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3.0 Design Criteria - Structures, Components, Equipment and Systems

(HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED)

The station structures, components, equipment and systems, collectively referred to as integral facilities, have a classification in accordance with their function and the degree of integrity required to protect the public.

The integral facilities design for normal conditions is governed by the applicable design codes. The design for loss of coolant accident, maximum seismic excitation, tornado wind, and missiles assures no loss of function.

Each of the engineered safety features is designed to tolerate a single failure during the period of recovery following an incident without loss of its protective function. This period of recovery consists of two segments; the short-term period and the long-term period.

During the short-term period, the single failure is limited to a failure of an active component to complete its function as required. Should the single failure occur during the long-term period rather than the short-term, the engineered safety features are designed to tolerate an active failure or a passive failure without loss of its protective function.

The following definitions are applicable to terms that pertain to the single failure criterion:

<u>Period of Recovery</u>: The time necessary to bring the unit to a cold shutdown and regain access to faulted equipment. The recovery period is the sum of the short and long-term periods defined below.

<u>Incident</u>: Any natural or accidental event of infrequent occurrence and its related consequences which affect the unit operation and require the use of engineered safety features. Such events, which are analyzed independently and are not assumed to occur simultaneously, include the loss-of-coolant accident, steam line ruptures, steam generator tube ruptures, etc. A blackout may be an isolated occurrence or may be concurrent with any event requiring engineered safeguards systems use.

<u>Short Term</u>: The time immediately following the incident during which automatic actions are performed, system responses are checked, type of incident is identified and preparations for long-term recovery operation are made. The short term is the first 24 hours following initiation of system operations.

<u>Long Term</u>: The remainder of the recovery period following the short term. In comparison with the short term where the main concern is to remain within NRC specified site criteria, the long-term period of operation involves bringing the unit to cold shutdown conditions where access to the Containment can be gained and repair effected.

<u>Active Failure</u>: The failure of a powered component such as a piece of mechanical equipment, component of the electrical supply system or instrumentation and control equipment to act on command to perform its design function. Examples include the failure of a motor-operated valve to move to its correct position, the failure of an electrical breaker or relay to respond, the failure of a pump, fan or diesel generator to start, etc. Equipment moving spuriously from the proper safeguards position without signal, such as a motor operated valve inadvertently shutting at the moment it is required is not considered credible.

<u>Passive Failure</u>: The structural failure of a static component which limits the component's effectiveness in carrying out its design function. When applied to a fluid system, this means a

break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm for 30 minutes. Such leak rates are consistent with limited cracks in pipes, sprung flanges, valve packing leaks or pump seal failures.

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3.1 Conformance with General Design Criteria

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

This section discusses briefly the design criteria for the facility structures, systems and components important to safety and how these criteria meet the NRC "General Design Criteria for Nuclear Power Plants" specified in Appendix A to 10CFR Part 50. The sections of the FSAR where more detailed information is presented are also referenced.

CRITERION 1 - QUALITY STANDARDS AND RECORDS

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

DISCUSSION:

Duke complies with Criterion 1. The structures, systems, and components of this facility are classified, as defined in ANS N18.2 according to their importance in the prevention and mitigation of accidents using generally recognized engineering codes and standards. Items, thus classified, are listed in Table 3-1, Table 3-2, Table 3-4 and Table 3-7. Duke's quality assurance program conforms with the requirements of 10CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants. This Quality Assurance program is described in Chapter 17. Included in this quality assurance program is specific direction for the maintenance of appropriate records.

Reference: Chapter 3 and Chapter 17.

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 structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quality, and period of time in which the historical data have been accumulated, (2) appropriate combination 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.

DISCUSSION:

Structures, systems and components designated Category 1 are designed to withstand, without loss of function, the most severe natural phenomena on record for the site with appropriate margins included in the design for uncertainties in historical data.

The Operating Basis Earthquake for the design of Category 1 structures systems and components is 0.08 g acting horizontally and 0.0533 g acting vertically. The Safe Shutdown Earthquake is 0.15 g acting horizontally and 0.10 g acting vertically.

Reference: Chapter 2 and Chapter 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. Non-combustible 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.

DISCUSSION:

The station is designed to utilize non-combustible and heat-resistant materials, wherever practical.

Duplication and physical separation of components to provide redundancy against other hazards also protects against simultaneous failures due to local fires. The Fire Protection system provides fire detection equipment for areas where potential for fire is greatest or areas not normally occupied by personnel. Also provided are reliable supplies of water, and halon to appropriate parts of the station.

Reference: Section 9.5.1.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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.

DISCUSSION:

Structures, systems and components important to safety are designed to function in a manner which assures public safety at all times. These structures, systems and components are protected for all worst-case postulated conditions by appropriate missile barriers, pipe restraints, and station layout. The Reactor Building is capable of withstanding the effects of missiles originating outside the Containment such that no credible missile can result in a loss-of-coolant accident. The Control Room is designed to withstand such missiles as may be directed toward it and still maintain the capability of controlling the units.

Emergency core cooling components are austenitic stainless steel or equivalent corrosion resistant material and hence are compatible with the containment atmosphere over the full range of exposure during the post-accident conditions.

Reference: Chapter 3 and Section 6.3.

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

DISCUSSION:

Structures, systems, and components, which are either shared (a) between the two units or (b) among systems within a unit, are designed such that there is not interference with basic function and operability of these systems due to sharing. This design protects the ability of shared structures, systems and components to perform all safety functions properly.

Reference: Chapter 3, Chapter 6, Chapter 8, Chapter 9 and Chapter 11.

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.

DISCUSSION:

The reactor core with its related coolant, control and protection systems is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The Reactor Protection System is designed to actuate a reactor trip for any anticipated combination of unit conditions when necessary to assure that fuel design limits are not exceeded. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions or normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, trip of the turbine-generator, loss of normal feedwater and loss of both normal and preferred power sources.

References: Chapter 4 discusses the design bases and design evaluation of reactor components. Chapter 5 discusses the Reactor Coolant System. The details of the Reactor Protection and Engineered Safety Features Actuation Systems design and logic are discussed in Chapter 7. This information supports the accident analyses presented in Chapter 15.

CRITERION 11 - REACTOR INHERENT PROTECTION

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

DISCUSSION:

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the non-positive operational limit on moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design using low-enrichment fuel; the non-positive moderator temperature coefficient of reactivity is assured by administratively limiting the dissolved absorber concentration.

Reference: Chapter 4.

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.

DISCUSSION:

Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and non-positive moderator temperature coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the radial, diametral and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and non-positive moderator temperature coefficients of reactivity.

Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided as a result of reactor trip functions using the measured axial power imbalance as an input.

Oscillations, due to xenon spatial effects, in axial modes higher than the first overtone, are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

Reference: Chapter 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.

DISCUSSION:

Plant instrumentation and control systems are provided to monitor variables in the reactor core, coolant system, and Containment building over their predicted range for all conditions to the extent required. The installed instrumentation provides continuous monitoring, warning, and initiation of safety functions. The following processes are controlled to maintain key variables within their normal ranges:

1. Reactor power level (manual or automatically by controlling thermal load).
2. Reactor coolant temperature (manual or automatically by rod control cluster assembly motion, in sequential groups).
3. Reactor coolant pressure (manual or automatically by heaters and spray in the pressurizer).
4. Reactor coolant water inventory, as indicated by the water level in the pressurizer (manual or automatic charging flow).
5. Reactor axial power balance (manual by rod motion).
6. Reactor Coolant System boron concentration (manual or automatic makeup of charging flow).
7. Steam generator water inventory on secondary side (manual or automatic feedpump flow through feedwater control valves).

The Reactor Control System is designed to automatically maintain a programmed average temperature in the reactor coolant during steady state operation and to insure that unit conditions do not reach reactor trip settings as the result of a transient caused by a design load change.

The Reactor Protection System Trip setpoints are selected so that anticipated transients do not cause a DNBR of less than 1.3.

Proper positioning of the control rods is monitored in the Control Room by bank arrangements of individual meters for each rod cluster control assembly. A rod deviation alarm alerts the operator of a deviation of one rod cluster control assembly from its bank position. There are also insertion limit monitors with visual and audible annunciation to avoid loss of shutdown margin. Each rod cluster control assembly is provided with a sensor to detect positioning at the bottom of its travel. This condition is also alarmed in the Control Room. Four ex-core long ion chambers also detect asymmetrical flux distributions indicative of rod misalignment.

Movable in-core flux detectors and fixed in-core thermocouples are provided as operational aids to the operator. Chapter 7 contains further details on instrumentation and controls. Information regarding the radiation monitoring system provided to measure environmental activity and alarm high levels is contained in Chapter 11.

Overall reactivity control is achieved by the combination of soluble boron and rod cluster control assemblies. Long term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short term reactivity control for power changes is accomplished by the Rod Control System which automatically moves rod cluster control assemblies. This system uses input signals including neutron flux, coolant temperature, and turbine load.

Reference: Chapter 7 and Chapter 11.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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

DISCUSSION:

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. In addition to the loads imposed on the piping under operating conditions; consideration is also given to abnormal loadings such as pipe rupture where postulated and seismic loadings as discussed in Sections 3.6 and 3.7. The piping is protected from overpressure by means of pressure relieving devices as required by applicable codes.

Reactor coolant pressure boundary materials selection and fabrication techniques assure a low probability of gross rupture or significant leakage.

The materials of construction of the reactor coolant pressure boundary are protected by control of coolant chemistry from corrosion which might otherwise reduce its structural integrity during its service lifetime.

The reactor coolant pressure boundary has provisions for inspections, testing and surveillance of critical areas to assess the structural and leaktight integrity.

Reference: Chapter 5.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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.

DISCUSSION:

Transient analyses are included in Reactor Coolant System design which conclude that design conditions are not exceeded during normal operation. Protection and control set points are based on these transient analyses.

Additionally, reactor coolant pressure boundary components achieve a large margin of safety by the use of proven ASME materials and design codes, use of proven fabrication techniques, nondestructive shop testing and integrated hydrostatic testing of assembled components.

The effect of radiation embrittlement are considered in reactor vessel design and surveillance samples monitor adherence to expected conditions throughout unit life.

Multiple safety and relief valves are provided for the Reactor Coolant System. These valves and their set points meet ASME criteria for over-pressure protection. The ASME criteria are satisfactory based on a long history of industry use.

Reference: Chapter 5.

CRITERION 16 - CONTAINMENT DESIGN

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

DISCUSSION:

Reactor Containment is a free-standing steel structure housing the ice condenser which limits Containment pressure to a safe level during a loss-of-coolant accident. A concrete Reactor Building surrounding the steel vessel provides collection of leakage for filtration. The Containment also contains a spray system which aids the ice condenser in limiting pressure and provides cooling as long as necessary following a loss-of-coolant accident. The design pressure is not exceeded during any pressure transients resulting from the combined effects of heat sources with minimal operation of the Emergency Core Cooling and Containment Spray Systems.

Reference: Sections 3.8, 6.2, and 6.3.

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 electrical 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 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 sources 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.

DISCUSSION:

Reliability of electric power supply is assured through several independent connections and a redundant source of standby emergency power from two diesel generators per unit. The specific design criteria applied in the design of systems and components are in accordance with IEEE Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations, IEEE No. 308, 1971.

Specific provisions which assure the required reliability are as follows:

1. Any circuit can be switched under normal or fault conditions without affecting any other circuit.
2. Any single circuit breaker can be isolated for maintenance without interrupting the power or protection to any circuit.
3. Short circuits of a single main bus is located without interrupting service to any circuit.
4. Short circuit failure of the tie breaker results in the loss of its two adjacent circuits until it is isolated by disconnect switches.
5. Short circuit failure of a bus side breaker results in the loss of only one circuit until it is isolated.
6. Circuit protection from failure of the primary protective relaying is assumed by redundant relaying.

Two separate 230 kV transmission lines for Unit 1 connect the 230 kV switchyard to two separate half size main transformers which transform the voltage to 24 kV. Similarly, two separate 525 kV transmission lines for Unit 2 connect the 525 kV switchyard to two separate half size main transformers which transform the voltage to 24 kV. The separation of the two supplies at the 24 kV voltage level for each unit is maintained by the two generator breakers which open when the generator is disconnected from the system. The two supplies are further reduced in voltage to 6900 volts by two full sized unit auxiliary power transformers. The two supplies are then separately connected through breakers to the normal auxiliary switchgear where they are connected through breakers and separate cables to the essential auxiliary power system switchgear. Each of the supplies is normally available within seconds following the tripping of the reactor and the opening of the generator breakers.

In the event one of the unit auxiliary transformers is out of service for maintenance, the other transformer is sized to carry all auxiliaries of one operating nuclear unit plus the safety shutdown loads of the other nuclear unit. In addition, a manually-initiated tie to the normal auxiliary busses of the other nuclear unit is available.

Two separate circuits from the transmission network are normally available to each nuclear unit. In the event one of the circuits is unavailable, a manual connection is provided to the other unit's

Normal Auxiliary Power System to provide the required second circuit from the transmission network in compliance with GDC 17 and Regulatory Guide 1.32.

Reference: Chapter 8 and Section 8.2.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

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

DISCUSSION:

Provisions are made for periodic testing of all important components of the emergency power system. Further provision is made for periodic testing of the emergency diesel generators to assure their capability to start within design limits and to accept loads.

The 24 kV, 230 kV, and 525 kV circuit breakers and their protective relays are inspected, maintained and tested on a routine basis. The 6900 volt and 4160 volt circuit breakers and associated equipment are tested in-service by opening and closing the circuit breakers so as not to interfere with the operation of the station. The 600 volt circuit breakers, motor contactors and associated equipment are tested in-service by opening and closing the circuit breakers or contactors so as not to interfere with operation of the station.

Systems are designed to allow as much testing of the various safety systems as is practical. The operation of the onsite power sources are conducted on a periodic basis and this includes starting each of the two diesel electric generating units assigned to each system and loading it to its continuous rating. Staggering of test periods is adhered to in order to avoid the testing of redundant equipment at the same time.

Reference: Chapter 8.

CRITERION 19 - CONTROL ROOM [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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

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

DISCUSSION:

The station is provided with a Control Room located in the Auxiliary Building where the nuclear power unit is operated under normal and accident conditions. The Control Room is designed and equipped to minimize the possibility of events which might preclude occupancy. In addition, provisions have been made for bringing both units to and maintaining them in a hot standby condition for an extended period of time from locations outside the main control room. Hot standby is a stable condition automatically reached following a unit shutdown. This capability is consistent with GDC 19 of 10CFR50. The term "hot shutdown" used in GDC 19 corresponds to the term "hot standby" as defined in the Standard Technical Specifications (Reference Regulatory Guide 1.68.2 "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants"). If necessary, the reactor may subsequently be placed in the cold shutdown condition.

The employment of non-combustible and fire retardant materials in the construction of the Control Room, the limitation of combustible supplies, the location of fire fighting equipment, and the continuous presence of a highly trained operator minimizes the possibility that the Control Room will become uninhabitable. Additionally, the Control Area Ventilation System is designed to maintain the control room at a positive pressure to minimize airborne radioactivity in-leakage. Under high radiation conditions, makeup air is recycled through a system of filters.

Sufficient shielding, distance, and Containment integrity are provided to assure that Control Room personnel shall not be subjected to doses under postulated accident conditions which would exceed 5 rem whole body.

Reference: Chapter 7 and Sections 3.8, 6.4, 12.1 and 12.2.

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.

DISCUSSION:

A fully automatic Reactor Protection System (with appropriate redundant channels) is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for Reactor Protection Systems meets the requirements of IEEE Standard No. 279. The Reactor Protection System automatically initiates a reactor trip when any monitoring variable or combination of variables exceeds its normal operating range. Setpoints are chosen to provide an envelope of safe operating conditions with adequate margin for uncertainties to assure that the DNBR does not go below 1.3 and that the linear heat generation rate is kept within limits discussed in Chapter 15 for ANS N18.2, Conditions 1 and 11.

Reactor trip is initiated by removing power to the rod mechanisms of all the full length rod cluster control assemblies. This allows the assemblies to free fall into the core, rapidly reducing the reactor power output. The protective actions which cause a reactor trip are detailed in Chapter 7.

The Engineered Safety Features Actuation System automatically initiates emergency core cooling, and other Engineered Safety Features functions, by sensing accident conditions using redundant analog channels measuring diverse parameters. Manual actuation of safeguards is relied upon where ample time is available for operator action. The Engineered Safety Features Actuation System also provides reactor trip on manual or automatic safety injection signal generation.

Reference: Chapter 7 and Chapter 15.

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protective 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 redundance unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit 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.

DISCUSSION:

The protection system is designed to comply with the intent of IEEE-279-1971 IEEE Criteria for Nuclear Power Generating Station Protection Systems. It provides high functional reliability and adequate independence, redundancy, and testability commensurate with the safety functions of the system. Actuation circuitry is provided with a capability of on-line testing. This extends to the final actuating device except where operational requirements prohibit actual operation of the device, e.g., turbine trip, steam line isolation, etc.

The Reactor Protection System is designed for high functional reliability by providing electrically isolated and physically separated, redundant analog channels and two separate and independent trip logic trains. This assures that no single failure results in the loss of any protection function. Except for certain defined backup trip functions detailed in Chapter 7, the redundancy and independence provided in the Reactor Protection System allows individual channel test or calibration to be made during power operation without negating reactor protection or the single failure criterion. This testing determines failures and losses of redundancy that may have occurred. This arrangement also permits removal from service of a channel while still maintaining the high reliability of the protection function. Details of the protection system design and testing provisions are contained in Chapter 7.

There are two series-connected circuit breakers which supply all power to the full length rod drive mechanisms. A reactor trip signal is fed to the undervoltage coils of both breakers simultaneously and opening of either breaker will trip the reactor.

The Engineered Safety Features Actuation System is also designed to meet IEEE-279 requirements.

The Engineered Safety Features Actuation System is testable at power with certain exceptions as detailed in Chapter 7. As with the components of the Reactor Protection System, both physical and electrical separation are practiced for the Engineered Safety Features Actuation System to provide a high degree of availability for its safety function.

Reference: Chapter 7.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

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

DISCUSSION:

Protection system components are designed and arranged so that the environment accompanying any emergency situation in which the components are required to function does not result in loss of the safety function. Various means are used to accomplish this. Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. Generally, two or more diverse protection functions would automatically terminate an accident before unacceptable consequences could occur.

For example, there are automatic reactor trips based upon neutron flux measurements, reactor coolant loop temperature measurements, pressurizer pressure and level measurements, reactor coolant pump bus under-frequency and under-power measurements and initiation of a safety injection signal.

Regarding the Engineered Safety Features Actuation System for a loss of coolant accident, a safety injection signal can be obtained manually or by automatic initiation from two diverse sets of signals:

1. Low pressurizer pressure.
2. High containment pressure.

For a steam line break accident, diversity of safety injection signal actuation is provided by:

1. Low pressurizer pressure
2. For a steam break inside Containment, high Containment pressure provides an additional parameter for generation of the signal.

All of the above sets of signals are redundant, physically separated and meet the intents of the criteria.

High quality components, suitable derating and applicable quality control, inspection, calibration and tests are utilized to guard against common mode failure. Qualification testing is performed on the various safety systems to demonstrate satisfactory operation at normal and post accident conditions of temperature, humidity, pressure and radiation. Typical protection system equipment is subjected to type tests under simulated seismic conditions using conservatively large accelerations and applicable frequencies.

Reference: Chapter 6 and Chapter 7.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fall 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.

DISCUSSION:

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of energy sources and the environment.

Each reactor trip channel is designed on the de-energize-to-trip principle so that a loss of power or disconnection of the channel causes that channel to go into its tripped mode. In addition, a loss of power to the full length rod cluster control assembly drive mechanisms causes them to insert by gravity into the core.

In the event of a loss of the preferred offsite power source, onsite diesel generators are available to power emergency loads and the station batteries to power the vital instrumentation loads. The diesels are capable of supplying power to the safety injection pumps, and associated valves. A loss of power to one train of safety injection equipment does not affect the ability of the other train to perform its function.

Reference: Chapter 7.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

The protection system shall be separated from control systems to the extent that failure of a 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, redundance, 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.

DISCUSSION:

Protection and control channels in the facility protection systems are designed in accordance with the IEEE-279-1971, "IEEE Criteria for Nuclear Power Plant Protection Systems".

The Reactor Protection system itself is designed to maintain separation between redundant protection channels and protection logic trains. Separation of redundant analog channels originates at the process sensors and continues along the wiring route and through Containment penetrations to analog protection racks and terminates at the Reactor Protection System logic racks. Isolation of wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Analog equipment is separated by locating components associated with redundant functions in different protection racks. Each redundant protection channel set is energized from a separate AC power feed.

The redundant reactor trip logic trains (two) are physically separated from one another. The Reactor Protection System is comprised of identifiable channels which are physically separated and electrically isolated.

Channel independence is carried throughout the system from the sensor to the logic interface. In some cases, however, it is advantageous to employ control signals derived from individual protection channels through isolation amplifiers contained in the protection channel. As such, a failure in the control circuitry does not adversely affect the protection channel.

The protection and control functions are thus separate and distinct. Test results proved that failure of any single control system component or channel including any short or ground or applying available AC or DC voltages to the control side (output) of the isolation amplifier, did not perceptibly disturb the protection side (input) of the amplifier.

The electrical supply and control conductors for redundant or back up circuits have such physical separation as is required to assure that no single credible event prevents operation of the associated function by reason of electrical conductor damage. Critical circuits and functions include power, control and analog instrumentation associated with the operation of Reactor Protection, Engineered Safety Features Actuation, Reactor and Residual Heat Removal Systems.

Reference: Chapter 7 and Chapter 8.

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.

DISCUSSION:

Reactor shutdown with control rods is completely independent of the control functions since the trip breakers interrupt power to the full length rod drive mechanisms regardless of existing control signals. The design is such that the system can withstand accidental withdrawal of control groups or unplanned dilution of soluble boron without exceeding acceptable fuel design limits.

Analyses of the effects of the other possible malfunctions are discussed in Chapter 15. The reactivity control systems, which are discussed further in Chapter 7, are such that acceptable fuel damage limits will not be exceeded even in the event of a single malfunction.

Reference: Chapter 7 and Chapter 15.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCE 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 or reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

DISCUSSION:

Two reactivity control systems, control rods and chemical shim, are provided.

The control rod positive insertion system relies on gravity-fall of the rods. In all analyses involving reactor trip, the single, highest-worth rod cluster control assembly is postulated to remain untripped in its full-out position.

The boron system can compensate for all xenon burnout reactivity transients without exception. The rod system can compensate for xenon burnout reactivity transients over the allowed range of rod travel. Xenon burnout transients of larger magnitude must be accommodated by boration or by reactor trip. The Boron System cannot compensate for the reactivity effects of fuel/water temperature changes accompanying power level changes. The rod system can compensate for the reactivity effects of fuel/water temperature changes accompanying power level changes over the full range from full load to no load at the design maximum ramp condition. Automatic control of the rods is, however, limited to the range of approximately 15 percent to 100 percent of rating for reasons unrelated to reactivity or reactor safety. The boron system maintains the reactor in the cold shutdown regardless of the disposition of the control rods.

Details of the construction of the rod cluster control assembly are included in Chapter 4, with the operation discussed in Chapter 7. The means of controlling the boric acid concentration are included in Chapter 9.

Reference: Chapter 4, Chapter 7, and Chapter 9.

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.

DISCUSSION:

Sufficient shutdown capability is provided to maintain the core subcritical for any anticipated cooldown transient, i.e., accidental opening of a steam bypass or relief valve or safety valve stuck open. This shutdown capability is achieved by a combination of RCCA and automatic boron addition via the Emergency Core Cooling System with the most reactive control rod assumed to be fully withdrawn. Manually controlled boric acid addition is used to supplement the RCCA in maintaining the shutdown margin for the long-term conditions of xenon decay and unit cooldown.

Reference: Chapter 4 and Chapter 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.

DISCUSSION:

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods and boron removal are limited to values that prevent rupture of the Reactor Coolant System boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The appropriate reactivity insertion rate for the withdrawal of RCCA and the dilution of the boric acid in the Reactor Coolant Systems are specified in the Technical Specifications for the facility, Chapter 16. The specification includes appropriate graphs that show the permissible mutual withdrawal limits and overlap of functions of the several RCCA banks as a function of power. These data on reactivity insertion rates, dilution and withdrawal limits are also discussed in Section 4.3. The capability of the Chemical and Volume Control System to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 9. The relationship of the reactivity insertion rates to unit safety is discussed in Chapter 15.

Assurance of core cooling capability following accidents, such as rod ejection, steam line break, etc., is given by keeping the reactor coolant pressure boundary stresses within faulted condition limits as specified by applicable ASME codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of needed safety features.

Reference: Chapter 4, Chapter 9, Chapter 15 and Chapter 16.

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 occurrences.

DISCUSSION:

The protection and reactivity control systems are designed to assure extremely high reliability in regard to their required safety functions in any anticipated operational occurrences. Likely failure modes of system components are designed to be safe modes. Equipment used in these systems is designed, constructed, operated and maintained with a high level of reliability. Loss of power to the protection system results in a reactor trip. Details of system design are covered in Chapter 7.

Reference: Chapter 7.

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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.

DISCUSSION:

Reactor coolant pressure boundary components are designed, fabricated, inspected and tested in conformance with Section III of ASME Boiler and Pressure Vessel Code. Major components are classified as ANS N18.2 Safety Class I and are accorded the quality assurance measures appropriate to this classification.

Leakage is indicated by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. This make-up is monitored. The reactor vessel closure joint is provided with a temperature monitored leak-off between double gaskets. Leakage inside the Containment is drained to the Containment sump where it is monitored.

Reference: Chapter 3, Chapter 5, Chapter 14, and Chapter 17.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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 rapid 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.

DISCUSSION:

Close control is maintained over material selection and fabrication for the Reactor Coolant System. The Reactor Coolant System materials which are exposed to the coolant are corrosion resistant stainless steel or Inconel. The nil ductility transition temperature of the reactor vessel material is established by Charpy V-notch and drop weight tests. These tests also insure that materials with insufficient toughness are not used.

1. *Ultrasonic Testing - Westinghouse requires the performance of 100 percent volumetric ultrasonic testing of reactor vessel plate for shear wave and a post-hydro test ultrasonic map of all welds in the pressure vessel. Also Westinghouse requires cladding bond ultrasonic inspection to more restrictive requirements than ASME Codes in order to preclude interpretation problems during inservice testing.*

2. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irridation and post-irridation testing of Charpy V-notch and tensile specimens. These programs are directed toward evaluation of the effects of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and is in accordance with ASTM-E185, “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels.”

The fabrication and quality control techniques used in the fabrication of the Reactor Coolant System are equivalent to those for the reactor vessel. The inspections of reactor vessel, pressurizer, reactor coolant pump casings piping and steam generator are governed by ASME code requirements.

Administrative controls are placed on plant heatup and cooldown rates, using conservative values for the change in ductility transition temperature due to irradiation to control vessel stresses below acceptable levels over the life of the plant while considering both allowable and postulated flows.

Details of the various aspects of the design and testing processes are included in Chapter 5.

Reference: Chapter 5.

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 leak-tight integrity, and (2) appropriate material surveillance program for the reactor pressure vessel.

DISCUSSION:

The design of the reactor coolant pressure boundary provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads, and external surfaces of the reactor coolant piping except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing the pressure boundary component's integrity. The reactor coolant pressure boundary is periodically inspected under the provisions of ASME B & PV Code, Section XI.

Monitoring of the RT_{NDT} properties of the reactor vessel core region plates forging, weldments and associated heat treated zones are performed in accordance with ASTM-E-185, Recommend Practice for Surveillance Testing on Structural Materials in Nuclear Reactors. Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation are used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below RT_{NDT}, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected RT_{NDT}, brittle fracture during normal operation is not considered to be a credible mode of failure. Additional details can be found in Section 5.2.

Reference: Chapter 5.

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.

DISCUSSION:

The Chemical and Volume Control System provides a means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below a present level. High pressure centrifugal charging pumps are provided which are capable of supplying the required makeup and reactor coolant seal injection flow with power available from either onsite or offsite electric power systems. These pumps also serve as high head safety injection pumps. In the event of a loss of coolant larger than the capacity of the normal makeup path, these pumps discharge into the larger safety injection piping and makeup line is automatically isolated. A high degree of functional reliability is assured by provision of standby components and assuring safe response to probable modes of failure. Details of system design are included in Chapter 6 and Chapter 9.

Reference: Chapter 6 and Chapter 9.

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 are 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.

DISCUSSION:

The Residual Heat Removal System, consisting of two redundant trains of pumps and heat exchangers, has appropriate heat removal capacity to ensure fuel protection. The system is Seismic Category 1 and is provided electric power by the diesel generators of the standby power system. This system supplements the normal steam and power conversion system which is used for the first stage cooldown (i.e., above 350°F and 400 psig). The Auxiliary Feedwater System complements the Steam and Power Conversion in this function. The systems together accommodate the single-failure criterion.

Reference: Chapter 15, Sections 5.5.7 and 6.3.

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.

DISCUSSION:

The ECCS design and safety analysis is in accordance with the NRC Acceptance Criterion for Emergency Core Cooling System for Light-Water Power Reactors of December 1973 (10CFR 50.46).

By combining the use of passive accumulators, centrifugal charging pumps, safety injection pumps and residual heat removal pumps, emergency core cooling is provided even if there should be a failure of any component in any system. The Emergency Core Cooling system (ECCS) employs a passive system of accumulators which do not require any external signals or source of power for their operation to cope with the short-term cooling requirements of large reactor coolant pipe breaks. Two independent and redundant high pressure flow and pumping systems, each capable of the required emergency cooling, are provided for small break protection and to keep the core submerged after the accumulators have discharged following a large break. These systems are arranged so that the single failure of any active component does not interfere with meeting the short-term cooling requirements.

The primary function of the ECCS is to deliver borated cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuelclad temperature and thereby ensures that the core remains intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical circumferential rupture of a reactor coolant loop.
2. A loss of coolant associated with a rod ejection accident.

References: Chapter 15 and Section 6.3.

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 integrity and capability of the system.

DISCUSSION:

Design provisions are made to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes, and valves for visual inspection and for non-destructive inspection where such techniques are desirable and appropriate, or required by codes.

The components located outside Containment are accessible for leaktightness inspection during operation of the reactor.

Details of the inspection program for the reactor vessel internals are included in Chapter 4 and Chapter 5. Inspection of the Emergency Core Cooling System is discussed in Chapter 6.

Reference: Chapter 4, Chapter 5 and Chapter 6.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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.

DISCUSSION:

The design provides for periodic testing of both active and passive components of the ECCS.

Preoperational performance tests of the components are performed in the manufacturer's shop. Initial system hydrostatic flow tests demonstrate structural and leaktight integrity of components and proper functioning of the system. Thereafter, periodic tests demonstrate that components are functioning properly.

Each active component of the ECCS may be individually actuated on the normal power source or transferred to standby power sources at any time during plant operation to demonstrate operability. The centrifugal charging/safety injection pumps are part of the charging system, and this system is in continuous operation during plant operation. The test of the safety injection pumps employs the minimum flow recirculation test line which connects back to the refueling water storage tank. Remote operated valves are exercised and actuating circuits are tested. The automatic actuation circuitry, valves and pump breakers also may be checked during integrated system tests performed during a planned cooldown of the Reactor Coolant System.

Design provisions include special instrumentation, testing and sampling lines to perform the tests during unit shutdown to demonstrate proper automatic operation of the ECCS. A test signal is applied to initiate automatic action and verification made that the safety injection pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers and automatic circuitry. In addition, the periodic recirculation to the refueling water storage tank can verify the ECCS delivery capability. This recirculation test includes all but the last valve which connects to the reactor coolant piping.

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions including transfer to alternate power sources for the ECCS to demonstrate the state of readiness and capability of the system. This functional test is performed with the water level below the safety injection signal set point in the pressurizer and with the Reactor Coolant System initially cold and at low pressure. The ECCS valving is set to initially simulate the system alignment for plant power operation.

Reference: Chapter 15 and Sections 6.3 and 13.6.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

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

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

DISCUSSION:

Containment heat removal is provided by (1) the ice condenser and (2) the Containment Spray System. The ice condenser is a passive system consisting of an energy absorbing ice bed in which steam is condensed during a loss-of-coolant accident. The condensation of steam in the ice bed limits pressure to a value less than Containment design pressure.

The Containment Spray System sprays cool water into the Containment atmosphere in the event of loss-of-coolant accident to assure that Containment pressure cannot exceed its design value. The recirculation mode allows for long-term heat removal by the Containment Spray System.

The loss of a single active component was assumed in the design of these systems. Emergency power system arrangements assure the proper functioning of the Containment Spray System during loss-of-power conditions.

Reference: FSAR Sections 6.2.2 and 6.5.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

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

DISCUSSION:

The ice condenser design includes provisions for visual inspections of the ice bed flow channels, door panels and cooling equipment. Where practicable, all active and passive components of the Containment Spray System are inspected periodically to demonstrate system readiness. Pressure containing systems are inspected for leaks for pump seals, valve packing, flanged joints and relief valves. During operational testing of the Containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks.

Reference: Sections 6.2.2 and 6.5.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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

DISCUSSION:

The ice condenser contains no active components required to function during an accident condition. However, samples of the ice are taken periodically to check the boron concentration. The door opening force is tested when the reactor is shutdown. The position of the ice condenser doors is monitored at all times. *All active components of the Containment Spray System are tested in the shop and again in place after installation. The system receives an initial flow test to assure proper dynamic functioning.* Further tests are conducted after any component maintenance.

Air test lines, located upstream of the isolation valves, are provided for checking that spray nozzles are not obstructed.

The transfer between normal and emergency power supplies is also tested.

Reference: Sections 6.2.2 and 6.5.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

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

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

DISCUSSION:

The Annulus Ventilation System consists of two full capacity fans, ducts, valves, moisture separators and filter trains for collecting and filtering contaminated gaseous leakage during accident conditions prior to its discharge to the unit vent. A failure of one train does not affect the operation of the other train. A crossover network is provided to cool the filter train associated with the redundant ventilation fan.

Hydrogen pocketing in subcompartments of the Containment is prevented by use of the Containment Air Return and Hydrogen Skimmer System.

The removal of iodine from containment atmosphere is accommodated by the combined effect of the Ice Condenser System, Reactor Building Purge System, and the Containment Spray System.

Reference:

Section 6.2.5
Section 6.2.2.14
McGuire Nuclear Station SER Section 6.2.3 Containment Heat Removal
McGuire Nuclear Station SER Section 6.2.4 Engineered Safety Feature Atmosphere Cleanup Systems
McGuire Nuclear Station SER, Amendment 51 to Facility Operating License NPF-9 And Amendment 32 to NPF-17

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

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

DISCUSSION:

All components of Containment atmosphere cleanup system are designed and located to facilitate scheduled inspections. All major components are located in the Auxiliary Building.

Reference: Section 6.2.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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.

DISCUSSION:

All active components are tested prior to initial plant installation and are tested periodically during unit life. In place testing of absolute and carbon filters assures that bypass flow paths have not developed and that filter material retains its capacity. The retentive capability of the carbon filter is tested by placing representative test carbon samples in the same air flow as the carbon bed.

Reference: Section 6.2.4.

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 redundance in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite 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.

DISCUSSION:

The cooling water systems important to safety are the Component Cooling System (CCS), a closed loop system which removes heat from the Residual Heat Removal System heat exchanger and other essential components, and the Nuclear Service Water (NSW) System, an open system which removes heat from the CCS, Containment spray heat exchangers, emergency diesel generator heat exchangers and the Auxiliary Feedwater System.

Component cooling water provides sufficient cooling capacity to fulfill all system requirements under normal and accident conditions. Adequate safety margins are included in the size and number of components to preclude the possibility of a component malfunction adversely affecting operating of safety features equipment.

The Nuclear Service Water System is designed to prevent any failure from curtailing normal unit operation or limiting the ability of the engineered safety features to perform their functions in the event of an accident. Design assures that loss of a complete header does not jeopardize plant safety.

Reference: Sections 9.2.1, 9.2.2 and Former Appendix 2G.

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

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

DISCUSSION:

The Component Cooling System (CCS) and the Nuclear Service System (NSW) are designed to permit periodic inspection, to the extent practical, of important components such as heat exchangers, pumps, valves and piping.

All heat exchangers, pumps, valves, piping, and instrumentation of the CCS are located outside the Containment with the exception of the excess letdown heat exchangers and the reactor coolant pump coolers. The nuclear service water pumps are located in the Auxiliary Building thus allowing proper maintenance and inspection.

Duke Energy's interpretation of Criterion 45 includes the following:

In many instances, the long term integrity of certain piping is assured by embedment in massive concrete, in which event the piping is not conducive to periodic inspection.

Reference: Section 9.2.1 and 9.2.2.

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 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.

DISCUSSION:

The Component Cooling System (CCS) and the Nuclear Service Water System (NSWS) are in operation during normal operation or shutdown. The structural and leaktight integrity of the CCS and NSWS and the operability and performance of active components are continuously demonstrated. The systems are designed to permit testing of system operation for reactor shutdown or loss-of-coolant accident conditions including the transfer between normal and emergency power.

Reference: Sections 9.2.1 and 9.2.2.

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.

DISCUSSION:

The Containment structure, including access openings and penetrations, is designed with sufficient conservatism to accommodate, without exceeding the design leakage rate, the transient peak pressure and temperature associated with a postulated reactor coolant piping break up to and including a double-ended rupture of the largest reactor coolant pipe.

The Containment structure and engineered safety features have been evaluated for various combinations of energy release. The analysis accounts for system thermal and chemical energy and for nuclear decay heat. The energy sink afforded by the Ice Condenser System and the Containment Spray System is adequate to prevent overpressurization of the structure.

Reference: Sections 3.8 and 6.2.

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.

DISCUSSION:

The Containment ventilation systems are sized to control the interior air temperature to 120°F during operation and 60°F during shutdown. The Containment is completely enclosed by thick concrete walls; therefore, it is not subject to sudden variations due to changes in external temperatures.

Safety of the structure under extraordinary circumstances and performance for the Containment at various loading stages are the main considerations in establishing the structural design criteria. In addition to providing for the leak tight integrity of the Containment under all loading conditions, the structural criterion for a low strain elastic response such that its behavior is predictable under all design loadings has been applied to the Reactor Building.

The Containment is designed for all credible conditions of loading, under normal and accident conditions. The load capacity of each load-carrying structural element is reduced by a yield capacity reduction factor that provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerance, may combine to result in undercapacity.

Reference: Sections 3.8 and 9.4.

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.

DISCUSSION:

The Containment design permits pre-operational integrated leak rate testing of the Containment and an inleakage rate of the Reactor Building. The integrated leak test at peak pressure verifies that the structure leaks less than the allowable value of 0.3 percent per day.

Duke Energy's interpretation of Criterion 52 includes the following: Some of the contents of Containment such as instrumentation, gauges, light bulbs, etc., cannot withstand leakage rate testing at the Containment design pressure.

Reference: Section 6.2.

CRITERION 53 - PROVISIONS FOR CONTAINMENT AND INSPECTION

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

DISCUSSION:

The Containment is designed such that integrated leak rates can be run during unit lifetime, and penetrations which have resilient seals or expansion bellows may be leak tested at design pressure at any time.

Reference: Section 6.2.

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.

DISCUSSION:

Piping penetrating the Containment is designed to withstand at least a pressure equal to the Containment design internal pressure. The design basis requires that no single failure or malfunction of an active component can result in loss of isolation or (intolerable) leakage.

Periodic closure and leakage tests are performed to assure that leakage is within specified limits.

Reference: Sections 3.8 and 6.2.

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 as acceptable on some other defined basis:

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

Isolation valve 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.

DISCUSSION:

Lines that are part of the reactor coolant pressure boundary and penetrate the primary Containment are provided with two barriers in series where they penetrate the Containment, so that failure of one active component does not prevent isolation. Isolation valves outside the Containment are located as close to the Containment as practical. Upon loss of actuating power automatic isolation valves are designed to take the position that provides the safety function in accordance with the single failure criterion.

Reference: Section 6.2.

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

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

1. One locked closed isolation valve inside and one locked closed isolation valve outside Containment, or
2. One automatic isolation valve inside and one locked closed isolation valve outside Containment, or
3. One locked closed isolation valve inside and one automatic isolation valve outside Containment. A simple check valve may not be used as the automatic isolation valve outside Containment, or
4. One automatic isolation valve inside and one automatic isolation valve outside Containment. A simple check valve may not be used as the automatic isolation valve outside Containment.

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

DISCUSSION:

Lines which connect directly to the Containment atmosphere and penetrate the primary Containment are provided with two barriers in series where they penetrate the Containment, so that failure of one active component does prevent isolation. Isolation valves outside the Containment are located as close to the Containment as practical. Upon loss of actuating power automatic isolation valves are designed to take the position that provides as the safety function in accordance with the single failure criterion.

Reference: Section 6.2.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

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

DISCUSSION:

Each line that penetrates the reactor Containment and is neither part of the reactor coolant pressure boundary nor connected directly to the Containment atmosphere has at least one Containment isolation valve located outside the Containment as close to the Containment as practical, the Residual Heat Removal System excepted.

Reference: Chapter 6 and Chapter 9.

CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIAL 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.

DISCUSSION:

Waste processing systems are incorporated in the facility design for processing and/or retention wastes generated during normal operation.

Chapter 11 describes the Waste processing system, and design criteria and amounts of estimated releases of radioactive effluents to the environment.

Reference: Chapter 11.

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.

DISCUSSION:

The spent fuel pool and cooling system, fuel handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions.

1. Components are designed and located such that appropriate periodic inspection and testing may be performed.

2. All areas of the station are design with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Chapter 12.

3. Individual components which contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems.

4. The Spent Fuel Cooling System provides cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed for testability to permit continued heat removal.

5. The spent fuel pool is designed such that no postulated accident could cause excessive loss of coolant inventory.

Reference: Sections 9.1, 11.2, 11.3, 11.5, and 12.1.

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.

DISCUSSION:

During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shutdown the core at a $k_{eff} \leq 0.95$. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. Borated water is used to fill the fuel pool at a concentration equal to that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure subcriticality even if unborated water is used to fill the pool. The design and operation of the new and spent fuel storage racks comply with 10CFR50.68(b).

Reference: Section 9.1.

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.

DISCUSSION:

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release operation. Radiochemical analyses are performed on all potentially radioactive wastes prior to their release to the environment. Radiation monitoring records and results of radiochemical analyses are maintain as permanent records of station releases.

The fuel pool cooling loop flow is monitored to assure proper operation.

A controlled ventilation system removes airborne radioactivity from the fuel storage and waste treating areas of the Auxiliary Building and discharges it to the atmosphere via the unit vent. Radiation monitors are in continuous service in these areas to actuate high-activity alarms in the main Control Room area.

Reference: Sections 9.1, 9.4, and 11.4.

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 station environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

DISCUSSION:

The Containment atmosphere, the unit vent, station water discharge and the waste disposal systems liquid effluent are monitored for radioactivity concentration during all operations.

All gaseous effluent from possible sources of accidental releases of radioactivity external to the Containment (e.g., the fuel pool) are exhausted from the unit vent which is monitored. All accidental spills of liquids are maintained within the Auxiliary Building and collected in a drain tank. Any contaminated liquid effluent discharged to the condenser circulating water is monitored. For the case of leakage from the Containment under accident conditions, the unit area radiation monitoring system, supplemented by portable survey equipment and the environmental radiation monitoring systems, provides adequate monitoring of accidental releases.

References: Sections 11.4, 11.6, and 12.2.

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3.2 Classification of Structures, Systems and Components

The term "nuclear safety related" or "safety related" is used throughout Duke Energy Carolinas documentation to indicate structures, systems, and components which QA Condition 1 processes and procedures outlined in Topical Report "Duke-1A" are applied. These structures, systems, and components are generally assigned to Duke Safety Classes 1, 2, and 3 (defined below). Mechanical and electrical system component safety classifications are fully described in Sections 3.2.2, 3.2.3, and 3.2.4.

3.2.1 Seismic Classification

All structures, systems and components required to shut down and maintain the reactor in a safe and orderly condition or prevent the uncontrolled release of excessive amounts of radioactivity have a seismic classification of Category 1. The recommendations of Regulatory Guide 1.29 are followed except as noted in section 3.2.4.2. Category 1 structures, systems, and components are tabulated in Table 3-1, Table 3-2, Table 3-4, and Table 3-7.

3.2.2 Mechanical System Quality Group Classification

Mechanical systems and components fall within four quality levels of groups A, B, C, and D which are directly related to codes and code classes as defined in Table 3-3. In addition, due to the complex nature of system functions and their importance to safety, mechanical systems and components are further classified by safety classification. Within a system, components or portions of systems may have differing classifications. In this sense, components will imply pressure vessels, tanks, piping, pumps, valves, and other equipment. The safety classifications - safety Class 1, Safety Class 2, Safety Class 3, and non-nuclear safety are applied relative to the design, materials, manufacture or fabrication, assembly, erection, construction, and operation. Systems and components which do not relate to nuclear safety are not assigned to a safety class. Table 3-3 correlates quality group classification and safety classification.

A safety system, as it is used below, is any system that is necessary to shut down the reactor, cool the core or cool another safety system or (after an accident) the containment, or that contains, controls, or reduces radioactivity released in an accident. Only those portions of a system are included that are designed primarily to accomplish one of the above functions or the failure of which could prevent accomplishing one of the above functions. A single system may have components in more than one Safety Class.

The following definitions of Safety Classes apply to fluid pressure boundary components and the reactor Containment. Supports which have a nuclear safety function are the same Safety Class as the components which they support. Selection of loading combinations and design methods for supports is the responsibility of the designer. (Reference: Section 3.1 and Section 3.9.3.2.9)

Duke complies with Regulatory Guide 1.26 except that position C.3 is modified as follows:

Piping and components in Group D are not safety-related. This fluid system classification is normally applied only to certain of radioactive waste management systems. Failure of a Group D component or piping would not result in an adverse effect on the health and safety of the public. Group D is typically applied to portions of liquid radioactive waste management systems handling fluids without entrained gases, fluids downstream from gas stripping processes, portions of solid waste management systems, or portions of gaseous waste management systems containing gases which are not normally held up for decay prior to release.

Portions of those radioactive waste management systems for which failure would result in an adverse effect on the health and safety of the public are in Group C.

Components and materials for Group D applications meet the code and standard requirements of Table 1. Manufacturers furnishing Group D components or materials are not required to have a quality assurance program meeting the 18 NRC criteria or ANSI N45.2-1971.

3.2.2.1 Safety Class 1

Safety Class 1, SC-1, applies to components whose failure could cause a Condition III or Condition IV loss of reactor coolant (Condition III and IV are defined in Chapter 15).

3.2.2.2 Safety Class 2

Safety Class 2, SC-2, applies to reactor Containment and to those components:

1. of the reactor coolant pressure boundary not in Safety Class 1 or
2. of safety systems that are necessary to: remove heat directly from the reactor or Containment, circulate reactor coolant from any safety system purpose, control within the reactor Containment radioactivity released, or control hydrogen in the reactor Containment.

3.2.2.3 Safety Class 3

Safety Class 3, SC-3, applies to those components not in Safety Class 1 or Safety Class 2 the failure of which would result in release to the environment of radioactive gases normally required to be held for decay, or those components that are necessary to:

1. provide or support a safety system function
2. control outside the reactor Containment airborne radioactivity released
3. remove decay heat from spent fuel.

3.2.2.4 Non-Nuclear Safety

Non-nuclear safety, NNS applies to portions of the nuclear power plant not covered by Safety Classes 1, 2, or 3 which can influence safe normal operation or which may contain radioactive fluids.

3.2.3 Safety Class Application

3.2.3.1 Safety Class 1

Safety Class 1 applies to reactor coolant pressure boundary components greater than 3/8" ID (unless protected by an orifice of this size) whose failure during reactor operations would prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal make-up systems only. Normal makeup systems are those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, power operation, or cooldown, using onsite power.

3.2.3.2 Safety Class 2

Safety Class 2 applies to:

1. The Containment including those valves and components of closed systems used to effect isolation of the Containment atmosphere from the outside environs.
2. Components of the reactor coolant pressure boundary not covered in Safety Class 1.
3. Safety system components of the following:
    a. Residual Heat Removal System.
    b. Portions of the reactor coolant auxiliary systems that form a reactor coolant letdown and makeup loop.
    c. Containment heat removal systems.
    d. Emergency Core Cooling System including injection and recirculation portions.
    e. Air cleanup systems used to reduce within the Containment radioactivity released in an accident.
    f. Containment hydrogen Control System.
    g. Portions of the steam and normal feedwater systems extending from and including the secondary side of the steam generator and including the outermost Containment isolation valves.

