description
Westinghouse	2	Annulus Exhaust Gas - Fan (Typ of 4)
	3	Miscellaneous Electrical - Supply Fan (Typ of 2)
	3	Miscellaneous Electrical - Return Fan (Typ of 2)
	3	Battery Room - Exhaust Fan (Typ of 2)
	3	Control Room - Supply Fan (Typ of 2)
	3	Control Room - Return Fan (Typ of 2)
	3	Control Room - Emergency Recirculation Fan (Typ of 2)
	3	Offgas Building - Exhaust Fan (Typ of 4)
	3	Fuel Handling Building - Supply Fan (Typ of 2)
	3	Fuel Handling Building - Exhaust Fan (Typ of 3)
CVI Corp.	2	Annulus Exhaust Gas - Plenum (Typ of 4)
	3	Control Room - Emergency Recirculation Plenum (Typ of 2)
	3	Offgas Building - Exhaust Plenum (Typ of 4)
	3	Fuel Handling Building - Exhaust Plenum (Typ of 3)
American Air Filter	3	Motor Control Center/Switchgear and Miscellaneous Electrical - Plenum (Typ of 2)
	3	Control Room - Supply Plenum (Typ of 2)
	3	Fuel Handling Building - Supply Plenum (Typ of 2)
Carrier	3	Emergency Closed Cooling Pump Area - Air Handling Unit (Typ of 2)
	3	ECCS Pump Rooms RHR A, B, C, RCIC, LPCS & HPCS Air Handling Units (Typ of 12)

TABLE 3.9-7 (Continued)

Vendor	Safety Class	Description
Carrier (Continued)	3	Control Complex Chilled Water System - Water Chiller (Typ of 3)
Buffalo Forge Co.	3	Emergency Service Water Pump House - Fan (Typ of 4)
	3	Diesel Generator Building - Fan (Typ of 12)

TABLE 3.9-8

FAN COMPONENT FACTORS OF SAFETY[1][2]

Fan Tag Number	Hold-Down Bolts Tension		Hold-Down Bolts Shear		Overturn Moment Tension		Bearing Bolts Tension		Bearing Bolts Shear		Shaft		Motor Bolts[3] Tension		Motor Bolts[3] Shear		Wheel Welds or Rivets		Bracing Weld		Bracing	
	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2	FS1	FS2
0-M23C0001A,B	20	17	11	7.8	4.6	3.5	36	31	17	12	13	7.8	26	18	12	7.5	6.4	4.2	33	24	2	1.3
0-M23C0002A,B	26	22	14	10	6.4	4.9	56	48	27	19	14	8.5	26	18	12	7.5	7.1	4.7	33	24	2	1.3
0-M24C0001A,B	162	139	88	63	44	34	187	160	86	60	18	11	36	26	17	11	9.4	6.1	184	130	14	9.3
0-M25C0001A,B	64	55	35	25	14	11	85	72	39	27	13	7.9	54	46	29	21	7.4	4.8	47	33	2.3	1.5
0-M40C0001A,B	396	408	122	111	63	60	486	500	101	76	20	11	353	364	108	99	7.3	4.7	250	209	8.8	7.3
1-M15C0001A,B	282	252	93	78	98	86	923	833	300	256	82	46	346	308	111	96	8.7	4.4	61	34	16	13
0-M26C0001A,B	46	39	25	18	15	11	117	100	53	36	25	15	54	46	29	21	14	7	82	58	3.6	2.4
2-M36C0001A,B	245	263	31	25	28	24	224	239	23	17	20	12	298	317	38	31	16	8.2	113	85	4.6	3.4
0-M40C0002A,B	132	90	41	34	26	20	110	75	26	20	20	11	74	45	20	13	16	8.2	137	97	6.2	4.6

NOTES:

[1] FS1 = Yield stress/SSE stress

[2] FS2 = Allowable working stress/OBE stress

[3] Factors of Safety for motor bolts have been linearly decreased to reflect the increased replacement motor weights, as applicable.

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TABLE 3.9-9

WESTINGHOUSE FAN MOTORS STRESS COMPARISON[1]

Component	Calculated Stress (psi)	Safety Limit Stress (psi)
Motor Mounting Bolts	788	9,200
Cantilever Foot	1,082	5,000
Inner Cap Screws	171	3,900
Top Cover Screws	19.5	2,125
Conduit Box Mounting Screws	18.7	2,125
Conduit Box Mounting - Pipe Nipple	None	None
Front End Bearing	568	13,500
Back End Bearing	2,030	13,500

NOTE:

[1] Results are from Reliance Electric Report No. 78-1-36 for the 100 HP motor originally
     furnished for the Westinghouse Sturdevant Fans. Replacement motors are seismically
     qualified separately in accordance with <IEEE Standard 344>, but the results are not
     reflected in this table.

TABLE 3.9-10

RESULTS OF PLENUM STRESS ANALYSIS FOR CVI CORP. FILTER UNITS

			Plenum Tag Number			
			1M15-D001A,B; 2M15-D001A,B	M26-D001A,B	1M36-D001A,B; 2M36-D001A,B	M40-D001A,B,C
Filter Adsorber Stress	-	OBE	11,195	11,731	11,532	11,547
	-	DBE	11,639	12,072	11,645	11,661
Plenum Housing Stress	-	OBE	344	2,140	2,520	1,925
	-	DBE	487	2,977	3,188	2,292
Housing Stress at Anchors	- Bending		14,122	7,044	14,382	2,613
	- Allowable		32,400	32,400	32,400	32,400
Anchor Bolt Stress	-	Shear	5,168	8,086	8,473	7,989
	-	Allowable Shear	15,000	15,000	15,000	15,000
Anchor Bolt Stress	-	Axial	3,651	1,821	3,719	676
	-	Allowable Axial	32,400	32,400	32,400	32,400

TABLE 3.9-11

RESULTS OF ANALYSIS FOR PLENUM ANCHORING

	OBE Allowable	OBE Calculated	SSE Allowable	SSE Calculated
Fuel Handling Building Supply Plenum				
Tensile Force (pounds)	12,110	794	18,165	1,423
Shear Force (pounds)	7,850	1,408	11,775	1,738
Motor Control Center/Switchgear and Miscellaneous Electrical Area Plenum and Control Room Supply Plenum				
Tensile Force (pounds)			18,165	9,865
Shear Force (pounds)			11,775	7,554

TABLE 3.9-12

CARRIER AIR HANDLING UNIT STRESS COMPARISON

	Air Handling Unit Size		
	39ED15	39ED18	39BA050
Beam Stress - Membrane and Bending - Calculated, psi	12,496	19,683	10,491
- Allowable, psi	18,000	27,000	18,000
Beam Stress - Shear - Calculated, psi	2,520	3,520	1,481
- Allowable, psi	15,000	15,000	15,000
Plate Stress - Membrane and Bending - Calculated, psi	3,764	4,994	7,193
- Allowable, psi	27,000	27,000	27,000
Plate Stress - Shear - Calculated, psi	1,780	2,348	5,413
- Allowable, psi	15,000	15,000	15,000
Anchor Bolts - Tension - Calculated, psi	3,505	3,591	2,262
- Allowable, psi	27,000	27,000	27,000
Anchor Bolts - Shear - Calculated, psi	2,280	2,012	1,528
- Allowable, psi	15,000	15,000	15,000
NF Analysis			
Maximum Plate Stress - Calculated, psi	3,764	4,994	7,193
- Safe Limit, psi	18,500	18,500	18,500
Maximum Beam Axial Stress - Calculated, psi	540	470	1,655
- Safe Limit, psi	18,400	18,400	16,200
Maximum Beam Shear - Calculated, psi	1,640	960	222
- Safe Limit, psi	16,000	16,000	12,000

TABLE 3.9-12 (Continued)

	Air Handling Unit Size		
	39ED15	39ED18	39BA050
Maximum Beam Bending Stress - Calculated, psi	3,240	10,030	5,075
- Safe Limit, psi	24,000	24,000	18,000
Maximum Weld Contact Stress - Calculated, psi	1,557	<1,557	8,416
- Safe Limit, psi	9,000	9,000	9,000
Maximum Weld Threat Stress - Calculated, psi	429	<429	11,900
- Safe Limit, psi	24,000	24,000	24,000
Maximum Bolt Stress - Calculated, psi	<4,439	4,439	22,500
- Safe Limit, psi	33,700	33,700	25,800

TABLE 3.9-13

CARRIER AIR HANDLING UNIT MOTOR STRESS COMPARISON

	Motor Horespower	
	5	20
Motor Mounting Bolts - Calculated, psi	3,389	11,990
- Safe Limit, psi	27,419	27,484
Canti-Lever Foot - Calculated, psi	832	585
- Safe Limit, psi	5,000	5,000
Inner Cap Screws - Calculated, psi	14.85	29.8
- Safe Limit, psi	875	2,125
Top Cap Screws - Calculated, psi	2.92	4.68
- Safe Limit, psi	875	875
Conduit Box Mounting - Screws - Calculated, psi	9.07	10.2
- Safe Limit, psi	875	875
Conduit Box Mounting - Pipe Nipple		
- Calculated, psi	None	None
- Safe Limit, psi	None	None
Front End Bearing - Calculated, psi	331	475
- Safe Limit, psi	1,170	5,048
Back End Bearing - Calculated, psi	921	1,820
- Safe Limit, psi	1,170	5,048

TABLE 3.9-14

WATER CHILLER STRESS COMPARISON

Component	Maximum Stress (psi)	Normal Allowable (psi)
Chiller Components		
Cooler Shell	10,024	12,000
Economizer Liquid Outlet Pipe to Cooler	16,328	20,550
Cooler Water Box Cover	18,111	26,250
Anchor Bolts - Axial Stress	12,552	62,500
Anchor Bolts - Shear Stress	10,297	25,800
Compressor Bolts - Axial Stress	32,101	52,500
Compressor Bolts - Shear Stress	10,340	21,700
Cooler Tube	2,881	6,000
Control Panel Support	2,006	13,700
Oil Filter Bracket Mounting Bolts - Axial Stress	8,931	57,500
Oil Filter Bracket Mounting Bolts - Shear Stress	14,223	23,767
Oil Cooler Bolts - Axial Stress	1,605	57,500
Oil Cooler Bolts - Shear Stress	1,105	23,767
Motor Shaft	2,123	12,000
Oil Pump Shaft	637	12,000
Component Support Structures		
Cooler to Base Tubesheet Support (NF 3320) [1]	1,885	12,000
Condenser to Cooler Tubesheet Support (NF 3320)	1,905	12,000

TABLE 3.9-14 (Continued)

Component	Maximum Stress (psi)	Normal Allowable (psi)
Component Support Structures (Continued)		
Cooler Gussets (NF 3320)	1,183	12,000
Base Box Beams (NF 3330)	5,605	12,000
Gusset Welds (NF 3390)	Negligible	18,000
Cooler to Base Tubesheet Support Welds (NF 3390)	3,612	18,000
Condenser Tubesheet Support Welds (NF 3390)	7,314	18,000
Condenser Tubesheet Support Welds (NF 3321.1(c)-1)	4,601	6,850

NOTE:

[1] References within parentheses refer to Subsection NF of ASME Code, Section III.

TABLE 3.9-15

BUFFALO FORGE CO. FAN SAFETY MARGINS

(Code Allowable/Calculated Maximum -1)

Component	Safety Margin (Fan 43B7)	Safety Margin (Fan 60D5)
Motor Shaft	9.8	4.2
Welds – Guide Vanes to Housing and Motor Support	2.44	5.8
Motor Bolts	6.00	7.5
Foundation Bolts	5.55	2.45
Vertical Legs	2.9	4.9
Welds – Fan to Support Structure	6.3	2.8
Motor Support Disc	.21	1.9
Motor Rear Bearing Seismic Loading	12.4	10.0
Motor Front Bearing Seismic Loading	22.1	10.1
Motor Shaft Stress (combined)	3.6	5.5

TABLE 3.9-16

NSSS COMPLIANCE WITH <Regulatory Guide 1.48>

Component	Plant Condition	Loading Combination [f]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC <Regulatory Guide 1.48>
Class 1 Vessels	U=upset	(NPC or UPC) + 0.5 SSE[1]	3.0 S_M (includes secondary stresses)	NB-3223	
	E=emergency	EPC	1.8 S_m or 1.5 S_y	NB-3224	Agree
	F=faulted	NPC + SSE + DSL	App. F-Sect. III	NB-3225	
Class 1 Piping	U	(NPC or UPC) + 0.5 SSE	3.0 S_m (includes secondary stresses)	NB-3654	
	E	EPC	2.25 S_m	NB-3655	Agree
	F	NPC + SSE + DSL	App. F – Sect. III	NB-3656	
Class 1 Pumps (Inactive)	U	(NPC or UPC) + 0.5 SSE	1.65 S_m (excludes secondary stresses)	NB-3223	
	E	EPC	1.8 S_m	NB-3224	Agree
	F	NPC + SSE + DSL	App. F – Sect. III	NB-3225	
Class 1 Pumps (Active)	U	(NPC or UPC) + 0.5 SSE	Not Applicable	Not Applicable	Not Applicable
	E	EPC			
	F	NPC + SSE + DSL			
Class 1 Valves (Inactive) by analysis	U	(NPC or UPC) + 0.5 SSE	Not Applicable	Not Applicable	Not Applicable
	E	EPC			
	F	NPC + SSE + DSL			

TABLE 3.9-16 (Continued)

Component	Plant Condition	Loading Combination [(f)]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC <Regulatory Guide 1.48>
Class 1 Valves (Inactive) Designed by either std. or alternative design rules	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	1.1 Pr 1.2 Pr 1.5 Pr	NB-3525 NB-3526 NB-3527	Agree
Class 1 Valves (Active) by analysis	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	Not Applicable	Not Applicable	Not Applicable
Class 1 Valves (Active) Designed by std. or alternative design rules	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	1.0 Pr 1.0 Pr[(a)] 1.0 Pr	NB-3525 NB-3526 NB-3527	Agree
Class 2 & 3 Vessels (Division 1) or ASME Section III	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	o-m = 1.1S See Note[(c)] o-m = 2.0S	Code Case 1607, NC/ND-3300	Agree except for Faulted Condition. NRC more conservative
Class 2 Vessels (Division 2) or ASME Section VIII	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	Not Applicable	Not Applicable	Not Applicable

TABLE 3.9-16 (Continued)

Component	Plant Condition	Loading Combination [f]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC <Regulatory Guide 1.48>
Class 2 & 3 Piping	U	(NPC or UPC) + 0.5 SSE	1.2 Sh	NC/ND-3611.3(b)	NRC more conserva-
	E	EPC	1.8 Sh	NC/ND-3611.3(c)	tive. GE
	F	NPC + SSE + DSL	2.4 Sh	Code Case 1606	reflects industry position
Class 2 & 3 Pumps (Inactive)	U	(NPC or UPC) + 0.5 SSE	Not Applicable	Not Applicable	Not Applicable
	E	EPC			
	F	NPC + SSE + DSL			
Class 2 & 3 Pumps (Active)	U	(NPC or UPC) + 0.5 SSE	o-m = 1.1S	Code Case 1636,	Agree
	E	EPC	See Note[a]	NC/ND-3423	See Note[a]
	F	NPC + SSE + DSL	o-m = 1.2S[c]		
Class 2 & 3 Valves (Inactive)	U	(NPC or UPC) + 0.5 SSE	o-m = 1.1S	Code Case 1635,	Equally
	E	EPC	See Note[c]	NC/ND-3521	conservative
	F	NPC + SSE + DSL	o-m = 2.0S		
Class 2 & 3 Valves (Active)	U	(NPC or UPC) + 0.5 SSE	o-m = 1.1S	Code Case 1635,	Equally
	E	EPC	See Note[a]	NC/ND-3521	conservative
	F	NPC + SSE + DSL	o-m = 2.0S[c]		

NOTES:

[1] An OBE or 0.5SSE intensity is classified as an emergency event. However, for design purposes it is treated as an upset condition as shown in <Table 3.9-3a>.

TABLE 3.9-16 (Continued)

NOTES FOR COMPARISON TABLE 3.9-16

Unless associated with an ASME Code reference, alphabetical indicators in (   ) correspond to the following:

(a) In addition to compliance with the design limits specified, assurance of operability under all design loading combinations shall be in accordance with Subsection 3.9.3.2.

(b) Not used.

(c) The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150 percent of that allowed for general membrane (except as limited to 2.4S for inactive components under faulted condition).

(d) Not used.

(e) Not used.

(f) When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in <Table 3.9-3>.

UPC = Upset Plant Conditions
NPC = Normal Plant Conditions
EPC = Emergency Plant Conditions
DSL = Dynamic System Loading
SSE = Safe Shutdown Earthquake

<TABLE 3.9-17>

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TABLE 3.9-18

MAXIMUM PRESSURE LIMITS FOR ACTIVE AND NONACTIVE ASME CODE CLASS VALVES

Safety Class 1 Valves:

Service Loading[1]	Design by Analysis Stress Limits	Standard Design Valve Pressure/Temperature Rating Limits[2]
Level A	NB-3222 (Active)	1.0 Pr (Active)
	NB-3223 (Nonactive)	1.1 Pr (Nonactive)
Level B	NB-3222 (Active)	1.0 Pr (Active)
	NB-3223 (Nonactive)	1.1 Pr (Nonactive)
Level C	NB-3222 (Active)	1.0 Pr (Active)
	NB-3224 (Nonactive)	1.2 Pr (Nonactive)
Level D	NB-3222 (Active)	1.0 Pr (Active)
	NB-3225 (Nonactive)	1.5 Pr (Nonactive)

TABLE 3.9-18 (Continued)

Safety Class 2 and 3 Valves:

Service Loading[1]	Valve Pressure/Temperature Rating Limits[2]
Level A	1.0 Pr (Active, Nonactive)
Level B	1.0 Pr (Active) 1.1 Pr (Nonactive)
Level C	1.0 Pr (Active) 1.1 Pr (Nonactive)
Level D	1.0 Pr (Active) 1.2 Pr (Nonactive)

NOTES:

[1] Service loadings are defined in <Section 3.9.3.1.2>.

[2] Pr represents the valve primary pressure rating as defined in ASME III NB, NC or ND as applicable.

TABLE 3.9-19

STRESS LIMITS FOR
ACTIVE ASME CODE CLASS 2 AND 3 PUMPS[1]

Service Loading[3]	Primary Membrane Stress Limit (P_m) Active Component	Primary Membrane + Bending Stress Limit ($P_m + P_b$) Active Component
A	1.0 S[2]	1.5 S
B	1.0 S	1.5 S
C	1.0 S	1.5 S
D	1.0 S	1.5 S

NOTES:

[1] Pump design satisfies the requirements of Section NC/ND-3400 of the ASME Code, Section III, for design conditions.

[2] S is the allowable stress value as specified in Appendix I of the ASME Code, Section III.

[3] Service loadings are defined in <Section 3.9.3.1.2>.

TABLE 3.9-20

STRESS LIMITS FOR
ASME CODE CLASS 2 AND 3 VESSELS[1]

Service Loading[3]	Primary Membrane Stress Limit (P_m)	Primary Membrane + Bending Stress Limit ($P_m + P_b$)
A	1.0 S[2]	1.5 S
B	1.1 S	1.65S
C	1.1 S	1.65S
D	1.5 S	1.8 S

NOTES:

[1] Pressure vessels designed in accordance with the ASME Code, Section III, satisfy the requirements of Section NC/ND-3300 of the ASME Code, Section III.

[2] S is the allowable stress value as specified in Appendix I of the ASME Code, Section III.

[3] Service loadings are defined in <Section 3.9.3.1.2>.

TABLE 3.9-21

BOP DESIGN LOADING COMBINATIONS OF ASME CODE
CLASS 1, 2 AND 3 PIPING[1]

Load[2] Case	N	SRV[3] 1	SRV ADS	OBE	SSE	SBA/IBA[4]	DBA	ASME Code Service Limit
1	X	X						B
2		X		X				B
3[7]		X			X			D[5]
4	X	X		X				B
5[7]	X	X			X			D[5]
6[7]	X	X				X		C[5]
7[7]	X		X			X		C[5]
8[7]	X		X	X		X		D[5]
9[7]	X		X		X	X		D[5]
10[7]	X				X		X[6]	D[5]
11	X							A

NOTES:

[1] Piping Component Bolting meets the allowables as defined by
Sections NC/ND 3658 of Section III of the ASME Code, July 1983
edition for Service Limits A, B, C, & D

[2] Refer to following legend for definition of terms.

[3] SRV_{ALL} or SRV - whichever is controlling will be used.

[4] SBA or IBA - whichever is greater.

[5] Functional capability of piping components in essential ASME Class 1,
2 and 3 piping will be demonstrated. This shall include use of the
criteria in the Interim Technical Position "Functional Capability of
Passive Piping Components," Mechanical Engineering Branch, Division
of Systems Safety. Where additional criteria are required to
evaluate functional capability, applicable criteria in NEDO-21985,
dated September 1978, shall be used.

[6] Annulus pressurization effects are included under DBA combinations
only

[7] Constraint of free end displacements and differential support motion
are not considered for this load case.

TABLE 3.9-21 (Continued)

<u>LOAD DEFINITIONS LEGEND</u>

N -   Normal load consists of pressure, dead weight and thermal
                  loads.

OBE -   Operational basis earthquake loads.

SSE -   Loads due to vibratory motion from safe shutdown
                  earthquake loads.

SRV_1 -   Safety/Relief valve discharge induced loads from one
                  valve's subsequent actuation.

SRV_{ALL} -   The loads induced by actuation of all safety/relief
                  valves which activate within milliseconds of each other
                  (e.g., turbine trip operational transient).

SRV_{ADS} -   The loads induced by the actuation of safety/relief
                  valves associated with the automatic depressurization
                  system which actuate within milliseconds of each other
                  during the postulated small or intermediate-size pipe
                  rupture.

DBA -   Design basis accident is the sudden break of the main
                  steam or recirculation lines (largest postulated breaks).
                  DBA related loads include main vent clearing and pool
                  swell, chugging, condensation oscillation, and annulus
                  pressurization.

SBA -   Small break accident.

IBA -   Intermediate break accident.

TABLE 3.9-21a

BALANCE OF PLANT DESIGN LOADING COMBINATIONS FOR ASME CODE
CLASS 1, 2 AND 3 COMPONENT SUPPORTS

I.   EQUIPMENT SUPPORTS -

Pump supports meet ASME Code Service Limits A or B for all loading
combinations. Vessel supports meet ASME Code Service Limits A, B
or C for all loading combinations.

II. PIPING SUPPORTS[1] -

Piping supports meet ASME Code Service Limits A, B, C, and D as
defined below:

Load Case[2]	ASME Code Service Limit	Plate & Shell Supports[3]	Linear Supports[4]	Component Standard
1	B	$1.5S^{(5)}$	$A^{(6)(7)}$	$B^{(6)}$
2	B	$1.5S^{(5)}$	$A^{(6)(7)}$	$B^{(6)}$
$3^{(8)}$	D	$1.8S^{(9)}$	$4/3\ A^{(10)}$	$D^{(11)}$
4	B	$1.5S^{(5)}$	$A^{(6)(7)}$	$B^{(6)}$
$5^{(8)}$	D	$1.8S^{(9)}$	$4/3\ A^{(10)}$	$D^{(11)}$
$6^{(8)}$	C	$1.8S$	$4/3\ A$	C
$7^{(8)}$	C	$1.8S$	$4/3\ A$	C
$8^{(8)}$	D	$1.8S^{(9)}$	$4/3\ A^{(10)}$	$D^{(11)}$
$9^{(8)}$	D	$1.8S^{(9)}$	$4/3\ A^{(10)}$	$D^{(11)}$
$10^{(8)}$	D	$1.8S^{(9)}$	$4/3\ A^{(10)}$	$D^{(11)}$
11	A	$1.5S^{(5)}$	$A^{(7)}$	A

TABLE 3.9-21a (Continued)

NOTES:

(1) Bolting is designed to NF 3280. Therefore, no increase in allowable stress is permitted for Code Service Limit C or D for bolting.
(2) Refer to <Table 3.9-21> for load cases.
(3) ASME Class 2 and 3 only.
(4) "A" is the allowable for bending, tension and shear. Appendix XVII of subsection NA presents equations for determining this value.
(5) When constraint of free end displacements are included, this allowable is multiplied by 2.0.
(6) The effects of differential support motion are not considered primary stresses, but for conservatism are treated as primary stresses.
(7) When constraint of free end displacements are included, this allowable ("A") is multiplied by 3.0. Buckling limits are not increased. Bolting limits are not increased.
(8) Constraint of free end displacements and differential support motion are not considered for this load case.
(9) If Code Service Limit D is used for the component, the plate and shell supports will be designed per equations:

$$\sigma_1 \quad \text{lesser of 1.5S or 0.4 Su}$$
$$\sigma_1 + \sigma_2 \quad \text{lesser of 2.25S or 0.6 Su}$$

Where:

Su = specified minimum ultimate tensile strength of the material (Table I-12.1)

Other terms as defined in NF 3321.

(10) If Code Service Limit D is used for the component, the linear support will be designed using the rules contained in F 1370 of Appendix F.
(11) If Code Service Limit D is used for the component, the component standard support will be designed to Level D limits.

TABLE 3.9-22

GE SUPPLIED SEISMIC CATEGORY ACTIVE PUMPS AND VALVES

COMPONENT NAME

MAIN STEAM ISOLATION VALVES	B21-F022
	B21-F028
MAIN STEAM SAFETY/RELIEF VALVES	B21-F041
	B21-F047
	B21-F051
CONTROL ROD DRIVE GLOBE VALVES	C11-F054
	C11-F062
	C11-F069
STANDBY LIQUID CONTROL PUMPS	C41-C001
STANDBY LIQUID CONTROL VALVE	C41-F004
RHR PUMP	E12-C002
LPCS PUMP AND MOTOR	E21-C001
HPCS PUMP AND MOTOR	E22-C001
HPCS GLOBE VALVE	E22-F010
	E22-F011
	E22-F023
HPCS GATE VALVE	E22-F001
	E22-F004
	E22-F012
	E22-F015
RCIC PUMP	E51-C001
RCIC TURBINE	E51-C002

TABLE 3.9-23

SUMMARY OF ACTIVE PUMPS (NON-NSSS)

Equipment Name	System Name	USAR Figure	Equipment Number	ASME III Code Class	Active Function	Qualification Method	Summary of Results
Screen Wash Pump	Emergency Service Water Screen Wash (P49)	-	C002A/B	3	Supplies high pressure water to clean traveling screens	Analysis	McDonald Engr. Report No. ME-452
Emergency Service Water Pump	Emergency Service Water System (P45)	<Figure 9.2-1(1)>	C001A/B	3	Supplies adequate head and flow for component cooling	Analysis	McDonald Engr. Report No. ME-454
Emergency Service Water Pump	Emergency Service Water System (P45)	<Figure 9.2-1(1)>	C002	3	Supplies adequate head and flow for component cooling	Analysis	McDonald Engr. Report No. ME-453
Water Leg Pumps							
RHR Water Leg Pumps	RHR System (E12)	<Figure 5.4-13(3)>	C003	2	Pressurizes ABC pump disch. lines	Analysis	Van Gulik and Associates, Inc. Report B40411
LPCS Water Leg Pumps	LPCS System (E21)	<Figure 6.3-8>	C002	2	Pressurizes ABC pump disch. lines	Analysis	Van Gulik and Associates, Inc. Report B40411
HPCS Water Leg Pumps	HPCS System (E22)	<Figure 6.3-7>	C003	2	Pressurizes ABC pump disch. lines	Analysis	Van Gulik and Associates, Inc. Report B40411
RCIC Water Leg	RCIC System (E51)	<Figure 5.4-9(1)>	C003	2	Pressurizes ABC pump disch. lines	Analysis	Van Gulik and Associates, Inc. Report B40411
Diesel Generator Fuel Oil Transfer Pumps	Standby Diesel Generator Fuel Oil System (R45)	<Figure 9.5-8>	C001A/B/C	3	Transfers fuel from tanks to diesel generator	Analysis	McDonald Engr. Report No. ME-532
		<Figure 9.5-15(1)>	C002/A/B/C	3	Transfers fuel from tanks to diesel generator	Analysis	McDonald Engr. Report No. ME-532

TABLE 3.9-23 (Continued)

Equipment Name	System Name	USAR Figure	Equipment Number	ASME III Code Class	Active Function	Qualification Method	Summary of Results
Fuel Pool Circu-lating Pumps	Fuel Pool Cooling & Cleanup (G41)	<Figure 9.1-9(3)>	C003A/B	3	Circulates water for cooling and cleanup of spent fuel pool	Analysis	McDonald Engr. Report No. ME-512
Control Complex Chilled Water Pumps	Control Complex Chilled Water (P47)	<Figure 9.4-20(1)>	C001A/B/C	3	Supplies cooling water for control room chillers	Analysis	I-R Order No. 016-36421
Emergency Closed Cooling Pumps	Emergency Closed Cooling System (P42)	<Figure 9.2-3(1)>	C001A/B	3	Provides cooling water for safety-related components	Analysis	I-R Order No. 016-36421

TABLE 3.9-24

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
SMALL EMERGENCY SERVICE WATER PUMP (1P45C0002)

Components	Vendor Report[1] Actuals	Allowables
Maximum Column Stress, psi	7,839	22,500
Maximum Column Flange Stress, psi	19,732	26,250
Bolt Stress, psi	29,553	37,500
Maximum Pump Casing Flange Stress, psi	9,158	21,000
Bolt Stress, psi	28,314	37,500
Nozzle Stress, psi	21,758	22,500
Anchor Bolt Stress, psi - Tensile	6,115	20,000
- Shear	4,054	10,000
Shaft Key Stress, psi	15,611	20,000
Pump Hub Coupling Key Stress, psi		
1.75" Design Key Engagement	16,210	18,750
1.52" Minimum Key Engagement	18,660	18,750
Pump Hold-Down Bolt Stress, psi – Tensile	10,531	36,000
- Shear	4,928	18,000
Motor Hold-Down Bolt Stress, psi - Tensile	4,304	20,000
- Shear	1,642	10,000
Discharge Head Base Plate Stress, psi	9,955	42,000
Discharge Head Stress, psi	3,398	22,500
Shaft Stress, psi (combined shear stress)		
Head Shaft Keyed End (top)	13,135	16,875
Head Shaft Threaded End (bottom)	16,869	22,500
Line Shaft Threaded Ends (top and bottom)	16,869	22,500
Pump Shaft Threaded End (top)	16,869	22,500
Discharge Nozzle Flange Moment, in-lbs.	30,242	305,550
Pump Casing Stress, psi	4,782	14,000

TABLE 3.9-24

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
SMALL EMERGENCY SERVICE WATER PUMP (1P45C0002)

Components	Vendor Report[1] Actuals	Allowables
Impeller Clearance, Inches	0.0000	0.009
Shaft Deflection, Inches	0.0027	0.03
Motor Support Hold-Down Bolt Stress, psi		
- Tensile	19,163	20,000
- Shear	3,181	10,000
Motor Support Stress, psi	4,465	22,500
Motor Support Flange Stress	14,073	26,250

NOTE:

[1] Supplemental calculations were also performed in conjunction with the Vendor Reports.

TABLE 3.9-24 (Continued)

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TABLE 3.9-25

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
LARGE EMERGENCY SERVICE WATER PUMP (1P45C0001A/B)

Components	Actuals	Allowables
Maximum Column Stress, psi	19,561	22,500
Maximum Column Flange Stress, psi	25,993	26,250
Bolt Stress, psi	34,240	37,500
Maximum Pump Casing Flange Stress, psi	10,426	21,000
Bolt Stress, psi	22,326	42,000
Nozzle Stress:		
Sustained Loads (ND-3652, Eqn 8), psi	3,695	22,500
Occasional Loads (ND-3652, Eqn 9), psi	5,687	22,500
Thermal Loads (ND-3652, Eqn 11), psi	10,409	22,500
Anchor Bolt Stress:		
Level D, Tensile, psi	20,452	39,860
Level D, Shear, psi	6,338	15,000
Level D + Thermal, Tensile, psi	24,977	26,720
Level D + Thermal, Shear, psi	14,544	15,000
Shaft Key Stress @ 42,000 in-lb, psi	12,491	42,000
Pump Hold-Down Bolt Stress		
Level D, Tensile, psi	16,373	37,500
Level D, Shear, psi	2,945	18,750
Level D + Thermal, Tensile, psi	21,763	60,000
Level D + Thermal, Shear, psi	6,953	30,000
Motor Hold-Down Bolt Stress, psi - Tensile	6,466	20,000
- Shear	3,349	10,000
Discharge Head Base Plate Stress		
Code, Level D, psi	16,672	26,250
Code, Level D + Thermal, psi	21,132	42,000
Discharge Head Stress, psi	17,506	22,500
Shaft Stress, psi	29,144	22,500

TABLE 3.9-25 (Continued)

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
LARGE EMERGENCY SERVICE WATER PUMP (1P45C0001A/B)

Components	Actuals	Allowables
Discharge Nozzle Flange Moment		
Level A, in-lb	422,496	590,000
Level B, in-lb	669,778	1,180,000
Level C, in-lb	422,496	1,730,000
Level D, in-lb	738,265	1,730,000
Pump Casing Stress, psi	4,434	14,000
Motor Mounting Plate Stress, psi	8,263	26,250
Motor Support Weld Stresses, psi	10,127	11,400

TABLE 3.9-25B

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TABLE 3.9-25B (Continued)

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TABLE 3.9-26

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
ESW SCREEN WASH PUMPS

Components	Vendor Report[1] Actuals	Allowables
Maximum Column Stress, psi	8,329	22,500
Maximum Column Flange Stress, psi	15,493	26,250
Bolt Stress, psi	21,765	37,500
Maximum Pump Casing Flange Stress, psi	7,786	21,000
Bolt Stress, psi	27,274	37,500
Nozzle Stress, psi	22,179	22,500
Anchor Bolt Stress, psi - Tensile	5,339	20,000
- Shear	1,454	10,000
Shaft Key Stress, psi	7,805	20,000
Pump Hold-Down Bolt Stress, psi - Tensile	8,862	15,000
- Shear	1,650	7,500
Motor Hold-Down Bolt Stress, psi - Tensile	2,601	20,000
- Shear	751	10,000
Discharge Head Base Plate Stress, psi	8,716	26,250
Discharge Head Stress, psi	2,367	22,500
Shaft Stress, psi	7,650	30,000
Discharge Nozzle Flange Stress, psi	20,536	26,250
Pump Casing Stress, psi	2,559	14,000
Impeller Clearance, Inches	.0001	.006
Shaft Deflections, Inches	.0020	.03
Motor Support Hold-Down Bolt Stress, psi - Tensile	12,475	20,000
- Shear	1,382	10,000

TABLE 3.9-26 (Continued)

Components	Vendor Report[1] Actuals	Allowables
Motor Support Stress, psi	2,847	22,500
Motor Support Flange Stress, psi	9,161	26,250

NOTE:

[1] Supplemental calculations were performed by Gilbert Associates Inc., in conjunction with the Vendor's report. Revised actual values are not presented in the above table. All revised values are within allowable limits.

TABLE 3.9-27

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
WATER LEG PUMPS

	Actual	Allowable
Pump Hold-Down Bolts Stress, psi - Shear	1,860	12,320
- Tensile	7,102	33,880
Pump Support (top weld), psi - Shear	520	7,560
- Tensile	2,002	12,600
Pump Support (bottom weld), psi - Shear	691	7,560
- Tensile	1,270	12,600
Motor Hold-Down Bolts, psi - Shear	764[1]	12,320
- Tensile	1,302[1]	33,880
Motor Support (weld), psi - Shear	99[1]	7,500
- Tensile	333[1]	12,600
Base Hold-Down Bolts, psi - Shear	1,005[1]	
- Tensile	1,737[1]	
Shaft Stress, psi - Tensile	2,154	18,750
Pump Nozzles, psi - Tensile	7,033	12,252
Pump Flanges, psi - Tensile	17,937	18,350
Pump Support Frequency, cps	270.0	>33
Motor Support Frequency, cps	70.1[1]	>33
Shaft Critical Frequency, cps	138.5	>33
Minimum Operating Clearance, inches	.0038	>.000

NOTE:

[1] These values are conservative values, due to the replacement RHR,
HPCS, and RCIC motor(s). Change in motor weight increased stresses
for the replacement motors. All ECCS motors are in the same report
and stress values have been updated in conjunction with the Vendor's
report. Revised actual values are presented in the above table. All
revised values are within allowable limits.

TABLE 3.9-28

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
CONTROL COMPLEX CHILLED WATER PUMPS & EMERGENCY CLOSED COOLING PUMPS

Component	Combined Stress Designation	Loads[1]	Calculated Stress, psi	Allowable Stress, psi
Casing	Bending	1	10,396	14,000
Flange Bolting	Direct	1	15,139	25,000
Suction Flange	Longitudinal	1 + 4	5,154	21,000
	Radial	1 + 4	1,400	21,000
	Tangential	1 + 4	1,901	21,000
	Longitudinal	1 + 5	5,733	31,500
	Radial	1 + 5	1,564	31,500
	Tangential	1 + 5	2,123	31,500
Discharge Flange	Longitudinal	1 + 4	8,959	21,000
	Radial	1 + 4	3,231	21,000
	Tangential	1 + 4	3,278	21,000
	Longitudinal	1 + 5	9,742	31,500
	Radial	1 + 5	3,528	31,500
	Tangential	1 + 5	3,580	31,500
Mounting Foot	Direct + Bending	2 + 3 + 4	4,227	14,000
	Shear	2 + 3 + 4	945	8,400
	Direct + Bending	2 + 3 + 5 + 6	5,118	21,000
	Shear	2 + 3 + 5 + 6	1,144	12,600

TABLE 3.9-28 (Continued)

Component	Combined Stress Designation	Loads[1]	Calculated Stress, psi	Allowable Stress, psi
Feet Bolts	Normal	2 + 3 + 4	35,140	46,620
	Normal	2 + 3 + 5 + 6	42,590	69,930
Feet Shear Pins	Shear	2 + 3 + 4	15,352	16,800
	Shear	2 + 3 + 5 + 6	18,464	25,200
Mounting Pad	Direct + Bending	2 + 3 + 4	2,995	18,000
Weld	Shear	2 + 3 + 4	837	12,000
	Direct + Bending	2 + 3 + 5 + 6	3,621	24,000
	Shear	2 + 3 + 5 + 6	1,006	16,000
Top plate	Bending	4 - 2	12,008	18,000
	Bending	5 + 6 - 2	16,129	24,000
Foundation Bolts	Normal	2 + 3 + 4	11,286	18,410
	Shear	2 + 3 + 4	5,990	10,800
	Normal	2 + 3 + 5 + 6	14,224	27,615
	Shear	2 + 3 + 5 + 6	7,403	16,200
Stuffing Box Bolts	Normal	1	6,786	25,000
Shaft	Shear	3	1,885	7,500

TABLE 3.9-28 (Continued)

NOTE:

[1] <u>LOADING LEGEND</u>

    1.   Design Pressure.

    2.   Deadweight.

    3.   Shaft Torque.

    4.   Normal or Steady-State Nozzle Loads.

    5.   OBE/SSE Nozzle Loads.

    6.   Safe Shutdown Earthquake.

TABLE 3.9-29

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
FUEL POOL CIRCULATING PUMPS

Components	Actual	Allowable
Motor Hold-Down Bolt Stress, psi - Shear	2,060	10,000
- Tensile	3,267	20,000
Pump Hold-Down Bolt Stress, psi - Shear	6,531	12,320
- Tensile	19,184	39,550
Anchor Bolt Stress, psi - Shear	7,058	10,000
- Tensile	13,188	16,707
Shaft Stress, psi	26,946	28,200
Frame Stress, psi	3,577	21,600
Thrust Retainer Bolt Stress, psi - Tensile	975	20,000
Pump Bearing Bolt Stress, psi - Shear	1,023	10,000
- Tensile	1,098	20,000
Pump Pedestal Stress, psi	2,139	21,600
Nozzle Stress, psi - Discharge	10,922	20,040
- Suction	1,772	20,040
Nozzle Flange Stress, psi - Discharge	16,835	20,040
- Suction	10,357	20,040
Pedestal Weld Stress, psi	5,304	10,800
Pump Bearing Loads, Lbs - Inboard	329	6,338
- Outboard	1,734	9,849
Flexible Coupling Misalignment, Radians	.00105	.017
Impeller Key Stress, psi - Shear	7,024	10,500
Impeller Relative Deflection, Inches	.003	.009

<TABLE 3.9-30>

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TABLE 3.9-31

PRESSURE DIFFERENTIALS ACROSS REACTOR VESSEL INTERNALS[1]

| | Maximum Pressure Differences Occurring During a Steam Line Break (psi) | |
Reactor Component	Case 1	Case 2
Core Plate and Guide Tube	27.5	28.5
Shroud Support Ring and Lower Shroud	47.0	54.0
Upper Shroud	23.0	28.5
Average Channel Wall (bottom)	15.0	12.0
Top Guide	7.4	13.4

Case 1 - Reactor initially at 102% rated thermal power, 105% rated core
         flow

Case 2 - Reactor initially at 33% rated thermal power, 110% rated core
         flow

NOTE:

[1] The values shown above are based on a GE12 full core configuration.
    [Analysis contained in Global Nuclear Fuels, DRF J11-03754-00,
    reviewed the application of GE14 and GNF2 fuel to Perry and concluded
    GE12 (with debris filters) fuel analysis bounds GE14 and GNF2(with
    debris filters) fuel.]

TABLE 3.9-32

DEFORMATION LIMIT

(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

General Limit

Permissible Deformation, DP
Analyzed Deformation
Causing Loss of Function, DL

$$\leq \frac{0.9}{SF_{min}}$$

Where:

DP = Permissible deformation under stated conditions of Service Levels, A, B, C, or D (normal, upset, emergency, or faulted).

DL = Analyzed deformation which could cause a loss of function can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they will be specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, and core support deformation causing fuel disarrangement.

SF_{min} = Minimum safety factor <Section 3.9.5.3.4>.

TABLE 3.9-33

PRIMARY STRESS LIMIT

(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Any One of (No More Than One Required)   General Limit

a. $\dfrac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}$ $\dfrac{\leq 2.25}{SF_{min}}$

b. $\dfrac{\text{Permissible load, LP}}{\text{Largest lower bound limit load, CL}}$ $\dfrac{1.5}{\leq SF_{min}}$

c. $\dfrac{\text{Elastic evaluated primary stress, PE}}{\begin{array}{l}\text{Conventional ultimate strength}\\\text{at temperature, US}\end{array}}$ $\dfrac{0.75}{\leq SF_{min}}$

d.   Elastic-plastic evaluated
     $\dfrac{\text{nominal primary stress, EP}}{\begin{array}{l}\text{Conventional ultimate strength}\\\text{at temperature, US}\end{array}}$ $\dfrac{0.9}{\leq SF_{min}}$

Where:

     PE   =   Primary stresses evaluated on an elastic basis. The
                effective membrane stresses are to be averaged through
                the load carrying section of interest. The simplest
                average bending, shear or torsion stress distribution
                which will support the external loading will be added to
                the membrane stresses at the section of interest.

     SF_{min} =   Mininum safety factor <Section 3.9.5.3.4>.

     PN   =   Permissible primary stress levels under Service Levels A
                or B (normal or upset) conditions under ASME Boiler and
                Pressure Vessel Code, Section III.

     LP   =   Permissible load under stated conditions of Service
                Levels A, B, C, or D (normal, upset, emergency, or
                faulted).

TABLE 3.9-33 (Continued)

PRIMARY STRESS LIMIT

(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Where: (Continued)

CL = Lower bound limit load with yield point equal to 1.5 S_m where S_m is the tabulated value of allowable stress at temperature of the ASME III code or its equivalent. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (nonstrain hardening material where deformations increase with no further increase in applied load). The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yeild strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yield to the uniaxial case.

US = Conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.

EP = Elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress strain curve may be used which everywhere has a lower stress for the same strain as the actual monotonic curve. Either the shear or strain energy of distortion flow rule may be used.

TABLE 3.9-34

BUCKLING STABILITY LIMIT

(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Any One of (No More Than One Required)   General Limit

a.   Permissible load, LP   2.25
     Service Level A (normal) permissible $\leq SF_{min}$
     load, PN

b.   Permissible load, LP   0.9
     Stability analysis load, SL $\leq SF_{min}$

Where:

     LP   =   permissible load under stated conditions of Service
             Levels A, B, C, or D (normal, upset, emergency, or
             faulted).

     PN   =   applicable Service Level A (normal) permissible load.

     SL   =   stability analysis load. The ideal buckling analysis is
             often sensitive to otherwise minor deviations from ideal
             geometry and boundary conditions. These effects shall be
             accounted for in the analysis of the buckling stability
             loads.   Examples of this are ovality in externally
             pressurized shells or eccentricity on column members.

     SF_{min} =   minimum safety factor <Section 3.9.5.3.5>.