3.2.3.3 Safety Class 3

Safety Class 3 applies to:

1. Safety system components of the following:
    a. Portions of the reactor auxiliary systems that provide boric acid for the letdown and makeup loop.
    b. Auxiliary Feedwater System.
    c. Portions of component and process cooling systems that cool other safety systems, the control room or safety related electrical components.
    d. Spent Fuel Cooling System.
    e. Onsite emergency power supply and support systems.
    f. Air cleanup systems other than those listed in Section 3.2.3.2.
2. Onsite system components the failure of which would result in uncontrollable release to the environment of gaseous radioactivity normally held up. Typically these systems are:
    a. Portions of the reactor coolant auxiliary systems that do not form the letdown and makeup loop.
    b. Portions of the radioactive water processing system.

3.2.3.4 Non-Nuclear Safety

Portions of the nuclear power plant not covered under Safety Class 1, 2, or 3 are non-nuclear safety. This applies primarily to components of the secondary systems and waste disposal systems not otherwise covered. Also, included are safety system components (for example, small components) whose failure would not degrade system performance or cause a release to the environment of gaseous activity normally held up.

3.2.3.5 System Piping Classification

System piping is divided into eight classes, depending on the required function of the system or portion of a system as described in Section 3.2.2 and as required to distinguish analysis and purchasing requirements. Four of the eight piping classes result from the combination of the preceding system classifications with and without design for seismic loading, as indicated in Table 3-5. Refer to system flow diagrams (listed in Figure 6-3) for piping safety or quality class boundaries.

3.2.3.6 System Valve Classification

System valves are divided into eight classes, depending on the required function of the system or portion of the system the same as piping discussed in Section 3.2.2; i.e., a valve is the same class as the portion of system piping which includes the valve. The eight valve classes result from the combination of the preceding system classifications with and without design for seismic loadings as indicated in Table 3-6. Table 3-6 relates Duke class, ANS safety class and NRC Quality Groups to code classes.

3.2.3.7 System Classification Identification

System classifications are shown on Chapter 5, Chapter 6, Chapter 9, Chapter 10, and Chapter 11 system flow diagrams by the method outlined on Figure 6-1.

3.2.4 Electrical Systems and Components

Electrical systems and components are classified in accordance with their importance to safety. Within a system, components or portions of systems have different classifications. The safety classes are used to establish the criteria by which the systems, components, and materials are selected. designed, manufactured, assembled, installed and operated. The safety classifications are Safety Class 1E, Safety Class 2E, Safety Class 3E, and Safety Class 4E. Systems and components not included in these safety classes are not given a safety classification; however, they are constructed and installed, employing good practice and applicable codes and standards.

Table 3-7 is a tabulation of various criteria for major systems and components. The table includes information relative to safety classification, quality assurance, radioactivity, and natural phenomena design.

3.2.4.1 Classification

With regard to the electrical power, control, and instrumentation requirements of the plant, these safety classes are defined as follows:

Safety Class 1E

Safety Class 1E applies to nuclear safety related systems and components. In general, nuclear safety related systems and components are those which prevent or mitigate the consequences of postulated accidents which could cause undue risk to the health and safety of the public.

Safety Class 2E

Safety Class 2E applies to those non-nuclear safety related systems and components important to the management and containment of liquid, gaseous, and solid radioactive waste.

Safety Class 3E

Safety Class 3E applies uniquely to fire protection systems and components. While the fire protection systems do not perform a nuclear safety related function, additional engineering and regulatory considerations are met for fire protection systems protecting areas important to safety.

Safety Class 4E

Safety Class 4E applies to those seismically designed/restrained non-nuclear safety related systems and components whose continued function is not required during and after a seismic event, but whose failure, could reduce the ability of a nuclear safety related system or component to perform its intended function or cause an incapacitating injury to control room occupants.

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3.3 Wind and Tornado Loadings

All Category 1 structures, except those structures not exposed to wind, are designed to withstand the effects of wind and tornado loadings, without loss of capability of the systems to perform their safety functions. The following sections provide the basis for the wind and tornado parameters and methods used in meeting the wind and tornado criteria.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

The design wind velocity for all Category 1 structures is 95 mph, at 30 feet above the nominal ground elevation. According to ASCE Paper No. 3269, "Wind Forces on Structures," this velocity is the fastest mile of wind with a recurrence interval of 100 years.

3.3.1.2 Basis for Wind Velocity Selection

The basis for the wind velocity selection is the ASCE Paper No. 3269, Reference 1, as defined in Section 3.3.1.1. This reference summarizes existing technical literature on wind velocities distribution extending back to the early 1800's.

3.3.1.3 Vertical Velocity Distribution and Gust Factor

Reference 1 recommends that buildings and structures with a height to minimum horizontal dimension ratio exceeding five should be dynamically analyzed to determine the effect of gust factors. (All Category 1 structures at McGuire have a height/width ratio of less than 5). Considering the above, a gust factor of unity is used in the analysis and design of all Category 1 structures for determining wind forces. It is also concluded that wind forces on Category 1 structures is not a controlling load condition in the design.

The vertical velocity distribution is discussed in Section 3.3.1.4.

3.3.1.4 Determination of Applied Forces

ASCE Paper No. 3269, "Wind Forces on Structures," assembles existing information on the factors that determine applied wind forces on structures. Rectangular structures are designed for a wind distribution as defined in the above reference. The wind pressure distribution for the Reactor Building above grade is shown in Figure 3-1.

The wind design pressure magnitude "p" is calculated as follows:

$$p = Cpe \ \text{x} \ f \ \text{x} \ v^2$$

where:

Cpe	=	the coefficient of the actual pressure distribution on the structure to the dynamic pressure of the free stream as given in ASCE Paper 4933, Reference 4. The magnitude of Cpe is as shown in Figure 3-1.
f	=	the constant obtained from Reference 1 for determining the dynamic pressure of the free stream. f = p/2 where p is the air density.
v	=	the wind design velocity as previously defined in Section 3.3.1.1.

The design pressure distribution (which is similar to the Cpe distribution shown in Figure 3-1) is represented by Fourier Series. Individual harmonics are analyzed by Kalnin's Computer Program as defined in Section 3.8.2.4, and are combined to produce the stress resultants of the total series.

3.3.2 Tornado Loadings

All Category 1 structures, except those structures not exposed to wind, are designed for tornado wind loads. Tornado wind loads are not postulated simultaneously with the design earthquake.

3.3.2.1 Applicable Design Parameters

The design tornado used in calculating tornado loadings is in conformance with Regulatory Guide 1.76 with the following exceptions:

1. Rotational wind speed is 300 mph.
2. Translational speed of tornado is 60 mph.
3. Radius of maximum rotational speed is 250 feet.
4. Tornado induced negative pressure differential is 3 psi, occurring in three seconds.

3.3.2.2 Determination of Forces on Structures

Tornado wind loadings are calculated in accordance with Section 3.3.1.4, using the tornado wind velocities given in Section 3.3.2.1. Category 1 structures have been designed according to the following combinations:

$W_t = W_w$

$W_t = W_p$

$W_t = W_w + W_p$

where:

W_t = Tornado load

W_w = Tornado wind load

W_p = Tornado differential pressure load

A further evaluation has been performed on selected Category 1 structures for overall stability for tornado missiles. This evaluation has been performed in accordance with Reference 5, and concludes structural stability for tornado missiles impact.

3.3.2.3 Ability of Category 1 Structures to Perform Despite Failure of Structures not Designed for Tornado Loads

The Turbine Building was investigated to determine the extent of failure of the structure in the direction of the Auxiliary Building due to the effect of tornado loading.

It was determined that the metal siding panels will fail and be blown off at loads considerably below the design tornado loading. The siding will fail prior to the girts being loaded to failure. The structural steel framing of the Turbine Building and the Turbine Building cranes will be exposed to tornado wind as an open structure following the failure of the siding. The design tornado has a peak rotational velocity of 300 mph at a radius of 250 feet and a translational velocity of 60 mph. Due to the size of the building in relation to the radius of the tornado maximum wind velocity, an average design velocity over the area of the building calculated utilizing the tornado center chosen results in a maximum tornado wind on column line "I-H". The characteristics of the design tornado are based on data presented in Reference 3.

The methods of converting the tornado wind into forces on the structure are determined on the basis of recommendations and data presented in ASCE Paper 3269, "Wind Forces on Structures."

As a result of this investigation, it is concluded that the resistance of the building is sufficient to prevent collapse of the structure in the direction of the Auxiliary Building.

The Turbine Building cranes, if not anchored, could possibly be blown out the end of the building, falling onto the Auxiliary Building roof.

To preclude the occurrence of this event and to protect the Auxiliary Building from the falling of the Turbine Room cranes, the Turbine Building cranes will be parked at the furthermost end of the Turbine Building from the Auxiliary Building and securely anchored any time they are not in use. The cranes are not postulated to be in use at the time of a tornado.

Local yielding of light secondary members of the Turbine Building may occur resulting in missiles which are of smaller magnitude in relation to the tornado missiles previously defined in Section 3.5.

3.3.3 References

1. "Wind Forces on Structures," Paper No. 3269, ASCE Transactions, Vol. 126, Part II, 1061, P. 1124.
2. American National Standard, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," ANSI A58.1-1972, New York, New York.
3. Hoecker, W. H., "Wind Speed and Air Flow Patterns in the Dallas Tornado and Some Resultant Implications," Monthly Weather Review, May 1960.
4. Maher, Francis J., "Wind Loads on Dome-Cylinder Dome-Cone Shapes," Journal of Structural Division, ASCE paper No. 4933, October 1966.
5. Williamson, R. A., and Alvy, R. R., "Impact Effects of Fragments Striking Structural Elements," Holmes and Narver, Inc. Anaheim, California, Revised November 1973.

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3.4 Water Level (Flood) Design

Category 1 structures and components are not subject to flooding due to the earth dike constructed north of the station site (FSAR Figure 2-4) and are, therefore, not required to be designed for flood conditions. The critical Lake Norman pond elevation for Criterion No. 1 (Probable Maximum Flood) as defined in Section 2.4.2 is Elevation 767.9. This provides a freeboard of 7.1 feet at the Cowans Ford embankment and 12.1 feet lower (Elevation 762.6) than Criterion No. 1. Wave height and runup due to a maximum recorded wind velocity of 57 mph added to the Criterion No. 1 flood elevation increases the feet at the McGuire dike. The maximum tailwater elevation of 698.5 is 61.5 lower than the McGuire station yard.

Duke's position generally follows Regulatory Guide 1.59 (Rev. 2) in that the flood study covers the PMF, PMF due to seismic event, and the effect of wind action. However, a PMF was not selected to meet the Corps of Engineers definition but was selected on basis of worst known area flood plus 40%. The study does not include the effects of coincident smaller event floods nor does it consider seismic event floods occurring at the recommended water levels. The water levels for the seismic event were not as high as recommended by the Regulatory Guide but were at a reasonable and realistic level due to Duke's total control of the water shed. The flood study report does not include extensive historical flood data for use as backup data.

To prevent flooding of Category 1 structures due to maximum water elevation at the site (Elevation 760.375 feet, Reference Section 2.4.10), the following design features are provided:

1. All low level piping into the Auxiliary Building, such as nuclear service water pipes, are encased in the structural foundation slabs or walls and do not require seals.
2. All exterior entrances are at El. 760.5 feet or above, or they are provided with curbs, drains, or inclined ramps to prevent the inflow of water..
3. The only piping penetrating the exterior wall of the Auxiliary Building that is not encased in the concrete structure is piping for the fire protection system. The lower elevation of which this piping penetrates the building, is Elevation 755 feet plus 4 inches. A typical detail of the fire protection piping at the structure interface is shown in Figure 3-3. This piping is encased in concrete poured against the outside surface of the exterior wall which prevents excessive inflow due to maximum water elevation at the structure. A flexible water seal is provided on the inside of the building in the penetration between the penetration sleeve and the pipe to prevent inflow of water. No means are provided for periodic checking of these penetration seals.

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3.5 Missile Protection

3.5.1 Missile Barriers and Loadings

3.5.1.1 Internal Missiles

The interior structural elements of all Category 1 structures, except those structural elements shielded from internal missiles are designed to withstand the internal missiles effect. For internal missiles characteristics refer to Section 3.5.2.9

3.5.1.2 Turbine-Generator Missiles

All Category 1 structures, with the exception of the New Fuel Storage Vault exposed to these missiles are designed to withstand their effect and meet Regulatory Guide 1.115, Rev. 1. The credible turbine-generator missiles are low trajectory and the associated properties are given in Section 3.5.2.

3.5.1.3 Tornado Generated Missiles

Category 1 structures exposed to these design basis missiles are designed to withstand their effect with the exception of those Structures, Systems and Components included in the TORMIS probabilistic tornado risk analysis listed in Table 3-63 and as discussed in Section 3.5.2.8.1.1. A tabulation of the tornado generated missiles is given in Table 3-8.

3.5.1.4 Site Proximity Missiles

For the McGuire Station, aircrafts are not considered as credible missiles due to the established flying patterns close to the station.

Table 3-9 provides a summary of the major Category 1 structures that are designed for missile protection, along with the types of missiles they are protected against.

3.5.1.5 Diesel Generator Missiles

Each Diesel Generator shall be protected against missiles produced by the adjacent diesel generator by the appropriate section of the block wall separating Diesel Generator rooms A from B. The credible diesel generator missiles are given in Section 3.5.2.9.

3.5.2 Missile Selection

The specific missiles and the basis for selection as credible missiles are discussed in this Section. Some missiles which are not credible are pointed out and justified as prescribed below.

3.5.2.1 Reactor Coolant Pump Flywheel

The following precautionary measures, taken to preclude missile formation from the reactor coolant pump flywheel, assure that the flywheel will not produce missiles under any anticipated accident conditions.

1. The flywheel is fabricated from rolled, vacuum-degassed, ASME SA-533.

2. Flywheel blanks are flame-cut from the plate, with allowance for exclusion of flame-affected metal.
3. A minimum of three Charpy tests are made from each plate parallel and normal to the rolling direction to determine that each blank satisfies design requirements.
4. An NDTT less than 10°F is specified.
5. The finished flywheel is subjected to 100 percent volumetric ultrasonic inspection.
6. The finished machined bores are also subjected to magnetic particle or liquid penetrant examination.

These design fabrication techniques yield flywheels with a primary stress at operating speed less than 50 percent of the minimum specified material yield strength at room temperature (100 to 150°F).

The reactor coolant pump is driven by an induction motor. Thus, its rotational speed is controlled by supply frequency. Normal operation speed of the pump is 1189 rpm with a synchronous speed of 1200 rpm; however, in accordance with NEMA standards, it is designed for an overspeed of 125 percent of synchronous speed, i.e., 1500 rpm. Integrity demonstration of the RCP flywheel is discussed in Section 5.2.6. Additional discussion demonstrating integrity of the entire RCP is presented in Reference 1.

Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (Reference 1.), to be 3900 rpm, more than three times the operating speed. Verification, by testing, of the flywheel evaluation analytical procedure is presented in the Appendix of Reference 1.

Deleted Paragraph(s) per 2003 update.

An ultrasonic inspection capable of detecting at least 1/2 in. deep cracks from the ends of the flywheel and a dye penetrant or magnetic particle test of the bore, both at the end of 20 years, will be more than adequate as part of a plant surveillance program.

The design specifications for the reactor coolant pumps include as design conditions the stresses generated by both the Operational Basis and Safe Shutdown Earthquakes. Besides examining the externally produced loads from the nozzles and support lugs, an analysis is made of the effect of gyroscopic reaction on the flywheel and bearings and in the shaft, due to rotational movements of the pump about a horizontal axis, during these seismic disturbances. For the SSE, the pump maintains its pressure boundary integrity and flow coastdown capability.

Evaluation for License Renewal:

To estimate the magnitude of fatigue crack growth during plant life, an initial radial crack length of 10% of the distance through the flywheel (from the keyway to the flywheel outer radius) was conservatively assumed. The analysis assumed 6000 cycles of pump starts and stops for a 60-year plant life. The existing analysis is valid for the period of extended operation.

3.5.2.2 Control Rod Mechanism

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

1. Each control rod drive mechanism is completely assembled and shop-tested at 3450 psig.

2. The mechanism housings are individually hydrotested to 3107 psig as they are installed on the reactor vessel to the head adapters and checked during the hydrotest of the completed Reactor Coolant System.
3. Stress levels in the mechanism are not affected by system transients at power or by thermal movement of the coolant loops.
4. The mechanism housings are made of Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

In addition, a missile shield structure is provided over the control rod drive mechanisms which will block any missiles which might be generated in the event of a fracture of the pressure housing of any mechanism.

3.5.2.3 Control Rod Drive Mechanism Missile Shield

The analysis performed to identify the potential missiles associated with the rupture of a control rod drive mechanism housing is presented in this section.

The criterion to be followed is that these missiles shall not jeopardize the Containment integrity. A concrete slab with steel facing is located on top of the CRDM housing, as close as possible to the housing to limit the velocity of the ejected missiles, to minimize the probability of missiles missing the shield and striking the Containment plate and to minimize the probability of missiles ricocheting and damaging other CRDM housings.

The assumptions, method of analysis, and results of the calculation performed to identify potential missiles should a CRDM housing break are summarized herein. This analysis can be applied to any Westinghouse PWR, using the same CRDM design and with a reactor coolant design pressure of 2500 psia.

1. Type of CRDM Missiles

   Three types of missiles are analyzed:

   a. Plug on top of the CRDM housing
   b. Drive shaft
   c. Drive shaft and drive mechanism latched together

   The worst case, assumed for design, is the following:

   The top plug on the CRDM housing is assumed to become loose to be accelerated by the water jet, until it reaches the underside of the missile shield and partially perforates it.

   In the meantime, as soon as the top plug clears the break, the drive shaft and control rod cluster are pushed out of the core by the differential pressure of 2500 psi across the drive shaft.

   The drive shaft and control rod cluster, latched together, are assumed fully inserted when the accident starts.

   After approximately twelve feet of travel, the RCC spider hits the underside of the upper support plate. Upon impact the flexure arms in the coupling joining the drive shaft and control cluster will fracture, completely freeing the drive shaft from the control rod cluster. It was assumed that the control cluster would be completely stopped by the upper support plate. The drive shaft, however, continues to be accelerated until its top hits the missile shield.

At this time the shaft pushes the plug through the non-perforated layer of steel and into the concrete slab.

2. Housing Plug Ejection

   The reactor coolant discharge flow rate from the break has been calculated using the Burnell equation. The coolant pressure has been assumed constraint, at the initial value. A fluid exit velocity of 534 ft/sec has been calculated. The velocity of the plug was calculated equating the increase of the plug momentum to the decrease of the water jet momentum. No spreading of the water jet was assumed. The characteristics of the plug and the velocity reached by the plug as a function of the travel are summarized in Table 3-10.

   The depth of penetration in the missile shield steel plate has been calculated as illustrated in ORNL-NSIC-5, page 6.158, using a value of 60,000 psi for the target plate ultimate tensile strength and 14 in. for the side length of the "square window." The depth of penetration is 0.5 in. and 0.6 in., for the missile shield located three feet and five feet above the top of the CRDM housing, respectively.

3. Drive Shaft Ejection

   The drive shaft and rod control cluster (RCC) have been assumed to be accelerated by the differential pressure of 2500 psi across the drive shaft. After a rod travel of twelve feet, the velocity of the drive shaft and cluster (W=270 lb) is 130 ft/sec. The drive shaft has been assumed to become loose after the impact of the RCC spider on the upper support plate. No credit was taken for the energy absorbed in breaking the flexure arms.

   Upon impact, the RCC (with the spider) is assumed to be completely stopped by the upper support plate. The drive shaft (with the disconnect rod) (W=120 lb) is assumed to be further accelerated by the differential pressure of 2500 spi. A clearance of one foot is assumed between the top of the drive shaft when fully withdrawn and the top of the housing. Table 3-11 gives the characteristics of this missile and its velocity as a function of travel out of housing. Table 3-11 also gives the missile shield steel plate and concrete slab-thicknesses required to stop the drive shaft as a function of the distance between the missile shield bottom and the housing top. The thickness of the steel plate is assumed constant and equal to one inch.

   The critical kinetic energy required for penetration is calculated as recommended in ORNL-NSIC-5, page 6.158, using a value of 60,000 psi for the target plate ultimate tensile strength and 14 in. for the side length of the "square window." A value of 48,000 ft lb has been found for the perforating energy for the one inch thickness of steel. This value has been deducted from the drive shaft kinetic energy at the time of impact and the new reduced drive shaft velocity has been determined. The depth of penetration in the concrete slab was calculated according to NAVDOCKS P-51, April 1951 and a slab thickness of three times the depth of penetration has been chosen as a design value.

4. Housing Plug and Drive Shaft Impact of the Same Missile Shield Spot

   For this case, which is the design case, it has been assumed that the plug perforates partially the steel plate as indicated in 2. Then the drive shaft hits the plug and pushes it through the non-perforated steel plate layer and into the concrete. Two solutions can be adopted. The first is to use the concrete slab thickness found in 3 and to increase the steel plate thickness by the plug perforation depth. This will over estimate the concrete thickness because the drive shaft pushes the plug instead of penetrating directly (plug OD = 2.75 in., drive shaft OD = 1.75 in.).

The second solution is to keep a one inch steel plate thickness, and increase the concrete slab thickness by two inches and seven inches for the missile shield located three feet and five feet from the housing top, respectively.

5. Ejection of Drive Shaft Latched to Drive Mechanisms

   The velocity of this missile has been calculated as in 3. The missile characteristics for this case are summarized in Table 3-12.

   The critical kinetic energy for perforation is 100,000 ft-lb. Therefore, no perforation is expected.

   The possibility of missile shield displacement under impact has been analyzed. Should the missile shield be located three feet above the housing top, a maximum vertical missile shield displacement of 0.3 in. has been calculated assuming an elastic collision. This displacement should not present any problem.

6. Jet Thrust on the Missile Shield

   The jet thrust is 6000 lbs and the weight of the missile shield is more that 50,000 lbs if located at a distance of three feet or more above the housing top; and therefore, no overturning will occur.

3.5.2.4 Valves

All the isolation valves installed in the Reactor Coolant System have stems with back seat. This rules out the probability of ejecting valve stems as, even if it were assumed that the stem threads fail, analysis shows that the back seat or the upset end cannot penetrate the bonnet and thereby become a missile. Additional interference is encountered with air and motor operated valves. For these reasons, valve stems are not considered to be credible sources of missile.

Valves have been designed against bonnet-body connection failure and subsequent bonnet ejection by means of (a) using the design practice of ASME Section III which limits the allowable stress of bolting material (b) using the design practice of ASME Section III for flange design; and (c) by controlling the load during the bonnet body connection stud tightening process.

The pressure containing parts are designed per criteria established by the pump and valve code. Materials of construction for these parts are in accordance with ASME Section III (Reference Table 5-12 for a listing of materials.)

Proper stud torquing procedures and the use of torque wrench, with indication of the applied torque, limit the stress of the studs to the allowable limits established in the ASME Code. This stress level is far below the material yield. The complete valves are hydrotested per pump and valve code. The cast stainless steel bodies and bonnets are radiographed and dye penetrant tested to verify soundness.

Valves with nominal diameter of two inches or smaller may be forged and have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads are designed to withstand the hydrostatic end force. The pressure containing parts are designed per criteria established by the pump and valve code specification.

For the above reasons valve body or bolt failures are not considered credible. To further decrease the risk associated with a loss of reactor coolant, valves will be oriented, to the extent practical so that no vital equipment or openings in the missile barrier are in their potential ejection trajectory. Considerations to be given to valves connected to the piping above the pressurizer are discussed separately in Section 3.5.2.6.

3.5.2.5 Instrument Wells and Thimbles and Heater Elements

3.5.2.5.1 Reactor Coolant Piping Temperature and Pressure Elements

The only credible source of jet-propelled missiles from the reactor coolant piping is that represented by the temperature and pressure element assemblies. The resistance temperature element assemblies can be of two types: with well and without well. Two rupture locations have been postulated: (1) Around the welding between the boss and the pipe wall; and (2) at the welding (or thread) between the temperature element assemnbly and the boss for the "without well" element, and the welding (or thread) between the well and the boss for the "with well" element. Table 3-13 gives characteristics of these missiles and the missile velocity as calculated for a jet with a ten degree expansion half angle. The missiles generated by the pressure taps are less severe than the missiles assumed generated from the temperature element assemblies.

3.5.2.5.2

(A) Reactor Coolant Pump Temperature Element

A temperature element is installed on the reactor coolant pumps close to the radial bearing assembly. A hole is drilled in the gasket and sealed on the internal end by a steel plate. Should this plate break, the pipe plug on the external end of the hold could become a missile. The characteristics of this pipe plug missile are:

Weight: 0.25 lb
Discharge Area: 0.50 in.^{2}
Thrust Area: 0.50 in.^{2}
Impact Area: 0.50 in.^{2}
Weight to impact area ratio: 0.5 psi
Velocity: 260 ft/sec

(B) Instrument Assemblies

Should the welding between the instrumentation well and the pressurizer wall fail, the well can be accelerated and become a jet-propelled missile. The potential missile considered is the instrumentation well and sensor assembly:

Flow Discharge Area: 0.442 in.^{2}
Thrust Area: 1.35 in.^{2}
Impact Area: 1.35 in.^{2}
Weight: 5.5 lb
Missile Weight divided by
Impact Area: 4.1 psi
Velocity: 100 ft/sec

3.5.2.5.3 Pressurizer Heaters

The heaters are normally installed underneath the bottom head of the pressurizer. Should they get loose, they would strike the concrete mat without causing any damage. The characteristics of this type of missile are as follows:

Flow Discharge Area: 0.80 in.^{2}
Thrust Area: 2.4 in.^{2}
Impact Area: 2.4 in.^{2}
Weight: 15 lb

Missile Weight divided by
Impact Area: 6.25 psi
Velocity: 55 ft/sec

3.5.2.5.4 Systems Connected to the Reactor Coolant System

The potential missiles that could be generated from the instrumentation assemblies attached to the Reactor Coolant System (piping, pump, and pressurizer) are described previously in this section.

Upon impact of any of these missiles on a wall, only a small wall area will be affected, because of the small kinetic energy involved. Therefore, only the depth of penetration of these missiles requires checking. Generally, the minimum thickness of the reactor compartment walls and the operation deck is two feet of concrete. Calculations based on a wall thickness of two feet and the listed missile characteristics show that the critical velocity required to penetrate this wall thickness is at least twice the maximum anticipated velocity with a ten degree expansion angel jet. Hence, these missiles are not of concern from a penetration standpoint. Should direct impact occur with the steam generator shell, the shell would not be perforated.

3.5.2.6 Pressurizer

The pressurizer extends above the operating deck and is enclosed in a pressure compartment which is an extension of the operating deck. This pressure enclosure acts as a missile barrier and is designed for jet force load. Equipment in this region consists of the pressurizer safety valves, the motor operated isolation valves in the relief line, the air operated relief valves, the air operated spray valves, instrumentation assemblies and associated piping.

Supports for these lines should be capable of restraining movement of components and piping, under action of reaction and jet forces from circumferential pipe rupture, in accordance with the criteria of Section 3.6.2.

Characteristics of valve bonnet missiles are given in Table 3-14. Pressurizer instrumentation assembly missile characteristics are included in Section 3.5.2.5.

3.5.2.7 Turbine-Generator Missiles

Turbine missiles can be generated by a turbine overspeed. The credible low-trajectory turbine missiles and the associated properties are defined in Table 3-15 and Figure 3-4. Basis for selecting these missiles is given in Section 10.2.3.

3.5.2.8 Tornado Generated Missiles

Table 3-8 provides a summary of the design basis tornado-generated missiles. The integrity of Category 1 structures is not impaired by these missiles. This is accomplished by designing the exposed structure of steel reinforced concrete capable of withstanding the impact of tornado-generated missiles. Modifications to existing or the design of new Category 1 structures shall conform to the requirements of NRC RIS 2008-14.

Table 3-63 provides a list of Category 1 structures, systems and components that have not been designed to withstand the impact of design basis tornado-generated missiles. These SSCs were probabilistically shown that they will not be impacted or will not be damaged beyond an acceptable criteria if impacted as discussed in Section 3.5.2.8.1.3.

(HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED)

The following was added as part of a NRC request for additional information in order to perform a comparability review. The request was to determine penetration velocities for 2 missiles which were not part of the design basis missiles used during the Construction Permit (CP) stage (Table 3-8). The requested velocities are for category 1 structures with wall or roofs less than 2 feet thick.

In order to assess the degree of comparability of protection against tornado missiles provided in the CP stage with that presently under review by the NRC, an additional investigation has been performed to evaluate the following missiles:

1. *Steel rods, one inch diameter by three feet long, weight eight pounds.*
2. *Utility pole, 13-1/2 inch diameter, 35 feet long, weight 1490 pounds.*

Structural concrete barriers designed to provide missile protection having thicknesses less than two feet are as follows:

1. *Slabs - None*
2. *Walls:*
   a. *1'- 0" thick located on column line AA between column lines 53 to 59 constructed to elevation 782 feet.*
   b. *1'- 6" thick, location on column lines 49 and 63 between column line AA (Turbine Building) and Reactor Building shield building constructed to elevation 782 feet.*

The maximum horizontal velocities required to penetrate the barrier or generate secondary missiles within the wall elevations are as follows:

Missile	*Horizontal Req'd Velocity 12"*	*Horizontal Req'd Velocity 18"*
1	*186*	*232*
2	*184*	*229*

The horizontal velocity (ft./sec) required for penetration or generation of secondary missiles is based upon a constructed thickness equal to three times the penetration depth.

Separation of redundant components is not considered in the design of barriers for tornado missiles.

3.5.2.8.1 Probabilistic Tornado Missile Risk Analysis

A probabilistic tornado missile risk analysis (Reference 7) was completed using the TORMIS computer code which is based on the NRC approved methodology detailed in References 8, 9, and 10. The TORMIS analysis was performed in accordance with the guidance described in NRC TORMIS Safety Evaluation Report (Reference 11) and as clarified by Regulatory Issue Summary (RIS) 2008-14 (Reference 12).

3.5.2.8.1.1 Scope

The TORMIS analysis (Reference 7) includes plant components identified as necessary to safely shutdown the plant and maintain a shutdown condition that are located in areas not fully protected by missile barriers designed to resist impact from design basis tornado generated missiles. The plant components (also referred to as, targets) included in the analysis are listed in Table 3-63 and additional details regarding these targets (i.e. specific identification, description, location, and portion) are included in Reference 7, Volume 3.

3.5.2.8.1.2 TORMIS Computer Code

The TORMIS (TORnado MISsile Risk Analysis Methodology) computer code uses a Monte Carlo simulation method that simulates tornado strikes on a plant. For each tornado strike the tornado field is simulated; missiles are injected and flown; and the missile impacts on structures, systems, and components (SSCs) are analyzed. These models are linked to form an integrated time history simulation methodology. By repeating these simulations, the frequencies of missiles impacting and damaging individual plant components (targets) and groups of targets are estimated. Statistical convergence of the results is achieved by performing multiple replications with different random number seeds.

3.5.2.8.1.3 Analysis

The TORMIS results show that the arithmetic sum of damage frequencies for all target groups affecting the individual Units are lower than the acceptable threshold frequency of 1.0E-06 per year per Unit as established in Reference 13.

The following limiting inputs and assumptions were used in the analysis (Reference 7):

a. A site specific tornado hazard curve and data set for McGuire was developed using statistical analysis of the NOAA/National Weather Service Storm Prediction Center tornado data for the years 1950 through 2016. The analysis utilizes the Enhanced Fujita (EF) Scale wind speeds in the TORMIS simulations.

b. The missile characteristics and locations are based on plant walk down surveys and plant drawings. The plant walk downs were conducted during both non-outage and outage periods to capture both conditions. A stochastic (time dependent) model of the missile population is implemented in TORMIS. The stochastic approach to the missile population varies the missile populations in each of the TORMIS replications to account for predictable changes in plant conditions (i.e. increased missiles during outages) and the randomness inherent in the total number of missiles present at the plant at any given time.

c. Finite element analysis calculations were performed to determine the missile damage threshold velocity for tornado generated missiles that would cause unacceptable damage to selected targets which is then used as an input in the TORMIS model.

d. Boolean combinations of targets were developed, and the logic was applied to targets or target groups to account for redundancies in the system design or for the TORMIS modeling of a component as multiple targets. The failure logic for redundancy of the MainSteam lines when missile damage to the PORVs and MSSVs is beyond acceptable criteria, is that the Unit can sustain damage to one of four MainSteam line and the damage can be in multiple places on the same MainSteam line (PORVs, MSSVs, or associated components). Damage, beyond the acceptable criteria, on more than one line is considered a failure in TORMIS space. The failure logic for the Control Room Air Ventilation System (CRAVS) Intakes (VC/YC Air Intakes) and Spent Fuel Pools (SPF) is simultaneous tornado generated missile impacts to all the Unit 1 and Unit 2 VC/YC Air Intakes AND the entry of a tornado generated missile into either the Unit 1 or Unit 2 SFP that would impact any Spent Fuel assemblies above acceptable critical velocities.

e. Any tornado generated missile strikes to the VC/YC Air Intakes were conservatively assumed to crimp the Intakes closed.

f. The Utility Port Barriers in the Doghouse Upper Openings are conservatively taken into account for their resistance to a conservative selection of tornado generated missiles entering the Doghouse Upper Openings.

g. All tornado generated missiles are conservatively assumed to strike with an end-on, colinear impact

3.5.3.9 Diesel Generator Missiles

Section 8.3.1.1.7 identifies the concrete block wall separating Diesel Generator Rooms A & B as missile barrier. This barrier is to protect each of the diesel generators from damage produced by missiles coming from the other. An evaluation provided by NORDBERG Mfg. Co. concluded that missiles produced by over-revving of the diesel generator was not plausible. Failure of internal diesel parts would occur and stop the diesel generator before destructive missiles could be ejected. An alternate scenario was proposed in which debris was dropped onto the generator flywheel causing partial fragmentation. Because the missile trajectories would lie in the plane of the generator flywheel, only those portions of the dividing wall between Column Lines 43 and 44 and between Column Lines 68 and 69 need to provide missile protection.

The postulated missile was flywheel fragment weighing 10 pounds with cross section of 9 square inches. The energy contained in this missile was given as 1460 ft-lbs. Using previously used formulas by Amerikian in NAVDOCKS P-51, the missile penetration given in Table 3-16 was calculated conservatively based upon a 150 fps velocity in lieu of the velocity applicable to the kinetic energy of the postulated missile.

3.5.4 Selected Missiles

The import parameters associated with the internal as well as the external missiles and basis for selecting them as credible missiles are discussed in Section 3.5.2. Knowing the velocity of each of these missiles, an estimate of their kinetic energy is calculated and the potential effects on the missile barriers can be assessed.

3.5.5 Barrier Design Procedure

3.5.5.1 Protection of Containment Function

The missiles that might be generated in coincidence with a loss of reactor coolant shall not cause loss of function of the Engineered Safety Features or loss of Containment integrity.

The systems located inside the Containment have been examined to identify and classify potential missiles. The basic approach is to assure design adequacy against generation of missiles, rather than allow missile formation and try to contain their effects.

Components which are examined from the standpoint of missile generation which would result in a loss of reactor coolant accident as are listed below:

Control rod drive shafts and/or housings (see Section 3.5.2.2 and Section 3.5.2.3)

Valves (see Section 3.5.2.4)

Instrument wells and thimbles (see Section 3.5.2.5)

Catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings and piping leading to generation of missiles is not postulated. The reason for not providing protection for these types of missiles is that massive and rapid failure of those components is incredible because of the material characteristics, inspections, quality control during fabrication, erection and operation, conservative design and prudent operation as applied to the particular component. The reactor coolant pump flywheel is not considered a source of missiles for the reasons discussed in Section 3.5.2.1.

Nuts and bolts are of no concern, because of the small amount of elastic energy that can be stored in the bolt material.

Sections of piping are not credible free missiles, however, consideration is to be given to the effect of whipping of pressurized piping as discussed in Section 3.6.2.

The principal barrier protecting the Containment structure from missiles is the secondary shield and operating deck. The layout of these structures and pressurized equipment is such that the path of potential missiles which might otherwise escape from the reactor compartments through openings around equipment, venting holes or any other opening in the missile barriers will be directed into the missile barriers.

The effects of these missiles on the reactor compartment walls are evaluated. The depth of penetration into and the effect of energy transfer to the concrete structures is analyzed. These structures are capable of stopping the potential missiles and still perform their function. (Refer to Section 3.5.4.2 for more details.)

Other than for the ECCS lines which must circulate cooling water to the vessel, the Engineered Safety Features are located outside the reactor compartment and are protected by the same barriers which protect the Containment. The ECCS lines which penetrate the secondary shield are routed around and outside the secondary shield to penetrate the secondary shield in the vicinity of the loop to which they are attached.

The steam generator shell thickness is ample to resist penetration by postulated missiles listed in Section 3.5.2. For the lower steam generator shell connecting lines, routing of these lines shall be such that they are not in the direct path of postulated missiles.

3.5.4.2 Penetration Depth Estimates

The depth to which a missile penetrates a concrete barrier is estimated by use of the modified Petry Formula (Reference 2). As shown in details in Reference 3, where several penetration formulas are studied, the modified Petry formula which has received general industry acceptance to date is reasonably conservative for estimating the penetration depth of missiles for velocity range below 1000 ft/sec.

3.5.5 Missile Barrier Features

Missile Penetration is evaluated in accordance with Reference 2. According to this reference, spalling or perforation of the missile will not occur if the barrier thickness is at least three times the penetration depth, D. All Category 1 structures, which are designed as missile barriers, have a combination of concrete thickness and strength to ensure that the slab or wall has a thickness of at least 3D. All portions of Category 1 structures which are considered as missile barriers meet the above criterion of missile penetration as outlined in Reference 2.

A further evaluation has been performed on selected structures to evaluate the overall structural response due to missile impact in accordance with Reference 5. This evaluation has confirmed that structural stability and functional requirements are maintained.

The reinforcing pattern used in missile shielding areas, as a minimum, consists of two-way reinforcement in each face of the structural wall or slab. Additional reinforcement has been provided as required by design.

Table 3-16 provides further information on missile types, barrier types and missile penetration as well as minimum barrier thickness required as calculated from Reference 2.

A tabulation of minimum missile barrier thicknesses provided for all Category 1 structures is given in Table 3-17.

3.5.6 References

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5. R. A. Wiliamson and R. R. Alvy, "Impact Effects of Fragments Striking Structural Elements," Holmes and Narver, Inc., Anahelm, California Revised January 1975.
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11. NRC Safety Evaluation Report, "Electric Power Research Institute (EPRI) Topical Reports Concerning Tornado Missile Probabilistic Risk Assessment (PRA) Methodology", October 26, 1983 (Adams ML080870291).

12. NRC Regulatory Issue Summary 2008-14, "Use of TORMIS Computer Code for Assessment of Tornado Missile Protection", June 16, 2008 (Adams ML080230578).

13. Memorandum from Harold Denton, NRR Director, to Victor Stello, Deputy Executive Director for Regional Operations and Generic Requirements, "Position of use of Probabilistic Risk Assessment in Tornado Licensing Action," dated November 7, 1983 (Adams ML030020331).

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3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Systems in which Design Basis Piping Breaks Occur

3.6.1.1 Reactor Coolant System

In the design of a pressurized water reactor, special provisions are made for protecting the public against the consequences of major mechanical accidents, including a loss-of-coolant or steam line break accident.

Section 3.6 of this SAR defines the extent of the allowable mechanical damage considered in these accidents, the various systems and equipment which are necessary to recover from these accidents and the mechanical provisions which are provided to prevent unacceptable extension of the accident consequences.

The particular arrangement of the Reactor Coolant System, building structures and mechanical restraints preclude the formation of plastic hinges for breaks postulated to occur at the branch connection. Consequently, pipe whip and jet impingement effects of the postulated pipe break at these locations will not damage necessary safety-related structures, mechanical or electrical systems and equipment required to mitigate the consequences of the postulated break.

The Reactor Coolant System as used in this portion of the SAR is limited to the main coolant loop piping and all branch connection nozzles out to the first butt weld. Dynamic loading effects are only considered for pipe breaks postulated at branch connections.

The application of criteria applied for protection against the effects of postulated breaks at the branch connection results in a system response which can be accommodated directly by the supporting structures of the reactor vessel, the steam generator and the reactor coolant pumps. The design basis for postulated breaks in the Reactor Coolant System are discussed in Section 3.6.2.1.

3.6.1.2 All Other Mechanical Piping Systems

This Section discusses all piping systems excluding the Reactor Coolant System as described in Section 3.6.1.1 and is in accordance with NRC Branch Technical Position APSCSB 3-1 and Regulatory Guide 1.46 except as noted in Table 3-20.

Other mechanical piping systems, both inside and outside Containment, which are reviewed and considered in the design with respect to a postulated pipe break are those normally operating high energy and moderate energy lines which are safety-related or which pass near safety-related structures, systems or components, and include the Reactor Coolant System branch piping terminating at the main coolant loop piping nozzle.

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

1. Maximum temperature exceeds 200°F, or
2. Maximum pressure exceeds 275 psig.

Except that (1) non-liquid piping systems (air, gas, steam) with a maximum pressure less than or equal to 275 psig are not considered high energy regardless of the temperature, and (2) for liquid systems other than water, the atmospheric boiling temperature can be applied.

Systems are classified as moderate energy if the total time that either of the above conditions are met is less than either of the following.

1. One (1) percent of the normal operating lifespan of the plant, or
2. Two (2) percent of the time period required to accomplish its system design function.

Moderate energy lines are defined as those which have:

1. A maximum operating temperature less than or equal to 200°F, and
2. A maximum operating pressure less than or equal to 275 psig.

Systems which do not contain mechanical pressurization equipment are excluded from moderate energy lines; i.e., systems without pumps, pressurizing tanks, boilers, etc., and which operate only from gravity flow or storage tank water head are not considered moderate energy. Open ended vents and drains and piping furnished as a part of equipment are also not considered moderate energy. Systems or appropriate portions which fall in either or both of the above categories are analyzed as described in Section 3.6.4.2 and protected in accordance with Section 3.6.5.2. Table 3-18 lists High Energy Systems or portions thereof and Table 3-19 lists moderate energy systems or portions thereof in accordance with the above definitions that are analyzed for the station.

Section 3.9.2.8 discusses Containment integrity with respect to breaks involving mechanical penetrations.

3.6.2 Design Basis Piping Break Criteria

3.6.2.1 Postulated Piping Break Location Criteria for the Reactor Coolant System

The design basis for postulated pipe breaks include not only the break criteria, but also the criteria to protect other piping and vital systems from the effects of the postulated break.

A loss of reactor coolant accident is assumed to occur for a pipe break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3-5) on outgoing line[1] and down to and including the second check valve (Case III Figure 3-5) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve does not result in an uncontrolled loss of reactor coolant if either of the two valves in the line close.

Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint does not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Case I and IV in Figure 3-5) a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

Engineered Safety Features are provided for core cooling and boration pressure reduction, and activity confinement in the event of a loss of reactor coolant or steam or feedwater line break accident to ensure that the public is protected in accordance with 10CFR100 guidelines. These safety systems have been designed to provide protection for a Reactor Coolant System pipe

[1] It is assumed that motion of the unsupported line containing the isolation valves could cause failure of the operation of both valves.

rupture of a size up to and including a double ended severance of the Reactor Coolant System main loop.

Branch lines connected to the Reactor Coolant System are defined as “small” if they have an inside diameter equal to or less than 4 inches. This size is such that Emergency Core Cooling System analyses using realistic assumptions show that no clad damage is expected for a break area of up to 12.5 square inches corresponding to 4 inches inside diameter piping.

In order to assure the continued integrity of the vital components and the engineered safety systems, consideration is given to the consequential effects of the pipe break itself to the extent that:

1. The minimum performance capabilities of the engineered safety systems are not reduced below that required to protect against the postulated break;
2. The Containment leaktightness is not decreased below the design value, if the break leads to a loss of reactor coolant[2]; and
3. Propagation of damage is limited in type and/or degree to the extent that:
    a. A pipe break which is not a loss of reactor coolant would cause a loss of reactor coolant or steam or feedwater line break, and
    b. A reactor Coolant System pipe break would cause a steam-feedwater system pipe break and vice versa.

In the unlikely event that one of the small pressurized lines should fail and result in a loss of reactor coolant accident, the piping is restrained or arranged to meet the following additional criteria in addition to (1 through 3) above.

1. Break propagation must be limited to the affected leg, i.e., propagation to the other leg of the affected loop and to other loops is prevented;
2. Propagation of the break in the affected leg is permitted but is limited to a total break area of 12.5 square inches (4 inch inside diameter). The exception to this case is when the initiating small break is the high head safety injection line. Further propagation is not permitted for this case;
3. Damage to the high head safety injection lines connected to the other leg of the affected loop or to the other loops is prevented; and
4. Propagation of the break to high head safety injection line connected to affected leg is prevented if the line break results in a loss of core cooling capability due to a spilling injection line.

The NRC issued IE Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," on June 22, 1988 and Supplements 1, 2 and 3 to this bulletin on June 24, 1988, August 4, 1988, and April 11, 1989, respectively. The purpose of this bulletin and supplements was to request that Licensees (1) review their reactor coolant systems (RCSs) to identify any connected, unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses and (2) take action, where such piping identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. The industry basis

[2] The Containment is here defined as the Containment vessel and penetrations, and the steam generator shell, the steam generator steam side instrumentation connection, the steam, feedwater, blowdown and steam generator drain pipes within the Containment structure.

for issuing this bulletin was the circumferential cracking of a short, unisolable section of emergency core cooling system (ECCS) piping connected to the cold leg of loop B in the RCS of the Farley Nuclear Plant. The root cause for this issue identified cooler water leaking by valves and into the RCS, thereby creating thermal stratification within the connecting piping and resultant piping fatigue failure. Further industry occurrences were identified in Supplements 1, 2, and 3. Initial reviews, inspections, and evaluations of the McGuire Nuclear Station applicable systems indicated these systems were not susceptible to this phenomenon (letters from H.B. Tucker to the NRC, dated September 9, 1988, December 28, 1988, and October 10, 1989). Temperature monitoring performed on selected safety injection piping indicated thermal stratification on certain class 2 sections (letter from M.S. Tuckman to the NRC dated January 2, 1991), resulting in additional monitoring and analysis. Further responses detailing evaluation of more recently acquired inspection data and evaluation methods, along with programmatic enhancements were submitted to the NRC, resulting in issue closure by the letter from the NRC to M.S. Tuckman, dated September 17, 1991.

3.6.2.1.1 Postulated Piping Break Locations and Orientations

Reference 1 defines the original basis for postulating pipe breaks in the Reactor Coolant System Primary Loop. References 3 and 4 provide the basis for eliminating previously postulated reactor coolant loop pipe breaks, with the exception of those breaks at branch connections, from consideration in the plant structural design basis by implementing leak-before-break (LBB) methodology. See Table 3-21 and Figure 3-6.

3.6.2.1.2 Postulated Piping Break Sizes

For a circumferential break, the break area is the cross-sectional area of the pipe at the break location, unless pipe displacement is shown to be less by analysis, experiment or physical restraint.

For a longitudinal break, a break area less than the cross-sectional area of the pipe may be assumed when analytically or experimentally substantiated. In the absence of this data, the break area shall be assumed to be the cross-sectional area of the pipe and the break length shall be assumed to be two pipe diameters.

3.6.2.1.3 Line Size Considerations for Postulated Piping Breaks

Branch lines connected to the Reactor Coolant System are defined as “large” for the purpose of this criteria as having an inside diameter greater than 6 inches up to the largest connecting line. Where postulated, pipe break of these lines results in a rapid blowdown of the Reactor Coolant System and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

3.6.2.2 General Design Criteria for Postulated Piping Breaks Other Than Reactor Coolant System

1. Station design considers and accommodates the effects of postulated pipe breaks with respect to pipe whip, jet impingement and resulting reactive forces for piping both inside and outside Containment. The analytical method utilized to assure that concurrent single active component failure and pipe break effects do not jeopardize the safe shutdown of the reactor are outline in Figure 3-7.
2. Station general arrangement and layout design of high energy systems utilizes the possible combination of physical separation, pipe bends, pipe whip restraints and encased or

jacketed piping for the most practical design of the station. These possible design combinations decrease postulated piping break consequences to minimum and acceptable levels. In all cases, the design is of a nature to mitigate the consequences of the break so that the reactor can be shutdown safely and maintained in a safety shutdown condition.

3. The environmental effects of pressure, temperature and flooding are controlled to acceptable levels utilizing restraints, level alarms and/or other warning devices, vent openings, etc.

4. Plant Operating Conditions

   a. Power Level - At the time of the postulated pipe break, the plant is assumed to be in the normal mode of plant operation, in which the piping under investigation experiences the maximum conditions of pressure and temperature. In cases where this mode is full power operation, the power level assumed is that assumed in the evaluation of the loss-of-coolant accident, steamline break accident, or feedwater line break accident, in Chapter 15 of the safety analysis report.

   b. Offsite Power - If the pipe break results in a loss-of-coolant accident, steam line break accident, or feedwater line break accident, a loss of offsite power is assumed to occur subsequent to the pipe rupture.

   c. Seismic Loadings - equivalent to either the Safe Shutdown Earthquake (SSE) or the Operating Basis Earthquake (OBE), as appropriate, will be used in the analysis of piping, equipment, protective devices, etc.

5. Consideration is given to the potential for a random single failure of an active component subsequent to the postulated pipe rupture. Where the postulated piping break is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system, i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping rupture, single failures of components in the other train or trains of the system only are not assumed, provided the system is designed to seismic Category 1 standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and in-service inspection standards appropriate for nuclear safety systems.

6. In the event of a postulated break in the piping in one unit, safe reactor shutdown of the affected unit cannot preclude the capability for safe shutdown of the reactor of the unaffected unit(s).

7. Containment structural integrity is maintained by limiting the combination of break sizes and types to the design basis capability (i.e., temperature, pressure, and leakage rate) of the containment.

8. For any postulated pipe break the structural integrity of the containment structure shall be maintained. In addition, for those postulated breaks classified as a loss of reactor coolant the design leak tightness of the containment fission product barrier shall be maintained.

9. The conditions within the control room or any other location where manual action is required to assure safe shutdown to the cold condition is such as to assure habitability and comply with the requirements of General Design Criterion 19.

10. A whipping pipe or jet is assumed not to cause failure of other pipes of equal or greater size and equal or greater thickness. Smaller and thinner pipes are assumed to encounter unacceptable damage upon impact. A whipping pipe or jet is considered capable of developing through-wall leakage cracks in larger nominal pipe sizes with thinner wall

thicknesses, except where experimental or analytical data for the expected range of impact energies demonstrate the capability to withstand the impact without failure.

11. Piping Breaks Within The LOCA Boundary
    a. All LOCA breaks are allowed to damage any non-LOCA line except essential systems, and steam and feedwater lines.
    b. Pipe breaks within the LOCA boundary are allowed to damage ECCS lines connecting to the ruptured line, providing the ECCS flow to other loops is maintained.
    c. For breaks in 6" nominal or larger piping, propagation of the break in the affected loop is not permitted if the resultant break area is more than 120% of the originating break area. If the originating break is a Reactor Coolant System main loop break, propagation is permitted to occur but must not exceed the design basis for calculating containment and subcompartment pressure, loop hydraulic forces, reactor internals reaction loads, primary equipment support loads, or ECCS performance. Propagation to any other loop is not permitted in any case.
    d. Pipe breaks within the LOCA boundary that are equal to or less than 4" nominal pipe size must meet the following criteria:
        1) Break propagation to the other leg of the affected loop and to other loops must be prevented.
        2) Propagation of the break in the affected leg is permitted but is limited to a total break area of 12.5 square inches (4-inch inside diameter). The exception to this case is when the initiating small break is the high head safety injection line. Further propagation is not permitted for this case.
        3) Damage to the high head safety injection lines connected to the other leg of the affected loop or to the other loops is prevented.
        4) Propagation of the break to high head safety injection line connected to the affected leg is prevented if the line break results in a loss of core cooling capability due to a spilling injection line.
12. Piping Breaks Outside the LOCA Boundary (Non-LOCA)
    a. A pipe break which is not a loss-of-reactor-coolant accident cannot cause a loss-of-reactor-coolant accident or steam or feedwater line break.
    b. All non-LOCA breaks (except steam and feedwater line breaks) are allowed to damage the non-LOCA portion of a single train of an ESF system, provided that unit shutdown can be achieved, when considering a single active failure.
    c. All non-LOCA breaks (excluding steam and feedwater line breaks) are allowed to damage any non-LOCA, non-essential lines (except steam and feedwater lines), provided that until shutdown can be achieved assuming any small active failure.
    d. A pipe break in one train of a redundant essential system or a pipe break which damages one train of a redundant essential system cannot result in damage to the opposite train of that system or any other essential system, considering a single acting failure.
    e. A pipe break in a non-seismic system (Duke System Piping Class D,E,G,H) cannot result in damage to an essential system necessary for the mitigation of the postulated pipe break.

13. Piping Breaks in Steam and Feedwater Lines

    a. Steam and feedwater line breaks are allowed to damage steam and feedwater lines, respectively, of the same steam generator, provided that the aggregate break size does not exceed the applicable maximum break size considered in the safety analysis.

    b. Steam and feedwater line breaks can damage any non-LOCA lines except required essential system lines.

14. Failure of any structure caused by the postulated line break is not allowed to adversely affect the mitigation of the consequences of the break nor the capability to safely shut down and maintain the reactor in a safe shutdown condition.

15. Loss of required redundancy in the protective system, engineered safety feature equipment, cable penetrations or their interconnecting cables due to postulated line breaks is not allowed to adversely affect the mitigation of the consequences of the break nor the capability to safely shutdown and maintain the reactor in a safe shutdown condition.

16. Loss of ability to cope with subsequent line ruptures due to an initial postulated line rupture is not allowed in electrical components.

17. Internal fluid energy level associated with the pipe break may take into account flow restrictors.

18. Environmental operability is assured for all electrical equipment in the immediate piping break area by the equipment specification requirements based on conservative design conditions.

19. Duke's Nuclear Generation Department prepares adequate emergency operating procedures that would be followed after a postulated piping break for high energy systems as required.

3.6.2.2.1 Postulated Piping Break Locations and Orientations

Systems identified as containing high energy or moderate energy piping are examined by a detailed design drawing review for a postulated pipe break or through-wall cracks as defined herein along their entire routing regardless of Code class. Systems analyzed for consequences of postulated piping breaks are listed in Table 3-18 and Table 3-19.

The requirement for arbitrary intermediate pipe breaks was eliminated by Reference 6. NRC Generic Letter 87-11 (Reference 7) was subsequently issued in which the NRC described conditions in which the dynamic and environmental effects resulting from arbitrary intermediate pipe ruptures may be eliminated from design basis without prior NRC approval.

1. Breaks in Duke Class A piping are postulated at the following locations: (See Table 3-5 for class correlations).

    a. The terminal ends of the pressurized portions of the run.

    b. At intermediate locations selected by either one of the following methods:

        1) At each location of potential high stress and fatigue such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments, or

        2) At all intermediate locations between terminal ends where the following stress and fatigue limits are exceeded,

            a) The maximum stress range should not exceed 2.4 Sm except as noted below.

b) The maximum stress range between any two load sets (including the zero load set) should be calculated by Eq. (10) in Paragraph NB-3653, ASME Code, Section III, for normal and upset plant conditions and an operating basis earthquake (OBE) event transient.

If the calculated maximum stress range of Eq. (10) exceeds the limit (2.4 Sm) but is not greater than 3 Sm, the limit of U < 0.1 should be met.

If the calculated maximum stress range of Eq. (10) exceeds 3 Sm, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 should not exceed 2.4 Sm and the limit of U < 0.1.

where:

S_n	=	Primary-plus-secondary stress-intensity range, as calculated from Equation (1) in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.
S_m	=	Allowable design stress-intensity value, as defined in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.
U	=	The cumulative usage factor, as calculated in accordance with Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

The requirement for arbitrary intermediate pipe breaks was eliminated by Reference 6.

2. Breaks in Duke Class B, C and D piping are postulated at the following locations: (see Table 3-5 for class correlations).

   a. The terminal ends of the pressurized portions of the run.

   b. At intermediate locations selected by either one of the following methods:

      1) At each location potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges and welded attachments, or

      2) At all locations where the stress, S, exceeds 0.8 ($1.2S_h + S_A$),

      where:

S	=	Stresses under the combination of loadings associated with the normal and upset plant condition loadings, as calculated from the sum of equations (9) and (10) in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.
S_h	=	Basic material allowable stress at maximum (hot) temperature from the allowable stress tables in Appendix 1 of the ASME Boiler and Pressure Vessel Code, Section III.
S_A	=	Allowable stress range for expansion stresses, as defined in Subarticle NC-3600 of the AMSE Boiler and Pressure Vessel Code, Section III.

3. Breaks in Duke Class E, F, G, and H piping are postulated at the following locations: (see FSAR Table 3-5 for class correlations).

   a. The terminal ends of the pressurized portion of the run.

   b. At intermediate locations by selecting one of the following methods.

      1) For Class E, F, G and H Piping:

At each location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments; or

2) For Class F Piping:

At all locations where the stress, S, exceeds 0.8 (1.2 S_h + S_A),

where:

S	=	Stresses under normal and upset plant loadings
S_h	=	Basic material allowable stress at maximum (hot) temperature, per ANSI B31.1.0.
S_A	=	Allowable stress range for expansion stresses, per ANSI B31.1.0.

For Cases 1, 2, 3b above - longitudinal and circumferential breaks shall be postulated, but not concurrently, unless from a detailed stress analysis (e.g., finite element analysis) the state of stress can identify the most probable type. If the primary plus secondary stress in the axial direction is found to be at least 1.5 times that in the circumferential direction for the most severe normal and upset load combination transients, then only a circumferential break need be postulated. Conversely, if the primary plus secondary stress in the circumferential direction is found to be at least 1.5 times that in the axial direction for the most severe normal and upset transients, then only a longitudinal break need be postulated. At terminal ends where piping has no longitudinal welds, no longitudinal breaks are postulated.

Where break locations are postulated at fittings without the benefit of a detailed stress calculation, breaks should be assumed to occur at each pipe-to-fittings weld. If a detailed stress analyses or tests are performed, the maximum stressed location in the fittings may be selected as the break location.

A circumferential break results in pipe severance with full separation except as limited by structural design features. The break shall be assumed perpendicular to the longitudinal axis of the pipe, and the break area assumed to be the cross-sectional flow area of the pipe at the break location. The break discharge coefficient used shall be substantiated analytically or experimentally. In the absence of this data, the discharge coefficient shall be assumed to be 1.0.

A longitudinal break results in an axial split without severance. The split shall be assumed to be orientated at any point about the circumference of the pipe, or alternatively at the point of highest stress as justified by detailed stress analyses. For the purpose of design, the longitudinal break shall be assumed to be circular or ellipical (2D x 1/2D) in shape, with an area equal to the largest piping cross-sectional flow area at the point of the break and have a discharge coefficient of 1.0. Any other values used for the area, diameter and discharge coefficient associated with a longitudinal break shall be verified by test data which defines the limiting break geometry.

For the purpose of analysis, circumferential and longitudinal breaks are assumed to reach full size within one (1) millisecond after break initiation unless otherwise analytically or experimentally substantiated.

Through-wall cracks are postulated in moderate-energy piping systems outside containment having a nominal diameter greater than one (1) inch. Cracks are not postulated in piping that contains no pressurization equipment; i.e., systems without pumps, pressurizing tanks, boilers, etc., and which operate only from gravity, flow or storage tank head. Also, cracks are not postulated in portions of Duke Class B, C, D, or F piping where the stresses are less than 0.4

(1.2 S_h + S_A). Through-wall cracks in moderate-energy piping systems are not postulated inside containment.

Terminal ends are considered at piping originating at structures or components (such as vessel and equipment nozzles and structural piping anchors) that act as rigid constraint to the piping thermal expansion. Typically, the anchors assumed for the piping code stress analysis would be terminal ends. The branch connection to the main run is one of the terminal ends of a branch run, except where the branch run may be classified as part of a main run.

Crack openings shall be assumed as a circular orifice of cross-sectional flow area equal to that of a rectangular one-half diameter in length and one-half pipe wall thickness in width. The orifice shall be assumed to be orientated at any point about the circumference of the pipe.

Pipe sizes and locations of postulated piping breaks in Duke Class A (ASME Class I) piping other than the reactor coolant loop are presented in FSAR Appendix 3P and Report No. MDS/PDG-77-1. The arbitrary intermediate breaks described in these documents were subsequently eliminated by reference 6.

Table 3-20 identifies differences between Duke criteria and NRC requirements contained in Branch Technical Position APCSB 3-1 (November 1975) and Regulatory Guide 1.46 (May 1973).

The analytical interface between Duke and Westinghouse for RCS pressure boundary is fully described in detail in Duke's ASME Class I piping design specification. The interface occurs at the weld end of all RC System branch nozzles. Analytical interfaces are defined to the extent that both Duke and Westinghouse are able to perform independent analysis without compromising allowable stress limits at the branch line connection.

3.6.2.2.2 Postulated Piping Break Sizes

Double ended and equivalent longitudinal pipe break areas are based on the nominal inside diameter (ID) of the piping system, i.e.,

$$A = \frac{\pi}{4}(ID)^2$$

Through-wall crack pipe break areas are based on length equal to one-half the nominal outside diameter (1/2 ID) and a width equal to one-half the minimum wall thickness (1/2 t) of the system piping materials, i.e.,

$$A = \frac{ID}{4} t$$

3.6.2.2.3 Line Size Considerations for Postulated Piping Breaks

For high energy systems, piping larger than 1" nominal pipe size (NPS) is reviewed for the consequences of a double ended break.

For high energy systems, piping 4" NPS and larger is reviewed for the consequences of double ended and equivalent area longitudinal breaks.

For moderate energy system, piping larger than 1" NPS is reviewed for the consequence of through-wall cracks.

3.6.2.3 Analysis and Results

The results of analyses of failure in fluid systems occurring inside containment for McGuire 1 are presented in FSAR Appendix 3P except Main Steam, Feedwater and Auxiliary Feedwater Systems which are included in Appendix C of Report No. MDS/PDG-77-1. The results of analysis of failure in fluid systems occurring outside containment are presented in the report "Evaluation of the Effects of Postulated Pipe Failures Outside Containment for McGuire Nuclear Station", Report No. MDS/PDG-77-1. These analyses are performed after design completion in accordance with the criteria presented in this section. The methods and results associated with the protection against the dynamic effects due to the postulated rupture of piping for McGuire Unit 2 are very similar to the methods and results associated with McGuire Unit 1, but they are not exactly the same. Specifically, field walkdowns were used more extensively to supplement drawing review on McGuire 2. Also on McGuire 2, the use of Table 3P-4A, "Devices Requiring Protection from Pipe Rupture", was more extensively supplemented by a special systems evaluation of each cable/impulse line interaction for acceptability .

Since the differences in approach produced similar (but not exactly the same) results for Unit 2, no specific data on Unit 2 is provided in Appendix 3P. The arbitrary intermediate breaks described in these documents were subsequently eliminated by reference 6.

3.6.3 Design Loading Combinations

3.6.3.1 Reactor Coolant System Design Loading Combinations

As described in Section 5.2, the dynamic forces associated with postulated reactor piping branch connection rupture are considered in the design of supports and restraints in order to assure continued integrity of vital components and Engineering Safety Features.

Reaction forces used in the design of supports and restraints are computed on the basis of an assumed break equal to the cross sectional flow area of the pipe.

The design stress limits applicable to postulated reactor coolant piping breaks and supports are discussed in WCAP-8172-A and are listed in Table 5-4.

3.6.3.2 All Other Mechanical Piping Systems Design Loading Combinations

Since all locations of consequences are reviewed and as detailed stress analysis information is extremely extensive, stress analysis information is only reviewed for special identified problem areas which might require additional restraints.

These additional consequential piping breaks posing safety-related problems to structures, systems or components in the immediate area are either restrained to mitigate the consequences of the break or reviewed in detail against existing stress analysis. If the stress allowables discussed in Section 3.6.2.2.1 are not exceeded, then the break is not considered to occur.

Loading and stress criteria for pipe whip restraints is fully described in Section 3.9. Postulated pipe breaks are considered a faulted condition with respect to the pipe whip restraint design and allowable restraint stresses.

3.6.4 Dynamic Analysis

3.6.4.1 Reactor Coolant System Dynamic Analysis

3.6.4.1.1 Westinghouse Methodology

This section summarizes the dynamic analysis as it applies to the LOCA resulting from the postulated design basis pipe breaks at reactor coolant piping branch connections. Further discussion of the dynamic analysis methods used to verify the design adequacy of the reactor coolant loop piping, equipment and supports is given in WCAP-8172, as it pertains to postulated breaks at branch connections.

The particular arrangement of the Reactor Coolant System for the McGuire Nuclear Station is accurately modeled by the standard layout used in WCAP-8172 and the postulated branch connection break locations do not change from those presented in WCAP-8172.

In addition, an analysis will be performed to demonstrate that at each postulated branch connection break location the motion of the pipe ends is limited so as to preclude unacceptable damage due to the effects of pipe whip or large motion of any major components. The loads employed in the analysis will be based on full pipe areas discharge except where limited by major structures. The effects of jet discharges will be analyzed to demonstrate that any structure, system or component required to safety shutdown the reactor or mitigate the consequences of an accident will not be impaired.

The dynamic analysis of the Reactor Coolant System employs displacement method, lumped parameter, stiffness matrix formulation and assumes that all components behave in a linear elastic manner.

The analysis is performed on integrated analytical models including the steam generator and reactor coolant pump, the associated supports, and the attached piping. An elastic-dynamic three-dimensional model of the Reactor Coolant System constructed. The boundary of the analytical model is, in general, the foundation concrete/support structure interface. The anticipated deformation of the reinforced concrete foundation supports is considered where applicable to the Reactor Coolant System model. The mathematical model is shown in Figure 5-8.

The steps in the analytical method are:

1. The initial deflected position of the Reactor Coolant System model is defined by applying the general pressure analysis;
2. Natural frequencies and normal modes of the broken branch connection are determined;
3. The initial deflection, natural frequencies, normal modes, and time-history forcing functions are used to determine the time-history dynamic deflection response of the lumped mass representation of the Reactor Coolant System;
4. The forces imposed upon the supports by the loop are obtained by multiplying the support stiffness matrix and the time-history of displacement vector at the support point; and
5. The time-history dynamic deflection at mass point are treated as an imposed deflection condition on the ruptured loop branch connection Reactor Coolant System model and internal forces, deflections, and stresses at each end of the members of the reactor coolant piping system are computed.

The results are used to verify the adequacy of the restraints at the branch connections. The general dynamic solution process is shown in Figure 3-126.

In order to determine the thrust and reactive force loads to be applied to the Reactor Coolant System during the postulated LOCA, it is necessary to have a detailed model of the hydraulic transient. Hydraulic forcing functions are calculated for the reactor coolant loops as a result of a postulated loss of coolant accident (LOCA) caused by a postulated branch connection break. These forces result from the transient flow and pressure histories in the Reactor Coolant System. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flow rates, and other hydraulic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates and is to calculate the time history of forces at appropriate locations in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire reactor coolant system. Key parameters calculated by the hydraulic model are pressure, mass flow rate, and density. These are supplied to the thrust calculation, together with appropriate station layout information to determine the concentrated time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account only in the evaluation of the local fluid conditions in the hydraulic model.

The blowdown hydraulic analysis is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the reactor system. The SATAN-V code was developed with a capability to provide this information.

The SATAN-V computer code performs a comprehensive space-time dependent analysis of a loss of coolant accident and is designed to treat all phases of the blowdown. The stages are: (i) a subcooled stage where the rapidly changing pressure gradients in the subcooled fluid exert an influence upon the Reactor Coolant System internals and support structures; and (ii) a two phase depressurized stage, and (iii) the saturated stage.

The code employs a one-dimensional analysis in which the entire Reactor Coolant System is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed to each element. Pump characteristics, pump coastdown and cavitation, core and steam generator heat transfer including the W-3 DNB correlation in addition to the reactor kinetics are incorporated in the code.

The blowdown hydraulic loads on primary loop components are computed from fluid transient information calculated using the following time dependent forcing function:

$$F = 144A[(P - 14.7) + \frac{(\dot{m}^2)}{pgA_m{}^2 144}]$$

which includes both the static and dynamic effects. The symbols and units are:

F = Force, Lb_f

A = Aperture area, Ft^2

P = System Pressure, PSIA

$\dot{m}$ = Mass flow rate, Lb_m/Sec

p = Density, Lb_m/Ft^3

g = Gravitational Constant = 32.174 Ft.sec^2

A_m= Mass Flow Area, Ft^2

The main Reactor Coolant System is represented by a similar nodal system as employed in the blowdown analysis. The entire loop layout is described in a global coordinate system. Each node is fully described by: (i) blowdown hydraulic information and (ii) the orientation of the streamlines of the force nodes in the system, which includes flow area, node numbers and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume, with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis. Further details are given in WCAP-8172.

The dynamic analysis described above for the reactor coolant loop piping has been completed and the results verify that the break locations and type postulated in WCAP-8082/8172 are the only ones that are required to be postulated for McGuire.

3.6.4.1.2 Steam Generator Replacement Methodology

This section summarizes the dynamic analysis as it applies to the LOCA resulting from the postulated design basis pipe breaks at the main reactor coolant branch line interconnections. The purpose of the analysis is to develop the thrust and reactive force loads to be applied to the Reactor Coolant System during the postulated LOCA.

The analyses are performed on an elastic three dimensional finite element model of the Reactor Coolant System. The model includes the replacement steam generators, reactor vessel, reactor coolant pumps, associated equipment supports and the attached piping. The NSSS piping, equipment, and equipment supports are coupled to the concrete Reactor Building interior structure finite element model (see Figures 3-122 through Figure 3-125).

The steps in the analytical method are:

1. The initial deflection, natural frequencies, normal modes, and time-history forcing functions are used to determine the time-history dynamic response of the mathematical representation of the Reactor Coolant System;
2. The forces imposed on the supports by the loop are obtained by multiplying the support stiffness matrix and the time-history of the displacement vector at the support points; and
3. The peak deflections at mass points are treated as an imposed deflection condition on the ruptured loop branch connection. Reactor Coolant System model internal forces, deflections, and stresses at each end of the members of the reactor coolant piping system are computed.

In order to determine the thrust and reactive force loads to be applied to the Reactor Coolant System during the postulated LOCA, it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the reactor coolant loops as a result of a postulated loss of coolant accident for a branch connection break. These forces result from the transient flow and pressure histories in the Reactor Coolant System. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flow rates, and other hydraulic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, to calculate the time history of forces at appropriate locations in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire reactor coolant system. Key parameters calculated by the hydraulic model are pressure, mass flow rate, and density. These are supplied to the thrust calculation, together with appropriate station layout information to determine the concentrated time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure, momentum flux, inertia, and gravitational terms are taken into account.

The blowdown hydraulic analysis is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics, and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the reactor system. The CRAFT2 (Reference 5 in Section 3.6.6) code was developed with a capability to provide this information.

The CRAFT2 computer code performs a comprehensive space-time dependent analysis of a loss of coolant accident and is designed to treat all phases of the blowdown. The stages are: (i) a subcooled stage where the rapidly changing pressure gradients in the subcooled fluid exert an influence on the Reactor Coolant System internals and support structures; (ii) a two phase depressurization stage; and (iii) the saturated stage. The code employs a one-dimensional analysis in which the entire Reactor Coolant System is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coastdown and cavitation, and core and steam generator heat transfer, in addition to the reactor kinematics, are incorporated in the code. The CRAFT2 computer code also computes the transient (blowdown) loads resulting from a LOCA.

The blowdown hydraulic loads on primary loop components are computed from the fluid transient information calculated using the following time dependent forcing function:

$$F = 144A\left\{(P - 14.7) + \left(\frac{m^2}{\rho_f g_c A^2 144}\right)\beta\right\}$$

$$\text{where} \quad \beta = \frac{(1-x)^2}{(1-\alpha)} + \frac{X^2}{\alpha}\frac{\rho_f}{\rho_g}$$

which includes both the static and dynamic effects. The symbols and units are:

F = Force, Lb_f

A = Aperture area, Ft^2

P = System pressure, psia

m = Mass flow rate, Lb_m/Sec

g_c = Gravitational Constant, = 32.174 $\frac{Lb_m Ft}{Lb_f Sec^2}$

x = Quality

α = Void fraction

ρ_f = Saturated liquid density, Lb_m/Ft^3

ρ_g = Saturated vapor density, Lb_m/Ft^3

The main Reactor Coolant System is represented by a similar nodal system as employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each node is fully described by: (i) blowdown hydraulic information and (ii) the orientation of the streamlines of the force nodes in the system, which includes flow areas, and projection coefficents along the three axes of the global coordinate system. Each node is modeled as a separate control volume, with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

3.6.4.2 All Other Mechanical Piping Systems Dynamic Analysis

Effects of pipe break are conservatively evaluated to determine the need for pipe whip restraints. Energy of the whipping pipe, its effect on targets, jet impingement forces and temperatures, and compartment pressurization and temperature effects establish the need and requirement for pipe whip restraints.

Dynamic analysis of Category 1 piping and supports is or is not performed depending on the conservatively determined consequences of the break. The need for dynamic analysis depends on the need for fully identifying the response of the system. The purpose of the analysis when required is to prove that the consequences of the break does not prevent mitigation of the break nor present the safe and continued shutdown of the reactor.

Dynamic analysis methods have been developed. These methods consider the energy of the whipping pipe using conservative forcing functions, gaps between pipe and restraint, and energy absorbers designed to absorb the major portion of the whipping pipe energy. The design of energy absorbers is based on test results under dynamic loading conditions. The response of the system with respect to its effect on Category 1 systems and equipment has been determined by analysis using a computer program such as PWHIP or equivalent. PWHIP is described in Section 3.9.2.3 of this SAR.

The dynamic analysis model used was one or more of three acceptable models specified by the NRC. Any one of these models was used depending upon the particular piping system being analyzed. A lumped-parameter model has been formulated and programmed, and is available for use should this option be elected. This model consists of lumped masses interconnected by bending stiffness springs. Modulus of elasticity for the bending stiffness springs is represented by a bilinear stress-strain curve. A suddenly applied load of constant value is currently programmed into the model with the constant value determiend as outlined by the NRC (i.e., F = KpA). A time-history numerical integration is performed using the Runge-Kutta-Gill technique. Newton's Second Law of Motion is applied to each of the lumped masses using the shear forces to accelerate the masses. From the accelerations, velocities are determined, and in turn displacements, elastic axis slopes, bending moments, and new shear forces are also determined. Extension of the model to include interaction with pipe whip restraints was accomplished once characteristics of the restraints were finalized. Restrain loadings were then determined.

Associated jet impingement forces on an object are treated as a suddenly applied load constant value and not a varying function of time.

In piping systems other than the Reactor Primary Coolant System the blowdown forces may be calculated by the following equation:

$$T = MV_E + P_E A_E$$

Any other method used for determining blowdown forces or thrust coefficients was based on justifiable analytical and/or experimental data such as the work of Henry and Fauske and Moody.

The above equation is applicable to all fluid flow but can be simplified for special conditions as follows:

1. Subcooled water:
   a. Temperature <212°F

$$T = \text{lesser of } MV_E = \frac{2P_R\ A_E}{1+K} \text{ or } MV_E = \frac{W^2 v}{g\ A}$$

2. Subcooled water:
   a. Temperature <212°F

$$T = \text{lesser of } MV_E = \frac{2P_R A_E}{1+K} \text{ or } MV_E = \frac{W^2 v}{g\ A}$$

3. Water - Steam mixture, low quality:

$$T = \leq \frac{2P_R A_E}{1+K}$$

4. Water - Steam mixture, high quality or superheated steam:

$$T = MV_E + P_E A_E$$

The flow is assumed to choked at the break area based on isentropic expansion from reservoir maximum operating condition for K = O. Where K ≠ O, Fanno Lines are used to determine flow conditions at exit or break location. Fluid properties are based Homogeneous Equilibrium Model.

The sonic velocity, mass flow rate, and thrust is calculated using SONVEL, which is a Fortran IV program written to solve, by iteration, the following equations. These equations are based on sonic flow through a convergent isentropic nozzle. Sonic velocity is calculated as follows:

$$V_R = 0(\text{Assumed})$$

$$V_E = 223.7(h_R - h_E)^{1/2}$$

$$V_E = 12 v_E [(\frac{\Delta P}{\Delta v})_E\, g]^{1/2}$$

Also,

$$S_R = S_E$$

where subscript R denotes reservoir conditions and subscript E denotes exit (or break) location.

A = Break area (in^2)

g = Gravitational constant (ft/sec^2)

h = Specific enthalpy (BTU/#)

K = Flow resistance coefficient based on flow velocity at exit

M $= \frac{W}{g}$

P = Static pressure (PSIG)

T = Thrust (Lb)

v = Specific volume ($ft^3/\#$)

V = Flow velocity (ft/sec)

W = Mass flow rate (#/sec)

$(\frac{\Delta P}{\Delta v})_E$ = Rate of change of pressure with specific volume at point "E" at constant entropy

The TMD Code is utilized in developing pressure transients for postulated piping breaks within containment and the main steam/feedwater penetration rooms (doghouses).

Assumption and considerations utilized in the analysis as applicable are:

1. The total volume being analyzed is subdivided into smaller compartments as required, and the time-dependent pressure rise for each individual subcompartment is assumed to be equal throughout;
2. Frictional effects, turning losses, vent losses, etc., are considered for flow through each subcompartment;
3. Condensation effects due to heat sinks are considered negligible for conservatism;
4. Calculations for mass flows from pipe ruptures do not consider frictional effects of piping; and
5. Two-phase mass flow (liquid and vapor phase) is assumed to be homogeneous.

The assumptions listed below are applied to the pressurization calculations for Auxiliary Building pipe ruptures.

1. A homogeneous mixture of air and steam or gas in each compartment, and thermodynamic equilibrium, are attained instantaneously.
2. Homogeneous or separated 2-phase flow models are used. A break discharge coefficient of 1.0 is used for all break sizes in blowdown analyses in the source compartment. The orifice discharge coefficient between compartments is assumed to be 0.6 unless other values can be justified, and is used for the determination of pressure differentials in the source compartment.
3. Potential energy and kinetic energy are negligible, and flow work is recovered and stored as integral energy.
4. Passive and active heat sinks are considered when justified.
5. Initial state of the contents of both the compartment and the pipe are known. Final state is saturated or super-heated vapor with liquid phase, if existing, at saturated or subcooled conditions.

A division of the building volume into compartments allows computation by computer program of the pressure buildup in each compartment due to the effects of postulated pipe break. The pressure at any point in time is calculated by obtaining the simultaneous solution of the mass balance, energy balance, and equations of state for each volume considered. This volume is either a total compartment volume or an arbitrary control volume assumed for computational purposes.

The results of the pipe rupture analysis for category 1 piping systems other than the reactor coolant loop are presented in Appendix 3P and Report No. MDS/PDG-77-1 for inside and outside containment, respectively.

3.6.4.3 Structural Analysis of Postulated Piping Breaks

Evaluation utilized to demonstrate the adequacy of or in the design of Category 1 structures subject to loadings of postulated piping breaks include:

1. Method of evaluating stresses;
2. Allowable design stresses and/or strains;
3. Load factors and combinations;
4. Design loads including pressure and temperature transients; and
5. Load reversal effects.

Details of the structural analysis involving the above combinations are discussed in Section 3.8.1.

3.6.5 Protective Measures

3.6.5.1 Reactor Coolant System

The fluid discharged from postulated branch connection breaks will produce reaction and thrust forces in branch line piping. The effects of these loadings are considered in assuring the continued integrity of the vital components and the engineered safety features.

To accomplish this in the design, a combination of component restraints, barriers, and layout are utilized to ensure that for a loss of coolant or steam-feedwater line break, propagation of damage from the original event is limited, and the components as needed, are protected and available.

3.6.5.1.1 Postulated Pipe Break Restraint Design Criteria for Reactor Coolant System

Piping connected to the Reactor Coolant System (six inches nominal or larger) and all connecting piping out to the LOCA boundary valve (Figure 3-5) is restrained to meet the following criteria.

1. Propagation of the break to the unaffected loops is prevented to assure the delivery capacity of the accumulators and low head pumps;
2. Propagation of the break in the affected loop is permitted to occur but must not exceed 20 percent of the area of the line which initially failed. This criterion is voluntarily applied so as not to substantially increase the severity of the loss of coolant; and

3. Where restraints on the lines are necessary in order to prevent impact on and subsequent damage to the neighboring equipment or piping, restraint type and spacing is chosen such that a plastic hinge on the pipe at the two support points closest to the break is not formed.

3.6.5.1.2 Protective Provisions for Vital Equipment

In addition to pipe restraints, barriers and layout are used to provide protection from pipe whip, blowdown jet and reactive forces for postulated pipe breaks.

Some of the barriers utilized for protection against pipe whip are the following. The polar crane wall serves as a barrier between the reactor coolant loops and the Containment liner. In addition, the refueling cavity walls, various structural beams, the operating floor, and the crane wall enclose each reactor coolant loop into a separate compartment; thereby preventing an accident, which may occur in any loop branch connection, from affecting another loop or the Containment. The portion of the steam and feedwater lines within the Containment have been routed behind barriers which separate these lines from all reactor coolant piping. The barriers described above will withstand loadings caused by jet forces, and pipe whip impact forces.

Other than Emergency Core Cooling System lines, which must circulate cooling water to the vessel, Engineered Safety Features are located outside the crane wall. The Emergency Core Cooling System lines which penetrate the crane wall are routed around and outside the crane wall to penetrate the crane wall in the vicinity of the loop to which they are attached.

In reviewing the mechanical aspects of these lines, it has been demonstrated by Westinghouse Nuclear Energy System tests that lines hitting equal or larger size lines of same schedule do not cause failure of the line being hit, e.g., a one-inch line, should it fail, does not cause subsequent failure of a one-inch or larger size line. The reverse, however, is assumed to be probable, discharged through the 4" line, could break smaller size lines such as neighboring three-inch or two-inch lines. In this case, the total break area shall be less than 12.5 square inches.

Alternately, the layout is planned such that whipping of the two free sections cannot reach equipment or other pipes for which protection is required; plastic hinge formation can be allowed to form. As another alternative, barriers can be erected to prevent the whipping pipe from impacting on equipment or piping requiring protection. Finally, tests and/or analyses may be performed to demonstrate that the whipping pipe does not cause damage in excess of the acceptable limits.

Whipping in bending of a broken stainless steel pipe section such as used in the Reactor Coolant System does not cause this section to become a missile. This design basis has been demonstrated by performing bending tests on large and small diameter, heavy and thin walled stainless steel pipes.

3.6.5.1.3 Criteria for Separation of Redundant Features

There are no redundant features associated with reactor coolant piping system. Redundant features of other mechanical piping systems are discussed in Section 3.6.5.2.

3.6.5.1.4 Separation of Piping

The Reactor Coolant System is separated from other piping systems and components by barriers, as discussed in Section 3.6.5.1.2.

3.6.5.2 All Other Mechanical Piping Systems

Measures to protect against pipe whip, jet impingement and resulting reactive forces to meet established criteria outlined in Section 3.6.2.2 are as follows:

1. Separation and remote location of fluid system piping from essential structures and equipment.
2. Structural enclosure of the fluid system piping with access provided for inservice inspection; or, alternatively, enclosure of the essential equipment.
3. Provision of system-redundant design features separated, or otherwise protected, from the effects of the postulated pipe rupture; or additional protection features such as restraints and barriers.
4. Design of essential structures and equipment to withstand the effects of the postulated pipe rupture.
5. Addition of guard piping for the main purpose of diverting or restricting blowdown flow.
6. In areas where none of the above can be met, or where unacceptable, more severe problems may be created, augmented inservice inspection may be used on a case by case basis to reduce the probability of failure to acceptable levels, and not postulate the failure. The augmented inservice inspection is in accordance with the guidelines presented in NRC MEB Branch Position No. 4 "Augmented Inservice Inspection and Secondary Protective Measures."

Table 3-24 identifies all cases where exceptions to the criteria of Section 3.6 have been taken.

See Table 3-18 and Table 3-19 for protection methods on a system basis.

Curbs are provided around passageways to the Auxiliary Building from the Turbine Building. These curbs are of adequate height to contain flood water caused by the break of the main consenser circulating water expansion joint, or the most severe Condensate System failure for a minimum of fifteen minutes. There are no pipe or cable chase entrances below the elevation of the top of the curbs. This flooding condition does not render any essential system or component inoperable.

3.6.5.3 Main Steam and Feedwater System Design

Design of the Main Steam and Feedwater System meets the general design criteria established in Section 3.6.2.2; however, additional specific information as follows applies to these systems.

1. Main Steam Lines are 100 percent cold pulled so that as lines heat up, all thermal expansion stresses are essentially eliminated throughout the system;
2. Overpressure capability of the piping based on actual wall thicknesses is as follows:

	Normal Operating Pressure	Actual Code Pressure Capability	Margin
Main Steam:	985 psig	1250 psig	19%
Feedwater:	1165 psig	1420 psig	22%

3. Safety-related portions of the Main Steam and Feedwater Systems are Duke Class B. Class B system materials, fabrication, nondestructive examinations and documentation are in accordance with ASME III, Class 2;

4. Proper piping system erection and function of safety-related supports and restraints are assured by several means:
   a. The Construction Department Reviews erection against design drawings, and
   b. QA surveillance is conducted by the Hanger-Contractor to verify correct location, direction of movement and proper hardware installation.
   c. Compliance with requirements of IE Bulletin 79-14.
5. Figure 3-9, Figure 3-10, Figure 3-11, Figure 3-13, Figure 3-14, and Figure 3-15 show design routing of the Main Steam and Feedwater Systems outside Containment to the Turbine Building. Piping for these two systems is isolated from other safety-related systems, equipment and the Control Room by a missile barrier as can be noted from the above listed figures.
6. SM system piping was originally designed for arbitrary intermediate breaks (AIB) in accordance with the NRC Branch Technical Position B.1.c.(2)(b)(ii). Generic Letter 87-11 revised MEB 3-1 such that AIB's are no longer mentioned or defined. In conjunction with reanalyses of main steam piping and supports for replacement steam generators, AIB loads have been eliminated.

3.6.5.4 Control Room Protection from Postulated Piping Breaks

The Control Room is located on the top floor of the Auxiliary building and is bounded on the east and west sides by Electrical Penetration Rooms which contain no piping. The north side of the Control Room is bounded by the equipment area housing the Control Room ventilation equipment. Piping in this area consists of low pressure, low volume chilled water and low pressure, low volume heating steam. On the south side, the Control Room is bounded by the computer room and supporting areas. Piping in this area consists of sanitary waste and vent piping, drinking water and instrument air, none of which are high energy systems.

Immediately below the Control Room is the cable room containing no piping. The Control Room ceiling is bounded by a missile barrier roof as denoted on Figure 3-9 and Figure 3-10.

Penetrations into the Control Room area consists of ducts, electrical cables and instrument air only. Openings around such penetrations are sealed.

Doors entering the Control Room area have pressure seals. A slight positive pressure will be maintained in the Control Room in the event of ESF actuation by a pressurizing fan such that any leakage is out-leakage. Momentary loss of pressure is experienced during ingress and egress but air flow is outward, i.e., air flow is from the Control Room to adjacent areas.

Based on the above physical parameters, the Control Room is structurally isolated from areas containing high energy systems; therefore, there are no related consequences to the Control Room from the postulated break of high energy piping systems.

3.6.5.5 Postulated Pipe Break Restraint Design Criteria for All Other Mechanical Piping Systems

Postulated pipe break restraints are considered to consist of four basic components. These are: “Process Pipe,” “Energy Absorbing Device,” “Structure Extension” and the “Anchorage” as further explained below. Related to the pipe break restraint is the “Structure” to which it is attached, which is also further discussed below.

1. The process pipe is the pipe which is to be restrained and includes all integral attachments which are welded, cast, or forged directly to the pipe wall.

2. The energy absorbing device is a structural, mechanical, hydraulic cushion or other energy absorbing device or material which is designed to minimize the forces imposed on the structure. In some cases, the process pipe itself may be the energy absorbing device if it can be quantitatively demonstrated that the local deformations of the pipe account for that portion of the induced energy required to maintain forces on the structure and structure extension below their design limits. In some cases, no energy absorbing device is employed when the structure and structure extension is designed to withstand the entire resisting force imposed by the pipe break phenomenon.
3. The structure extension is the structural assemblage which connects the anchorage to the energy absorbing device or process pipe. In general, it may be considered as an extension of the anchorage. It is designated as a separate component because it can be an extensive structure and may be designed using different rules, applicable to the type of material used, than used for the anchorage. In rare cases, the process pipe or energy absorbing device may be directly connected to the anchorage in which case there is not structure extension.
4. The anchorage is that component which connects the structure extension to the structure. Generally for a concrete structure, it is an embedded plate. For a steel structure, it generally consists of welding or bolting.
5. The structure is that feature of the building which is a necessary part of the building but <u>also</u> is designed to accommodate the loads transmitted through the anchorage caused by the postulated pipe break. It may be either a steel or concrete component and is characterized by being relatively stiff and massive when compared to the pipe break restraint.
6. Allowable stresses used in the design of the pipe break restraint components are consistent with the component function. In general, the allowable stresses associated with the total reaction force, including impact, on the structure extension, anchorage and structure is taken as the minimum yield stress for structural steel and concrete embedments. For those situations where structure load limiting features cannot be provided to maintain the allowable stresses to within yield, plastic deformation in structural components is tolerated so long as the structure is capable of continuing its functional requirement after the deformation occurs. The upper design limit for pipe break restraint is 50 percent of the restraint material ultimate strain.

3.6.5.5.1 Typical Pipe Whip Restraints

A description of the typical pipe whip restraints and a summary of number and location of all pipe ruptures requiring restraints in each system is presented in 3.6.4.2 and Appendix 3P.

3.6.6 References

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2. “Multiply Aperture Small Break ECCS Analysis,” Letter from W.O. Parker, Jr. to Harold R. Denton, June 6, 1980.
3. Letter from H. B. Tucker to H. R. Denton, August 30, 1985, Transmitting Westinghouse Report *WCAP-10585*.
4. Letter from B.J. Youngblood to H.B. Tucker, May 8, 1986, Permitting Elimination of Large Primary Loop Pipe Rupture.

5. BWNT Computer Software Manual for Program NPGD-TM-287, "CRAFT2 Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant", Manual Revision AK, Software Versions 31.0HP, 32.1HP, and 34.0HP (September 1992).
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8. Nuclear Regulatory Commission, Letter to All Holders of Operating Licenses or Construction Permits for Light-Water-Cooled Nuclear Power Reactors, from Charles E. Rossi, June 22, 1988, NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems."
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13. Nuclear Regulatory Commission, Letter to M.S. Tuckman (DPC), September 17, 1991, NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems."

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3.7 Seismic Design

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The site-smoothed response spectra for the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) are defined in Former Appendix 2E (refer to Section 2.6, "Former Appendix 2A-H"). The spectra defined in former Appendix 2E are the responses at the top of sound rock.

Former Figure 2E-2C gives the smoothed spectra for the OBE at two percent damping. The amplification of base motion to the peak response is approximately 3.5 in the period range of 0.17 to 0.5 seconds. The amplification in the period range of 0.03 to 0.17 seconds is greater than 1.0. The response spectra do not reflect the response in the period range 0.03 to 0.05 seconds, however, the response at 0.05 seconds is used for the design of structures, systems and components with a period of vibration between 0.03 and 0.05 seconds.

Ground response is used for the design of structures, systems and components with a period of vibration less than 0.03 seconds.

3.7.1.2 Design Response Spectra Derivation

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

All Category 1 systems and components supported by structures are designed for seismic response by the use of response spectra generated, at the respective structure support elevation, from four synthetic earthquakes at the respective period. Figures 2E-2A through 2E-2D of Former Appendix 2E provide a comparison of the response spectra generated from the synthetic earthquake motions and the site response spectra for 1/2, 1, 2, and 5 percent damping for the OBE.

The following system period interval cases were used as a comparison to establish the final system period intervals for the calculated response spectra:

CASE I. FROM (Rad/Sec)	*TO (Rad/Sec)*	*STEP SIZE (Rad/Sec)*
125	*26*	*3*
26	*6*	*0.5*
6	*1*	*0.25*
CASE II. FROM	*TO*	*STEP SIZE (Rad/Sec)*
125	*25*	*2*
25	*4*	*0.3*
4	*1*	*0.15*
CASE III. FROM	*TO*	*STEP SIZE (Rad/Sec)*
125	*65*	*2*
65	*20*	*1*

20	*4*	*0.3*
4	*1*	*0.15*

The response to the synthetic motions for 0.5 percent damping was used to establish the response spectra used for comparison.

As can be seen from the comparison of Cases I and II in Figure 3-16 the step sizes in Case II produced additional peaks in the 0.1 to 0.25 period range.As shown in the comparison of Clases II and III in Figure 3-17, the step sizes used in Case III produced some additional peaks also in the 0.1 to 0.25 period range. A comparison of the results for Cases I and III is shown in Figure 3-18.

An evaluation of Figure 3-16, Figure 3-17, and Figure 3-18 indicates that the results of Case III represent all peaks calculated by Case I and Case II. An evaluation of the shape of Case III also indicates that Case III is a good representation of all peaks that might exist.

All structural response spectra calculations have been based upon the step sizes given in Case III.

3.7.1.3 Critical Damping Values

The following damping values are used for the seismic design of Category 1 structures, systems and components:

ITEM	PERCENT CRITICAL DAMPING
Containment Vessel	1.0
Welded Steel Structures	2.0
Concrete Structures	5.0

The stress levels in structural elements are not the same for all the elements of a whole structure, therefore, a single value cannot be accurately assigned to a total structure based upon a single stress level. The damping values mentioned above are the average based on the lower stress level in the structure. These values are the same for the OBE and the SSE.

The specific percentage of critical damping values used for Category 1 systems and components by Westinghouse are provided in Table 3-25. Damping values for the ice condenser system structure are presented in Chapter 6.

For analysis cases when ISM methodology (described in 3.7.2.1.3) is utilized, damping values of three percent are used in accordance with Reg. Guide 1.61. Otherwise, in analyses not using ISM, equipment and large diameter piping systems (pipe diameter greater than 12 in.) are analyzed using two percent damping data. Small diameter piping systems (less than or equal to 12 in. diameter) are analyzed using one percent damping data. Higher damping values that are in accordance with Regulatory Guide 1.61 may be used as applicable for SSE-specific analyses.

Optionally the damping values given in Code Case N-411 may be used. This option is generally used for reanalysis of piping systems for either modifications or support/snubber optimization. No combination of the two criteria is used. Code Case N-411 damping values are not used for time history analysis (Reference 18).

Duke complies with Regulatory Guide 1.61 except for the damping values used for concrete structures. Westinghouse exceptions to Regulatory Guide 1.61 are noted in WCAP-7921.

3.7.1.4 Bases of Site Dependent Analysis

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

All major Category 1 structures, such as the Reactor Building and Auxiliary Building are supported on sound rock. The site design response spectra are based upon rock motion and are defined in Former Appendix 2E and Section 3.7.1.1.

3.7.1.5 Soil-Supported Category 1 Structures

All major Category 1 structures such as the Reactor and Auxiliary Buildings are founded on sound rock. The rock characteristics are as defined in Former Appendix 2E. Other Category 1 foundations for equipment and tanks are discussed in Section 4.1.2 of Former Appendix 2D.

3.7.1.6 Soil-Structure Interaction

The Reactor Building, Auxiliary Building and Fuel Handling Building are founded on sound rock. The rock characteristics are defined in Section 2.5."

According to the data obtained from the finite element analysis presented in Section 3.7.2.4, it is seen that the effect of soil-structure interaction on the structure's frequencies and mode shapes is small and can be neglected. Consequently, for the purpose of seismic analysis, each structure has been considered individually and with a fixed base.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

1. Reactor Building and Containment Vessel:

   The stresses, stress resultants and displacements of the response of a shell of revolution to the excitation of an earthquake are calculated by superposing the normal modes of free-vibration of the shell. The modes of vibration are calculated by the general bending theory of shells derived by E. Reissner. The translatory inertia terms in the normal, meridional and circumferential direction of the shell are taken into account. The mass distribution in the mathematical model is the actual mass distribution of the shell and no approximations are made. E. Reissner's shell theory predicts the complete spectrum of natural frequencies of the shell.

   The differential equations given by E. Reissner are solved by the multisegment direct integration method of solving eigenvalue problems, which was published by A. Kalnins (Reference 1 and 2). The eigenvalue problem of a shell of revolution is reduced to the solution of a frequency equation which approaches zero at a natural frequency. The frequency equation consists of a solution of E. Reissner's equations. The calculation of the natural frequencies and the corresponding mode shapes of each mode of free vibration is performed by a computer program written by A. Kalnins. The computer program is used for the calculation of the dynamic characteristics of many types of shells of revolution and its results have been verified with experiments. The program calculates the natural frequencies of any rotationary symmetric thin shell within a given frequency interval and gives all the stresses, stress resultants and displacements corresponding to a natural frequency, at any prescribed point on the meridian of the shell.