TABLE 3.9-35

FATIGUE LIMIT (FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Design fatigue cycle usage from analysis using the method of ASME Code	Limit for Service Levels A and B (Normal and Upset) Design Conditions
Cumulative Damage in Fatigue	≤1.0

TABLE 3.9-36

ASSURANCE OF OPERABILITY ANALYSIS
DIESEL GENERATOR FUEL OIL TRANSFER PUMP
(SUMMARY OF RESULTS)

Components	Actuals	Allowables
Motor Hold-Down Bolt Stress, psi - Shear	4,109	10,000
- Tensile	8,685	20,000
Pump Hold-Down Bolts Stress, psi - Shear	4,441	12,320
- Tensile	10,153	40,000
Anchor Bolt Stress, psi - Shear	3,551	10,000
- Tensile	5,752	20,000
Shaft Stress, psi	4,305	17,500
Frame Stress, psi	6,088	21,600
Thrust Retainer Bolt Stress, psi	2,446	20,000
Pump Frame Bolt Stress, psi - Shear	4,148	12,320
- Tensile	11,280	40,000
Frame Adapter Bolt Stress, psi	15,906	25,000
Frame Adapter Flange Stress, psi	17,270	21,000
Maximum Nozzle Stress, psi - Discharge	15,533	20,640
- Suction	16,682	20,640
Adapter Frame Ring Bolt Stress, psi	4,568	40,000
Pump Bearing Loads, lb - Inboard	307	4,443
- Outboard	1,788	8,855
Flexible Coupling Misalignment, Radians	.010	.017
Impeller Connection Stress, psi - Shear	4,950	8,570
- Tensile	1,614	17,500
Impeller Relative Deflection, Inches	.004	.025

TABLE 3.9-36 (Continued)

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TABLE 3.9-37

COMPONENT CYCLIC OR TRANSIENT LIMITS[1]

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	40 hydrostatic pressure or leak tests	Pressurized to ≥930 psig and ≤1250 psig

NOTES:

[1] This table retained for historical purposes. Refer to <Table 3.9-1> or figures referenced in <Table 3.9-2> for updated cycle information.

3.10   SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND
          ELECTRICAL EQUIPMENT

This section describes the program for seismic and dynamic qualification
of mechanical and electrical equipment to assure its operability under
the full range of normal and accident loadings. Mechanical and
electrical equipment to be qualified includes equipment associated with
systems that are essential to:

a.   Emergency reactor shutdown,

b.   Containment isolation,

c.   Reactor core cooling,

d.   Containment heat removal,

e.   Reactor heat removal, or

f.   Preventing significant release of radioactive material to the
      environment.

That is, equipment that:

a.   Performs the above functions automatically,

b.   Is used by the operator to perform these functions manually, or

c.   The failure of which can prevent the satisfactory accomplishment of
      one or more of the above safety functions.

A list of these systems is provided in <Table 3.2-1>. The mechanical
and electrical equipment associated with the safety functions of these
systems is designated as Safety Class 1, 2 or 3 (see <Section 3.2.3> for

definitions) in <Table 3.2-1> and is referred to generally as "safety-related" equipment. The safety-related mechanical and electrical equipment and its associated supports are classified Seismic Category I, except those portions of the radioactive waste treatment handling and disposal systems, whose postulated simultaneous failure would not result in conservatively calculated offsite exposures comparable to the guideline exposures of <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>).

For seismic and dynamic qualification, "safety-related" equipment is categorized in three groups by safety function:

a.   Safety-related Electrical Equipment designated as "Class 1E" per IEEE Standard 279, 1971.

b.   Safety-related Mechanical Equipment

   1.   "Active" Mechanical Equipment - is that which must move or change position to perform its safety function (examples are pumps, motor-operated valves, safety-relief valves, or check valves).

   2.   "Passive" Mechanical Equipment - is that which must only maintain its pressure integrity to perform its safety function (examples are tanks, heat exchangers or manual valves).

A discussion of the listing of the specific equipment by MPL No. that was prepared to demonstrate the seismic and dynamic qualification is provided in <Section 3.10.4>.

Recognizing that the hydrodynamic loads associated with a
loss-of-coolant accident (LOCA) and safety/relief valve (SRV) discharge
can have a significant effect on the design of structures, systems and
equipment, both seismic and hydrodynamic loads are addressed in this
section. The hydrodynamic loads are applicable to equipment in the

Reactor Building only. Reference to the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE) in this section includes the combined seismic and hydrodynamic loads. These hydrodynamic loads are described in detail in <Appendix 3A>.

3.10.1   SEISMIC AND DYNAMIC QUALIFICATION CRITERIA

This Section describes:

a.   The adequacy of the seismic and dynamic qualification program through conformance with the regulatory requirements,

b.   The deciding factors for choosing between tests and analyses, and

c.   The considerations in defining the seismic and dynamic load input motions.

3.10.1.1   Conformance with Regulatory Requirements

3.10.1.1.1   <10 CFR 50, Appendix A>

3.10.1.1.1.1   Criterion 1 - Quality Standards and Records

The seismic and dynamic qualification program is in compliance with the project Quality Assurance Program and its conformance to GDC 1 as discussed in <Section 3.1.2.1.1.1>.

3.10.1.1.1.2   Criterion 2 - Design Bases for Protection Against Natural Phenomena

The consideration of seismic loadings in the input motion is discussed in <Section 3.10.1.3>. Mechanical and electrical equipment is qualified to assure operability under the full range of normal and accident loadings (including seismic) in accordance with the requirements of GDC 2.

3.10.1.1.1.3   Criterion 4 - Environmental and Dynamic Effects Design
                    Bases

The consideration of the dynamic effects of missiles, pipe whipping and
discharging fluids in the input motion is discussed in
<Section 3.10.1.3>. Mechanical and electrical equipment is qualified to
assure operability under the full range of normal and accident loadings
in accordance with the requirements of GDC 4.

3.10.1.1.1.4   Criterion 14 - Reactor Coolant Pressure Boundary and
                    Criterion 30 - Quality of the Reactor Coolant Pressure
                    Boundary

Mechanical and electrical equipment associated with the reactor coolant
pressure boundary is identified as discussed in the beginning paragraph
of <Section 3.10>, and is qualified to assure operability under the full
range of normal and accident loadings in accordance with the
requirements of GDC 14 and 30.

3.10.1.1.2   <10 CFR 50, Appendix B>

The Mechanical and Electrical Equipment Seismic and Dynamic
Qualification Program is in compliance with the project Quality
Assurance Program. The Quality Assurance Program meets the requirements
of <10 CFR 50, Appendix B>. See <Chapter 17> for a detailed discussion.

3.10.1.1.3   Regulatory Guides

3.10.1.1.3.1   <Regulatory Guide 1.61>

Damping values associated with mechanical and electrical equipment are
selected in accordance with <Regulatory Guide 1.61>.

3.10.1.1.3.2   <Regulatory Guide 1.89>

Compliance with <Regulatory Guide 1.89> is discussed in
<Section 3.11.2.1.3.5>.

3.10.1.1.3.3   <Regulatory Guide 1.92>

Modal responses are combined in accordance with the methods defined in
<Regulatory Guide 1.92>.

3.10.1.1.3.4   <Regulatory Guide 1.100>

The seismic and dynamic qualification program for mechanical and
electrical equipment was designed to conform to the requirements of IEEE
Standard 344-1975. In addition, the program is responsive to the
guidance of IEEE Standard 323-1974.

The seismic and dynamic qualification program has been determined to be
in accordance with the guidelines outlined in this section as documented
in Supplement 7 to the PNPP SER. BOP equipment meets IEEE
Standard 344-1975 as modified by <Regulatory Guide 1.100>. NSSS
equipment in a harsh environment is qualified to IEEE Standard 344-1975,
while that equipment in a mild environment is qualified to IEEE
Standard 344-1971 and has been evaluated to the requirements of
344-1975.

3.10.1.2   Selection of Qualification Method

Seismic and dynamic qualification of mechanical and electrical equipment
is accomplished by test, analysis or a combination of the two methods.
In general, analysis is used to supplement test data although simple

components may lend themselves to analysis in lieu of full-scale
testing. The deciding factors for choosing between tests or analysis
include:

a.   Magnitude and frequency of the seismic and hydrodynamic loadings,

b.   Environmental conditions <Section 3.11.1> associated with the
     dynamic loadings,

c.   Nature of the safety function(s),

d.   Size and complexity of the equipment,

e.   Dynamic characteristics of expected failure modes (structural or
     functional).

3.10.1.3   Input Motion

The input motion for the qualification of equipment and supports is
defined by either response spectra or static g-levels. The floor
response spectra are generated from the building dynamic analysis. They
are grouped by buildings and by elevations. The required response
spectra (RRS) for the seismic and hydrodynamic loads were provided to
the equipment vendors.

When one type of equipment is located at several elevations and/or in
several buildings, the governing response spectra are specified.

3.10.1.4   Qualification for Hydrodynamic Loads

3.10.1.4.1   Uncertainty in Hydrodynamic Loads

Uncertainties in hydrodynamic load amplitude and frequency that arise
from the use of an axisymmetric model, variations in the properties of

structure and foundation, and the inability of analytical model-forcing function combinations to accurately predict high frequency response are accounted for as follows:

a. Non-axisymmetric forcing functions are employed.

b. Envelopes of radial and tangential response are used in combination with azimuthal envelopes.

c. The conservative GE load definition has generally been "tuned" to provide maximum response.

d. Events are enveloped.

e. Equipment RRS have been broadened in accordance with standard procedures <Regulatory Guide 1.122>.

f. The general problem of high frequency "exceedances" is being addressed in a manner similar to that described in "Generic Criteria for High-Frequency Cut-off for BWR Equipment," NEDE-25250.

g. Where equipment is qualified by test, frequencies up to at least 100 Hz are considered.

3.10.1.4.2   Input Load for Equipment

Various hydrodynamic loads are combined with SSE or OBE for equipment qualification as follows:

a. OBBA = OBE + SRV (2% damping)

b. DE1 = SSE + Pool Swell (3% damping)

c. LC3 = SSE + SRV + CO (3% damping)

d.   LC4 = SSE + SRV + Chug (3% damping)

e.   DE2 = Higher of LC3 or LC4

f.   SSBA = Higher of DE1 or DE2

The combinations are by SRSS, the basis for which is contained in "Study to Demonstrate the Generic Applicability of SRSS Combination of Dynamic Responses for Mark III Nuclear Steam Supply System and Balance-of-Plant Piping and Equipment Components," SMA 12109.01-R001.

Individual spectra were developed by time history analysis for the following dynamic loads:

a.   OBE

b.   SSE

c.   SRVA (1, 8 and 19 valves)

d.   Pool Swell (including main vent cleaning)

e.   Chug

f.   Condensation Oscillation

The hydrodynamic loads are discussed in detail in <Appendix 3A>. Whenever envelopes are employed, they are based on the highest acceleration at any given frequency.

3.10.1.4.3   Effect of Number of Cycles

Where qualification is by analysis, all stress cycles on critical sections, plant normal, abnormal, and accident conditions, including the

effect of thermal cycles and hydrodynamic and earthquake load cycles, are considered. Where qualification is by testing, shake tables using random motion compatible with the required response spectra in accordance with IEEE 344-1975 are preferred.

3.10.2   METHODS AND PROCEDURES FOR SEISMIC AND DYNAMIC QUALIFICATION
            OF MECHANICAL AND ELECTRICAL EQUIPMENT

The seismic and dynamic qualification of mechanical equipment is discussed in <Section 3.9.2> and <Section 3.9.3>.

The remainder of the section that follows describes the methods and procedures used to qualify by test or analysis, Seismic Category I instrumentation and electrical equipment for operation during seismic and hydrodynamic events.

Safety-related electrical equipment and instrumentation is shown to function as designed during and after an earthquake even though several other earthquakes may have occurred during its operating history.

The typical seismic test procedure includes exploratory tests (to search for resonant frequencies), five OBE tests and one SSE test. The structural and functional integrity of the equipment must be maintained during and after the test. The test procedure also compensates for the limitations of current facilities, e.g., limited multi-directional capability.

Safety-related equipment to be qualified by analysis will be qualified for OBE, SSE and other vibratory loads.

3.10.2.1   Qualification by Testing

The testing methodology for Seismic Category I instrumentation and electrical equipment includes the hardware interface requirements and the test methods.

3.10.2.1.1   Interface Requirements

Intervening structures or components (such as interconnecting cables, bus ducts, conduit, etc.) that serve as interfaces to the equipment to be qualified are not qualified as part of this program. However, the effects of interfaces are taken into consideration. When applicable, accelerations and frequency content at locations of interfaces with interconnecting cables, bus ducts, conduits, etc., are determined and documented in the test report.

To minimize the effects of interfaces of the equipment, standard configurations bottom cable entries are utilized whenever possible. Where non-rigid interfaces are located at the equipment support top, equipment qualification is based on top entry requirements. Equipment support outline drawings with the equipment embedment loads and mounting requirements for the equipment support are supplied by the vendor.

3.10.2.1.2   Test Methods

The test methods include biaxial or single axial (as appropriately justified) random multi-frequency, or single-frequency excitation that envelope the RRS levels in accordance with the intent of IEEE Standard 344.

When biaxial testing is used for qualification, it applies input motions to both the vertical and one of the horizontal axes simultaneously. Independent random inputs are preferred and, when used, the test is performed in two steps with equipment rotated 90 degrees in the horizontal plane for the second step.

3.10.2.1.2.1   Selection of Test Specimen

Representative samples of equipment and supports are selected for use as test specimens. Variations in the configuration of similar equipment

are reviewed to determine worst case. For example, variations may occur in mass distributions from one cabinet to another. Before testing, it is determined which mass distribution results in the maximum acceleration and/or frequency content and this worst case configuration is used as the test specimen.

3.10.2.1.2.2   Mounting of Test Specimens

The test specimen is mounted to the vibration table so that inservice mounting, including interfaces, is simulated.

For interfaces that cannot be simulated on the test table, the acceleration and any resonances at such interface locations are recorded during the equipment test and documented in the test report.

3.10.2.1.3   Seismic (including hydrodynamic loads) Testing Sequence

The test sequence includes exploratory resonance search, and the seismic including hydrodynamic loads testing.

3.10.2.1.3.1   Exploratory Tests

Exploratory tests may be run on equipment to aid in the determination of the test method. Exploratory tests are sine-sweep tests to determine resonant frequency and transfer factors at locations of Seismic Category I devices in the equipment support. The exploratory tests are run at a low acceleration level at a sweep rate of 2 octaves per minute or less. This is intended to excite all modes between approximately 1 and 33 Hz for seismic only, and a higher cutoff frequency for seismic plus hydrodynamic loads. This cutoff frequency for combined seismic and hydrodynamic loads is 100 Hz. An acceleration level is chosen to provide a usable signal-to-noise ratio for the sensing equipment to allow accurate detection of the natural frequencies of the test specimens.

These tests are run for one axis at a time in the three mutually
perpendicular major axes corresponding to the side-to-side,
front-to-back and vertical directions.

Accelerations and frequencies are monitored at locations determined by
the test engineer and the equipment vendor. Following the exploratory
tests, the devices are either tested to determine the malfunction limit
or subjected to a seismic test, including hydrodynamic loads.

3.10.2.1.3.2   Malfunction Limit Test

Malfunction Limit Tests are usually performed on devices that will
become part of one or more assemblies (especially control panels). The
malfunction limit test is performed at each resonant frequency
determined by the frequency scan. In this test the acceleration level
is gradually increased until either the device malfunctions or the limit
of the vibration machine is reached. If no resonances are detected,
the device is considered to be rigid (all parts move in unison) and the
malfunction limit is, therefore, independent of frequency. To achieve
maximum acceleration from the vibration machine, rigid devices are
malfunction tested at the cutoff test frequency since that allows the
maximum acceleration to be obtained from deflection limited machines.

3.10.2.1.3.3   Seismic Including Hydrodynamic Loads Testing

When a seismic including hydrodynamic loads test is performed, the
cutoff frequency is extended beyond 33 Hertz to the cutoff frequency for
hydrodynamic loads as discussed in <Section 3.10.2.1.3.1>. This test is
conducted to demonstrate that equipment will continue to perform its
safety function during and after seismic events (OBE and SSE) combined
with hydrodynamic loads. Operation of equipment is verified as
described in <Section 3.10.2.1.3.4>.

3.10.2.1.3.4   Demonstration of Operation

Equipment is tested in an operational condition. Most Seismic
Category I instrumentation and electrical equipment have safety function
requirements before, during and after seismic events. Other equipment
(such as plant status display equipment) have requirements only before
and after seismic events. All equipment is operated at appropriate
times to demonstrate ability to perform its safety function. Digital
equipment is cycled while analog equipment is stepped in discrete steps
from zero to full scale or ramped from zero to full scale.

If a malfunction is experienced during any test, the effects of the
malfunction are determined and documented in the final test report. If
necessary, repairs and/or modifications are made so the equipment will
meet the performance requirements and a retest is performed. The
results of the retest are noted in the final test report and, if
successful, the modified equipment is considered qualified.

3.10.2.1.4   Final Test Report

The final test contains a summary of test/analysis results which is
readily available for audit. The report normally includes but is not
limited to the following:

a.   Equipment Identification

b.   Equipment Specification

c.   Test Facility, including:

    1.   Location

    2.   Test equipment and calibration

d.   Test Methods and Procedures

e.   Test data (including performance data)

f.   Summary of results, including:

   1.   Resonant frequencies, if any, and transfer ratios,

   2.   Calculation of equipment damping coefficient if there is
         resonance in the frequency range of interest,

   3.   Conclusion as to equipment seismic and dynamic qualification.

3.10.2.2   Qualification by Analysis

The procedure presented in the sections that follow apply to the
qualification of equipment by analysis.

3.10.2.2.1   Analysis Methods

Analysis is employed in two phases for equipment qualification. First,
a dynamic analysis or an equivalent static analysis is made of the
equipment to establish the inertia forces acting on the equipment. In
general, the choice of the analysis is based on the expected design
margin since the static coefficient method (the easiest to perform),
when applicable, is generally more conservative than the dynamic
analysis method. After the inertia forces are determined, a stress
analysis is performed to verify the equipment can perform its intended
function under these forces.

If the fundamental frequency of the equipment is above the input cutoff
excitation frequency (as defined in <Section 3.10.2.1.3.1>), the
equipment is considered rigid. In this case, the loads on each
component can be determined statically by concentrating its mass at its

center of gravity and multiplying the values of the mass by the appropriate maximum floor acceleration at the equipment support point. The maximum floor acceleration is the acceleration at the ZPA of the RRS.

A static coefficient analysis may also be used for certain equipment in lieu of the dynamic analysis. No determination of natural frequencies is made in this case. The seismic loads are determined statically by multiplying the actual distributed weight of the equipment by a static coefficient equal to 1.5 times the peak value of the RRS at the equipment mounting location.

The static coefficient analysis is applicable to equipment with simple frame-type structures and can be represented by a simple model. For equipment having configurations other than a simple frame-type structure, this method may be applied when justification can be provided for the static factor which is used on a case-by-case basis.

If the equipment is determined to be flexible (i.e., within the frequency range of the input spectra) and not simple enough for equivalent static analysis, a dynamic analysis method is required. Dynamic analysis by the response spectrum method is outlined in <Section 3.7.2>.

3.10.2.2.2   Seismic Analysis Including Hydrodynamic Loads

An analysis is performed assuming seismic including hydrodynamic loads. The analysis must show that the seismic events (OBE and SSE) with hydrodynamic loads do not result in failure of the equipment to perform its safety function(s).

3.10.2.2.3   Documentation of Analysis

The demonstration of qualification is documented and includes the
requirements of the equipment specification, the results of the
qualification, and the justification that the methods used are capable
of demonstrating that the equipment will not malfunction.

3.10.2.3   Qualification by Combined Testing and Analysis

In some instances, it is not practical to qualify Seismic Category I
instrumentation and electrical equipment solely by testing or analysis.
This may be because of the size of the equipment, the complexity or the
large number of similar configurations. The sections that follow
address the cases in which combined analysis and testing may be
warranted.

3.10.2.3.1   "In Situ" Testing

Large equipment may be impractical to test due to limitations in
vibration equipment loading capability. With the equipment mounted to
simulate service mounting, a number of exciters are attached at points
which will best excite the various modes of vibration of the equipment.
Data is obtained from sensors for subsequent analysis of the equipment
seismic plus hydrodynamic load performance. The amplification of
resonant motion is used to determine the appropriate modal frequencies
and damping for a dynamic analysis of the equipment.

3.10.2.3.2   Extrapolation of Similar Equipment

In many instances, similar equipment has already been qualified but with
changes in size or in specific qualified devices in a fixed assembly or
structure. In such instances, a full test program <Section 3.10.2.1> is
conducted on a typical piece of equipment. The test results combined
with some analysis allows a model of the similar equipment to be

adjusted to produce a revised stiffness matrix and to allow refinement
of the analysis for the modal frequency of the similar equipment. The
result is a verified analytical model that is used to qualify the
similar equipment.

3.10.2.3.3   Extrapolation of Dynamic Loading Conditions

Test results can be extrapolated for dynamic loading conditions in
excess of, or different from, previous tests on a given piece of
equipment when the test results are in sufficient detail to allow an
adequate dynamic model of the equipment to be generated. The model
provides the capability of predicting failure under the increased or
different dynamic load excitation.

3.10.3   METHODS AND PROCEDURES FOR SEISMIC AND DYNAMIC QUALIFICATION
            OF SUPPORTS OF MECHANICAL AND ELECTRICAL EQUIPMENT

Methods and procedures for seismic and dynamic qualification of
mechanical equipment supports are discussed in <Section 3.9.3.4>.

Combined stresses of the mechanically designed electrical equipment
supports are maintained within the limits of ASME Code Section III,
Division 1, Subsection NF, up to the interface with building structure.
The combined stresses of the structurally designed component supports
defined as building structure in the project design specifications are
maintained within the limits of the AISC Specification for the Design,
Fabrication and Erection of Structural Steel for Buildings, 1969 edition
plus addenda.

3.10.3.1   NSSS Equipment Supports (Other than Motors and
              Valve-Mounted Equipment)

The equipment support seismic and hydrodynamic load qualification tests
are performed over the frequency range of interest. When possible, the

supports are tested with the equipment installed. Otherwise, a dummy is employed to simulate inertial mass effects and dynamic coupling to the support.

Some GE-supplied Seismic Category I supports are qualified by analysis only. Analysis is used for passive mechanical devices and is sometimes used in combination with testing for larger assemblies containing Category I devices. For instance, a test can be run to determine if there are natural frequencies in the support equipment within the critical frequency range. If the support is determined to be free of natural frequencies (in the critical frequency range), then it is assumed to be rigid and a static analysis is performed. If it has natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations are determined to see if Seismic Category I supports still maintain functional integrity.

In general, the testing of Seismic Category I supports is accomplished using the procedure that follows.

Assemblies (e.g., control panels) containing devices which have established dynamic load malfunction limits are tested by mounting the assembly on the table of a vibration machine, in the manner it is to be mounted when in use, and running a low-level resonance search. The assemblies are tested in the three major orthogonal axes.

The resonance search is run in the same manner as performed for devices. If resonances are present, the transmissibility between the input and the location of each device is determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response of any Seismic Category I device location for any given input. (It is assumed that the transmissibilities are linear as a function of acceleration even though

they actually decrease as acceleration increases; therefore, it is a conservative assumption.)

Control panels and racks constitute the majority of Seismic Category I electrical assemblies supplied by GE. There are basically four generic panel types. One or more of each type is tested using these procedures. <Figure 3.10-1>, <Figure 3.10-2>, <Figure 3.10-3> and <Figure 3.10-4> illustrate the four basic panel types and show typical accelerometer locations.

From many full acceleration level tests, it can be concluded that most of the panel types have more than adequate structural strength and that a given panel design acceptability is just a function of its amplification factor and the malfunction levels of the devices mounted in it.

Subsequent panels can, therefore, be tested at lower acceleration levels and the transmissibilities measured to the various devices as described. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input, the panel dynamic qualification level can be determined. Several high level tests have been run on selected generic panel designs to assure the conservativeness in using the transmissibility analysis described.

3.10.3.2   Balance of Plant Electrical Equipment Supports

3.10.3.2.1   Battery Racks, Instrument Racks, Control Cabinets, and Panels

Class 1E battery racks, instrument racks, control cabinets, panel supports, and other methods of attachment are analyzed or tested with the accompanying equipment, and are qualified with the equipment mounted or attached in a manner that simulates the intended service mounting.

3.10.3.2.2   Cable Tray and Conduit Support Criteria

Safety-related cable tray and conduit (raceway) supports are designed by
the equivalent static load method of analysis described in
<Section 3.7.3>. The structural capacity of the raceway is used to
determine the maximum spacing of the support points. Simply supported,
single span, beam models are used to represent the raceways. Provisions
are made for differential motion between buildings by breaks in the tray
and flexible connections in the conduit.

The following criteria are used in the design of safety-related raceway
supports:

a.   Method of Analysis

     Safety-related raceway supports are designed to satisfy the
     requirements of Seismic Category I structures by means of
     equivalent static load analysis using the appropriate seismic
     response spectra.

b.   Raceway Loading Criteria

     Cable tray loading of 15 lb/ft^2 of tray bottom is used for 4-inch
     deep tray and 25 lb/ft^2 of tray bottom for 6-inch deep tray.
     Conduit loading considers the maximum cable loading that can occur
     within the conduit.

c.   Raceway Support Spacing

     Raceway support spacing for horizontal or vertical runs do not
     exceed 10 feet unless noted otherwise on design drawings (design
     drawings generally locate supports not more than 9'-0" apart).

Raceway supports provide resistance to three earthquake directions by means of vertical, transverse and longitudinal support and bracing systems.

d. Support Attachments

Raceway supports are attached in the following manner:

1. Support members may be attached to existing structural steel.

2. Support members may be attached to supplementary steel members spanning between existing floor beams or between existing floor beams and walls.

3. Support members may be attached to concrete wall/columns/floor using:

    (a) Embedded steel plates with stud connectors.

    (b) Steel plates attached to concrete with drilled in expansion anchors.

e. Load Combinations

1. Dead load + cable load + 200 lbs concentrated load placed at any point in the span (tray only).

2. Dead load + cable load + OBE + hydrodynamic load.

3. Dead load + cable load + SSE + hydrodynamic load.

3.10.4   SEISMIC AND DYNAMIC QUALIFICATION RESULTS

The results of the seismic and dynamic qualification program for each
type of harsh environment safety-related electrical equipment are
recorded in the Equipment List and stored in an auditable file.
Retention of auditable file packages and listing in the EQ equipment
list is not required for any mechanical components or for any mild
environment electrical components.

3.10.4.1   Listing of Safety-Related Equipment

See <Section 3.11.3.1>

3.10.4.2   Auditable File

Auditable file for seismic and dynamic qualification is not required by
<10 CFR 50.49>.

3.10.5   REFERENCES FOR SECTION 3.10

1.   <10 CFR Part 50, Appendix A>, General Design Criterion 1, "Quality Standards and Records."

2.   <10 CFR Part 50, Appendix A>, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3.   <10 CFR Part 50, Appendix A>, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

4.   <10 CFR Part 50, Appendix A>, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

5. <10 CFR Part 50, Appendix A>, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

6. <10 CFR Part 50, Appendix B>, "Quality Assurance Criteria for Nuclear Power Plants and Reprocessing Plants."

7. NRC <Regulatory Guide 1.61>, "Damping Values for Seismic Design of Nuclear Power Plants."

8. NRC <Regulatory Guide 1.89>, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants."

9. NRC <Regulatory Guide 1.92>, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

10. NRC <Regulatory Guide 1.100>, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants."

11. IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

12. IEEE Std. 344-1971, "Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

13. IEEE Std. 344-1975, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

3.11   ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL
         EQUIPMENT

This section describes the program for environmental qualification of
mechanical and electrical equipment. The program identifies the
mechanical and electrical equipment to be qualified, defines the
environmental conditions under normal, abnormal and accident
conditions, and documents the qualification tests and analysis employed
to demonstrate the equipment's capability to perform design safety
functions when exposed to normal, abnormal, accident, and postaccident
environments. Seismic qualification is addressed in <Section 3.10> for
mechanical and electrical equipment.

3.11.1   EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

This section identifies: 1) the mechanical and electrical systems and
equipment that are required to perform a design safety function and
2) the environmental design bases for that equipment, including the
definition of the normal, abnormal, accident, and postaccident
environments.

3.11.1.1   Equipment Identification

Mechanical and electrical equipment to be qualified includes equipment
associated with systems that are essential to:

a.   Emergency reactor shutdown.

b.   Containment isolation.

c. Reactor core cooling.

d. Containment heat removal.

e. Reactor heat removal, or

f. Preventing significant release of radioactive material to the environment.

That is, equipment that:

a. Performs the above functions automatically.

b. Is used by the operator to perform these functions manually, or

c. The failure of which can prevent the satisfactory accomplishment of one or more of the above safety functions.

A list of these systems is provided in <Table 3.2-1>. The mechanical and electrical equipment associated with the design safety functions of these systems are designated as Safety Class 1, 2 or 3 (see <Section 3.2.3> for definitions) in <Table 3.2-1> and are designated generally as "safety-related" equipment.

For environmental qualification, "safety-related" equipment is categorized in three groups by design safety function:

a. Safety-related electrical equipment - designated as "Class 1E" per IEEE Standard 279-1971 (Reference 1).

b.   Safety-related mechanical equipment

   1.   "Active" Mechanical Equipment - that equipment which must
        move or change position to perform its design safety function
        (examples are pumps, motor-operated valves, safety/relief
        valves, or check valves).

   2.   "Passive" Mechanical Equipment - that equipment which must
        only maintain its pressure integrity to perform its design
        safety function (examples are tanks, heat exchangers and
        manual valves).

The design safety functions for specific equipment items are discussed
on a system basis in <Section 3.4>, <Section 3.5>, <Section 3.6>,
<Chapter 5>, <Chapter 6>, <Chapter 7>, <Chapter 8>, <Chapter 9>,
<Chapter 12>, and <Chapter 15>. A discussion of the listing of the
specific equipment by MPL No. that was prepared to demonstrate that the
equipment is qualified to perform its design safety function for the
normal, abnormal, accident, and postaccident environments in which it
is located is provided in <Section 3.11.3>.

3.11.1.2   Environmental Conditions

Safety-related mechanical and electrical equipment is located for
environmental qualification purposes by environmental zones within the
various plant buildings. <Figure 3.11-10> provides a list of
environmental zone descriptions including typical equipment and systems
located in each zone. The zone boundaries were determined based on
contiguous areas with similar environmental conditions by review of
<Section 3.6> for pipe rupture locations, <Section 9.4> for HVAC system
boundaries and <Chapter 12> for radiation doses. The environmental
zone boundaries are shown on plant layout drawings called
"Environmental Zone Maps" in <Figure 3.11-39>, <Figure 3.11-40>,
<Figure 3.11-41>, <Figure 3.11-42>, <Figure 3.11-43>, <Figure 3.11-44>,

<Figure 3.11-45>, <Figure 3.11-46>, and <Figure 3.11-47>. Areas where
no safety-related equipment is located are not assigned an
environmental zone.

The normal, abnormal and accident (including postaccident)
environmental conditions for each environmental zone are provided in
terms of temperature, pressure, relative humidity, and radiation in
<Figure 3.11-11>, <Figure 3.11-12>, <Figure 3.11-13>, <Figure 3.11-14>,
<Figure 3.11-15>, <Figure 3.11-16>, <Figure 3.11-17>, <Figure 3.11-18>,
<Figure 3.11-19>, <Figure 3.11-20>, <Figure 3.11-21>, <Figure 3.11-22>,
<Figure 3.11-23>, <Figure 3.11-24>, <Figure 3.11-25>, <Figure 3.11-26>,
<Figure 3.11-27>, <Figure 3.11-28>, <Figure 3.11-29>, <Figure 3.11-30>,
<Figure 3.11-31>, <Figure 3.11-32>, <Figure 3.11-33>, <Figure 3.11-34>,
<Figure 3.11-35>, <Figure 3.11-36>, <Figure 3.11-37>, and
<Figure 3.11-38>. Definitions used in determining the environmental
conditions follow:

a.   Normal Conditions - planned, purposeful, unrestricted reactor
     operating modes that include startup, power range and hot standby
     (condenser available), shutdown, and refueling modes.

b.   Abnormal Conditions - any deviation from normal conditions
     anticipated to occur often enough that design should include a
     capability to withstand the conditions without operation
     impairment; planned testing including preoperational tests are
     also considered abnormal conditions (loss of nonsafety-related
     HVAC is an example of an abnormal condition)

c.   Accident Conditions - a single event not reasonably expected
     during the course of plant operations that has been hypothesized
     for analysis purposes or postulated from unlikely but possible
     situations or that has the potential to cause a release of
     radioactive material (a reactor coolant pressure boundary rupture

may qualify as an accident; a fuel cladding defect does not).
Accident conditions are calculated for a postaccident period
sufficient to ensure that steady-state conditions have been
reached.

The environmental parameters shown are based on verified design
calculations and do not include margins required in qualification
testing or analysis as described in <Section 3.11.2>. The design basis
used for preventing the loss of ventilation for some zones is discussed
in <Section 3.11.4>. The basis for the estimated chemical and
radiation environmental conditions is discussed in <Section 3.11.5>.
Environmental zones and conditions are described in <Figure 3.11-10>,
<Figure 3.11-11>, <Figure 3.11-12>, <Figure 3.11-13>, <Figure 3.11-14>,
<Figure 3.11-15>, <Figure 3.11-16>, <Figure 3.11-17>, <Figure 3.11-18>,
<Figure 3.11-19>, <Figure 3.11-20>, <Figure 3.11-21>, <Figure 3.11-22>,
<Figure 3.11-23>, <Figure 3.11-24>, <Figure 3.11-25>, <Figure 3.11-26>,
<Figure 3.11-27>, <Figure 3.11-28> <Figure 3.11-29>, <Figure 3.11-30>,
<Figure 3.11-31>, <Figure 3.11-32>, <Figure 3.11-33>, <Figure 3.11-34>,
<Figure 3.11-35>, <Figure 3.11-36>, <Figure 3.11-37>, and
<Figure 3.11-38>. Environmental zone boundaries are shown in
<Figure 3.11-39>, <Figure 3.11-40>, <Figure 3.11-41>, <Figure 3.11-42>,
<Figure 3.11-43>, <Figure 3.11-44>, <Figure 3.11-45>, <Figure 3.11-46>,
and <Figure 3.11-47>. These figures will be revised only when a
modification in equipment or components affects environmental
qualification conditions.

To aid in the application of the Qualification Program acceptance
criteria that is discussed in <Section 3.11.2>, the environmental zones

listed in Figures 3.11-10 through 3.11-38 have been classified as either a harsh or mild environment per the following definitions:

a. Harsh environments - those zones where the environmental conditions exceed significantly the normal or abnormal range as a result of a DBE.

b. Mild environments - those zones where the environmental conditions do not exceed significantly the normal or abnormal range as a result of a DBE.

3.11.2   ENVIRONMENTAL QUALIFICATION PROGRAM ACCEPTANCE CRITERIA

This section describes the environmental qualification program acceptance criteria that were employed to meet the following general requirements:

a. The equipment was designed to have the capability of performing its design safety functions under all normal, abnormal, accident, and postaccident environments for the length of time for which its function is required.

b. The equipment environmental capability was demonstrated by appropriate testing and/or analyses.

c. A quality assurance program meeting the requirements of <10 CFR 50, Appendix B>, was established and implemented to provide assurance that all requirements have been satisfactorily accomplished.

3.11.2.1   Conformance with Regulatory Requirements

3.11.2.1.1   <10 CFR 50, Appendix A> - Criterion 4 - Environmental
                 and Dynamic Effects Design Bases

The scope of mechanical and electrical equipment and the environmental
requirements for GDC 4 are addressed in <Section 3.11.1>.

Mechanical and electrical equipment that is required to perform a
design safety function is designed to accommodate the effects of, and
to be compatible with, the environmental conditions associated with
normal operation, maintenance, testing, and postulated accidents,
including loss-of-coolant accidents in accordance with GDC 4.

3.11.2.1.2   <10 CFR 50, Appendix B>

Mechanical and electrical equipment qualification is in compliance with
the project Quality Assurance Program. The Quality Assurance Program
meets the requirements of <10 CFR 50, Appendix B>. See <Chapter 17>
for a detailed discussion.

3.11.2.1.3   Regulatory Guides

3.11.2.1.3.1   <Regulatory Guide 1.9>

<Regulatory Guide 1.9>, concerning diesel generator units, is also
discussed in <Section 1.8> and <Section 8.1>. The diesel generator
units are qualified in accordance with IEEE Standard 387-1977
(Reference 2), as modified by <Regulatory Guide 1.9>.

3.11.2.1.3.2   <Regulatory Guide 1.40>

<Regulatory Guide 1.40> concerning continuous duty motors inside
containment is also discussed in <Section 1.8> and <Section 8.1>.

Continuous duty motors inside containment are type tested in accordance with IEEE Standard 334-1971 (Reference 3), as modified by the regulatory positions of <Regulatory Guide 1.40>.

3.11.2.1.3.3   <Regulatory Guide 1.63>

<Regulatory Guide 1.63>, concerning electrical penetrations, is also discussed in <Section 1.8> and <Section 8.1>. Containment electrical penetrations are qualified in accordance with IEEE Standard 317-1976 (Reference 4), as modified by <Regulatory Guide 1.63>.

3.11.2.1.3.4   <Regulatory Guide 1.73>

<Regulatory Guide 1.73>, concerning electric valve operators inside containment, is also discussed in <Section 1.8> and <Section 8.1>. Electric valve operators are qualified in accordance with IEEE Standard 382-1972 (Reference 5), as modified by the position of <Regulatory Guide 1.73>.

3.11.2.1.3.5   <Regulatory Guide 1.89>

<Regulatory Guide 1.89>, which pertains to the qualification of Class 1E equipment, is also discussed in <Section 1.8> and <Section 8.1>.

Class 1E equipment is qualified in accordance with IEEE Standard 323-1974 (Reference 7), as endorsed by <Regulatory Guide 1.89> with the following specific exceptions:

a.   NSSS Class 1E equipment located in mild environmental zones was procured and qualified to IEEE Standard 323-1971 (Reference 6).

b.   Regulatory Position C2. - The basis for radiological source terms used is discussed in <Section 3.11.5.2.2>.

c.   Additional specific guidance for type testing of cables, field splices and terminations is provided by IEEE Standard 383-1974 (Reference 8), <Table 8.1-2>.

d.   Specific criteria for assessing the acceptability of the environmental qualification program for safety-related electrical equipment in a harsh environment is provided by <NUREG-0588> Category I (Reference 9).

e.   The acceptance criteria for the environmental qualification of all safety-related equipment other than harsh environment electrical equipment is the following:

1.   The documentation required to demonstrate qualification of safety-related equipment other than Class 1E harsh are the "Design/Purchase" specifications. The specifications contain a description of the functional requirements for its specific environmental zone during normal and abnormal environmental conditions. A well supported maintenance/surveillance program in conjunction with a good preventive maintenance program will ensure that equipment that meets the specifications is qualified for the designed life.

2.   The maintenance/surveillance program data and records will be reviewed periodically (not more than 24 months) to ensure that the design qualified life has not suffered thermal and cyclic degradation resulting from the accumulated stresses triggered by the abnormal environmental conditions and the normal wear due to its service condition. Engineering judgment shall be used to modify the replacement program and/or replace the equipment deemed necessary.

3.11.2.2   Qualification Methodologies for Class 1E Harsh Equipment

Safety-related electrical equipment in a harsh environment supplied by
GE under the NSSS contract is qualified as outlined in GE Licensing
Topical Report NEDE-24326-P, dated August 1982, General Electric
Environmental Qualification Program. All other safety-related
electrical equipment located in a harsh environment is qualified using
the methodologies in this section. Safety-related electrical and
mechanical equipment located in a mild environment is qualified as
discussed in Exception e to <Regulatory Guide 1.89>,
<Section 3.11.2.1.3.5(e)>, and <Section 1.8>. The documentation of the
application of the methodologies for the specific equipment identified
in <Section 3.11.1.1>, to demonstrate qualification to the
environmental conditions defined in <Section 3.11.1.2>, is presented in
<Section 3.11.3>.

Qualification of safety-related electrical equipment in a harsh
environment is accomplished by type testing, analysis and/or documented
operating experience. Although type testing is the preferred method of
qualification, equipment qualification usually involves some
combination of the three methods. The qualification methods used
depend on a number of factors, including:

a.   Material used in construction of the equipment.

b.   Applicable normal, abnormal and accident conditions.

c.   Operational requirements (during and after accidents).

d.   Nature of safety function(s).

e.   Size of equipment.

f.   Dynamic characteristics of expected failure modes (structural or functional).

In general, analysis is used to supplement test data, although simple components may lend themselves to analysis in lieu of full scale testing. The role of operating experience is generally limited to aiding in determining realistic performance goals.

Equipment samples selected for qualification are of the same basic design and materials as the equipment to be installed at Perry. The sample is manufactured using similar techniques and processes as those used for the installed equipment. Any significant variations or deviations are noted in the qualification results with justification provided as necessary.

3.11.2.2.1   Type Testing

Type testing is designed to demonstrate that the safety-related electrical equipment can perform its safety function(s) within the accuracy and response time requirements applicable for normal, abnormal and accident conditions. The type test consists of a demonstration of safety functions under a planned sequence of conditions, both before and after age conditioning.

A test plan is prepared at the beginning of the type testing program, which includes the basic methodology of the program, its intent and purpose, and sufficient detail to describe the tests and demonstrate compatibility with the requirements. As a minimum, it includes:

a.   Equipment description.

b.   Number of test specimens.

c.   Acceptance criteria.

d.   Failure definition.

e.   Testing sequence.

f.   Aging technique with appropriate justification.

g.   Test levels and service conditions.

h.   Parameters to be monitored.

i.   Test equipment to be used.

j.   Mounting and connection methods.

k.   Qualified life goal and design life.

l.   Documentation to be maintained.

3.11.2.2.1.1   Sequence

Type testing is done in the following sequence:

a.   Demonstration of required safety functions during normal service
      conditions.

b.   Demonstration of required safety functions at the extreme of the
      abnormal service conditions.

c.   Age conditioning.

d.   Seismic testing.

e.   Demonstration of required safety functions during and after
      accident conditions.

The same units are used for steps (a) through (e).

3.11.2.2.1.2   Test Specification

The type test is performed in accordance with a manufacturer prepared
test specification. This specification expands the approved test plan
into a detailed step-by-step description of how to implement the test
plan. The specification includes details of the tests and is submitted
as part of the test result.

3.11.2.2.1.3   Margin

Margin is the difference between the most severe calculated service
condition of the plant and the conditions used in type testing. Margin
is necessary to account for normal variations in production and
reasonable error in calculating service conditions. Margin has been
achieved, where practicable in the suppliers' generic type tests, by
increasing the levels of test parameters, the number of test cycles and
the test duration.

3.11.2.2.1.4   Acceptance Criteria

The type test program is designed to show that the equipment can
perform its design safety function under the environmental conditions,
specified under <Section 3.11.1>, with margin as defined above.

3.11.2.2.1.5   Testing Under Normal Conditions

The equipment is installed and operated in a manner which simulates the
intended normal inservice conditions. The equipment is exercised to
demonstrate performance of its safety functions. Data is recorded for
later reference.

3.11.2.2.1.6   Testing Under Abnormal and Accident Conditions

After receiving the accumulated exposure associated with normal and
abnormal environmental conditions, the equipment is installed and
operated while being exposed to at least the environmental limit
conditions sequentially for the function times specified.

3.11.2.2.1.7   Aging

There are four basic options that can be exercised pertaining to aging.
The first option is an analysis of equipment design and service
conditions to determine if long term operation affects design safety
function performance during the anticipated service conditions. If
such effects could occur, the second option is to examine similar units
already qualified to see if the previous results demonstrate that such
effects will not affect the performance of the unit. Third, if no
similar units have been qualified, an attempt is made to locate units
that have been in operation under similar conditions for an extended
period of time. When no natural aged units are available, the fourth
option is to develop and implement an age conditioning procedure.
These four options are discussed in the following sections. An
alternate method of addressing aging is presented in
<Section 3.11.2.2.1.7.8>.

3.11.2.2.1.7.1   Analysis

Analysis is used to demonstrate that the equipment suffers no appreciable change in ability to perform, due to the service conditions associated with high stress events at any time in the qualified life. This method is limited to the following classes of equipment:

a.   Equipment which is simple in design and construction (e.g., cabinets, frame structures, panels, instrument racks).

b.   Equipment where the design basis event does not impose stresses additive to those imposed during normal operation in such a manner as to cause a common mode failure.

c.   Equipment that is similar to the existing qualified equipment and the differences are minor.

d.   Equipment which has no significant aging mechanisms over its qualified life.

If no appreciable degradation is anticipated, no age conditioning is performed. Such analyses reference applicable vendor test results, operating experience and test results on equipment of similar design and include a systematic examination of the significant aging mechanisms.

3.11.2.2.1.7.2   Similarity

For some equipment types, the only additional stress during life is vibration (nonseismic). The aging mechanisms of such equipment can be divided into the effects at the component level, and the interaction between components when subjected to vibration induced stresses. For example, the aging mechanisms of seals, carbon resistors, printed circuits, junctions, solder joints, and wiring may not differ from one

module to a similar module. If the qualified life of one module can be established, then modules of similar types will have an equivalent qualified life if all modules have similar failure mechanisms. For the modules to be qualified, various types of equipment are compared for similarity or grouping by comparing the following items:

a.   Type of technology used to design and manufacture the module.

b.   Type of critical components.

c.   Packaging, mounting and type of connections.

d.   Service conditions.

e.   Safety functions.

For similar types of modules, all modules are type tested excluding aging. Some representative modules have an additional specimen type tested including aging. If the representative modules show no change in test results, whether aged or not aged, aging is deemed to have no effect on safety function performance and, therefore, aging would have no effect on safety function performance of the remainder of the similar group. However, if significant differences in performance between aged and unaged modules are found, similarity is not used.

3.11.2.2.1.7.3   Natural Aging

An effort is made to locate identical or similar equipment that has been exposed to typical service conditions over an extended period of time. Where not all service conditions were present in sufficient severity, conditioning is performed to account for the missing effects.

3.11.2.2.1.7.4   Age Conditioning

Age conditioning is a process of controlled physical deterioration that
can provide a qualitative evaluation of the equipment vulnerability to
aging effects that may affect its ability to perform its design safety
functions. Age conditioning stresses are intended to produce equipment
degradation levels that equal or exceed expected inservice degradation.
Age conditioning addresses the effects of temperature, humidity,
pressure, radiation, vibration, chemical atmosphere, power supply,
monitored process, and operational cycles.