The normal modes of free vibration need only be added in order to construct the response of the shell to an earthquake. The relationship between free vibration and a given excitation is given by the following equation:

$$Y(x) = \sqrt{\sum_{i=1}^{N} \left[Y_i(x) \, \frac{C_i S_{vi}}{\omega_i N_i} \right]^2}$$

where:

$Y(x)$	=	Fundamental variables of the response such as deflections, moments, membrane forces or shears.
$Y_i(x)$	=	Fundamental variables of the i^{th} mode such as deflections, moments, membrane forces or shears.
C_i	=	Constant for the i^{th} mode.
ω_i	=	Natural frequency of the i^{th} mode.
N_i	=	Constant for the i^{th} mode.
S_{vi}	=	Maximum velocity from the response spectrum for a single-degree-of-freedom system for a given value of ω_i for the i^{th} mode.
N	=	Number of modes considered.

The Dynamic Analysis was used by the following companies for the analysis of thin shells:

a. Martin Company - Orlando, Florida

b. Pratt and Whitney Aircraft - East Hartford, Connecticut

c. Central Electricity Generating Board - London, England

The Dynamic Analysis was described and its results compared to experiment by: J. J. Williams, "Natural Draught Cooling Towers - Ferry Bridge and After," in the institution of Civil Engineers' Publication, 12 June 1967.

2. Containment Interior Structure and Auxiliary Building

The seismic loads on the Containment Interior Structure and Auxiliary Building as a result of a base excitation are determined by a dynamic analysis. The dynamic analysis is made by idealizing the structure as a series of lumped masses with weightless elastic columns acting as spring restraints. The base of the structure is considered fixed.

The steps used in conducting the dynamic analysis are as follows:

a. The formulation of a mathematical model consisting of lumped masses connected with elastic members. The choice of the location of these mass points depends on the distribution of masses in the real structure (see Section 3.7.2.3). Between these locations, properties are calculated for moments of inertia, cross-sectional area, effective shear area and length.

b. The derivation of the model's influence coefficients (the flexibility matrix). The contributions of flexure, as well as shearing deformations are considered.

The resulting matrix is inverted to obtain the stiffness matrix, which is used together with the mass matrix to obtain the eigenvalues and associated eigenvectors.

The natural frequencies and mode shapes are determined by solving for eigenvalues and eigenvectors from the equations of motion:

$$([K] - \omega^2{}_n [M])\, \phi_n = \underline{o}$$

where,

$[K]$ = Stiffness matrix

$[M]$ = Diagonal mass matrix

ϕ_n = Mode shape vector for the n^{th} mode

$\underline{0}$ = Zero vector

ω_n = Natural circular frequency for the n^{th} mode

Having obtained the frequencies and mode shapes and obtaining the appropriate damping factors, Section 3.7.1.3, the spectral acceleration for each mode can be obtained from spectra curves in Former Appendix 2E. The standard response spectrum technique is used to determine inertial forces, shears, moments and displacements for each mode.

The acceleration response at mass point i is obtained from:

$$\ddot{A}_{ij} = y_j \ \phi_{ij} \ S_{aj}$$

where,

$\ddot{A}_{ij}$ = Response acceleration at mass point i, for mode j

γ_j = Participation factor for mode j

ϕ_{ij} = Mode shape magnitude at mass point i, mode j.

S_{aj} = Spectral acceleration for j^{th} mode as obtained from Former Appendix 2E.

The response displacement at mass point i, for mode j, may be obtained by:

$$A_{ij} = y_j \, \phi_{ij} \, S_{dj}$$

where,

S_{dj} = Spectral displacement for the j^{th} mode

$$= S_{aj} / \omega_j^2$$

The effective earthquake inertial force at mass point i, for the j^{th} mode, is

$$Q_{ij} = m_i \ddot{A}_{ij}$$

The effective shear at mass point i, for the j^{th} mode is

$$V_{ij} = \sum_{y=i}^{N} Q_{yj}$$

and the effective moment is

$$M_{ij} = \sum_{y=i}^{N} Q_{yj} x_y$$

where,

V_{ij} = Shear at mass point i, for mode j

Q_{yj} = Inertia force at mass point y, for mode j

M_{ij} = Moment at mass point i, for mode j

x_y = Distance from mass point i to mass point n

N = Number of mass points

The structural response is obtained by combining the modal contributions of all the modes considered. The combined effect is represented by the square root of the sum of the squares,

$$R_i = \sqrt{\sum_{j=1}^{M} R_{ij}^2}$$

M	=	Number of modes considered
R_i	=	Structural response such as acceleration ($\ddot{A}$), displacement (A), force (Q), shear (V) or moment (M) at mass point i.

The stresses due to moments are calculated based upon a linear strain distribution about the neutral axis of the section of the structure considered. The stresses due to shears are calculated based upon the shear area of the section considered.

For applicable stress criteria, refer to Section 3.8.

Figure 3-20 and Figure 3-21 show sketches of mathematical models used for Category I structures.

3.7.2.1.1 Seismic Analysis Methods for Category I (Safety Class) Systems and Components

Seismic classification of safety related systems and components as per ANS-N18.2 “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants” are presented in Section 3.2. Classification of systems and components by the ANS Safety Classes provides an adequate and proper determination of the applicable seismic design requirements.

In general, the dynamic analyses are performed using a modal analysis plus either the response spectrum analysis or integration of the uncoupled modal equation, as described in Sections 3.7.2.1.1.3 and 3.7.2.1.1.4, respectively, or by direct integration of the coupled differential equations of motion described in Section 3.7.2.1.1.1. However, the first two methods can only be used if the system under study is linear. If non-linearities are involved, such as gaps between components or plasticity, then the equations of motion must be integrated simultaneously.

3.7.2.1.1.1 Dynamic Analysis - Mathematical Model (NSSS Scope)

The first step in any dynamic analysis is to model the structure or component, i.e., covert the real structure or component into a system of masses, springs, and dash pots suitable for mathematical analysis. Essentially, the problem is to select mass points so that the displacements obtained will be a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure or component. Modeling techniques are presented in Reference 3.

Equations of Motion

Consider the multidegree of freedom system shown in Figure 3-19. Making a force balance on each mass point r, the equations of motion can be written in the form (Reference 4)

$$m_r \ddot{y}_r + \sum^{i} c_{ri}\dot{u}_i + \sum^{i} k_{ri} u_i = 0$$

Equation 3.7-1

where,

m_r	=	The value of the mass (rotational inertia) at mass point r.
$\ddot{y}_r$	=	Absolute translational (angular) acceleration of mass point r.
c_{ri}	=	Damping coefficient - external force (moment) required at mass point r to produce a unit translational (angular) velocity as mass point i, maintaining zero translational (angular) velocity at all other mass points. Force (moment) is positive in the direction positive translational (angular) velocity.
$\dot{u}_i$	=	Translational (angular) velocity of mass point i relative to the base.
k_{ri}	=	Stiffness coefficient - the external force (moment) required at mass point r to produce a unit deflection (rotation) at mass point i, maintaining zero displacement (rotation) at all other mass points.
		Force (moment) is positive in the direction of the displacement (rotation).
u_i	=	Displacement (rotation) of mass point i relative to the base.

Note that Figure 3-19 does not attempt to show all of the springs (and none of the dashpots) which are represented in Equation 3.7-1

Since

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s$$

Equation 3.7-2

where,

$\ddot{y}_s$	=	Absolute translational (angular) acceleration of the base,
$\ddot{u}_r$	=	Translational (angular) acceleration of mass point i relative to the base, Equation 3.7-1 can be written as

$$m_r \ddot{u}_r + \sum^{i} c_{ri} \dot{u}_i + \sum^{i} k_{ri} u_i = - m_r \ddot{y}_s \qquad \text{Equation 3.7-3}$$

For a single degree of freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad \text{Equation 3.7-4}$$

3.7.2.1.1.2 Modal Analysis (NSSS Scope)

1. Natural Frequencies and Modal Shapes

The first step in the modal analysis method is to establish the normal modes, which are determined by Equation 3.7-3, with the right hand side equal to zero. The damping terms may be omitted for this purpose. (Reference 4). With the above terms equal to zero, Equation 3.7-3 becomes

$$m_r \ddot{u}_r + \sum^{i} k_{ri} u_i = 0 \qquad \text{Equation 3.7-5}$$

The equation given for each mass point r in Equation 3.7-5 can be written as a system of equations in matrix form as

$$[M](\ddot{\Delta}) + [K](\Delta) = 0 \qquad \text{Equation 3.7-6}$$

where,

[M]	=	Mass and rotational inertia matrix
(Δ)	=	Column matrix of the general displacement and rotation at each mass point relative to the base
[K]	=	Square stiffness matrix
$(\ddot{\Delta})$	=	Column matrix of the general translational and angular accelerations at each mass point relative to the base, $d^2 (\Delta) / dt^2$

Harmonic motion is assumed and the (Δ) is expressed as

$$(\Delta) = (\delta) \sin \omega t \qquad \text{Equation 3.7-7}$$

where,

(δ)	=	Column matrix of the spatial displacement and rotation at each mass point relative to the base.
and		
ω	=	Natural frequency of harmonic motion in radians per second.

The displacement function and its second derivative are substituted into Equation 3.7-6 and yield:

$$[K](\delta) = \omega^2[M](\delta)$$ Equation 3.7-8

The determinant $|[K] - \omega^2[M]|$ is set equal to zero and is then solved for the natural frequencies and the associated mode shapes. (Reference 4). This yields n natural frequencies and mode shapes where n equals the number of masses of the system. The mode shapes are all orthogonal to each other and are sometimes referred to as normal mode vibrations. (Reference 4). For a single degree of freedom system, the stiffness matrix and mass matrix are single terms and the determinant $|[K] - \omega^2[M]|$ when set equal to zero yields simply

$$k - \omega^2 m = 0$$
$$\omega = \sqrt{\frac{k}{m}}$$ Equation 3.7-9

where ω is the natural frequency in radians per second. The natural frequency in cycles per second is therefore

$$f = \frac{1}{2\pi}\sqrt{\frac{k}{m}}$$ Equation 3.7-10

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation 3.7-8, however, only n - 1 of these are independent. This means that the elements of (δ) can be expressed only as multiples of one another. Normalizing (δ) such that the maximum displacement (rotation) of any element is unity gives

ϕ_{rn} = Displacement (rotation) of mass point r in mode n relative to the base

2. Modal Equations

The response of a structure or component is always some combination of its normal modes. The combination method is described in Section 3.7.3.4. Good accuracy can usually be obtained by using only the first few modes of vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as, (Reference 4),

$$\ddot{A}_n + 2\omega_n p_n \dot{A}_n + \omega_n^2 A_n = -\gamma_n \ddot{y}_s$$ Equation 3.7-11

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u_{rn}, by the equation

$$u_{rn} = A_n \phi_{rn}$$ Equation 3.7-12

where,

ω_n = Natural frequency of mode n in radians per second.

λ_n = Critical damping ratio of mode n.

γ_n = Modal participation factor of mode n given by

$$\gamma_n = \frac{\sum^{n} m_r \phi'_{rn}}{\sum^{n} m_r \phi^2_{rn}} \qquad \text{Equation 3.7-13}$$

and

ϕ'_{rn} = Component of ϕ_{rn} in the direction of the earthquake.

The essence of the modal analysis lies in the fact that Equation 3.7-11 is analogous to the equation of motion for a single degree of freedom system that will be developed from Equation 3.7-4. Dividing Equation 3.7-4 by m gives

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \qquad \text{Equation 3.7-14}$$

The critical damping ratio of a single degree of freedom system, λ, is defined by the equation

$$\lambda = \frac{c}{c_c} \qquad \text{Equation 3.7-15}$$

where the critical damping coefficient is given by the expression

$$c_c = 2m\omega \qquad \text{Equation 3.7-16}$$

Substituting Equation 3.7-16 into Equation 3.7-15 and solving for c/m gives

$$\frac{c}{m} = 2\omega\lambda \qquad \text{Equation 3.7-17}$$

Substituting this expression and the expression for k/m given by Equation 3.7-9 into Equation 3.7-14 gives

$$\ddot{u} + 2\omega\lambda\dot{u} + \omega^2 u = -\ddot{y}_s \qquad \text{Equation 3.7-18}$$

Note the similarity of Equations 3.7-11 and 3.7-18. Thus each mode may be analyzed as though it were a single degree of freedom system and all modes are independent of each other. By this method a fraction of critical damping, i.e., c/c_c, may be assigned to each mode and it is not necessary to identify or evaluate individual damping coefficients, i.e., c. However, assigning only a single damping ratio to each mode has a drawback. Normally, there are two ways used to overcome this limitation when considering slightly damped structures (e.g., steel) supported by a massive moderately dampened structure (e.g., concrete). The first method is to develop and analyze separate models for both structures using their respective damping values. The massive moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures are used as forcing functions for their subsequent detailed analysis. The second method is to inspect the mode shapes to determine which modes correspond to the supports and which modes correspond to the supported structures.

3.7.2.1.1.3 Response Spectrum Analysis (NSSS Scope)

The response spectrum is a plot showing the variation in the maximum response (displacement, velocity, and acceleration) of a single degree of freedom system versus its natural frequency of vibration when subjected to a time history motion of its base (Reference 5).

The spectrum concept can best be explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree of freedom system with a given damping and natural frequency be calculated for a given base motion. The variations in response are established, and the maximum value of each is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency in sufficient detail to establish the complete curve. Other curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis, and the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with that base motion. When these curves are generated mathematically, the actual curves are not smooth and require a certain degree of judgement in smoothing them out. The spectra acceleration, velocity, and displacement are related by the equation:

$$S_{a_n} = \omega_n S_{v_n} = \omega_n{}^2 S_{d_n}$$ Equation 3.7-19

There are two types of response spectra that must be considered. If a given building is shown to be rigid and to have a hard foundation, the response spectrum based on a time history of ground motion is used. It is referred to as a ground response spectrum. If the building is flexible and/or has a soft foundation, the ground response spectrum is modified to include these effects. The response spectrum at various support points must be developed. This is called a floor response spectrum. The specific response spectrum curves used are discussed in Section 3.7.1.3 and Section 3.7.2.6.

3.7.2.1.1.4 Integration of Modal Equations (NSSS Scope)

This method can be separated into the following two basic parts:

1. Integration procedure for the uncoupled modal Equations 3.7-11 to obtain the modal displacements and accelerations as a function of time. Integration of these uncoupled modal Equations 3.7-11 can be done by electronic simulation (analog computer) or by step-by-step numerical integration. The electronic simulation method is well documented in the literature (Reference 6) and, therefore, does not need to be discussed here. The step-by-step numerical integration procedure (Reference 7) consists of selecting a suitable time interval, Δt, and calculating modal acceleration, $\ddot{A}_n$, modal velocity, $\dot{A}_n$, and modal displacement, A_n, at discrete time stations Δt apart, starting at t = 0 and continuing through the range of interest for a given time history of base acceleration.

To illustrate, once the modal displacement, modal velocity, and the base acceleration Y_s are known at one such station, the modal acceleration is computed from Equation 3.7-11. The displacement at the next station is then calculated by a recurrence formula such as the following constant velocity procedure:

$$(A_n)_{m+1} = 2(A_n)_m - (A_n)_{m-1} + (\ddot{A}_n)_m[\Delta t]^2 \qquad \text{Equation 3.7-20}$$

where,

$(A_n)_{m+1}$	=	Modal displacement (rotation) in mode n relative to the base at the $(m+1)^{th}$ time step.
$(A_n)_m$	=	Modal displacement (rotation) in mode n relative to the base at the m^{th} time step.
$(A_n)_{m-1}$	=	Modal displacement (rotation) in mode n relative to the base at the $(m-1)^{th}$ time step.
$(\ddot{A}_n)_m$	=	Modal translational (angular) acceleration in mode n relative to the base at the m^{th} time step.
Δt	=	Time interval between m^{th} time step and $(m + 1)^{th}$ time step.

It is noted that the use of such recurrence formula for the first time step requires special consideration, as is stated in the literature (Reference 7). Thus the complete modal time history is obtained.

Other time integration techniques are available in the literature. For example, the Westinghouse Information Systems Laboratory has at least two computer programs, ICE and NICE, in its library.

2. Using these modal displacements and accelerations to obtain the total displacements, accelerations, forces, and stresses.

   From the modal displacements and accelerations, the total displacements, accelerations, forces, and stresses can be determined as follows:

   1) Displacement of mass point r in mode n as a function of time is given by Equation 3.7-12 as

   $$u_{rn} = A_n \phi_{rn} \qquad \text{Equation 3.7-21}$$

   with the corresponding acceleration of mass point r in mode n as

   $$\ddot{u}_{rn} = \ddot{A}_n \phi_{rn} \qquad \text{Equation 3.7-22}$$

   2) The displacement and acceleration values obtained for the various modes are superimposed algebraically to give the total displacement and acceleration of each time interval.

3) The total acceleration in each time interval is multiplied by the mass to give an equivalent static force. Stresses are calculated by applying these forces to the model or from the deflections at each time interval.

3.7.2.1.1.5 Integration of Coupled Equations of Motion (NSSS Scope)

The coupled equations of motion given by Equation 3.7-3 can be integrated using methods similar to those outlined in Section 3.7.2.1.1.4 for integrating the uncoupled model equations.

3.7.2.1.1.6 Systems Components (Duke Energy Mechanical Scope)

In accordance with Table 3-4, mechanical system components are designed to be capable of resisting the earthquake loads imposed by the applicable response spectra curves developed by methods described in Section 3.7.1. System components have been evaluated to verify their capability to withstand design loadings as described in Table 3-30 and Table 3-47, Table 3-48, Table 3-49, Table 3-50, and Table 3-51. The manufacturers have evaluated the mechanical components of their equipment by analysis or static testing. Static testing is employed by using conservatively applied loads to verify the operability of a specific mechanical component. The manufacturers have submitted to Duke reports summarizing the results of their analyses. These reports have been reviewed and verified by Duke or their consultant, EDS nuclear. In general, the manufacturer has designed his equipment and its structural support system such that the fundamental frequency of the system is above 30 Hz and the equipment with its supports have been considered “rigid”. Manufacturers have submitted calculations which verify this frequency assumption. Or, in some cases, Duke has performed calculations to verify this frequency assumption. Instrumentation, control and electrical systems which are part of the mechanical system subject to qualification by analysis, whether supported on the equipment or not, have been qualified in accordance with the procedures set forth in Section 3.10.

3.7.2.1.1.7 Cylindrical Shell Type Equipment and Components and Their Supports (Duke Energy Mechanical Scope)

The design specification for each tank, heat exchanger and pressure vessel specifies the particular loads for which the particular component must be designed. The loading combinations and stress criteria to which the component must be designed are shown in Table 3-51. Seismic loading for each component is specified in the design specification in the form of response spectra. The design specification requires that the manufacturer perform or have performed a modal analysis using the spectra provided. The spectra provided is general for all equipment at a given elevation. However, the procedure for determining the effects of torsional building modes is explicitly described in Section 3.7.2.10 and the location of the component within the building is specified. The result of piping flexibility analysis show the calculated loads on nozzles. These loads have been compared with the design loads to assure a compatible design.

3.7.2.1.1.8 Valves (Duke Energy Mechanical Scope)

The valve design specifications contain requirements for the operability of the valves for seismic loadings. These requirements are considered to be more than adequate to assure capability for all calculated seismic loadings. After completion of piping design, the accelerations to which the valves would be subjected under the postulated seismic event, are compared with the specification limitations. In order to verify the operability of valve operators, static tests may be employed to verify that operators will function when distorted under the specified loads. In general, the natural frequency of valve operators is greater than 30 Hz. For valves suspended in pipe lines, the imposed accelerations are obtained from the dynamic analysis of the piping systems in which the valves are located. The accelerations of valves obtained by this method are compared with those set forth in the specifications to verify the adequacy of the design.

The capability of valves to resist the moment loading introduced from the adjacent piping is assured due to its greater wall thickness requirements for pressure-retaining capability. Because of the relatively low stress levels imposed on the body of the valve, it is not anticipated that local valve body distortions will have any influence on the operability of the valve. Where this assumption is not valid, and in cases where stresses are excessively high, the manufacturer is required to verify the capability of the valve for the design loads and may be required to demonstrate by tests the operability of critical valves for the faulted conditions.

3.7.2.1.1.9 Pumps (Duke Energy Company Scope)

The pump specifications contain the criteria for determining the nozzle loads and seismic accelerations under which pumps must be designed to operate. The design nozzle loads vary with the operating condition and the size of the pump nozzles. Three design conditions are given: (1) normal operating, (2) normal operating plus OBE, and (3) normal operating plus SSE. The results of piping flexibility analyses show the calculated loads on the pump nozzles for each of the above conditions. The calculated loads are compared with the design loads to assure a compatible design. Pumps are considered to be rigid bodies rigidly mounted when compared to seismic frequencies and are therefore considered not susceptible to modal analysis. Consequently, the only significant seismic design requirement is that each pump be required to operate under the influence of the lateral and vertical accelerations of the floor on which it rests.

It is anticipated that seismically induced stresses in the body of a pump will be relatively small and will have little influence on the operability of the pump. Where stresses are excessively high, or where rigidity cannot be clearly demonstrated, the manufacturer is required to verify the capability of the pump for the design loads and may be required to demonstrate by test the operability of critical pumps for faulted conditions.

3.7.2.1.1.10 General Methods of Evaluation

For seismic analysis, a single conservatively-determined acceleration value is used for pumps and other mechanical components subjected to a static load equivalent to the weight of the equipment multiplied by the acceleration. Stresses, displacements, and loadings are determined on the basis of this static calculation. Verification that the frequency of vibration of the equipment is in the "rigid" range is accomplished using a conservative analytical procedure. The procedures used in analyzing a particular piece of equipment are justified by the manufacturer. No dynamic testing is required.

3.7.2.1.2 System Piping

Duke's ASME III Design Specification describes the loading conditions for which the Class I nuclear piping is designed. This specification considers both static and dynamic loadings and establishes the combinations of loadings that are considered credible. These loading conditions are categorized according to the classifications - Normal, Upset, Emergency, and Faulted as defined in Table 3-30, in order that loadings may be related to allowable stresses.

In accordance with Section 3.2.2 and applicable response spectra curves as developed from the method described in Section 3.7.2.6, and enveloped for conservatism, system piping is analyzed as follows:

3.7.2.1.2.1 Seismic Criteria

All seismically designed piping includes earthquake loads represented by horizontal earthquake response spectra at the various floor elevations in the Category 1 structures. For a piping system spanning between two or more elevations, an upper bound envelope of all applicable individual spectra is used. (Reference 15, 16). The spectra used to represent the vertical seismic accelerations are equal to 2/3 the horizontal ground spectra where no vertical floor spectra are developed. Each piping system is evaluated using: a) the envelope of results from an N-S earthquake combined with a vertical earthquake and an E-W earthquake combined with a vertical earthquake; or b) the results from the simultaneous application of three orthogonal directions of earthquake.

For the evaluation of relative support motions in the seismic analysis of piping systems interconnecting two or more primary structures, the maximum relative movement between structures is assumed, and the piping system is subjected to these movements through the piping system supports and restraints. Separate cases for N-S earthquake and E-W earthquake are considered. Support movements are based on the maximum of the floor movements immediately above and below the support location, with the interpolation optional. The stresses in the piping resulting from these imposed restraint movements are considered to act concurrently with other seismic and thermal stresses; however, these stresses are considered to be secondary stresses and as such are combined directly with the stresses resulting from thermally induced movement.

All piping is classified into either one of two categories, rigid or flexible. Rigid piping is that which has a period of less than 0.033 seconds (corresponding to a modal frequency of 30 Hz). All piping with periods greater than 0.033 seconds is classified as flexible.

3.7.2.1.2.2 Method of Analysis - Rigid Piping

All rigid piping is designed for a uniform static coefficient equal to the maximum floor acceleration corresponding to the appropriate building elevation for each piping system.

3.7.2.1.2.3 Method of Analysis - Flexible Piping Greater than Four Inches in Diameter

A dynamic seismic analysis is performed on applicable flexible piping systems by the response spectrum method. The method employed is described below:

Each pipe loop is idealized as a mathematical model consisting of lumped masses connected by elastic members. Lumped masses are located at carefully selected points in order to adequately represent the dynamic and elastic characteristics of the pipe system. Using the elastic properties of the pipe, the flexibility matrix for the pipe is determined. The flexibility calculations include the effects of the torsional, bending, shear, and axial deformations. In addition, for curved members, the stiffness is decreased in accordance with ASME III for applicable nuclear piping systems.

Once the flexibility and mass matrices of the mathematical model are calculated, the frequencies and mode shapes for all significant modes of vibration are determined. All modes having a period greater than 0.033 seconds (corresponding to a modal frequency of 30 Hz) are used in the analysis. The mode shapes and frequencies are solved in accordance with the following equation:

$$([K] - \omega_n^2 [M])\phi_n = \underline{0}$$

in which:

[K] = Square stiffness matrix of the pipe loop

[M] = Mass matrix for the pipe loop

ω_n = Frequency for the n^{th} mode

ϕ_n = Mode shape matrix of the n^{th} mode

After the frequency is determined for each mode, the corresponding spectral acceleration is read from the appropriate response spectrum for the pipe. Using these spectral accelerations, the response for each is found by solving the following equation:

$$Y_n \max = \frac{R_n Sa_n [D]}{[M_n]\omega_n^2}$$

in which:

$Y_n max$ = Response of the n^{th} mode

R_n = Participation factor for the n^{th} mode = $\Sigma m_i \phi_{in}$

Sa_n = Spectral acceleration for the n^{th} mode

[D] = Earthquake direction matrix

$[M_n]$ = Generalized mass matrix for the n^{th} mode = $\Sigma m_i \phi_{in}{}^2$

Using these results, the maximum displacements for each mode are calculated for each mass point in accordance with the following equation:

$$V_{in} = \phi_{in} Y_n \max$$

in which:

V_{in} = Maximum displacement of mass i for mode n

The total displacement for each mass is determined by taking the square root of the sum of the squares of the maximum deflection for each mode;

$$V_i = \sqrt{\sum V_{in}^2}$$

in which:

V_i = Maximum displacement of mass i due to all modes calculated

For closely spaced modes, where modal frequencies are within 10% of one another, modal responses are combined by absolute sum.

The inertia forces for each direction of earthquake for each mode are then determined from:

$[Q_n]$ = [K][V]

in which:

$[Q_n]$ = Inertia force matrix for mode n

[V] = Displacement matrix corresponding to Q_n

Each mode's contribution to the total displacement, internal forces, moments, and stresses are determined from standard structural analysis methods using the inertia forces for each mode as an external loading condition. The total combined results are obtained by taking the square root of the sum of the squares of each parameter under consideration except where the modal frequencies are within 10% of one another. For these closely spaced modes, modal responses are combined by absolute sum.

3.7.2.1.2.4 Method of Analysis - Flexible Piping Nominal Size Four Inches and Less

The alternate methodology described below is a conservative, approximate analysis that may be used to analyze piping and is included in the UFSAR for reference only. This information is provided to give a general description of the methodology, however, since it is for reference only, it is not being maintained. For the current analysis methodology and its applicability, see Specification MCS 1206.02-04-0000, "Alternate Analysis Criteria for Duke Energy Piping Classification B, C, & F" x 4" and smaller piping in reactor and auxiliary buildings.

A conservative approximate analysis procedure was used to determine the seismic response of piping systems of four-inch nominal diameter and less. The analysis included the effects of pressure and dead weight, and of horizontal and vertical seismic loadings.

The piping system was divided into a series of equal pipe spans between supports. The natural frequencies of the pipe spans were determined for all pipe sizes and schedule numbers for various support spacings. Pipe spans were determined such that the natural frequencies of the pipes were relatively high in comparision with the response spectra of the structures in which they are located.

For those spans containing concentrated weights located at the midspan, the length of span was calculated such that the period of the span is equal to or less than the period of a maximum seismic span of the same pipe size and schedule.

For those spans containing a single concentrated weight located near the support, the maximum concentrated weight was calculated such that the first period of the span results in a specified response acceleration. The accelerationis 1.6g (OBE) for the Auxiliary Buildig and 2.0g (OBE) for the Reactor Building. For those spans containing a single concentrated weight on a change in direction, the same spans determined for concentrated weights on a span with no change in direction were used. However, the maximum concentrated weight was determined such that the maximum response acceleration (OBE) would not exceed 1.6g for Auxiliary Building and 2.0g for Reactor Building for the various cases considered in analysis.

The horizontal Operating Basis Earthquake (OBE) spectra applied to the Reactor Building piping, was developed by enveloping the 1% critical damping horizontal floor response spectra for all elevations of the Reactor Interior, the Reactor Building, and the Containment Vessel. Due to geometric symmetry of the Reactor Building, no seismic torsional effects were considered in the analysis.

The horizontal OBE spectra applied to Auxiliary Building piping was developed by enveloping the 1% critical damping horizontal floor response spectra for all elevations of the Auxiliary Building. Seismic torsional effects, caused by the asymmetrical Auxiliary Building, are only significant if the piping system is excited at the building's fundamental frequency. Pipes analyzed in this alternate analysis have been support such that the natural frequencies of the pipe spans are relatively high in comparison with the fundamental frequency of the Auxiliary Building. Therefore, the torsional effects were considered to be insignificant.

The vertical OBE spectrum applied to both the Reactor Building piping and the Auxiliary Building piping is the 1% critical damping Ground Response Spectrum for the McGuire Nuclear Station.

The maximum dynamic response of the piping was conservatively assumed to occur in the first mode. The fundamental period was used to enter the applicable seismic response spectra curves to obtain the horizontal and vertical accelerations to be applied to the pipe. To account for the possible influence of higher mode excitation of the piping, the horizontal and vertical accelerations obtained in this manner were increased by twenty percent. The accelerations values so obtained were then used to determine pipe deflections, bending stresses, and support reactions.

In addition to the integral vertical and lateral supports described above, long straight runs of piping were assumed to be provided with axial restraints to ensure that no significant seismic excitation of the large pipe mass would occur in the axial direction. To maintain reasonable support loads on these axial restraints, straight runs of pipe were limited as specified in Appendix A for the development of these criteria. Seismic loads on axial restraints were calculated for both horizontal and vertical 50-foot straight runs of pipe.

Out-of-plane supports perpendicular to the plane containing the pipe bend were assumed where supports were located at the recommended spans away from the bend in each direction. These out-of-plane supports provide seismic and/or dead load support to the piping at such locations while maintaining in-plane thermal flexibility.

All maximum acceptable lateral support spacings allow the piping to satisfy the stress criteria of the Upset Condition described in the next paragraph. This was confirmed for spans with and without concentrated weights.

For the Upset Condition, the material allowable stress at the maximum operating temperature (s_h @ 300°F) is multiplied by 1.2 as permitted paragraph NC-3611.1 (c) of ASME Section III, for loads occurring during aone percent of the operating period. $S_{OBE} + S_G + S_{1p} \leq 1.2S_h$.

Deflection criteria were applied in accordance with the McGuire Nuclear Station Preliminary Safety Analysis Report. The criteria are based on engineering judgement to minimize the possibility of excessive deflections causing interference with other pipes, or other undesirable performance characteristics. For the Upset Condition, the maximum allowable deflection is one inch.

$$\delta_{OBE} + \delta_G \leq 1.0 \text{ inch}$$

The seismic stresses and displacements due to OBE loading were multiplied by 15/8 to determine the response of the piping to Safe Shutdown Earthquate (SSE) excitation. Three-dimensional dynamic coupling effects in the piping system were eliminated by placing restraints near all changes of direction of the piping. Three-dimensional dynamic coupling effects in those piping spans containing concentrated weights were eliminated by limiting the distance between the center of gravity of the concentrated weight and pipe centerline.

For the Faulted Condition, the material allowable stress at the maximum operating temperature (S_h @ 300°F) is multiplied by 2.4 as permitted by paragraph NC-3652 of ASME Section III, for loads occurring during a Safe Shutdown Earthquake (SSE) occurrence.

$$S_{SSE} + S_G + S_{1p} + S_{EXT} + S_{DBA} \leq 2.4\ S_h$$

It is noted that the scope of these criteria, defined in Section 2.6, specifically excludes piping requiring additional analysis for the consideration of external loads, including Design Basis Accident loading. The stress evaluation performed for the Faulted Condition within the scope of this report reduces to the following:

$$S_{SSE} + S_G + S_{1p} \leq 2.4\ S_h$$

Due to the exclusion of external loads, and SSE loading being less than double the OBE loading, the stress criteria for the Faulted Condition are automatically satisfied when those for the Upset Condition are satisfied.

There are no deflection criteria currently specified for the Faulted Condition. The limitations used in the Upset Condition evaluation have been imposed upon the Faulted Condition as well.

$$\delta_{SSE} \leq 1.0 \text{ inch}$$

If actual conditions warrant, larger deflections may be permitted. However, an evaluation of the consequences of such larger deflections should be performed.

Comparisons of the results using this conservative approximate analysis method with those obtained using the response spectra mode superposition method are shown for two typical systems in Section 3.7.3.9.

3.7.2.1.3 Alternative Analysis Methodologies

As an alternative to the method described in Section 3.7.2.1.2 of the McGuire UFSAR, the independent support motion methodology may be used.

A piping subsystem which is supported in more than one building structure and/or is supported at varying elevations within a single structure may be analyzed using the independent support motion (ISM) methodology. Inertial response as well as relative anchor motion effects are combined to determine the total response of the piping. For the inertial response, the ISM methodology allows the specific input of response spectra at the support locations. Supports are classified into groups or levels based on structure and elevation. X, Y and Z direction spectra are correlated to each group and input in the analysis as applied loadings. For each direction, the response is calculated based on the absolute sum of the group responses and a SRSS modal combination method including missing mass effects. The total inertial response is determined by the SRSS of the directional responses. For the relative anchor motion effects, a static analysis is performed. The inertial and anchor motion responses are developed by the SRSS combination. This methodology conforms to that described in NUREG-1061 (Reference 19) and approved for use by NRC's letter of October 13, 1995 (Reference 20).

3.7.2.2 Natural Frequencies and Response Loads

In the following, the natural frequencies, critical mode shapes and the response loads of some Category I structures are given:

1. The Reactor Building:

   a. Natural Frequencies and Mode Shapes

      Figure 3-22 illustrates the first and second horizontal mode shapes of the Reactor Building due to ground excitation. Also, shown on the same figure are the magnitudes of the first and second natural horizontal frequencies of the Reactor Building.

      The first and second vertical mode shapes of vibration of the Reactor Building are shown in Figure 3-23. On the same figure the magnitudes of the first and second natural vertical frequencies are shown.

   b. Response Loads

      The response loads of the Reactor Building due to the safe shutdown earthquake (SSE) are shown in Figure 3-24 through Figure 3-29. The response loads are calculated based on the combined modal effects. The critical mode shapes used in the analysis are the first and second horizontal modes as well as the first and second vertical modes. Refer to Section 3.7.2.1 for more details on the seismic analysis of the Reactor Building, and the method of combining the individual modal responses.

Since no critical equipment or support points are attached to the Reactor Building, generating the response spectra at different elevations of the Reactor Building is not necessary.

2. The Containment Interior Structure:

The mathematical model of the Containment interior structures is shown in Figure 3-20. The seismic analysis procedure is fully outlined in Section 3.7.2.1. Some of the numerical results include:

a. Natural Frequencies and Mode Shapes

North-South

The first four horizontal mode shapes of vibration of the Containment interior structure are shown in Figure 3-30. The magnitude of the first four horizontal natural frequencies are also shown on the same figure.

East-West

For this direction, the first four horizontal mode shapes of the Containment interior structure and the associated magnitudes of the first four frequencies are shown in Figure 3-31.

Vertical

Figure 3-32 illustrates the first two vertical mode shapes of vibration of the Containment interior structure and the magnitude of their natural frequencies.

b. Response Loads

The response loads of the Containment interior structure due to the Safe Shutdown Earthquake (SSE) are calculated according to the procedure of seismic analysis outlined in Section 3.7.2.1. The first four horizontal modes and the first vertical mode of the interior structure are combined in calculating the following response loads:

1) Inertia forces.

2) Acceleration at different elevations.

3) Displacements.

4) Shearing forces including interior structure base shear.

5) Moments at different elevations including the overturning moment at the fixed base.

These response loads are shown in Figure 3-33 for the North-South direction SSE and in Figure 3-34 for the East-West direction SSE.

Figure 3-35 through Figure 3-39 illustrate the response spectra of the interior structure at important equipment elevations as well as at other critical points of support.

3. Systems and Components (by Westinghouse)

Natural frequencies of Westinghouse supplied components are considered in the system seismic analysis. The natural frequencies of the components themselves are above the seismic cutoff frequency and listings of the natural frequencies are presented in the components' stress reports.

3.7.2.3 Procedures Used to Lump Masses

The procedure used to lump masses for the seismic structural model is dependent upon the actual mass distribution and structural characteristics of the structure.

Mass locations are established at elevations in the structure where there are concentrations of mass such as floor slabs and/or equipment. Mass locations have also been established when there are changes in structural properties such as moments of inertia, shear area or elastic properties.

The mass of the equipment is lumped at that elevation at which it is supported such as lateral supports for the steam generators, reactor vessel, reactor coolant pumps, pressurizer and polar crane. When equipment is supported on a floor slab, the equipment mass is lumped with the structural mass of the slab.

The mass of the structural members, elastic members between masses, is distributed to the adjacent mass locations.

The structural connection between equipment and structure is considered rigid for the seismic analysis of the structure. A response spectrum has been generated as defined in Section 3.7.2.6 at mass locations where equipment or piping is supported. This response spectrum is used for the seismic design of equipment and piping as defined in Sections 3.7.2.1.1 and 3.7.2.1.2.

Refer to Section 3.7.2.1.1 for criteria to lump masses for systems and components. Westinghouse methods and procedures used to lump masses are presented in Section 3.7.2.1.1.

Refer to Section 3.7.5 for the procedures used to assure that all the required inputs and/or responses required by different design organizations for all Category I structures are compatible.

3.7.2.4 Rocking and Translational Response Summary

The effect of rocking and translational response on the structures founded on sound rock, is investigated. The Reactor Building shell and foundation are represented by shell elements of revolution, and the base rock is represented by solid elements of revolution. The dimensions of the base rock considered are selected in such a manner that the free-field conditions exist in the model for joints located away from the structure and the influence of the boundary conditions do not affect the rocking or translational motion of the structure. In accordance with recommendations (Reference 8) the radius and depth of the base rock for the model are not less than 1.5 and 1.0 times the diameter of the structure respectively.

The finite element representation of the Reactor Building and base rock is shown in Figure 3-40.

The horizontal frequencies and mode shapes for the Reactor Building and base rock, and the Reactor Building fixed at the base, are calculated as defined in Reference 8.

A plot of the normalized mode shape for the first horizontal mode is shown in Figure 3-41; a plot of the second horizontal mode is shown in Figure 3-42.

As shown in the comparison of the first mode, the base rock (soil interaction) results are nearly equal to that of the fixed base condition.

The comparison of the second mode reflects a difference in the mode shape at the base of the structure; however, the overall difference in mode shape and frequency is minor.

The first mode has the most influence on the structural responses and minor differences in the higher modes would not materially influence the total design of the structure.

A tabulation of the maximum accelerations for the different spectra is shown in Table 3-26. The response spectra for the reactor vessel support (elevation 738.22), steam generator lateral support (elevation 774.60) and a typical high penetration support (elevation 768.22) for the fixed base models and combined interaction models are shown in Figure 3-43 through Figure 3-48.

3.7.2.5 Methods Used to Couple Soil with Seismic - System Structures

Refer to Section 3.7.2.4 for a description of the finite elements method employed to couple the soil (base rock) and the seismic - system structures.

3.7.2.6 Development of Floor Response Spectra

Figures 2E-2A through 2E-2D of former Appendix 2E reflect the time-history spectra and site design spectra.

The synthetic earthquakes used to generate the time-history spectra in Former Figures 2E-2A through 2E-2D were used to generate response spectra at elevations in structures that house systems and components that are designed for seismic excitation.

The analytical technique used to generate the response spectra at specified elevations in a structure is the time-history method. The acceleration time-history of each elevation is retained for the generation of response spectra reflecting the maximum acceleration of a single-degree-of-freedom system for a range of frequencies at the respective elevation.

Damping values for the structural model are selected from Section 3.7.1.3.

<u>TIME-HISTORY ANALYSIS</u>

The time-history of the specified mass points is determined by the modal method in which the responses in the normal modes are determined separately, then superimposed to provide the total response to a specified base input motion.

The displacement a_{rn} for any arbitrary mass point r, in the n^{th} mode, can be represented as a function of the modal displacement A_n, therefore,

$$a_{rn} = A_n \left[\frac{a_{rn}}{A_n} \right] = A_n \phi r_{rn}$$

$$\dot{a}_{rn} = \dot{A}_n \phi_{rn}$$

$$\ddot{a}_{rn} = \ddot{A}_n \phi_{rn}$$

where:

a_{rn} = Displacement of the r^{th} mass point in the n^{th} mode

ϕ_{rn} = Mode shape magnitude at mass point r, for the n^{th} mode

Dots indicate differentiation with respect to time.

The generalized displacement (coordinate) response of the structure is obtained by solving the modal equation for support motion. For the n^{th} mode this equation is:

$$\ddot{A}_n + \omega_n^2 A_n + 2B_n \dot{A}_n = -\ddot{y}_s(t) n \gamma_n$$

where:

ω_n = Natural circular frequency of the n^{th} mode

B_n = $\lambda_n \ \omega_n$

λ_n = Ratio of damping to critical damping for the n^{th} mode

$\ddot{y}_s(t)$ = Support acceleration time history

γ_n = Modal participation factor for the n^{th} mode

$$\delta_n = \frac{\sum_{r=1}^{j} m_r \ \phi_{rn}}{\sum_{r=1}^{j} mr\,\phi^2_{rn}}$$

j = Number of mass points

m_r = Mass value at the mass point r.

The modal relative displacement of mass point r is:

$$u_{rn}(t) = A_n(t)\phi_{rn}$$

and the relative acceleration

$$\ddot{u}_{rn}(t) = \ddot{A}_n(t)\phi_{rn}$$

The response of each mass for each mode at each increment of time is retained, and the total response for each increment of time is obtained by summing the responses of each mode for a particular time. The total relative displacement of mass point r is:

$$u_r(t) = \sum_{n=1}^{M} u_{rn}(t)$$

and the relative acceleration is

$$\ddot{u}_r(t) = \sum_{n-1}^{M} \ddot{u}_{rn}(t)$$

where M = the number of modes considered. The time-history method gives the exact combination of mode participation and therefore the time-history of each mass is defined.

RESPONSE SPECTRA

A response spectrum can be defined as the representation of the maximum response of a single mass system for a varying frequency range to a defined base motion.

The time-history of the mass points is used as the base motion to obtain the response spectra. The numerical average for the response of the four earthquake time-histories was used to generate the final response spectra used in the seismic design.

A typical structural mass model of the Containment Interior Structure is shown in Figure 3-20.

Response spectra are generated, for structures that require the generation of response spectra, in the horizontal and vertical direction for structures with modes of vibration less than 20 Hz. For structures with fundamental modes of vibration in a particular direction equal to or greater than 20 Hz, the ground time-history response spectra are used.

When the ground response spectra are used the acceleration values corresponding to 20 Hz are used as a minimum value for the design of piping and components. The acceleration values at 20 Hz are greater than the values corresponding to a rigid system and therefore are conservative.

Typical horizontal response spectra for five elevations of the Containment interior structure are shown in Figure 3-35 through Figure 3-39.

3.7.2.7 Differential Seismic Movement of Interconnected Components

Refer to Section 3.7.2.1.2 for the description of the analytical consideration of the differential seismic movement of the interconnected components between floors.

The effect of differential seismic movement of interconnected components (supplied by Westinghouse) is considered in the analysis. The interconnected components, subjected to differential movement, are within the applicable stress and deformation limits.

3.7.2.8 Effects of Variations on Floor Response Spectra

To take into account the possible variations in structural properties, damping, soil or rock properties and soil-structure interaction, the calculated floor response spectrum is shifted ± ten percent of the period at points on the curve. In addition, the peak of the curve is increased ten percent. An adjusted design typical floor spectra is shown in Figure 3-49.

Alternatively, analysis with the unbroadened response may be used as described in Code Case N-397 and in the Summer 1984 addendum to Section III, Appendix N of the ASME Code. This option is generally used for the reanalysis of piping systems for either modifications or support/snubber optimization (Reference 18)

3.7.2.9 Use of Constant Load Factors

The vertical modes of vibration are considered in the seismic design of structures.

The vertical modes of vibration for the Containment Vessel and Reactor Building are determined as defined in Section 3.7.2.1. The vertical frequencies of these structures are less than 20 Hz and are considered to influence the seismic design. All vertical modes contributing significantly to the seismic loads are used.

Lumped mass structures with vertical modes of vibration less than 20 Hz are designed by performing a dynamic analysis in the vertical direction. The dynamic analysis is performed as defined in Section 3.7.2.1.

Lumped mass structures with vertical fundamental frequencies equal to or greater than 20 Hz are designed as rigid structures with a constant vertical acceleration equal to the acceleration corresponding to 20 Hz on the vertical response spectrum. The acceleration response at 20 Hz was greater than the response of an infinitely stiff structure and was conservative.

The response spectrum used for the design of vertical modes is equal to two-thirds of the horizontal spectrum.

The maximum horizontal and vertical seismic responses are considered to act simultaneously.

The method of analysis for systems and components for vertical seismic excitation is described in Section 3.7.2.1.

The constant load factors are not used as the vertical floor response load for the seismic design of Category I systems and components within the scope of responsibility of Westinghouse.

3.7.2.10 Method Used to Account for Torsional Effects

Category I structures are designed so as to minimize the distance between the center of mass and the center of rigidity. Torsional moments for structural design were computed by multiplying the seismic forces by the distance between center of rigidity and center of mass. The shears due to torsional moments are applied to the frames by the relative stiffness method as presented in Reference 9, "Design of Multistory Reinforced Concrete Buildings for Earthquake Motion" by Blume, Newmark and Corning.

COUPLED TRANSLATIONAL AND TORSIONAL MODES

The dynamic analysis is performed by idealizing the structure as a series of lumped masses with weightless elastic columns acting as spring restraints. The base of the structure is considered fixed. The steps used in conducting the dynamic analysis are as follows:

1. Formulate a mathematical model consisting of lumped masses connected with elastic members. The choice of the location of these mass points depends on the distribution of masses in the real structure (see Section 3.7.2.3). Between these locations, values are calculated for flexural moments of inertia, cross-sectional area, effective shear area, torsional moment of inertia and length. The masses and mass moments of inertia are calculated at the mass locations. The eccentricity of the center of mass relative to the center of rigidity at each mass location is also determined.
2. Derive the model's stiffness matrix which is used together with the mass matrix (which includes the mass moment of inertia) to obtain eigenvalues and associated eignenvectors.

The natural frequencies and coupled torsional and translational mode shapes are determined by solving for eigenvalues and eigenvectors from the equations of motion:

$$([K^*]-\omega_n^2[\underline{M}^*])\phi_n = \underline{0}$$

where,

$[K^*]$ = Total stiffness matrix (translation + torsion)

$[M^*]$ = Diagonal total mass matrix (including mass moment of inertia)

ϕ_n = Coupled mode shape vector for the n^{th} mode

$\underline{0}$ = Zero vector

ω_n = Natural circular frequency for the n^{th} mode

Having obtained the frequencies, mode shapes and the appropriate damping factors (Section 3.7.1.3), the spectral acceleration for each mode can be obtained from spectra curves in Section 2.5. The standard response spectrum technique is used to determine inertial forces, shears, moments and displacements for each mode. Refer to Section 3.7.2.1 for detailed description.

3.7.2.11 Comparison of Responses

The seismic design of Category I structures is performed by the response spectrum technique.

The generation of response spectra for support elevations on structures for the seismic analysis of systems and components is made by the time-history method.

The contribution of each mode for the response spectrum analysis was combined as defined in Section 3.7.2.1 and is not consistent with the technique used in the time-history analysis.

Considering the differences in basic principles of the two methods, it is reasonable to assume that the results did not necessarily coincide.

Table 3-27 gives a tabulation of the maximum acceleration, shears and moments as calculated by the two methods for the Containment Interior Structure.

As can be seen from this tabulation, the time-history technique produces greater response in the lower portion of the structure. The response in the upper portion of the structure is compatible and in some cases the response spectrum technique produces greater responses than the time-history technique.

An evaluation of Figure 2E-2C of former Appendix 2E verifies that the response of structural systems to the time-history input is always greater than the response due to the spectrum technique for any given mode of vibration.