3.11.2.2.1.7.4.1   Thermal Aging

Thermal aging accelerates those aging processes that are the result of
chemical reactions and various physical processes at the molecular
level, such as diffusion.

Thermal acceleration factors are extracted from published test data
when available. Where unavailable, or the validity is doubtful, the
Arrhenius approach is used to relate accelerated thermal aging time to
equivalent natural aging time. The formula is:

$$L_N = L_A \exp\left[-\frac{\Delta E}{K}\left(\frac{1}{T_A} - \frac{1}{T_N}\right)\right]$$

where:

    T_N = normal temperature, °K

    T_A = elevated temperature, °K

    L_N = life at normal temperature

L_A = life at elevated temperature

ΔE = activation energy in electron-volts for life determining
aging process

K = Boltzmann's constant = 8.617×10^{-5} eV/°K

Where the activation energy, ΔE, is not known, one of the following
procedures is employed:

a. ΔE is determined experimentally in accordance with the procedures
in IEEE Standard 98-1972 (Reference 10), IEEE Standard 99-1970
(Reference 11), and IEEE Standard 101-1972 (Reference 12), or
EPRI NP-1558 (Reference 13), or other established standard
procedures, or

b. Based on known activation energies for thermal degradation of
similar materials, a conservative lower bound is determined and
used inplace of the actual value of ΔE.

Acceleration factors and activation energies may be determined by aging
tests performed on individual components and material samples.

The normal temperature used for the accelerated life calculations is
greater than or equal to that actually experienced by the individual
components of the device, due to the combined effects of ambient
temperature, process fluid temperature, heat internally generated by
operation, and heat transmitted from other equipment located in the
proximity of the device.

The normal and accelerated lives, L_N and L_A, are determined by the
deterioration of some relevant material property beyond a predetermined
level.

Where thermal aging rates for different materials comprising a device vary significantly, one of the following procedures may be used to prevent failure due to excessive aging:

a.   The component(s) in question may be replaced with new items at such a time or times during the aging procedure as to ensure that the remaining period of thermal aging for the device as a whole will adequately age these components; or

b.   Components which age relatively slowly will be individually subjected to a period of accelerated aging prior to incorporation in the test unit which will then be subjected to further accelerated aging.

3.11.2.2.1.7.4.2   Radiation Aging

Radiation aging consists of exposure to a radiation dose equivalent to the expected integrated dose to which the device is exposed during its installed life. Design basis event radiation exposure is added to the normal service exposure so that only one period of radiation exposure is required during the testing. The test radiation includes all significant types of radiation that occur in normal/abnormal and DBE exposure <Section 3.11.5.2>.

In calculating the equivalent test radiation dose for an accelerated exposure rate, the effects of oxidation-gas diffusion are accounted for by applying a greater total dose than the expected integrated dose during the service life. Either IEEE Standard 278-1967 (ANSI N4.1-1967), "Guide for Classifying Electrical Insulating Materials Exposed to Neutron and Gamma Radiation" or ASTM D 2953-71, "Classification System for Polymeric Materials for Service in Ionizing Radiation," are used as applicable.

3.11.2.2.1.7.4.3   Operational Cycling

Accelerated operational cycling is performed for each type of stress
that varies significantly and which may thereby constitute an aging
factor. Such stresses include:

a.   Process fluid temperature and pressure variations.

b.   Power on-off cycling.

c.   Operation of electromechanical devices such as relays and
     switches.

d.   Temperature cycling (generally associated with power on-off
     cycling).

The number of cycles is based on the number expected during the
projected qualified life.

3.11.2.2.1.7.4.4   Process Fluid (Applicable Equipment)

Long term normal exposure to process fluid may result in chemical
deterioration (e.g., corrosion of metal or deterioration of seals).
Such effects may be shown by analysis to be negligible on the basis of
published test data.

Where such effects are significant, a period of exposure at elevated
temperatures is included as part of the general aging procedure. The
provisions and criteria relating to thermal aging are applied here.

Variations in process fluid pressure and temperature are a part of the
operational cycling procedure.

3.11.2.2.1.7.5   Aging Procedures for Specific Components and
                        Materials

3.11.2.2.1.7.5.1   Structural Metal

In general, no environmental testing or aging will be performed on
structural metal except as incidental to such procedures being
performed on supported components.

3.11.2.2.1.7.5.2   Nonmetallic Materials

The primary aging procedures are thermal aging and radiation aging when
applicable to cable and wire jacketing, terminal blocks, insulating and
structural components in electromechanical devices, seals, printed
wiring boards, connector installing materials, etc.

3.11.2.2.1.7.5.3   Solid State Electronics Components

Radiation aging is performed where applicable. Except for radiation,
in general, it is not expected that aging will degrade these components
in such a way as to make them more susceptible to environmental stress.
Consequently, analysis may be used to determine the effect of
electronic component aging on the qualified life of the equipment. If
aging is performed, the relevant procedures include thermal aging.

3.11.2.2.1.7.5.4   Printed Circuit Boards

Thermal aging, radiation aging, when applicable, and vibration aging,
when applicable, are applied.

3.11.2.2.1.7.5.5   Capacitors

Thermal aging, voltage aging, temperature/humidity aging and radiation
aging, when applicable, are applied.

3.11.2.2.1.7.5.6   Transformers

Thermal aging, temperature/humidity aging, radiation aging, when applicable, and gas exposure are applied.

3.11.2.2.1.7.5.7   Resistors

Thermal aging only is applied.

3.11.2.2.1.7.5.8   Moving Parts

Operational cycling only (see electrical contacts) is performed.

3.11.2.2.1.7.5.9   Electrical Contacts

Temperature/humidity aging, thermal aging, operational cycling, and gas exposure is applied.

3.11.2.2.1.7.6   Implementation of Age Conditioning Methods

This section describes the approaches that form the basis for implementing the age conditioning methods described in <Section 3.11.2.2.1.7.4>.

3.11.2.2.1.7.6.1   Component Approach

Each component of the module is conditioned separately and then the module is assembled. The requirements to justify this approach are to show that the significant aging mechanisms of each component are not amplified or accelerated by the surrounding components and no significant aging mechanism involves more than one component at a time.

3.11.2.2.1.7.6.2   Critical Components Approach

The module is examined for a component with a clearly dominant age
related failure mode in the various service conditions to which the
module will be subjected. This is supported by MIL handbooks, vendor
test results, literature on test results of similar components, or
testing samples at the various environmental levels required. The
conditioning, determined for the critical components, is applied to the
entire assembled module. The requirements to justify this approach
include a determination of the amount of conditioning given to the
balance of the components.

3.11.2.2.1.7.6.3   Split-Phase Approach

Components are individually conditioned with respect to significant
aging mechanisms and at acceleration rates applicable to the
components. This is done in Phase I for durations so that each
component reaches a prespecified apparent time (advanced age). The
prespecified apparent time may be different from component to component
at the end of Phase I, and further advanced conditioning is done on the
module until all the components in the module arrive at the desired
condition.

3.11.2.2.1.7.6.4   Module-Level Approach

The entire module is conditioned to a point where some of the
components are conditioned to the end of qualified life and the
remainder are conditioned to a point of known or estimated life.

3.11.2.2.1.7.7   Disposition of Failures During Age Conditioning

Equipment failures during age conditioning are considered to exist if
testing indicates that the equipment cannot be calibrated within
specification. Random failures are allowed during age conditioning.

Should a failure occur during the age conditioning, the following steps are taken to determine the necessary action:

a. Identify the failed part(s) and replace. Verify proper operation by baseline tests after replacement.

b. Determine the failure mechanism and evaluate the relationship to the aging process.

c. If unrelated to the aging stresses, the failure is to be considered random and conditioning may be continued.

d. If the failure could have been prevented by periodic maintenance, each maintenance is specified as a requirement on the final user. Continue with conditioning.

e. If the component has been overstressed by the conditioning procedure, the conditioning procedure must be suitably revised and started over again.

Otherwise, the equipment has failed. Either equipment redesign or reduction of qualified life may resolve the problem. Age conditioning is reperformed after such resolution.

When Items c, d or e, above, are determined to be applicable, justification of such determination is included in the test results report.

3.11.2.2.1.7.8   Qualified Life

When no significant aging mechanisms have been found, a qualified life equal to the plant life is assigned. When similarity is used, a qualified life is assigned equal to that of the age conditioned similar unit. When natural aging is used, the qualified life will not exceed

the actual age of the unit. When age conditioning is used, the determination of qualified life is based upon conservative engineering analysis which takes the following into account as available and applicable:

a.   Results of age conditioning.

b.   Equipment operating data.

c.   Existing test results.

d.   Reliability data.

e.   Physical understanding of significant aging mechanisms that have been identified.

f.   The equipment expected inservice duty cycle.

g.   The estimated rate of aging based on the results of the conditioning process.

h.   The expected inservice maintenance procedure and schedule. The qualified life will be expressed in the most purposeful terms for the particular application.

Justification is provided in the qualification documentation of the means used to estimate the qualified life.

3.11.2.2.1.7.9   Service Conditions

For the aging program, the following service conditions are used to determine the contributing factors for the equipment qualified life:

a.   Normal service conditions.

b.   Abnormal service conditions.

3.11.2.2.2   Qualification Analysis

Analyses are used for environmental qualification when such analyses
can be shown to be conservative. In general, qualification by analysis
is limited to simple constructions and used to supplement the type
testing <Section 3.11.2.2.1>. Such analyses, performed for full
environmental qualification or to supplement the type testing, are
justified in the test plan, test specification or qualification report.

Class 1E equipment, classified as safety-related by association and not
required to function to mitigate the consequences of an accident, may
be exempted from qualification for harsh environment conditions by
analysis of the possible failure modes present when subject to that
harsh environment. Each of the possible failure modes is identified
and analyzed to determine if it has an effect on performance of the
safety functions of other safety-related equipment.

3.11.2.2.3   Operating Experience

Operating experience, if available, serves as a basis for determining
the qualified life of equipment (e.g., systems or elements, components,
modules, and other constituent parts of systems). IEEE
Standard 323-1974 defines operating experience as an "accumulation of
verifiable service data for conditions equivalent to those for which
the equipment is to be qualified."

In order for equipment to be qualified by reason of operating experience, auditable data must be available confirming that the following criteria have been met:

a. The equipment cited for operating experience is identical or justifiably similar to the equipment to be qualified.

b. The equipment, cited for operating experience, has operated under all service conditions which equal or exceed, in severity, the service conditions for which the equipment is to be qualified and performed its design safety functions under these conditions.

c. The normal and abnormal service condition requirements were satisfied prior to the occurrence of the accident conditions.

d. Margin is considered in determining the levels for which qualification is being sought.

When documentation of auditable data demonstrates that the criteria are met, the equipment is considered qualified by reason of operating experience for a time no longer than that length of time from the start of operation until the accident condition. This type of operating experience is a very limited method of qualifying equipment. However, when all criteria are not met, the equipment is a prospect for natural aging per <Section 3.11.2.2.1.7.3> or combined qualification per <Section 3.11.2.2.4>.

3.11.2.2.4   Combined Qualification

Equipment is qualified by test, analysis, previous operating experience, or any combination of these three methods. There are

various rationales for qualifying equipment by combining test, analysis and previous operating experience. Some examples are:

a.   Equipment is too complex for analysis alone or too large for testing alone.

b.   Tests of selected samples of a particular design group, that may vary in size but are made of similar materials and have similar design principles, may be augmented by extrapolation techniques to other sizes that have not been tested.

c.   Verification of a mathematical model by a partial test to determine mode shapes and resonant frequencies.

d.   Use of operating experience to develop a basis for simulated aging techniques.

e.   Analysis of an assembly to determine the environment to which components are to be tested.

f.   Analysis correlating the errors and response time between two subassemblies that have been qualified separately to demonstrate the required functional operability when the two subassemblies are combined.

The combined qualification will demonstrate that the equipment can perform its design safety function under normal, abnormal and accident conditions throughout its qualified life. Certain portions of the qualification may be demonstrated by operating experience. Some portions may be demonstrated by test. Still others may be demonstrated by analysis. Combined qualification provides suitable data by which the various primary qualification methods may be brought together to satisfy the requirements of the qualification program.

3.11.2.2.5   (Deleted)

3.11.2.2.6   Supplemental Qualification Concerns

3.11.2.2.6.1   Maintenance of Qualified Life

Qualification is not a guarantee of performance for each component of a
system. It is rather assurance that the system can perform its safety
function under all specified service conditions. To maintain the
qualified status of such equipment requires scheduled maintenance to
prevent components from exceeding their qualified life, and periodic
testing to locate components that may have failed. The minimum
scheduled maintenance required to maintain equipment qualification is
specified for all the harsh environment safety-related electrical
equipment. Nonscheduled maintenance will be required as a result of
out-of-spec performance, and the malfunctioning part(s) shall be
located and replaced by one equivalent to the original part. Such
replacement followed by an in-spec test result does not affect the
qualified status of the equipment unless evidence of common mode
failure exists. If analysis, retest or operation experience verifies
that a common mode failure exists, the failure is eliminated by
appropriate modifications,

redesign of the equipment, respecification of required maintenance, or reduction of qualified life. Such modifications are reviewed for impact on qualification in accordance with <Section 3.11.2.2.6.3>.

3.11.2.2.6.2   Extension of Qualified Life

The original qualified life of certain safety-related control and instrumentation equipment may be less than the plant design life. Numerous methods exist by which the qualified life for such equipment may be extended to meet the plant design life. This may be accomplished by a replacement program, additional tests, further analysis, operating experience, or other demonstration that the equipment can perform its safety function for an additional specified period of time. When modules have components with a qualified life that limits overall equipment qualified life or when experience determines that the chance of successful extension is low, a replacement program is initiated.

Analysis is used when physically measurable parameters are identified that accurately reflect the state of deterioration. By periodic examination of these parameters, a more accurate determination of actual in situ aging rate may be obtained with a correspondingly more accurate evaluation of qualified life, based on this analyzed operating experience.

3.11.2.2.6.3   Design Modifications during the Qualification Program

Due to the nature of a qualification test program, changes to both procedures and equipment are not uncommon during the progress of the test. During qualification, such modifications may be required to provide a more realistic or practicable method of conducting the program. Such modifications may be made without affecting the validity of the results only when the adequate justification is provided.

Such modification to procedures or equipment is evaluated to determine its impact on the equipment qualification. This evaluation determines the extent of requalification required. The results of this evaluation, with a full description of the modification, is to be included in the test report. Any equipment changes made are also to be made to all production units.

Where changes neither affect the significant aging mechanisms nor affect the possible failure mode, requalification is not required. Where there is such an effect, sufficient requalification is performed to determine that the equipment can perform its design safety function for the specified period of time.

3.11.2.2.6.4   Design Modifications During Construction/Operations

During the course of the plant construction it may be necessary to modify equipment that has already been qualified and installed. Such modifications may be made without affecting the validity of existing qualification results only when appropriate justification is provided.

Such modification to existing equipment is evaluated to determine its impact on the existing qualification report. This evaluation determines the extent of requalification required. During construction, the results of this evaluation, with a full description of the modification, were added to the qualification report. During operations, the results of this evaluation, with a reference to the modification, will be added to the auditable file for all harsh environment, safety-related, electrical equipment.

Where changes neither affect the significant aging mechanisms nor affect the possible failure mode, requalification is not required. Where there is such an effect, sufficient requalification is performed to determine that the modified equipment can perform its design safety function for the specified period of time.

3.11.3   QUALIFICATION TEST RESULTS

The results of qualification testing are only required for each type of
harsh environment safety-related electrical equipment. These results
are recorded in the Equipment List and stored in an auditable file.

3.11.3.1   Equipment List

The Equipment List is a data base used to identify the qualified
safety-related electrical equipment in the plant. In addition to
identifying the equipment, the equipment list will provide other
information to support the qualification.

a.   The identified Class 1E equipment will have the following
      information provided:

      1.   Equipment Number (MPL No.).

      2.   Equipment Description.

      3.   Manufacturer and Model Number.

      4.   Equipment location by building, column line cross and
            elevation.

      5.   Equipment support by parent MPL or mounting configuration.

      6.   Equipment category as defined by <NUREG-0588>, Appendix E,
            Item 2.

      7.   Equipment function time required to perform the design safety
            function.

      8.   Type of environmental zone (harsh).

9.   Zone relating to environmental conditions <Figure 3.11-10>.

10.   Reference to the Auditable File Package.

The above format is required for all 1E equipment located in a harsh environment.

3.11.3.2   Auditable File

The auditable file is arranged in packages by harsh environment safety-related electrical equipment specifications. The Auditable File Package (AFP) format outlined below is required for all 1E equipment installed in a harsh environment to establish environmental qualification in accordance with the requirements of <10 CFR 50.49>.

a.   A cover sheet identifying the type of package, the type of equipment covered and a unique package identification number.

b.   Package revision sheets indicating the revision level of the package and documenting the continued qualification of the equipment.

c.   A package index.

d.   The original statement of equipment qualification.

e.   An equipment listing of the safety-related equipment covered by the package by Equipment Number (MPL No.). This is a sort by specification of the Equipment List discussed in <Section 3.10.4.1>.

f.   The checklist(s) used to perform the evaluation of the qualification report. (The Environmental Qualification Report checklist meets the intent of <NUREG-0588>).

g.   A listing or reference to the specific performance criteria the equipment is required to meet to be qualified.

h.   Reference to the Vendor's Qualification Report(s).

i.   Documentation, such as calculations, which support the acceptability of the package.

j.   Correspondence which supports the acceptability of the package.

k.   Special maintenance requirements applicable to the equipment covered by the package.

l.   A design modification cross-reference for modifications discussed in <Section 3.11.2.2.6.4>.

m.   Industry experience notices applicable to the qualification of the equipment covered by the package.

3.11.4   LOSS OF VENTILATION

Loss of ventilation systems during accident conditions will not affect
the qualification of safety-related mechanical and electrical equipment
in the environmental zones listed below, that are served by
safety-related HVAC systems.

a.   Control room (Zone CB-1).

b.   MCC, switchgear and miscellaneous equipment areas (Zone CB-2).

c.   ECCS pump rooms (Zones AB-2, 3 and 4).

d.   Diesel generator building (Zone DG-1).

e.   Fuel handling area (Zones FB-1, 2, 3).

f.   Emergency closed cooling pump area (Zone CB-5).

g.   Emergency service water pumphouse (Zone ES-W).

This section discusses the design of the safety-related HVAC system in
these environmental zones where redundant systems are provided to
preclude loss of HVAC to redundant safety-related mechanical and
electrical equipment in that zone. All other environmental zones have
considered loss of the nonsafety-related HVAC system in determining the
limiting accident condition.

However, loss of ventilation, both safety-related and
nonsafety-related, has been considered as a limiting abnormal condition
except in those zones where the redundant safety-related systems both
serve the entire zone. These zones are:

a.   Control room (Zone CB-1).

b. Emergency closed cooling pump area (Zone CB-5).

c. Emergency service water pumphouse (Zone ES-W).

3.11.4.1   Control Room HVAC/Emergency Recirculation System

The control room is air conditioned and shielded against radiation to
allow the operators safe and continued occupancy under optimum
environmental conditions <Section 6.4>. Redundant control room HVAC
components are designed to satisfy Safety Class 3, Seismic Category I
requirements. The control complex chilled water system is designed and
fabricated in accordance with the ASME Code, Section III
<Section 9.4.9>. Upon loss of offsite power, emergency power from the
onsite diesel generator sets is automatically supplied to the
equipment. No single failure can result in a loss of control room
HVAC, which ensures that the operability of safety-related control and
electrical equipment located in the control room is not adversely
affected.

3.11.4.2   MCC Switchgear and Miscellaneous Electrical Equipment
            Areas HVAC/Battery Room Exhaust Rooms Ventilation

Redundant Class 1E equipment that supplies power for safety-related
equipment is located in separate MCC, switchgear and miscellaneous
electrical equipment areas of the control complex. Redundant HVAC
equipment for these areas is designed to satisfy Safety Class 3,
Seismic Category I requirements <Section 9.4.1>. The control complex
chilled water system is designed and fabricated in accordance with the
ASME Code, Section III <Section 9.4.9>. The HVAC system is supplied
with onsite emergency power from the associated standby diesel
generator should loss of offsite power occur. No single failure will
result in loss of HVAC in both MCC, switchgear and miscellaneous
electrical equipment areas.

3.11.4.3   Emergency Core Cooling System Pump Rooms

Each train of each system of the ECCS is located in separate
compartments of the auxiliary building. The ECCS pump rooms are
maintained between the temperatures shown on <Figure 3.11-11>,
<Figure 3.11-12>, <Figure 3.11-13>, and <Figure 3.11-14> for normal
operating conditions (pumps not running) and under operating conditions
with the ECCS pumps and associated electrical equipment operating at
full design capacity under accident conditions or during pump testing.

The ECCS pump room coolers are designed to satisfy Safety Class 3,
Seismic Category I requirements <Section 9.4.5>. Onsite emergency
power is supplied to each pump room cooler from the associated standby
diesel generator should offsite power be lost. No single failure can
result in loss of cooling in more than one division of ECCS pump rooms.

3.11.4.4   Diesel Generator Building Ventilation System

The diesel generator building ventilation system is designed to
maintain the area within the temperatures provided in <Figure 3.11-27>
when the diesel generators are operating and outdoor temperature is
less than or equal to 95°F. Relative humidity corresponding to this
temperature will be less than 50 percent since the building load will
consist almost entirely of sensible heat.

Each diesel generator is provided with an independent ventilation
system consisting of outdoor air intake louvers and two redundant
ventilating fans. These ventilation systems are designed to satisfy
Safety Class 3, Seismic Category I requirements and are supplied with
onsite emergency power from the associated standby diesel generators
should loss of offsite power occur <Section 9.4.5>. No single failure
can result in loss of cooling in more than one diesel generator room.

3.11.4.5   Fuel Handling Area Ventilation System

The fuel handling area ventilation system is designed to maintain the
area at or below the values shown on <Figure 3.11-32 (1)> and
<Figure 3.11-32 (2)> with an outdoor temperature of 95°F, and to
mitigate the consequences of a fuel handling accident <Section 15.7.4>
and <Section 15.7.6>. The redundant ventilation systems are designed
to satisfy Safety Class 3, Seismic Category I requirements. They may
be supplied by operator action with onsite emergency power from the
standby diesel generators should loss of offsite power occur
<Section 9.4.2>. No single failure can result in loss of ventilation
to the fuel handling area.

3.11.4.6   Emergency Closed Cooling Pump Area HVAC

The emergency closed cooling pump area HVAC is designed to maintain the
area at or below the values shown on <Figure 3.11-19> with both Unit 1
and Unit 2 ECC pumps running, piping heat loss, and instrument air
compressors, service air compressors, control complex chillers, and
chilled water pumps running. Unit 2 was subsequently abandoned. The
redundant HVAC units are designed to satisfy Safety Class 3, Seismic
Category I requirements <Section 9.4.5>. The control complex chilled
water system is designed and fabricated in accordance with ASME Code,
Section III <Section 9.4.9>. Both systems are supplied with onsite
emergency power from the standby diesel generators should loss of
offsite power occur. No single failure can result in loss of HVAC to
the ECC pump area.