The maximum ground acceleration for the time-history is increased some 36 percent above the maximum site ground acceleration in order to produce conservative spectra in structures for the design of piping and equipment. Therefore, a comparison of the responses of structures from the time-history and response spectrum techniques is not an indication of the conservatism of the design of structures.

3.7.2.12 Method for Seismic Analysis of Dams

The seismic analysis of the Standby Nuclear Service Water Dam is performed according to the dynamic method of stability check described by Newmark (Reference 10). Law Engineering Testing Company recommends a minimum factor of safety of 1.05 for the SSE. This factor of safety is considered conservative because: (1) the analysis does not include the shear resistance of the sides of the failure zone and (2) the analysis does not account for periodic short duration reversals of motion inherent in earthquakes.

An additional check for the factor of safety against seismic loading is also made using the pseudo-dynamic analysis, as recommended by Law Engineering Testing Company Foundation Report, Former Appendix 2D. Details of the Standby Nuclear Service Water Dam are presented in Former Appendix 2G.

3.7.2.13 Methods to Determine Category I Structure Overturning Moments

Category I structures overturning moments due to seismic base excitation are determined as outlined in Section 3.7.2.1. The overturning moments for the Containment Interior Structure due to other loading conditions, e.g., LOCA, is determined by summing the contributions of the differential pressures on various portions of the structure at different times. This way, a time-history of the overturning moments, shearing forces and uplift forces on the structure was established. The maximum values of these forces and moments which are used for design are shown in Table 3-28.

The overturning moments for shell type structures, e.g., Reactor Building and Containment Vessel, are automatically included in the shell analysis of such structures.

The maximum differential pressures across different segments of the Interior Structure and time at which each occurs are shown in Tables 6-2 and 6-3.

3.7.2.14 Analysis Procedure for Damping

1. Structure

   Refer to Section 3.7.1.3 for values of the critical damping for Category I structures and the assumptions on which these values are based.

2. NSSS Scope

   In a coupled system with different structural elements, either the lowest damping value associated with the elements of the system is used for all modes, or equivalent modal damping values are determined according to the energy distribution of the structural elements in each mode. In the case of the Reactor Coolant Loop/Support Systems, the damping values are established as described above and confirmed by comparison with existing test results (Reference 11). A summary of damping values for Westinghouse supplied equipment is presented in Table 3-25.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Determination of Number of Earthquake Cycles

1. Category 1 Systems and Components Other Than NSSS

   For the design of Category 1 structures, systems and components, the number of earthquake cycles during one operating basis earthquake (OBE) is assumed to be 40. The number of assumed postulated events of this loading is assumed to be 5, resulting in a total number of 200 full cycles. In addition, for the safe shutdown earthquake (SSE), one event is assumed, resulting in 100 cycles.

2. NSS System

   Where fatigue analyses of mechanical systems and components are required, Westinghouse specifies in the equipment specification the number of cycles of the Operating Basis Earthquake (OBE) to be considered. The number of cycles for NSSS components is given in Table 5-49. The fatigue analyses are performed and presented as part of the components stress report.

3.7.3.2 Basis for Selection of Forcing Frequencies

The frequencies of component response are selected so as to avoid resonance as described in Section 3.7.2.

3.7.3.2.1 Basis for Selection of Forcing Frequencies (NSSS Scope)

The analysis of equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports based upon the mass and stiffness characteristics of the system, will determine the seismic accelerations which the equipment must withstand.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low-period region of the floor response spectra.
2. If the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant and the equipment behaves as though supported on the ground.
3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

The considering of equipment as rigid, flexible, or resonant is based on the ratio of the fundamental frequency of the equipment to the fundamental frequency of the supporting structure. The rigid category is considered applicable for frequency ratios having a value greater than 2.0. The flexible category is considered applicable for frequency ratios having values less than The resonant category includes frequency ratios having a value between 0.5 and 2.0.

When feasible, the resonant region of the subsystem is designed to occur beyond the regions of the forcing frequencies. The shifting of the resonant region of the subsystem is accomplished by alternating its mass and/or stiffness characteristics. Certain components may qualify even under peak resonance conditions. In either case, components under earthquake loading are designed to be within code allowable stresses. Calculation of forcing frequencies is presented in Section 6.3.

3.7.3.3 Root Mean Square Basis

The term "root-mean-square" basis is not used in structures, components or equipment seismic analyses. The modal responses are combined by using the square root of the sum of the square method. The procedure for combining modal responses is presented in Section 3.7.3.4.

3.7.3.4 Procedure for Combining Modal Responses

3.7.3.4.1 Duke Supplied Piping

Modal responses for piping are combined by taking the square root of the sum of the squares of each parameter under consideration except where modal frequencies are within ten percent of each other. For these closely spaced modes, modal responses are combined by the absolute sum. This is automatically performed by the piping analysis computer programs PISOL-1A and SUPER PIPE. Duke-supplied piping analysis meets all requirements of IE Bulletin 79-07 (Reference 16).

3.7.3.4.2 NSSS Scope

The total seismic response in the Reactor Coolant System Analysis is obtained by combining the individual responses utilizing the square root of the sum of the square method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10% of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root sum of the squares of all modes the product of the responses to the modes in each group of closely spaced modes and a coupling factor ε. This can be represented mathematically as

$$R_T^{\,2} = \sum_{i=1}^{N} R^2{}_i + 2\sum_{j=1}^{S} \sum_{K=M_j}^{n_j-1} \sum_{\lambda=K+1}^{N_j} R_K R_{\lambda} \varepsilon_{K\lambda}$$

where

R_T = total response

R_i = absolute value of response of mode i

N = total number of modes considered

S = number of groups of closely spaced modes

M_j = lowest modal number associated with group j of closely spaced modes

N_j = highest modal number associated with group j of closely spaced modes.

$\varepsilon_{K\lambda}$ = coupling factor with

$$E_{K\lambda} = 1 + \left[\left(W'_k + W'_\lambda \right)^2 / (B'_k w_K + B'_\lambda w_\lambda) \right]^{-1}$$

and

w'_j = $w_j [1-(B'_j)^2]^{1/2}$

B'_j = $B_j + \dfrac{2}{w_j t_d}$

w_j = frequency of closely spaced mode j

B_j = fraction of critical damping in closely spaced mode j

t_d = duration of the earthquake

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Freq	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20

There are two groups of closely spaced modes, namely with modes (2, 3, 4) and (6, 7). Therefore,

S = 2 number of groups of closely spaced modes

M_1 = 2 lowest modal number associated with group 1

N_1 = 4 highest modal number associated with group 1

M_2 = 6 lowest modal number associated with group 2

N_2 = 7 highest modal number associated with group 2

N = 8 total number of modes considered

The total response for this system is, as derived from the expansion of equation (3.K-22a).

$$R_T^2 = (R_1^2 + R_2^2 + ... + R_8^2) + 2R_2R_3\varepsilon_{23} + 2R_2R_4\varepsilon_{24}$$

$$+ 2R_3R_4\varepsilon_{34} + 2R_6R_7\varepsilon_{67}$$

For equipment and components, the method described above for the Reactor Coolant System analysis is used for closely spaced modes when the modes occur all component bending in the horizontal x direction and another mode with frequency within 10% of the frequency of the first, due to component internals (e.g., pump shaft, heat exchange tubes, etc.) response also in the x direction, the responses will be combined using the closely spaced modes equations.

However, if these modes are not in the same direction, for instance, if one mode is in the horizontal x direction and another is in the horizontal z direction, or if one mode is horizontal and another vertical, the contributions of the modes are combined by the square root sum of the squares (SRSS) method. The Westinghouse-supplied analysis meets all requirements of IE Bulletin 79-07 (Reference 16).

Modal responses for the NSSS reactor coolant loop seismic analyses performed for steam generator replacement, which include Babcock and Wilcox steam generators, are combined in accordance with Regulatory Guide 1.92.

3.7.3.5 Significant Dynamic Response Mode

Static loads equivalent to the peak of the floor spectrum will be used only where equipment can be modeled as a single degree of freedom system.

3.7.3.5.1 Significant Dynamic Response Modes (NSSS Scope)

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as a single-degree-of-freedom system are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50 percent to account conservatively for the increased modal participation.

3.7.3.6 Design Criteria and Analytical Procedure for Piping

1. Westinghouse Supplied Primary Coolant Loop

   The effect of different floor response spectra at different elevations, seismically induced relative building displacements, are conservatively included in the analysis. Seismic analysis of the primary coolant loop is presented in Reference 12, a topical report entitled, “Westinghouse Technical Position on discrete Break Locations and Types for the LOCA Analysis of the Primary Coolant Loop”.

   The reactor coolant loop model is reanalyzed to incorporate the Babcock and Wilcox International replacement steam generators. The ground response spectra provides seismic input to the primary coolant loop model that is coupled to the Reactor Building interior structure model. The seismic analysis is described in Section 5.2.1.10 and Reference 22 of Section 3.7.6.

2. Duke Supplied Piping Systems

For the design criteria and analytical procedures pertaining to support displacements on piping resulting from building displacement, see Section 3.7.2.1.2. Generally, the relative displacements of floors within a building are considered to have a minor effect on piping from the standpoint of stress because the building is considered to be far more rigid than the piping. However, it is recognized that certain equipment with extremely low allowable loads such as pumps could possibly be affected by relative floor movements. These are evaluated by the same method as described for relative building movements.

3.7.3.7 Basis for Computing Combined Response

1. Westinghouse Supplied Systems and Components

   The seismic design of the reactor coolant loop/support systems, piping and components includes the modeling of the support effects. The modeled structure is then analyzed using the horizontal and vertical seismic spectra which are prepared to properly and conservatively excite the piping system at the attachment points to the building structure. The system is analyzed for the simultaneous occurrence of the horizontal and vertical motions. The horizontal and vertical response loadings are conservatively combined. The resulting stresses are held below the appropriate code allowable limits.

2. Duke Supplied Equipment and Piping Systems

   The effects of seismic response of supports and equipment are not directly included in the seismic analysis of piping initially as equipment and supports are normally designed and analyzed subsequent to the piping analysis. After the equipment has been analyzed, the results are reviewed for response in the frequency spectrum of interest. The finding of significant response from the equipment will result in a re-analysis of the piping with the model revised to include the equipment stiffness and mass characteristics. The methods for combining horizontal and vertical response loading are described in Section 3.7.2.1.2 for piping and equipment.

3.7.3.8 Amplified Seismic Responses

1. Westinghouse Supplied Components and Equipment

   Constant vertical load factors are not used as the vertical floor response load for the seismic design of safety related components and equipment within Westinghouse's scope of responsibility.

2. Duke Supplied Piping and Equipment

   The use of a constant load factor as the vertical floor response load for the seismic design of Category 1 piping and equipment is limited according to Section 3.7.3.5.

3.7.3.9 Use of Simplified Dynamic Analysis

As described in Section 3.7.2.1.2.4, a simplified dynamic analysis is used for flexible Category 1 piping less than six-inch nominal size. Presented below is a comparison of results for a piping system which was analyzed using both the dynamic analysis method of Section 3.7.2.1.2.2 and the simplified method of Section 3.7.2.1.2.4. These results show that the simplified analysis yield conservative results compared to the dynamic analysis method of Section 3.7.2.1.2.2.

Analyses of two typical piping systems have been performed by the Approximate Seismic Analysis method and the Response Spectra Mode Superposition method. The analysis procedures used are as discussed in Section 3.7.2.1.2. Results for the full dynamic analysis have been combined with a static analysis for dead load to provide a consistent basis for comparison with the Approximate Seismic method:

SAMPLE SYSTEM SHOWN IN Figure 3-50

	Dynamic Analysis (Seismic & Deadload)	Approximate Seismic Analysis
First Mode Period (Seconds)	0.069	0.069
Maximum Stress (psi)	1771	7800
Maximum Displacement (in.)	0.04	0.40
Maximum Reaction (lb)	60	72
Number of Modes Considered in Dynamic Analysis = 11		
Least significant Period = 0.033 seconds		

SAMPLE SYSTEM SHOWN IN Figure 3-51

	Dynamic Analysis (Seismic & Deadload)	Approximate Seismic Analysis
First Mode Period (Seconds)	0.071	0.074
Maximum Stress (psi)	2030	7800
Maximum Displacement (in.)	0.04	0.45
Maximum Reaction (lb)	276	334
Number of Modes Considered in Dynamic Analysis = 24		
Least significant Period = 0.033 seconds		

Spectra used for the analysis of both systems by the Dynamic and Approximate Methods are as shown in Figure 3-52 and Figure 3-53.

3.7.3.9.1 Use of a Simplified Dynamic Analysis (NSSS Scope)

The typical Westinghouse supplied Safety Related mechanical components are checked for seismic adequacy as follows:

1. If a component falls within one of the many categories which have been previously analyzed using a multi-degree-of-freedom model and shown to be relatively rigid (all natural frequencies greater than 33 Hz) then the equipment specification for that component is checked to ensure that the appropriate response spectrum is smaller than the specified values.
2. If the component cannot be categorized as similar to previously analyzed components, then an analysis is performed as described in Section 3.7.2.1.

3.7.3.10 Modal Period Variation

1. Structures

To take into account the possible variations in structural properties, damping, soil or rock properties and soil-structure interaction, the calculated floor response spectrum is shifted ± ten percent of the period at points on the curve. In addition, the peak of the curve has been increased by ten percent. An adjusted design typical floor spectrum is shown in Figure 3-49.

2. Westinghouse Supplied System

   The materials employed in safety related systems under Westinghouse scope of supply are standard. The material properties which can effect a variation in modal period are well known, and the known variation in these properties does not account for any measurable or significant shift in period or increase in seismic loads.

3.7.3.11 Torsional Effects of Eccentric Masses

1. Westinghouse Supplied Piping System

   The seismic mass model accounts for the effect of masses that are offset from the pipe centerline. Components with eccentric masses are modeled by placing the component's mass at its calculated center of gravity and connecting this mass to the pipe centerline with a rigid connection. The inertia forces calculated from the response spectra curves are applied at this lumped mass point. Therefore, any forces or moments, including torsion, resulting from eccentric masses are accounted for in the seismic analysis.

2. Duke Supplied Piping Systems

   Piping systems are modeled to include projecting masses such as valve motor operators. For valves that have natural frequency less than 30 Hz a detailed model is used which represents the member properties and mode shapes of the actual valve operator. The operators of valves with natural frequency greater than or equal to 30 Hz are modeled as rigid components.

3.7.3.12 Piping Outside Containment Structure

3.7.3.12.1 Buried Piping

Seismic design criteria for buried piping from the Standby Nuclear Service Water Pond to the station are as follows:

1. Intake structure is designed such that the differential movement between this structure and the earth is negligible and the seismic response spectrum utilized is the ground response.
2. Allowable structural and piping stresses after the line penetrates the Auxiliary Building is assured by the use of expansion joints.

Other buried lines penetrating structures are designed similarly to the Standby Nuclear Services Water line described above or alternatively by the use of flexible seals as the lines pass through pipe sleeves in the structure.

Important factors considered are the flexibility, supports and restraints of lines which are virtually anchored in earth but which penetrate a structure. A flexibility analysis of these lines is performed to show that the piping and structures are not overstressed under the additive differential movement of the earth and structure.

3.7.3.12.2 Above Ground Piping

Seismic design criteria and methods of accounting for the effects of differential movement of buildings on piping and penetrations are described in Sections 3.7.2.1.2, 3.7.2.7. and 3.7.2.8.

3.7.3.13 Interaction of Other Piping with Category 1 Piping

The protection of Category 1 piping from possible adverse effects of other piping during an earthquake is accomplished by several methods. Specifically, these methods are:

1. Category 1 lines are physically separated from other lines insofar as possible such that failure of a line has no effect on Category 1 lines.
2. All Category 1 boundary valves are designed to meet seismic criteria. A valve always serves as a pressure boundary and constitutes the seismic/non-seismic boundary. If failure in the non-seismic portion of the system could cause loss of function of the safety system, then an appropriate automatic or remote manual operator would be used if the valve is open during normal reactor operation.
3. The pressure boundary valve is protected by restraining or anchoring the non-seismic portion of the system as required.

3.7.3.14 Field Location of Supports and Restraints

The location, type and design loads of restraints as determined by dynamic analysis are forwarded to the hanger designer for his use in the design of the supports and restraints. The hanger designer produces the necessary designs, prepares, checks and approves fabrication and erection sketches describing the supports and restraints and issues them for erection. After the piping is erected, a quality assurance inspection is made by Duke Quality Assurance to verify that the supports and restraints are installed as designed. Duke Design Engineering checks as-erected sketches against the design information for location, type, and design loads. Assurance that design documents reflect as-built conditions is detailed in Duke's response to IE Bulletin 79-14 (Reference 17).

3.7.3.15 Seismic Analysis For Fuel Elements, Control Rod Assemblies and Control Rod Drives (NSSS Scope)

Fuel assembly component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling. The time history floor response based on a standard seismic time history normalized to Safe Shutdown Earthquake levels is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems. The seismic response of the fuel assemblies is analyzed to determine the design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in Reference 13, “Fuel Assembly Safety Analysis for Combined Seismic and Loss of Coolant Accident”. A detailed description of the analyses performed for the Mark-BW fuel assembly is contained in Reference 15.

The Control Rod Drive Mechanisms (CRDM) are seismically analyzed to confirm that system stresses under seismic conditions do not exceed allowable levels as defined by the ASME Boiler and pressure Vessel Code Section III for “Upset” and “Faulted” conditions. Based on these stress criteria, the allowable seismic stresses in terms of bending moments in the structure are determined. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation and the resultant seismic bending moments along the length of the CRDM are calculated. These values are then compared to the allowable seismic bending moments for the equipment, to assure adequacy of the design.

Duke Fuel Assembly Compatibility Evaluation for the supply of 17x17 Westinghouse Robust Fuel Assemblies.

Seismic Evaluation

The non-linear dynamic seismic analysis of the reactor pressure vessel system includes the development of the system finite element model and the synthesized time history accelerations. The development of the system finite element model and the synthesized time history accelerations is given in Section 3.9.3.2.2.

3.7.4 Seismic Instrumentation Program

3.7.4.1 Comparison with Regulatory Guide 1.12

Seismic instrumentation has been provided to conform with the intent of the Regulatory Guide 1.12, Revision 2.

3.7.4.2 Location and Description of Instrumentation

Five strong motion triaxial accelerographs are used to obtain seismic event data at the station site. One of these instruments is located on the Reactor Building basement in the annulus area outside the Containment, and a second is located directly above the first, attached to the Containment divider at about Containment midheight. The sensor orientation of all instruments is identical.

A network control system is used for rapid interrogation of accelerograph data and provides digital storage for recording the history. The time-history data can be used to determine peak acceleration values at the above locations. Also, for rapid determination of peak acceleration, triaxial recording accelerometers are used in other selected locations for rapid determination of the effect of a seismic event.

In addition to the seismic instrumentation defined above, two triaxial sensor/recorders are placed at selected locations of Category 1 structures, systems and components. These instruments are used to provide responses for a specified damping and frequencies and are used to establish sufficient points to generate response spectra at the same locations. Peak responses from calculated design response spectra are used to establish instrument locations.

The major Category 1 structures, Reactor Building, Containment and Auxiliary Buildings are founded on a common rock foundation and have similar base motions. The dynamic structural properties and responses of these structures are generated using similar assumptions and analytical techniques. Therefore, the response of these structures can be determined based upon the instrumentation in one structure.

Top of soil (free field) responses do not provide useful analytical data for the evaluation of major Category 1 structures founded on rock. Therefore, free field instrumentation does not contribute to the evaluation of these structures.

3.7.4.3 Control Room Operator Notification

The seismic monitoring system is activated when two of the three accelerograph sensors located outside containment exceed a predetermined acceleration value. Upon activation, Control Room operators will receive OAC alarm “Seismic System Actuated”. The activation setpoint is based on the seismic equipment manufacture's recommendation and guidance provided by Regulatory Guide 1.12, Rev 2. The seismic Network Control Center console located in the Control Room will record and display peak acceleration values for each axis of displacement. The system records absolute acceleration verse time data for each accelerograph sensor location.

In the event that OBE is exceeded, a Control Room annunciator alarm “OBE Exceeded” is actuated by contact closure of the Network Control Center. Setpoints for the OBE exceedance annunciator alarm are based on 100% OBE values of 0.08 g (horizontal) and 0.053 g (vertical).

3.7.4.4 Comparison of Measured and Predicted Responses

In the event of an earthquake, the data is analyzed to determine the magnitude of the earthquake. If the Operating Basis Earthquake is exceeded, the units are shut down and structures, systems and equipment thoroughly investigated. Responses from instruments located on selected structures, systems and components are compared to calculated responses for those structures, systems and components at the respective location when subjected to the same base response.

The recorded seismic data is used for comparison and verification of seismic analysis assumptions, damping characteristics and the analytical model used for the station seismic design.

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3.7.4.5 Test and Inspection

Periodic calibration and alignment is performed on this instrumentation in order to assure proper operation.

3.7.5 Seismic Design Control Measures

3.7.5.1 Duke Procedure for Seismic Design Control

The Civil- Mechanical-, Nuclear, and Electrical Divisions are responsible for the seismic design of the station.

The Civil Division is responsible for the site seismology and maximum accelerations. The site seismological data, maximum accelerations and time-history for the synthetic earthquakes are provided by Law Engineering Testing Company and reviewed and approved by the Civil Division.

The site response spectra and spectra for the design of systems and components located in structures are generated by the Civil Division. These spectra, with a transmittal letter, are forwarded to the sections in the Design Engineering Department and to Westinghouse, who is responsible for the seismic design of the NSS systems and components.

Revisions in the generated spectra are made by the Civil Division and forwarded with a transmittal letter to the concerned parties, reflecting the revisions and instructions to update the existing spectra.

The Civil, Mechanical, Nuclear and Electrical Divisions are responsible for defining those structures, systems and components requiring seismic analysis, outlining acceptable design and testing techniques and writing respective specifications and/or performing seismic design calculations.

Specifications for structures, systems and components requiring seismic design or testing are checked for compliance with the latest seismic design data by the Supervising Engineer within the respective Engineering Division. Specifications received Engineering checks and approvals as outlined in Chapter 17.

Seismic design performed within the Engineering Department receives the design control as outlined in Chapter 17.

Seismic design calculations and/or test results submitted by vendors and suppliers and checked for compliance with approved specifications, design techniques and seismic criteria by the responsible Engineering Division.

The seismic design techniques, criteria and design controls performed by Westinghouse are reviewed and approved by the Engineering Department.

3.7.5.2 Seismic Design Control (NSSS Scope)

The following procedure is implemented for Westinghouse supplied safety-related mechanical equipment that falls within one of the many categories which have been analyzed as described in Sections 3.7.2 and 3.7.3 and has been shown to be relatively rigid with all natural frequencies greater than 33 Hz:

1. Equivalent static acceleration factors for the horizontal and vertical directions are included in the equipment specification. The vendor must certify the adequacy of the equipment to meet the seismic requirements as described in Section 3.7.3.
2. When the floor response spectra are developed the cognizant engineer responsible for the particular component checks to ensure that the acceleration factors are less than those given in the equipment specification.

All other Westinghouse supplied safety-related equipment is analyzed or tested as described in Sections 3.7.2, 3.7.3 and 3.10.

Westinghouse design control generally and seismic design control specifically is discussed in detail in Chapter 17.

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3.8 Design of Category I Structures

3.8.1 Concrete Containment

The concrete containment outside the steel Containment Vessel, separated from each other by annulus space, is named, here and throughout, the Reactor Building.

3.8.1.1 Description of the Containment

The concrete containment (Reactor Building) is a reinforced concrete structure composed of a right cylinder with a shallow dome and flat circular foundation slab. The Reactor Building houses the Containment Vessel and is designed to provide biological shielding as well as missile protection for the steel Containment shell.

A five foot annulus space is provided between the Containment Vessel and Reactor Building shell for control of Containment external temperatures and pressures. The annulus space also provides a controlled air volume for filtering and access to penetrations for testing and inspection. The annulus nominal volume is 427,000 $ft.^3$

The Reactor Building has a cylinder inside radius of 62 ft. 6 in., a thickness of 3 ft. 0 in. and a dome thickness of 2 ft. 3 in. The dome inside radius is 87 feet. The height of the Reactor Building is approximately 177 feet. The structural outline of the Reactor Building is shown on Figure 3-54.

The annulus space is kept at a slight negative pressure following a loss-of-coolant accident to control and filter radioactive leakage, if any, from the Containment Vessel and penetrations. For further details on the Annulus Ventilation System, see Section 6.2.

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The arrangement patterns and amount of reinforcing steel for the Reactor Building are shown in Figure 3-55 through Figure 3-64.

3.8.1.2 Applicable Codes, Standards and Specifications

The applicable codes, standards and specifications employed in the design of the Reactor Building are given in Table 3-31.

3.8.1.3 Loads and Loading Combinations

3.8.1.3.1 Static Analysis

The Reactor Building is statically analyzed as a shell of revolution subject to axisymmetric and asymmetric loadings. These loadings are detailed in Section 3.8.1.4. Refer to Section 3.8.2.4 for description of the static analysis of shells of revolution.

3.8.1.3.2 Dynamic Analysis

Refer to Section 3.7.2.1 for the description of the dynamic analysis of shells of revolution subject to base excitation.

3.8.1.3.3 Loading Combinations

Loading combinations and code requirements for the Reactor Building are summarized in Table 3-32.

3.8.1.4 Design and Analysis Procedures

The Reactor Building is designed to maintain its function for the following loadings:

1. Dead Loads (DL)
2. Seismic Loads (OBE or SSE)
3. Design Basis Accident Loads (DBA)
    a. Pressure
    b. Temperature
4. Operating Loads (OL)
    a. Live Loads
    b. Snow and Ice Loads
    c. Penetration Loads and Pipe Reactions
    d. Soil and Water Pressure
    e. Temperature
5. Tornado Loads (W_t)
6. Wind Loads (W)
7. Construction Loads (CL)
8. Pipe Rupture Loads (Y)

Dead Loads

The dead load includes all dead loads during and after construction.

Operating Loads

Operating loads are those loads associated with the operation of the unit, which include normal thermal loads and penetration loads due to pipe reactions.

Design Basis Accident Loads

The design basis accident loads on the Reactor Building include the thermal loads due to the rise in temperature of the annulus. The design temperature gradient across the containment vessel is 77.2°F. Since the annulus is maintained at a negative pressure during a DBA, DBA pressure on the Reactor Building wall is not considered for the analysis and design. Following a DBA, there will be a slight increase in annulus pressure, but due to the Annulus Ventilation System, this will be reduced to a slight negative pressure within a short time following the accident (see FSAR Section 6.2.3). Due to the small magnitude of these pressures, they will not be considered in the Reactor Building Wall analysis or design.

Wind Loads

The wind loads are based upon ASCE Paper 3269, "Wind Forces on Structures," using 95 mph as the fastest mile of wind for a 100 year period of recurrence.

The non-axisymmetric loads considered in the design of the Reactor Building are the normal and tornado wind loads.

The wind loads are analyzed by approximating the wind distribution of the Reactor Building, as defined in ASCE Paper 3269, by a Fourier Series. The wind distribution curve and Fourier Series used in the design are given in Figure 3-1. Individual harmonics are analyzed by Kalnin's program and combined to produce the stress resultants for the total series. The analytical techniques used by the program are given in Section 3.8.2.4.

Tornado Loadings

The tornado loadings are described in Section 3.3.2.

Snow and Ice Loads

The Reactor Building dome is designed for a snow and ice load of 20 pounds per square foot.

Soil and Water Pressure

The Reactor Building is designed for earth pressures due to backfill and groundwater pressure as defined in Section 2.4.

Seismic Loads

Refer to Section 3.7, for the seismic loadings.

The Reactor Building is designed in accordance with the loading combinations and code requirements of Table 3-32.

Refer to Section 3.8.3.4 for listing of the computer programs used for the structural analysis as well as the testing procedure of these programs.

In addition to the analysis performed on the Reactor Building as a whole, areas around large penetrations in the Reactor Building wall are examined.

The area around a large penetration (e.g., the equipment hatch) is modeled as a space frame. The forces and moments at the boundaries of this area, which were previously obtained using shell theory, are used in the analysis. The model used in the analysis is as shown in Figure 3-65. Adequate reinforcement is provided to carry the design forces and moments obtained from the analysis. For the small penetrations of the Reactor Building which are subject to major pipe loads, the stresses in the vicinity of these penetrations are evaluated by performing an analysis using a space frame or by applying well established analytical procedures. As a result of these analyses additional reinforcement configurations around these penetrations are provided, as necessary.

The overpressure load of 1.8 psi as given in Section 2.2.3 due to a postulated explosion is combined with dead load and operating load. This pressure acting horizontally is less than the pressure of the tornado wind (2.3 psi). Considering the 1.8 psi as external pressure, the forces and moments due to the postulated explosion are found to be smaller in magnitude than those from DL+OL+W_t (Table 3-32). It is concluded that the Reactor Building design is adequate to withstand this load condition.

3.8.1.5 Structural Acceptance Criteria

The Reactor Building is designed to remain within the elastic limit of the structural material when subjected to any of the loading combinations defined in Table 3-32.

For loading combinations 1 through 4 of Table 3-32 the working stress method is used. The allowable stress limits of the concrete and the reinforcing steel are as specified in Chapter 10 of the ACI-318-1963 code. Loading combinations 5, 6 and 7 of Table 3-32 are ultimate (faulted) conditions and require that the structure remains functional.

The working stress method of design for reinforced concrete is based upon cracked sections of reinforced concrete structural members. Therefore, cracked sections are assumed when working stress design methods are used in the design of the Reactor Building wall.

For considering the tangential shear, the Reactor Building shell is assumed to be a cantilever with a circular cross section. The Reactor Building shell is not exposed to a net uplift; therefore vertical membrane stresses are small and created only by horizontal loadings. Considering the above, the tangential shear is considered the same as the shear and diagonal tension as defined in the ACI-318-63 code. The shearing stresses are small and are within the allowable shear stress limits defined in Chapter 10 of the ACI-318-63 Code.

3.8.1.6 Materials, Quality Control and Construction Techniques

The following materials are used for the design and construction of the Reactor Building:

1. Concrete - 3000 psi and 5000 psi based upon 28 day test.
2. Reinforcing Steel - ASTM A615, Grade 40 and/or Grade 60.
3. Structural Steel Shapes - ASTM A-36.
4. Embedded Plates - ASTM A-36 and/or SA-516.

The materials and quality control procedures are in accordance with the requirements of ACI-318-63, and the quality assurance requirements of Chapter 17 of the FSAR. Wherever mechanical rebar splicing is used, such splices (e.g., Cadwelds) meet the requirements of Regulatory Guide 1.10. There are no special construction techniques employed in the construction of the Reactor Building.

3.8.1.7 Testing and Inservice Surveillance Requirements

Although the Reactor Building is here listed as a containment, it is not an actual containment and not subject to 10CFR 50 Appendix J. The steel Containment, completely described in Section 3.8.2, is the real Containment subject to all the design requirements and provisions for pressurized Containment Vessels.

The Reactor Building is visually inspected on a periodic basis as defined by Improved Technical Specifications as noted in Table 18-1.

3.8.2 Steel Containment System

3.8.2.1 Description of the Containment

The Containment Vessel is a freestanding welded steel structure with a vertical cylinder, hemispherical dome and a flat base. The Containment shell is anchored to the Reactor Building foundation by means of anchor bolts around the circumference of the cylinder base. The base of the Containment is 1/4 in. liner plate encased in concrete and anchored to the Reactor Building foundation. The base liner plate functions only as a leak-tight membrane and is not designed for structural capabilities. The Containment Vessel has a nominal inside diameter of 115 ft., overall height of 171 ft. 3 in., nominal wall thickness of 0.75 inch, nominal dome thickness of 0.6875 inch, nominal bottom thickness of 0.25 inch, and net free volume of 1.2 x 10^6 cubic feet. Other details are as shown in Figure 3-54 and Figure 3-66.

The Containment penetrations are:

1. EQUIPMENT HATCH

   The equipment hatch is composed of a cylindrical sleeve in the Containment shell and a dished head 20.0 feet in diameter with mating, bolted flanges. The flanged joint has double compressible seals with an annulus space for pressurization and testing.

   The equipment hatch is designed, fabricated and tested in accordance with Section III, Subsection B, of the ASME Boiler and Pressure Vessel Code, 1968 Edition, including all addenda through Summer 1970.

   Details of the equipment hatch are shown on Figure 3-67.

2. PERSONNEL LOCKS

   Two personnel locks are provided for each unit. Each lock has double doors with an interlocking system to prevent both doors being opened simultaneously. Remote indication is provided to indicate the position of each door.

   Double, inflatable seals are provided on each door. A top connection between the seals provides the capability for local leak rate testing as required. The use of double inflatable seals allows testing of the annulus space without the use of external strongbacks or other remote devices.

   The personnel locks are a completely prefabricated and assembled welded steel subassembly designed, fabricated, tested and stamped in accordance with Section III, Subsection B of the ASME Code.

   Details of the personnel locks are shown on Figure 3-67.

3. FUEL TRANSFER PENETRATION

   A 20-inch fuel transfer penetration is provided for transfer of fuel to and from the fuel pool and the Containment fuel transfer canal.

   The fuel transfer penetration is provided with a double gasketed blind flange in the transfer canal and a gate valve in the fuel pool.

   Expansion bellows are provided to accommodate differential movement between the connecting buildings. Figure 3-68 shows conceptual details of the fuel transfer penetration.

4. SPARE PENETRATIONS

   Spare penetrations are provided to accommodate future piping and electrical penetrations. The spare penetrations consist of the penetration sleeve and head.

5. PENETRATION SLEEVES

   All penetration sleeves are pre-assembled into Containment Vessel shell plates and stress relieved prior to installation into the Containment Vessel as shown on Figure 3-69 and Figure 3-70.

6. PURGE PENETRATIONS

   The purge penetrations have one interior and one exterior quick-acting tight-sealing isolation valve. Details of the purge penetrations are shown on Figure 3-68.

7. ELECTRICAL PENETRATIONS

Medium voltage electrical penetrations for reactor coolant pump power (shown on Figure 3-68) use sealed bushings for conductor seals. The assemblies incorporate dual seals along the axis of each conductor.

Low voltage power, control and instrumentation cable enter the Containment Vessel through penetration assemblies which have been designed to provide two leak tight barriers in series with each conductor.

All electrical penetrations have been designed to maintain Containment integrity for Design Basis Accident conditions including pressure, temperature and radiation. Double barriers permit testing of each assembly as required to verify that Containment integrity is maintained.

The conformance of the electrical penetrations to Regulatory Guide 1.63 is discussed in Section 8.3.1.2.7.7.

Qualification tests which may be supplemented by analysis are performed and documented on all electrical penetration assembly types to verify that Containment integrity is not violated by the assemblies in the event of a design basis accident. Existing test data and analysis on electrical pene- tration types may be used for this verification if the particular environ- mental conditions of the test are equal to or exceeded those for the McGuire Nuclear Station.

8. MECHANICAL PENETRATIONS

   Typical mechanical penetrations are shown on Figure 3-68.

   Mechanical penetration functional requirements, code considerations, analysis and design criteria are defined in Section 3.9.2.8.

   Figure 3-69 through Figure 3-72 provide details of the Containment Vessel plate thickness, size and spacing of ring and vertical stiffeners and other information for the as-built structure.

   The Containment Vessel overall dimensions and plate thicknesses are shown on Figure 3-73.

3.8.2.2 Applicable Codes, Standards and Specifications

The Containment Vessel is designed, fabricated, constructed and tested in accordance with Subsection B, Section III, of the ASME Boiler and Pressure Vessel Code, 1968 Edition, including all addenda and code cases through Summer 1970.

The Containment Vessel is analyzed as defined in Section 3.8.2.4.

Subsection B, Section III, does not make provisions for stamping pressure vessels of this geometry, and therefore the Containment Vessel has not received an ASME Code Stamp. The personnel lock has received a code stamp.

The shop fabrication, field erection, non-destructive testing, pressure testing and quality assurance documentation are in accordance with the ASME Code.

Regulatory Guide 1.19 is used for nondestructive testing of the Containment bottom liner with the following additions or exceptions:

1. C.I.b - Add liquid penetrant method as an acceptable means of testing liner seal welds.

   Liquid penetrant is used more successfully in detecting circular defects.

2. Delete C.I.c

   Non-destructive testing as required in C.I.b and leak chase pressure testing to peak Containment pressure as required in C.I.d have been successful in detecting leaks in seal welds. Vacuum box tests run at five psi do not necessarily detect leaks that might occur at peak Containment pressure.

   In addition to the pressure test required in C.I.d an additional ten minute peak Containment pressure leak chase pressure test is performed prior to and after placing concrete over the leak chase system. These tests are performed in order to detect leaks, if any, created during construction activities after the completion of C.I.d and during placement of concrete around the chase system.

3.8.2.3 Loads and Loading Combinations

The Containment Vessel steel shell is designed for the following loads:

1. Dead loads and construction loads.
2. Design basis accident.
3. External pressure.
4. Seismic loads.
5. Penetration loads.

Dead Load

The dead load includes the weight of the Containment shell and attachments. Construction loads include all loads imposed on the Containment shell during construction.

Design Basis Accident

The Design Basis Accident loads are the peak pressure and temperature developed inside Containment as a result of a rupture in the primary coolant system up to and including a double-ended rupture of the largest pipe.

The Containment Vessel peak pressure is 15 psig and the design temperature is as follows:

1. The water accumulated in the lower compartment after a loss-of-coolant accident has the peak sump temperature of < 200 degrees.
2. The Containment atmosphere temperature in the lower compartment below the ice condenser is 250 degrees after LOCA.
3. The Containment atmosphere temperature in the upper compartment adjacent to the ice condenser is 190 degrees.

See Chapter 6 for details of the Containment Design Basis Accident.

External Pressure

The external pressure is the internal vacuum created by an accidental trip of a portion of the Containment Spray System during normal unit operation.

The maximum design pressure is 1.5 psig. For details of the design vacuum pressure conditions refer to Section 7.6.4.

Seismic Loads

See Section 3.7, for the seismic design loadings.

Penetration Loads

Those loads are imposed upon the Containment Vessel due to penetration dead load or pipe reactions.

Table 3-33 lists the Containment Vessel load combinations and code requirements.

The material, fabrication (except for weld details) and allowable stresses for the transition torus meet the requirements of Subsection B, Section III of the ASME Code, 1968 Edition.

The welds and test channels on the torus are as shown in Figure 3-66. The torus is considered part of the bottom liner plate and all welds have been tested as specified in Section 3.8.2.7.

Plates used to transfer loads through the thickness of the plate are ultrasonically tested in accordance with Subparagraph N-321.1 of Subsection III of the ASME Code.

3.8.2.4 Design and Analytical Procedures

The Containment shell is designed based on the loads and loading combinations of Section 3.8.2.3 using the codes, standards and specifications defined in Section 3.8.2.2.

The Containment Vessel shell is analyzed to determine all membrane forces, moments and shears as a result of all specified static loadings.

The static load stresses and deflections that are in a thin, elastic shell of revolution are calculated by a numerical solution of the general bending theory of shells. This analysis employs the differential equations derived by E. Reissner and published in the "American Journal of Mathematics," Volume 63, 1941, pp. 177-184. These equations are generally accepted as the standards for the analysis of thin shells of revolution. The equations given by E. Reissner are based on the linear theory of elasticity and consider bending as well as membrane action of the shell.

The method of solution is the multisegment method of direct integration, which is capable of calculating the stress resultants and stresses of an arbitrary thin, elastic shell of revolution when subjected to any given edge, surface and temperature loads. This method of analysis was published in the "Journal of Applied Mechanics," Volume 31, 1964, pp. 467-476, and has found wide application by many engineers concerned with the analysis of thin shells of revolution.

The actual calculation of the stresses and stress resultants produced in the shell is determined by a computer program written by Professor A. Kalnins of Lehigh University, Bethlehem, Pennsylvania. The program makes use of the exact equations given by E. Reissner, and solves them by means of the multi-segment method. Applied loads can vary in meridional and circumferential directions. Boundary conditions and discontinuity loads may be specified and varied around the circumference of the shell.

The toroidal transition section between the cylindrical shell and the floor plate is designed by considering a torus fixed at the floor plate and imposing the movements of the Containment shell (at the point of attachment of the torus) as the opposite end boundary conditions. The surface loads on the torus are the Containment design pressure and temperature. The method of analysis for the torus is by Kalnins' program as defined above.

The stresses, stress resultants and displacements due to the response of a shell of revolution to the excitation of an earthquake are calculated by the method described in Section 3.7.2.1.

Localized Areas Around Large and Small Openings of the Containment Vessel

The localized areas around large and small openings of the Containment Vessel are analyzed and designed to meet the requirements of the ASME Code N450 of Section III, 1968 Edition, including the addenda through the Summer of 1970. A systematic numerical procedure is set up in order to analyze the small penetrations reinforcement and stress analysis in accordance with the ASME code requirements. The numerical procedure provides detailed requirements for thickening the shell around the opening or using a reinforcement ring. Separate analysis and design are performed on the equipment hatch penetration and the personnel air lock penetration.

Transient Dynamic Pressure Due to LOCA

The stresses, stress resultants and displacements of the steel Containment Vessel due to the transient dynamic pressure associated with a loss-of-coolant accident are determined by performing a dynamic analysis as follows:

1. Design Considerations:

   The rapid energy absorption capability of the Containment Vessel, due to the use of the Ice Condenser, maintains the Containment Vessel design at a low level as well as reducing the peak duration. This reduction in peak pressure keeps the shell thickness below the stress-relieving requirements of the ASME Boiler and Pressure Vessel Code.

2. Loss-of-Coolant Accident:

   A LOCA is a hypothetical double-ended rupture of a reactor coolant pipe in which the pressurized water flashes immediately into steam causing a pressure transient build-up in the Containment compartments.

   The Containment is divided into 37 compartments (see Figure 3-83 through Figure 3-85 for compartments layout).

   A pressure transient analysis is performed to determine the various compartment pressure transients resulting from a reactor coolant pipe break in each of the possible six breaks of the lower compartment elements.

3. Analytical Representation of the Shell:

   The Containment Vessel shell is idealized as an assemblage of horizontal conical frusta joined together at their nodal circles. As part of the model the stiffening girders are also included as finite elements. To incorporate the vertical stringers into the solution, the shell material is assumed to be elastically orthotropic, which implies that the value of the modulus of elasticity E has different magnitudes in the longitudinal and circumferential directions. The value of $E = 29 \times 10^6$ psi is used for the circumferential direction, while an equivalent value, E_{eq}, is used in the longitudinal direction, and is given by

   $$E_{eq} = Eh_{eq} / h_{shell}$$

   where, h_{eq} = the equivalent thickness of a smooth shell (without stringers) whose cross-sectional area in the longitudinal direction equals that of the real shell with stringers.

4. Analysis Procedure:

   To analyze the discrete Containment shell, the Hamilton's variational principle is used to derive the structure's equations of motion. This leads to the formation of the mass matrix, stiffness matrix and the load vectors. The equations of motion are solved numerically by the direct integration procedure in the time domain. The transient loading on the shell is first approximated by a Fourier Series with a finite number of terms. For each Fourier component, the stiffness and mass matrices and the corresponding load vector are formed and the equation of motion is solved. After solving for the response of all the Fourier terms, their contributions are summed to obtain the total response. The computer program

(Reference 1) employed for this analysis was originally written in Fortran IV language at the University of California at Berkeley. This program was modified and verified at Duke Energy.

5. Dynamic Loads:

   The time-dependent loads applied to the Containment Vessel are those loads caused by a blowdown of a major pipe of the reactor coolant system. Referring to Figure 3-83 through Figure 3-85 the compartments to be considered for the transient load analysis of the Containment Vessel are only those compartments in contact with the Containment shell inner surface. The pressure in each of these compartments varies with time as shown in Figure 3-75 for Compartments 7, 8 and 9. Figure 3-76 through Figure 3-81 show the pressure transients, due to break in element No. 1 of the Reactor Coolant System, in the remaining compartments in contact with the Containment shell. The dynamic load at each node of the discrete Containment shell is the resultant of pressures on an area extending between mid-points of adjacent elements. A time-history of dynamic forces at each node is developed for each specified break location. Since the load varies around the circumference, it is resolved into its Fourier components, both symmetrical and asymmetrical terms are used in the final Fourier representation of the pressure transient. Six Fourier components were employed in the presentation since convergence was found satisfactory. A typical comparison of the actual pressure distribution around the circumference versus the Fourier Series distribution for a given time step is shown in Figure 3-82.

Stability of the Containment Shell Under LOCA Conditions:

Since the LOCA loads on the Containment Vessel are not of a symmetrical nature, the stability of the Containment shell is investigated. Two basic stability criteria are employed. These criteria are:

1. Stability of the Overall Shell:

   Reference 4 is utilized in order to investigate the overall stability of the stiffened Containment shell. As a result of this investigation, it is concluded that the actual compressive stresses due to LOCA are much lower in magnitude than the critical buckling stresses as calculated from Reference 4.

2. Stability of Individual Shell Panels:

   Extensive investigation is performed on the stability of the individual shell panels isolated between two adjacent vertical and circumferential stiffeners known as local buckling. Two different buckling criteria are considered: buckling of panels as flat plate, Reference 5, and buckling of curved panels as prescribed in Reference 6.

For both cases of shell panels, the following panel loadings are investigated:

1. Axially loaded panels;
2. Axial and shear loaded panels.

The critical buckling stresses as determined in References 4, 5 and 6 are based on experimental data from tests performed on similar shell panels undergoing similar loading conditions. These buckling stresses are lower in magnitude than the critical buckling stresses calculated from theoretical and closed form mathematical solutions. Considering the above, it is concluded that the buckling factors of safety calculated for the McGuire Containment Vessel based upon References 4, 5 and 6 are conservative. Table 3-36 represents typical buckling factors of safety at several points on the Containment Vessel shell. The most critical factor of safety (the minimum value) is close to the base of the Containment. As shown in Table 3-36, the minimum buckling factor of safety for the Containment Vessel is 3.27. It can also be concluded from Table 3-36 that the minimum value of the buckling factor of safety of the shell panels result from considering the panels as curved plates under the combined action of axial and shear loadings.

Regulatory Guide 1.57 did not exist during the licensing, analysis and design of the McGuire Containment. As a result, Regulatory Guide 1.57 is not adopted in the Containment design of the McGuire Station.

Regulatory Guide 1.57 references Subsection NE of Section III of the ASME Code. Subsection B of Section III of the ASME Code was in effect during the licensing of McGuire; therefore, Subsection NE was not used in the design of the Containment Vessel. Regulatory Guide 1.57 references a minimum factor of safety for stability of 2.0 computed based upon analytical solutions for the appropriate load combinations. The factors of safety for stability used in the actual design of the Containment Vessel are based upon test results which are more conservative than being based upon analytical results.

In order to evaluate the design of the McGuire Containment Vessel for 49 compartments as compared to the original 37 compartments, a comparison is provided for the McGuire Containment Vessel and the Catawba Containment Vessel

The layout of the compartments inside the McGuire Containment Vessel is as outlined in Figure 3-83 to Figure 3-85. The compartment layout inside the Catawba Containment is shown in Figure 6-29 to Figure 6-32. The two Containments are identical except for the circumferential stiffeners arrangement and plate thickness as shown in Figure 3-74. The two Containments are analyzed using the same techniques as previously discussed in Section 3.8.2.4. The mass and energy release values currently tabulated in Section 6.2 of the McGuire FSAR were used to generate the compartment pressures utilized in the Catawba Containment analysis.

The results of the preliminary Catawba containment analysis can be used for comparison utilizing the buckling analysis procedures previously discussed (References 4, 5 and 6). The model of the two containments and a summary of the resulting factors of safety against buckling are shown in Figure 3-74.

From this comparison, it is concluded that the buckling factor of safety at the most critical point on the McGuire Containment Vessel is not significantly altered by the change from a 37 subcompartment to 49 subcompartment layout.

As discussed in Section 6.2, the TMD model was revised to include 53 subcompartments. Containment has been analyzed using the pressures described in Section 6.2.1. The minimum factor of safety using these pressures was between the values given in Figure 3-74 for the 37 and 49 subcompartment pressures.

Design Bases

The Containment Vessel is designed to assure that an acceptable upper limit of leakage of radioactive material is not to be exceeded under design basis accident conditions, including the LOCA.

The Containment Vessel utilizes the ice condenser concept for energy absorption during a loss-of-coolant accident. The rapid energy absorption capability maintains the Containment Vessel design pressure at a low level as well as reducing the peak duration. See Section 6.2 for details and description of the ice condenser design and function.