3.11.4.7   Emergency Service Water Pumphouse Ventilation

The emergency service water pumphouse ventilation system is designed
with redundant fans, such that one fan can maintain the area at or
below the values shown on <Figure 3.11-31> with both ESW pumps "A" and
"B". The systems are designed to satisfy Safety Class 3, Seismic

Category I requirements <Section 9.4.5>. The system is supplied with onsite emergency power from standby diesel generators should loss of offsite power occur. No single failure can result in loss of ventilation to the ESW pumphouse.

3.11.5   ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

This section presents the justification for the estimated chemical and radiation environments of <Section 3.11.1.1> and <Figure 3.11-11>, <Figure 3.11-12>, <Figure 3.11-13>, <Figure 3.11-14>, <Figure 3.11-15>, <Figure 3.11-16>, <Figure 3.11-17>, <Figure 3.11-18>, <Figure 3.11-19>, <Figure 3.11-20>, <Figure 3.11-21>, <Figure 3.11-22>, <Figure 3.11-23>, <Figure 3.11-24>, <Figure 3.11-25>, <Figure 3.11-26>, <Figure 3.11-27>, <Figure 3.11-28>, <Figure 3.11-29>, <Figure 3.11-30>, <Figure 3.11-31>, <Figure 3.11-32>, <Figure 3.11-33>, <Figure 3.11-34>, <Figure 3.11-35>, <Figure 3.11-36>, <Figure 3.11-37>, and <Figure 3.11-38>.

3.11.5.1   Chemical Environment

3.11.5.1.1   Normal Operation

Water of the reactor, suppression pool, upper containment pool, fuel storage pools, fuel transfer system, residual heat removal system, and emergency core cooling systems is not chemically inhibited and is controlled by ion exchange systems to be compatible with the normal operating limits listed in <Table 3.11-13>.

Sampling capabilities are provided for periodic analysis of this water to assure compliance with operational limits.

3.11.5.1.2   Design Basis Accident

Water released from the reactor to the suppression pool, following a design basis accident and used for the containment spray, is calculated on the basis of <Regulatory Guide 1.7> to have a pH range of 4.5 to

7.0, a conductivity of ≤21 μS/CM, oxygen content of ≤8 ppm, a carbon dioxide content of ≤1 ppm, dissolved hydrogen of ≤60 ppb, dissolved salts of ≤2x10^{-5} g mole/L, and undissolved solids ≤9 ppm. No significant concentrations of airborne or waterborne deleterious chemicals have been identified due to the post-LOCA fission products.

For the design basis LOCA analysis, the suppression pool pH is maintained at 7 or above to minimize the conversion of cesium iodide to elemental iodine. The SLCS is used following the design basis LOCA for postaccident containment water chemistry management <Section 15.6.5.5.1.8>.

The containment spray system provides demineralized water as described above (for containment depressurization), at 5,250 gpm per train (A and B), 120 psig and 132°F from the containment spray headers. Provided that other aspects of the initiation logic are satisfied, the train A spray may be initiated automatically as soon as 10 minutes after a LOCA signal or manually at any time. The train B spray initiation logic is identical to train A except that an additional time delay is used in the design.

3.11.5.2   Radiation Environment

3.11.5.2.1   Normal Operation

Radiation sources during normal plant operations are identified in <Chapter 11> and <Chapter 12> for the various plant systems.

For the neutron and gamma sources, energy spectra information is provided in <Table 12.2-2>, <Table 12.2-3>, <Table 12.2-4>, and <Table 12.2-5>. Alpha and beta sources do not contribute to the integrated doses. The resulting normal radiation environments,

integrated over 40 years, are given in <Figure 3.11-11>,
<Figure 3.11-12>, <Figure 3.11-13>, <Figure 3.11-14>, <Figure 3.11-15>,
<Figure 3.11-16>, <Figure 3.11-17>, <Figure 3.11-18>, <Figure 3.11-19>,
<Figure 3.11-20>, <Figure 3.11-21>, <Figure 3.11-22>, <Figure 3.11-23>,
<Figure 3.11-24>, <Figure 3.11-25>, <Figure 3.11-26>, <Figure 3.11-27>,
<Figure 3.11-28>, <Figure 3.11-29>, <Figure 3.11-30>, <Figure 3.11-31>,
<Figure 3.11-32>, <Figure 3.11-33>, <Figure 3.11-34>, <Figure 3.11-35>,
<Figure 3.11-36>, <Figure 3.11-37>, and <Figure 3.11-38>

3.11.5.2.2   Design Basis Accident

The radiation doses from recirculating fluid lines used to determine
the equipment qualification environmental conditions are in accordance
with <NUREG-0588> for comment (dated December 1979) and <Regulatory
Guide 1.89>, and are based on the radiation sources given in
<NUREG-0737>, Section II.B.2. The postaccident radiation doses (for
equipment qualification purposes) are based on these more limiting
<NUREG-0737> sources as opposed to the source terms used for the
alternative accident source terms used in the LOCA submittal
(Reference 14). Equipment qualification environmental conditions are
reflected in the Environmental Tables on <Figure 3.11-11>,
<Figure 3.11-12>, <Figure 3.11-13>, <Figure 3.11-14>, <Figure 3.11-15>,
<Figure 3.11-16>, <Figure 3.11-17>, <Figure 3.11-18>, <Figure 3.11-19>,
<Figure 3.11-20>, <Figure 3.11-21>, <Figure 3.11-22>, <Figure 3.11-23>,
<Figure 3.11-24>, <Figure 3.11-25>, <Figure 3.11-26>, <Figure 3.11-27>,
<Figure 3.11-28>, <Figure 3.11-29>, <Figure 3.11-30>, <Figure 3.11-31>,
<Figure 3.11-32>, <Figure 3.11-33>, <Figure 3.11-34>, <Figure 3.11-35>,
<Figure 3.11-36>, <Figure 3.11-37>, and <Figure 3.11-38>. Accident
radiation environments are provided for both gamma and beta integrated
over six months.

3.11.6   REFERENCES FOR SECTION 3.11

1.   Institute of Electrical and Electronics Engineers (IEEE),
     "Criteria for Protection Systems for Nuclear Power Generating
     Stations," Standard 279 (ANSI N.42.7), dated 1971.

2.   Institute of Electrical and Electronics Engineers (IEEE),
     "Standard Criteria for Diesel Generator Units Applied as Standby
     Power Supplies for Nuclear Power Generating Stations,"
     Standard 387, dated 1977.

3.   Institute of Electrical and Electronics Engineers (IEEE),
     "Standard for Type Tests of Continuous Duty Class 1E Motors for
     Nuclear Power Generating Stations," Standard 334, dated 1971.

4.   Institute of Electrical and Electronics Engineers (IEEE),
     "Electric Penetration Assemblies in Containment Structures for
     Nuclear Power Generating Stations," Standard 317, dated 1976.

5.   Institute of Electrical and Electronics Engineers (IEEE),
     "Standard for Qualification of Safety-Related Valve Actuators,"
     Standard 382, dated 1972.

6.   Institute of Electrical and Electronics Engineers (IEEE), "General
     Guide for Qualifying Class 1E Electrical Equipment for Nuclear
     Power Generating Stations," Standard 323, dated 1971.

7.   Institute of Electrical and Electronics Engineers (IEEE),
     "Standard for Qualifying Class 1E Equipment Nuclear Power
     Generating Stations," Standard 323, dated 1974.

8.   Institute of Electrical and Electronics Engineers (IEEE), "Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations," Standard 383, dated 1974.

9.   <NUREG-0588>, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."

10. Institute of Electrical and Electronics Engineers (IEEE), "Guide for the Preparation of Test Procedures for the Thermal Evaluation and Establishment of Temperature Indices of Solid Electrical Insulating Materials," Standard 98, dated 1972.

11. Institute of Electrical and Electronics Engineers (IEEE), "Guide for the Preparation of Test Procedures for the Thermal Evaluation of Insulation Systems for Electrical Equipment," Standard 99, dated 1970.

12. Institute of Electrical and Electronics Engineers (IEEE), "Guide for Statistical Analysis of Thermal Life Test Data," Standard 101, dated 1972.

13. Electrical Power Research Institute (EPRI), "A Review of Equipment Aging Theory and Technology" NP-1558, Franklin Research Center.

14. Amendment No. 166 to Facility Operating License No. NPF-58 – Perry Nuclear Power Plant Unit 1 – Issuance of Amendment Concerning Full Implementation of Alternative Source Term (TAC No. MF3197).

<Table 3.11-1>

<Table 3.11-2>

<Table 3.11-3>

<Table 3.11-4>

<Table 3.11-5>

<Table 3.11-6>

<Table 3.11-7>

<Table 3.11-8>

<Table 3.11-9>

<Table 3.11-10>

<Table 3.11-11>

<Table 3.11-12>

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TABLE 3.11-13

COMPARISON OF PRIMARY COOLANT AND SUPPRESSION
POOL WATER QUALITY LIMITS

Parameter	Reactor Water Limits Shutdown Condition	Refueling and Storage Pool Water Quality	Suppression Pool Water Quality Maximum Limit
Conductivity	≤10 µmho/cm at 25°C 2 wks/12 mo. period	≤3 µmho/cm at 25°C	≤10 µmho/cm at 25°C
Chlorides as Cl	≤0.5 ppm Time above 0.2 ppm 2 wks/12 mo. period	≤0.5 ppm	≤0.5 ppm
pH	≥4.0, ≤1 0 Time <5.6, >8.6, 24 continuous hours	5.3 to 7.5 at 25°C	5.3 to 8.6 at 25°C
Total Suspended Solids		<1 ppm	≤5 ppm
Heavy Elements (Fe, Cu, Hg, Ni)		≤0.1 ppm	

<APPENDIX 3A>

HYDRODYNAMIC LOADS

FOR PNPP

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1.0   DESCRIPTION OF HYDRODYNAMIC LOADING FUNCTIONS

The complete description of hydrodynamic loading on the reactor building
structures due to safety/relief valve actuation and hypothetical
loss-of-coolant accidents is outlined in GESSAR II, Appendix 3B.
Deviations for PNPP are described in the attached <Appendix 3B>.

1.1   GENERAL LOADING DESCRIPTION FOR SAFETY/RELIEF VALVE ACTUATION
           ANALYSIS

The basis of all safety/relief valve loading is the idealized pressure
time history and pressure distribution described in <Appendix 3B> for
the actuation of a single safety/relief valve. All input time histories
for safety/relief valve reactor building analysis use a time increment
(Delta T) of 0.0025 second.

1.1.1   MONTE CARLO METHOD (RANDOM)

The complete description of the Monte Carlo method for analysis of
multiple safety/relief valve actuations is contained in Attachment N of
<Appendix 3B>. The complete set of critical load cases was run for the
19 valve, ADS and one-valve-second actuation.

1.1.2   IN-PHASE ANALYSIS

The input for in-phase analysis for the 19 valve case is generated by
superimposing the single valve pressure field of the 19 valves at their
respective locations and combined, as described in <Appendix 3B>. The
resultant pressure field is multiplied by the idealized time history and
factored by the maximum pressure shown in <Appendix 3B>.

The in-phase analysis was performed for the 19 valve case only, since
the vertical response is greater than the one, two or ADS valve
actuations. The greatest horizontal response is from the

one-valve-subsequent actuation case, but there is no difference between in-phase and Monte Carlo one-valve since the variable used in Monte Carlo do not affect a single valve actuation. Therefore, Monte Carlo one-valve analyses' results were used for the in-phase cases.

<Appendix 3B> states that a variation of frequencies from 5 to 12 Hz is necessary to completely analyze the in-phase valve actuations. Integer frequencies from 5 to 12 Hz were run as well as three natural frequencies of the containment vessel.

1.2   LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSIS

Based on the hypothetical LOCA described in <Appendix 3B>, there are four discrete cases:

a.   Main vent clearing

b.   Pool swell

c.   Condensation oscillation

d.   Chugging

1.2.1   MAIN VENT CLEARING AND POOL SWELL

The complete description of main vent clearing and pool swell phenomena is contained in <Appendix 3B>. The analysis included main vent clearing, pool swell impact, drag and fallback loads, and containment pressurization.

1.2.1.1   Platform Reactions

A separate platform dynamic analysis due to pool swell direct impact force was made using the DYNAL program and using the loading described

in <Appendix 3B>. The time step was 0.003 second for the range of 0.0 to 6.0 seconds. At 6.0 seconds most of the impact and fallback loads on the platforms have dissipated.

The reactions of the platforms at the drywell wall interface were then applied to the drywell wall of the reactor building model, in conjunction with the direct pressure loading as input for the main vent clearing pool swell analysis.

1.2.1.2   Distribution of Loads

The loading generated for the model starts with the pressure time history described in <Appendix 3B>. The loading was interpolated for a time step of 0.003 second for the range of 0.0 to 6.0 seconds.

1.2.2   CONDENSATION OSCILLATION

The phenomena is described in <Appendix 3B>. The pressure loading for condensation oscillation is shown in Attachment F of <Appendix 3B>. The analysis was performed with a time step of 0.010 second for the range of 0.0 to 27.0 seconds.

1.2.3   CHUGGING

The chugging phenomena is described in <Appendix 3B>. There are two different chugging areas which were analyzed: the weir annulus area and the suppression pool area. There were two analyses performed for suppression pool chugging: 10 Hz and 12 Hz suppression pool chug.

2.0   ANALYTICAL METHOD

2.1   MODEL DESCRIPTION

The mathematical model used for the analysis of the reactor building for
the loads caused by suppression pool dynamics is shown in <Figure 3A-1>.
For the containment design, the model including fluid elements is as
shown in <Figure 3A-2>.

2.1.1   HORIZONTAL AND VERTICAL REACTOR PRESSURE VESSEL MODELS

The horizontal and vertical General Electric reactor pressure
vessel (RPV) models are presented in <Figure 3A-3> and <Figure 3A-4>.
The geometries of the two models were incorporated into separate reactor
building models.

2.1.2   SOIL PROPERTIES AND ELEMENT ASPECT RATIO

The soil properties for the model are obtained from <Section 2.5>. The
maximum quadrilateral element aspect ratio for the soil elements in the
production model of the reactor building is approximately 1:6, which is
within the guidelines presented in (Reference 1).

2.1.3   BOUNDARY DISTANCE

The boundary distance is governed by the frequency content of the
dynamic loads and the wave velocity (shear wave velocity and compression
wave velocity) of the foundation material. The following properties
were used for the selection of the boundary distance and in the
analysis:

Shear Wave Velocity, V_s = 4,900 ft/sec

Unit Weight, γ = 152 lb/ft^3

Shear Modulus, G_s = 7.88 x 10^5 lb/in.^{2}

Young's Modulus, E = 21.38 x 10^5 lb/in.^{2}

Poisson's Ratio, μ = 0.36

The compression wave velocity (V_c) is calculated to be 8,077 ft/sec. A curve is presented in <Figure 3A-5> which shows the relationship of frequency and wave length for the Chagrin shale of the plant site. Based upon this information, a distance is chosen to allow for a sufficient number of waves to occur within the soil, to allow the energy of the loads to dissipate before it is reflected back towards the reactor building.

The dimensions chosen for the boundary distances were:

a.   Radial Direction = 2,840 feet

b.   Vertical Direction = 2,206.25 feet

In order to determine if the boundary distance selected produces acceptable reactor building accelerations, a study was made to obtain structure accelerations for the boundary distances of the production model and for boundary distances less than those of the production model.

Since the aspect ratio for the soil elements was reduced from 1:6 to 1:3 and by not increasing the number of soil elements, the boundary distance was reduced to:

a.   Radial Direction = 1,681 feet

b.   Vertical Direction = 1,783 feet

The boundary distances selected for the production model are 6.96 and 5.39 times the soil shear wave length in the radial and vertical directions, respectively, while the boundary distances selected for the study model are 4.12 and 4.37 times the soil shear wave length in the radial and vertical directions, respectively.

The results indicate that the mat responses are 15 percent higher with the shorter boundary distances than with the longer boundary distances. Based upon these results, the conclusion can be reached that selected boundary distances are required in order to provide production results which do not overestimate responses.

2.1.4   FLUID MODELING

Three different modeling techniques were used to represent the water:

a.   Fluid Element Model

b.   Sloshing Mass Model

c.   Concentrated Mass Model

The ANSYS computer program was used for all three analyses. Springs were used to represent the soil in the finite element models.

2.1.4.1   Fluid Element Model

The fluid element model has the water modeled by finite elements. Only the containment vessel, drywell and mat are represented in the model.

2.1.4.2   Sloshing Mass Model

The sloshing mass model has the water modeled as two masses. The one mass is a stationary mass which is distributed by tributary area to the

nodes in the lower portion of the suppression pool. The second mass is a moving mass which is connected to the drywell and containment vessel by two springs, at a specific elevation above the bottom of the suppression pool. The moving water mass simulates the sloshing effect of the pool water. This method of modeling the water is described in (Reference 2).

2.1.4.3   Concentrated Mass Model

The concentrated mass model, shown in <Figure 3A-1>, has the water mass distributed by tributary area to the node points along the containment vessel, mat and drywell wall boundary of the suppression pool.

2.1.4.4   Conclusion

A comparison of accelerations of the structures resulting from a representative SRV load showed that when this concentrated mass methodology was used it produced higher radial and vertical accelerations than the other methodologies (Reference 3). Therefore, the concentrated mass methodology was used to produce conservative design input for reactor building analysis and design.

Based on the results of these analyses, the concentrated mass method adequately represents the fluid and is used in the reactor building analysis. For the containment vessel analysis, fluid elements, using ASHSD2, were used for a more detailed investigation <Figure 3A-2>.

2.1.5   DAMPING VALUES

The method by which damping is considered in the direct integration of the equations of motion, within the ASHSD2 computer program, is by the

alpha beta method. The damping values used for dynamic analysis of the reactor building are the following:

alpha = 1.955

beta = 0.0003537

2.2   ANALYSIS

2.2.1   INTEGRATION TIME STEP, ΔT

The integration method used in the ASHSD2 computer program is accurate, if the integration time step is small compared with the natural period of finite element system associated with the highest frequency expected to participate significantly. If the time step is reduced in order to accurately integrate the stiff components of a particular problem, then the time step will be much too small for the low frequency responses, resulting in excessive computer time for the calculations.

A parametric study, using the model shown in <Figure 3A-1>, was used to obtain results which indicate what integration time step should be used in the production runs for the safety/relief valve analysis. The structure accelerations from two computer runs, Δt = 0.0025 second and Δt = 0.0010 second, demonstrated that Δt = 0.0025 second is adequate.

2.2.2   FOURIER COEFFICIENTS

A parametric study was undertaken to determine the number of Fourier terms of the input load time histories required to obtain accurate results from the production computer runs for multiple safety/relief valve analysis. The model shown in <Figure 3A-1> was used for the study with a random 19 valve case.

The study consisted of four computer runs in which the following numbers of Fourier coefficients are used for the input load time history:

a.   Run No. 1: 3 cosine terms and 2 sine terms

b.   Run No. 2: 5 cosine terms and 4 sine terms

c.   Run No. 3: 7 cosine terms and 6 sine terms

d.   Run No. 4: 9 cosine terms and 8 sine terms

The study is made by comparing the reactor building radial, vertical and tangential accelerations (ZPA), and the radial, vertical and tangential response spectra at the containment vessel support, the base of the pedestal and the drywell side of the refueling bellows for the variation caused by the number of Fourier coefficients used for each computer analysis.

The summary indicated that 5 cosine terms and 4 sine terms provide results that have converged for all substructures of the reactor building.

2.2.3   SPECTRA

The structural acceleration responses of SRV and LOCA analyses are used to calculate floor response spectra (FRS) to be used for equipment analysis and design.

2.2.3.1   Circumferential Variation

The results of the ASHSD2 analysis include acceleration time histories for each node at each of eight separate angles. For each node, a response spectra was generated for each of eight angles around the

circumference. The FRS were then created by enveloping the eight individual spectra. The frequency range is 0.5 Hz to 100 Hz using the frequency values specified by <Regulatory Guide 1.122>.

2.2.3.2   Enveloping

The FRS used in design are the envelopes of all controlling safety/relief valve cases.

2.2.3.3   Broadening

The FRS and envelopes were broadened in accordance with <Regulatory Guide 1.122>.

3.0   CONTAINMENT FIX

Based on the large accelerations of the steel containment vessel due to SRV loads, a fix has been designed in order to qualify containment-mounted equipment. The fix required that concrete be poured to a height of 23 feet 6 inches above the foundation mat filling the annulus between the containment vessel and the shield building. This is discussed in <Section 3.8.1>.

4.0   REFERENCES FOR <APPENDIX 3A>

1.   Ghosh, S. and Wilson, E., with revisions by Lin, C. J.,
     1975 Revisions, Dynamic Stress Analysis of Axisymmetric Structures
     Under Arbitrary Loading, p. 98.

2. George W. Housner, February 1973, <u>The Dynamic Behavior of Water Tanks</u>, <u>Bulletin of the Seismological Society of America</u>, V. 53, N. 2, pp. 381-387.

3. Dr. Ecozomos (Brookhaven National Laboratories), September 24-25, 1981, ACRS Transcript-Subcommittee Meeting on Hydrodynamic Loads, San Francisco, California.

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<APPENDIX 3B>

CONTAINMENT LOADS

TABLE OF CONTENTS

LIST OF TABLES

PREFACE

This section has been revised to delete the GESSAR Appendix 3B which had been reproduced as <Appendix 3B> of the PNPP USAR. The following section follows the format of GESSAR II - Rev. 2, Appendix 3B, and provides a step-by-step Perry specific comparison. GESSAR II, as modified by Draft Acceptance Criteria (Reference 1): Appendix C to the Draft Technical Evaluation Report on Mark III LOCA-Related Hydrodynamic Load Definition provided by memo dated 10/8/82 from Mr. Themis P. Speis (NRC) to Mr. Hank Pfefferlen (GE)) is the basis for the PNPP design.

APPENDIX 3B

CONTAINMENT LOADS

3B.1   INTRODUCTION

No deviations.

3B.1.1   CONFIRMATORY TESTING

No deviations.

3B.1.2   DEFINITION OF LOCA

No deviations.

3B.1.3   DESIGN MARGINS

No deviations.

3B.2   REVIEW OF PHENOMENA

No deviations.

3B.2.1   DESIGN BASIS ACCIDENT (DBA)

GESSAR II Figures 3B-2 through 3B-6 are not applicable to PNPP. See
USAR <Figure 3B-1>, <Figure 3B-2>, <Figure 3B-3>, <Figure 3B-4>, and
<Figure 3B-5>.

3B.2.2   INTERMEDIATE BREAK ACCIDENT (IBA)

No deviations.

3B.2.3   SMALL BREAK ACCIDENT (SBA)

No deviations.

3B.2.4   SAFETY/RELIEF VALVE ACTUATION

No deviations.

3B.2.5   OTHER CONSIDERATIONS

No deviations.

3B.3   DYNAMIC LOAD TABLE

No deviations.

3B.4   DRYWELL STRUCTURE

No deviations.