The use of the ice condenser requires that the Containment Vessel be divided into three major volumes. The lower volume houses the Reactor Coolant System, the intermediate volume houses the ice condenser energy absorption system, and the upper volume contains the air after passing from the lower volume through the ice condenser. Compartments have been designed for peak differential pressures (preliminary design pressures furnished by Westinghouse plus a 40 percent margin) due to a severance of the largest pipe within the enclosure or flow into the compartment from a break in an adjacent compartment.

The Containment Vessel is designed to accommodate all calculated external pressures. Vacuum breakers are not required.

The Containment shell plate (cylinder and dome) is not exposed to ground water and is protected by the Containment Annulus. The Containment bottom liner plate is anchored to the Reactor Building foundation which is constructed of reinforced concrete with waterstop in all construction joints. No water-proofing is provided.

The Containment liner is designed to function as a leak-tight membrane and is not required to function as a structural component.

The bottom liner plate is 1/4 inch carbon steel of which its total thickness is available for corrosion allowance.

The materials used for the design and construction of the Containment Vessel are given in Table 3-34.

Containment Vessel Coatings

The interior steel surface of Containment Vessel and penetrations are cleaned and coated with materials meeting ANSI N101.2-1972, Section 1.4.2.2, Design Basis Accident Environmental Conditions for PWR's. The environmental conditions for the Containment are listed in Section 6.2.1.1 for normal operating conditions and Section 6.2.1.2 for DBA conditions. The maximum integrated radiation dose is 3×10^7 Rads during normal operating conditions and 2×10^8 Rads for DBA conditions.

All exterior surfaces of the Containment Vessel and penetrations are coated with a suitable system for outdoor exposure.

3.8.2.5 Structural Acceptance Criteria

The Containment Vessel is designed and fabricated in accordance with the provisions of Subsection B, Section III, of the ASME Code. Refer to Section 3.8.2.4 for more details on the Containment design basis and its compliance with codes and specifications.

3.8.2.6 Design Loading Combination Stress Limits

Table 3-33 summarizes the Containment Vessel loading combinations and code requirements for the Containment design. As shown in the table, the stress limits are as defined in ASME Section III, Subsection B.

3.8.2.7 Steel Containment Tests and Inspection

3.8.2.7.1 Preoperational Testing and Inspection

1. Structural Testing

   The Containment shell, personnel airlocks and equipment hatch are inspected and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection B.

2. Leakage Rate Tests

Bottom Liner Plate: The bottom liner plate welds are inspected, prior to placing fill concrete, in accordance with the following:

1. Dye penetrant examinations are performed in accordance with Appendix 8 of Section VIII of the ASME Boiler and Pressure Vessel Code.
2. Upon completion of the dye penetration test, the weld seams are covered with test channels and pressure tested. All detected leaks are repaired and retested.

Personnel Airlocks and Equipment Hatch: The personnel airlocks are pressurized and a Type B leak rate test is performed as described in Section 6.2.1.6.

The double o-ring seals in the equipment hatch are tested for leakage.

Containment Leakage Rate Test: Upon completion of all penetration, personnel airlocks, equipment hatch, bottom liner plate and structural testing, a leakage rate test is performed on the Containment as described in Section 6.2.1.6.

3.8.2.7.2 Postoperational Surveillance

1. Structural Integrity

   The Containment Vessel shell has been designed, fabricated and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection B.

   The Containment shell is protected by the Reactor Building from adverse environmental conditions. In addition, under operating conditions, the shell does not experience design pressure and temperature load cycling. It is therefore contemplated that additional structural testing of the Containment shell other than the initial structural test is not necessary. Visual examinations are conducted as outlined in the Technical Specifications.

2. Leakage Rate Testing and Inspection

   Periodic leakage rate tests of the Containment Vessel, testable penetrations, personnel locks and equipment hatch are conducted to verify leak tightness integrity as described in Section 6.2.1.6.2.

3.8.3 Concrete and Structural Steel Internal Structures of the Steel Containment

3.8.3.1 Description of Internal Structures

The Internal Structures enclose the primary coolant system and provide biological shielding and pressure boundaries for the lower, intermediate and upper volumes of the Containment interior. The Internal Structures are anchored to the Reactor Building foundation as shown in Figure 3-54. The Internal Structures are primarily poured-in-place reinforced concrete.

Further details of the Internal Structures are as follows:

1. Containment Basement Floor Slab

   The Containment basement floor slab is two feet thick reinforced concrete, circular in shape. The crane wall and reactor vessel cavity wall dowels are embedded in the slab.

2. Reactor Vessel Cavity Wall

   The reactor vessel cavity wall is an eight feet-six inches thick cylindrical wall. The reactor vessel cavity wall provides steel support pads for the Reactor Vessel. The steam generators lower lateral support embedments are anchored to the reactor vessel cavity wall (see Figure 5-30 for embedments of the steam generator lower lateral support).

3. Upper Reactor Cavity

The upper reactor cavity wall is approximately four feet thick reinforced concrete wall lined with 3/16 inch stainless steel plate and is part of the refueling canal.

Openings are located on top of the cavity venting into the lower containment. These vent openings provide pressure relief during a LOCA in the primary cavity. The control rod drive mechanism missile shield covers the compartment and is anchored by means of embedded anchor bolts.

4. Refueling Canal

The refueling canal provides water storage for refueling and a storing facility for the upper and lower reactor internals. The floor of the refueling canal is 3 ft. thick and the wall thickness is four feet. The inside face of the walls and floor of the refueling canal are lined with 3/16 inch stainless steel plate. The canal area is separated from the upper reactor cavity by means of a three feet thick reinforced concrete gate which is removed during refueling.

5. Crane Wall

The crane wall provides secondary shielding for the primary coolant system, support for the polar crane and the ice condenser components. The crane wall is a three feet thick reinforced concrete cylinder with an inside diameter of 83 feet. The wall has openings as required by the ice condenser and penetrations. Embedments are located in the crane wall for the steam generators and reactor coolant pumps lateral supports as well as piping, cable trays and miscellaneous ladders and platforms.

6. Steam Generator Compartments

The steam generator compartments are enclosures for the steam generator and an extension of the lower Containment. The end walls of the steam generator enclosures are two feet thick reinforced concrete and the center wall between two adjacent compartments is three feet thick. Each steam generator compartment has a cylindrical steel shell 3/4 inch in thickness which is anchored to the end wall and the center wall of the compartment. A steel dome is provided on top of each of the steam generator compartments. The steel dome can be removed for inservice inspection and regular maintenance of the steam generators. The shell and dome material is SA-516, Grade 70, and they are anchored to the compartment walls by high strength anchor bolts SA-320, Grade L43. Pressure seals and/or gaskets are provided between the steel and the concrete surfaces in order to limit the bypass leakage from the lower to upper compartments during a LOCA.

7. Pressurizer Compartment

The pressurizer compartment houses the pressurizer and provides an extension of the lower compartment. The compartment is surrounded by a cylindrical two feet thick reinforced concrete wall, and a roof slab which is the same thickness. Lateral pressurizer supports are anchored to the cylindrical wall of the enclosure. A pressure hatch is provided in the top slab for access to the pressurizer compartment in order to perform inservice inspection and general maintenance.

8. Divider Deck

The divider deck is the main divider barrier between the lower and upper compartments inside the Containment. The divider deck is a two and one-half feet thick reinforced concrete slab. Several pressure hatches are provided for access and pump removal. The pressure hatches are anchored to the deck by high strength bolts, and are designed for the peak differential pressure between the upper and lower compartments. Gaskets are provided between the surface of the pressure hatches and the concrete in order to limit the bypass leakage from the lower to the upper compartment during a LOCA.

9. Equipment Floor

The equipment floor (elevation 738 + 3) provides support for the accumulator tanks, lower Containment ventilation units and miscellaneous light equipments. The equipment floor is a two feet thick reinforced concrete slab. At one end the equipment floor is cantilevered from the crane wall and supported by steel columns at the other end.

10. Ice Condenser Floor

The ice condenser floor (elevation 766 + 8 1/2) provides the main support for the ice condenser lower support structure as well as the vertical loads of the ice baskets. The ice condenser floor is a two and one-half feet reinforced concrete slab cantilevered from the crane wall at one side and supported on steel wide flange columns at the other end. High strength bolts embedded to the floor provide the anchorage for the ice condenser lower support structure.

11. Pressure Seals and Gaskets

Pressure seals and/or gaskets are provided at all locations where it is necessary to limit or eliminate bypass leakage during a LOCA. The locations of the pressure seals between the different structural components inside the Containment are indicated on general arrangement drawings as follows:

Ice Condenser Seals	Figure 1-13, Figure 1-15 & Figure 1-16
Steam Generator Enclosures Seals	Figure 1-13, & Figure 1-16
Operating Deck Hatches and Access Opening Seals	Figure 1-11 to Figure 1-16
Refueling Canal Seals	Figure 1-12
Pressurizer Enclosure Seals	Figure 1-15

The seals are required to remain functional to ensure that the maximum permissible bypass leakage area, between the upper and lower compartments, is not exceeded. The locations of all seals and typical seal details are shown on Figures 3-99 to 3-101. The figures show for each seal; its location, operating temperature, design pressure, radiation level and maximum movement, as applicable.

Based on the normal environment during unit operation, the seals are expected to last the life of the unit (40 years). However, test coupons located in the vicinity of the functional seals necessary to eliminate bypass leakage are used to determine the degree of degradation of the material due to normal operating conditions. The design properties of the seal materials are shown in Table 3-40 for the membrane seals as well as the compressible seals. The seals would be replaced if the test coupon results indicate a significant change in the applicable material properties, such as tensile properties for membrane seals and durometer readings for compressible seals. Table 3-41 lists the minimum acceptable seal properties.

12. Accumulator Wing Walls

The accumulator wing walls are two feet thick walls that house the accumulator tanks. They extend from the equipment floor (elevation 738' + 3") up to the ice condenser floor (elevation 766' + 8 1/2"). (See Figure 3-54)

After construction and prior to operation, a detailed and thorough visual inspection is performed on all potential leak paths to determine the need for additional seals and to verify that existing seals are installed properly. The inspection is repeated during the startup test program with the Reactor Coolant System at hot conditions. As a part of this inspection, a check is made for any unexplained airflow between the upper and lower containment.

Overall sketches of the Internal Structures are as shown in Figure 1-9 through Figure 1-16.

The volume boundaries are designed to withstand the differential volume pressures as outlined in Section 3.8.3.3.

3.8.3.2 Applicable Codes, Standards and Specifications

The interior structures have been designed according to the applicable codes and specifications tabulated in Table 3-31.

3.8.3.3 Loads and Loading Combinations

3.8.3.3.1 Static Analysis

The Containment interior structural components have been designed for operating and accident loads as follows:

1. Operating Loads - The interior structure has been designed for dead and equipment loads and reactions from piping systems.

   The structural components exposed to thermal gradients due to the presence of the ice condenser are designed for the maximum gradient occurring during normal operation.

2. Accident Loads - The interior structural components are designed for the peak pressure differential occurring during the accident. Each break location, as defined in Section 3.6 within the lower compartment is evaluated to establish the peak pressure differential occurring across a structural component (see Figure 3-83 through Figure 3-85 for the compartment's layout). The total number of compartments used for the design is 37 compartments. Table 3-29 provides a tabulation of the peak positive and negative differential pressure between compartments, the time after the start of the blowdown at which the peak occurs, the break location which produces the peak pressure and if the peak pressure is produced by a hot or cold leg break. Also, shown on the same table are the design differential pressures which are 40 percent larger than the preliminary pressures furnished by Westinghouse. The peak design temperature in the compartments is 240 degrees F, however, the peak does not occur until the pressure across a component has stabilized (see Section 6.2).

   NEW COMPARTMENT LAYOUT

   At a later date a new Containment interior layout of compartments was introduced. The total number of compartments in the new layout is 53 (versus 37 compartments originally). Refer to Section 6.2.1.3.1.2 for details of the new compartments layout and the corresponding differential pressure across the compartments (see Table 6-2). Table 3-37 shows a comparison of the design pressures based on the 37 compartments layout, the design pressures based on the 53 compartments layout, and the differential design pressures actually employed in the design and analysis of the Containment interior structural components.

It is concluded, therefore, that the pressures used for the design of the interior structures are adequate.

3. Jet Forces - The interior structure subcompartment structural elements are designed for pipe rupture jet impingment forces as defined herein.

   Jet impingement forces are established and defined as follows:

   a. Pipe break size and locations are as described in Section 3.6 of the FSAR.

   b. Jet force orientation was established as follows:

      1) For circumferential breaks, the jet orientation is limited to elastic deflections at the ruptured end of the pipe relative to the nearest anchor point if the pipe section remains elastic at the anchor when a hypothetical load equal to the jet load is applied perpendicular to the axis of the pipe at the ruptured end. If the pipe section is partially yielded but a full plastic hinge does not form, the jet orientation is established by a hypothetical lateral displacement of one pipe diameter relative to the axis of the pipe. If a plastic hinge does form at the anchor, lateral restraints are provided or the jet orientation will be established considering all orientations relative to the hinged anchor.

      2) For longitudinal breaks, the jet orientation shall be established at the break location in a direction perpendicular to the axis of the pipe.

      Pipe sleeves, physical restraints and obstructions have been considered in evaluating jet orientation and impingement area.

      Jet impingement pressures were established as defined in Reference 1 except that the jet dispersion angle has been a ten degree one-half angle relative to the orientation axis.

      To account for the dynamic response of an object subjected to a jet force, a dynamic load factor (DLF) based upon the ratio of natural frequency of the model to the duration of the jet force as defined in Reference 3, has been used to establish an equivalent static load.

      The overall interior structure is designed for the maximum uplift, horizontal shear and overturning moment.

      Each break location in the lower compartment has been evaluated to establish the maximum uplift, horizontal shear and overturning moments on the interior structure. Table 3-29 gives a tabulation of the maximums, the time at which the maximum occurs, and identified the break producing the maximum. The related forces occurring at the same time as the maximums are combined with the maximum for the final design.

      The loadings described above were utilized in the design of the interior structure. Subsequent to this design, revised postulated pipe break criteria were introduced in Section 3.6. The load resultants and differential pressures presented in Table 3-28 and Table 3-29, respectively, are not applicable as listed, but represent an upper bound for loadings resulting from a postulated pipe break. The final compartment differential pressures are in all cases less than those used for design.

3.8.3.3.2 Dynamic Analysis

The loads on the Containment internal structure as a result of a base excitation are determined by a dynamic analysis as described in Section 3.7.2.1 (2).

3.8.3.3.3 Loading Combinations

Loading combinations and code requirements for the interior structure are summarized in Table 3-39.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Design Loads

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

The Containment interior structure has been designed for the following loads:

1. *Dead load*
2. *Compartment pressure and/or pressure differentials as described in Section 6.2.1*
3. *Equipment operating loads*
4. *Pipe reactions*
5. *Seismic: See Section 3.7 for seismic criteria*
6. *Pipe rupture loads: The Containment interior structure dynamic model has been combined with the dynamic model of the primary system to evaluate the overall structural effect of a loss-of-coolant accidents. For a more detailed description of the combined analysis, refer to Section 6.2.1.*
7. *Internal Missiles: The Containment interior structure has been designed to withstand internal missiles as defined in Section 3.5.*

   The analysis of the Containment interior floor slabs at elevation 778 + 10 and 782 + 4 between the lower and upper compartments is performed by two independent and separate engineering groups (A and B on Figure 3-86) of the Civil and Environmental Division. Each of the engineering groups is responsible to the Structural Principal Engineer. Each independent analysis is checked by qualified engineers within the respective group. The structural Principal Engineer having overall responsibility for the design and analysis provided surveillance by review of concept, conformity with codes and criteria, execution. Any discrepancy is viewed by the Structural Section Principal Engineer and the Chief Engineer. Assumptions, calculations, design techniques and methods are examined for corrections and further work performed, as necessary to resolve any disagreements. Figure 3-86 shows a partial organization chart of the Civil-Environmental Division and illustrates the independence of the two design groups.

8. *Pipe rupture jet impingement:*

   The interior structure is designed for jet impingement as defined in Section 3.8.3.3. Typical jet impingement and analyses of a structural component are as follows:

The design of the Internal Structure is in accordance with the requirements of the applicable codes as shown in Table 3-39. The forces and moments of each of the Internal Structure components are determined from established analytical procedures and computer structural analysis programs. A brief outline of the analysis and design methods are as follows:

1. *Containment Floor Slab:*

   The Containment floor slab is a fill slab, therefore, minimum reinforcement is provided as specified in the ACI-318-63 Code.

2. *Reactor Vessel Cavity Wall:*

   The reactor vessel cavity wall is designed to accommodate loads generated by compartment pressure from postulated pipe break and support. The reactor cavity wall is analyzed as a thick cylindrical wall and the steel reinforcement is basically a hoop reinforcement pattern.

3. *Upper Reactor Cavity:*

   The upper reactor cavity is analyzed as a space finite elements model together with the refueling canal walls and floor. The upper cavity wall is designed to accommodate loads due to postulated pipe break which consist of internal and external pressures.

4. *Refueling Canal:*

   The refueling canal walls and floor are analyzed as an integral part with the upper reactor cavity space finite elements model.

5. *Crane Wall:*

   A space finite elements mathematical model is utilized in order to perform the analysis on the crane wall. The crane wall is designed as a secondary barrier for differential pressures from postulated pipe breaks across the various compartments of the Interior Structure. Asymmetric loadings due to postulated pipe breaks are induced on the crane wall. For this type of loading, the computer program No. 2 of Section 3.8.3.4.2 of the FSAR is utilized to perform the crane wall analysis as a shell of revolution subject to asymmetric loads. The polar crane is designed for the loadings as defined in Table 3-2. Use of the polar crane is described in Sections 9.1.4 and 9.1.5. This use does not have an influence on the safe shutdown of the reactor.

6. *Steam Generator Compartments:*

   The steel dome of the steam generator compartment is analyzed as a thin shell of revolution employing Kalnin's program (No. 1) described in Section 3.8.3.4.2. The steel and the reinforced concrete parts of the compartment are idealized together as a plane finite elements model. The steel dome, the side walls and the cylindrical steel shell of the steam generators compartments are designed to accommodate the internal pressure due to main steam line rupture. The design of the steel components meet the requirements of the ASME Code, 1968 Edition, Subsection B. (The high strength embedded anchor bolts meet requirements of Section A for normal operation and 1986 code, Appendix 'F', for faulted condition.) The connection between the steel and concrete portions of the steam generator compartment is designed to permit removal of the steel portion. The connections consist of embedded anchor bolts and exposed nuts which can be removed when desired.

7. *Pressurizer Compartment:*

   The pressurizer compartment is designed for the internal pressure due to pipe rupture inside the compartment. A plane finite element model is utilized in analyzing the pressurizer compartment.

8. *Divider Deck:*

   A plane finite elements plate bending model is utilized to perform the analysis of the deck. The divider deck, which is the main pressure barrier between the lower and upper

Containment interior compartments, is designed to accommodate the compartment differential pressure from postulated pipe breaks and the jet impingement loads.

9. *Ice Condenser Floor:*

   The ice condenser floor is designed for the differential pressures from postulated pipe breaks, as well as the weights of the ice condenser components and the reaction loads from the ice condenser lower support structure. A plane finite element model is utilized in performing the analysis of the ice condenser floor.

10. *Equipment Floor:*

    A plane finite elements mesh is utilized to represent the equipment floor. The analysis is performed by employing some of the computer programs outlined in Section 3.8.3.4.2. The equipment floor is designed for the compartment differential pressure from postulated pipe breaks and the weight of the equipment.

11. *Accumulator Wing Walls*

    The accumulator wing walls are designed for differential pressures from postulated pipe breaks. A plane finite element model is utilized in performing the analysis of the walls. The adequacy of the walls has been demonstrated using the ultimate strength design method as outlined in Standard Review Plan 3.8.3 for the load combination: U=D+L+Ta+Ra+1.5Pa.

3.8.3.4.2 Computer Programs for the Structural Analysis

The following computer programs are employed in the analysis of Category 1 structures:

1. For the stresses, stress resultants and displacements produced in a thin shell of revolution due to static and seismic loads: A computer program written by Professor A. Kalnins of Lehigh University, Bethlehem, Pennsylvania. Refer to Section 3.7.2 and Section 3.8.2.4 for description of the program.
2. For the stresses, stress resultants and displacements of a shell of revolution due to the transient dynamic pressures associated with a loss-of-coolant accident: A computer program originally written at the University of California, Berkeley. Refer to Section 3.8.2.4 for description of the program.
3. For seismic response of structures that can be idealized as multi-mass systems: A computer program based on the theory presented in Sections 3.7.2.1 and 3.7.2.6.
4. For stresses and displacements of frames subject to static loads: The Structural Design Language (STRUDL) computer program, latest version and modification. Refer to the STRUDL manuals published by MIT for program description.
5. For linear equilibrium problems of general structures: The ELAS75 program. Refer to latest manuals published by Duke University, School of Engineering, December 1971.

3.8.3.4.3 Computer Program Testing

Prior to using any computer program for production problems, it is Duke's standard procedure to test the applicability and validity of the program using test problems which are similar to the production problem. The computer programs' solutions to the test problems are compared to other solutions as follows:

1. Solutions obtained by other computer programs which are published in technical literature.
2. Closed form solutions found in standard text books.

3. Solved examples in standard text books.
4. Hand calculations for the simpler test problems.

In each case, the results are found to be substantially identical.

Whenever possible, the same production problem is analyzed using two different computer programs from the list of programs in Section 3.8.3.4.2 to verify the validity of the results. For example, the pressure deck between the upper and lower compartments is analyzed by two separate groups using two separate methods as defined in Section 3.8.3.4.2. The two programs are the STRUDL and ELAS programs. Results obtained from these two programs indicated good correlation which verifies the validity of results.

3.8.3.4.4 Summary Comparisons of Computer Program Test Problems

1. Programs 1) and 2) of the list in Section 3.8.3.4.2 were used to obtain the natural frequencies and mode shapes of a model of the Containment Vessel. Comparisons of Kalnins' program results and those obtained by the Finite Element Method of the program in 2) are shown in Table 3-38 and Figure 3-87 through Figure 3-94. The results are compatible and verify the results obtained from the programs. It should be noted that the two programs use two completely different approaches to obtain the solution.
2. The programs number 4) and 5) of the list in Section 3.8.3.4.2 (STRUDL and ELAS programs) were used to obtain the solution of the pressure deck between the upper and lower compartments. The finite element representation of plate bending for each program is shown in Figure 3-95 and Figure 3-96. However, different element types and layouts are employed. The results, as shown in Figure 3-97 for the fixed condition and Figure 3-98 for the pinned condition, compare well enough to assure the validity of the answers.
3. Numerous other test problems were compared with standard closed form solutions found in text books as well as published literature.

3.8.3.5 Structural Acceptance Criteria

The interior structural elements are designed in accordance with the codes, standards and specifications of Table 3-31, under any of the loading combinations of Table 3-39. The limits on the stresses, strains and deformations are as prescribed in the codes and specifications of Table 3-31.

3.8.3.6 Materials, Quality Control and Special Construction Techniques

The following materials are used for design and construction of the Containment interior structure:

1. Concrete - 3000 psi and/or 5000 psi strength based upon 28-day test.
2. Reinforcing Steel - ASTM A615, Grade 40 and/or 60.
3. Structural Steel Shapes - ASTM A-36.
4. Embedded Plates - ASTM A-36 and/or SA-516.
5. Stainless steel standard shapes - ASTM A276, Type 304.
6. Anchor Bolts - ASME SA-320 L43 or A36.
7. Stainless Steel Plates - ASTM A240, Type 304.

The materials and quality control procedures are in accordance with the requirements of the codes, standards and specifications of Table 3-31, and the quality assurance requirements of Chapter 17. Since mechanical rebar splicing (e.g., Cadwelds) are used in the interior structure, such splices meet the requirements of Regulatory Guide 1.10.

There are no special construction techniques employed in the construction of the interior structure.

Aging effects associated with the structures and components are managed through periodic inspections of the Reactor Building Interior Concrete, Foundation Slab, and Interior Structures (as committed to in Section 4.22 of MCS-1274.00-00-0016, Revision 2, "McGuire License Renewal Commitments").

3.8.3.7 Testing and Inservice Surveillance Requirements

There are no testing or inservice surveillance of the interior structure beyond those quality control tests performed during the construction of this structure.

3.8.4 Other Category I Structures

3.8.4.1 Auxiliary Building

3.8.4.1.1 Description of the Structure

The Auxiliary Building is a poured-in-place reinforced concrete structure as shown in Figure 1-2 through Figure 1-7. The Auxiliary Building houses the Nuclear Steam Supply System auxiliary equipment, electrical equipment, control building, fuel pools, diesel generators related piping and cabling. The structure is designed to provide biological shielding and missile protection in applicable areas. Components of the Auxiliary Building are as follows:

1. <u>Diesel Generator Building</u>

   The Diesel Generator Buildings which house the diesel generators are reinforced concrete Category I structures with plan dimensions of approximately 63 feet by 77.5 feet (see Figure 3-102). The Diesel Generator Buildings are founded on rock on/or fill concrete with a seven feet thick reinforced concrete mat. The mat includes various equipment trenches, pits and two high capacity sumps. Structural walls and roof slabs have concrete thicknesses of 36 inches and 28 inches respectively. Each building is flanked on two sides by compacted fill and on the opposite sides by the Auxiliary Building and the Turbine Building. The maximum roof slab clear span is 27 feet and the building length of 77.5 feet is divided into four bays with center-to-center spacings of from 12 feet to 29 feet. Approximately 82 percent of the Diesel Generator Buildings is below plant grade.

2. <u>Spent Fuel Building</u>

   The spent fuel buildings house the spent fuel pool and cask handling area. A 125 ton bridge crane is provided for fuel cask handling. Each pool has four feet reinforced concrete walls lined with 3/16 inch thick stainless steel liner plates. The stainless steel liner has a leak chase system that provides a method of continuous testing for leaks throughout the life of the station. The spent fuel pool can be separated from the fuel transfer upending canal. The upending canal can be dewatered independent of the main pool. Provisions for maintaining water level and pool protection are in compliance with the requirements of Regulatory Guide 1.13. The physical dimensions of the spent fuel pool are approximately 67 feet in length, 21 1/2 feet in width and the maximum depth of the water in the pool is approximately 40 feet. The thickness of the roof of the spent fuel pool is 28 inches for turbine missile protection. For further details of the pool, refer to Figure 3-104 through Figure 3-107.

The concrete structure encloses the spent fuel pool except for the north end of the structure which opens to the cask handling area. This area is enclosed by a non-Category I steel structure that also encloses the new fuel storage vault. An evaluation has been performed to confirm that the failure of the steel structure would not cause a decrease in the degree of subcriticality provided for the spent fuel or new fuel storage arrays.

3. The Control Building

The Control Building houses the control room, battery room and the cable room. The Control Building is a reinforced concrete Category I structure, consisting of grid of frames connected by continuous slabs, walls and columns. The roof slab of the Control Building is 28 inches. Refer to Figure 3-108 through Figure 3-110 for further details on the location of the Control Building as well as necessary details of wall thicknesses and floor dimensions.

3.8.4.1.2 Applicable Codes, Standards and Specifications

The applicable codes, standards and specifications in the design of the Auxiliary Building are given in Table 3-31.

3.8.4.1.3 Loads and Loading Combinations

1. Static Analysis

The Auxiliary Building is statically designed as a series of rigid frames subject to the loadings outlined in Section 3.8.4.1.4. The Structural Design Language (STRUDL) computer program is used to perform this analysis.

2. Dynamic Analysis

The seismic loadings on the Auxiliary Building are determined in accordance with the dynamic analysis procedure described in Section 3.7.2.1.

3. Loading Combinations

Loading conditions, combinations and code requirements for the Auxiliary Building design and construction are summarized in Table 3-42 and Table 3-43.

3.8.4.1.4 Design and Analysis Procedures

The Auxiliary Building is designed for the loading combinations outlined in Table 3-43. The wind loadings, tornado loadings, snow and ice loads, soil and water pressure are the same as for the Reactor Building presented in Section 3.8.1.4. Refer to Section 3.7 for the seismic loadings on the Auxiliary Building. The ultimate strength method of design is used with the load factors, maximum allowable stresses and load combinations as defined in Table 3-43. Masonry construction is designed and reinforced to remain functional under the above applicable loading conditions.

The Preliminary Safety Analysis Report was submitted and structural analysis and design of the structural elements outside the Containment were initiated prior to establishing design criteria as defined in “Structural Design Criteria for Evaluating the Effects of High-Energy Pipe Breaks on Category I Structures Outside the Containment” issued by the Regulatory Staff. However, Category I structures and structural elements outside the Containment (Auxiliary Building) are designed for the effects of high-energy breaks. Table 3-42 and Table 3-43 are updated to include the criteria actually used with respect to loading conditions and combinations for the design of Structural elements subjected to the effects of high-energy pipe breaks.

The overpressure loads of 1.8 psi as given in Section 2.2.3 due to a postulated explosion is combined with dead load and operating load. This pressure acting horizontally is less than the pressure of the tornado wind pressure. Considering the 1.8 psi as external pressure, the forces and moments due to the postulated explosion are found to be smaller in magnitude than those from the tornado wind pressure (Table 3-43). It is concluded that the structural design is adequate to withstand this load condition.

Fuel cask area is separated from fuel storage area to assure that fuel remains flooded in the event an accidental cask drop ruptures the bottom liner. The separation is a concrete wall with a gated slot for handling fuel.

The fuel pool and fuel unloading area are shown on (Figure 9-1 and Figure 9-2). The structureal design of the cask crane conforms to the criteria established in Electric Overhead Crane Institute Specification Number 70. In addition, the crane and its components are designed for seismic forces in the unloaded condition. (Hold-down devices are provided.)

Mechanical parts of the crane are designed to have a minimum safety factor of 5 when under rated load and based on the ultimate strength of the materials used. The main hook was load-tested to twice the rated load and then magnaflux-examined. The crane is load-tested in the Field at rated load prior to normal operation.

The arrangement of the fuel pool and unloading area prohibits the cask from being moved over stored fuel as shown in Figure 9-1 and Figure 9-2.

The maximum height to which the cask may be lifted is less than 30 feet when being moved through the unloading area. The cask is designed to survive, leak tight, a drop from the maximum height in the unloading area. No vital equipment is located in the path of the shipping cask as it travels from the fuel pool to a rail car or a truck.

The fuel pool and fuel cask area are lined with 3/16 inch thick stainless steel conforming to ASTM A-240, Type 304.

The cask crane stops are located in a position to prevent the cask from being moved into the fuel pool area. In the event of an accident with the cask in the cask handling area, the center of gravity of the cask remains over the cask area assuring that the cask does not fall into the fuel pool.

In Section 3.8.3.4.2 a list of the computer programs used in the structural analysis is given, as well as the testing procedure of these computer programs is given in Section 3.8.3.4.3.

3.8.4.1.5 Structural Acceptance Criteria

The Auxiliary Building is designed in such a way that the elastic behavior of the structure is maintained when subjected to any of the loading combinations employed for the design as defined in Table 3-42 and Table 3-43.

The stress limits used for the design of the Auxiliary Building are as defined in the codes, standards and specifications summarized in Table 3-31.

3.8.4.1.6 Materials, Quality Control and Special Construction Techniques

The following materials are used in the design and construction of the Auxiliary Building:

1. Concrete - 3000 psi and/or 5000 psi strength based upon 28-day test.
2. Reinforcing Steel - ASTM A615, Grade 40 and/or 60.
3. Structural Steel Shapes - ASTM A-36.
4. Embedded Plates - ASTM A-36 and/or SA-516.
5. Stainless Steel Standard Shapes - ASTM - Type 304.
6. Anchor Bolts - ASME SA-320 L43.
7. Stainless Steel Plates - ASTM A240, Type 304.

The materials and quality control procedures are in accordance with the requirements of the codes, standards and specifications of Table 3-31, and the quality assurance requirements as outlined in Chapter 17 of the FSAR.

Wherever Cadwelds are used, the requirements of Regulatory Guide 1.10 are met. There are no special construction techniques employed in the construction of the Auxiliary Building.

3.8.4.1.7 Testing and Inservice Surveillance Requirements

The spent fuel pool liner plate welds are inspected in accordance with the following:

1. Dye penetrant examination is performed in conformance with Appendix VIII of Division 1 of Section VIII of the ASME Boiler and Pressure Vessel Code.
2. Upon completion of the dye penetrant test, the weld seams are covered with a leakage chase system and then pressure tested. All detected leaks are repaired and retested.

Duke Energy Experience with Mechanical Splicing

Duke Energy has an extensive experience with reinforcement bar mechanical splicing (Cadwelds). The following table presents a history of the tensile strength tests performed on Cadweld splices at the McGuire Nuclear Station as of July, 1974:

Bar Grade	Bar Size	Splice Position	Number Tests	Number Rejects
40	11	Vertical	72	0
40	11	Horizontal	3	0
40	9	Horizontal	1	0
40	7	Vertical	2	0
60	18	Vertical	19	0
60	18	Horizontal	3	0
60	11	Vertical	28	0
60	7	Vertical	1	0

3.8.4.2 New Fuel Storage Vault

3.8.4.2.1 Description of Structure

The New Fuel Storage Vault is a poured in place reinforced concrete structure as shown in Figure 9-1 and Figure 9-2. The New Fuel Storage Vault houses the new fuel before the fuel is transferred to the spent fuel pool area.

3.8.4.2.2 Applicable Codes, Standards, and Specifications

The applicable codes, standards, and specifications used in the design of the New Fuel Storage Vault are given in Table 3-31.

3.8.4.2.3 Loads and Loading Combinations

Loading conditions are the same as for the Auxiliary Building and are defined in Table 3-42. The loading combinations for which the New Fuel Storage Vault is designed are defined in Table 3-44. Tornado missiles are defined in Table 3-8.

3.8.4.2.4 Design and Analysis Procedures

The New Fuel Storage Vault is designed as a series of rigid frames and is subject to the requirements as prescribed for a Category I structure. The structural design language (STRUDL) computer program is used to perform this analysis.

The seismic loadings on The New Fuel Storage Vault are determined in accordance with the dynamic analysis procedure described in Section 3.7.

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The overpressure load of 1.8 psi as given in Section 2.2.3 due to a postulated explosion is combined with dead load and operating load. This pressure acting horizontally is less than the pressure of the tornado wind pressure. Considering the 1.8 psi as external pressure, the forces and moments due to the postulated explosion are found to be smaller in magnitude than those from the tornado wind pressure (Table 3-44). It is concluded that the structural design is adequate to withstand this load condition.

The area is serviced by the fuel handling bridge crane, which is described in Section 3.8.4.1.4. For a discussion of the cask handling in this area, refer to Section 9.1.1.3.2. The New Fuel Storage Vault is constructed to prevent fuel assemblies from being stored in other than the prescribed locations. Drainage of the New Fuel Storage Vault is provided to preclude the buildup of water within the vault.

The New Fuel Storage Vault is not subject to effects of high energy pipe breaks.

3.8.4.2.5 Structural Acceptance Criteria

The New Fuel Storage Vault is designed in such a way that the elastic behavior of the structure is maintained when subjected to the loading combinations employed for design as defined in Section 3.8.4.2.3.

The stress limits used for the design of the New Fuel Storage Vault are as defined in the codes, standards, and specifications summarized in Table 3-31.

3.8.4.2.6 Materials, Quality Control and Special Construction Techniques

The New Fuel Storage Vault utilizes the same materials as the Auxiliary Building as listed in Section 3.8.4.1.6.

The materials and quality control procedures are in accordance with the requirements of the codes, standards and specifications of Table 3-31, and the quality assurance requirements as outlined in Chapter 17 of the FSAR.

Where Cadwelds are used, the requirements of Regulatory Guide 1.10 are met. There are no special construction techniques employed in the construction of the Auxiliary Building.

3.8.4.2.7 Testing and Inservice Surveillance Requirements

The New Fuel Storage Vault complies with the Testing and Inservice Surveillance Requirements as defined in Section 3.8.4.1.7.

3.8.5 Foundations and Concrete Supports

3.8.5.1 Description of Foundations and Supports

3.8.5.1.1 Foundations

1. Reactor Building

   The Reactor Building foundation is a six feet thick reinforced concrete slab, based on sound rock. The foundation is reinforced in both directions of both faces. Refer to Figure 3-111 for the reinforcement arrangement at the junction of the foundation with (i) the Reactor Building vertical shell (ii) the crane wall.

2. Auxiliary Building

   The foundation of the Auxiliary Building consists of a four feet thick reinforced concrete slab, based on sound rock. The foundation slab is reinforced in both directions of both faces of the foundation. Figure 3-112 illustrates a typical reinforcement arrangement between the Auxiliary Building foundation and a typical vertical wall.

3.8.5.1.2 Concrete Supports

Refer to Section 5.5.14 for details of the equipment supports.

3.8.5.2 Applicable Codes, Standards and Specifications

The codes, standards and specifications employed in the design of the foundations of all Category I structures are defined in Table 3-31.

3.8.5.3 Loads and Loading Combinations

3.8.5.3.1 Foundations

1. Reactor Building

   The Reactor Building foundation is designed to properly sustain the following loads which are transmitted to the foundation from:

a. Reactor Building, refer to Section 3.8.1.3 for details of the loads and loading combinations of the Reactor Building.

b. Containment Vessel, Section 3.8.2.3 lists the details of loads and loading combinations of this structure.

c. Interior structure, refer to Section 3.8.3.3 for loads and loading combinations of the interior structure.

d. Operating loads due to equipments or systems directly supported by the foundation.

The foundation design takes into account the effect of the gross overturning moment and torsional moments as well as base shear of the interior structure. The foundations are also designed to accommodate the local effects of the equipment supports anchored to them.

2. Auxiliary Building

The Auxiliary Building foundation is designed to properly sustain the following loads:

a. The loads transmitted to the foundation from the Auxiliary Building itself subject to the appropriate loads and loading combinations outlined in Section 3.8.4.1.3.

b. The operating loads due to equipments and systems directly supported on the foundation.

3.8.5.3.2 Equipment Supports

For equipment supports loads and loading combinations, refer to Section 5.5.14.

3.8.5.4 Design and Analysis Procedure

3.8.5.4.1 Foundations

1. Reactor Building

The Reactor Building foundation is designed as a circular plate based on a sound rock base (for rock characteristics, refer to Former Appendix 2E of the FSAR). The effect of the rock base is considered in the design of the foundation. The loads and loading combinations transmitted to the foundation from the Reactor Building, the interior structure and the Containment Vessel are idealized as axisymmetric or sum of Fourier Harmonics, Kalnin's computer program is employed for the analysis of the foundation (for more details of the program, refer to Section 3.8.2.4). Testing procedure for the computer programs used in the structural analysis are outlined in Section 3.8.3.4.

2. Auxiliary Building

The Auxiliary Building foundation is designed as a reinforced concrete slab on sound rock. The design is according to the codes and specifications subscribed in Section 3.8.5.2. The complete list of the computer programs used in the structural analysis and testing procedure of these programs are given in Section 3.8.3.4.

3.8.5.5 Structural Acceptance Criteria

The Category I structures foundations are designed in accordance with the code requirements of Table 3-31. The loads and loading combinations employed in the foundation analysis and design are outlined in Section 3.8.5.3.

The factors of safety against overturning and sliding for all Category I structures are as shown in Table 3-45. These factors of safety are the absolute minimum and are calculated based upon faulted loading conditions for Category I structure.

3.8.5.6 Materials, Quality Control and Special Construction Techniques

1. Foundations

   The materials used for the design and construction of the foundations of Category I structures are:

   a. Concrete - strength 3000 psi to 5000 psi after 28-day test.

   b. Reinforcing Steel - ASTM - A615, Grades 40 and/or 60.

   c. Standard Steel Shapes - ASTM A-36.

   d. Embedded Plates - ASTM A-36 and/or SA-516.

   e. Stainless Steel Standard Shapes - ASTM A276, Type 304.

   f. Anchor Bolts - ASME SA320-L43.

   The materials and quality control procedures are in accordance with the requirements of the codes, standards and specifications of Table 3-31, and the quality assurance requirements of Chapter 17 of the FSAR.

   There are no special construction techniques employed in the construction of the Category I structures foundations.

3.8.5.7 Testing and Inservice Surveillance Requirements

There are no testing or inservice surveillance requirements for the foundations of Category I structures.

3.8.6 References

1. Wilson, Edward and Ghosh, Sukmar; “Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading,” Report No. EERC 69-10, College of Engineering, University of California, Berkeley, California, September 1969.
2. Deleted
3. Biggs, J. M., “Introduction to Structural Dynamics,” McGraw-Hill Book Company, 1964.
4. NACA TN3786, “Handbook of Structural Stability, Part VI - Strength of Stiffened Curved Plates and Shells,” by Herbert Becker, New York University, July 1958.
5. NACA TN3781, “Handbook of Structural Stability, Part I - Buckling of Flat Plates,” by G. Gerard and H. Becker, New York University, July 1957.
6. NACA TN3782, “Handbook of Structural Stability, Part II, Buckling of Composite Elements,” by G. Gerard and H. Becker, New York University, July 1957.

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3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

Information presented in this section supplements the basic analytical methods and testing program requirements described in Section 3.7.2 and Section 3.7.3. Equipment was designed to insure structural integrity and operability; however, it must be realized that the load combination and stress limits that were used reflect AEC requirements that were in effect when the construction permit for this plant was issued and when the components were purchased and subsequently designed. Furthermore, the codes and procedures which were available when the components were purchased are based on conservative design requirements rather than detailed stress analyses. These codes and procedures have been widely used by the nuclear industry for the design of components which are installed in plants that are presently operating.

Active pumps were designed in accordance with the ASME Code for Pumps and Valves for Nuclear Power Plants. The stress levels in the pumps did not exceed those allowed by the Code. Forces resulting from seismic acceleration in the horizontal and vertical directions were included in the analyses of the pumps and their supports. To eliminate any amplification of the seismic floor accelerations in the pump support structure, the supports were designed to have natural frequencies in excess of 30 cps.

The pumps are subjected to a series of tests prior to installation and after installation in the plant. In-shop tests include hydrostatic tests to 150% of the design pressure, seal leakage tests, net positive suction head (NPSH) tests to qualify the pumps for the minimum available NPSH, and functional performance tests. For the NPSH and functional performance tests, the pumps are placed in a test loop and subjected to operating conditions. After installation of the pumps in the plant, they undergo cold hydrostatic tests, hot functional tests to verify operation, and the required periodic inservice inspection and operation.

Active valves are analyzed to determine the stresses in the extended structures when the valves are subjected to seismic accelerations in the horizontal and vertical directions. For these analyses, primary stress intensities/stresses are limited to 1.5 Sm for Class 1 valves and 1.8 S for Class 2 and Class 3 valves. For non-pressure retaining high strength alloy pull down bolts installed on the Main Steam Isolation Valves (MSIVs) primary stresses are limited to 90% of yield strength. Both limits preclude plastic deformation in all but the extreme fibers of Class 2 and Class 3 valves. Deformation in the valves will thus be small and the valves will operate as required.

Prior to installation, the valves are subjected to shell hydrostatic tests, seat leakage tests, and functional tests to show that the valves will open and close within the specified time limits when subjected to the design differential pressure. After installation, the valves undergo cold hydrostatic tests, hot functional tests to verify operation, and periodic inservice inspection and operation to assure the continued ability of the valves to operate.

The above design procedures and qualification tests are, therefore, adequate to insure the structural integrity and operability of the pumps and valves for this plant.

3.9.1.1 System Vibration Operational Test Program

ASME III requires that piping design minimize vibration and that piping systems be observed under startup or initial operating conditions to insure that steady state vibration in piping systems is not excessive. As part of the preoperational test program described in Chapter 14, steady state piping vibration and transient response of piping due to valve closures, pump starts, and other changing configurations are observed. If excessive movement or vibration is noted, additional restraints or hydraulic suppressors are designed and added to eliminate the problem.

[HISTORICAL INFORMATION – NOT REQUIRED TO BE REVISED]

A preoperational test procedure, prepared by the Steam Production Department, with acceptance criteria determined by Design Engineering, is used for these observations. This procedure contains the following information:

Identification of piping systems to be monitored
Identification of system functional tests during which piping is to be monitored.
Test equipment
Acceptance criteria
Methods of observation and measurement
Data sheets for recording results

Testing action consists of measuring the overall vibration velocity (unfiltered) at several points along each pipe. This velocity reading is then compared with acceptance criteria based on the piping material (carbon or stainless steel). If the unfiltered velocity reading exceeds the acceptance criteria, a spectrum analyzer is used to make a hard copy vibration spectrum printout at the point where the acceptance criteria are exceeded. This location, along with the pertinent thermal and hydraulic conditions of the system at that time, is noted and the results are sent to Design Engineering for evaluation and recommendations. Following the implementation of a solution, the vibration measurement is repeated to insure satisfactory resolution.

Acceptance criteria are based on conservatively estimated stresses which are derived from measured displacements and conservatively assumed mode shapes.

System vibration measurement described above is performed on Duke Class A, B, C, and F piping systems identified in Table 3-46. Duke Class A, B, C, and F systems not in this table have been omitted for one or more of the following reasons:

1. *Vibration measurements are not performed on piping with nominal size 1" or less. The consequences of the failure of small lines does not justify the expense of designing them to meet the vibration requirements.*
2. *Vibration measurements are not performed on piping containing gases, rather than liquids, because the relatively small forces exerted by flowing gases preclude the development of excessive vibration.*
3. *Vibration measurements are not performed on piping systems which have no flow, or have flow less than 1% of the normal operating life span of the station, because of the lack of or relatively short duration of flow induced vibration in these pipes.*

If any acceptance criteria are exceeded, or if a structural failure, such as an improperly grouted hanger, causes the initial vibration measurements to be questionable, the vibration measurement will be repeated during the unit Hot Functional Test or power ascension testing so that proper performance can be verified after corrective action has been taken.

Reactor Internals Preoperational Tests and Reactor Internals Vibration Monitoring are presented in Sections 3.9.1.3.1 and 3.9.1.4.1.

3.9.1.2 Dynamic Response Testing

[HISTORICAL INFORMATION – NOT REQUIRED TO BE REVISED]

Verification of proper dynamic response of piping systems to changing system configurations (pump starts, valve closures, etc.) is performed at the same time as the System Vibration Operational Test Program, described in Section 3.9.1.1.

ASME III requires that piping system design take into account dynamic effects due to rapid changes in temperature. The systems subject to rapid changes in temperature during plant transients are the Main Steam System and Feedwater System; therefore, these systems receive additional visual inspections following the plant transients described in Chapter 14 to insure that the system dynamic response is within design limits. The results of this inspection following the most severe of these transients (Unit Loss of Electrical Load) are documented.

Acceptance criteria for the dynamic response of piping systems are:

1. No permanent deformation or damage in any system, structure, or component important to nuclear safety.
2. All suppressors and restraints respond within their expected ranges.

In general, no seismic testing is performed on piping or equipment except for electrical controls and instrumentation. Seismic qualification of electrical equipment and instrumentation is discussed in Section 3.10. The analytical procedures used in designing Category 1 equipment include the dynamic effects of seismic events, postulated accidents, and operation. During preoperational testing of equipment, the extent of vibration is checked and corrective action taken where vibration is excessive. The operational vibration testing program is described in Section 3.9.1.1.

Section 3.7.1.2 describes the derivation of design response spectra, and critical damping values for equipment and components are given in Section 3.7.1.3. Dynamic analysis is discussed in Section 3.7.2.1.1. A description of the analyses used in the design of safety-related mechanical equipment such as pumps and heat exchangers to withstand seismic loadings is given in Sections 3.7.2 and 3.7.3.

The tubes in the steam generator are subject to a possible flow-induced vibration that does not exist in the primary coolant loop. This vibration could result from flow across the tubes due to vortex shedding. Parallel flow vibration is also analyzed. Analyses are discussed in greater detail in Section 5.5.2.3.5.

3.9.1.3 Dynamic System Analysis Methods for Reactor Internals

The reactor internals are modeled dynamically for: a) load produced by a double-ended pipe rupture of the reactor coolant loop, (the Design Basis Accident, DBA), for both cold and hot leg breaks; b) response due to an Operating Basis Earthquake (OBE); and c) for the most unfavorable combination of DBA and OBE. Seismic analysis of the reactor vessel and its internals are described in Sections 3.7.2 and 3.7.3.

The reactor internals structures have been designed to withstand the stress and be within deflection limits originating from a full double-ended RCS primary loop pipe break even though such pipe breaks are no longer considered for dynamic effects, in accordance with Reference 14.

The upper internals support structure is made of two plates. The upper support plate and the upper core plate are connected by columns bolted to the plates, with the guide tubes pinned to the core plate. This structure compresses the fuel assemblies and the annular hold-down spring during assembly and is subjected to vertical upward forces from these springs.

During operation, normal and transverse flow drag forces are applied to the columns and guidetubes and differential pressure exists across the horizontal plates. The forces on the columns and guide tubes vary with the distance from the outlet nozzles. Because of the complexity of the upper package geometry and loading conditions, the modeling of the reactor internals was performed by using the method of analysis based on the finite element idealization of the structure and matrix displacement for each finite element. This finite-element structural analysis computer program permits static elastic and plastic analysis, steady state and transient heat transfer, dynamic mode shape analysis, linear and nonlinear dynamic analysis, and plastic dynamic analysis. Descriptions of the techniques used to model the various parts of the internals follow.

The top structure, deep beam, and the upper core plate have been modeled with flat shell elements, the support columns with “three dimensional” beam elements and the fuel assemblies and hold-down spring with “three dimensional” spring elements. Because of symmetry, a one-eighth slice of the upper package has been modeled. The core plate is perforated and is modeled as a geometrically equivalent solid plate which has modified elastic constants according to the theory of perforated plates.

Columns of two different lengths are modeled, the long columns connecting the plates and the short columns connecting the beam grid with the upper core plate.

Under the loads used for design, according to the operating condition under study, the previously described computer program provides stresses and deflections at all nodal points.

The study of the lower internals structure which supports the core is another application of the system code to determine the behavior of a complex structure subjected to a given load. This is a sandwich type structure and consists essentially of the perforated core support plate, support columns, and lower perforated core plate. To obtain a realistic representation of the interaction of the components, the lower support structure was also modeled using the finite element structural analysis computer program. The core plate and core support plate, as well as the lower part of the core barrel, are represented by flat triangular shell elements. Reduced plate strength, due to the perforations, is accounted for by using an equivalent elastic modulus and Poison ratio in the calculations. This structure is loaded with various vertical forces, due to normal and abnormal operation, and the deflections and stresses are obtained for each case. Experimental data from core support assembly testing are available (Reference 1) from tests carried out on a one-seventh scale model. The experimental values have been converted according to basic scaling laws and applied to the prototype structure. The comparison of experimental and theoretical vertical deflections shows good agreement. The test values are larger, as expected, since they are obtained in the absence of the core plate and support columns structures, making the core support plate more flexible. Using the same model, this type of system code is also used to compute stresses and deformation due to non-uniform temperature distributions. With temperature at the component surfaces and the gradient generated by the heat generation as input for the system code, the deflected shape of the structure is obtained. Stresses in components such as the perforated upper and lower core plates, core support plate and top support plate are then computed using the stress intensification factor provided by the standard theory of perforated plates.

3.9.1.3.1 Reactor Internals Preoperational Tests

The program used to establish the integrity of reactor internals involves extensive design analysis, model testing and pre and post hot functional inspection. Additionally, full-size reactors have been instrumented: (References 2 and 3) to measure the dynamic behavior during preoperational testing and the results were compared with predicted values. This program was instituted as part of a basic philosophy of instrumenting the internals of the first of a kind of the current nuclear steam supply system designs for power plants. The previous “first-of-a-kind” plants that were instrumented are R. E. Ginna (two loops), H. B. Robinson (three loops), Indian Point Unit II (four loops), Trojan 1 and Sequoyah 1.

The Indian Point II reactor has been established as the prototype for a four loop unit internals verification program. Subsequent four loop units are similar in design. Past experience indicates that units of similar designs behave in a similar manner. For these reasons a comprehensive instrumentation program was conducted on the Indian Point Plant to confirm the behavior of the reactor internals. The main objectives of this tests were to increase confidence in the adequacy of the internals by determining stress or deflection levels at key locations and to obtain data that can be used to develop improved analytical tools for prediction of internals vibration. In addition, the Trojan 1 and Sequoyah 1 instrumentation programs provide information applicable to McGuire.

The McGuire plants are similar to Indian Point 2; the significant differences in the internals are the modifications resulting from the use of 17 x 17 fuel, replacement of the annular thermal shield with neutron shielding pads, and the change to the Upper Head Injection (UHI – removed from service), inverted top hat upper internals configuration. In addition, McGuire units have a higher flow rate than Indian Point. The effects of these differences are discussed below.

17 x 17 Fuel

The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and control rod drive line. The new 17 x 17 guide tubes are stronger and more rigid, hence, they are less susceptible to flow induced vibration problems. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation is expected from the 15 x 15 fuel assembly vibration characteristics.

Neutron Shielding Pads Lower Internals

The primary cause of core barrel excitation is flow turbulence, generated in the downcomer annulus (Reference 11). The vibration levels due to core barrel excitation for Trojan and McGuire, both having neutron shielding pads, are expected to be similar. Since McGuire has greater velocities than Trojan, the McGuire vibration level due to the core barrel excitation is expected to be somewhat greater than that for Trojan (proportional to flow velocity raised to a small power) (Reference 10). However, scale model test results (Reference 10) and results from Trojan (Reference 12) show that core barrel vibration of units with neutron shielding pads is significantly less than of plants with thermal shields. This information and the fact that low core barrel flange stresses with large safety margins were measured at Indian Point Unit 2 (Reference 3) (thermal shield configuration) show that low stresses approximately equal to those of Indian Point Unit 2 are expected on McGuire.

Upper Internals

The components of the upper internals are excited by turbulent forces due to axial and cross flows in the upper plenum (Reference 11) and pump speed related excitations. Sequoyah and McGuire have the same upper internals configuration; therefore, the vibration behavior is not changed. Data on upper internals vibration have been obtained during hot functional testing of Sequoyah 1. A report on analysis of the data has been submitted (Reference 3). A reduction of the data and the post hot functional inspection results provide assurance of the design adequacy. The increased flow rate of McGuire with respect to Sequoyah is reflected in upper internals vibration primarily as a change in fluid velocity. The vibration at the upper internals due to flow turbulence is approximately proportional to the product of density and velocity squared Reference 10. This product is approximately 5% higher in McGuire than Sequoyah 1.

Applying a 5% increase in level to the high factors of safety deducted from Sequoyah 1 data Reference 13 results in adequate margins for McGuire upper internals. The change in fluid density and elastic modulus due to outlet temperature differences results in a very small change in structural natural frequencies.

Further data have been obtained during initial start-up testing of Sequoyah 1. These data indicate lower vibration levels (and consequently higher factors of safety) than those deduced from hot functional data.

Because the McGuire reactor internals design configuration is well characterized as discussed above, it is not considered necessary to conduct instrumented tests of the McGuire plant hardware. The recommendations of Regulatory Guide 1.20 are satisfied by conducting the confirmatory pre- and post hot functional examination for integrity.

This examination will include the 35 points shown in Figure 3-113. These 35 points include the following:

1. All major load bearing elements of the reactor internals relied upon to retain the core structure in place;
2. The lateral, vertical, and torsional restraints provided within the vessel;
3. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals; and
4. Those other locations on the reactor internal components which were examined on the Prototype Indian Point II design.

The interior of the reactor vessel is also examined for evidence of loose parts or foreign materials.

Specifically, the inside of the vessel is inspected before and after the hot functional test, with all the internals removed to verify that no loose parts or foreign material are in evidence.

1. Lower Internals - A particularly close inspection is made on the following items or areas, using a 5X or 10X magnifying glass or penetrant testing where applicable. The locations of these areas are shown in Figure 3-113.
    a. Upper barrel to flange and girth weld.
    b. Upper barrel to lower barrel girth weld.
    c. Upper core plate aligning pin. Examine bearing surfaces for any shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.
    d. Irradiation specimen guide screw locking devices and dowel pins; check for lockweld integrity.
    e. Baffle assembly locking devices. Check for lockweld integrity.
    f. Lower barrel to core support girth weld.
    g. Neutron shielding pad screw locking devices and dowel pin cover plate welds. Examine the interface surfaces for evidence of tightness and for lockweld integrity.
    h. Radial support key welds.
    i. Insert screw locking devices. Examine soundness of lockwelds.
    j. Core support columns and instrumentation guide tubes. Check all the joints for tightness and soundness of the locking devices.

k. Secondary core support assembly screw locking devices for lockweld integrity.

l. Lower radial support keys and inserts. Examine for any shadow marks, burnishing, buffing, or scoring. Check the integrity of the lockwelds. These members supply the radial and torsion constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing, or shadowing marks which would indicate pressure loading and relative motion between the two parts. Some scoring of engaging surfaces is also possible and acceptable.

m. Gaps at baffle joints. Check for gaps between baffle and top former and at baffle to baffle joints.

2. Upper internals - A particularly close inspection is made on the following items or areas, using a magnifying glass of 5X or 10X magnification, where necessary. The locations of these areas are shown in Figure 3-113.

   a. Guide tube, support column, and thermocouple column assembly locking devices.

   b. Upper core plate alignment inserts. Examine for any shadow marks, burnishing, buffing or scoring. Check the locking devices for integrity of lockwelds.

   c. Guide tube closure welds, tube-transition plate welds and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals are subjected to a total operating time at greater than normal full flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition, there is some operating time with only one, two, and three pumps operating.

Therefore, when no signs of abnormal wear are found or of harmful vibration being present in the core support structures, and with no apparent structural changes taking place, the four loop core support structures are considered adequate.

3.9.1.4 Correlations of Reactor Internals Vibrations Tests With the Analytical Results

As stated in Section 3.9.1.3, it is not considered necessary to conduct instrumented tests of the McGuire reactor vessel internals. The original test and analysis of the 4-loop configuration is augmented by References 9 through 13 to cover the effects of successive hardware modifications and confirmed by prototype operating experience. These studies, which utilize analytical models, scale model test results, component tests, and results of previous plant tests, are used to characterize the forcing functions and establish component structural characteristics and to estimate the flow induced vibratory behavior and response levels of McGuire.

3.9.1.4.1 Vibration Monitoring

Since internals of a given type (i.e., two, three, or four loop) are designed and manufactured to essentially the same procedures, processes, and similar drawings, the response of these structures within a pressurized water reactor environment is similar.

Performance data from the instrumentation of actual reactors as well as mechanical and flow scale models, are available. (References 1, 2, 3, 4, 5)

The preoperational flow test on the Indian Point -11 Plant, the four-loop prototype unit, Trojan 1, the neutron panel prototype and Sequoyah 1, the UHI internals prototype have been completed.

The pre and post-preoperational flow test examination of the internals have been completed indicating that all the components performed as predicted. No evidence of damage or incipient failure has been found.

The testing programs consisted of measurements of the stresses, deflections and responses of select key points in the internals structures during hot functional and initial startup tests. The main purpose of these testing programs was to assure that no unexpected large amplitudes of vibration existed in the internals structure during operation. The tests were intended to provide data and results on indicators of overall core support structure performance and to verify particular stress and deflection quantities.

3.9.1.5 Analysis Methods Under LOCA Loadings

The dynamic system analysis methods and procedures used to confirm the structural design adequacy of the Reactor Coolant System and reactor internals are discussed in the following and in Sections 3.6, 3.9.2 and 5.2.

Specifically, mathematical modeling of piping, pipe supports, and reactor internal structures that are used in the analysis are discussed in Sections 3.6, 3.9.2 and 5.2, respectively. The basis for any structural partitioning and directional decoupling of components is also discussed in Section 5.2.

In Section 5.2 a description of the forcing functions that are used for the LOCA dynamic analysis is presented. These forcing functions are derived from applicable combinations of system pressure differential, direction, rise time, magnitude, duration and initial conditions.

A description of the methods and procedures used to compute dynamic responses are discussed in this section.

A summary of the results of the dynamic analysis including applicable loading combinations that govern the designing of the system are presented in Section 5.1.

The acceptability of computer codes used by Westinghouse is documented in Reference 9.

The scope of the different dynamic analysis techniques and methods used to evaluate mechanical systems and components of the Westinghouse Pressurized Water Reactor for loads produced by a double-ended pipe rupture of the main coolant loop (DBA) is very extensive.

Reactor Internals Analysis

Analysis of the reactor internals blowdown loads resulting from a loss of coolant accident is based on the time history response of the internals to simultaneously applied blowdown forcing functions. The forcing functions are defined at points in the system where changes in cross section or direction of flow occur such that differential loads are generated during the blowdown transient. The dynamic analysis can employ the displacement method, lumped parameters, stiffness matrix formulations and assumes that all components behave in a linearly elastic manner.

In addition, because of the complexity of the system and the components, it is necessary to use finite element stress analysis codes to provide more detailed information at various points.

A comprehensive explanation of all the techniques and analytical methods used cannot be included in the scope of the FSAR. The more important and relevant methods are presented as an overview in Section 3.9.1.3 and summarized in the following.

Blowdown Forces Due to Cold and Hot Leg Break

A blowdown digital computer program which is developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in Pressurized Water Reactor coolant systems during a loss of coolant accident is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM (Reference 7) which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. This blowdown code is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time.

Although spatially one dimensional conservation laws are employed, the code can be applied to describe three dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surface (such as in the pressurizer). System losses such as friction contraction, expansion, etc. are considered.

The blowdown code evaluates the pressure and velocity transients for locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are post-processed with the programs LATFORC and FORCE2 which utilize a detailed geometric description in evaluating the loadings on the reactor internals.

Horizontal forces in the x and y directions were calculated with the LATFORC program. The downcomer annulus is sub-divided into cylindrical segments through circumferential and axial zones. The LATFORC program uses coolant property history data generaged by the blowdown code along with geometric component radial and axial length information to calculate the horizontal forces on the vessel wall and core barrel.

FORCE2 calculates forces in the vertical direction for reactor components. Each reactor component for which FORCE2 calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element.
2. Flow stagnation on, and unrecovered orifice losses across the element.
3. Friction losses along the element.

Input to the code, in addition to the blowdown pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the hydraulic diameter and area of the element along which the shear forces act.

The mechanical analysis has been performed using conservative assumptions in order to obtain results with extra margin. Some of the most significant are:

1. The mechanical, thermal and hydraulic analyses have been performed separately without including the effect of the water-solid interaction. Peak pressures obtained from the hydraulic analysis are attenuated by the deformation of the structures.
2. When applying the hydraulic forces, no credit is taken for the stiffening effect of the fluid environment which reduces the deflections and stresses in the structure.
3. The multi-mass model described below is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the vertical direction. This

model is conservative in the sense that further mass-spring resolution of the system would lead to further attenuation of the shock effects obtained with the present model.

The reactor internals are represented by a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs.

Appropriate dynamic differential equations for the multi-mass model describing the aforementioned phenomena are formulated and the results obtained using a digital computer program which computes the response of the multi-mass model when excited by the set of time dependent forcing functions generated by the LATFORC and FORCE2 programs.

The results from the program give the forces, displacements and deflections as functions of time for the reactor internals (lumped masses).

Reactor Coolant Loop (RCL) Analysis - Westinghouse Methodology

A flow diagram representing the procedure for the complex time-history dynamic solution is shown in Figure 3-115. The procedure for dynamic-solution is iterative in nature since the definition of support stiffness matrices for dynamic behavior (to be incorporated in the RCL model) depends upon the response of the support points which is not known a priority.

The initial displacement configuration of the mass points is defined by applying the initial steady-state hydraulic forces to the unbroken RCL model. For this calculation, the support thickness matrices for the static behavior are incorporated into the RCL model. For dynamic solution, the unbroken RCL model is modified to simulate the physical severances of the pipe due to the postulated LOCA under consideration. The static support cases (i.e., steam generator columns and reactor coolant pump columns) are included in the dynamic model as stiffness matrices. Other supports such as tie rods, bumper blocks, and hydraulic snubbers, which go directly to ground, are represented in FIXFM by non-linear elements which correctly define the restraint of the physical element. For supports which cannot be represented by non-linear elements, the stiffness matrix for dynamic behavior is selected on the basis of anticipated displacement response at the support points.

The natural frequencies and normal modes for the modified RCL dynamic model are determined. The time-history hydraulic forces at appropriate node points are combined to determine the forces and moments at structural lumped-mass points of interest. After proper coordinate transformation to the RCL global coordinate system, the hydraulic forcing functions are stored on magnetic tape for later use as input to the FIXFM program.

The initial displacement conditions, natural frequencies, normal modes, and the time-history hydraulic forcing functions form the input to the FIXFM program which calculates the dynamic time-history displacement response for the dynamic degrees-of-freedom in the RCL model. The displacement response at support points is reviewed to validate the use of support stiffness matrices for dynamic behavior. If the calculated support point response does not match with the anticipated response, the dynamic solution is revised using a new set of support stiffness matrices for dynamic behavior. This procedure is repeated until a valid dynamic solution is obtained.

The time-history displacement response from the valid solution is stored on magnetic tape for later use to compute the support loads and to analyze the RCL piping stresses.

The support loads, [F], are computed by multiplying the support stiffness matrix, [k], and the displacement vector, [δ], at the support point. The support loads are stored on magnetic tape for use in the support member evaluation. The time-history displacement response from the

FIXFM program is used as input to the WESDYN-2 program. The program treats this input as an imposed deflection condition on the RCL model and computes the time-history of internal forces, deflections, and stresses at each end of the members of the RCL piping systems. The results of this solution are stored on magnetic tape for later use in piping stress evaluation.

Reactor Coolant Loop Analysis - Steam Generator Replacement

Large reactor coolant loop pipe ruptures (double-ended guillotine breaks) were eliminated for steam generator replacement by the application of leak-before-break-criteria to the reactor coolant loop piping. This was permitted by the NRC as described in Reference 16 in Section 3.9.4.

3.9.1.6 Analytical Methods for ASME Code Class I Components

Elastic system analyses are generally being used to establish loads for the evaluation of ASME Code Class I components by elastic stress analysis methods.

The design of systems is being performed using elastic dynamic analysis procedures and the response of systems is normally maintained within the elastic range by selection of support systems which avoid resonance conditions.

Should inelastic methods be deemed necessary in cases where local inelastic response is significant, the elastic system analysis will be reviewed to see if the analysis requires modification. In these cases, the system analyses will be modified to include the reduced stiffness value of the inelastic component corresponding to the inelastic deformation.

3.9.1.6.1 Analytical Methods for ASME Code Class I Components - Westinghouse

No plastic instability allowable limits given in ASME Section III are used when dynamic analysis is performed. The limit analysis methods have the limits established by ASME Section III for normal, upset and emergency conditions. For these cases, the limits are sufficiently low to assure that the elastic system analysis is not invalidated. For ASME Code Class I components, the stress limits for faulted loading conditions are specified in Section 5.2.

3.9.1.6.2 Analytical Methods for ASME Code Class 1 (Duke Class A) Components - Duke Scope

Categorized design loading combinations and stress limits for Duke Class A piping are in accordance with the ASME Boiler and Pressure Vessel Code Section NB. See Table 3-4. Table 3-49 gives the stress criteria for Duke Class A Supports, Restraints, and Anchors.

3.9.2 ASME Code Class 2 and 3 Components

Evaluation for License Renewal:

McGuire has a number of systems that were designed to ASME Code Class 2 and 3. Piping analyses for these systems include stress range reduction factors to provide conservatism in the design to account for thermal cyclic operations. Thermal fatigue of mechanical systems designed to ASME Code Class 2 and 3 is considered to be a time-limited aging analysis because all six of the criteria contained in 10 CFR 54.3 are satisfied. From the license renewal review, it was determined that the analyses of thermal fatigue of these mechanical systems are valid for the period of extended operation.

3.9.2.1 System and Component Functional Design Basis

Design pressure, temperature, and other functional design conditions for mechanical systems and equipment are described on a system basis in Chapters 6.0 and Chapter 9 and Section 3.2.2.

3.9.2.2 Design Loading Combinations

Categorized design loading combinations and associated stress limits for ASME Code Class 2 and 3 components are given in Table 3-47, Table 3-48, Table 3-50, Table 3-51, and Table 3-52 and are further discussed in Section 3.7.2.1.

3.9.2.3 Design Stress Limits

Conventional methods of analyses are used for Category 1 components analyzed by Duke. STRUDL is used to perform all finite element analyses when such analyses are warranted. However, in general, the design approach is taken such that individual elements of a design configuration are each required to resist a specific load or set of loads. The sum of the elements is then capable of resisting the sum of the loads for each design condition. The ability of each of the elements to carry its specific load independent of every other element eliminates the need for costly detailed interactive stress analyses. It is recognized that unintentional interaction could be detrimental, especially for cyclic loading, and must therefore be evaluated.

Most piping analyses on Duke Class B, C and F piping are performed using computer programs PISOL-3A and SUPERPIPE. Dynamic analyses on Duke Class B, C and F piping are performed on PISOL-1A and SUPERPIPE. These programs were originally from Impell Corporation (formerly EDS nuclear) and are well recognized and utilized throughout the industry. Each program is described below for further clarification.

1. PISOL-1A--for the dynamic elastic analysis of piping systems subject to seismic excitation.

   EDS Program PISOL-1A analyzes arbitrary, three-dimensional piping systems for seismic excitation using the dynamic analysis technique known as the response spectrum mode super-position method. In this technique, the earthquake excitation is characterized by acceleration response spectra, and the total response of the system is evaluated as a square root of the sum of the squares combination of the response of the significant natural modes of vibration of the system. The results for earthquakes acting in both horizontal directions separately, each combined with vertical motion are computed, or alternatively, earthquakes acting in all three directions simultaneously may be computed.

   A piping system is idealized as a mathematical model consisting of lumped masses connected by massless elastic members. The location of the lumped masses is chosen to adequately represent the dynamic characteristics of the system. The direct stiffness method of structural analysis is used to form the system stiffness matrix, including stiffness modifications for curved components, and diagonal mass and damping matrices are assumed. The dynamic properties of the system (periods of vibration and normal mode shapes) are determined using the Householder-QR method, and the system response is then computed by the modal superposition procedure.

   PISOL-1A has been used for dynamic seismic piping analysis for more than 15 nuclear power plants, and has been verified independent analysis by the Bechtel Power Corporation of San Francisco for several of these plants. In addition, the program has been bench marked by EDS against the ASME Sample Problem #1 contained in ASME publication, “Pressure Vessel and Piping 1972, Computer Programs Verification”, and the bench mark data has been submitted to the ASME. Typical plants for which the program has been used for dynamic seismic piping analysis include Donald C. Cook, Rancho Seco, Trojan and Calvert Cliffs Unit 1.

2. PISOL-3A--for the static elastic analysis of piping systems subject to static loading.

   PISOL-3A analyzes arbitrary, three-dimensional piping systems subject to applied static loadings and displacements. The program is based on the direct stiffness method of structural analysis.

   A piping system is idealized as a mathematical model consisting of lumped weights connected by weightless elastic members. The location of the lumped weights is chosen to adequately represent the weight distribution of the system for dead load analysis. The direct stiffness method of structural analysis is used to form the stiffness matrix including stiffness modifications for curved components. The equations of equilibrium are solved to determine the system displacements, and hence member forces and moments for the applied loading and/or displacements, using a Gaussian reduction procedure.

   PISOL-3A has been used for static piping analysis for more than 20 nuclear power plants. The program has been used for independent verification of the programs of the Bechtel Power Corporation of San Francisco for several plants, and was included on the Monticello docket. In addition, the program has been bench marked by EDS against other programs such as EDSGAP and MEL-40. Typical plants for which the program has been used for static piping analysis include Monticello, Donald C. Cook, Rancho Seco, Trojan and Calvert Cliffs Unit 1.

   [HISTORICAL INFORMATION – NOT REQUIRED TO BE REVISED]

3. *TRANS-1A--for the determination of temperature and flow transient effects in piping components.*

   TRANS-1A analyzes the one-dimensional time variation of temperature in piping components subject to temperature and flow transient conditions to determine the T_{avg}, Delta T1, and T_a-T_b parameters for use in piping stress analysis. The program is based on the solution for one-dimensional heat flow presented by John E. Brock in "A Temperature Chart and Formulas Useful With the USAS1 B31.7 Code for Thermal Stress in Nuclear Power Piping", extended to cover arbitrary temperature and flow transient time-histories. The program allows the user to select the solution grid and in addition to determine the temperature variation of the material and fluid properties.

   TRANS-1A has been used for temperature transient analysis of piping components for five nuclear power plants. The program has been verified by EDS Nuclear both against the results presented by Brock in the above reference and against the results of temperature transient analyses obtained using EDS program TAPAS (a two-dimensional finite element heat transfer program). Typical plants for which the program has been used for temperature transient analyses include Rancho Seco, Calvert Cliffs Unit 1, and Indian Point Unit 1 ECCS Modifications.

4. PISOL-7A--for the combination of forces and moments, and effects induced in the components of a piping system by the various loadings and the evaluation of the compliance of stress and fatigue conditions in accordance with Section NB3600 of ASME III.

   PISOL-7A is a stress and fatigue evaluation program which contains the various stress and fatigue equations of Section NB3600 of ASME III. The program uses as input the piping system component force and moment results obtained from static and dynamic piping analyses using EDS Program PISOL-3A and PISOL-1A, in addition to the temperature transient parameters from EDS Program TRANS-1A.

The Program evaluates at each data point the compliance with the relevant sections of ASME III for cyclic load histogram specified by the user. Primary, secondary, peak and elastic-plastic stresses and fatigue usage compliance are evaluated as requested.

The program has been used for the stress evaluation of five nuclear power plants including Rancho Seco, Calvert Cliffs Unit 1, and Indian Point Unit 1 ECCS Modifications. The program results are amenable to hand calculation verification and extensive hand calculation verification has been performed by EDS Nuclear.

[HISTORICAL INFORMATION – NOT REQUIRED TO BE REVISED]

5. *PWHIP--for the Dynamic, Nonlinear, Inelastic Analysis of Three Dimensional Piping Systems subjected to arbitrary time varying forces.*

PWHIP

EDS Program PWHIP is a general-purpose program for the dynamic, nonlinear, inelastic analysis of three-dimensional piping systems subjected to arbitrary time-varying forces. The program may be used to determine time histories of pipe displacements, member forces and strains, as well as restraint force and deformation time histories. The effects of inelastic pipe response, inelastic restraint behavior, and rupture restraint clearances are included in the analysis procedure.

PWHIP has been verified for a comprehensive set of example problems in accordance with the EDS Nuclear Quality Assurance Procedures. Three example problems solved using the PWHIP program are described in Figure 3-116 through Figure 3-120. PWHIP results are described and compared against theoretical solutions in Table 3-54 through Table 3-56. A detailed description of the program, approximations used in the inelastic behavior algorithm and usage restrictions are presented below.

The system to be analyzed is idealized as an assemblage of three-dimensional "beam" type finite elements. The use of beam elements allows the analysis of both process piping and structural frame works which may be required for transmitting impact loads from energy-absorbing devices to building structures. The beam element locations and connectivities between nodal points of the system. Each node may have up to six degrees of freedom (three translation and three rotation) and nodal fixities may be specified for any of the nodal degrees of freedom at two or more different nodes may be specified to be identical in order to similar constraints on motion between different portions of the piping system or adjacent piping systems.

Initial stiffness matrices are calculated for each element of the system using virtual work principles and the elastic portion of the piping and restraint material properties. They are assembled to form a global stiffness matrix by the direct stiffness method.

The mass matrix for the system is assembled from mass point data supplied by the user, including both translational and rotational inertia terms. This enables the user to specify concentrated masses for non-structural system components, such as valves, and also permits a higher degree of refinement, if desires, in the distribution of system stiffness properties than in the distribution of mass properties. Because the required time step for analysis is controlled by the minimum mass point spacing and instantaneous material moduli, this feature of the program allows greater use control of the time step required for problem solution. This avoids excessive solution time associated with very small time steps that might otherwise be required for numerical stability.

The damping matrix for the system is specified in the form of Rayleigh damping using the mass and the initial and/or instantaneous stiffness matrices of the system. Parametric studies have been performed to investigate the influence of the assumed Rayleigh damping data on the results, and have indicated that the calculated pipe whip responses are insensitive to the assumed damping values. This results from the fact that the primary system response takes place over a very short period of time and virtually no energy is dissipated by material damping.

The nonlinear stiffness properties are specified in the form of multilinear moment-curvature and torque-twist relationships for the piping and structural members, and force-deformation relationships for one-dimensional pipe whip restraint elements. It should be noted that, although several types of one-dimensional restraint elements are available in the program, a general three-dimensional nonlinear restraint can also be simulated through the use of structural members. Local member axis orientation may be arbitrarily specified, thus permitting consideration of strong and weak axis orientation for general structural shapes. Local member axis orientation may be arbitrarily specified, thus permitting consideration of strong and weak axis orientation for general structural shapes. Any number of “strength” types may be used for defining nonlinear properties, thus permitting consideration of different pipe sizes and schedules, structural shapes, reinforced components such as tees and reducers, and increased pipe flexibilities at elbows.

Dynamic response of the system is obtained by direct integration of the equilibrium equations. Arbitrary force time histories may be applied to any of the nodal points. At each time step, each element of the system is checked for yielding (or unloading, if yield has occurred), and total displacements of the pipe at each of the restraint locations are monitored for impact or rebound. In addition, member forces and deformations are checked at the end of each time step and compared to the member properties used in the calculation. Equilibrium corrects are made at the end of each time step if it is found that the stiffness properties of the member have changed. Large deformations of the pipe whip restraints are permitted and geometry corrections are included in the instantaneous restraint stiffness properties. In the current program version, geometry corrections are not made for the piping system, and hence piping response is restricted to “small” displacements. However, this is not considered to be significant restriction on usage as accuracy of system response can be maintained for displacements of up to one-tenth of pipe span lengths, which is sufficient for virtually all practical problems.

Yielding of one-dimensional restraint elements is determined directly from the force-deformation characteristics defined for the element. In the current program version, yielding of pipe or structural elements is detected at any time step when the following relationship is satisfied:

$$\left(\frac{M1}{M_{1p}}\right)^2 + \left(\frac{M2}{M_{2p}}\right)^2 + \left(\frac{T}{T}\right)^2 \geq 1$$

where M1, M2, and T are the calculated bending and torsional moments at the end of a member, and M_{1p}, M_{2p} and T_p are the corresponding bending and torsional moments at initial yield. This elliptical interaction relationship permits detection of yielding for relatively complex system response. If the moment increases at a subsequent time step lead to a reduction in the value of the yield function, unloading occurs along a line parallel the elastic portion of the moment-curvature characteristics for the member. The above yield criterion produces accurate results provided no reverse curvature occurs in the member at a particular time step. This constraint is readily avoided by specifying member lengths which are sufficiently small to preclude the development of high moment gradients between the ends of the member.

6. SUPERPIPE

   Superpipe is a comprehensive computer program for the structural analysis and design checking of piping systems, with particular emphasize on nuclear piping. It can be used for both static and dynamic elastic analysis.

The safety-related mechanical components are classified as Safety Class 1, 2 or 3, as shown in Table 3-4 and are designed in accordance with the codes listed. The loading conditions shown in Table 3-4 are considered in the overall station design.

The safety-related piping systems are classified as shown in Table 3-5 and system flow diagrams located in Chapter 6 and Chapter 9.

The stress criteria for Class 2 and Class 3 items are shown in Table 3-47, Table 3-48, Table 3-50 and Table 3-51.

3.9.2.3.1 Westinghouse Design Stress Limits

The membrane plus bending stress limits for vessels, and the membrane plus bending faulted stress limits for pumps permit inelastic deformation to occur in the outer fibers of the section under consideration. However, gross inelastic deformation cannot occur since the limits listed in Table 3-52 for membrane stress insure that stresses are below yield everywhere except in the outer fibers.

3.9.2.4 Analytical and Empirical Methods for Design of Pumps and Valves

Class 2 and 3 pumps and valves are designed in accordance with the standards and requirements for the applicable safety class and the ASME Boiler and Pressure Vessel Code, Section III, Sections NC or ND, respectively. Performance of components, required to perform during the transients or events considered in the respective operating condition categories, is assured by selecting the stress limits, following the code intent, sufficiently low so as to preclude gross deformation, as described in Sections 3.9.2.2 and 3.9.2.3.

1. Active Class A valves are designed to function properly during normal upset and faulted conditions described in Table 5-4 and listed in Table 5-5. There are no active Class A Pumps.

   Class B & C active pumps and valves are designed to properly operate during normal, upset and faulted conditions as described above.

The NRC issued Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance”, on June 28, 1989 which extended the scope of Bulletin 85-03, “Motor Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings”, dated November 15, 1985 to all safety related MOVs as well as all position-changeable MOVs as defined in the generic letter. The subsequent seven supplements provided additional NRC guidance and more clearly defined the scope to be a subset of safety-related motor-operated valves. The NRC staff closed its review of the GL89-10 program for McGuire in Reference 31. Switch settings and valve operational capabilities are verified on an on-going basis through a periodic verification program in accordance with Generic Letter 96-05, “Periodic Verification of Design Basis Capability of Safety Related Motor-Operated Valves”, as described in Reference 35. The NRC accepted McGuire’s Generic Letter 96-05 program in Reference 36.

The NRC issued Generic Letter 95-07, “Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves”, on August 17, 1995 to request licensees take action to identify safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding and ensure that they are capable of performing their safety functions.

Evaluations of the valves within this category were completed with responses to the NRC submitted in References 25, 32, and 33. The NRC closed this issue in Reference 34.

2. The seismic design of active components is described in Section 3.9.1. Active pumps in Duke's scope of supply consider the Ground Response Spectra, (Figure 2E-2B of former Appendix 2E of FSAR). Additional information is contained in Sections 3.7.2.1.1.8 and 3.7.2.1.1.9 for valves and pumps respectively.

3. Active pump motors and vital appurtenances are designed and analyzed to function properly in connection with their respective pumps for the same conditions which the pumps are designed as stated above.

   Active valve, electric motor operators are tested as specified in Section 3.11.2.1.3, though some of the valves are located outside the reactor containment. All valve appurtenances vital to the operation of the valves are designed and analyzed to function properly with their respective valves and for the same conditions.

4. The specification, design, testing and inspection of active pumps and valves is adequately described in Section 3.9.1.

Westinghouse equipment as designed complies with the intent of R. G. 1.48 (withdrawn), i.e., it was designed to insure structural integrity and operability. However, it must be realized that the load combinations and stress limits that were used reflect NRC requirements that were in effect when the construction permit for this plant was issued and when the components were purchased and subsequently designed. Furthermore, the codes and procedures which were available when the components were purchased are based on conservative design requirements rather than detailed stress analyses. These codes and procedures have been widely used by the nuclear industry for the design of components which are installed in plants that are presently operating.

Class 2 and 3 valves for McGuire are purchased to the design requirements of the ASME Code, Section III (with addenda through Summer, 1972). Section III with Summer 1972 addenda does not address design limits as such for Class 2 and 3 valves. Instead, Paragraphs NC-3500 and ND-3500 invoke the requirements of ANSI B16.5. B16.5 design rules are based on the selection of a standard pressure-temperature rating value using appropriate design temperature and pressure values. Each pressure-temperature standard value covers a number of acceptable pressure-temperature standard value covers a number of acceptable pressure-temperature combinations. Selection of the particular design pressure and temperature is the basis for determining the amount of conservatism (above operating or accident pressures) for a B16.5 valve.

Regarding stress limits, Class 2 and 3 valves are purchased with design specifications which require analysis for normal operating, upset and faulted conditions with applicable limits as follows:

1. <u>NORMAL OPERATING</u> - This loading includes design flow, dead weight loads and other mechanical loads.

2. <u>UPSET CONDITION</u> - This loading includes applied sinusoidal accelerations of 2g through a frequency range of 1 to 30 H_z in both horizontal and vertical directions. For this loading the stresses shall not exceed allowable working stresses given in Section III.

3. <u>FAULTED CONDITION</u> - This loading includes applied sinusoidal accelerations of 4g through a frequency range of 1 to 30 H_z. For this loading the primary stresses would not exceed a valve of $.9\sigma\gamma$ (yield strength of material as given in Section III at the proper temperature).

When the above limits cannot be on specific valves, the capabilities of these valves are entered into the affected system analyses to assure compatibility with the accelerations produced by postulated seismic events.

Duke considers the above design, stress and pressure rating limits, as well as the basic design philosophy, acceptable for McGuire Nuclear Station.

Deleted paragraph(s) per 2002 revision.

3.9.2.4.1 Operability Assurance of Duke Safety-Related Active Pumps

1. Safety-Related Active Pumps

The following criteria assure that the safety related active pumps will function as designed:

a. Safety-related active pumps are subjected to stringent tests both prior to and after installation in the plant. The in-shop tests include (a) hydrostatic tests of pressure-retaining parts to 150% of the design pressure, (b) seal leakage tests, and (c) performance tests conducted while the pump is operating with flow to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirement, and other pump/motor properties. Bearing temperatures and vibration levels are also monitored during these operating tests to insure values are within limits specified by the manufacturer. After installation, the pump undergoes start-up tests and required inservice inspections and operation, Section 3.9.1.

b. During and after faulted conditions, the safety related active pumps are qualified for operability by the test results and or analyses described in Section 3.9.1 and Table 3-4, and the following:

   1) The pump manufacturer will be required to meet the seismic conditions of the original equipment purchase specification. UFSAR Sections 3.7.1 and 3.7.2.1.1.6 describe the seismic response. Specific procedures for mathematical analysis are outlined in Section 3.7.2. Seismic qualification by testing may be performed instead, Section 3.7.2.1.1.9.

   2) The pump motor and all electrical appurtenances vital to operation of the pump are seismically qualified as described in Section 3.9.2.4.

   3) The functional ability of active pumps after a faulted condition is assured by specifying that faulted condition pump nozzle loads shall be considered as the design pump nozzle (end connection) loads. The pump manufacturer must demonstrate by test or analysis that the pump will operate normally under faulted condition loads, Sections 3.9.2.4 and 3.7.2.1.1.9.

The NRC issued IE Bulletin 88-04, "Potential Safety-Related Pump Loss," on May 5, 1988. The purpose of this bulletin was to request licensee investigation and correction, as applicable, of two miniflow design concerns for plant safety-related pumps. The first concern involved the potential for dead-heading of one or more pumps in safety-related systems that have a miniflow line common to two or more pumps or other piping configurations that do not preclude pump-to-pump interaction during miniflow operations. The second concern was whether or not the installed miniflow capacity is adequate for even a single safety-related pump in operation. Plant evaluations of the McGuire Nuclear Station safety-related pumps and system configurations in conjunction with manufacturer data evaluations formed the basis of Duke Energy's preliminary responses to this bulletin. Final evaluations and operability justifications per the requirements of this bulletin were presented in response to the NRC by letter on January 15, 1990 (letter from H.B. Tucker to the NRC, dated January 15, 1990). Further programmatic enhancements and long-term corrective actions committed to in this response were verified complete/closed out in the letter from M.S. Tuckman to the NRC, dated January 10, 1991.

3.9.2.5 Design and Installation Criteria, Pressure-Relieving Devices

3.9.2.5.1 Mainsteam and Reactor Coolant System Pressurizer Safety and Relief Valves

Safety-related safety and relief valve systems consist of the open discharge Main Steam and the closed discharge Reactor Coolant System Pressurizer safety and relief valve systems. For both the Main Steam and the Reactor Coolant System, pressure relief is accomplished through the use of relief valves and sufficient safety valves to meet ASME Code requirements. All three of the pressurizer safety valves are set to relieve pressure at 2485 psig. The main steam safety valves are set for progressive relief at increasing pressures within the ASME Code allowed range of pressure to avoid more than one valve actuating simultaneously. However, it is recognized that the shock to the system produced by the sudden opening of a valve could possibly induce other valves to open. Therefore, for this dynamic condition, it is assumed that all valves on a single header may initiate discharge simultaneously. This combination of thrust loading would produce the most severe thrust loading condition for the relief system and its support system.

For the Reactor Coolant pressure relief system, the piping stress analyses of the safety and relief valve discharge cases are based on force/time histories generated by rigorous thermal hydraulic analyses. For the Main Steam valve discharge case, a dynamic load factor of 1.22 is used for the analysis of piping and structural member affected by the load.

For the Main Steam System the safety valve discharge piping is designed in a configuration such that the steady state discharge thrust minimizes bending in the valve inlet nozzle. For the pressurizer valves, the discharging piping system is closed and thus the steady state discharge thrust is automatically balanced within the system and therefore produces no steady-state bending moment.

Pressurizer safety valves are assumed to discharge simultaneously for maximum design loadings. For stress analysis purposes, the maximum stresses produced are conservatively combined using peak stresses without regard to their occurring at different times and locations along the piping system.

The dynamic condition existing during the opening and immediately after the opening of a valve is evaluated considering the factors listed below:

1. Installed configuration
2. Valve opening time characteristics
3. Valve thrust
4. Valve station supports and restraints
5. Supporting structure
6. Sequencing of valve operation
7. Dynamics of compressible fluid flow in discharge piping for closed systems
8. Effect of water seal where applicable

The above loading conditions are considered to act in combination with normal operating conditions and the OBE to produce an upset plant condition. Also, the above loading conditions are considered to act in combination with normal operating conditions and the SSE to produce a faulted plant condition.

The stress analyses procedures that are used for qualification of the Main Steam relief valve header nozzles are as follows:

1. The nozzle opening in the header is reinforced in accordance with standard code practice for internal pressure.
2. Stresses due to the loading as determined by the dynamic analysis are calculated in accordance with Welding Research Council Bulletin 107.

   The relief valve header with a mean radius of 16.808 inches and the extruded nozzle with a mean radius of 3.75 inches results in a ratio of the nozzle mean radius to header mean radius of 0.223. This is well within the limits for using the Bijlaard analysis techniques of Welding Research Council Bulletin Number 107.
3. The primary stresses as determined in (2) are combined with the appropriate stresses resulting from the remainder of the primary loads for the plant loading conditions.
4. Since it is not intended that a finite element analysis be performed, it is assumed that the primary pressure stresses around the nozzle opening in the header can be determined by smearing the load caused by the pressure over the metal area provided to resist the load.
5. The allowable stresses are in accordance with Section 3.9.2.2.

The steady state thrust for the Main Steam relief valve discharge is calculated based on Regulatory Guide 1.67 (withdrawn) which states that ASME Code Case 1569 provides guidance for design and analysis which may be used.

From the pressure source (main steam line or pressurizer), the flow to the valve orifice is assumed to be isentropic.

From the valve orifice to the discharge nozzle outlet, the process is assumed to be at constant enthalpy.

The thrust at the nozzle outlet is calculated using the equation:

$$F=(w/g)\ Ve + (Pe)A$$

It is anticipated that corrections for friction drop will not be accounted for; however, in the event that it becomes necessary to do so, Fanno lines will be used to predict flow relationships.

<u>Nomenclature</u>

g	Gravitational constant
F	Reaction force
W	Mass flow rate
P	Static gage pressure
e	Subscript denoting end of discharge nozzle
A	Exit flow area of discharge nozzle (at point e)

V Flow velocity

The design and installation criteria for the Main Steam System Relief Valves includes compliance with Regulatory Guide 1.67 (withdrawn) with the following exceptions:

1. Duke Energy takes exception to ASME Code Case 1569 which states: “It is the opinion of the Committee that the following may be used for design and analysis of piping for a pressure relief/safety valve station meeting the requirements of Section III Construction”. Figure 1 of Code Case 1569 depicts a typical safety valve station configuration. In as much as Duke does not use this configuration in the design of the McGuire relief/safety valve design, Duke must take exception to the applicability of code Case 1569 to the McGuire Nuclear Station.

   In particular, Duke takes exception to the requirement that the M_b term is defined to be no less than the product of force times nominal discharge pipe size times dynamic load factors. Duke's design has placed emphasis on obtaining the minimum moment arm for the discharge thrust force, and we have designed to a moment, M_b, equal to the product of the thrust force times the nominal inlet pipe size times dynamic load factor. A dynamic load factor of 1.22 has been used in the design.

2. Duke Energy also takes exception to paragraph B.3 which states: “a reasonable position to assure adequate strength is to require consideration of the most severe potential sequence of discharge”. Duke does not feel that this is a reasonable position since the permutations and combinations of analyses necessary to define the condition resulting in the maximum effect at any point would be astronomical. Duke's position is given in the first paragraph of this Section.

3.9.2.5.2 Class B and C Systems Overpressure Protection

Safety Valves, Relief Valves, or Safety Relief Valves are used as necessary to provide protection against credible overpressure events, consistent with Articles NC-7000 and ND-7000 of the ASME Section III edition specified in Table 3-5. Testing of relief devices is conducted per ASME/ANSI OM-1 (1987) including OMc (1994). An identified non-compliance with ASME Section III relates to the necessary placement of block valves in the inlet or outlet lines of various systems' relief valves. Such placement, where necessary, was originally found acceptable in ASME Code Interpretation III-1-80-67 per the "controls and interlocks" provision as found in NC-7142. These devices are no longer in compliance with ASME Section III, as now understood per Code Interpretations III-1-89-25 and III-1-80-67R (revision to the original interpretation rendered in 1980).

A comprehensive review of all ASME Section III relief valve applications was conducted in response to NRC request for additional information during their review of the Duke Relief Request submitted for NRC approval of the subject non-compliance with the ASME Section III Code block valve provisions. Such Code Relief had been requested to reflect the As-Built configuration of certain ASME Section III Code Overpressure protection devices. This Request for Relief was returned by NRC transmittal dated August 22, 1997 back to Duke with the explanation that the ASME Code Section III requirements as stated in 10CFR 50.55a(d) and (e) apply to power plants whose applications for construction permits were docketed after May 14, 1984. Therefore, the regulations in 10CFR 50.55a(d) and (e) concerning ASME Section III design requirements do not apply to McGuire. The ASME Code systems' design at McGuire is described in UFSAR Table 3-5. The subject non-compliance with ASME Code block valve provisions has been evaluated and found to be acceptable in each case, based on a combination of administrative controls (controlled documents, directive and procedures) and physical barriers such as locks where necessary.

3.9.2.6 Stress Levels for Category 1 Components

Stress level comparisons to established design loading combinations at those locations where intermediate breaks are postulated are provided as part of the analysis of pipe break effects.

3.9.2.7 Field Run Piping Systems

Duke's practice is to detail the routing of all safety-related process lines, non-nuclear safety and conventional system process lines regardless of size, except as follows:

1. Process piping - All main run process piping in Duke System Classifications A, B, C, D, E, F and G is detailed on engineering drawings; however, items such as vents, drains, valve bypass warming lines, and steam leakoffs for all systems are “field run”. In general, all lines 1" and smaller are field routed.
2. Instrument impulse and air lines - end points and specific routing requirements of safety-related instrument impulse and air lines are established and defined by Engineering. The actual path is established in the field during installation.

It is not practical to limit “field run” piping to an extent greater than this for the following reasons:

1. Obstruction to desirable routing would be difficult to determine; and documentation of a precisely designed path would be lengthy, difficult to prepare, and difficult to follow.
2. Revision to major process piping would cause changes in routing of small lines, resulting in drawing changes without significant improvement in the final result.
3. Sloping of impulse lines would be difficult to accomplish and document.

Thus, field routing of small lines results in a superior job since obstructions and other design revisions are clearly visible and easier to consider while meeting design requirements as established by Engineering.

The special rigorous quality assurance measures and performance tests that are conducted to assure satisfactory installation of field run piping and instrument impulse lines are as follows:

1. All field-engineered lines are schematically shown either on a system flow diagram, an instrumentation detail or a piping drawing such that mistakes in valving, connection termination points and materials are virtually eliminated.
2. All field run piping requiring seismic design is reviewed after erection by appropriate Engineering personnel and applicable seismic controls are detailed by installation specifications.
3. Field run piping is hydrostatically tested as required by ANSI B31.1.
4. An engineering surveillance program is conducted after erection to review all safety-related piping as well as non-safety-related piping in the area to assure that appropriate criteria have been followed. Instrument impulse line installation is inspected by an independent group on site and stamped approval and signoff are required before the system can be turned over to operating personnel.
5. Anytime piping is routed in a Radiation Control Area, all ALARA considerations are properly addressed.

If for some reason the piping may pose a radiological problem/hazard, after being installed or during implementation, a problem report can be generated and transmitted to the appropriate personnel for review and resolution. Typical resolutions may include but are not limited to:

a. Reclassify and restrict the area in question appropriately.

b. Reroute the field run piping as required to comply with original radiation level limits.

c. Shield the radioactive piping as required to comply with original radiation level limits.

6. Instrumentation testing programs are well defined by Duke test procedures. These tests document conclusively that the instrument loops are correctly installed and operate properly.

7. Engineering walkdowns include inspection of instrumentation installations with corrective measures taken as necessary.

8. Both “field run” piping and impulse line installation work are monitored during installation, functional verification and normal daily activities such as operator rounds, radiation protection surveys and field walkdowns.

This practice of “controlled field routing” of small piping and instrument impulse lines produces the best possible overall results. It is not practical to limit “field run” piping to a greater extent.

3.9.2.8 Mechanical Penetrations

Mechanical penetrations are treated as fabricated piping assemblies meeting the requirements of ASME III Section NC and which are assigned the same classification as the piping system that includes the assembly, i.e., Class A through H as defined in Table 3-5 except that any Class C through H lines are upgraded to Class B between Containment isolation valves.