3B.4.1   DRYWELL LOADS DURING A LARGE BREAK ACCIDENT

No deviations.

3B.4.1.1   Sonic Wave

No deviations.

3B.4.1.2   Drywell Pressure

GESSAR II Figure 3B-10 is not applicable to PNPP. USAR <Figure 6.2.11>
shows the PNPP drywell short term pressure response to a main steam line
break (DBA).

3B.4.1.3   Hydrostatic Pressure

No deviations.

3B.4.1.4   Loads on the Drywell Wall During Pool Swell

No deviations.

3B.4.1.5   Condensation Oscillation Loads

GESSAR II Figure 3B-17 is not applicable to PNPP. See USAR
<Figure 3B-6> for the distribution of condensation oscillation loads on
the drywell.

3B.4.1.6   Fallback Loads

No deviations.

3B.4.1.7   Negative Load During ECCS Flooding

No deviations.

3B.4.1.8   Chugging

No deviations.

3B.4.1.9   Loads Due to Chugging

No deviations.

3B.4.1.9.1   Chugging Loads Applied to Top Vent

No deviations.

3B.4.1.9.2   Pool Boundary Chugging Loads

GESSAR II Figures 3B-28 through 3B-31, 3B-34 and 3B-35 are not
applicable to PNPP. See USAR <Figure 3B-7>, <Figure 3B-8>,
<Figure 3B-9>, <Figure 3B-10>, <Figure 3B-11>, and <Figure 3B-12>.

3B.4.2   DRYWELL LOADS DURING INTERMEDIATE BREAK ACCIDENT

No deviations.

3B.4.3   DRYWELL DURING SMALL BREAK ACCIDENT

No deviations.

3B.4.3.1   Drywell Temperature

No deviations.

3B.4.3.2   Drywell Pressure

No deviations.

3B.4.3.3   Chugging

No deviations.

3B.4.4   SAFETY/RELIEF VALVE ACTUATION

No deviations.

3B.4.5   DRYWELL ENVIRONMENTAL ENVELOPE

No deviations.

3B.4.6   TOP VENT TEMPERATURE (CYCLING) PROFILE DURING CHUGGING

No deviations.

3B.4.7   DRYWELL MULTICELL EFFECTS

No deviations.

3B.5   WEIR WALL

No deviations.

3B.5.1   WEIR WALL LOADS DURING DESIGN BASIS ACCIDENT

No deviations.

3B.5.1.1   Sonic Wave

No deviations.

3B.5.1.2   Outward Load During Vent Clearing

No deviations.

3B.5.1.3   Outward Load Due to Vent Flow

No deviations.

3B.5.1.4   Chugging Loads

No deviations.

3B.5.1.5   Inward Load Due to Negative Drywell Pressure

Deviations from GESSAR II as required by the NRC Draft Acceptance
Criteria.

3B.5.1.6   Suppression Pool Fallback Loads

No deviations.

3B.5.1.7   Hydrostatic Pressure

No deviations.

3B.5.1.8   Safety/Relief Valve Actuation

No deviations.

3B.5.1.9   Condensation

No deviations.

3B.5.2   WEIR WALL LOADS DURING AN INTERMEDIATE BREAK ACCIDENT

No deviations.

3B.5.3   WEIR WALL LOADS DURING A SMALL BREAK ACCIDENT

No deviations.

3B.5.4   WEIR WALL ENVIRONMENTAL ENVELOPE

No deviations.

3B.5.5   WEIR ANNULUS MULTICELL EFFECTS

No deviations.

3B.6   CONTAINMENT

No deviations.

3B.6.1   CONTAINMENT LOADS DURING A LARGE STEAMLINE BREAK (DBA)

GESSAR II Figures 3B-2 through 3B-6 are not applicable to PNPP. See
USAR <Figure 3B-1>, <Figure 3B-2>, <Figure 3B-3>, <Figure 3B-4>, and
<Figure 3B-5>.

3B.6.1.1   Compressive Wave Loading

No deviations.

3B.6.1.2   Water Jet Loads

No deviations.

3B.6.1.3   Initial Bubble Pressure

No deviations.

3B.6.1.4   Hydrostatic Pressure

No deviations.

3B.6.1.5   Local Containment Loads Resulting from the Structures at
                or Near the Pool Surface

Deviation from GESSAR II as required by the NRC Draft Acceptance
Criteria.

3B.6.1.6   Containment Load Due to Pool Swell at the HCU Floor
                (Wetwell Pressurization)

The PNPP HCU floor is approximately 27 feet above the suppression pool
surface and has been designed for 10 psid across the total area of the
platform (structural steel plus grating). This was reduced from the
GESSAR II specification because the PNPP HCU floor is seven feet higher
than the GESSAR II standard. In addition, a plant unique analysis
showed a peak calculated pressure differential equal to approximately
5.4 psid based upon a design open area ratio of 30 percent of the total
HCU floor area, (Reference 2), (Reference 3) and (Reference 4).
GESSAR II Figure 3B-58 is not applicable to PNPP.

3B.6.1.7   Fallback Loads

No deviations.

3B.6.1.8   Post Pool-Swell Waves

No deviations.

3B.6.1.9   Condensation Oscillation Loads

GESSAR II Figure 3B-17 is not applicable to PNPP. See USAR
<Figure 3B-6> for condensation oscillation loads on containment.

3B.6.1.10   Chugging

No deviations.

3B.6.1.11   Long Term Transient

No deviations.

3B.6.1.12   Containment Environmental Envelope

No deviations.

3B.6.2   CONTAINMENT LOADS DURING AN INTERMEDIATE BREAK ACCIDENT

No deviations.

3B.6.3   CONTAINMENT LOADS DURING A SMALL BREAK ACCIDENT

No deviations.

3B.6.4   SAFETY/RELIEF VALVE LOADS

No deviations.

3B.6.5   SUPPRESSION POOL THERMAL STRATIFICATION

No deviations.

3B.6.6   CONTAINMENT WALL MULTICELL EFFECTS

No deviations.

3B.7   SUPPRESSION POOL BASEMAT LOADS

No deviations.

3B.8   LOADS ON STRUCTURES IN THE SUPPRESSION POOL

No deviations.

3B.8.1   DESIGN BASIS ACCIDENT

No deviations.

3B.8.1.1   Vent Clearing Jet Load

The ECCS strainer intrudes into the water jet zone of exclusion and is designed for vent clearing loading.

3B.8.1.2   Drywell Bubble Pressure and Drag Loads Due to Pool Swell

The PNPP design basis for drywell bubble pressure and drag loads conservatively uses the LOCA bubble pressure. A comparison of the PNPP load methodology and the GESSAR II methodology is given in <Section 3BL.2.3> of the USAR.

3B.8.1.3   Fallback Loads

No deviations.

3B.8.1.4   Condensation Loads

LOCA condensation-oscillation drag loads are bounded by the PNPP LOCA bubble pressure drag load methodology (Reference 5), with the exception of the ECCS strainer. The ECCS strainer is designed for condensation-oscillation loads, determined using the GESSAR II methodology.

3B.8.1.5   Chugging

Chugging drag loads are bounded by the PNPP LOCA bubble pressure drag load methodology (Reference 5) and (Reference 6), with the exception of the ECCS strainer. The ECCS strainer is designed for chugging loads, determined using the acoustic wave methodology (Reference 7) (Reference 8) (Reference 9) (Reference 10) (Reference 11).

3B.8.1.6   Compressive Wave Loading

No deviations.

3B.8.1.7   Safety/Relief Valve Actuation

The PNPP design basis for safety/relief valve quencher air bubble drag loads is conservatively based on the maximum quencher bubble pressure, except the ECCS strainer which utilizes the GESSAR II methodology for determination of quencher bubble loads. A comparison of the PNPP maximum bubble pressure load methodology and the GESSAR II load methodology is given in <Section 3BL.3.2> of the USAR.

3B.9   LOADS ON STRUCTURES AT THE POOL SURFACE

As required by the NRC Draft Acceptance Criteria, the PNPP analysis used a velocity ranging from zero feet per second at the pool surface to a maximum of 50 feet per second as a function of height instead of the

constant 40 feet per second velocity specified in GESSAR II Table 3B-2, to calculate pool swell drag loads.

3B.10 LOADS ON STRUCTURES BETWEEN THE POOL SURFACE AND THE HCU FLOORS

No deviations.

3B.10.1 IMPACT LOADS

Impact loads are calculated in accordance with GESSAR II as modified by the requirements of the NRC Draft Acceptance Criteria.

The design basis for bulk pool swell impact loads on small structures less than 4 feet long and/or 6 feet above the pool has been evaluated using an alternative method (Reference 6).

3B.10.2 DRAG LOADS

Drag loads are calculated in accordance with GESSAR II as modified by the requirements of the NRC Draft Acceptance Criteria.

3B.10.3 FALLBACK LOADS

No deviations.

3B.11 LOADS ON EXPANSIVE STRUCTURES AT THE HCU FLOOR ELEVATION

Other than the HCU floor, <Section 3B.6.1.6> the only expansive structure in the pool swell region is the steam tunnel, which is designed in accordance with the NRC Draft Acceptance Criteria (Reference 1) (Reference 3) and (Reference 4).

3B.12   LOADS ON SMALL STRUCTURES AT AND ABOVE THE HCU FLOOR

ELEVATION

Deviation from GESSAR II as required by the NRC Draft Acceptance
Criteria.

3B.13   REFERENCES

1.   Draft NRC Acceptance Criteria for LOCA Related Mark III Containment
      Pool Dynamic Loads, Appendix C of Attachment to NRC letter from
      T. P. Speis, NRC, to H. Pfefferlen, GE, dated October 8, 1982.

2.   CEI letter, PY-CEI/NRR-0010L from M. R. Edelman, CEI, to
      B. J. Youngblood, NRC, dated January 31, 1983.

3.   CEI letter, PY-CEI/NRR-0055L from M. R. Edelman, CEI, to
      B. J. Youngblood, NRC, dated June 20, 1983.

4.   CEI letter, PY-CEI/NRR-0123L from M. R. Edelman, CEI, to
      B. J. Youngblood, NRC, dated July 11, 1984.

5.   CEI letter, PY-CEI/NRR-0235L from M. R. Edelman, CEI, to
      B. J. Youngblood, NRC, dated May 16, 1985.

6.   CEI letter, PY-CEI/NRR-0336L from M. R. Edelman, CEI, to
      B. J. Youngblood, NRC, dated September 13, 1985.

7.   Mark II Containment Program Lead Plant Program Load Evaluation and
      Acceptance Criteria, Generic Technical Activity A-8, <NUREG-0487>,
      Supplement No. 2, U. S. Nuclear Regulatory Commission, Office of
      Nuclear Reactor Regulation, Washington, D.C., February 1981.

8.   Mark II Improved Chugging Methodology, NEDE-24822-P Class III
      General Electric Company, May 1980; this document was prepared for

the Mark II Utility Owners' Group by Bechtel Power Corporation under contract with General Electric Company, (Proprietary).

9.   Mark II Containment Program Load Evaluation and Acceptance Criteria, Generic Technical Activity A-8, <NUREG-0808>, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C., August 1981.

10. An Approach to Chugging, Assessment of RHR Steam Discharge Condensation Oscillation in Mark III Containments, Prepared for the Mark III Containment Issues Owners' Group, Job 16031, Bechtel Power Corporation, San Francisco Power Division, Nuclear Engineering Staff, March 1984, (Proprietary).

11. Safety Evaluation Report related to the operation of Perry Nuclear Power Plant, <NUREG-0887>, Supplement No. 8, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C., January 1986.

ATTACHMENT A TO APPENDIX 3B

SAFETY/RELIEF VALVE LOADS (QUENCHER)

3BA.1   INTRODUCTION

No deviations.

3BA.2   SUMMARY AND CONCLUSIONS

GESSAR II Table 3BA-1 is not applicable to PNPP. USAR <Table 3BA-1>
provides the SRVDL information for PNPP. GESSAR II Table 3BA-2 is not
applicable to PNPP. USAR <Table 3BA-2> (located in this attachment)
provides the results of the analysis to determine the maximum quencher
bubble pressures for PNPP. The SRVDL peak pressure is limited to
570 psid for PNPP.

3BA.3   DESCRIPTION OF THE PHENOMENA

The SRVDL peak pressure is limited to 570 psid for PNPP.

3BA.4   ARRANGEMENT

3BA.4.1   DISTRIBUTION IN POOL (QUENCHER ARRANGEMENT)

GESSAR II Figures 3BA-2 through 3BA-4 are not applicable to PNPP. PNPP
USAR <Figure 3BA-1>, <Figure 3BA-2>, and <Figure 3BA-3> show the
elevation and plan views of the PNPP quencher arrangement.

3BA.4.2   SRVDL ROUTING

GESSAR II Figure 3BA-7 is not applicable to PNPP. USAR <Figure 3BA-4>
shows the SRVDL routing for PNPP.

3BA.4.2.1   Line Lengths and Volume

GESSAR II Table 3BA-1 and GESSAR II Figure 3BA-7 are not applicable to
PNPP. USAR <Table 3BA-1> shows PNPP SRVDL line lengths and volumes
based on the PNPP SRVDL layout shown in USAR <Figure 3BA-4>.

The SRVDL from the 45° elbow just above the pool to the quencher is a
10 inch, Schedule 40, stainless steel pipe.

3BA.4.2.2   Drywell Penetration Sleeve

The SRVDL drywell penetration sleeve for PNPP is shown in USAR
<Figure 3BA-1>.

3BA.4.2.3   SRVDL Vacuum Breaker

No deviations.

3BA.5   QUENCHER LOAD ON POOL BOUNDARY

3BA.5.1   PRESSURES ON DRYWELL, BASEMAT AND CONTAINMENT

GESSAR II Table 3BA-2 is not applicable to PNPP. USAR <Table 3BA-2>
identifies the maximum and minimum bubble pressures for PNPP.

3BA.5.1.1   Single SRV Loads

GESSAR II Table 3BA-6 and Figure 3BA-8 are not applicable to PNPP. USAR
<Table 3BA-3> and <Figure 3BA-5> show PNPP design values.

3BA.5.1.2   Two Adjacent SRV Loads

GESSAR II Table 3BA-7 and Figure 3BA-11 are not applicable to PNPP.
USAR <Table 3BA-4> and <Figure 3BA-6> show PNPP design values.

3BA.5.1.3   Eight SRV Loads (ADS)

GESSAR II Table 3BA-8 and Figures 3BA-14, 3BA-15 and 3BA-16 are not
applicable to PNPP. USAR <Table 3BA-7>, and <Figure 3BA-7>,
<Figure 3BA-8>, and <Figure 3BA-9> show the normalized dynamic peak
pressure field and radial and circumferential peak values for the eight
ADS SRV's for PNPP.

3BA.5.1.4   All (19) SRV Loads

GESSAR II Table 3BA-9 and Figures 3BA-17, 3BA-18 and 3BA-19 are not
applicable to PNPP. USAR <Table 3BA-8>, <Figure 3BA-10>,
<Figure 3BA-11>, and <Figure 3BA-12> show the normalized dynamic peak
pressure field and radial and circumferential peak values for PNPP.

3BA.5.2   LOAD ON WEIR WALL

No deviations.

3BA.5.3   LOADS ON SUBMERGED STRUCTURES

See USAR <Appendix 3BL> for definition of PNPP load methodology.

3BA.5.4   NORMALIZED PRESSURE TIME HISTORY (THEORETICAL RALEIGH
               BUBBLE)

No deviations.

3BA.5.5   REPRESENTATIVE PRESSURE TIME HISTORY

No deviations.

3BA.5.6   ESTIMATED MARGINS

3BA.5.6.1   Peak Bubble Pressures

No deviations.

3BA.5.6.2   Bubble Pressure Amplitude

No deviations.

3BA.5.6.3   95% Confidence

No deviations.

3BA.5.6.4   Margin

No deviations.

3BA.6   OTHER LOADS ON STRUCTURES IN THE POOL

3BA.6.1   LOCA, POOL SWELL, CONDENSATION OSCILLATION, AND CHUGGING

See USAR <Appendix 3BL> for definition of PNPP load methodology.

3BA.6.2   FORCES ON PIPES DUE TO VENT CLEARING, POOL SWELL AND
            FALLBACK

Upper bound values for submerged structure loads were derived using the
procedures defined in USAR <Appendix 3BL>.

3BA.6.3   THERMAL EXPANSION LOADS

No deviations.

3BA.6.4   SEISMIC LOADS

No deviations.

3BA.6.5   SEISMIC SLOSH LOADS

No deviations.

3BA.7   <u>QUENCHER ANCHOR LOADS</u>

GESSAR II Figures BA-2 through 3BA-6 and 3BA-28 and Tables 3BA-13 and
3BA-14 are not applicable to PNPP. USAR <Figure 3BA-1>, <Figure 3BA-2>,
and <Figure 3BA-3> show the general arrangement of the quenchers in the
pool. The quencher anchor loads are defined in <Table 3BA-3> and
<Table 3BA-4>.

3BA.7.1   QUENCHER ARM LOADS AND QUENCHER LOADING APPLICATION

See USAR <Table 3BA-3> and <Table 3BA-4>.

3BA.7.2   QUENCHER DESIGN INFORMATION

USAR <Figure 3BA-1>, <Figure 3BA-2>, and <Figure 3BA-3> show the
quencher side elevation, top elevation and angular locations in the
suppression pool. GESSAR II, Figures 3BA-2, 3BA-3 and 3BA-4 are not
applicable to PNPP.

3BA.7.2.1   Codes and Standards

No deviations.

3BA.7.2.2   Design Pressures, Temperatures, Loads, Configuration, and
                 Performance

3BA.7.2.2.1   Component Data for Safety/Relief Valve, Discharge
                   Piping and Quencher

The plant specific value for maximum flow at 1,190 psig is
1×10^{6} pounds per hour (approximately 453 metric tons per hour).

3BA.7.2.2.2   SRVDL Geometry

No deviations.

3BA.7.2.2.3   Quencher Design Criteria

a.   Forces   See USAR <Table 3BA-3> and <Table 3BA-4>

b.   Fatigue   16,200 cycles

c.   Cycles of operation   1,800

3BA.7.2.2.4   Quencher Configuration and Location

The PNPP design rating is 570 psig and the minimum radial distance from the centerline of the quencher to the flange of the ECCS strainers is 11 feet 6 inches.

3BA.8   SRV VALVE LOAD COMBINATIONS

GESSAR II Figure 3BA-4 is not applicable to PNPP. See USAR <Figure 3BA-3> for the SRVDL discharge locations for PNPP.

3BA.8.1   SYMMETRIC AND ASYMMETRIC LOAD CASES

No deviations.

3BA.8.2   SSE AND OBE CONSIDERATIONS

No deviations.

3BA.8.3   LOCA CONSIDERATIONS

No deviations.

3BA.8.3.1   DBA With MS Line Break

No deviations.

3BA.8.3.2   DBA With Recirculation Line Break

No deviations.

3BA.8.3.3   Other SRV Conditions

No deviations.

3BA.8.3.3.1   Water Clearing Pressure Spike for SRV First Actuation,
                Normal Operating Conditions

No deviations.

3BA.8.3.3.2   SRV First Actuation With a Pressurized Containment

No deviations.

3BA.8.3.3.3   Water Clearing Pressure Spike for SRV, Second Actuation
                Normal Operating Conditions

No deviations.

3BA.8.3.3.4   Second Actuation of one SRV With a Pressurized
                Containment

No deviations.

3BA.8.3.3.5   First Actuation of One SRV, Leaking Valve Condition

No deviations.

3BA.8.3.3.6   SRV Steam Condensation

No deviations.

3BA.8.4   DESIGN LOAD SUMMATION

No deviations.

3BA.9   FATIGUE CYCLES

No deviations.

3BA.10   CALCULATIONAL PROCEDURES FOR MARK III CONTAINMENT

No deviations.

3BA.11   PARAMETRIC STUDIES

No deviations.

3BA.12   BASIS AND JUSTIFICATION FOR DEVELOPED QUENCHER LOADS

No deviations.

3BA.13   REFERENCES

No deviations.

PNPP SRV DISCHARGE LINE

| | | Air Leg Length (feet) | | | | Max. F $\left(\dfrac{L}{D}\right)$ | |
SRV	Total Length[3]	10" S/40S[4]	12" S/40S[4]	1.4" S/40[4]	Volume (ft³)	10"[1] S/40S	12"[2] S/40S
F041G	60.781	15.699	28.377	–	42.826	1.929	3.968
F041E	52.43	17.00	–	16.37	38.713	1.27	2.62
F051C	69.918	48.293	14.872	–	44.178	1.795	3.691
F051A	62.423	11.372	–	16.074	37.45	1.54	3.166
F051G	60.682	16.852	26.588	–	42.789	2.101	4.321
F047C	60.405	21.255	28.583	–	41.991	1.271	2.614
F041C	75.002	43.824	20.427	–	46.932	2.312	4.755
F041B	74.544	45.037	19.74	–	46.519	2.09	4.296
F041F	46.052	15.21	–	22.441	43.465	1.211	2.492
F051B	54.688	24.18	19.541	–	37.267	1.389	2.857
F047B	79.42	54.654	15.00	–	48.062	2.227	4.581
F047D	45.301	14.424	–	13.902	36.327	1.473	3.029
F041D	44.543	13.006	–	17.591	37.940	.97	1.994
F051D	63.876	35.607	16.87	–	43.278	1.86	3.826
F041K	49.101	10.219	26.1	–	39.501	1.284	2.643
F047H	60.441	24.259	21.04	–	41.447	1.442	2.966
F047F	60.135	21.414	23.979	–	41.864	1.337	2.751
F047G	57.844	18.303	24.399	–	41.708	1.417	2.915
F041A	45.219	13.23	–	17.652	38.121	1.101	2.264

NOTES:

[1] Normalized to 10 inch Schedule 40S pipe.
[2] Normalized to 12 inch Schedule 40S pipe.
[3] Total length stated accounts for straight pipe and arc-lengths of elbows.
[4] Stated lengths of pipe in Columns 3, 4 and 5 account for straight pipe only.

INDUCED LOADS ON POOL BOUNDARIES

Case Description	Design Value - Bottom Maximum Pressure (psid)		Containment Normalized Factor @ Point 10[b]	Containment Peak Pressure @Point 10 (psid)[b]		Affected Quenchers/ Design Cycles
	$P_B(+)$	$P_B(-)$		P+	P-	
Single Valve First Actuation, at T_{srv}=95°F Pool Temperature	10.63	-6.5	0.711	7.7	-4.6	V10/33
Single Valve Subsequent Actuation	18.3	-7.8	0.711	13.0	-5.5	V10/1,560
Two Adjacent Valves First Actuation[g] at T_{srv}=95°F Pool Temperature	10.8	-6.5	0.856	9.2	-5.5	V9,V10/1
19 Valves (All Valve Case) First Actuation,[g] at T_{srv}=95°F Pool Temperature	12.1	-6.4	1.0	12.1	-6.4	All Quenchers/ 159
8 ADS Valves First Actuation at T_{srv}=120°F Pool Temperature	11.3	-6.8	0.821	9.3	-5.6	V2,V4,V7,V9, V11,V13,V16, V18/1

TABLE 3BA-3

QUENCHER LOADS

NOTE:

Load F_e may act in any direction. Moment M_e may act in any vertical plane.

TABLE 3BA-4

QUENCHER ANCHOR LOADS

GAI Load Case	G.E. Load Case		Combination
Deadweight	Deadweight		Addition
-	Arm Loads Case A		
-	Arm Loads Case B	Screen	Use Screened G.E. Values
-	Arm Loads Case C		
SRVI	Structural Response		SRSS
OBEI	OBE		SRSS
SSEI	SSE		SRSS
SRV Blowdown	Transient Wave		SRSS
-	Water Clearing		G.E. Values
-	Adjacent Quencher Water Jet Impingement		G.E. Values
-	LOCA Water Jet		G.E. Values
Pool Swell Drag	LOCA Air Bubble		SRSS
Pool Swell Drag	Pool Fallback		SRSS
PSI	Pool Swell		SRSS
-	Condensation Oscillation		G.E. Values
CHUGI	Chugging		SRSS
API	-		GAI Values
Thermal Normal Ambient	-		GAI Values
Thermal Normal SRV	-		GAI Values

TABLE 3BA-4 (Continued)

GAI Load Case	G.E. Load Case	Combination
Thermal Stuck Open Valve	–	GAI Values
Thermal Post LOCA	–	GAI Values
SRV Drag (Active)	–	GAI Values
SRV Drag (Inactive)	–	GAI Values

QUENCHER BASEMAT LOADINGS – GE & GAI COMBINED

Load Type	Forces (kips)			Moments (inch-kips)			Load ID
	Fx	Fy	Fz	Mx	My	Mz	
DEAD WEIGHT	0.39	-7.27	0	-41.0	-3.0	0	DW
ARM LOADS							
Max Case A	±0.97	0	±5.49	±296.5	±1,244	±52.37	ALA
Max Case B	±25.3	0	0	0	0	±817	ALB
Max Case C	0	±15.17	0	0	0	0	ALC
STRUCT. RESPONSE MAX	±3.16	±4.16	0	0	±39.95	±164.88	SR
SEISMIC							
OBE Max	±5.81	±7.82	0	0	±76.50	±286.24	OBE
SSE Max	±8.30	±11.23	0	0	±108.9	±410.12	SSE
TRANSIENT WAVE	+0.67	+19.21	0	0	±11.0	±72.0	TW
	-0.29	-28.32					
WATER CLEARING	0	+2	0	0	0	0	WC
		-150					
ADJACENT QUENCH-ER JET IMPINGE-MENT	±36.1	0	0	0	±619.8	±1,770	AQJI
LOCA EVENT							
Water Jet	±36.5	0	0	0	0	±1,375	LWJ
Air Bubble	±26.87	±14.2	0	0	±74	±1,498	LAB
Pool Swell	+0.25	±37.92	0	0	±4	±26	LPS
Pool Fallback	±5.9	-37.82	0	0	±74	±608	LPF
Condensation Oscillation	±19.48	±4.5	0	0	0	±999	LCO
Chugging	±6.32	±8.42	0	0	±4	419.24	LCG
Annulus Press.	±0.62	±3	0	0	±20	±61	LAP
SRV DRAG ACTIVE	±2.96	±7.02	0	0	±37	±303	SRVD
THERMAL NORMAL AMB.	10.73	-4.31	0	0	-4	-1,239	TH3
THERMAL NORMAL SRV	22.57	-16.86	0	0	-24	-2,550	TH1
THERMAL STUCK OPEN V	25.67	-17.79	0	0	-24	-2,909	TH2
THERMAL POST LOCA	19.33	-7.64	0	0	-7	-2,233	TH4
SRV DRAG INACTIVE	±2.0	±4.75	0	0	±25.08	±205.4	SVDI

[1]Supplemental PNPP Calculations have been performed. The actual loads are not presented in the Tables above. All revised loads are within allowable limits.

TABLE 3BA-6

<u>QUENCHER BASEMAT LOADINGS</u>[1][2]

Condition	Fe (kips)	Fv (kips)	Me (inch-kips)	Mv (inch-kips)
Service Level A (P)	36.6	+0 -11.43	1,818.7	624.1
Service Level A (P+S)	47.3	-15.7	3,057.7	628.1
Service Level B (P) Comb. 1 - Transient Wave	6.6	+14.0 -37.0	400.6	84.3
Service Level B (P+S) Comb. 1 - Transient Wave	29.1	+14.0 -53.86	2,950.6	108.3
Service Level B (P) Comb. 1 - Case A Arm Loads	9.3	+3.0 -157.6	532.9	1,250.2
Service Level B (P+S) Comb. 1 - Case A Arm Loads	31.8	+3.0 -174.5	3,082.9	1,274.2
Service Level B (P) Comb. 1 - Case B Arm Loads	26.6	+3.0 -157.6	449.1	92.9
Service Level B (P+S) Comb. 1 - Case B Arm Loads	49.2	+3.0 -174.5	3,499.1	116.9
Service Level B (P) Comb. 1 - Case C Arm Loads	7.3	+11.0 -158.4	430.0	92.9
Service Level B (P+S) Comb. 1 - Case C Arm Loads	29.9	+11.0 -175.2	2,980.0	116.9

<u>NOTE</u>:

[1] Refer to <Table 3BA-3> for orientation.

[2] Supplemental PNPP Calculations have been performed. The actual loads are not presented in the Tables above. All revised loads are within allowable limits.

Condition	Fe (kips)	Fv (kips)	Me (inch-kips)	Mv (inch-kips)
Service Level B (P) Comb. 1 - Bubble Loads Active	7.6	+4.0 -18.6	489.2	96.9
Service Level B (P+S) Comb. 1 - Bubble Loads Active	30.2	+4.0 -35.5	3,039.2	120.9
Service Level B (P) Comb. 2 - Adjacent Quencher	37.1	+1.6 -16.1	1,841.6	628.8
Service Level B (P+S) Comb. 2 - Adjacent Quencher	47.8	+1.6 -20.4	3,080.6	632.8
Service Level B (P) Comb. 2 - Bubble Loads Inactive	7.3	+2.8 -17.3	430.0	92.9
Service Level B (P+S) Comb. 2 - Bubble Loads Inactive	18.0	+2.8 -21.6	1,669.0	96.9

Service Level C (P) Comb. 1
Service Level C (P+S) Comb. 1

Service Level C (P) Comb. 2
Service Level C (P+S) Comb. 2 ⎱ Bounded by Service Level B

Condition	Fe (kips)	Fv (kips)	Me (inch-kips)	Mv (inch-kips)
Service Level D (P) Comb. 1 - Transient Wave	38.0	+37.1 -56.3	1,617.9	143.3
Service Level D (P+S) Comb. 1 - Transient Wave	60.6	+37.1 -73.2	4,167.9	167.3

Condition	Fe (kips)	Fv (kips)	Me (inch-kips)	Mv (inch-kips)
Service Level D (P) Comb. 1 - Case A Arm Loads	38.4	+32.8 -162.5	1,644.8	1,254.8
Service Level D (P+S) Comb. 1 - Case A Arm Loads	61.0	+32.8 -179.4	4,194.8	1,278.8
Service Level D (P) Comb. 1 - Case B Arm Loads	45.7	+32.8 -162.5	1,815.6	142.9
Service Level D (P+S) Comb. 1 - Case B Arm Loads	68.3	+32.8 -179.4	4,365.6	166.9
Service Level D (P) - Comb. 1 - Case C Arm Loads	38.0	+35.6 -163.3	1,616.3	142.9
Service Level D (P+S) - Comb. 1 - Case C Arm Loads	60.6	+35.6 -180.1	4,166.3	166.9
Service Level D (P) - Comb. 2 - Adjacent Quencher	52.5	+32.5 -47.0	2,401.6	637.9
Service Level D (P+S) - Comb. 2 - Adjacent Quencher	71.8	+32.5 -54.7	4,634.6	644.9
Service Level D (P) - Comb. 2 - Bubble Loads Inactive	38.0	+32.8 -47.3	1,616.3	142.9
Service Level D (P+S) - Comb. 2 - Bubble Loads Inactive	57.3	+32.8 -55.0	3,849.3	149.9

TABLE 3BA-7

PERRY NUCLEAR POWER PLANT DYNAMIC PRESSURE FIELD FOR EIGHT SAFETY/RELIEF VALVES

TIME = 0.15 sec (positive pressure psid) ΔP (r)

0.08 sec (negative pressure psid) ΔP (r)

SRV Angle (degrees) Reference Point	4.5	13.5	V-11 22.5	31.5	40.5	49.5	V-13 58.5	67.5	76.5	85.5	94.5	103.5
1	0	0	0	0	0	0	0	0	0	0	0	0
2	.800	.950	1	.950	.800	.909	1	.893	.657	.598	.657	.893
3	.894	1	1	1	.894	1	1	1	.720	.635	.720	1
4	.912	1	1	1	.912	1	1	1	.733	.641	.733	1
5	.846	1	1	1	.846	1	1	.983	.687	.615	.687	.983
6	.914	1	1	1	.885	1	1	1	.725	.666	.725	1
7	.857	1	1	1	.846	1	1	.971	.701	.633	.701	.971
8	.755	.860	.988	.842	.745	.845	.966	.792	.628	.583	.628	.792
9	.667	.714	.764	.720	.657	.698	.740	.674	.564	.533	.564	.660
10	.692	.752	.824	.758	.681	.737	.801	.712	.581	.544	.581	.699
11	.686	.741	.805	.748	.676	.726	.782	.702	.577	.541	.577	.688
12	.653	.688	.728	.695	.642	.672	.704	.650	.554	.525	.554	.635
13	0	0	0	0	0	0	0	0	0	0	0	0

TABLE 3BA-7 (Continued)

SRV Angle (degrees) Reference Point	V-16 112.5	121.5	130.5	139.5	V-18 148.5	157.5	166.5	175.5	184.5	193.5	V-2 202.5	211.45
1	0	0	0	0	0	0	0	0	0	0	0	0
2	1	.909	.800	.909	1	.893	.657	.598	.657	.850	.984	.850
3	1	1	.894	1	1	1	.720	.635	.720	1	1	1
4	1	1	.912	1	1	1	.733	.641	.733	1	1	1
5	1	1	.846	1	1	.983	.687	.615	.687	.944	1	.944
6	1	1	.856	1	1	1	.725	.636	.688	1	1	1
7	1	1	.835	1	1	.971	.681	.604	.688	.956	1	.956
8	.953	.826	.734	.826	.953	.792	.608	.555	.615	.776	.930	.776
9	.725	.691	.646	.691	.725	.646	.543	.524	.551	.644	.700	.629
10	.786	.730	.670	.730	.786	.685	.561	.535	.568	.682	.762	.668
11	.767	.720	.665	.720	.767	.674	.556	.532	.564	.672	.743	.657
12	.688	.665	.631	.665	.688	.620	.533	.516	.541	.618	.663	.603
13	0	0	0	0	0	0	0	0	0	0	0	0

SRV Angle (degrees) Reference Point	220.5	229.5	238.5	247.5	V-4 256.5	265.5	274.5	283.5	292.5	V-7 301.5	310.5	319.5
1	0	0	0	0	0	0	0	0	0	0	0	0
2	.657	.598	.657	.850	.984	.850	.657	.598	.657	.893	1	.909
3	.720	.635	.720	1	1	1	.720	.635	.720	1	1	1
4	.733	.641	.733	1	1	1	.733	.641	.733	1	1	1
5	.687	.615	.687	.944	1	.944	.687	.615	.687	.983	1	1
6	.688	.604	.688	1	1	1	.688	.636	.725	1	1	1
7	.667	.574	.667	.956	1	.956	.688	.604	.681	.971	1	1
8	.594	.526	.594	.776	.930	.776	.615	.555	.608	.792	.966	.845
9	.530	.514	.530	.629	.700	.644	.551	.524	.543	.660	.740	.698
10	.548	.525	.548	.668	.762	.682	.568	.535	.561	.699	.801	.737
11	.543	.523	.543	.657	.743	.672	.564	.532	.556	.688	.782	.726
12	.520	.507	.520	.603	.663	.618	.541	.516	.533	.635	.704	.672
13	0	0	0	0	0	0	0	0	0	0	0	0

SRV Angle (degrees) Reference Point	328.5	337.5	V-9 346.5	355.5
1	0	0	0	0
2	.800	.950	1	.950
3	.894	1	1	1
4	.912	1	1	1
5	.846	1	1	1
6	.885	1	1	1
7	.846	1	1	1
8	.745	.842	.988	.860
9	.657	.720	.764	.714
10	.681	.758	.824	.752
11	.676	.748	.805	.741
12	.642	.695	.728	.688
13	0	0	0	0

TABLE 3BA-8

PERRY NUCLEAR POWER PLANT DYNAMIC PRESSURE FIELD FOR NINETEEN SAFETY/RELIEF VALVES

TIME = 0.15 sec (positive pressure psid) ΔP (r)

0.08 sec (negative pressure psid) ΔP (r)

SRV Angle (degrees) Reference __Point__	V-10 4.5	13.5	V-11 22.5	31.5	V-12 40.5	49.5	V-13 58.5	67.5	V-14 76.5	85.5	V-15 94.5	103.5
1	0	0	0	0	0	0	0	0	0	0	0	0
2	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1
6	1	1	1	1	1	1	1	1	1	1	1	1
7	1	1	1	1	1	1	1	1	1	1	1	1
8	1	1	1	1	1	1	1	1	1	1	1	1
9	1	1	1	1	1	1	1	1	1	1	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	1	1	1	1	1	1	1	1	1	1
12	.989	.992	.989	.992	.989	.992	.989	.992	.989	.992	.989	.992
13	0	0	0	0	0	0	0	0	0	0	0	0

SRV Angle (degrees) Reference Point	V-16 112.5	121.5	V-17 130.5	139.5	V-18 148.5	157.5	V-19 166.5	175.5	V-1 184.5	193.5	V-2 202.5	211.45
1	0	0	0	0	0	0	0	0	0	0	0	0
2	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1
6	1	1	1	1	1	1	1	1	1	1	1	1
7	1	1	1	1	1	1	1	1	1	1	1	1
8	1	1	1	1	1	1	1	1	1	1	1	.983
9	1	1	1	1	1	1	1	1	.990	.971	.931	.860
10	1	1	1	1	1	1	1	1	1	1	.989	.896
11	1	1	1	1	1	1	1	1	1	1	.972	.886
12	.989	.992	.989	.983	.978	.978	.967	.953	.935	.896	.835	.783
13	0	0	0	0	0	0	0	0	0	0	0	0

SRV Angle (degrees) Reference Point	220.5	229.5	V-3 238.5	247.5	V-4 256.5	265.5	V-5 274.5	283.5	V-6 292.5	301.5	V-7 310.5	319.5
1	0	0	0	0	0	0	0	0	0	0	0	0
2	.929	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1
5	.972	1	1	1	1	1	1	1	1	1	1	1
6	1	1	1	1	1	1	1	1	1	1	1	1
7	.992	1	1	1	1	1	1	1	1	1	1	1
8	.888	.983	1	1	1	1	1	1	1	1	1	1
9	.797	.860	.931	.971	.990	1	1	1	1	1	1	1
10	.821	.896	.989	1	1	1	1	1	1	1	1	1
11	.815	.886	.972	1	1	1	1	1	1	1	1	1
12	.783	.835	.896	.935	.953	.967	.971	.978	.978	.983	.989	.992
13	0	0	0	0	0	0	0	0	0	0	0	0

SRV Angle (degrees) Reference Point	V-8 328.5	337.5	V-9 346.5	355.5
1	0	0	0	0
2	1	1	1	1
3	1	1	1	1
4	1	1	1	1
5	1	1	1	1
6	1	1	1	1
7	1	1	1	1
8	1	1	1	1
9	1	1	1	1
10	1	1	1	1
11	1	1	1	1
12	.989	.992	.989	.992
13	0	0	0	0

ATTACHMENT B TO APPENDIX 3B

SCALING ANALYSES AND SMALL STRUCTURE

POOL SWELL DYNAMIC LOADS

No deviations for the entire Attachment.

ATTACHMENT C TO APPENDIX 3B

WEIR ANNULUS BLOCKAGE

No deviations for the entire Attachment.

ATTACHMENT D TO APPENDIX 3B

DRYWELL PRESSURE DISTRIBUTION

No deviations for the entire Attachment.

ATTACHMENT E TO APPENDIX 3B

SUPPRESSION POOL SEISMIC-INDUCED LOADS

No deviations for the entire Attachment.

ATTACHMENT F TO APPENDIX 3B

DIGITIZATION OF FORCING FUNCTION FOR

CONDENSATION OSCILLATION

No deviations for the entire Attachment.

ATTACHMENT G TO APPENDIX 3B

DRYWELL NEGATIVE PRESSURE CALCULATION

No deviations for the entire Attachment.

ATTACHMENT H TO APPENDIX 3B

CONTAINMENT ASYMMETRIC LOADS

No deviations for the entire Attachment.

ATTACHMENT I TO APPENDIX 3B

SUPPRESSION POOL THERMAL STRATIFICATION

USAR <Figure 3BI-1> shows the suppression pool temperature profile for large breaks at Perry.

ATTACHMENT J TO APPENDIX 3B

WEIR WALL LOADS DURING DRYWELL DEPRESSURIZATION

No deviations for the entire Attachment.

ATTACHMENT K TO APPENDIX 3B

WETWELL ASYMMETRIC PRESSURES

No deviations to the entire Attachment.

ATTACHMENT L TO APPENDIX 3B

SUBMERGED STRUCTURE LOADS DUE TO

LOCA AND SRV ACTUATIONS

3BL.1   <u>INTRODUCTION</u>

The subject loads are discussed in Attachment L to Appendix 3B. The
following is a comparison between the PNPP design basis and the GE
design methodology described in Attachment L.

3BL.2   <u>SUBMERGED STRUCTURE LOADS DUE TO LOCA</u>

3BL.2.1   COMPRESSIVE WAVE LOADING

No deviations from Attachment L criteria.

3BL.2.2   LOCA WATER JET LOAD

The ECCS strainer intrudes into the water jet zone of exclusion and is
designed for vent clearing loading.

3BL.2.3   LOCA BUBBLE LOADS

The submerged design drag loads for all piping and structures in the
suppression pool is conservatively based on the LOCA bubble pressure
given in GESSAR II Figure 3B-67. The following is a comparison between
the PNPP design basis and the GE design criteria described in
Attachment L.

The sample structure used for comparison is a two-foot long, one-foot
diameter pipe located four feet from the drywell wall as depicted in
GESSAR II Figure 3BL-8.

The PNPP design methodology yields 4,810 lb_f. The GE methodology yields
1,235.8 lb_f as shown in Table 3BL-5 of GESSAR II. The GE criteria
recommends multiplying this result by a factor of 2 which results in a
load of 2,471.6 lb_f or approximately 50 percent of the PNPP design
basis. In addition, appropriate dynamic load factors have been applied
to each load used in the design.

3BL.2.4   FALLBACK LOADS

No deviations from Attachment L criteria.

3BL.2.5   LOCA CONDENSATION OSCILLATIONS LOADS

LOCA condensation oscillation drag loads are bounded by the PNPP LOCA
bubble pressure drag load methodology, with the exception of the ECCS
strainer. The ECCS strainer is designed for condensation oscillation
loads.

3BL.2.6   LOCA CHUGGING LOADS

LOCA chugging drag loads are bounded by the LOCA bubble pressure drag
loads, with the exception of the ECCS strainer. The ECCS strainer is
designed for chugging loads.

3BL.3   <u>SUBMERGED STRUCTURE LOADS DUE TO SRV ACTUATIONS</u>

3BL.3.1   QUENCHER WATER JET LOAD

No deviation from Attachment L criteria.

3BL.3.2   QUENCHER BUBBLE LOAD

The design submerged drag loads for all piping and structures in the
suppression pool, with the exception of the ECCS strainer, is

conservatively based on applying the peak SRV bubble pressure (reduced by the ratio of r_o (peak bubble pressure) to r (distance of structure from bubble center) to the component.

For these loads, the PNPP unique maximum quencher bubble pressure as calculated per Section 3BA.12.6 was used to generate design loads for submerged piping and structures. The bubble pressure was attenuated by distance as described in Section 3BA.10.3.1.

The sample structure used for comparison is shown in GESSAR II Figure 3BL-10.

The PNPP design basis yields 10.63 psi. The GE methodology, as shown on Page 3BL-33/3BL-34 of GESSAR II, yields a peak load of 0.36 psi or less than 4 percent of the PNPP design basis. In addition, appropriate dynamic load factors have been applied to each load used in the design.

The quencher bubble loads for the ECCS strainer are determined utilizing the GESSAR II methodology.

ATTACHMENT M TO APPENDIX 3B

POOL SWELL VELOCITY

This is not applicable to Perry. Determination of maximum pool swell velocity ranging from zero feet per second at the pool surface to a maximum of 50 feet per second as a function of height is based on the provisions of the NRC Draft Acceptance Criteria.

ATTACHMENT N TO APPENDIX 3B

MULTIPLE SAFETY/RELIEF VALVE ACTUATION FORCING

FUNCTION METHODS

3BN.1   INTRODUCTION

No deviations.

3BN.2   RANDOM PARAMETERS

3BN.2.1   Reactor Vessel Pressure Rise Rate (PRR)

No deviations.

3BN.2.2   Valve Setpoint Tolerance (VST)

GESSAR II Figures 3BA-4 through 3BA-6 and GESSAR II Figures 3BA-84
through 3BA-87 are not applicable to PNPP. Safety/relief valve
discharge locations are shown in USAR <Figure 3BA-3>.

3BN.2.3   Valve Opening Time (VOT)

No deviations.

3BN.2.4   Quencher Bubble Frequency Distribution (QBF)

No deviations.

3BN.3   MONTE CARLO TRIAL SIMULATIONS

No deviations.

3BN.4   FACTORS AFFECTING PRESSURE DISTRIBUTION ON THE SUPPRESSION
           POOL BOUNDARY

No deviations.

3BN.5   FORCING FUNCTIONS FOR NSSS EQUIPMENT

No deviations.

3BN.6   STRUCTURAL RESPONSE ANALYSIS

Resulting dynamic responses for the critical cases selected in
Subsection 3BN.5.4 are enveloped and used for both BOP and NSSS
equipment and structural evaluations.

3BN.7   EXAMPLE OF TYPICAL TIME SEQUENCING APPLICATION

The use of PNPP unique parameters and the procedures in this subsection
result in PNPP unique SRV forcing functions which have been used to
determine the dynamic response necessary for NSSS and balance of plant
equipment evaluation.

3BN.8   COMPARISON OF SELECTED TRIALS WITH THE FOURIER SPECTRA OF THE
           59 MONTE CARLO SIMULATIONS

No deviations.

3BN.9   CONSERVATISM OF SRV METHODOLOGY

No comments.

ATTACHMENT O TO APPENDIX 3B

DATA AND ANALYSES PERTAINING TO SRV ACTUATION

RANDOM PARAMETERS

No deviations for the entire Attachment.