The process line and flued heads making up the pressure boundary are consistent with the system piping materials; fabrication, inspection and analysis requirements are as required by ASME III, Section NC.

Critical high temperature lines and selected engineered safety system and auxiliary lines (regardless of temperature) require the “Hot Penetration” assembly as shown in Figure 3-68 which features the exterior guard pipe for the purpose of returning any fluid leakage to the Containment and for protection of other penetrations in the building annular space. Other lines are treated as cold penetrations since a leak into the annular space would not cause a personnel hazard or damage other penetrations in the immediate area.

Penetration assemblies and their anchorages are analyzed in accordance with Table 3-5 and applicable response spectra curves as developed from the method described in Section 3.7.2 and envelopes for conservatism except that any Class C through H lines are upgraded to Class B between Containment isolation valves as discussed above. Loading combinations and stress criteria for penetrations are shown in Table 3-48. The design of guard pipes consider the simultaneous effects of pressure and jet loadings resulting from a postulated pipe break within the guard pipe and the SSE loadings. An independent analysis is performed on penetrations with guard pipes to enhance the reliability of these critical penetrations. The detailed design analysis of penetrations with guard pipes is performed by an independent engineering group. The group is responsible to a Principal Engineer. The design analysis is checked by qualified engineers within the group. The Supervising Engineer having overall responsibility for the design gives surveillance by review of concept, conformity with codes and criteria, and execution. Any discrepancies are reviewed by the engineering management. Assumptions, calculations, design techniques, and methods are examined for correctness and further work performed as necessary to resolve any disagreements.

The NRC issued Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Conditions," on September 30, 1996, requesting that licensees determine if containment air cooler cooling water systems are susceptible to either water-hammer or two-phase flow conditions during postulated accident conditions and to determine if piping systems that penetrate containment are susceptible to thermal expansion of fluid so that over-pressurization of piping could occur. The Generic Letter 96-06 evaluation was documented by References 38 and 39. The evaluations determined that: i) McGuire containment air cooler cooling water systems were not susceptible to either water-hammer or two-phase flow conditions during postulated accident conditions, ii) that the piping systems that penetrate the containment are not susceptible to over-pressurization due to fluid thermal expansion. The NRC documented that the McGuire generic letter response was acceptable per Reference 37.

3.9.2.8.1 General Design Information for All Mechanical Penetrations

The following definitions are utilized to distinguish the categories of mechanical penetrations.

1. Primary System - Reactor Coolant System and any line connecting to same which penetrates the Containment.
2. Secondary System - All other piping penetrations and systems within the Reactor Building; this includes also the Nuclear Auxiliary Systems.

Design requirements as follow are applicable to piping between the Containment boundary (steel Containment shell or concrete wall, whichever is applicable for anchorage) and the crane wall only.

1. All penetrations are designed to maintain Containment integrity for any loss-of-coolant accident combination of Containment pressures and temperatures.
2. All penetrations 4" NPS and smaller are designed to withstand line rupture forces and moments (via Reactor Building anchors and blowdown restraint as shown in Figure 3-68) generated by their own rupture as based on their respective operating pressures and temperatures except that for process pipe sizes greater than 4 inches NPS, additional moment limiting pipe break restraints are utilized to meet this criteria based on detailed analytical loadings.
3. All primary penetrations and all secondary penetrations that would be damaged by a primary break are designed to maintain Containment integrity.
4. All secondary lines whose break could damage a primary line and also break Containment are designed to maintain Containment integrity.
5. Quality Assurance measures for penetration design calculations, criteria, documentation and procedures are in accordance with the design control requirements of Chapter 17. The design of penetrations is complex in nature but no more so than other safety-related design items; therefore, Duke does not plan on, nor feel that, an independent review of the design is required other than that described in Chapter 17 and Section 3.9.2.8.
6. The minimum guardpipe thicknesses were developed using the criteria of ASME Code case 1606 and the appropriate loading combinations and stress limits of Table 3-48. Table 3-53 lists all cases where guard pipe design utilizes the criteria of ASME Code Case 1606 in lieu of designing for the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.

7. Flued head design is based on the same criteria as the guard pipe design. Design criteria for bellows expansion joints consider operational differential movements between primary and secondary containment and movements as appropriate. All bellows expansion joints are designed for primary containment pressure and are of two ply construction with a wire mesh between plys for testability.
8. Mechanical penetration design features for precluding bypass leakage are as follows:
   a. All mechanical penetrations are designed, fabricated, non-destructively examined and erected to the requirements of ASME Section III, Class 2, Section NC.
   b. All mechanical penetrations and their anchorages are analyzed in accordance with the requirements of ASME Section III, Class 2, Section NC for pipe whip, and associated loadings to assure containment integrity for any loss of coolant accident.
   c. All bellows expansion joints are of two-ply construction with a wire mesh between plys for testability of bellows and bellows weld to piping.

3.9.2.8.2 Hot Penetrations

Typical hot penetration assemblies as shown on Figure 3-68 consists of three major components: a) process line and flued head, b) guard pipe, and c) expansion joint containment seal.

Design requirements for hot penetrations are as follows:

1. The guard pipe and bellows assembly constitute an extension of the containment and as such meet containment design conditions.
2. A guard pipe is required for lines that can overpressurize the annulus (assuming a postulated pipe break) and/or release unacceptable amounts of radioactivity to the atmosphere.
3. Guard pipe contains and returns any process line leakage back to the containment.
4. Bellows design accommodates both axial and lateral displacements between the containment and Reactor Building for thermal, seismic, and containment test conditions.
5. The guard pipe and process line are anchored and guided to act as a single unit under thermal, seismic, and pipe rupture loads.
6. Stress levels for process lines meet requirements of Table 3-48.
7. Stress levels for guard pipes and other penetration structural components meet the requirements of Table 3-48.
8. Exterior bellows cover and impingement plate protects the bellows assembly from foreign objects during construction and station operation.
9. The process pipe was designed to meet the requirement of Table 3-47 for stress levels and applicable loading combinations. The process pipe is of seamless construction made from SA 376 GR 304 or 316 stainless steel.

Design codes applicable to hot penetrations are as described below:

1. Penetrations are in accordance with ASME III, Section NC. Process lines including flued head, guard pipe, and bellows assemblies including dished heads, are designed, fabricated, and inspected to ASME III, Section NC, with the allowable stresses as defined above.

2. The Reactor Building anchor section is considered a structural component. Attachment welds to the guard pipe meet and are inspected to ASME III, Section NC. Field welds between the guard pipe attachment and Reactor Building anchor section are structural welds. Field welds between the bellows and containment meet and are inspected to ASME III, Section NC, except that the bellows stub end to containment weld does not receive third party inspection and the weld is subject to containment pressure test in lieu of NC test requirements.

3.9.2.8.3 Residual Heat Removal Recirculation Line Penetration

Residual heat removal recirculation line penetrations consist of guard pipes surrounding the recirculation lines.

Design requirements for these penetrations are as follow:

1. The recirculation line is an extension of Containment up through the first valve; therefore, a guard pipe is provided from the sump intake structure through the annulus to just outside the Reactor Building wall.
2. Augmented inservice inspection is utilized for weld locations in unguarded piping between the guard pipe and valve where break locations are postulated (see Section 3.6) to provide assurance that critical cracks do not form and integrity of the piping is maintained.
3. All guard pipe is of seamless construction.
4. These valves are Safety Class 2 and are conservatively designed (600 psig design pressure) to withstand the Containment design pressure of 15 psig.
5. Valves are located in an accessible area for maintenance during the post-accident period.
6. Expansion joints are utilized in the penetration design.

3.9.2.8.4 Fuel Transfer Tube Penetration

The fuel transfer tube penetration design as shown on Figure 3-68 consists of two major components: a) transfer tube including blind flange and valve, and, b) the Containment anchor assembly.

Design Requirements for the Fuel Transfer Tube Penetrations are as follows:

1. The transfer tube and anchor assembly constitute an extension of the Containment and as such meet Containment design conditions.
2. Inside Containment bellows design accommodates both axial and lateral displacements between the Containment and spent fuel pool for thermal, seismic, annular, and pool pressure conditions.
3. Outside Containment bellows design accommodates both axial and lateral displacements between the Containment and spent fuel pool for thermal, seismic, annular, and pool pressure conditions.
4. Stress levels for the transfer tube meet requirements of ASME III, Class 2.
5. Stress levels for the anchor ring component meet the requirements of Table 3-48.

Design codes applicable to the Fuel Transfer Tube Penetrations are as follows:

1. The penetration is in accordance with ASME III Section NC. The fuel transfer tube is designed, fabricated, and inspected to ASME III, Section NC with the allowable stresses as defined above.
2. Attachment welds to the fuel transfer tube and field welds between the Containment and the anchor ring component meet and are inspected to ASME III, Section NC.

3.9.3 Components Not Covered by ASME Code

Safety-related mechanical components not covered under the ASME Boiler and Pressure Vessel Code are tabulated in Table 3-4. Design criteria for these components are as stated in the various system descriptions containing these components in Chapter 6, Chapter 9, Chapter 10, Chapter 11, and Chapter 12. Specifically and as can be determined from Table 3-4, components requiring seismic reliability are designed to the same criteria as an ASME component. As described in Section 3.7, the manufacturer has the option of performing detailed seismic design calculations or conducting seismic testing. In addition, all manufacturers and suppliers of components must undergo successfully a quality assurance audit conducted by Duke as described in Chapter 17. Suppliers must be capable of producing components and appropriate QA documentation in strict accordance of Duke's specifications which fully describe necessary NDE and QA documentation requirements. A summary of the design calculations or experimental testing performed to confirm their structural integrity of functional capability is presented in the form of a topical report once these components are ordered and the work has been performed.

3.9.3.1 Heating, Ventilation and Air Conditioning

HVAC equipment is designed as described in Section 3.9.3 and in accordance with Table 3-57.

3.9.3.2 Reactor Core and Vessel Internals Not Covered By ASME Code

The response of the reactor core and vessel internals under excitation produced by a simultaneous complete severance of a reactor coolant pipe and seismic excitation for a typical Westinghouse Pressurized Water Reactor Unit internals has been determined. The following mechanical functional performance criteria apply:

1. Following a postulated design basis accident, the basic operational or functional criterion to be met for the reactor internals is that the unit shall be shutdown and cooled in orderly fashion so that fuel cladding temperature is kept within specified limits. This criterion implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling. The reactor internals structures have been designed to withstand the stress and be within deflection limits originating from a full double-ended RCS primary loop pipe break even though such pipe breaks are no longer considered for dynamic effects, in accordance with Reference 14.
2. For large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. The subsequent refilling of the core by the Emergency Core Cooling System uses borated water to maintain the core in the subcritical state. Therefore, the main requirement is to assure effectivenesss of the Emergency Core Cooling System. Insertion of the control rods, although not needed, gives further assurance of ability to shut the unit down and keep it in a safe shutdown condition.
3. The functional requirements for the core structures during the design basis accident are shown in Table 3-58. The inward upper barrel deflections are controlled to insure no contacting of the nearest rod cluster control guide tube. The outward upper barrel

deflections are controlled in order to maintain an adequate annulus for the coolant between the vessel inner diameter and core barrel outer diameter.

4. The rod cluster control guide tube deflections are limited to insure operability of the control rods.
5. To insure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited to the value shown in Table 3-58.
6. The reactor has mechanical provisions which are sufficient to maintain the design core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from the design basis accident operating conditions (References 1, 5).
7. The core internals are designed to withstand mechanical loads arising from operating basis earthquake, safe shutdown earthquake and pipe ruptures (References 1, 2, 3, 5).

3.9.3.2.1 Faulted Conditions

The following events are considered in the faulted conditions category:

1. Loads produced by a postulated double ended pipe rupture of the main coolant loop design basis accident, for both cases: cold and hot leg break. The methods of analysis adopted are related to the type of accident assumed (cold leg break or hot leg break).
2. Response due to a safe shutdown earthquake.
3. Most unfavorable combination of safe shutdown earthquake and design basis accident. Maximum stresses obtained in each case are added in the most conservative manner.

Maximum stress intensities are compared with the allowable stresses given in Table 3-59 for each of the above conditions. When fatigue is of concern, the applicable stress concentrations factors and peak stresses are used to establish the usage factor. Elastic analysis is used to obtain the response of the structure and the stress analysis on each component is performed on an elastic basis. For faulted conditions, stresses are above yield in a few locations. For these cases only, when deformation requirements exist, a plastic analysis is independently performed to ensure that functional requirements are maintained (guide tube deflections and core barrel expansion). The elastic limit allowable stresses are used to compare with the result of the analysis. No inelastic stress limits are used.

The above-described analyses show that the stresses and deflections which would result following a faulted condition are less than those which would adversely affect the integrity of the structures. Also, the natural and applied frequencies are such that resonance problems should not occur.

3.9.3.2.2 Reactor Internals Response Under Blowdown and Seismic Excitation

A loss of coolant accident would result from a postulated rupture of reactor coolant piping. During the blowdown of the coolant, critical components of the core are subjected to vertical and horizontal excitation as a result of rarefaction waves propagating inside the reactor vessel. For these large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. The subsequent refilling of the core by the Emergency Core Cooling System uses borated water to maintain the core in a subcritical state. Therefore, the main requirement is to assure effectiveness of the Emergency Core Cooling System. Insertion of rod cluster control assemblies, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A one millisecond severance time is taken as the limiting case.

In the case of the hot leg break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefaction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic instability (buckling) or large deflections of the upper core barrel, or both, is the possible response of the barrel during hot leg blowdown. In addition to the above effects, the hot leg break results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg break, a rarefaction wave propagates along a reactor inlet pipe, arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a non-axisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg break, the initial steady state hydraulic lift forces (upward) decrease rapidly (within a few milliseconds) and then increase in the downward direction. These cause the reactor core and lower support structure to move initially downward.

If a seismic event with the intensity of the safe shutdown earth-quake is postulated simultaneously with the loss of coolant accident, the imposed loading on the internals component may be additive in certain cases and therefore the combined loading must be considered. In general, however, the loading imposed by the earthquake is small compared to the blowdown loading.

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Seismic Excitations and Synthesized Time Histories

For a time history response of the reactor pressure vessel and its internals under seismic excitations, synthesized time history accelerations are required. The synthesized time history accelerations used in the McGuire RPV system analysis were based on the seismic response spectra provided in Reference 26. The time history accelerations were developed using DEBLIN2 Computer code, Reference 27. In DEBLIN2, the spectrum amplification and suppression techniques are used to modify the initial transients supplied as input to the code as described in Reference 28. The records of a real earthquake, Taft, are the basis for the synthesized time history accelerations. The spectral characteristics of the synthesized time histories are similar to the original "Taft" earthquake records. The spectrum ordinates are computed using suggested frequency intervals given in Regulatory Guide 1.122, Reference 29. The spectra corresponding to the synthesized time history motions meet the acceptance criteria given in Safety Review Plan (SRP) 3.7.1, Reference 30. Note that the input excitations, which were developed, are for ten (10) second long seismic events.

Seismic Results

The results of system seismic analysis include time history displacements and impact forces for all major components. The time history displacements of upper core plate, lower core plate and core barrel at the upper core plate elevation are provided as input for the reactor core evaluations. The impact forces calculated at the vessel-internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals.

For fuel grid impact loads, time history motions for the lower core plate, upper core plate, and the core barrel at upper core plate elevation were transmitted to Westinghouse Fuels Division in Columbia for fuel/grid impact analysis.

3.9.3.2.3 Acceptance Criteria

The criteria for acceptability in regard to mechanical integrity analyses is that adequate core cooling and shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and stability of the parts in addition to a stress criterion to assure integrity of the components.

1. Allowable Deflection and Stability Criteria

   For the loss of coolant plus the safe shutdown earthquake condition, deflections of critical internal structures are limited to the values given in Table 3-58. In a hypothesized downward vertical displacement of the internals, energy absorbing devices limit the displacement to 1.25 inches by contacting the vessel bottom head. The reactor internals structures have been conservatively designed to withstand the stresses originating from a LOCA (full double-ended primary loop pipe break) even though such pipe breaks are no longer considered for dynamic effects according to Reference 14.

   Upper barrel. The upper barrel deformation has the following limits:

   a. To insure a shutdown and cooldown of the core during blowdown, the basic requirement is a limitation on the outward deflection of the barrel at the locations of the inlet nozzles connected to the unbroken lines. A large outward deflection of the barrel in front of the inlet nozzles, accompanied with permanent strains, could close the inlet area and stop the cooling water coming from the accumulators. Consequently, a permanent barrel deflection in front of the unbroken inlet nozzles, larger than a certain limit called the "hot-loss-of function" limit, could impair the efficiency of the Emergency Core Cooling System.

   b. To assure rod insertion and to avoid disturbing the control rod cluster guide structure, the barrel should not interfere with the guide tubes. This condition also requires a stability check to assure that the barrel does not buckle under the accident loads.

   Control Rod Cluster Guide Tubes. The guide tubes in the upper core support package house the control rods. The deflection limits are established from tests and are provided in Table 3-58.

   Fuel Assembly. The limitations for this case are related to the stability of the thimbles in the upper end. The upper end of the thimbles must not experience stresses above the allowable dynamic compressive stresses. Any buckling of the upper end of the thimbles due to axial compression could distort the guide line and thereby affect the free fall of the control rod.

   Upper Package. The local vertical deformation of the upper core plate, where a guide tube is located, shall be below 0.100 in. This deformation causes the plate to contact the guide tube since the clearance between plate and guide tube is 0.100 in. This limit will prevent the guide tubes from undergoing compression. For a plate local deformation of 0.150 in., the guide tube is compressed and deformed transversely to the upper limit previously established; consequently, the value of 0.150 in. is adopted as the no-loss-of-function local deformation, with an allowable limit of 0.100 in. These limits are given in Table 3-58.

2. Allowable Stress Criteria

   The allowable stress limits during the loss of coolant accident used for the core support structures are based on 10 CFR50, Section 50.55a as referenced in Section 5.2.1.3.

3.9.3.2.4 Methods of Analysis

The internals structures are analyzed for loads corresponding to normal, upset, and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the stress analysis problem is very large, requiring many different techniques and methods, both static and dynamic. The more important and relevant methods used are presented in Section 3.9.1 and summarized in the following sections.

3.9.3.2.5 Blowdown Forces Due to Cold and Hot Leg Break

A blowdown digital computer program (Reference 6), developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in Pressurized Water Reactor Coolant systems during a loss of coolant accident, is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM (Reference 7) which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. This blowdown code is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy are solved numerically using a fixed mesh in both space and time.

Although spatially one dimensional conservation laws are employed, the code can be applied to describe three dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansion, orifices, pumps, and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc., are considered.

Predictions of this code have been compared with numerous test data (Reference 8) and the results show good agreement in both the subcooled and the saturated blowdown regimes.

1. FORCE Model for Blowdown

   The blowdown code evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the program FORCE (Reference 1) which uses a detailed geometric description in evaluating the loadings on the reactor internals.

   Each reactor component for which FORCE calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of: a) the pressure differential across the element, b) the flow stagnation on, and unrecovered orifice losses across the element, and c) friction losses along the element.

   Input to the code, in addition to the blowdown pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The mechanical analysis is performed (Reference 1) using conservative assumptions in order to obtain results with extra margin. Some of the most significant are:

a. The mechanical and hydraulic analyses are performed separately without including the effect of the water-solid interaction. Peak pressures obtained from the hydraulic analysis are attenuated by the deformation of the structures.

b. When applying the hydraulic forces, no credit is taken for the stiffening effect of the fluid environment which reduces the deflections and stresses in the structure.

c. The multi-mass model described below is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the vertical direction. This model is conservative in the sense that further mass-spring resolution of the system would lead to further attenuation of the shock effects obtained with the present model.

2. Vertical Excitation Model for Blowdown

For the vertical excitation, the reactor internals are represented by a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. Also, incorporated in the multi-mass system is a representation of the motion of the fuel elements relative to the fuel assembly grids. The fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers. Coulomb type friction is assumed in the event that sliding between the rods and the grid fingers occurs. A spring-mass system is used to represent the internals. In order to obtain an accurate simulation of the reactor internals response, the effects of internal damping, clearances between various internals, snubbing action caused by solid impact, Coulomb friction induced by fuel rod motion relative to the grids, and preloads in hold down springs have been incorporated in the analytical model. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs.

The appropriate dynamic differential equations for the multi-mass model describing the aforementioned phenomena are formulated and the results obtained using a digital computer program (Reference 1) which computes the response of the multi-mass model when excited by a set of time dependent forcing functions. The appropriate forcing functions are applied simultaneously and independently to each of the masses in the system. The results from the program give the forces, displacements and deflections as functions of time for all the reactor internals components (lumped masses). Reactor internals response to both hot and cold leg pipe ruptures is analyzed. The forcing functions used in the study are obtained from hydraulic analyses in the pressure and flow distribution around the entire Reactor Coolant System as caused by double ended severance of a Reactor Coolant System pipe.

3. Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

Core Barrel. For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse. The barrel is then analyzed for dynamic buckling using the following conservative assumptions:

1. The effect of the fluid environment is neglected (water stiffening is not considered).

2. The shell is treated as simply supported.

During the leg blowdown, the upper barrel is subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

1. The upper core barrel is treated as a simply supported cylindrical shell of constant thickness between the upper flange weldment and the lower core barrel weldment. No credit is taken for the supports at the barrel midspan offered by the outlet nozzles. This assumption leads to conservative deflection estimates of the upper core barrel.
2. The upper core barrel is analyzed as a shell with four variable sections to model the support flange, upper barrel, reduced weld section, and a portion of the lower core barrel.
3. The barrel with the core and thermal shielding pads is analyzed as a beam fixed at the top and elastically supported at the lower radial support and the dynamic response is obtained.

Guide Tubes. The dynamic loads on rod control guide tubes are more severe for a loss of coolant accident caused by hot leg rupture than for an accident by cold leg rupture since the cold leg break leads to much smaller changes in the transverse coolant flow over the rod cluster control guide tubes. Thus, the analysis is performed only for a hot leg blowdown.

The guide tubes in closest proximity to the rupture outlet nozzle are the most severely loaded. The transverse guide tube forces during the hot leg blowdown decrease with increasing distance from the ruptured nozzle location.

A detailed structural analysis of the rod cluster control guide tube is performed to establish the equivalent cross section properties and elastic end support conditions. An analytical model is verified both dynamically and statically by subjecting the control (Reference 1) rod cluster tube to a concentrated force applied at the transition plate. In addition, the guide tube is loaded experimentally using a triangular distribution to conservatively approximately the hydraulic loading. The experimental results consist of a load deflection curve for the rod cluster control guide tube plus verification of the deflection criteria to assure rod cluster control insertion.

The response of the guide tubes to the transient loading due to blowdown may be found by utilizing the equivalent single degree of freedom system for the guide tube using experimental results for equivalent stiffness and natural frequency.

The time dependence of the hydraulic transient loading has the form of a step function with constant slope front with a rise time to peak force of the same order of the guide tube fundamental period in water. The dynamic amplification factor in determining the response is a function of the ramp impulse rise time divided by the period of the structure.

Upper Support Columns. Upper support columns located close to the broken nozzle during hot leg break are subjected to transverse loads due to cross flow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes, i.e., by taking into consideration the increase in flow across the column during the accident. The columns are studied as beams with variable section and the resulting stresses are obtained using the reduced section modulus at the slotted portions.

3.9.3.2.6 Methods and Results of Blowdown Analysis (Mechanical)

The results obtained from the linear analysis indicate that during blowdown, the relative displacement between the components closes the gaps; consequently the structures impinge on each other, making the linear analysis unrealistic and forcing the application of nonlinear methods to study the problem. Although linear analysis does not provide information about the impact forces generated when components impinge on each other, it can be, and is, applied prior to gap closure. The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, between fuel and baffle plates, and between the control rods and their guide paths are considered in the analysis. References (1) and (4) provide further details of the blowdown method used in the analysis of the reactor internals.

Results of these analyses indicate that both static and dynamic stress intensities are within acceptable limits. In addition, the cumulative fatigue usage factor is also within the allowable usage factor of unity.

The stresses due to the safe shutdown earthquake (vertical and horizontal components) are combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection.

These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to assure control rod insertion. These limits are shown in Table 3-58. For the guide tubes deflected above the no-loss-of-function limit, it must be assumed that the rods do not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown is aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no-loss-of-function limit of Table 3-58.

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Mathematical Model of the Reactor Pressure Vessel (RPV)

The mathematical model of the RPV is a three-dimensional nonlinear finite element model, which represents the dynamic characteristics of the McGuire reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN computer code. The WECAN computer code is a general-purpose finite element code. Shown in Figure 3-127 are the loop layout and the global coordinates of the WECAN model. The WECAN model consists of three concentric structural submodels connected by nonlinear impact elements and stiffness matrices. The first submodel, shown in Figure 3-128, represents the reactor vessel shell and associated components. The reactors vessel is restrained by four reactor vessel supports (situated beneath alternate nozzles) and by the attached primary coolant piping. A linear horizontal stiffness and a vertical impact element model each reactor vessel support. A stiffness matrix represents the attached piping.

The second submodel, shown in Figure 3-129, represents the reactor core barrel, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations. The third and innermost submodel, shown in Figure 3-130, represents the upper support plate, guide tubes, support columns, upper and lower core plates, and fuel. This submodel includes the specific properties of the Westinghouse 17x17 Robust fuel assembly with Intermediate Flow Mixers (IFMs). The third submodel is connected to the first and second by stiffness matrices and nonlinear elements.

Fluid-structure or hydro-elastic interaction is included in the reactor pressure vessel model for seismic evaluation. The horizontal hydro-elastic interaction is significant in the cylindrical fluid flow region between the core barrel and reactor vessel (the downcomer). Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel and reactor vessel shell.

Two concentric cylinders are presumed to displace the X_1 and X_2 directions for inner and outer cylinders, respectively. For the case of an incompressible, frictionless fluid displaced in the annulus due to motion of the cylinders, the following expression is derived for the hydrodynamic mass matrix connecting the inner and outer cylinders:

$$M_f = \begin{matrix} M_H & -(M_1 + M_H) \\ -(M_1 + M_H) & (M_1 + M_2 + M_H) \end{matrix}$$

where

$M_1 = \pi R_i^2 L\rho$

$M_2 = \pi R_o^2 L\rho$

M_H = hydrodynamics mass = $\dfrac{R_o^2 + R_i^2}{R_o^2 - R_i^2}\rho\pi R_i^2 L$

L = Length of cylinders

ρ = density of fluid

R_i = inner radius of annulus

R_o = outer radius of annulus

The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel shell and core barrel. The off-diagonal terms reflect the fact that all the water mass does not partipate when there is no relative motion of the vessel and core barrel. It should be pointed out that the hydrodynamic mass matrix has no artificial virtual mass effect and is derived in a straight-forward, quantitative manner.

The matrices are a function of the properties of two cylinders with a fluid in the cylindrical annulus, specifically; inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the reactor core barrel (RCB) allows inclusion of radii variations along the RCB height and approximates the effects of RCB beam deformation. These mass matrices were inserted between selected nodes on the core barrel and reactor vessel shell as shown in Figure 3-131.

The WECAN computer code, which is used to determine the response of the reactor vessel and its internals, is a general-purpose finite element code. In the finite element approach, the structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array are first calculated for each element in the local coordinates. Employing appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure:

$$[M]\{\ddot{U}\} + [D]\{\dot{U}\} + [K]\{U\} = \{F\} \qquad \text{Equation (1)}$$

where:	[M]	=	Global inertia matrix
	[D]	=	Global damping matrix
	[K]	=	Global stiffness matrix
	$\{\ddot{U}\}$	=	Acceleration array
	$\{\dot{U}\}$	=	Velocity array
	{U}	=	Displacement array
	{F}	=	Force array, including impact, thrust forces, hydraulic forces, constraints and weight.

WECAN solves equation (1) using the nonlinear modal superposition theory, described in section 2.5.2.1 of the WECAN User's Manual. An initial computer run is made to calculate the eigenvalues and eigenvectors for the mathematical model. This information is stored, and is used in a subsequent computer run, which solves equiation (1). The first time step performs a static solution of equation (1) to determine the initial vertical displacements of the structure due to deadweight and normal operating hydraulic forces. After the initial time step, WECAN calculates the dynamic solution of equation (1). Nodal displacements and impact forces are stored for post-processing.

The following elements from the WECAN finite element library are used to represent the reactor vessel and internals components.

- Three-dimensional elastic pipe
- Three-dimensional mass with rotary inertia
- Three-dimensional beam
- Three dimensional linear spring
- Concentric impact element
- Linear impact element
- 6 x 6 stiffness matrix
- 18 Card stiffness matrix
- 18 Card mass matrix
- Three-dimensional friction element

The finite element models shown in Figures 3-127 through 3-130 were used to perform the LOCA analysis. Since, McGuire takes credit for leak-before-break (LBB), the LOCA analyses due to main line breaks for the reactor pressure vessel system are not required. Then the next limiting breaks to be considered are the branch line breaks which consists of (a) accumulator line, and (b) residual heat removal (RHR) line breaks.

Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront with low pressure on one side and high pressure on the other. The wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The LOCA loads applied to McGuire reactor pressure vessel system consists of (1) reactor internal hydraulic loads (vertical and horizontal) and (2) reactor coolant loop mechanical loads. All the loads are calculated individually and combined in a time-history manner.

RPV Internal Hydraulic Loads

Depressurization waves propagate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle.

After a postulated break in the cold leg, the depressurization path for waves entering the reactor vessel is through the nozzle which contains the broken pipe and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, around, and down the downcomer annulus, then up through the region circumferentially enclosed by the core barrel; that is, the fuel region.

The region of the downcomer annulus close to the break depressurizes rapidly but, because of restricted flow areas and finite wave speed (approximately 3,000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and RPV. As the depressurization wave propagates around the downcomer annulus and up through the core, the barrel differential pressure reduces, and similarly, the resulting hydraulic forces drop.

In the case of a postulated break in the hot leg, the waves follow a dissimilar depressurization path, passing through the outlet nozzle and directly into the upper internals region, depressurizing the core and entering the downcomer annulus from the bottom exit of the core barrel. Thus, after a break in the hot leg, the downcomer annulus would be depressurized with very little difference in pressure across the outside diameter of the core barrel.

A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the downcomer annulus is not directly involved) and internal differential pressures are not as large as for a cold leg break. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for a hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves would continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX computer code calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and two-phase (saturated) blowdown regimes. The MULTIFLEX program employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture.

The MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator systems. A beam model of the core support barrel has been developed from the structural properties of the core barrel; in this model, the cylindrical barrel is vertically divided into various segments and the pressure as well as the wall motions are projected onto the plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into 10 segments; each segment consistes of 3 separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces, which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible walls is determined by solving the global equations of motion for the masses representing the forced vibration of an undamped beam.

Reactor Coolant Loop Mechanical Loads

The reactor coolant loop mechanical loads are applied to the RPV nozzles by the primary coolant loop piping. The loop mechanical loads result from the release of normal operating forces present in the pipe prior to the separation as well as transient hydaulic forces in the reactor coolant system. The magnitudes of the loop release forces are determined by performing a reactor coolant loop analysis for normal operating loads (pressure, thermal, and deadweight). The loads existing in the pipe at the postulated break location are calculated and are "released" at the initiation of the LOCA transient by application of the loads to the broken piping ends. These forces are applied with a ramp time of 1 millisecond because of the assumed instantaneous break opening time. For breaks in the auxiliary lines, accumulator and RHR, the force applied at the reactor vessel would be insignificant. The restraints on the main coolant piping would eliminate any force to the reactor vessel caused by a break in the auxiliary line breaks.

The severity of a postulated break in a reactor vessel is related to two factors: the distance from the reactor vessel to the break location, and the break opening area. The nature of the reactor vessel decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reacotor vessel.

LOCA Results

The loads described in the previous sections were applied to the WECAN model of the reactor pressure vessel system shown in Figures 3-126 through 3-129 and the input to the analysis was specifically applicable to McGuire. The core plate motions for this analysis were transmitted to Westinghouse Fuels Division in Columbia for fuel/grid impact analysis.

The transition of McGuire to Westinghouse 17 x 17 robust fuel with IFMs will not adversely impact the response of the reactor internals system and components due to LOCA excitations.

3.9.3.2.7 Control Rod Drive Mechanisms

The control rod drive mechanisms are Class A components designed to meet the stress limits of the ASME Boiler and Pressure Vessel Code therefore are presented in Section 4.2.

3.9.3.2.8 Ice Condenser System

The Ice Condenser System structures are described and discussed in Section 6.2.

3.9.3.2.9 Supports, Restraints, and Anchors

Design information pertaining to the principal Reactor Coolant System component supports, restraints, guides, and snubbers is as follows:

1. NSS Vessel, structural type supports and restraints are defined in Section 3.6.4.1 and Section 3.6.5.1.
2. Mechanical type hanger supports, restraints, guides, and snubber devices for the Reactor Coolant System will meet ANSI B31.7 Class 1 (1969), MSS SP-58, MSS SP-69, and Table 3-49 with respect to movements, stresses, and materials for the following loads or applicable combination of loads:
    a. Dead weight effect
    b. Thermal effect

c. Seismic effect

d. Applicable safety valve thrust effects

e. Postulated pipe break and pipe whip effects

f. Maximum differential movement between structures as applicable

Design information pertaining to other safety-related system component supports, restraints, guides, and snubbers is as follows:

Mechanical type hanger supports, restraints, guides and snubber devices will meet ANSI B31.7 (1969), Class II and III, as applicable, ANSI B31.1.0 (1967), MSS SP-58, MSS SP-69, and Table 3-50 with respect to movements, stresses and materials for the following loads or applicable combination of loads:

1. Dead weight effect
2. Thermal effect
3. Seismic effect
4. Applicable safety valve thrust effects
5. Postulated pipe break and pipe whip effects
6. Maximum differential movement between structures as applicable

Loading combinations and stress criteria for mechanical supports, restraints, and anchors are shown in Table 3-49 for Duke Class A systems and in Table 3-50 for Duke Classes B, C, and F systems.

The extent of all restraints includes the components attaching to the piping, support structure or vessel, the main hanger component and all necessary rods and turnbuckles. Structural steel and concrete embedments are considered structural. Exceptions to the extent of restraints would be attachment lugs, plates, etc., which are designed and fabricated with vessels or containment in accordance with ASME III.

The final revision to Duke's response to USNRC IE Bulletin 79-02 was completed in September, 1982 and submitted to the commission. All items of concern have been addressed and resolved.

3.9.3.3 Category 1 Equipment

As identified in Table 3-4 certain Category 1 components were purchased before applicable portions of ASME Section III became effective. These components could not be updated to reflect the requirements of Section III, however a seismic analysis for Category 1 equipment is required of the vendors. These analyses use the methods described in Section 3.7.2.1. The stress criteria for these components are the same as for comparable ASME Section III components.

Category 1 components and their supports were analyzed since testing would not have been appreciably more meaningful considering the lack of availability of the required testing facilities.

Input motion is obtained from the design acceleration response spectra for major components.

All equipment seismic calculations and stress analyses are reviewed independently with approval required prior to component acceptance.

Electrical equipment such valve actuators, battery and instrument racks, control consoles, cabinets, panels, and cable trays are discussed in Section 3.10.

3.9.4 Inservice Testing of Pumps and Valves

Inservice testing of pumps and valves is necessary to assure that these components will be in a state of operational readiness and could perform their safety function throughout the life of the station. Inservice inspection of these components is required by 10 CFR 50.55a (g). In accordance with paragraph (g) (iii) pumps and valves which are classified as ASME Code Class 1, 2 or 3, have been designed and been provided access to enable the performance of inservice testing to assess operational readiness set forth in ASME OM Code 1998 Revision through 2000 Addenda applied to the construction of the particular pump or valve. At Duke Energy Company option, a later edition of the code may be adopted.

3.9.4.1 Relief Requests

As noted in Section 3.9.4, the inservice testing program will be periodically updated to meet requirements of ASME OM Code 1998 Revision through 2000 Addenda. However, if it proves impractical to implement this criteria; requests for relief will be submitted on a case-by-case basis.

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3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

3.10.1 Seismic Design Criteria

3.10.1.1 Equipment Identification

Category 1 instrumentation and electrical equipment requiring seismic qualification are identified in Table 3-7.

3.10.1.2 Criteria

The Reactor Protective System, Engineered Safety Feature circuits and the Emergency Power System are designed to;

1. Assure initiation of protective action during the Safe Shutdown Earthquake when required, and
2. Maintain the unit in a safe shutdown condition (post-accident) during either an Operating Basis or Safe Shutdown Earthquake.

For the Safe Shutdown Earthquake, there may be permanent deformation of the equipment provided that the criteria in 1 and : 2 are satisfied.

3.10.1.3 IEEE Standard 344-1971

Category 1 instrumentation and electrical equipment supplied by the applicant and identified in Table 3-7 is seismically qualified in accordance with the procedure and documentation requirements of IEEE Std 344-1971. When testing is conducted, the normal method on most electrical equipment is the biaxial random multifrequency seismic excitation method similar to that described in IEEE 344-1975.

For equipment which is seismically qualified by testing, the equipment is arranged in an operational mode which demonstrates proper function, e.g., switchgear is mechanically cycled during test, inverters operate under load but not necessarily full load.

The safety related electrical and control equipment in Table 3-7 furnished by the Nuclear Steam Supply System supplier is qualified in accordance with References 2 thru 17 in Section 3.10.3. Responses to the concerns of the NRC evaluation of the above references are being addressed by the Nuclear Steam System supplier for McGuire. These responses (i.e., additional seismic testing or other seismic qualification justification) have been submitted by the above supplier to the NRC for generic resolution as part of the Westinghouse Seismic and Environmental Supplemental Qualification Program. Results of the seismic portion of this program are documented in References 10, 11 and 12

3.10.2 Seismic Analysis, Testing Procedures and Restraint Measures

3.10.2.1 Equipment Supports

The seismic design adequacy of Category 1 electrical equipment supports (e.g., cable trays, battery racks, instrument racks and control consoles) is established by analysis and/or testing. In that cable trays are, in most cases, common to more than one component of safety-related electrical equipment, the necessary seismic analysis or testing may be applied to the tray systems rather than to each individual equipment test or analysis.

3.10.2.2 Amplified Design Loads for Vendor Supplied Equipment

In both the testing and analysis procedure, the possible amplified design loads for vendor supplied equipment is considered as follows:

1. The support is tested with the actual components mounted or with the component loads simulated.
2. The support analysis includes the component loads.

Seismic restraints are used as applicable with their adequacy verified by either testing or analysis.

3.10.3 Reference [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

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3.11 Environmental Design of Mechanical and Electrical Equipment

This section presents information to demonstrate that the mechanical and electrical portions of the Engineered Safety Features and the Reactor Protection Systems are capable of performing their designated safety related functions while exposed to applicable accident and post-accident environmental conditions.

Evaluation for License Renewal:

Some qualification analyses for safety-related equipment identified in Section 3.11.1.1 were found to be time-limited aging analyses for license renewal. The existing EQ process, in accordance with 10 CRF 50.49, will adequately manage aging of EQ equipment for the period of extended operation.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Equipment Identification

3.11.1.1.1 Electrical Equipment

Electrical equipment that is required to perform a safety function(s) in a harsh environment is identified in the McGuire NUREG 0588 submittal (Reference 1). Electrical equipment within the scope of 10CFR50.49 is identified in the McGuire Equipment Database (EDB). The EDB and the Environmental Qualification Maintenance Manual (EQMM) in conjunction with administrative controls of the EQ Program under Duke Nuclear Fleet Procedures AD-EG-ALL-1612 and PD-EG-ALL-1612, provide the documentation required under 10CFR50.49.

3.11.1.1.2 Mechanical Equipment

Mechanical equipment including qualification requirements are identified in Section 3.2.2.

3.11.1.2 Environmental Conditions

The McGuire Plant Environmental Parameters (PEP) Manual, MCS-1240.03-00-0001, contains specific station normal and postulated accident environmental parameters.

3.11.1.2.1 Environmental Conditions Inside Containment

The environmental conditions inside the containment following a design basis accident are determined from analyses performed by Duke and Westinghouse. The containment analyses including methods, assumptions, and results are discussed in Section 6.2.

The environmental parameters that compose the overall worst-case containment accident environment are as follows:

Temperature (Upper Compartment): Time history as shown in Figure 6-9

Temperature (Lower Compartment): Time history as shown in Figure 6-24

Temperature (Break Compartment): Time history as shown in Figure 6-25

Pressure (Upper and Lower Compartment): Time history as shown in Figure 6-8.

Note: The worst-case lower compartment average temperature is due to a Main Steam Line Break (MSLB). The worst-case break compartment temperature is also taken from the MSLB transient. The worst-case upper compartment temperature result, as well as the worst-case pressure results, are due to a Loss of Coolant Accident (LOCA).

Relative Humidity: 100%

Radiation: Total integrated radiation dose for the equipment location includes the 40 year normal operating dose plus the appropriate accident dose based on equipment operability requirements. The bases for determining the containment radiation environment are discussed in Chapter 12.

Chemical Spray: Boric acid and sodium tetraborate spray resulting from mixing in the containment sump of borated water from the refueling water storage tank and sodium tetraborate solution from ice bed melt. Refer to Section 6.1.3 for additional information on chemical spray.

3.11.1.2.2 Environmental Conditions in the Annulus

The environmental conditions in the annulus (primarily temperature and radiation) are dictated by the containment environment because of the physical arrangement of the annulus with respect to the containment. Therefore, the worst case annulus temperature and radiation environment is based on the worst case containment accident environment.

The parameters that compose the overall worst case annulus environment are as follows:

Temperature: 142°F peak.

Relative Humidity: 100%

Radiation: Total integrated radiation dose for the equipment location includes the 40 year normal operating dose plus the appropriate accident dose based on equipment operability requirements. The bases for determining the radiation environment are discussed in Chapter 12.

3.11.1.2.3 Environmental Conditions Outside Containment - Pipe Break

The environmental conditions outside the containment that result from a moderate or high energy system pipe break vary throughout the Auxiliary Building depending upon the specific routes of moderate or high energy system piping, the system in which the break is postulated to occur, and the postulated size and location of the break.

The criteria and method for determining moderate and high energy system pipe breaks and the resulting environmental conditions are discussed in Section 3.6.2, and the resultant environments are documented in the McGuire Plant Environmental Parameters (PEP) Manual, MCS-1240.03-00-0001.

3.11.1.2.4 Environmental Conditions Outside Containment - Radiation

The bases for determining the outside containment normal and post-LOCA recirculation radiation environment are discussed in Chapter 12. The resultant radiation environments are documented in the McGuire Plant Environmental Parameters (PEP) Manual, MCS-1240.03-00-0001.

3.11.2 Qualification Tests and Analyses

The McGuire environmental qualification program for electrical equipment required to perform a safety function in a harsh environment is in accordance with IEEE 323-1971. Qualification is achieved by testing analysis, or a combination of these methods. Additionally, consideration is also given to the capabilities of a manufacturer's specific design in determining qualification. Initial environmental qualification was in accordance to IEEE 323-1971, which remains applicable to most of the station electrical equipment. However, should a device require replacement with a different or newer device, the replacement device will be environmentally qualified with IEEE 323-1974. The Environmental Qualification Maintenance Manual (EQMM) provides a reference to the supporting qualification documentation for each item of equipment located in a postulated harsh environment. The qualification documentation identifies the method of qualification.

The McGuire NUREG 0588 submittal (Reference 1) identifies the method of qualification and provides a reference to the supporting qualification documentation for each item of equipment located in a postulated harsh environment. In general, the qualification references contain a description of the qualification program for their respective equipment.

3.11.2.1 Qualification Criteria and Standards

3.11.2.1.1 10CFR50 Appendix A, General Design Criteria

The implementation of the General Design Criteria (GDC) is discussed in Section 3.1.

3.11.2.1.2 10CFR50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

The implementation of the quality assurance criteria is discussed in Chapter 17.

3.11.2.1.3 10CFR50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.

The NRC has established in 10CFR 50.49 the specific requirements which must be met in order to satisfy General Design Criterion 4 relating to the environmental qualification of electrical equipment. These requirements are met as outlined in Nuclear Fleet Procedures AD-EG-ALL-1612 and PD-EG-ALL-1612. The McGuire NUREG 0588 submittal (Reference 1) contains a discussion of compliance with 10CFR 50.49(b). Additionally, consistent with 10CFR 50.49(k), Duke's position with respect to the Category II (IEEE 323-1971) guidelines of NUREG 0588 is also presented in the McGuire NUREG 0588 submittal (Reference 1).

3.11.2.1.4 NRC Regulatory Guides

<u>Regulatory Guide 1.30 (Safety Guide 30)</u>

The quality assurance requirements for the installation, inspection, and testing of Class 1E electrical equipment are discussed in Chapter 17.

<u>Regulatory Guide 1.40 (Revision 0)</u>

Continuous-duty motors installed inside the containment that are required to function in a harsh environment are qualified in accordance with Regulatory Guide 1.40.

<u>Regulatory Guide 1.63 (Revision 0)</u>

McGuire electrical penetrations have been qualified in accordance with environmental qualification requirements of Regulatory Guide 1.63. Refer to Chapter 8 for compliance with penetration protection requirements.

<u>Regulatory Guide 1.73 (Revision 0)</u>

Electric valve operators installed inside the containment that are required to function in a harsh environment are qualified in accordance with Regulatory Guide 1.73.

Regulatory Guide 1.89 (Revisions 0 and 1)

The recommendations of Regulatory Guide 1.89 Rev. 0, are not applicable to McGuire based on the implementation date of the guide. Qualification of electrical equipment at McGuire in accordance with NUREG 0588, Category II, has been reviewed and accepted by the NRC. Replacement equipment will be qualified in accordance with Regulatory Guide 1.89, Revision 1.

Regulatory Guide 1.131 (August 1977)

The recommendations of Regulatory Guide 1.131 are not applicable to McGuire based on the implementation date of the guide.

3.11.3 Qualification Program Results

The results of the qualification tests and/or analyses for electrical equipment required to perform a safety function in a postulated harsh environment are presented in the individual qualification reports referenced in the McGuire NUREG 0588 submittal (Reference 1). These qualification results have been reviewed with respect to the Duke position on the Category II (IEEE 323-1971) guidelines of NUREG 0588. The results of this review are presented in Reference 1 along with a summary of the qualification program results for each item of electrical equipment required to perform a safety function in a harsh environment is provided in Reference 1.

3.11.4 Air Conditioning and Ventilation Criteria for Control Area

Temperature in the control area (Control Room and Cable Room) is maintained under normal conditions for personnel comfort at 75± 5°F and 45± 10 RH. Protective equipment in this space is designed to operate within design tolerance over this temperature and humidity range. Design specifications for this equipment specify no loss of protective function over the temperature range 40°F to 90°F and humidity range of 15 to 95 RH. Thus, there is a wide margin between design limits and the normal operating environment for Control Room equipment. Should the control area temperature exceed the equipment design temperature, the station will be shut down.

Two 100% Safety Class 3 redundant air handling systems are provided for the Control Room and two 100% Safety Class 3 redundant air handling systems are provided for the control room area. Two 100% Safety Class 3 redundant chilled water systems provide cooling for the above handling systems. If only one air handling unit in each area and one chilled water system remain operable, the control area ambient temperature is not affected.

Refer to FSAR Chapter 7 for further discussion of qualification of safety equipment located in the Control Room.

3.11.5 Chemical and Radiation Environment

3.11.5.1 Chemical Environment

Reference Section 6.5.2 for Containment chemical conditions.

3.11.5.2 Radiation Environment

The bases for determining the radiation environment for normal operation is described in Chapter 12 from which generic environmental conditions are determined. Specific radiation environments are contained in the McGuire Plant Environmental Parameters (PEP) Manual, MCS-1240.03-00-0001.

The design basis accident fission product sources used to determine the containment and annulus accident radiation environment and the post-LOCA recirculation radiation environment are based on the assumptions stated in TID-14844, namely; 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission product inventory. Additional assumptions applied are:

1. Removal of airborne activity by sprays and ice condenser
2. Fission product decay

For additional information regarding the design basis accident radiation environment, refer to Chapter 12.

3.11.6 References

1. Duke Power Company - McGuire Nuclear Station - Response to NUREG 0588 (H.B. Tucker letter to H.R. Denton dated October 15, 1984).
2. Deleted Per 2008 Update.
3. Ellis, A E and Miller, R. B.; "Environmental Qualification of Safety Related Class 1E Process Instrumentation", *WCAP 9157*, August, 1977.
4. Miller, R. B., Qualification Testing of ITT/Barton Transmitters Production Lot No. 2, *WCAP 9885*, April 1981.
5. McGuire Nuclear Station, Response to NUREG-0588.

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